FPL

L-2012-383
10 CFR § 50.73
OCT 2 4 2012

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

Re: Turkey Point Unit 3
Docket No. 50-250
Reportable Event: 2012-003-00
Condition Prohibited by Technical Specifications Due to Instrument Isolation Valve Mispositioning

The attached Licensee Event Report 05000250/2012-003-00 is submitted in accordance with 10 CFR 50.73(a)(2)(i)(B).

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

Very truly yours,

Michael Kiley
Vice President
Turkey Point Nuclear Plant

Attachment

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

IE22
NRR

Florida Power & Light Company
9760 SW 344th St., Florida City, FL 33035

NRC FORM 366 (10-2010) | U.S. NUCLEAR REGULATORY COMMISSION | APPROVED BY OMB: NO. 3150-0104 | EXPIRES: 10/31/2013

LICENSEE EVENT REPORT (LER)

Estimated burden per response to comply with this mandatory collection request: 80 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the FOIA/Privacy Section (T-5 F53), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects.resourse@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 an 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.

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4. TITLE
Condition Prohibited by Technical Specifications Due to Instrument Valve Mispositioning

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	25	2012	2012	003	00	10	24	2012	FACILITY NAME	DOCKET NUMBER

9. OPERATING MODE: Mode 2

10. POWER LEVEL: < 3 %

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)0	☐ 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)
Robert J. Tomonto	305-246-7327

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 8/25/12, at approximately 1140, Turkey Point Unit 3 was in Mode 2. The Operations Department was performing the Main Turbine Valve Alignment, in preparation for turbine start-up following a refueling outage. During the alignment verification, Operations discovered the root isolation valves for the Turbine inlet pressure transmitters closed when they were required to be open. The Main Steam pressure transmitters, PT-3-446 and PT-3-447, provide input to various protection and control functions. Upon discovery of this condition, operators entered Technical Specification (TS) 3.0.3 for Unit 3 because the Minimum Channels Operable requirements of TS 3.3.2, Table 3.3-2, Functional Unit 1.f (Safety Injection, Steam Line flow – High coincident with SG pressure Low or Low T_{avg}) and TS 3.3.2, Table 3.3-2, Functional Unit 4.d (Steam Line Isolation) were not met. The isolation valves were then opened and TS 3.0.3 was exited at approximately 1239.

The cause was determined to be lack of rigor in ensuring a proper follow-up review of a modification, which added the new root isolation valves at the High Pressure Turbine inlet pressure tap locations.

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NARRATIVE

DESCRIPTION OF THE EVENT

On 4/28/12, an Engineering modification was created to incorporate carbon steel piping and new root isolation valves, 3-10-292/293/294/295, at the High Pressure Turbine inlet pressure tap locations. A change request notice (CRN) was prepared by a non-FPL organization that did not identify plant procedure 3-NOP-089, Main Turbine, as an affected document. In addition, the CRN was not formally sent to either the Procedures group or Operations group for review, as required by design control process. The new root isolation valves were installed in the plant on 7/21/12 and declared operational.

On 8/10/12 at approximately 0904, Unit 3 entered Mode 3. Unit 3 subsequently entered Mode 2 on 8/15/12 at approximately 1607.

During preparation for entry into Mode 1 on 8/25/12, while performing a Main Turbine [EIIS: TA, TRB] valve alignment, the field operator realized that there were two valves shown on the related plant drawing that were not listed in the valve alignment procedure. These root isolation valves [EIIS: JC, RTV], 3-10-292 and 3-10-294, which were shown as open on the Piping and Instrumentation Diagram (PI&D), were found in the closed position, thereby isolating Main Steam pressure transmitters [EIIS: SB, PT], PT-3-446 and PT-3-447, which provide input to steam line break detection. The pressure transmitters are required to be OPERABLE in Modes 1, 2 and 3.

The discrepancy was identified as a result of the plant process which requires the operator performing a valve alignment to use two independent sources to verify valve lineup. In addition to the procedure, the operator was using the plant PI&D to satisfy the requirement for two independent sources. While performing this task, the operator noticed the two valves, 3-10-292 and 3-10-294, on the PI&D were not included on his walkdown verification checklist. The operator located the valves and recognized that their position did not match the drawing.

The operator immediately notified the Control Room. TS 3.0.3 was entered because TS 3.3.2, Table 3.3-2, Functional Units 1.f and 4.d were not met and the time in the condition exceeded the Action requirement for one channel less than the total number of channels.

This report is submitted in accordance with 10 CFR 50.73(a)(2)(i)(B) as a condition which was prohibited by the plant's TS.

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CAUSE OF THE EVENT

The root cause identified a weakness in the CRN process with respect to the Operations Department "impact of change determination" and review requirements. [Note: CRN documents are part of the plant design configuration control program.]

The following programmatic contributing causes were identified:

1. There is no detailed guidance in the CRN procedure to make the "impact of change" determination. In addition, the CRN form incorrectly specified Post Maintenance Test, Maintenance and Operations requirements on the same line item, masking the specific areas that had been identified as affected.
2. The responsible engineer handed off the task of identifying the affected departments impacted by the CRN to another group and did not follow-up to ensure documents were updated.
3. Operations Department processes were relied upon for configuration control to ensure the proper position of the newly installed valve, instead of configuration management processes and post maintenance testing.

ANALYSIS

System Description

Turkey Point has two pressure transmitters, PT-446 and PT-447, which measure the pressure in the first stages of the high pressure turbine. These transmitters provide signals equivalent to the turbine load for indication, control and protection. Both PT-446 and PT-447 signals are used by the steam line break detection logic to provide an indication of the turbine load. This signal is used to develop the maximum steam flow set point for a given turbine load.

The engineered safeguards system [EIIS: JE], in conjunction with the Reactor Protection System (RPS) [EIIS: JC], is provided to sense impending accident situations and avoid exceeding safety limits during anticipated transients by activating appropriate engineered safeguards features (ESF) as necessary and initiating a reactor trip. The ESFs include the Emergency Core Cooling System [EIIS: BP and BQ], Containment Ventilation and Heat Removal Systems [EIIS: BK], and Component Cooling Water System [EIIS: CC].

The types of major postulated accidents for which the engineered safeguards system provides protection are:

1. Loss of primary coolant accident,
2. Loss of feedwater accident,
3. Main steamline break accident, and
4. Other accidents resulting in the actuation of safeguards logic.

Turbine inlet pressure provides a pressure signal used in the following control systems:

- ESF Actuation System (ESFAS) High Steam Flow Setpoint Program
- 70 % Load Status Light
- ATWS Mitigation System Actuation Circuitry (AMSAC)
- T_{ref} signal to compare T_{avg} to T_{ref} for Steam Dump modulation
- 10% Load Drop (Load Rejection) setpoint to open air supply to Steam Dump Condenser Valves CV-3-2827 and CV-3-2828
- 30% Load Drop (Load Rejection) setpoint to open air supply to Steam Dump Condenser Valves CV-3-2829 and CV-3-2830
- Turbine Power Mismatch
- T_{avg}/T_{ref} deviation alarm

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- Rod Control circuit T_{avg}/T_{ref} comparison

Safety Injection (SI) – The High Steam Flow Steam Line Isolation setpoint is increased as normal steam flow increases. Since turbine inlet pressure is proportional to steam flow, it is used in the setpoint calculation circuit. High steam flow coincident with either low steam generator pressure or low T_{avg} is required to provide an SI actuation signal. The function is set for a delta pressure corresponding to 41.2% steam flow at a turbine load between 0% and 20%, then increasing linearly from 20% turbine load to a value corresponding to 114% steam flow at full load. (Reactor was never greater than 5%.)

The RPS uses power range detectors and turbine inlet pressure to allow RPS permissive P-7 to block reactor trips when reactor power and turbine load are less than 10%. RPS will also un-block reactor trip circuits when the turbine load is above 10%. (Reactor power was never greater than 5%.)

ATWS Mitigation System Actuation Circuitry – The Anticipated Transient Without Scram (ATWS) System senses turbine load through PT-3-446 and PT-3-447. Pressure modifiers PM-3-446A and PM-3-447A are part of this loop and the system is armed when the power level is greater than 40%. (Reactor power was never greater than 5%.)

Main Steam – PT-3-447 provides input to the arming signal for the steam dumps to condenser. PT-3-446 inputs into the actual program for opening the steam dumps.

Reactor Control System [EIIS: JD] - Turbine inlet pressure provides the reference temperature signal supplied to the Rod Control circuit. Rods are adjusted automatically (inserted only) to match T_{avg} to T_{ref}.

Reportability

The unplanned entry into Turkey Point Unit 3 TS 3.0.3 lasted from approximately 1140 on 8/25/12 to 1239 on 8/25/12. The ESFAS instrumentation associated with PT-3-446 and PT-3-467 was inoperable in Modes 3 and 2 for a time greater than the 6 hour limit with the channel(s) not tripped. This event exceeded TS 3.3.2, Table 3.3-2, Action 15 allowed outage time of 6 hours to address TS 3.3.2 Limiting Conditions for Operation, Table 3.3-2, Functional Units 1.f and 4.d not met. The condition is reportable in accordance with 10 CFR 50.73(a)(2)(i)(B).

ANALYSIS OF SAFETY SIGNIFICANCE

With the Unit 3 turbine inlet pressure transmitter manual isolation valves 3-10-292 and 3-10-294 in the closed position, the turbine Inlet pressure transmitters PT-3-446 and PT-3-447, were not capable of measuring changes in turbine load in Modes 2 and 3. The safety significance for the event is as follows:

- The ESFAS Hi steam flow signal is a fixed value equal to 41.2% of full steam flow when turbine load (measured by turbine inlet pressure) is less than 20%. Reactor power was never greater than 5%. Therefore, the Hi Steam Flow setpoint was unaffected and would have performed its intended function as assumed in the safety analysis.
- Reactor permissive P-7 is only required to be operable in Mode 1. Since the reactor was not in Mode 1 during the period the root isolation valves were closed, the low power reactor trip features would provide adequate core protection as assumed in the safety analysis.
- AMSAC circuitry is not required to be operable below turbine load of 40%. Since the reactor was not in Mode 1 during the period the root isolation valves were closed, turbine load was always less than 40%. Therefore, adequate core protection was provided with manual operator actions using existing Emergency Operating Procedures.
- The turbine bypass system would still reduce RCS temperature to no load conditions as designed since the isolated turbine inlet pressure transmitters provide zero turbine load signal.

Based on the above, there was no risk to personnel safety, radiological safety, or nuclear safety.

LICENSEE EVENT REPORT (LER) CONTINUATION SHEET

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CORRECTIVE ACTIONS

Corrective actions are in documented in Condition Report 1797236 and include the following:

1. Modify plant procedure to include specific guidance for making the impact of change determination in the CRN process.
2. Revise the CRN process to require an Operations approval signature on CRN which add new component, deletes/abandons/removes an existing component, or changes the normal position, plant location, nomenclature or parameter/setpoint of an existing component. Similar requirement will be added for the maintenance organization.
3. Enhance the process of identifying and tracking all documents affected by the design change.
4. Ensure the equivalent valves on Unit 4 for that unit EPU outage incorporates the respective valve lineup and are verified in their proper position prior to the required mode.

FAILED COMPONENTS IDENTIFIED: None

PREVIOUS SIMILAR EVENTS: None