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

March 11, 1996
3F0396-09

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

Subject: Licensee Event Report (LER) 96-003-01

Dear Sir:

Please find the enclosed Licensee Event Report (LER) 96-003-01. This report is submitted by Florida Power Corporation in accordance with 10 CFR 50.73. It provides additional explanation of operator actions relative to exceeding the Reactor Coolant System cooldown rate on January 11, 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

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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 6

TITLE (4)

Operator Error Results in Exceeding Reactor Coolant System Cooldown Rate During Shutdown

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	0 5 0 0 0
0 1	1 1	9 6	9 6	0 0 3	0 1	0 3	1 1	9 6	N/A	0 5 0 0 0

OPERATING MODE (9): 5

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)	[X] 50.73(a)(2)(i)	[ ] 50.73(a)(2)(viii)(A)	
[ ] 20.405(a)(1)(iv)	[ ] 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)
R.O. Enfinger, Manager, Nuclear Safety Assessment Team	3 5 2 5 6 3 - 2 9 4 3

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	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 January 11, 1996, Florida Power Corporation's (FPC) Crystal River Unit 3 (CR-3) was in MODE 5 (COLD SHUTDOWN) and cooling down. The unit had been shut down on January 9, 1996 due to a failure of a main condenser tube. During the cooldown, the cooldown rate was exceeded for approximately 40 minutes. Exceeding the cooldown rate is a violation of Improved Technical Specification 3.4.3 and is reportable in accordance with 10CFR50.73(a)(2)(i)(B) as a condition prohibited by technical specifications. Analysis of the effects of this event show that required safety margins were maintained; therefore, the health and safety of the general public was not affected. The primary cause of this event was the use of a temperature indication by operators that is inconsistent with analysis assumptions when determining Reactor Coolant System temperature for transition between cooldown rates. Corrective actions completed include instructing operators involved of the temperature utilized in the analysis and issuance of interim administrative guidance to clarify existing procedure guidance. Additional corrective actions will address revisions to procedures and expansion of existing engineering guidance for determining limits for temperature changes that occur while swapping Decay Heat Removal system trains.

EVENT DESCRIPTION

On January 11, 1996, Florida Power Corporation's (FPC) Crystal River Unit 3 (CR-3) was in MODE 5 (COLD SHUTDOWN) and cooling down to ambient temperature. The unit had been shut down on January 9, 1996 at 1939 hours due to a failed main condenser [SG,COND] tube which caused out-of-specification secondary chemistry. The shutdown and cooldown were progressing smoothly. The time to conduct the cooldown was extended due to concerns with secondary chemistry. The operating crew was using operating procedures OP-209 "Plant Cooldown" and OP-404 "Decay Heat Removal System", and surveillance procedure SP-422 "RC System Heatup and Cooldown Surveillance" to conduct the evolution. At 1849 hours, Decay Heat (DH) Removal system pump [BP,P] DHP-1A was in service and the last two Reactor Coolant System (RCS) pumps [AB,P], RCP-1A and RCP-1B were secured. With all RCS pumps secured, the operating crew is directed by OP-209 and SP-422 to utilize DH Cooler [BP,CLR] outlet temperature to determine RCS temperature. Discussions were held between licensed operators and a senior operator as to which temperature indication should be used to determine when RCS temperature reached 200 degrees Farenheit (F) for the approaching transition from MODE 4 (HOT SHUTDOWN) to MODE 5. The senior operator believed the procedure guidance to use DH Cooler outlet temperature pertained only to RCS cooldown rate determination. The senior operator reasoned that DH Cooler outlet temperature, approximately 15 degrees F lower than the current Reactor Coolant System bulk fluid temperature, is non-conservative for RCS temperature determination when making mode changes and subsequent removal of Engineering Safeguards (ES) equipment from service. The senior operator directed the licensed operators to use DH Pump suction temperature indication for RCS temperature determination as this parameter indicates RCS bulk fluid temperature.

As the cooldown progressed, RCS temperature approached 150 degrees F. When RCS temperature is </= 150 degrees F, the rate limit becomes more restrictive. The limit changes from 25 degrees F per half hour to 10 degrees F per hour. The senior operator directed use of DH Pump suction indication to determine RCS temperature when the 150 degree F point was reached. The use of DH Pump suction temperature indication to determine RCS temperature for rate limit changes had the unintended consequence of conflicting with assumptions used in the cooldown analysis. Assumptions of RCS temperature for cooldown rate transition is based on Reactor Vessel temperature, of which the DH Pump cooler outlet temperature is the best indication. At approximately 2200 hours, DH cooler outlet temperature decreased less than 150 degrees F. The limit of 10 degrees F per hour was then exceeded twice as recorded in SP-422: 18 degrees F at 2215 hours and 32 degrees F at 2245 hours. The operating crew recorded RCS bulk temperature per DH Pump suction indication </= 150 degrees F at 2315 hours and then they applied the more restrictive limit.

Exceeding RCS pressure and temperature limits is a violation of the Limiting Condition for Operations (LCO) associated with Improved Technical Specification

(ITS) 3.4.3 "RCS Pressure and Temperature (P/T) Limits". The operating crew conducting the cooldown was not aware of the violation at the time.

An independent review of SP-422 data on January 12, 1996 by a different senior operator questioned the use of the indication used to determine when 150 degrees F was reached. Discussions ensued over the weekend and appropriate support staff were asked to look into the difference in temperature indications used in the cooldown analysis. A Problem Report (PR) was written on Monday, January 15, 1996, after sufficient discussions concluded the wrong temperature indication was used. Research of the cooldown analysis commenced and Required Action ITS 3.4.3.C was entered at 1930 on January 15th, after it was determined and communicated to the Shift Supervisor on Duty that use of the RCS bulk fluid temperature conflicted with the analysis which uses DH Cooler outlet temperature. This action requires parameters to be restored within limits immediately and a determination to be made that the RCS is acceptable for continued operation prior to entering MODE 4 (HOT SHUTDOWN).

This report is being provided in accordance with 10CFR50.73(a)(2)(i)(B) to document a condition prohibited by CR-3's Improved Technical Specifications.

EVENT EVALUATION

Reactor vessel cooldown limits are provided to assure analysis assumptions used to calculate the RCS pressure/temperature limits are not exceeded. The calculation of these limits is based on RCS fracture toughness properties. The applicable cooldown rates are:

RCS TEMPERATURE (T)	COOLDOWN RATE
T > 280 degrees F	</= 100 degrees F per hour (</= 50 degrees F per 1/2 hour)
280 degrees F >/= T >/= 150 degrees F	</= 50 degrees F per hour (</= 25 degrees F per 1/2 hour)
150 degrees F >/= T	</= 10 degrees F per hour

Pressure and temperature data starting from 1800 hours on January 11, 1996 and going through the next 550 minutes was retrieved from the plant computer [ID,CMP]. A fracture mechanics evaluation was conducted per ASME Code Section XI Appendix E which provides specific guidance for performing an engineering evaluation of the effects of an out-of-limit condition on the structural integrity of the reactor vessel [RPV]. The unit was maintained in MODE 5 (COLD SHUTDOWN) until the analysis was completed. The evaluation was performed by Framatome Technologies (CR-3's Nuclear Steam System Supplier) and showed that

required safety margins had been met throughout the entire transient. FPC concluded, based on the results of the evaluation, that CR-3's reactor vessel had maintained adequate structural integrity and continued normal operation was acceptable. As a result, ITS Required Action 3.4.3.C was exited on January 17, 1996 at 2148 hours.

CAUSE

The cause of this event is personnel error in that the operating crew misinterpreted the procedure and failed to use DH Cooler outlet temperature to determine the transition temperature for the more restrictive cooldown rate limit. DH Cooler outlet temperature is the best indicator of the reactor pressure vessel temperature change. The operating crew mistakenly decided that Reactor Coolant System bulk fluid temperature should be used. To a lesser extent, training of the operators was not specific enough to preclude the misinterpretation. Additionally, although clear in hindsight, the procedure itself did not provide a sufficient barrier to prevent the occurrence of this event. Another contributing factor was insufficient corrective actions from a 1983 event where the actual calculation of the cooldown rate used the DH Pump suction temperature instead of the DH Cooler outlet temperature. The corrective action from the previous event, revision of SP-422, was not broad enough to preclude the wrong temperature from being used to determine the allowable cooldown rate limit.

IMMEDIATE CORRECTIVE ACTION

The RCS temperature was stable when the event was discovered. ITS Required Action 3.4.3.C was entered and a fracture analysis was performed.

ADDITIONAL CORRECTIVE ACTION

The licensed operators involved were counseled and instructed that DH Cooler outlet temperature is the best, conservative indicator of the reactor pressure vessel temperature.

Short Term Instruction (STI) 96-0004 was issued January 17, 1996 to re-emphasize management's expectations and to clarify the requirements of OP-409 and SP-422 regarding use of DH Cooler outlet temperature for determining RCS temperature relative to cooldown rate limits when the decay heat system is operating with no RC pumps operating.

ACTION TO PREVENT RECURRENCE

1. The guidance provided in response to Request for Engineering Assistance (REA) 93-0667 has been reviewed and expanded upon. REA 93-0667 was initiated after CR-3's March, 1993 event involving an RCS cooldown which

exceeded technical specification requirements. It provided guidance on limits for swapping from one Decay Heat Removal (DHR) system train to the other DHR train. More specific guidance was necessary to ensure brief temperature drops that occur when swapping DH trains are fully addressed.

2. An interpretation has been provided to Operations regarding the use of Average Reactor Coolant Temperature values given in ITS Table 1.1-1 for defining MODES. This interpretation addresses which indications are used when the decay heat system is operating with no RC pumps operating.

3. SP-422 and OP-404 have been revised for human factors enhancements, clarity and technical accuracy.

PREVIOUS SIMILAR EVENTS

Previous events at CR-3 involving RCS cooldown rates occurred in April, 1977, January, 1979, March, 1983 and March, 1993. The April, 1977 event involved a cooldown greater than 100 degrees F in 20 minutes caused by valve leakage while conducting a shutdown from outside the Control Room test. The January, 1979 event involved cooldown greater than 50 degrees F in less than 30 minutes while attempting to recover from a reactor/turbine trip. The 1983 event involved operator error in that cooldown rate was logged using DH pump suction temperature instead of DH cooler outlet temperature. The 1993 event, reported in LER 93-01, involved an overcooling of 25 degrees F greater than permitted by technical specifications, and was caused by failure of a control valve controller.

ATTACHMENT

Attachment 1 -Abbreviations, Definitions and Acronyms

ATTACHMENT 1 - ABBREVIATIONS, DEFINITIONS AND ACRONYMS

CR-3	Crystal River Unit 3
DHR	Decay Heat Removal
FPC	Florida Power Corporation
ITS	Improved Technical Specifications
MODE FOUR	HOT SHUTDOWN
MODE FIVE	COLD SHUTDOWN
OP	Operating Procedure
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.
RCS	Reactor Coolant System
REA	Request for Engineering Assistance
SP	Surveillance Procedure
STI	Short-Term Instructions are part of the Shift Order program which provides a means for operations management to communicate short-term information and administrative instructions to shift personnel.

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])