FLORIDA POWER & LIGHT CO.

Turkey Point Units 3 & 4

Operability Evaluation

Control Rod Operability Evaluation as a Result of Incomplete Rod Insertion at Other Westinghouse Plants

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Safety Related

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REVIEW AND APPROVAL RECORD

PLANT Turkey Point UNIT 3 & 4

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REVIEW/APPROVAL:

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CIVIL			x				
NUC**		x					
CSI			x				
NUC FUEL	x						

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OTHER INTERFACES

Control Rod Operability Evaluation as a Result of Incomplete Rod Insertion at Other Westinghouse Plants

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1.0 Background

Between December 1995 and February 1996, three events involving stuck rod cluster control assemblies (RCCAs) occurred in Westinghouse Plants. This prompted the NRC to issue NRC Bulletin 96-01 (Reference 7.1) which addresses incomplete RCCA insertion.

1.1 South Texas Project

On December 18, 1995, South Texas Unit 1 experienced a turbine trip and a reactor trip from 100% Rated Thermal Power. While verifying control rod insertion, operators noted that the rod bottom lights of three control rod assemblies did not indicate full insertion; the digital rod position indication for each rod indicated six steps withdrawn. A step is equivalent to 1.59 cm [5/8 inch], and the top of the dashpot begins at 38 steps. One rod did drift into the fully inserted rod bottom position within 1 hour, and the other two rods were manually inserted later. During subsequent testing of all control rods in the affected banks, the rod position indication for the same three locations, as well as a new location, indicated six steps withdrawn. As compared to prior rod drop testing, no significant differences in rod drop times were noted before reaching the upper dashpot area for any of the control rods. Within 1 hour after the rod drop tests, two of the rods drifted to the rod bottom position and the other two were manually inserted. All four control rods were located in 17X17 XLR fuel assemblies that were in their third cycle, with burnup greater than 42,880 megawatt days per metric ton uranium (MWD/MTU).

1.2 Wolf Creek Plant

On January 30, 1996, after a manual scram from 80 percent power, five control rod assemblies at the Wolf Creek plant failed to insert fully. Two rods remained at 6 steps withdrawn, two at 12 steps, and one at 18 steps. At Wolf Creek, a step is equivalent to 1.59 cm [5/8 inch] and the top of the dashpot begins at approximately 30 steps. Three of the affected rods drifted to fully inserted within 20 minutes, one within 60 minutes, and the last one within 78 minutes. The results also indicate that there was some slowing down of affected rods before they reached the dashpot. After the scram, the licensee initiated emergency boration because all rods did not insert fully. During subsequent cold rod drop tests, the same five rods, plus an additional three rods, failed to fully insert. All of the affected rods were in 17x17 VANTAGE 5H fuel assemblies, with burnup greater than 47,600 MWD/MTU.

1.3 North Anna Plant

On February 21, 1996, during the insert shuffle in preparation for loading North Anna 1, Cycle 12, two new control rod assemblies could not be removed with normal operation of the handling tool from the fuel assemblies in the spent fuel pool in which they were temporarily stored. The control rod assemblies were removed using the rod assembly handling tool in conjunction with the bridge crane hoist. The two affected fuel assemblies were 17X17 VANTAGE 5H assemblies, which had achieved 47,782 MWD/MTU and 49,613 MWD/MTU burnup during two cycles of irradiation.

2.0 Description and Purpose

In Reference 7.1, the NRC requested that utilities "promptly determine the continued operability of control rods based on current information". This operability evaluation is intended to fulfill this requirement.

3.0 Licensing Requirements

The following provides the applicable licensing requirements for incomplete RCCA insertion.

3.1 Technical Specifications 3.1.1.1 requires that the shutdown margin be greater than or equal to 1.77 $\%\Delta\rho$ at End of Cycle (EOC).

3.2 Technical Specifications 3.1.3.1 requires that all full length rods shall be operable and positioned within $\pm$ 12 steps of the group step counter demand position within 1 hour after rod motion.

3.3 Technical Specifications 3.1.3.4 requires that the individual full length rod drop time from the fully withdrawn position shall be less than or equal to 2.4 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry.

4.0 Evaluation

Based on a review of Reference 7.1, the following conclusions are drawn:

a. The phenomena is associated with fuel assemblies having high exposures (> 40,000 MWD/MTU) and core locations where RCCAs reside.

b. The phenomena is so far isolated to 17X17 Westinghouse fuel assemblies. There has been no indication that this phenomena affects the Westinghouse 15X15 fuel assemblies used at Turkey Point.

Table 4.1 provides the fuel assembly exposures for Unit 3 Cycle 15, Unit 4 Cycle 15 and Unit 4 Cycle 16 in the core locations where the RCCAs reside.

4.1 Unit 3 Cycle 15

Based on Table 4.1, Unit 3 Cycle 15 does not currently have any fuel assembly with exposure greater than 40,000 MWD/MTU residing in RCCA locations. The Unit tripped from 60% power on 2/9/96 (cycle burnup of approximately 3600 MWD/MTU) with all RCCAs fully inserting into the core.

At EOC 15, Unit 3 is projected to have 13 RCCAs that will reside in fuel assemblies with exposures greater than 40,000 MWD/MTU. These include the center RCCA in CBD, 4 RCCAs in SBB and 8 RCCAs in CBC.

4.2 Unit 4 Cycle 15

Unit 4 Cycle 15 is in a refueling outage with all RCCA fully inserted. At the EOC, the Unit had 21 RCCAs residing in fuel assemblies with exposures greater than 40,000 MWD/MTU. These included 4 RCCAs in SBB, 8 RCCAs in CBA, 8 RCCAs in CBC and the center RCCA in CBD with a fuel assembly exposure of approximately 50,800 MWD/MTU. During the shutdown sequence on 3/4/96, plant management instructed the operators to perform a trip test of the RCCAs. The test was performed with CBD at 74 steps withdrawn, CBC at 202 steps withdrawn and the remaining of the RCCAs fully withdrawn. The results of the test indicated that all RCCAs fully inserted.

4.3 Unit 4 Cycle 16

After the refueling outage, Unit 4 Cycle 16 will initially operate with no RCCA residing in high exposure fuel assembly locations. At the EOC only 5 RCCAs are projected to reside in fuel assemblies with exposures greater than 40,000 MWD/MTU. These include 4 RCCAs in SBB and the center RCCA in CBD with assembly exposure of 50,100 MWD/MTU.

For Turkey Point, the top of the dashpot is located approximately 24" from the top of the bottom nozzle (Reference 7.2). Based on Reference 7.3, this distance corresponds to approximately 28 steps withdrawn. Reference 7.3 determined, assuming that all RCCAs residing in fuel assemblies with exposures greater than 40,000 MWD/MTU get stuck at 28 steps withdrawn (200 steps inserted), the impact on EOL shutdown margin is less than 100 pcm for Unit 3 Cycle 15 and Unit 4 Cycle 16. A reduction of 100 pcm is reasonable because both Units have 6" natural uranium blankets at the bottom of the fuel rods. In addition, at HZP the axial power shape is top peaked resulting in minimum worth of the RCCAs in the bottom of the core.

Using this result and the shutdown margin results from References 7.4 and 7.5, Table 4.2 was developed. Table 4.2 indicates that the Technical Specifications shutdown margin is maintained even after conservatively assuming that RCCAs residing in fuel assemblies with exposures greater than 40,000 MWD/MTU remained 28 steps withdrawn.

TABLE 4.1

Turkey Point Unit 3 Cycle 15

Core Location	Rod Bank	Type of Fuel	Current Burnup (as of 3/10/96) (MWD/MTU)	Projected EOC Burnup (MWD/MTU)	Burnup at Last Scram (2/9/96) (MWD/MTU)	Observations
E-05	SBB	OFA/DRFA	23,330	37,500	22,500	Full Insertion
L-05	SBB	OFA/DRFA	23,330	37,500	22,500	Full Insertion
L-11	SBB	OFA/DRFA	23,330	37,500	22,500	Full Insertion
E-11	SBB	OFA/DRFA	23,330	37,500	22,500	Full Insertion
F-08	SBB	OFA/DRFA	35,300	46,900	34,600	Full Insertion
H-06	SBB	OFA/DRFA	35,300	46,900	34,600	Full Insertion
K-08	SBB	OFA/DRFA	35,300	46,900	34,600	Full Insertion
H-10	SBB	OFA/DRFA	35,300	46,900	34,600	Full Insertion
C-07	SBA	OFA/DRFA	20,800	34,000	20,000	Full Insertion
G-03	SBA	OFA/DRFA	20,800	34,000	20,000	Full Insertion
J-03	SBA	OFA/DRFA	20,800	34,000	20,000	Full Insertion
N-07	SBA	OFA/DRFA	20,800	34,000	20,000	Full Insertion
N-09	SBA	OFA/DRFA	20,800	34,000	20,000	Full Insertion
J-13	SBA	OFA/DRFA	20,800	34,000	20,000	Full Insertion
G-13	SBA	OFA/DRFA	20,800	34,000	20,000	Full Insertion
C-09	SBA	OFA/DRFA	20,800	34,000	20,000	Full Insertion
E-07	CBA	OFA/DRFA	23,600	37,400	22,900	Full Insertion
G-05	CBA	OFA/DRFA	23,600	37,400	22,900	Full Insertion
J-05	CBA	OFA/DRFA	23,600	37,400	22,900	Full Insertion
L-07	CBA	OFA/DRFA	23,600	37,400	22,900	Full Insertion
L-09	CBA	OFA/DRFA	23,600	37,400	22,900	Full Insertion
J-11	CBA	OFA/DRFA	23,600	37,400	22,900	Full Insertion
G-11	CBA	OFA/DRFA	23,600	37,400	22,900	Full Insertion
E-09	CBA	OFA/DRFA	23,600	37,400	22,900	Full Insertion
B-06	CBB	OFA/DRFA	18,300	26,800	17,800	Full Insertion
F-02	CBB	OFA/DRFA	18,300	26,800	17,800	Full Insertion
K-02	CBB	OFA/DRFA	18,300	26,800	17,800	Full Insertion
P-06	CBB	OFA/DRFA	18,300	26,800	17,800	Full Insertion
P-10	CBB	OFA/DRFA	18,300	26,800	17,800	Full Insertion
K-14	CBB	OFA/DRFA	18,300	26,800	17,800	Full Insertion
F-14	CBB	OFA/DRFA	18,300	26,800	17,800	Full Insertion
B-10	CBB	OFA/DRFA	18,300	26,800	17,800	Full Insertion
D-06	CBC	OFA/DRFA	30,700	43,100	30,000	Full Insertion
F-04	CBC	OFA/DRFA	30,700	43,100	30,000	Full Insertion
K-04	CBC	OFA/DRFA	30,700	43,100	30,000	Full Insertion
M-06	CBC	OFA/DRFA	30,700	43,100	30,000	Full Insertion
M-10	CBC	OFA/DRFA	30,700	43,100	30,000	Full Insertion
K-12	CBC	OFA/DRFA	30,700	43,100	30,000	Full Insertion
F-12	CBC	OFA/DRFA	30,700	43,100	30,000	Full Insertion
D-10	CBC	OFA/DRFA	30,700	43,100	30,000	Full Insertion
D-08	CBD	OFA/DRFA	23,400	36,000	22,700	Full Insertion
H-04	CBD	OFA/DRFA	23,400	36,000	22,700	Full Insertion
M-08	CBD	OFA/DRFA	23,400	36,000	22,700	Full Insertion
H-12	CBD	OFA/DRFA	23,400	36,000	22,700	Full Insertion
H-08	CBD	OFA/DRFA	38,200	49,400	37,600	Full Insertion

TABLE 4.1

Turkey Point Unit 4 Cycle 15

Core Location	Rod Bank	Type of Fuel	Current Burnup (as of 3/10/96) (MWD/MTU)	Approximate EOC Burnup (MWD/MTU)	Burnup at Trip Test (3/4/96) (MWD/MTU)	Observations
E-05	SBB	OFA/DRFA	In Refueling	43,800	43,800	Full Insertion
L-05	SBB	OFA/DRFA	In Refueling	43,800	43,800	Full Insertion
L-11	SBB	OFA/DRFA	In Refueling	43,800	43,800	Full Insertion
E-11	SBB	OFA/DRFA	In Refueling	43,800	43,800	Full Insertion
F-08	SBB	OFA/DRFA	In Refueling	34,300	34,300	Full Insertion
H-06	SBB	OFA/DRFA	In Refueling	34,300	34,300	Full Insertion
K-08	SBB	OFA/DRFA	In Refueling	34,300	34,300	Full Insertion
H-10	SBB	OFA/DRFA	In Refueling	34,300	34,300	Full Insertion
C-07	SBA	OFA/DRFA	In Refueling	31,600	31,600	Full Insertion
G-03	SBA	OFA/DRFA	In Refueling	31,600	31,600	Full Insertion
J-03	SBA	OFA/DRFA	In Refueling	31,600	31,600	Full Insertion
N-07	SBA	OFA/DRFA	In Refueling	31,600	31,600	Full Insertion
N-09	SBA	OFA/DRFA	In Refueling	31,600	31,600	Full Insertion
J-13	SBA	OFA/DRFA	In Refueling	31,600	31,600	Full Insertion
G-13	SBA	OFA/DRFA	In Refueling	31,600	31,600	Full Insertion
C-09	SBA	OFA/DRFA	In Refueling	31,600	31,600	Full Insertion
E-07	CBA	OFA/DRFA	In Refueling	42,000	42,000	Full Insertion
G-05	CBA	OFA/DRFA	In Refueling	42,000	42,000	Full Insertion
J-05	CBA	OFA/DRFA	In Refueling	42,000	42,000	Full Insertion
L-07	CBA	OFA/DRFA	In Refueling	42,000	42,000	Full Insertion
L-09	CBA	OFA/DRFA	In Refueling	42,000	42,000	Full Insertion
J-11	CBA	OFA/DRFA	In Refueling	42,000	42,000	Full Insertion
G-11	CBA	OFA/DRFA	In Refueling	42,000	42,000	Full Insertion
E-09	CBA	OFA/DRFA	In Refueling	42,000	42,000	Full Insertion
B-06	CBB	OFA/DRFA	In Refueling	26,000	26,000	Full Insertion
F-02	CBB	OFA/DRFA	In Refueling	26,000	26,000	Full Insertion
K-02	CBB	OFA/DRFA	In Refueling	26,000	26,000	Full Insertion
P-06	CBB	OFA/DRFA	In Refueling	26,000	26,000	Full Insertion
P-10	CBB	OFA/DRFA	In Refueling	26,000	26,000	Full Insertion
K-14	CBB	OFA/DRFA	In Refueling	26,000	26,000	Full Insertion
F-14	CBB	OFA/DRFA	In Refueling	26,000	26,000	Full Insertion
B-10	CBB	OFA/DRFA	In Refueling	26,000	26,000	Full Insertion
D-06	CBC	OFA/DRFA	In Refueling	47,900	47,900	Full Insertion
F-04	CBC	OFA/DRFA	In Refueling	47,900	47,900	Full Insertion
K-04	CBC	OFA/DRFA	In Refueling	47,900	47,900	Full Insertion
M-06	CBC	OFA/DRFA	In Refueling	47,900	47,900	Full Insertion
M-10	CBC	OFA/DRFA	In Refueling	47,900	47,900	Full Insertion
K-12	CBC	OFA/DRFA	In Refueling	47,900	47,900	Full Insertion
F-12	CBC	OFA/DRFA	In Refueling	47,900	47,900	Full Insertion
D-10	CBC	OFA/DRFA	In Refueling	47,900	47,900	Full Insertion
D-08	CBD	OFA/DRFA	In Refueling	34,000	34,000	Full Insertion
H-04	CBD	OFA/DRFA	In Refueling	34,000	34,000	Full Insertion
M-08	CBD	OFA/DRFA	In Refueling	34,000	34,000	Full Insertion
H-12	CBD	OFA/DRFA	In Refueling	34,000	34,000	Full Insertion
H-08	CBD	OFA	In Refueling	50,800	50,800	Full Insertion

TABLE 4.1

Turkey Point Unit 4 Cycle 16

Core Location	Rod Bank	Type of Fuel	Burnup at BOC (MWD/MTU)	Projected EOC Burnup (MWD/MTU)	Burnup at Last Scram (MWD/MTU)	Observations
E-05	SBB	OFA/DRFA	30,400	46,600	N/A	N/A
L-05	SBB	OFA/DRFA	30,400	46,600	N/A	N/A
L-11	SBB	OFA/DRFA	30,400	46,600	N/A	N/A
E-11	SBB	OFA/DRFA	30,400	46,600	N/A	N/A
F-08	SBB	OFA/DRFA	18,100	38,000	N/A	N/A
H-06	SBB	OFA/DRFA	18,100	38,000	N/A	N/A
K-08	SBB	OFA/DRFA	18,100	38,000	N/A	N/A
H-10	SBB	OFA/DRFA	18,100	38,000	N/A	N/A
C-07	SBA	OFA/DRFA	15,500	34,500	N/A	N/A
G-03	SBA	OFA/DRFA	15,500	34,500	N/A	N/A
J-03	SBA	OFA/DRFA	15,500	34,500	N/A	N/A
N-07	SBA	OFA/DRFA	15,500	34,500	N/A	N/A
N-09	SBA	OFA/DRFA	15,500	34,500	N/A	N/A
J-13	SBA	OFA/DRFA	15,500	34,500	N/A	N/A
G-13	SBA	OFA/DRFA	15,500	34,500	N/A	N/A
C-09	SBA	OFA/DRFA	15,500	34,500	N/A	N/A
E-07	CBA	OFA/DRFA	18,100	37,400	N/A	N/A
G-05	CBA	OFA/DRFA	18,100	37,400	N/A	N/A
J-05	CBA	OFA/DRFA	18,100	37,400	N/A	N/A
L-07	CBA	OFA/DRFA	18,100	37,400	N/A	N/A
L-09	CBA	OFA/DRFA	18,100	37,400	N/A	N/A
J-11	CBA	OFA/DRFA	18,100	37,400	N/A	N/A
G-11	CBA	OFA/DRFA	18,100	37,400	N/A	N/A
E-09	CBA	OFA/DRFA	18,100	37,400	N/A	N/A
B-06	CBB	OFA/DRFA	17,600	29,100	N/A	N/A
F-02	CBB	OFA/DRFA	17,600	29,100	N/A	N/A
K-02	CBB	OFA/DRFA	17,600	29,100	N/A	N/A
P-06	CBB	OFA/DRFA	17,600	29,100	N/A	N/A
P-10	CBB	OFA/DRFA	17,600	29,100	N/A	N/A
K-14	CBB	OFA/DRFA	17,600	29,100	N/A	N/A
F-14	CBB	OFA/DRFA	17,600	29,100	N/A	N/A
B-10	CBB	OFA/DRFA	17,600	29,100	N/A	N/A
D-06	CBC	OFA/DRFA	17,800	36,500	N/A	N/A
F-04	CBC	OFA/DRFA	17,800	36,500	N/A	N/A
K-04	CBC	OFA/DRFA	17,800	36,500	N/A	N/A
M-06	CBC	OFA/DRFA	17,800	36,500	N/A	N/A
M-10	CBC	OFA/DRFA	17,800	36,500	N/A	N/A
K-12	CBC	OFA/DRFA	17,800	36,500	N/A	N/A
F-12	CBC	OFA/DRFA	17,800	36,500	N/A	N/A
D-10	CBC	OFA/DRFA	17,800	36,500	N/A	N/A
D-08	CBD	OFA/DRFA	18,100	36,600	N/A	N/A
H-04	CBD	OFA/DRFA	18,100	36,600	N/A	N/A
M-08	CBD	OFA/DRFA	18,100	36,600	N/A	N/A
H-12	CBD	OFA/DRFA	18,100	36,600	N/A	N/A
H-08	CBD	OFA/DRFA	34,500	50,100	N/A	N/A

OFA = 15x15 Optimized Fuel Assembly
DRFA = Debris Resistant Fuel Assembly (FPL Design)

TABLE 4.2

SHUTDOWN REQUIREMENTS AND MARGINS

Control Rod Worth (%Δρ)	Unit 3 Cycle 15 EOL	Unit 4 Cycle 16 EOL
All Rods Inserted Less Worst Stuck Rod	6.23	6.12
(1) Less 7%	5.79	5.69
Control Rod Requirements (%Δρ)		
Reactivity Defects (Doppler, Tavg Void, Redistribution)	2.78	2.75
Rod Insertion Allowance	0.50	0.50
RCCAs Incomplete Insertion	0.10	0.10
(2) Total Requirements	3.38	3.35
Shutdown Margin (1)-(2) (%Δρ)	2.41	2.34
Required Shutdown Margin (%Δρ)	1.77	1.77
Excess Shutdown Margin (%Δρ)	0.64	0.57

5.0 Safety Analysis

In addition to shutdown margin, the impact on Safety Analysis needs to be considered.

5.1 Uncontrolled RCCA Withdrawal from Sub-critical

A trip reactivity of 1.5% $\Delta\rho$ is assumed in this analysis with the worst stuck rod assumed. The transient is not sensitive to small changes in trip reactivity since the transient is essentially turned around as a result of the Doppler defect.

5.2 Uncontrolled RCCA Withdrawal at Power

A trip reactivity of 4.0% $\Delta\rho$ is assumed in this analysis with the worst stuck rod assumed. The transient is not sensitive to small changes in trip reactivity.

5.3 RCCA Misoperation (Dropped)

No impact.

5.4 CVCS Malfunction

For a boron dilution event the reduction in rod worth can increase the required boron concentration. However, this event is limiting at BOC and not at EOC.

5.5 Feedwater System Malfunction

A trip reactivity of 4.0% $\Delta\rho$ is assumed in this analysis with the worst stuck rod assumed. The transient is not sensitive to small changes in trip reactivity.

5.6 Excessive Increase in Secondary Steam Flow

No impact.

5.7 Partial / Complete Loss of Forced Reactor Coolant Flow

A trip reactivity of 4.0% $\Delta\rho$ is assumed in this analysis with the worst stuck rod assumed. The transient is not sensitive to small changes in trip reactivity.

5.8 Locked Rotor

A trip reactivity of 4.0% $\Delta\rho$ is assumed in this analysis with the worst stuck rod assumed. The transient is not sensitive to small changes in trip reactivity.

5.9 Loss of Load and/or Turbine Trip

A trip reactivity of 4.0% $\Delta\rho$ is assumed in this analysis with the worst stuck rod assumed. The transient is not sensitive to small changes in trip reactivity.

5.10 Loss of Normal Feedwater Flow

A trip reactivity of 4.0% $\Delta\rho$ is assumed in this analysis with the worst stuck rod assumed. The transient is not sensitive to small changes in trip reactivity.

5.11 Rupture of Steam Pipe

The most important parameter is the assumed 1.77%$\Delta\rho$ shutdown margin and as discussed in Section 4.0. This margin is not challenged by the postulated reduction in rod worth.

5.12 RCCA Ejection

The transient assumed a worst stuck rod for trip reactivity. The transient is not sensitive to small changes in trip reactivity since the transient is essentially turned around as a result of the Doppler defect.

5.13 Large and Small LOCAs

No impact since no credit is taken for RCCAs.

6.0 Conclusions

Based on the previous analysis, the following conclusions can be drawn.

6.1 The impact of the uninserted worth on shutdown margin is small (< 100 pcm).

6.2 Shutdown margin continues to be met in the event that the RCCAs residing in fuel assemblies with exposures greater than 40,000 MWD/MTU only reached 28 steps withdrawn (200 steps inserted) rather than fully inserted. This is applicable to Unit 3 Cycle 15 and Unit 4 Cycle 16.

6.3 A RCCA trip test performed at the EOC for Unit 4 Cycle 15 showed no indication of incomplete rod insertion. For this cycle, there were 21 RCCAs residing in fuel assemblies with exposures greater than 40,000 MWD/MTU. Worth noting is that the center RCCA resided in a fuel assembly with 50,800 MWD/MTU at the time of the trip test. This seem to indicate that the phenomena experienced in 17X17 Westinghouse fuel assemblies is not manifested in 15X15 fuel assemblies. It is judged that the results of this test are applicable to Unit 3 Cycle 15 due to the identical fuel designs (see Table 4.1).

6.4 The current safety analyses will remain valid for the kinds of trip scenarios that could be postulated to occur.

In summary, the RCCAs remain operable and continued operation is acceptable.

7.0 References

7.1 NRC Bulletin 96-01, "Control Rod Insertion Problems," March 8, 1996.

7.2 Westinghouse Drawing 2D32938, "Zircaloy Single Dashpot Guide Thimble Tube," Revision 29.

7.3 JPN Calculation PTN-BFJF-96-066, "Shutdown Margin Assessment with Incomplete Rod Insertion," Revision 0, Approved 3/12/96.

7.4 PC/M 94-134, "Turkey Point Unit 3 Cycle 15 Reload," Revision 1.

7.5 PC/M 95-066, "Turkey Point Unit 4 Cycle 16 Reload," Revision 2.

FLORIDA POWER AND LIGHT COMPANY
TURKEY POINT NUCLEAR
UNIT 3, CYCLE 15

FIGURE 1: CORE CONFIGURATION

15	14	13	12	11	10	9	8	7	6	5	4	3	2	1	
						EE05 HF-08	EE08 HF-04	EE15 HF-24							R
				EE41	FF31 R-04	GG38	FF21	GG40	FF30 R-05	EE14					P
			EE28	GG46	GG54	FF52 R-15	GG33	FF43 R-06	GG56	GG48	EE23				N
		EE27	FF49	GG03 12P337WZ	EE45 R-02	FF03	FF16 R-03	FF06	EE51 R-27	GG10 12P343WZ	FF48	EE29			M
	EE06	GG49	GG13 12P344WZ	FF18 R-09	GG23 16P52WZ	FF33 R-18	GG24 12P347WZ	FF39 R-11	GG02 16P41WZ	FF22 R-12	GG31 12P351WZ	GG50	EE38		L
	FF32 R-37	GG57	EE47 R-40	GG05 16P42WZ	EE56	GG18 16P48WZ	EE33 R-14	GG19 16P49WZ	EE55	GG15 16P45WZ	EE52 R-35	GG58	FF27 R-39		K
EE01 HF-26	GG41	FF44 R-10	FF08	FF37 R-42	GG16 16P46WZ	FF14	GG09 12P342WZ	FF04	GG12 16P44WZ	FF25 R-17	FF02	FF42 R-07	GG42	EE09 HF-12	J
EE04 HF-14	FF24	GG34	FF10 R-16	GG01 12P336WZ	EE43 R-46	GG22 12P346WZ	EE03 R-41	GG08 12P341WZ	EE35 R-36	GG07 12P340WZ	FF15 R-22	GG35	FF20	EE07 HF-25	H
EE20 HF-03	GG43	FF47 R-08	FF09	FF36 R-33	GG29 16P54WZ	FF11	GG27 12P349WZ	FF05	GG21 16P51WZ	FF40 R-26	FF07	FF46 R-24	GG37	EE02 HF-17	G
	FF35 R-100	GG59	EE48 R-96	GG32 16P56WZ	EE54	GG20 16P50WZ	EE44 R-20	GG17 16P47WZ	EE53	GG26 16P53WZ	EE46 R-10A	GG53	FF34 R-29		F
	EE36	GG51	GG06 12P339WZ	FF23 R-28	GG30 16P55WZ	FF38 R-23	GG04 12P338WZ	FF29 R-45	GG11 16P43WZ	FF19 R-97	GG25 12P348WZ	GG45	EE16		E
		EE25	FF50	GG14 12P345WZ	EE50 R-19	FF01	FF12 R-44	FF13	EE49 R-21	GG28 12P350WZ	FF45	EE22			D
			EE31	GG52	GG60	FF51 R-32	GG36	FF41 R-43	GG55	GG47	EE30				C
				EE11	FF28 R-38	GG44	FF17	GG39	FF26 R-01	EE37					B
						EE17 HF-19	EE19 HF-21	EE10 HF-18							A

FIGURE 1

TURKEY POINT NUCLEAR
UNIT 4 – CYCLE 15
CORE LOADING

↑
N

	15	14	13	12	11	10	9	8	7	6	5	4	3	2	1
R							RR35 HF01	RR13 HF10	RR40 HF11						
P					SS29	TT42 R61	UU46	UU33	UU48	TT41 R76	SS41				
N				SS25	UU26	UU38 20P42WZ	TT47 R64	SS12	TT52 R73	UU40 20P44WZ	UU28	SS24			
M			SS23	TT22	UU03 12P328WZ	SS09 R59	TT09	TT20 R68	TT10	SS10 R77	UU08 12P330WZ	TT28	SS22		
L		SS32	UU29	UU10 12P332WZ	SS45 R58	UU16 20P35WZ	SS35 R65	UU17 20P36WZ	SS33 R75	UU02 20P26WZ	SS46 R81	UU20 12P335WZ	UU30	SS44	
K		TT44 R51	UU41 20P45WZ	SS08 R55	UU04 20P27WZ	TT36	TT03	TT23 R69	TT06	TT35	UU13 20P32WZ	SS06 R86	UU42 20P46WZ	TT39 R91	
J	RR33 HF02	UU49	TT45 R52	TT14	SS38 R66	TT16	TT29	UU21 20P37WZ	TT25	TT04	SS37 R82	TT08	TT51 R89	UU50	RR39 HF20
H	RR03 HF05	UU34	SS03	TT21 R54	UU14 20P33WZ	TT26 R60	UU22 20P38WZ	RR05 R70	UU23 20P39WZ	TT31 R78	UU11 20P30WZ	TT30 R87	SS13	UU35	RR04 HF15
G	RR38 HF13	UU51	TT50 R109	TT07	SS30 R84	TT11	TT18	UU24 20P40WZ	TT19	TT12	SS43 R83	TT05	TT49 R90	UU45	RR37 HF07
F		TT38 R53	UU43 20P47WZ	SS01 R56	UU12 20P31WZ	TT34	TT01	TT24 R71	TT13	TT33	UU07 20P29WZ	SS07 R88	UU37 20P41WZ	TT40 R92	
E		SS40	UU31	UU09 12P331WZ	SS47 R57	UU01 20P25WZ	SS42 R101	UU15 20P34WZ	SS36 R93	UU06 20P28WZ	SS48 R85	UU05 12P329WZ	UU25	SS39	
D			SS20	TT32	UU19 12P334WZ	SS15 R62	TT15	TT27 R72	TT02	SS02 R79	UU18 12P333WZ	TT17	SS18		
C				SS26	UU32	UU44 20P48WZ	TT48 R67	SS16	TT46 R74	UU39 20P43WZ	UU27	SS21			
B					SS31	TT37 R63	UU52	UU36	UU47	TT43 R80	SS34				
A							RR36 HF06	RR07 HF16	RR34 HF23						

Key:
RR## Reload Cycle 12
SS## Reload Cycle 13
TT## Reload Cycle 14
UU## Feed Cycle 15
R## Control Rod
HF## Hafnium Insert
##P###WZ Waba Insert

LEGEND:

---	ASSEMBLY ID.
---	INSERTS

ATTACHMENT 7
(Page 1 of 1)

REACTOR FUEL LOCATION DIAGRAM
TURKEY POINT UNIT NO. 4
CYCLE NO. 16

	15	14	13	12	11	10	9	8	7	6	5	4	3	2	1
R							TT15 HF23	TT21 HF16	TT02 HF06						
P					TT29	UU42 R76	VV39	UU34	VV44	UU41 R61	TT23				
N				TT45	VV46	VV55 16P62WZ	UU51 R77	VV61 16P65WZ	UU45 R64	VV56 16P63WZ	VV38	TT51			
M			TT47	VV03	VV08 12P357WZ	UU10 R73	UU31	UU04 R51	UU25	UU20 R59	VV10 12P359WZ	VV16	TT52		
L		TT26	VV59	VV17 12P365WZ	TT17 R81	VV02 12P353WZ	UU17 R75	VV21 12P368WZ	UU22 R55	VV20 12P367WZ	TT32 R65	VV04 12P354WZ	VV40	TT25	
K		UU44 R69	VV50 16P58WZ	UU03 R86	VV13 12P362WZ	TT43	VV26 8P225WZ	UU16 R60	VV28 8P227WZ	TT38	VV14 12P363WZ	UU08 R66	VV51 16P59WZ	UU39 R90	
J	TT08 HF07	VV47	UU48 R91	UU28	UU24 R74	VV29 8P228WZ	TT40	VV34 8P233WZ	TT37	VV35 8P234WZ	UU11 R58	UU26	UU46 R52	VV48	TT14 HF13
H	TT27 HF15	UU36	VV62 16P66WZ	UU01 R93	VV22 12P369WZ	UU12 R72	VV25 8P224WZ	TT13 R109	VV36 8P235WZ	UU13 R79	VV23 12P370WZ	UU02 R53	VV63 16P67WZ	UU33	TT20 HF05
G	TT05 HF20	VV43	UU47 R92	UU27	UU14 R85	VV27 8P226WZ	TT41	VV32 8P231WZ	TT44	VV33 8P232WZ	UU21 R62	UU32	UU52 R101	VV45	TT07 HF02
F		UU38 R71	VV37 16P57WZ	UU19 R88	VV11 12P360WZ	TT39	VV30 8P229WZ	UU06 R63	VV31 8P230WZ	TT42	VV12 12P361WZ	UU18 R67	VV54 16P61WZ	UU40 R57	
E		TT18	VV42	VV07 12P356WZ	TT28 R87	VV09 12P358WZ	UU23 R83	VV24 12P371WZ	UU15 R80	VV01 12P352WZ	TT22 R84	VV15 12P364WZ	VV41	TT31	
D			TT48	VV06	VV05 12P355WZ	UU09 R78	UU29	UU07 R68	UU30	UU05 R70	VV19 12P366WZ	VV18	TT46		
C				TT50	VV58	VV57 16P64WZ	UU49 R54	VV64 16P68WZ	UU50 R56	VV53 16P60WZ	VV60	TT49			
B					TT24	UU37 R89	VV52	UU35	VV49	UU43 R82	TT19				
A							TT09 HF11	TT30 HF10	TT10 HF01						

*/AOC/nn/bvc/ev

TURKEY POINT FUEL ASSEMBLY DESIGN

Description	
Fuel Assembly Array/Design	15 x 15 Debris Resistant Fuel Assembly (Optimized Fuel Assembly)
Fuel Rod Material	Zircaloy
Spacer Grid Material	Top and Bottom Grids: Inconel Intermediate Grids: Zircaloy
Guide Thimble Material	Zircaloy
Guide Thimble Inside Diameter	Above Dashpot - 0.499 in. Below Dashpot - 0.455 in. Length of the Dashpot - 23.245 in.
Length of Guide Thimbles	152.970 in.

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Turkey Point Unit 4 - Reportable Event: 2000-001-00, on January 24, 2000, Manual React or Trip due to Main Feedwater Flow Control Valve Cage Disengagement

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L-2000-043
10 CFR § 50.73

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
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Re: Turkey Point Unit 4
Docket No. 50-251
Reportable Event: 2000-001-00
Date of Event: January 24, 2000
Manual Reactor Trip due to Main Feedwater Flow Control Valve Cage Disengagement

The attached Licensee Event Report 2000-001 is being submitted pursuant to the requirements of 10 CFR § 50.73 to provide notification of the subject event.

If there are any questions, please contact us.

Very truly yours,

R. J. Hovey
Vice President
Turkey Point Nuclear Plant

RJH/SM
Attachment

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

ML003687003

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LICENSEE EVENT REPORT (LER)

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FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Turkey Point Unit 4	05000251	Page 1 of 7

TITLE (4)

Manual Reactor Trip due to Main Feedwater Flow Control Valve Cage Disengagement

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 NAME	DOCKET NUMBER
01	24	2000	2000 -	001 -	00	02	23	2000	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	1
POWER LEVEL (10)	95

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

	20.2201(b)		20.2203(a)(2)(v)		50.73(a)(2)(i)		50.73(a)(2)(viii)
	20.2203(a)(1)		20.2203(a)(3)(i)		50.73(a)(2)(ii)		50.73(a)(2)(x)
	20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
	20.2203(a)(2)(ii)		20.2203(a)(4)	x	50.73(a)(2)(iv)		OTHER
	20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Stavroula Mihalakea, Licensing Engineer	(305) 246 - 6454

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX
D	SJ	FCV	C635	Y	-	-	-	-	-
-	-	-	-	-	-	-	-	-	-

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

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

At approximately 7:30 AM on January 24, 2000, FPL's Turkey Point Unit 4 reduced power to 95% to investigate main feedwater flow instabilities caused by the "A" Steam Generator (SG) Feedwater Flow Control Valve, FCV-4-478. At approximately 11:14 AM feedwater flow appeared to increase causing a SG level deviation. The Operators placed FCV-4-478 in manual operation. A preliminary determination of valve internal problems versus control problems resulted in the decision to shut down the Unit by performing a fast load reduction. At approximately 11:42 AM, the Reactor Control Operator (RCO) manually tripped the reactor due to difficulty in controlling SG levels. All rods were fully inserted and all systems except Feedwater functioned as designed.

The immediate cause of the reactor trip was a manual action taken by the RCO in response to Main Feedwater flow instabilities. The underlying cause of the trip was a failure in FCV-4-478 valve internals. The valve cage had disengaged from the valve body web. The root cause of the failure of FCV-4-478 is inadequate change management in the 1980's when the practice of periodic replacement of the cage was stopped; specifically, FPL failed to require periodic re-torque of a re-used FCV cage.

FCV-4-478 was repaired. FPL established inspection controls to monitor for signs of valve degradation. Cage torque will be verified for all feedwater FCV at the next opportunity.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Turkey Point Unit 4	05000251	2000	- 001 -	00	Page 2 of 7

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

Event Description

On January 24, 2000, FPL's Turkey Point Unit 4 was operating at 100% power.

At approximately 6:00 AM, the Feedwater Flow Control Valve (FCV) [SJ:fcv] to the "A" Steam Generator (SG)[AB:sg], FCV-4-478, was at 100% demand with SG level decreasing. The Reactor Control Operator (RCO) reduced SG blowdown [WI] and started the third condensate pump [KA:p] to recover SG level. As a result of a conservative management decision to provide additional operating margin, the RCO commenced a load reduction to 95% power and an Event Response Team (ERT) was formed. The Unit reached 95% power successfully, and FCV-4-478 seemed to provide stable SG level control.

At approximately 11:14 AM, while the ERT was investigating the source of earlier flow instabilities, FCV-4-478 appeared to initiate another flow transient, causing flow instabilities in all SGs. The RCO placed FCV-4-478 in manual and stabilized levels in all SGs. However, FCV-4-478 indicated deteriorating flow control stability. A preliminary field determination of valve internal problems versus control loop problems resulted in the decision to reduce power by using Off Normal Operating Procedure 4-ONOP-100, Fast Load Reduction. At approximately 11:42 AM, the RCO manually tripped the reactor due to difficulty in controlling SG levels.

The manual reactor trip was initiated in Mode 1 at 95% power with automatic reactor coolant system (RCS) [AB] pressure control operational. All rods [AA:rod] were fully inserted and all systems other than Feedwater functioned as designed. The Main Turbine [TA] automatically tripped in response to the manual reactor trip. The SG "A" and "B" Feedwater flow control valves were taken to manual prior to the reactor trip in response to unstable level control. Following the reactor trip, a feedwater isolation signal was generated on reactor trip with low RCS average temperature of 554 degrees F, as expected. FCV-4-478 did not fully isolate for approximately 100 seconds, allowing approximately 10% of the nominal feedwater flow into the "A" SG. In accordance with 4-EOP-E-0, Reactor Trip or Safety Injection, the RCO closed the Feedwater Isolation Valve MOV-4-1407 [SJ:isv], and terminated the FCV leakage flow. All SG levels were restored to desired levels.

A walkdown was performed on the affected piping and components associated with valve FCV-4-478. A Feedwater Flow Transmitter [JB:ft] tube associated with the SG "B" loop was broken off at the interface of the 3/8 inch port connector [JB:ft,con] with a 3/4-inch x 3/8-inch adapter. The port connector failure likely occurred as a result of the nearby SG "A" piping deflections during FCV-4-478 flow instabilities. FPL found no further evidence of damage to any other major components (piping/supports). This was determined from the observation of no insulation damage, no bent or misaligned supports, and no evidence of excessive movement.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Turkey Point Unit 4	05000251	2000	- 001	- 00	Page 3 of 7

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Background

The Turkey Point Unit 4 Main Feedwater System consists of two SG feedwater pumps [SJ:p], two high pressure feedwater heaters [SJ:hx], the Feedwater FCVs and the controls associated with these components. Feedwater leaving the high pressure feedwater heaters splits into three feedwater headers, which supply feedwater to the three SGs. The Feedwater FCV, one for each SG, controls flow to each SG. Upstream of each FCV is a motor operated valve, which is the feedwater isolation valve. Normally the FCV controller will be in "Auto" with power between 15-100%. During plant operations, the FCV maintains a programmed level of water in the steam generator by controlling feedwater flow to the SG depending upon the steam flow demand and actual level in the SG.

On December 25, 1999, Operations discovered that with Unit 4 at 100% power, the demand for FCV-4-478 was between 98% and 100%, while the demand for the FCVs on the other two SGs was about 90%. A condition report was initiated to document a high demand condition identified for FCV-4-478. Investigation was underway to determine the validity of the demand and to isolate the problem to either the control system or the valve. Review of maintenance and calibration records, field inspections, measurements of feedwater flow for all FCVs (for comparison purposes), and an examination of performance data had been completed without identifying any anomaly associated with FCV-4-478. Additional investigation was conducted for other potential flow restrictions, including feedwater isolation MOV and check valves, and for bypass flow paths or undocumented demand. No problem was identified. However, the investigation confirmed that FCV-4-478 continued to adequately maintain SG levels at full power conditions.

On January 16, 2000, blowdown was increased in all Unit 4 SGs to correct SG chemistry due to increased sodium concentrations. When blowdown was increased to 60,000 lbm/hr, a SG "A" level deviation alarm [SG:la] was received. Operations discovered that FCV-4-478 could not maintain level with blowdown at 60,000 lbm/hr. Although level was slowly decreasing, both Steam Flow and Feed Flow channels were matched and operating correctly. "A" SG level could be maintained at 50,000 lbm/hr. The investigation activities planned in response to the December 25, 1999 condition report were augmented based on this event. Feed pump performance was monitored and manual valve positions were verified. Preparation was underway for both a non-intrusive radiographic inspection of the valve internals and a performance test to evaluate FCV response to varying blowdown conditions.

On January 24, 2000 an instability in SG Feedwater flow occurred. FPL decided to conservatively reduce power to provide additional operating margin. An Event Response Team (ERT) was formed. The initial ERT activities were underway when feedwater flow control stability deteriorated without a corresponding valve position change (indicative of internal problems), and significant vibration of the feedwater piping occurred. At 11:42 AM, Turkey Point Unit 4 was manually tripped from 95% power. The stem position and the valve indicating lights indicated that FCV-4-478 did not fully close.

Failure Analysis

The problem originally identified for FCV-4-478 was one of high demand when compared with similar valves for the "B" and "C" SGs. Further examinations of the control functions and the actual valve position also validated the demand signal. A number of potential failure mechanisms were under investigation. However, prior to performing a non-intrusive internal examination, the valve began to exhibit extreme control instability, which then led to the Turkey Point Unit 4 manual reactor trip on January 24, 2000.

FCV-4-478 is a 12 inch Copes-Vulcan double web valve. The valve body has an upper and a lower set of threads. The valve cage threads into the web at these two locations. The valve plug rides in the cage. When the valve cage is threaded into the web (at installation), it is held in place by torque alone.

When FCV-4-478 was disassembled, the cage was found loose in the web. The upper set of threads on the valve body web were destroyed. The lower set of threads were damaged.

FPL believes that the cage could have come loose from the web only by relaxation of the torque over time. Until the 1980's FPL's practice was to replace the valve plug and valve cage at each refueling outage. The new cage was thus torqued into place approximately every 18 months. Because the cage rarely showed wear, FPL changed its maintenance practice sometime in the 1980's (the exact time is unknown), and began replacing only the plug, leaving the cage in place unless it showed signs of wear. FPL did not recognize that periodic re-torquing of the valve cage was necessary to correct torque relaxation. The last known torque of FCV-4-478 took place in 1986.

The root cause of the failure of FCV-4-478 is inadequate change management in the 1980's when the practice of periodic replacement of the cage was stopped.

The phenomenon of relaxation of a threaded fastener over time following application of an installation torque is not uncommon. In the case of flow control valve cages, the most probable cause for this relaxation is time in service and flow induced loading. For FCV-4-478, the condition of the lower threads may have aggravated this phenomenon. It was documented in 1986 that the cage thread engagement was degraded from original condition. Such degradation would reduce the stability of the cage, permitting greater influence from flow instability and perhaps accelerating the cage disengagement. An examination of the procedures and work packages documentation used to overhaul FCV-4-478 did not identify any requirement to verify the cage torque on a periodic basis. The most recent documented verification of the torque occurred in June 1986. It is likely that thirteen years of service, without re-torque of the cage and under constant hydraulic loading, is sufficient time for torque relaxation and cage disengagement.

Other evidence of relaxation of installation torque is available within the work history at Turkey Point. A work package and an Operating Experience Feedback report review identified two other recorded loose cages during routine valve inspection activities: FCV-3-478 in April 1992 and FCV-3-488 in February 1991. This phenomenon likely became applicable after changing the maintenance practices in the 1980s. Previously, the valve internals, including the cage and plug, were changed on a routine basis. The practice was changed since the cage seldom evidenced any degradation that would require replacement. The replacement of the cage was eliminated in the 1980s without changing the inspection requirements, since the potential for torque relaxation was not recognized.

Cause of the Event

The immediate cause of the reactor trip was a manual action taken by the RCO in response to Main Feedwater flow instabilities. The underlying cause of the trip was a failure in FCV-4-478 valve internals. The valve cage had disengaged from the valve body web. The root cause of the disengagement of the cage in FCV-4-478 was the failure to recognize that reuse of the FCV cage should have been accompanied by torque rechecks whenever plug replacements were scheduled. This resulted in specifying inadequate maintenance activities in the inspection/overhaul procedure for the feedwater FCVs. Procedure 0-PMM-074.10, Main Feedwater System Flow Control Valve Inspection, which is performed every 18 months on each FCV, permits a visual inspection of the cage to accept its condition. The visual inspection should have been augmented with verification that the cage remained properly torqued into the valve body web. Implementation of that change would have eliminated the potential for torque relaxation and the subsequent potential for flow instabilities to loosen the cage within the threaded body.

Safety Consequences and Safety Analysis Impact

Disengagement of the valve cage does not impact the function of the FCV until the loose cage becomes a restriction on flow. Such a condition can be identified by the external symptoms of high valve demand and valve position. The other two Unit 4 SG FCVs were monitored as part of the investigation of FCV-4-478. Normal stroke was verified for valves FCV-4-488 and FCV-4-498 during this reactor trip outage. The available data confirms no operability concern exists for the Unit 3 SG FCVs: FCV-3-478, FCV-3-488, FCV-3-498, or for the other two Unit 4 SG FCVs: FCV-4-488, and FCV-4-498. There are no other systems at Turkey Point which have double web Copes-Vulcan FCVs. Continued monitoring will ensure no operability concerns develop. Interim monitoring will ensure proper valve function until the next refueling outage when valve cage torque can be verified. Permanent monitoring will continue to track valve performance to detect any deteriorating trends in FCV performance.

The manual reactor trip resulted in an automatic turbine trip. The trends of nuclear power, pressurizer pressure, pressurizer water volume, RCS average temperature, RCS inlet temperature, and SG pressure for this trip compared very conservatively to the trends in the Updated Final Safety Analysis Report (UFSAR).

Following the reactor trip, a feedwater isolation signal was generated on reactor trip with RCS average temperature of 554 degrees F. FCV-4-478 did not fully isolate, allowing approximately 10% of the nominal feedwater flow into the "A" SG for approximately 100 seconds (normally the valve closes in 20 seconds). In case of failure of the FCV, termination of feedwater flow to the SG can be accomplished by closing the feedwater isolation MOV, or by tripping the Main Feedwater pump. For this case, and in accordance with EOP E-0, the RCO closed the feedwater isolation valve, which also terminated the leakage flow. The potential impact on the Safety Analyses of the additional feedwater flow to the SG has been evaluated. The two UFSAR safety analyses directly impacted by feedwater malfunction are the Feedwater Malfunction Event and the Main Steamline Break (MSLB) Event.

The Feedwater Malfunction Event is assumed to result in excessive feedwater reaching one SG. The excessive feedwater flow increases the heat removal capability of the secondary system thus resulting in a primary system cooldown. The cooldown of the primary system will cause a power increase due to negative reactivity feedback. The current analysis assumes one feedwater FCV malfunctions resulting in a step increase to 200% of the nominal feedwater flow to one SG. The assumptions and results of the analysis in the UFSAR bound the conditions of the actual event, i.e., the total amount of feedwater added to the SG in the safety analysis is significantly greater than the amount of feedwater added as a result of the FCV malfunction. Therefore, the RCS cooldown predicted in the Safety Analysis for this event envelops the cooldown caused by the FCV malfunction.

The Main Steamline Break analysis results in an RCS depressurization, cooldown and corresponding reactivity addition initiated from hot standby conditions. The analysis assumes that the positive reactivity resulting from the Steamline Break could exceed the minimum plant shutdown margin. The analysis assumes that the faulted SG is conservatively supplied with twice the nominal feedwater flow, with the intact SG receiving the nominal feedwater flow. The results of the analysis (factoring in the malfunction of the FCV occurring in either the faulted or intact SGs) conclude that fuel cladding damage is not likely to occur since the 95/95 Departure from Nucleate Boiling (DNB) ratio limit is satisfied. Therefore, because the assumptions and results of the analyses in the UFSAR bound the conditions of the actual event, this event did not compromise the health and safety of plant personnel or the general public.

Corrective Actions

1. The cage for the SG "A" Main Feedwater Flow Control Valve, FCV-4-478, was repaired and properly secured, ensuring acceptable operation by implementing a temporary design change and modification. FPL and the vendor are evaluating the acceptability of the temporary repair as a permanent modification of FCV-4-478.

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

2. FPL reviewed maintenance records for all six valves, which represents all of the valves of this design at Turkey Point Units 3 and 4. Records indicate that four of the valves had their valve cage torque inspected. No records of any inspection were found for the Unit 3 FCV-3-498. Interim monitoring of all (total of six) Turkey Point Units 3 and 4 Main Feedwater FCVs demand will be used to identify unusual trends in demand, indicative of potential cage disengagement. FPL will monitor and record the demand position for each Feedwater FCV once per shift until each unit's next refueling outage when cage torque will be verified.

3. Permanent monitoring of the loose cage symptom will be incorporated in the System Engineer trends for the Feedwater system by trending valve position on a monthly basis. Experience with FCV-4-478 indicates that both position and demand will yield an extended warning of potential valve cage movement.

4. Procedure 0-PMM-074.10, Main Feedwater System Flow Control Valve Inspection, will be revised to require verification of the cage installation torque during FCV inspections or overhauls. Incorporation of torque verification within the standard valve inspection/overhaul will prevent torque relaxation and eliminate the potential for flow instability to move the valve cage.

5. All of the Unit 4 feedwater flow transmitter port connectors have now been replaced with 0.065-inch thick 3/8-inch tubing. The Unit 3 port connectors will be replaced during the next Unit 3 refueling outage.

Additional Information

There has been one earlier event reported related to Feedwater FCV failure: LER 250/94-006-00. This failure was due to intermittent open circuit in the transducer.

The Institute of Nuclear Power Operations (INPO) LER data base has been searched and no other LERs were found which identify the cause of a reactor trip as the disengagement of the FCV cage from the valve body web.

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