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10 CFR 50.73

W3F1-2015-0072

August 21, 2015

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
11555 Rockville Pike
Rockville, MD 20852

Subject: Licensee Event Report (LER) 2015-004-01, Emergency Feedwater System Flow Oscillations and LER 2015-005-01, Manual Reactor Trip due to Low Steam Generator Levels
Waterford Steam Electric Station, Unit 3 (Waterford 3)
Docket No. 50-382
License No. NPF-38

Dear Sir or Madam:

On June 3, 2015, Waterford Steam Electric Station, Unit 3 (Waterford 3) experienced a loss of main feedwater to both steam generators. Entergy is hereby submitting revisions to two LERs, 2015-004-01 and 2015-005-01, for events that occurred during this transient. The LERs were revised to add an abstract section per NUREG-1022.

LER 2015-004-01 provides details associated with a condition that resulted in a common cause inoperability of both trains of the Emergency Feedwater (EFW) System and could have impacted the past operability of both trains of the Emergency Feedwater System and the Atmospheric Dump Valves. It was determined that this condition is reportable pursuant to 10 CFR 50.73(a)(2)(i)(B), 10 CFR 50.73(a)(2)(v)(B), 10 CFR 50.73(a)(2)(vii) and 10 CFR 50.73(a)(2)(ix)(A). A follow up to LER 2015-004-01 is due by January 28, 2016 to provide the safety significance determination that is not yet complete.

LER-2015-005-01 provides details associated with a condition that resulted in the following System Actuations: Reactor Protection System Including Reactor Trip, Emergency Feedwater System, and Emergency Diesel Generators. It was determined that these conditions are reportable pursuant to 10 CFR 50.73 (a)(2)(iv)(A); specifically: 10 CFR 50.73 (a)(2)(iv)(B)(1), 10 CFR 50.73 (a)(2)(iv)(B)(6), and 10 CFR 50.73 (a)(2)(iv)(B)(8).

This report contains no new commitments. Please contact John P. Jarrell, Regulatory Assurance Manager, at (504) 739-6685 if you have questions regarding this information.

Sincerely,

JPJ/MMZ

Attachments: (1) LER 2015-004-01
(2) LER 2015-005-01

cc: Mr. Mark L. Dapas, Regional Administrator
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Attachment 1

to

W3F1-2015-0072

Licensee Event Report 2015-004-01

NRC FORM 366 (02-2014)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB: NO. 3150-0104 EXPIRES: 01/31/2017

LICENSEE EVENT REPORT (LER)

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4. TITLE
Emergency Feedwater System Flow Oscillations

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 05000
06	03	2015	2015 -	004 -	01	08	21	2015	FACILITY NAME	DOCKET NUMBER 05000

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

LICENSEE CONTACT	TELEPHONE NUMBER *(Include Area Code)*
John Jarrell	5047396685

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		01	28	2016

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

On June 3, 2015, at 1707, following a manual reactor scram from 100% power, an Emergency Feedwater Actuation Signal (EFAS) was automatically actuated to both Steam Generators (SGs). Following flow initiation, the Emergency Feedwater (EFW) Backup Flow Control Valves (BFCVs) for both trains exhibited wide, frequent oscillations. Operators took manual control of both trains and stabilized flow. Both channels of EFAS flow control logic and both trains of EFW BFCVs were subsequently declared INOPERABLE and Technical Specifications (TS) 3.3.2.b and 3.7.1.2.d were entered, respectively. Analysis has determined that the identified valve cycling would have exceeded the assumed nitrogen consumption rate and, without operator intervention, would have exhausted the accumulators prior to the credited 10 hour analyzed mission time. The specified safety functions of both trains of EFW and both Atmospheric Dump Valves (ADVs) would not have been fulfilled. The cause of this event was that tuning of the flow control system was not adequate to cope with changes to system operating parameters. Compensatory measures have been put in place to station a dedicated operator to control the EFW BFCVs in manual following a reactor trip to establish and maintain the SG level in accordance with emergency operating procedures; system modification will follow.

NARRATIVE

INITIAL CONDITIONS

On June 3, 2015, Waterford 3 was in Mode 1 at approximately 100% power. There were no structures, components, or systems that were inoperable at the start of the event that contributed to the event.

The manual reactor [RCT] trip due to low SG levels is reported in LER-WF3-2015-0005.

EVENT DESCRIPTION

At 1704 on June 3, 2015, Waterford 3 experienced a loss of the “A” Main Feedwater Pump. At 1705, the reactor was manually tripped when SG levels were continuing to lower. At 1707, an EFAS [JE] was automatically actuated to both SGs as SG level decreased. Flow initially stabilized at 250 gpm with the Primary Flow Control Valves (PFCVs) [FCV] providing flow and the BFCVs [FCV] closed. Both SG levels continued to decrease and the EFW [BA] control logic shifted the operation of the BFCVs to flow control mode to maintain flow to each SG. EFW flow stabilized and then SG levels began recovering. At 1709, the EFW AB pump [P] reached rated speed and EFW header discharge pressure increased. Shortly afterward, wide, frequent fluctuations in EFW flow were observed which was not in accordance with the expected system response.

Operations personnel observed that the controller [FIK] outputs for both BFCVs were oscillating frequently and widely. Both PFCVs operated correctly in automatic (the controller outputs remained steady and the valves remained in their fixed position). To prevent further oscillations, both BFCV controllers were taken to manual and then closed at 1715. The oscillations stopped concurrent with taking the valves to manual control. Operations personnel cycled both BFCVs in manual with no further flow oscillations noted. After confirming that they could control both BFCVs in manual, the operators closed the PFCVs. EFW flow to the SGs was controlled by operation of both BFCVs in manual for the remainder of the event, until EFW was secured.

The EFW flow control logic [JB] and both BFCVs were declared INOPERABLE due to the FCV oscillation and actions per TS 3.3.2.b and 3.7.1.2.d were entered, respectively. The EFW system functioned adequately to fill the SGs and maintain the specified safety function (Reactor Coolant System Heat Removal).

This event and the manual reactor trip were immediately reportable (reference EN # 51116) under 10 CFR 50.72(b)(3)(iv)(A), Specified System Actuation, and 10 CFR 50.72(b)(2)(iv)(B), RPS Actuation (scram), respectively.

Investigation has revealed that the components comprising the EFW flow control system were not configured to appropriately respond to the changes observed in the system operating parameters. Both EFW BFCVs cycled more than assumed in the nitrogen accumulator [ACC] sizing calculation. The excessive cycling has the potential for exhausting the accumulators prior to their 10 hour analyzed mission time in the event of a loss of Instrument Air (IA) [LD]. These accumulators also supply backup nitrogen to the ADVs. Periodic testing to confirm the stability of the BFCVs in the automatic flow control mode has not been performed. It is therefore reasonable to assume that this condition has likely existed within three

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years of the time of discovery. It was concluded that the EFW FCVs and the ADVs [V] have been inoperable for the time in which the reactor operated in the modes requiring applicability during the three year time period leading up to this event.

SYSTEM DESCRIPTION

The safety function of the EFW system is to provide sufficient supply of cooling water to one or both SGs for the removal of decay heat from the Reactor Coolant System (RCS) [AB] in response to any event causing low SG level coincident with the absence of a low pressure trip. The EFW system supplies this demand via three EFW pumps through two supply paths. Both supply paths are supplied with redundant IA operated FCVs and isolation valves, all of which fail open on loss of air. The FCVs modulate EFW flow in response to SG level. These valves are designated as primary and backup. The FCVs change operating modes and setpoints based on changes in SG level indication. If IA is lost, backup positioning of the valves is provided utilizing nitrogen from dedicated accumulators (V and VIII) which are sized for a minimum of 10 hours operation.

The ADVs are used to remove reactor decay heat from the SG in the event of loss of condenser [COND] cooling. They are also credited with reducing RCS pressure during certain small break loss of coolant accident scenarios, but not actuated by any engineered safety feature actuation system signal. The valves are electro-pneumatically operated and are controlled automatically or manually. The valves are designed to fail closed on loss of IA. Each valve has a handwheel which can be operated locally to override the actuator spring. The nitrogen accumulators that supply backup nitrogen for the EFW valves also supply backup nitrogen to the ADVs.

Once the accumulators are depleted, manual local control is necessary, and analyzed in the design basis, to position the EFW BFCVs and the other valves fed by the accumulator (the PFCVs and the ADVs).

REPORTABLE OCCURRENCE

TS 3.3.2 requires that the EFAS control valve logic shall be OPERABLE in Modes 1, 2, and 3. Action b requires that with control valve logic inoperable, restore within 48 hours or be in HOT STANDBY within 6 hours, and in HOT SHUTDOWN within the following 6 hours. This also requires entry into TS 3.7.1.2 action d.

TS 3.7.1.2 requires that three EFW pumps and two flow paths shall be OPERABLE in Modes 1, 2, and 3. Action d requires that with the EFW system inoperable and able to deliver at least 100% flow to either SG, restore EFW to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

TS 3.7.1.7 requires each ADV be OPERABLE in Modes 1, 2, 3, and 4. The condition of both ADVs inoperable for reasons other than the automatic actuation channels is not addressed by the ACTION statements; however, it is mentioned in the TS basis. For this condition, TS 3.0.3 is entered. TS 3.0.3 requires that when a Limiting Condition for Operation is not met, except as provided in the associated ACTION requirements, within 1 hour, action shall be initiated to place the unit in a MODE in which the specification does not apply by placing it, as applicable, in: (1) At least HOT STANDBY within the next 6 hours, (2) At least HOT SHUTDOWN within the following 6 hours, and (3) At least COLD SHUTDOWN within the subsequent 24 hours.

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During the event, the EFW flow control logic was declared inoperable due to the observed erratic behavior and TS 3.3.2.b and TS 3.7.1.2.d were entered. This condition is reportable under 10 CFR 50.73(a)(2)(vii), Common Cause Inoperability of Independent Trains or Channels, because this resulted in both channels of the EFAS and all EFW control valves to become inoperable.

The following additional reportability criteria that were met are based on the conclusion that the EFW FCVs and the ADVs have been inoperable for the time in which the reactor operated in the modes requiring applicability during the three year time period leading up to this event.

This condition is reportable under 10 CFR 50.73(a)(2)(i)(B), Operation or Condition Prohibited by Technical Specifications, because the potential condition has existed for longer than the allowed outage time of TS 3.3.2 action b (15), TS 3.7.1.2 action d, and 3.7.1.7 actions per 3.0.3.

This condition is also reportable under 10 CFR 50.73(a)(2)(v)(B), Event or Condition that Could Have Prevented Fulfillment of a Safety Function, and 10 CFR 50.73(a)(2)(ix)(A), Single Cause that Could Have Prevented Fulfillment of the Safety Functions of Trains or Channels in Different Systems, because the potential condition could have prevented the fulfillment of the safety function of both trains of the EFW system, both channels of the EFAS, and both ADVs.

CAUSAL FACTORS

A root cause evaluation was completed for this condition.

The direct cause of this condition was an instability in the control system setup of the EFW BFCVs that occurred when the valves were operating in the flow control mode. This resulted in the continuous cycling of the EFW BFCVs.

The root cause of this event was that the components comprising the EFW flow control system were not configured to appropriately respond to the changes observed in the system operating parameters. Analysis has concluded that the unstable behavior seen in the flow control mode is explained if the valve gain or controller gain was improperly selected for both loops. With the gain improperly selected, a perturbation in EFW flow would have caused a feedback effect that would have setup a varying output signal to the FCVs. The analysis eliminated all other potential causes except the following all tied to system gain:

- Controller proportional gain and reset interval
- Valve trim (linear) results in greater than desirable flow when the valve is operating close to its seat
- Valve stroke time set too fast
- Volume booster setup improper for both valves.

Follow on corrective actions are assigned to determine the sensitivity of the control system to these parameters and to determine if the potential identified causes may have shifted the system to unstable operation. These actions are also directed to determine whether changes in the valve or controller characteristics could alter the stability of the system.

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One contributing cause of this event was that there is no periodic testing that confirms the stability of the BFCVs in the automatic flow control mode. No startup test exists where the system was allowed to shift the BFCVs to the flow control mode and control in this mode. Calibration checks of the flow control loops and actuator are periodically performed; however, these tests do not provide sufficient intrusiveness to determine instabilities in system operation in all modes.

A second contributing cause of this event is that previous corrective actions were ineffective at determining the cause of the EFW flow instabilities and confirming the oscillations were corrected. EFW flow oscillations of similar magnitude and frequency were observed following a plant trip on January 21, 2013. There were missed opportunities noted in response to this event that may have led to earlier discovery of the causes.

EXTENT OF CONDITION

Other safety-related FCVs were identified. Since the BFCVs operate in multiple modes, each mode was considered in the extent of condition.

The possibility of flow oscillations in the manual mode was eliminated by direct observation of flow with the BFCVs in manual mode. EFW flow was varied over a wide range of conditions and flow was stable. The probability of oscillations in level control mode is considered low. Level is a slowly changing variable and is not expected to produce oscillations. Observation of level control mode was made during previous events and no oscillations were noted.

The PFCVs were considered as similar items, since they also control EFW flow and could be a source of oscillations. The probability of oscillations is considered low because the primary EFW valves do not have a flow control mode, and their observation in manual was directly observed. Level control mode is also considered low risk for the same reasons given for the BFCVs.

In extending the extent of condition beyond the EFW system, it was determined that systems where two valves operate in parallel to control flow would be of particular concern.

A review was conducted of major plant systems with operations input for similar applications. The review identified main feed regulating valves and startup feed regulating valves as a similar application. This application is considered low risk because the system is normally in flow control mode and the performance of the feed regulating valves is continuously monitored. This behavior is not currently observed in the system.

The extent of condition review determined that additional corrective action is necessary to verify that the same-similar condition does not exist on the BFCVs. The specific concern is that the flow oscillations may have led to some degradation of the actuator or connected piping. An action will be assigned to perform post event valve diagnostics to determine that no damage has occurred to the actuators. A walkdown of the adjacent piping has been conducted and discovered no discrepancies.

NARRATIVE

CORRECTIVE ACTIONS

Operations has issued a standing order to implement a manual compensatory action for this condition. This requires that an additional operator to be stationed to operate the BFCVs in manual following a reactor trip to establish and maintain the SG water level in accordance with emergency procedures. This manual action is required to protect the associated nitrogen accumulators from depletion due to the excessive cycling of the BFCVs during an EFW actuation. Crediting the established compensatory measures, the EFW and ADVs are capable of performing their specified safety functions for the evaluated mission times. This will remain in effect until the condition is resolved.

An operability evaluation was completed by engineering personnel providing analysis supporting continued operation of the EFW system with the BFCVs in the manual closed position and specifying manual operator action to control the EFW BFCVs in manual upon an EFAS actuation.

A walkdown of the EFW piping adjacent to the BFCVs was conducted to confirm that no collateral damage to the equipment occurred as a result of this condition. No damage was noted.

Additional corrective actions:

(1) Develop a modification to the EFW system that ensures control stability is maintained under all expected operating conditions.

(2) Perform analysis to confirm the identified potential operating condition changes that led to unstable flow conditions.

(3) Perform diagnostic testing on the BFCVs to confirm valve operating sensitivity to control changes.

(4) Develop requirements for periodic testing of the EFW BFCVs.

(5) Perform a comprehensive design review of the EFW control logic for the PFCVs and BFCVs.

(6) Provide this condition as a case study to engineering personnel to emphasize the lessons learned.

(7) Perform a review of air operated control valves to determine if control system instabilities currently exist with the valves in automatic.

(8) Perform a review of surveillance testing of air operated FCVs to determine if the current surveillances are adequate to detect changes in system or component parameter that could lead to unstable operation in the automatic mode.

(9) Perform a review of significant condition reports (category A and B) associated with the EFW system to ensure that if multiple condition reports were closed to another condition report, all conditions were adequately corrected.

(10) Complete safety significance determination.

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SAFETY SIGNIFICANCE

Industrial Safety: There was no industrial safety significance associated with this issue.

Radiological Safety: There was no radiological safety significance associated with this issue.

Environmental Safety: There was no environmental safety significance associated with this issue.

Nuclear Safety: The safety significance determination is not yet completed. Safety significance will be included as a planned update to this LER.

PREVIOUS OCCURRENCES

Plant operating history was reviewed for other possible plant transients in which an EFAS occurred and water was injected into the SGs. Three previous EFAS actuations (1998, 2005, & 2013) have been identified.

January 21, 2013 (Condition Report CR-WF3-2013-0451): A plant trip on Low SG Level occurred following the inadvertent closure of the SG 1 feedwater regulating valve. EFAS was actuated for SG 1 only. SG 2 levels remained above the EFAS actuation setpoint. All three EFW pumps started and the BFCV and the PFCV initially went to the full open position due to SG wide range indication dropping. When SG level recovered, the BFCV should have shifted to flow control mode to maintain EFW total flow at the setpoint. EFW flow was instead observed to widely fluctuate. These oscillations exhibited a similar frequency and amplitude to the oscillations observed during the most recent June 3, 2015 event. The condition report documents that the PFCV position remained steady during this period. The BFCV remained in the automatic mode of operation during the entire 13 minute period in which oscillations were occurring. The oscillations remained relatively constant in amplitude and frequency until SG level reached the level control mode setpoint at which the valve shifted modes and the oscillations stopped. Subsequent troubleshooting revealed that the positioner for the BFCV had been degraded by a loose screw within its pneumatic operating mechanism resulting in a shift in the positioner’s calibration. The apparent cause of the oscillations observed was attributed to the degraded positioner.

November 11, 2005 (CR-WF3-2005-4598): A plant trip and loss of main feedwater occurred due to a loss of circulating water. EFAS was initiated to both SGs, all three EFW pumps started, and the BFCVs for both SGs opened. The PFCVs failed to open due to a rapid, short duration downward spike in wide range level and a vulnerability inherent to the design of the EFW control logic. The BFCVs were observed to control flow during this event at the intended rate with no oscillations in flow observed. Since the PFCVs were closed in this event rather than at their fixed position, the BFCVs operated farther off their closed seat in order to maintain the flow setpoint. Operating the BFCVs closer to mid-stroke position placed the valves in a more favorable position to control flow. It is believed that operating the BFCVs in this manner prevented flow oscillations from occurring.

July 16, 1998 (CR-WF3-1998-0948): A loss of main feedwater occurred resulting in a plant trip and EFW actuation. EFAS was initiated for both SGs, all three EFW pumps started, and both the PFCVs and BFCVs for both SGs opened. The BFCVs were observed to operate in flow control mode at the setpoint without any oscillations present. The PFCVs operated as designed at their fixed position. This event is significant because it provides evidence that the flow control system for the BFCVs previously exhibited

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stable operation, even with the PFCVs opened at a fixed setpoint. The differences between this event and the event that occurred on June 3, 2015 were: 1) The speed setpoint for EFW AB pump had been increased after the 1998 EFAS actuation; and 2) The event started with the EFW BFCVs full open. This may have prevented initial overshoot of the system since the valves were closer to the setpoint. One anomaly was noted during this event. One BFCV went to the full open position when the valve switched from flow control mode to level control mode. This was caused by the failure of a relay card in the process analog control system. This failure is unrelated to the oscillations observed on June 3, 2015.

ADDITIONAL INFORMATION

Energy industry identification system (EIIS) codes and component function identifiers are identified in the text with brackets [ ].

Attachment 2

to

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Licensee Event Report 2015-005-01

NRC FORM 366 (02-2014) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB: NO. 3150-0104 EXPIRES: 01/31/2017

LICENSEE EVENT REPORT (LER)

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4. TITLE

Manual Reactor Trip due to Low Steam Generator Levels

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 05000
06	03	2015	2015	- 005	- 01	08	21	2015	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE	11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*			
1	☐ 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)
10. POWER LEVEL	☐ 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)
100	☐ 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

LICENSEE CONTACT	TELEPHONE NUMBER *(Include Area Code)*
John Jarrell	5047396685

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
E	SN	LCV	W255	N	B	EA	RLY	A160	N

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 June 3, 2015, at 1705, the reactor was manually tripped due to low Steam Generator (SG) levels caused by a loss of the ‘A’ Main Feedwater (MFW) Pump. During the transfer of electrical buses, the ‘B’ train electrical buses failed to transfer to the Startup Transformer and were de-energized. The loss of ‘B’ power caused the loss of the remaining MFW pump and the Emergency Diesel Generator (EDG) on that side to automatically start and load the safety buses. The loss of MFW resulted in an Emergency Feedwater Actuation Signal (EFAS) with both SGs being fed by Emergency Feedwater (EFW) to restore level. The Emergency Operating Procedure (EOP) for a reactor trip due to the loss of MFW was entered. This event was caused by (1) the failure to identify the failure mechanism for repeated failures of non-safety related normal level control valves and (2) the relay’s timed contact sets not changing state due to an unknown equipment problem. Corrective actions include rebuilding valves with identified similar deficiencies, changing the frequency of the Preventive Maintenance (PM) procedures for level control valves, and establishing a relay replacement PM and identifying electrical control circuit system enhancements.

NARRATIVE

INITIAL CONDITIONS

The plant was in Mode 1 at 100% power with Reactor [RCT] Power Cutback (RXC) removed from service due to time in core life. Preventive maintenance was being performed on the Feed Heater Drain (FHD) [SN] 2C alternate level control valve [LCV]. Circulating Water Pump A was removed from service for maintenance on the traveling screen system.

EVENT DESCRIPTION

On June 3, 2015, at approximately 1700, the control room received multiple alarms due to FHD 2C experiencing high levels. Extraction steam (ES) [SE] to #2 heaters isolation isolated on high high level in the 2C FHD and the FHD 1C alternate drain valve [LCV] opened as designed. The control room entered the Off-Normal Procedure for Secondary System Transient. At 1702, the control room received annunciators for all three heater drain pump [P] low suction pressure and both MFW pumps [P] low suction pressure. The Control Room Supervisor (CRS) directed the Balance of Plant (BOP) operator to commence removing 100 MW's at a rate of 40 MW/min. At 1704, the heater drain pumps tripped on low suction pressure followed by the trip of MFW pump 'A'. The CRS entered the Off-Normal Procedure for a Rapid Plant Power Reduction. At 1705, the BOP operator reported that both SG [SG] levels were at 50% narrow range level and dropping rapidly (automatic reactor trip occurs at 27.4% narrow range). The CRS directed a manual reactor trip in anticipation of an automatic reactor trip from the Reactor Protection System (RPS) [JC] and entered the EOP for Standard Post Trip Actions. All rods fully inserted as designed and all required safety systems actuated as designed.

At the time of the manual reactor trip and main generator trip, the electrical buses [BU] [EA] [EB] were to transfer from the Unit Auxiliary Transformers (UATs) [XFMR] (supplied by the main generator) to the Startup Auxiliary Transformers (SUTs) [XFMR], which are fed from offsite power. The 'A' train electrical buses transferred as required, but the 'B' train buses failed to transfer to the Startup Transformer causing the loss of the 'B' train safety [EB] and non-safety buses [EA]. At 1705, EDG 'B' [DG] [EK] started and energized the safety buses as designed.

The loss of the 'B' non-safety buses caused the remaining FW pump to trip, resulting in a loss of all normal FW to both SGs. At 1707, the EFW [BA] system started on receipt of an EFAS [JE] and commenced feeding both SGs. The loss of the non-safety bus also caused a loss of 2 out of 3 circulating water pumps leading to a loss of main condenser vacuum. At 1706, the non-safety buses were manually transferred to the 'B' SUT but non-safety related bus loads remained de-energized.

EFW was feeding both SGs at 1715 when the BOP operator noted that the EFW backup flow control valves [FCV] exhibited wide, frequent oscillations. The BOP subsequently placed both trains of backup flow control valves in manual (this event is covered in LER-WF3-2015-004). At 1716, the crew entered into the EOP for Loss of Main Feedwater.

The non-safety train 'B' electrical buses were energized at 1817 from offsite power and the safety buses were transferred to offsite power at 1820. EDG 'B' was secured at 1824.

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At 2143, the Auxiliary Feedwater (AFW) [SJ] pump (non-safety related) was started and commenced feeding both SGs in preparation for securing the EFW Pumps. At 2231, EFW was secured with SG levels being maintained in their normal band using the AFW pump. On June 4, 2015, at 0224, the EFAS system was reset and the EFW system was placed in standby.

On June 4, 2015, at 0527, the plant exited the EOP for Loss of Main Feedwater and entered the Normal Operating Procedure for Plant Shutdown. The plant was stabilized in Mode 3 at normal operating temperature and pressure with heat removal being controlled on Atmospheric Dump Valves.

INVESTIGATION POST TRIP

The cause for FW heater 2C high level was that the 2C normal level control valve [LCV] stem and plug assembly had disengaged from the actuator stem. The stem threads were damaged and the lock pin was sheared. This caused the normal level control valve to fail closed. With the alternate level control valve removed from service for maintenance, all paths for removal of condensed water were unavailable, thus resulting in the high high level condition.

The failure of the ‘B’ UAT to transfer to ‘B’ SUT was determined to be a failure in the SUT ‘B’ time delay relay. Two sets of contacts would not change state when the relay was bench tested post removal. This failure mode would have prevented power to be transferred from UAT ‘B’ to SUT ‘B’ during a fast bus transfer. This relay is normally energized and the failure of the relay also explains why both the 6.9kV and 4kV feeder breakers from the SUT ‘B’ did not transfer.

SYSTEM DESCRIPTION

<u>Feedwater Heater Drain System</u>

The purpose of the extraction steam system is to supply steam to preheat the FW prior to entering the SGs. Preheating of the FW increases the cycle thermodynamic efficiency and minimizes thermal shock to the SGs. Extracting steam from the various stages of the turbine also removes moisture that collects as the enthalpy of the steam is reduced along the blade path.

The heater normal drains are arranged in a cascading fashion. The #1 heater drains to the #2 heater under normal conditions, with an alternate drainage valve that will automatically open to divert drainage to the condenser when heater water level is too high. The normal drains from the #2 heater are aligned to the suction of the heater drain pump which pumps the drain flow to the FW pump suction. Under abnormal high water level conditions, the drains from the #2 heater are automatically diverted to the condenser.

<u>Emergency Feedwater System</u>

The safety function of the EFW system is to provide a sufficient supply of cooling water to one or both SGs for the removal of decay heat from the reactor coolant system [AB] in response to any event causing low SG level coincident with the absence of a low pressure trip. The EFW system supplies this demand via three EFW pumps through two supply paths. Both supply paths are supplied with redundant instrument air operated flow control valves (FCVs) and isolation valves, all of which fail open on loss of air. The

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FCVs modulate EFW flow in response to SG level. These valves are designated as primary and backup. The FCVs change operating modes and setpoints based on changes in SG level indication.

Reactor Power Cutback System

The RXC system is actuated upon receiving coincident 2/2 logic signals indicating either large turbine load rejection or loss of one of two main FW pumps. The actuation logic initiates the insertion of the preselected pattern of CEAs. Subsequent insertion of other groups will automatically occur after the operator performs the immediate action of placing the control element drive mechanism control system control switch to the auto sequential position. On loss of a FW pump, a rapid turbine power reduction to 50 percent power is initiated, followed by a further reduction if necessary to balance turbine power with reactor power.

Electrical Distribution for Transfer of Unit Auxiliary Transformers to Startup Transformers

The automatic transfer uses a sequential design that requires the SUT supply breaker to receive its close signal from the opening UAT supply breaker. The nominal “dead time” for the sequential bus transfer is 3.3 cycles. The bus loads cannot distinguish between a "live bus" transfer and a "fast dead bus" transfer and will remain energized throughout the bus transfer process.

Bus transfer as a result of abnormal conditions is done automatically. Main generator lockout relays 86G1 and 86G2 will initiate a fast dead bus transfer of plant loads from the UATs to the SUTs. An automatic fast dead bus transfer will occur when the main generator lockout relays operate, as long as the trip was not caused by an overcurrent condition on the UAT feeder breakers, voltage is present at the SUTs, the UAT feeder breaker is open, and bus and incoming voltage/frequency are synchronized and in phase, as detected by synchronizing check relays.

REPORTABLE OCCURRENCE

Initial reportability (Message EN# 51116) was performed within 4 hours per 10CFR50.72(b)(2)(iv)(B), 4-hour Non-Emergency RPS Actuation (Scram), and 10CFR50.72(b)(3)(iv)(A), 8-hour Non-Emergency Specified Systems Actuation.

This condition is reportable under 10CFR50.73(a)(2)(iv)(A) (System Actuation); specifically: 10CFR50.73(a)(2)(iv)(B)(1), Reactor Protection System (RPS) Including Reactor Scram or Reactor Trip, due to a manual reactor trip prior to initiation of and automatic scram, 10CFR50.73(a)(2)(iv)(B)(6), PWR Auxiliary or Emergency Feedwater System, due to automatic initiation of EFW system and the addition of EFW to the SGs, and 10CFR50.73(a)(2)(iv)(B)(8), Emergency AC Electrical Power Systems, Including Emergency Diesel Generators, due to the automatic start of the EDG on loss of offsite power to the ‘B’ train electrical buses.

CAUSAL FACTORS

Failure of the Normal Level Control Valve

Root Cause: The root cause of the event is the failure to identify the failure mechanism for repeated failures of non-safety related normal level control valves.

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Contributing Causes: First contributing cause - Inadequate causal and technical evaluations for repeated failures of the non-safety related normal level control valves were accepted and tolerated by the site leadership team.

Extent of Condition:

This extent of condition is bounded by consideration of failures of normal and alternate level control valves for low, intermediate, and high pressure heaters. These valves are non-safety related valves and similar in purpose and design. The review identified that the normal feedwater intermediate pressure heater control valve (FHD-455A) is currently vulnerable to failure. The valve is not currently exhibiting lateral movement or chattering as was exhibited by the failed valve. Corrective action needed to address this condition is provided below. Several other normal level control valves for intermediate, low, or high pressure heaters with different size or similar design were identified that could fail closed while their alternate control valve is tagged out. The probability of occurrence of this happening is deemed medium with consequence from low to high. An interim action to address this is already in place to perform weekly schedule risk reviews for single point vulnerabilities and items that could result in an action requiring immediate shutdown therefore no further action is required at this time. Several other alternate level control valves of similar design were identified that could fail closed, or in a position other than closed, while their normal level control valve is tagged out. The probability of occurrence of this happening is deemed low with consequence from low to high. An interim action to address this is already in place to perform weekly schedule risk reviews for single point vulnerabilities and items that could result in an action requiring immediate shutdown therefore no further action is required a this time.

Corrective Actions:

(1) Change the frequency and due dates for the inspection and rebuild PMs for normal level control valves to be performed each refueling outage starting with the next refueling outage until the underlying failure mechanism for valves is determined and corrected.

(2) Develop and implement a “scheduling strategy” that contains expectations for scheduling online work to identify, assess and mitigate risks associated with trip/event initiators.

(3) Rebuild normal feedwater intermediate pressure heater control valve as soon as parts and plant conditions allow.

(4) Develop additional actions if needed after the failure mechanism of the failed level control valve is determined.

(5) Create a Trip/Event Initiator (TEI), requirement code in Asset Suite to identify work orders and PMs as TEIs to enable this information to be pulled into the P6 schedule layout.

<u>Failure of UAT ‘B’ to Auto Transfer to SUT ‘B’</u>

Apparent Cause: The apparent cause for the Allen Bradley 700RTC11200U1 relay’s timed contact sets C1-C2 and C3-C4 not changing state was due to an unknown equipment cause. Potential causes are failure of the relays C1-C2 and C3-C4 contact sets, or coil failure.

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Contributing Cause: A contributing cause for the lack of identification of the failure of the 4KVEREL2237 B RELAY (TL) prior to the loss of offsite power was due to a design change that was not adequate. A latent design deficiency (1997 plant change) did not take into account the observed failure mode where the two coil design allows one coil to fail thus preventing the dead bus transfer function (timed contacts) without actuating the contacts for the alarm circuit. This resulted in the alarm not being received prior to the failure to dead bus transfer from UAT ‘B’ to SUT ‘B’.

Extent of Condition:

There are 38 Allen Bradley 700RTC relays installed at Waterford 3:

- Four relays are installed for fast bus transfer of the 4KV and 7KV busses. It was verified during plant shutdown that power was transferred from UAT ‘A’ to SUT ‘A’.
- Thirty-three of these relays are installed in Engineered Safety Features Actuation System (ESFAS) applications. These relay failures are self-revealing (alarm or equipment start/stop/trip) and actions are in place to correct this condition.
- One relay was installed in core protection calculator panel D under temporary modification. The application is an alarm only circuit.

Corrective Actions:

(1) Generate an action request to establish a relay replacement PM task frequency for three years for four Allen Bradley 700RTC relays installed in the auxiliary relay cabinet.

(2) Fund an engineering study to make a recommendation for a relay to replace the existing fast bus transfer Allen-Bradley 700RTC series relays within auxiliary panel 4. The study should also pursue a new design that will provide proper alarm function should the coil for the control circuit fail.

(3) Funding an engineering study to make a recommendation for a relay to replace the existing fast bus transfer Allen-Bradley 700RTC series relays. The study should also pursue a new design that will provide proper alarm function should the coil for the control circuit fail.

SAFETY SIGNIFICANCE

Industrial Safety: There was no industrial safety significance associated with this issue.

Radiological Safety: There was no radiological safety significance associated with this issue.

Environmental Safety: There was no environmental safety significance associated with this issue.

Nuclear Safety: The actual consequence of this event was a manual reactor trip due to loss of MFW [SJ] pump ‘A’. This unplanned reactor trip did not have any nuclear safety significance, based upon the operating shift’s proper and timely diagnosis of the loss of MFW Pump ‘A’, and the initiation of a manual reactor trip. Since the operating shift was able to properly analyze the event and respond by manually tripping the unit, there are no potential consequences if an additional barrier failed or response actions

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were delayed. The RXC during this event was not in service. Because of this condition, the plant transient would have led to a SG low level condition resulting in an automatic reactor trip.

PREVIOUS OCCURRENCES

Waterford 3 Licensee Event Report history was reviewed.

LER WF3-2013-001-00: MFW regulating valve failed closed on loss of instrument air. This caused a manual reactor trip to be initiated.

There have been no previous occurrences where a reactor trip occurred due to a FW heater normal level control valve failing closed. Previous occurrences had only caused a reduction in plant power with unit remaining on line. This is based on a search of the Waterford 3 corrective action program database.

ADDITIONAL INFORMATION

Energy industry identification system (EIIS) codes and component function identifiers are identified in the text with brackets [ ].