FPL.

POWERING TODAY.
EMPOWERING TOMORROW.®

10 CFR § 50.73
L-2008-217
October 6, 2008

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

Re: Turkey Point Unit 4
Docket No. 50-251
Reportable Event: 2008-003-00
Date of Event: August 7, 2008
Class 1 Weld Leak Due to Fatigue and Completion of Technical Specification Required Shutdown

The attached Licensee Event Report 05000251/2008-003-00 is being submitted pursuant to 10 CFR 50.73(a)(2)(i)(A), 10 CFR 50.73(a)(2)(i)(B) and 10 CFR 50.73(a)(2)(ii)(A) to provide notification of the subject event.

If there are any questions, please call Mr. Robert Tomonto at 305-246-7327.

Very truly yours,

William Jefferson, Jr.
Vice President
Turkey Point Nuclear Plant

Attachment

cc: Regional Administrator, USNRC, Region II
Senior Resident Inspector, USNRC, Turkey Point Nuclear Plant

IE22
NRR

NRC FORM 366 (9-2007)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB: NO. 3150-0104 EXPIRES: 08/31/2010

LICENSEE EVENT REPORT (LER)

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Turkey Point Unit 4	05000251	1 of 6

4. TITLE

Class 1 Weld Leak Due to Fatigue and Completion of Technical Specification Required Shutdown

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
8	7	2008	2008	- 003 -	00	10	6	2008	FACILITY NAME	DOCKET NUMBER

9. OPERATING MODE: 1

10. POWER LEVEL: 100

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

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

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER (Include Area Code)
Paul F. Czaya	305-246-7150

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

14. SUPPLEMENTAL REPORT EXPECTED		15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)*	☒ NO		-	-	-

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

On August 7, 2008, at approximately 1230 with Unit 4 at 100% reactor power a small weld crack was discovered on a ¾ inch test connection line for the 4B reactor coolant pump (RCP) seal water injection (SWI) header inside containment. On August 15, 2008, FPL was informed by the NRC that Technical Specification (TS) 3.4.10, Structural Integrity, was applicable to the Class 1 leak. As TS Limiting Condition for Operation 3.4.10 and its related Action for Class 1 components could not be met, entry into TS 3.0.3 was required. A Unit 4 shutdown commenced at approximately 1629 on August 15, 2008, and was completed on August 17, 2008 at approximately 0402 when Unit 4 reached Mode 5. Entry into TS 3.4.10 was not made on August 7, 2008 as FPL had previously determined that TS 3.4.10 did not apply to operational leakage. The socket weld fracture initiated at the weld root and propagated outward by fatigue, initiated by inherent minor positive displacement charging pump induced system vibration over an extended period of time. The cause of the failure to enter TS 3.4.10 and TS 3.0.3 on August 7, 2008 is attributed to a lack of clarity in TS 3.4.10. FPL and the NRC came to different conclusions regarding the applicability of TS 3.4.10 to operational leakage. Corrective actions include the following: 1) The 4B RCP SWI test connection line has been repaired. 2) Initial extent of condition inspections are complete on Unit 4. 3) A license amendment request will be submitted to the NRC to delete or relocate TS 3.4.10.

NARRATIVE

DESCRIPTION OF THE EVENT

On August 7, 2008 at approximately 1230 with Unit 4 at 100% reactor [AC, RCT] power, a Unit 4 containment [NH] walkdown was performed to investigate elevated reactor coolant system (RCS) [AB] leakage of approximately 0.16 gpm. During the investigation, a small weld crack was discovered on a ¾ inch test connection line for the 4B reactor coolant pump (RCP) [AB, P] seal water injection (SWI) header inside containment. The affected weld is located on a ¾ inch piping socket weld near test connection valve [CB, TV] 4-285E where the ¾ inch test connection line is welded to a half coupling [CB, CPLG] branch connection on the 4B SWI header. The affected weld is within the site designated Quality Group A (ASME Class 1) boundary.

The inlet to the leak location was able to be isolated, if needed, by manual closure of upstream valve [CB, ISV] 4-297B in the pipe and valve room or a manual SWI filter [CB, FLT] valve [CB, ISV] in the charging pump [CB, P] room. Downstream isolation was via check valve [CB, SHV] 4-298E. The leak was deemed isolable and so was not considered pressure boundary leakage. Unit 4 remained in power operation while Florida Power and Light Company (FPL) evaluated repair options.

On August 15, 2008, during a teleconference with NRC representatives, FPL was informed by the NRC that Technical Specification (TS) 3.4.10, Structural Integrity, was applicable to the Class 1 leak. As TS Limiting Condition for Operation (LCO) 3.4.10 and its related Action for Class 1 components could not be met, entry into TS 3.0.3 was required. A Unit 4 shutdown commenced at approximately 1629 on August 15, 2008. The shutdown was completed on August 17, 2008 at approximately 0402 when Unit 4 reached Mode 5. Entry into TS 3.4.10 was not made on August 7, 2008 as FPL had previously determined that TS 3.4.10 did not apply to operational leakage.

The leak was reported to the NRC Operations Center in accordance with 10 CFR 50.72(b)(3)(ii)(A), however, the eight hour requirement for this report was not met. The Event Notification (EN) number is 44398. Condition Report (CR) 2008-25246 was initiated in response to the leak condition and CR 2008-25373 was initiated to document the EN and address its lateness. The initiation of the Unit 4 shutdown was reported to the NRC Operations Center in accordance with 10 CFR 50.72(b)(2)(i). The EN number is 44418. CR 2008-27020 was initiated in response to the failure to enter TS 3.4.10 on August 7, 2008.

CAUSE OF THE EVENT

The socket weld fracture initiated at the weld root and propagated outward by fatigue, initiated by inherent minor positive displacement charging pump induced system vibration over an extended period of time.

The apparent cause of the failure to enter TS 3.4.10 and TS 3.0.3 on August 7, 2008 is attributed to a lack of clarity in TS 3.4.10. FPL and the NRC came to different conclusions regarding the applicability of TS 3.4.10 to operational leakage.

NARRATIVE

ANALYSIS OF THE EVENT

Background

The socket weld leak location was at a branch connection off the SWI line to the 4B RCP. This is considered part of the Chemical and Volume Control System (CVCS) [CB] that connects to the RCS at the 4B RCP seal.

The CVCS a) adjusts the concentrations of chemical neutron absorber [CB, ABS] for chemical reactivity control, b) maintains the proper water inventory in the RCS, including makeup for system leakages, c) provides the required seal water flow for the reactor coolant pump shaft seals, d) processes reactor coolant letdown, e) maintains the proper concentration of corrosion inhibiting chemicals in the reactor coolant and f) maintains the reactor coolant and corrosion activities to within design levels. The system is also used to fill and hydrostatically test the RCS.

The RCS transfers the heat generated in the reactor core to the steam generators where steam is generated to drive the turbine generator [TA, TB]. Borated water is circulated at the flow rate and temperature consistent with achieving the desired reactor core thermal-hydraulic performance. The water also acts as a neutron moderator and reflector, and as a solvent for the neutron absorber used in chemical shim control.

The RCS provides a boundary for containing the coolant under operating temperature and pressure conditions. It serves to confine radioactive material and limits to acceptable values its release to the secondary system and to other parts of the unit under conditions of either normal or abnormal reactor operation. During transient operation the system's heat capacity attenuates thermal transients generated by the core or extracted by the steam generators. The RCS accommodates coolant volume changes within the protection system criteria. The layout of the system ensures natural circulation capability following a loss of flow to permit cooldown without overheating the core. Part of the system's piping is used by the Emergency Core Cooling System to deliver cooling water to the core during a loss-of-coolant accident.

Analysis

Class 1 Leak

The identified cracked weld is located on a 4B RCP SWI test connection line upstream of valve 4-285E. The affected socket weld joins a ¾ inch, Schedule 160 SS test line to a ¾ inch half coupling in the 2 inch, Schedule 160 SS SWI line. The test line branch is oriented vertically up and includes an unsupported 12 lb. isolation valve (4-285E), which was located approximately 6 inches from the affected weld. The half coupling is located between check valve 4-298B and globe valve 4-298H in the 2 inch SWI line. Based on maintenance history review, the affected weld and all associated components mentioned above are suspected to be original plant equipment.

A metallurgical visual exam and preliminary failure analysis was performed by FPL Engineering. Visual examinations confirmed a circumferentially oriented flaw approximately 3/8 - 1/2 inches long. The general

NARRATIVE

appearance of the fillet weld was an as-welded surface condition with a ground area approximately ¾ inch in length in the area of the leak. The weld measured approximately ¼ inch fillet all around. The crack location was entirely on the face of the fillet weld close to the half coupling side weld toe. Visually, the crack appeared to be tight with no evidence of pitting or corrosion on the surface. The ¾ inch test line did not display any evidence of significant plastic deformation in the pipe or weld region.

The presence of the ground area suggests that this weld had original fabrication complications. The ground area may have locally reduced the weld cross section. The orientation of the crack and its features suggests that the probable failure mechanism of the socket weld was low stress/high cycle fatigue. The confinement of the flaw to the face of the fillet weld suggests that the crack was ID initiated. Environmentally assisted cracking is not a suspected contributor since the line is constantly flowing with deoxygenated RCS fluid.

Based on preliminary engineering evaluation, the probable failure mechanism of the socket weld was low stress/high cycle fatigue initiated by inherent minor positive displacement charging pump induced system vibration over an extended period of time.

The removed fillet weld sample was sent to an independent laboratory to perform metallurgical failure analysis to confirm the failure mechanism. The metallurgical failure analysis concluded that the socket weld fracture initiated at the weld root and propagated outward by fatigue. The circumferential fracture exited just below the weld throat centerline. There were two contributing factors identified, localized within the failure quadrant: the lack of penetration at the weld root (stress riser) and weld metal surface grinding. Both of these conditions reduced the throat thickness, but it was still within minimum calculated throat thickness.

The 4B RCP SWI test connection line upstream of valve 4-285E has been successfully repaired. To help minimize cyclic fatigue, repair activities modified the configuration of the branch connection by reducing the branch connection cantilever to valve pipe length from six to three inches, and replaced socket weld profile with a 2:1 fillet leg size.

An initial extent of condition evaluation included similar unsupported cantilevered configuration welds for charging and RCP SWI piping downstream of the charging pumps for Unit 4 that have been exposed to similar charging pump induced service cyclic fatigue.

ASME Code Class 1 SWI cantilevered configurations inside containment were inspected and/or determined to be well supported. Although the crack associated with 4-285E was ID initiated, these failures can be either OD initiated at the weld toe or ID initiated at the weld root. These initiation sites often display weld fabrication defects that act as stress risers such as undercut at the weld toe or lack of penetration at the weld root. Completed NDE inspections will only identify OD initiated flaws. Liquid penetrant tests performed on similar welds were satisfactory with no indication of degradation. Other CVCS branch connections located outside containment were visually inspected based on radiological conditions. Results at these locations were satisfactory.

NARRATIVE

Other similar configurations, long term cyclic impacts and a review of similar conditions in Unit 3 are being evaluated.

Late Entry into TS 3.4.10

On August 15, 2008, the NRC staff informed FPL that TS LCO, 3.4.10, Structural Integrity, was applicable to the Unit 4 leak. Since the LCO and the required Action could not be met, Unit 4 entered TS 3.0.3 and a shutdown was initiated and completed.

A prior FPL interpretation of TS LCO 3.4.10 determined that it did not apply to operational leakage but only to defects found in all Class 1, 2 and 3 components as a result of ASME Code required examinations and inspections. The NRC communicated to FPL that TS 3.4.10 applies only to components in the RCS and in all modes regardless of the means of identifying defects. As the actions of TS 3.4.10 presume the plant is shutdown and as the surveillance requirements are specific to the ASME Code Section XI inservice inspection program, TS 3.4.10 is unclear in its applicability and subject to interpretation. FPL will propose to delete or relocate TS 3.4.10 from the Turkey Point TS.

Reportability

10 CFR 50.73(a)(2)(ii)(A) requires the reporting within 60 days of discovery:

"Any event or condition that resulted in:

(A) The condition of the nuclear power plant, including its principal safety barriers, being seriously degraded."

Section 3.2.4 of NUREG-1022, Revision 2, provides several examples of conditions that are reportable. One of these examples is as follows:

"Welding or material defects in the primary coolant system which cannot be found acceptable under ASME Section XI, IWB-3600, "Analytical Evaluation of Flaws" or ASME Section XI, Table IWB-3410-1, "Acceptance Standards.""

Since a through-wall leak is not acceptable under ASME Code rules, the leak is reportable in accordance with 10 CFR 50.73(a)(2)(ii)(A).

Unit 4 entered TS 3.0.3 since the requirements of TS LCO 3.4.10 and its associated Action for Class 1 components could not be met. The completion of a nuclear plant shutdown required by the TSs is reportable to the NRC in a Licensee Event Report in accordance with 10 CFR 50.73(a)(2)(i)(A).

Since Unit 4 did not enter TS 3.4.10 on August 7, 2008 when the through-wall leak was first identified, it is a condition prohibited by TS and is reportable in accordance with 10 CFR 50.73(a)(2)(i)(B).

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Turkey Point Unit 4	05000251	2008	- 003	- 00	6 of 6

NARRATIVE

ANALYSIS OF SAFETY SIGNIFICANCE

The leak location was able to be isolated, if needed in the event of catastrophic failure, by manual closure of upstream valve 4-297B in the pipe and valve room or a manual SWI filter valve in the charging pump room and downstream via check valve 4-298E. The actual leak rate was well within normal makeup capability.

The effect of this event on core damage frequency is a forced manual shutdown to repair the leak. The ICCDP (Incremental Conditional Core Damage Probability) estimates vary with assumptions (whether time-dependent or time-independent of the degradation and failure probability of the check valve). Bounding estimates of ICCDP indicate that the risk impact is less than 1.0E-6. The safety significance is considered very low.

CORRECTIVE ACTIONS

Corrective actions include the following:

1. The 4B RCP SWI test connection line upstream of valve 4-285E has been repaired.
2. Initial extent of condition inspections on Unit 4 are complete.
3. A root cause evaluation is in progress. The evaluation and any additional corrective actions, including similar locations on Unit 3, will be documented in CR 2008-25246.
4. A license amendment request will be submitted to the NRC to delete or relocate TS 3.4.10.

ADDITIONAL INFORMATION

EIIS Codes are shown in the format [IEEE system identifier, component function identifier, second component function identifier (if appropriate)].

FAILED COMPONENTS IDENTIFIED: None

PREVIOUS SIMILAR EVENTS: None