Florida Power
A Progress Energy Company

Crystal River Nuclear Plant
Docket No. 50-302
Operating License No. DPR-72

Ref.: 10 CFR 50.73

December 3, 2001
3F1201-07

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

Subject: Crystal River Unit 3 - Licensee Event Report 50-302/01-003-00, Small Pressure Boundary Leak Found in Reactor Coolant Pump Heat Exchanger

Dear Sir:

Please find enclosed Licensee Event Report (LER) 50-302/01-003-00. The LER discusses a small Reactor Coolant System pressure boundary leak found in a Reactor Coolant Pump heat exchanger. This report is being submitted pursuant to 10CFR50.73(a)(2)(i)(B) and 10CFR50.73(a)(2)(ii)(A).

If you have any questions regarding this submittal, please contact Mr. Sid Powell, Supervisor, Licensing & Regulatory Programs at (352) 563-4883.

Sincerely,

James H. Terry

James H. Terry
Manager Engineering

JHT/pei

Enclosure

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

IE22

NRC FORM 366 (7-2001) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104 EXPIRES 7-31-2004

Estimated burden per response to comply with this mandatory information collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records Management Branch (T-6 E6), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to bjs1@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

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4. TITLE

Small Pressure Boundary Leak Found in Reactor Coolant Pump Heat Exchanger Resulting in Radionuclide Intrusion into Closed Cycle Cooling System

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MO	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
										05000
10	04	01	01	– 003	– 00	12	03	01	FACILITY NAME	DOCKET NUMBER
										05000

9. OPERATING MODE	5
10. POWER LEVEL	0%

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply)

	20.2201(b)		20.2203(a)(3)(ii)	50.73(a)(2)(ii)(B)	50.73(a)(2)(ix)(A)
	20.2201(d)		20.2203(a)(4)	50.73(a)(2)(iii)	50.73(a)(2)(x)
	20.2203(a)(1)		50.36(c)(1)(i)(A)	50.73(a)(2)(iv)(A)	73.71(a)(4)
	20.2203(a)(2)(i)		50.36(c)(1)(ii)(A)	50.73(a)(2)(v)(A)	73.71(a)(5)
	20.2203(a)(2)(ii)		50.36(c)(2)	50.73(a)(2)(v)(B)	OTHER Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iii)		50.46(a)(3)(ii)	50.73(a)(2)(v)(C)	
	20.2203(a)(2)(iv)		50.73(a)(2)(i)(A)	50.73(a)(2)(v)(D)	
	20.2203(a)(2)(v)	X	50.73(a)(2)(i)(B)	50.73(a)(2)(vii)	
	20.2203(a)(2)(vi)		50.73(a)(2)(i)(C)	50.73(a)(2)(viii)(A)	
	20.2203(a)(3)(i)	X	50.73(a)(2)(ii)(A)	50.73(a)(2)(viii)(B)	

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER (Include Area Code)
Paul Infanger, Project Engineer	(352) 795-6486

13 COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
K	AB	P	B580	Y					

14. SUPPLEMENTAL REPORT EXPECTED			15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE).	X NO				

16. ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)

On October 4, 2001, Florida Power Corporation's (FPC) Crystal River Unit 3 (CR-3) was in MODE 5 (COLD SHUTDOWN) at zero percent RATED THERMAL POWER. While investigating the source of elevated levels of radionuclide activity in the Nuclear Services Closed Cycle Cooling System (SW), CR-3 personnel identified a small leak in the drilled-hole heat exchanger on Reactor Coolant Pump 1B (RCP-1B), which is part of the Reactor Coolant System (RCS) pressure boundary. Although this leakage was not identified during power operation, FPC concludes that the condition is RCS pressure boundary leakage and is reportable under 10CFR50.73(a)(2)(i)(B) and 50.73(a)(2)(ii)(A). This condition does not represent a reduction in the public health and safety due to the small size of the leak (less than 0.02 gallons per minute). The location of the leakage site precluded direct observation or acquiring metallurgical samples. Consequently, the root cause of the leak and failure mechanism is uncertain. The leak site was isolated by plugging the RCP-1B drilled-hole heat exchanger and verified by post-modification testing. CR-3 resumed power operation on October 25, 2001. Since startup, radionuclide activity levels in the SW system have been decreasing, providing confirmation that the RCS leakage has been eliminated. There have been no previous occurrences of an RCS leak to the SW system through an RCP heat exchanger.

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17. NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)*

Description

On October 4, 2001, Florida Power Corporation's (FPC) Crystal River Unit 3 (CR-3) was in MODE 5 (COLD SHUTDOWN) at zero percent RATED THERMAL POWER. While investigating the source of elevated radionuclide activity in the Nuclear Services Closed Cycle Cooling System (SW) [CC], CR-3 personnel identified a small leak in the drilled-hole heat exchanger [AB, HX] (see Attachment 3 for drawing) of the Reactor Coolant Pump 1B (RCP-1B) [AB, P] (Byron Jackson Type DFSS). The drilled-hole heat exchanger is part of the Reactor Coolant System (RCS) [AB] pressure boundary.

CR-3 began to experience elevated levels of radionuclide activity on June 3, 2001. Due to the small size of the leak, normal volumetric methods of locating the leak were ineffective. Some potential in-leakage sources were eliminated because the systems operate at a lower pressure than the SW system. The letdown cooler [CB, CLR] and pressurizer sample coolers [PZR, CLR] had been the sources of all previous RCS to SW leakage events. A series of tests were performed by isolating SW flow to these coolers. The results of these tests were not conclusive. Similar testing for the RCPs could not be performed because operating procedures and the CR-3 licensing basis do not allow isolation of SW cooling while RCPs are in operation. Contingency plans were established to perform additional diagnostic testing during the September 2001 refueling outage. Since the testing was unable to specifically locate the source of the increased radionuclides into the SW system, the leakage remained classified as unidentified.

Prior to the refueling outage that began September 29, 2001, CR-3 developed a method to more effectively sample the return flow from the RCP heat exchangers. This test involved throttling the SW to the RCP cooling, one pump at a time. A sample of the individual pump return flow was taken while the mixing flow was reduced. Initial sampling was inconclusive but a second attempt gave indications that RCP-1B was a possible leakage site. During the refueling outage, CR-3 confirmed a small leak in the drilled-hole heat exchanger on RCP-1B. The leakage was identified by a more conclusive pressure decay test. However, due to the nature of the test, the exact location of the leakage site could not be found.

The drilled-hole heat exchanger is located in the pump cover-thermal barrier, below the mechanical seals and above the pump bearing. This heat exchanger is a series of interconnecting angular drilled holes. During fabrication these holes were machined from above through the stuffing box bore. These holes are encapsulated by the installation of a welded ring in the bottom of the stuffing box bore, forming the pressure boundary between the RCS fluid and the SW system fluid. The SW fluid flows into and around this interconnecting ring of drilled holes before rejoining the fluid from the integral heat exchanger and then returning to the SW system.

The RCP integral heat exchanger is of a tube-in-tube design, external to the pump mechanical seals. A recirculation impeller, located just under the RCP seals, pushes the reactor coolant fluid (when the pump is running) through the inner tube. The SW system fluid passes through the outer tube prior to joining the drilled-hole heat exchanger fluid and returning to the SW system.

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The RCP drilled-hole heat exchanger was determined to be the source of the RCS leakage into the SW system. The heat exchanger is part of the RCS pressure boundary required by Improved Technical Specifications (ITS) 4.6.12, which states that no pressure boundary leakage is allowed. This leakage began while CR-3 was in MODE 1, Power Operation. Therefore, CR-3 was outside of the Limiting Condition for Operation for ITS 3.4.12. This condition is reportable under 10CFR50.73(a)(2)(i)(B) as a condition prohibited by ITS. This condition was also reported at 1558 on October 4, 2001 as a non-emergency eight-hour notification to the NRC Operations Center (NRC Event #38348) for being an event or condition that results in the degradation of a principal safety barrier (RCS pressure boundary). Therefore, this condition is also reportable under 10CFR50.73(a)(2)(ii)(A).

Assessment of Safety Consequences

The nuclear safety aspects of a very small leak (less than 0.02 gpm) from the RCS into the SW system were evaluated and it was concluded that leakage of this magnitude did not represent a significant safety concern. The leakage was too small to be detected by normal volumetric measurements and was detected only by an increasing trend on the radiation monitor in the SW system. The leakage was treated as unidentified leakage because the source was not known. ITS 3.4.12 allows up to 1 gpm of unidentified leakage before action is required. However, it was later determined that this leakage was through the RCS primary pressure boundary (through RCP-1B's drilled-hole heat exchanger), which is prohibited by ITS.

The information provided by pressure testing and isotopic assays indicates that this is a very small defect, but the information cannot be used to positively locate or size the flaw. Hence, it was conservatively assumed to be in the ligament between the RCS and a drilled hole extending for the entire vertical length. This condition was analyzed by the pump designer (Flowserve SR-0919, Rev. 0), reviewed and approved by Framatome (38-1288504-00), and owner accepted by CR-3 as calculation M-01-0008, Rev. 0. This calculation concludes:

1. The analysis very conservatively assumes that the flaw is a crack, which extends for the entire vertical length of a drilled hole.
2. Once the maximum length was attained, the crack could not propagate further.
3. With this maximum crack length, and with the drilled-hole system blocked, there is no possibility that the crack will extend through the wall of the RCP cover and allow leakage of reactor coolant into containment.
4. The postulated axial crack would result in the loss of hoop restraint at the affected drilled hole. This effect could result in some localized distortion; this would, however, be limited to the vicinity of the affected hole and would have no effect on the gross behavior of the cover. The structural integrity of the cover is considered assured for two years of operation.

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There are no short-term safety concerns due to having a crack in the ligament of the drilled-hole heat exchanger. The RCP cover will perform its design function for at least two years. The two-year limitation was imposed as a conservative measure until a full analysis of the impact of cycle-related stresses could be completed. A detailed computer aided analysis is in progress to quantify the low cycle fatigue stress range that develops with the assumed full-length flaw during plant heat-up and cool down. Preliminary analysis shows that the cover's service life, with the assumed flaw, will exceed the remaining plant life. Therefore, this condition does not constitute a safety system functional failure and there was no reduction in the public health and safety.

Based on RC pump heat exchanger thermal analysis, the tube-in-tube heat exchanger provides the majority of cooling capacity. The analysis indicates that the RCP seal temperature will remain below the seal maximum operating temperature with the drilled-hole heat exchanger isolated following a loss of seal injection (SI) [CB, HX] flow. Therefore, the affected pump may operate indefinitely without impact on seal longevity even without SI flow.

Cause

The cause of the leakage is unknown. The location of the leak site was confirmed to be in the drilled-hole section of the RCP seal heat exchanger. The leak was repaired by plugging the inlet and outlet to the drilled-hole heat exchanger. However, the location of the leakage site precluded direct observation or acquiring metallurgical samples. Consequently, the root cause of the leak and failure mechanism is uncertain.

Corrective Actions

The leaking section of RCP-1B was isolated using two ASME Class I plugs, one at the inlet and one at the outlet of the heat exchanger. This repair design ensures that the leak site is isolated from the SW system. Post modification testing was performed by pressurizing the plugged area and verifying that the pressure did not significantly decay. Monitoring of SW activity level since return to power operation October 25, 2001, indicates that there is no further RCS leakage into the SW system. Analysis will be completed to demonstrate that the isolated flaw in the drilled-hole heat exchanger will not propagate to the RCP cover. This item is being tracked in the Corrective Action Program under Nuclear Condition Report 43024.

Previous Similar Events

CR-3 has had previous leakage from the RCS into the SW System (one Pressurizer Sample Cooler leak and five Letdown Cooler Leaks). This is the first time that an RCP contained the leakage site. Industry wide, this style RCP has approximately 1000 pump years of operation at pressurized water reactors. CR-3 is the first plant with a confirmed case of an RCP drilled-hole heat exchanger leak.

ATTACHMENTS

Attachment 1 - Abbreviations, Definitions, and Acronyms
Attachment 2 - List of Commitments
Attachment 3 - Drawing of RCP Seal Area

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ATTACHMENT 1

ABBREVIATIONS, DEFINITIONS AND ACRONYMS

ASME	American Society of Mechanical Engineers
CFR	Code of Federal Regulations
CR-3	Crystal River Unit 3
FPC	Florida Power Corporation
gpm	Gallons per minute
ITS	Improved Technical Specifications
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
SI	Reactor Coolant Pump Seal Injection from the Makeup (Charging) System
SW	Nuclear Services Closed Cycle Cooling (Component Cooling Water)

NOTES: Improved Technical Specifications defined terms appear capitalized in LER text, {e.g., MODE 1}

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

EIIS codes appear in square brackets, {e.g., reactor building penetration [NH, PEN]}.

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ATTACHMENT 2

LIST OF COMMITMENTS

The following table identifies those actions committed to by Florida Power Corporation in this document. Any other actions discussed in the submittal represent intended or planned actions by Florida Power Corporation. They are described to the NRC for the NRC's information and are not regulatory commitments. Please notify the Supervisor, Licensing & Regulatory Programs of any questions regarding this document or any associated regulatory commitments.

RESPONSE SECTION	COMMITMENT	DUE DATE
	No regulatory commitments are being made in this submittal.	

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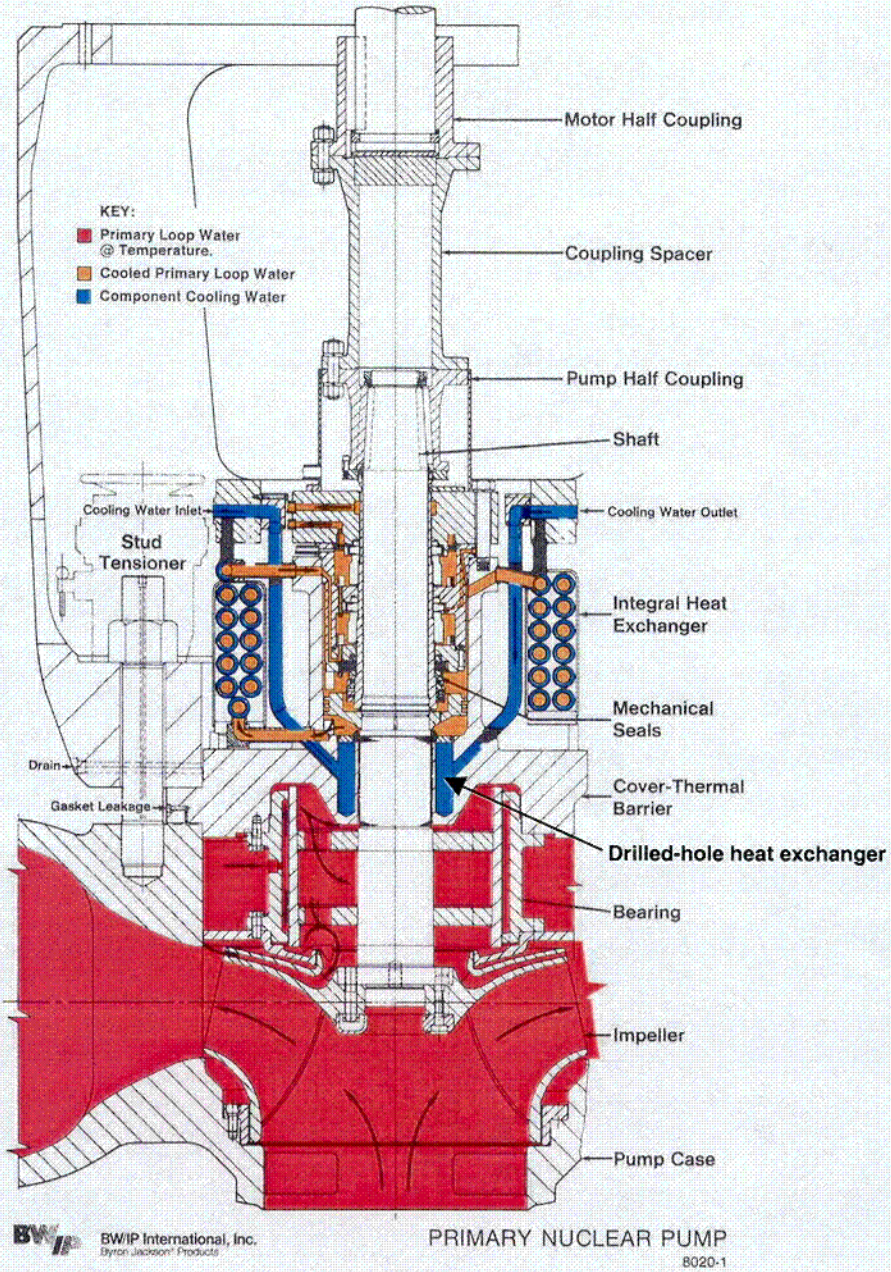
ATTACHMENT 3 - Drawing of RCP Seal Area



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