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APR 2 4 2003

L-2003-103
10 CFR § 50.73

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

Re: Turkey Point Unit 3
Docket No. 50-250
Reportable Event: 2003-005-00
Date of Event: March 1, 2003
Disabling Both Auxiliary Feedwater Trains Inadvertently During Mode 3

The attached Licensee Event Report 250/2003-005-00 is being submitted pursuant to the requirements of 10 CFR § 50.73(a)(2)(i)(B) to provide notification of the subject event.

Very truly yours,

Terry Jones

Terry O. Jones
Vice President
Turkey Point Nuclear Plant

SM

Attachment

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

JE22

NRC FORM 366 (7-2001)

U.S. NUCLEAR REGULATORY COMMISSION

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

# LICENSEE EVENT REPORT (LER)

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

Estimated burden per response to comply with this mandatory information collection request 50 hrs Reported lessons learned are incorporated into the licensing process and fed back to industry. Forward comments regarding burden estimate to the Records Management Branch (T-6 F33), U S Nuclear Regulatory Commission, Washington, DC 20555-0001, and to the Paperwork Reduction Project (3150-0104), Office of Management and Budget, Washington, DC 20503. If an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

| 1. FACILITY NAME | 2. DOCKET NUMBER | 3. PAGE |
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4. TITLE

Disabling Both Auxiliary Feedwater Trains Inadvertently During Mode 3

| 5. EVENT DATE MONTH | DAY | YEAR | 6. LER NUMBER YEAR | SEQUENTIAL NUMBER | REVISION NUMBER | 7. REPORT DATE MONTH | DAY | YEAR | 8. OTHER FACILITIES INVOLVED FACILITY NAME | DOCKET NUMBER |
|---|---|---|---|---|---|---|---|---|---|---|
| 03 | 01 | 2003 | 2003 | 005 | 00 | 04 | 29 | 2003 | | |

9. OPERATING MODE: 3

10. POWER LEVEL: 0

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

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

12. LICENSEE CONTACT FOR THIS LER

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

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

| CAUSE | SYSTEM | COMPONENT | MANUFACTURER | REPORTABLE TO EPIX | CAUSE | SYSTEM | COMPONENT | MANUFACTURER | REPORTABLE TO EPIX |
|---|---|---|---|---|---|---|---|---|---|
| D | BA | EI | | N | | | | | |

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

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


On March 1, 2003, Turkey Point Unit 3 was in Operational Mode 3 in accordance with the planned reactor shutdown for the Cycle 20, Refueling Outage. A clearance executed for planned maintenance activities on the Rod Control system opened two DC breakers, which disabled both trains of Reactor Protection Relay Racks. This was an expected condition. However, it also disabled both trains of Auxiliary Feedwater (AFW) automatic actuation logic and relays from the Steam Generator water (SG) water level low-low signal. This was not expected. Two Technical Specifications (TS) for the AFW actuation logic and actuation relays Item 6.a of TS Table 3.3-2 and TS 3.7.1.2 for the AFW steam supply valves are applicable in Modes 1-3. This condition was discovered in Mode 5 when no TS applied. There was no recognition that TS action statements applied during the transition to Mode 5. As a result, the TS actions while in Mode 3 were not performed. No operability concerns existed at the time of discovery. Unit 3 continued to Mode 6 for refueling activities. The root cause of this event is a latent organizational weakness in the level of detail contained in the vital DC load list drawing. A contributing cause is the inherent design of the SG water level low-low signal circuitry. It was determined that the health and safety of the public were not affected by this event. This event is reportable in accordance to 10 CFR 50.73(a)(2)(i)(B).

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

## EVENT DESCRIPTION

On March 1, 2003, Turkey Point Unit 3 was in Operational Mode 3 in accordance with the planned reactor shutdown for the Cycle 20, Refueling Outage. Operations personnel were executing an outage clearance, zone 28-01, on the Rod Control System [AA], which opened two DC breakers [EI:BKR] disabling both trains of Reactor Protection Relay System (RPS) Racks [JC:83]. The first train was disabled at 00:59 hours on March 1, 2003 and the second train was disabled at 01:13 hours on March 1, 2003 while the plant was still in Operational Mode 3. Operations was cooling down the Steam Generators by filling and draining. This evolution drains the Steam Generators [SG] below the 10% Steam Generator water level low-low Setpoint. Since this setting is also the Auxiliary Feedwater (AFW) initiation setpoint, the AFW Steam Supply Motor Operated Valves (MOVs) [FCV:20] are expected to open. The Reactor Operator noticed that these MOVs did not open as expected. The Outage Control Center was contacted to investigate the reason why these MOVs did not open. On March 1, 2003, at approximately 23:00, the investigation determined that the clearance on zone 28-01 had inadvertently disabled both trains of AFW Auto-Start actuation circuitry from Steam Generator water level low-low input. Specifically, the clearance directed the Operator to open breakers 3D01-40 and 3D23-08. These clearance steps disabled both trains of AFW automatic actuation logic and actuation relays, impacting automatic operation of the AFW system on a Unit 3 Steam Generator water level low-low signal. It was determined that two Technical Specifications (TS) applied. The applicable Limiting Condition of Operation (LCO) for the AFW actuation logic and actuation relays is provided as Item 6.a of Table 3.3-2. Technical Specification 3.7.1.2 provides the LCO for the AFW steam supply valves.

The LCO associated with Item 6.a of TS Table 3.3-2 requires that both trains of AFW automatic actuation logic and actuation relays be operable in Modes 1, 2, and 3. Breaker 3D01-40 was opened at 00:59 on 3/1/03 while the unit was in Mode 3. This rendered one of the two AFW logic trains required by Technical Specification Table 3.3-2, Item 6.a, inoperable. With one less than the required two trains of automatic actuation logic and actuation relays operable, Action statement 20 of Technical Specification 3.3.2 requires that the affected unit be in at least Hot Standby within 6 hours and in at least Hot Shutdown within the following 6 hours. Breaker 3D23-08 was opened at 01:13 on 3/1/03 while the unit was in Mode 3 and rendered the second AFW logic train required by Technical Specification Table 3.3-2, Item 6.a inoperable. Technical Specification Table 3.3-2 does not address the loss of both trains. The actions of Technical Specification Section 3.0.3 would generally be applicable under these conditions. However, as indicated above, Technical Specification 3.7.1.2 is also applicable to the subject event. The LCO for Technical Specification 3.7.1.2 is similarly applicable in Modes 1, 2, and 3. Action 2 of this Technical Specification specifically addresses the conditions that must be met if both trains of AFW are rendered inoperable. Action 2 states that if neither train can be restored to an OPERABLE status within 2 hours, verify the OPERABILITY of both Standby Feedwater pumps and place the affected unit(s) in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

Both trains of AFW were rendered inoperable in Mode 3 at 01:13 on March 1, 2003. Unit 3 entered Mode 4 at 07:05 on March 1, 2003, 5 hours and 52 minutes after both trains were rendered inoperable. The subject condition was not discovered until 2300 on March 1, 2003. The unit was already in Mode 5 when the discovery was made. At this time the above Technical Specifications no longer applied. There was no cognitive recognition that this action statement applied during the transition to Hot Shutdown. As a result, the requisite action to verify operability of both Standby Feedwater pumps within 2 hours was not performed.

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

However, when the inadvertent entry into these action statements was recognized, investigation revealed that the Field Operator had taken normal log readings at 00:25 on March 1, 2003. Both Steam Generator Standby Feedwater pumps had passed their last surveillances and were operable at the time the AFW circuitry was disabled. Since Unit 3 was already being shut down, the TS 3.0.3 requirement for placing the unit in Hot Shutdown was met. No operability concerns existed at the time of discovery. Unit 3 continued to Mode 6 for refueling activities.

The failure to comply with plant Technical Specifications is reportable under 10 CFR 50.73(a)(2)(i)(B).

SYSTEM DESCRIPTION

This event impacts the Engineered Safety Features Actuation System (ESFAS) components associated with AFW actuation. Operation of the AFW system is initiated from the following signals:
1. Bus Stripping (loss of offsite power)
2. AMSAC
3. Low-low Steam Generator water level
4. Trip of Last Running Main Feedwater Pump [P]
5. Safety Injection
6. Manual Initiation

For the low-low Steam Generator water level signal, the actuation of AFW requires that two out of three level transmitters [LT] on a single Steam Generator measure water level below 10%. If this condition occurs, each affected Steam Generator level comparator [CPT] will trip a bistable. The output of these bistables energizes relays in the AFW actuation logic matrix and AFW automatic start actuation relays. The output of the AFW actuation relays provides a start signal to the AFW pump steam admission valves. A simplified diagram, Figure 1, of this process is provided.

Furthermore, a simplified overview of the Steam Generator low-low water level AFW initiation circuit is provided in the attached sketch, Figure 2. The sketch reflects one transmitter on one Steam Generator and one train of actuation logic and actuation relays. Each of the three Steam Generators has three level transmitters. These nine transmitters are connected to two trains of logic and actuation relays such that the AFW is initiated if a low-low level is sensed on two out of three channels on any Steam Generator.

Technical Specification Table 3.3-2, Item 6 addresses operability requirements for the AFW ESFAS Instrumentation. Item 6.b of Table 3.3-2 is specific to the AFW low-low Steam Generator water level instrumentation. This is considered to encompass the level transmitter, level comparator and bistable. This instrumentation is powered from 120 VAC vital instrumentation power and was not part of the subject clearance. Item 6a of Table 3.3-2 is specific to AFW automatic actuation logic and actuation relays. This is considered to encompass the remainder of relays. As shown in the sketch, the actuation logic relay coil is powered from the 120 VAC vital instrumentation power, which was not part of the subject clearance. The opening of Breaker 3D01-40 removed power from the Train 3A low-low level actuation relays for all three Steam Generators. Breaker 3D23-08 removed power from the Train 3B low-low level actuation relays for all three Steam Generators. As shown in the attached sketch, the opening of the subject breakers only disabled the low-low level actuating relays. Power to the AFW automatic start actuating relays was not affected by opening the subject breakers. Therefore, only the low-low Steam Generator water level AFW auto start was affected by the subject incident. All other AFW automatic start signals (safety injection, bus stripping, and trip of Last Running Main Feedwater pump, AMSAC) remained available.

EVENT ANALYSIS AND ROOT CAUSE

The clearance steps of zone 28-01 opened two DC breakers that removed power from both Reactor Protection Relay Racks for planned work on the Reactor Trip Breakers and circuitry. These breakers also affected both trains of the AFW Auto Start actuation circuitry from Steam Generator Water level Low-Low relays disabling both trains when power was removed. The breakers were added to the clearance zone to provide electrical isolation for work that was planned on the Reactor Trip Breakers and circuitry. The decision to add these breakers was based on the desire to add additional personnel protection, and an effort to minimize changes and revisions to the clearance after it was hung.

The RPS and ESFAS circuitry associated with this event are not independent. The circuitry shares the same power supplies and utilizes different contacts off of the same component. The complexity of the circuitry involved and the supporting documentation typically available to the operations staff make the process necessary to arrive at the correct determination difficult and cumbersome. Numerous individuals from multiple departments involved with the process missed the shared power supply configuration, and thereby the TS implications.

The supporting documentation needed to determine the impact on the unit is extensive and key documentation, such as the Vital DC Bus Load List and operating diagrams only reference the RPS not the ESFAS. They do not provide any apparent reference to AFW Auto start circuitry. This lack of detailed information made the process of determining the impact on the unit more complex.

The de-energizing of the Reactor Protection Relay Racks was a planned outage activity. Since Steam Generator low-low water level is a reactor trip associated with this circuitry, it was expected that the relays associated with this would be disabled. Based on the above, the impact of disabling these relays on the AFW functions was not recognized.

The root cause of this event is a latent organizational weakness in the level of detail contained in the vital DC load list drawing. There was a lack of detailed information referencing the AFW ESFAS functions of the breakers that were opened.

A contributing cause is the inherent design of the relay. There is no redundancy feature. The one relay serves a dual purpose (both the RPS and the ESFAS) and it is contained in the RPS racks. Removal of control power to the reactor protection racks disabled both the RPS and ESFAS circuitry in this case.

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## GENERIC IMPLICATIONS

A review of PTN Condition Report Database revealed other condition reports were written due to insufficient or inadequate description in the Breaker List for both AC and DC breakers. These condition reports were assigned significance levels requiring corrective action for the specific discrepancy only. Consequently, the corrective actions from these condition reports addressed the specific discrepancies and did not address generic implications for other AC / DC breakers containing inadequate description.

Due to the complexity of some circuitry and the numerous components often supplied from one power supply breaker, electrical clearances are more likely to impact multiple components than less complex mechanical clearances. An immediate corrective action taken upon discovery of this event included a review of all electrical, I&C, and mechanical clearances that were already hung and that were to be hung for generic implication. Clearance zone 28-01 on Unit 4 was similarly affected. A history check on this clearance was performed for Unit 4. It was determined that during Unit 4 refueling outage, March 2002, clearance zone 28-01 was hung. However, Unit 4 was not in Operational Mode 1-3, so no Technical Specification action statements were entered due to opening these breakers. Both zone 28-01 clearances for Unit 3 and Unit 4 have been deleted from the clearance database.

## SAFETY SIGNIFICANCE

The loss of automatic AFW actuation on Unit 3 low-low Steam Generator water level in Mode 3 was determined to not be a safety significant event. The 3A Main Feedwater Pump was providing decay heat removal. Both Standby Steam Generator Feedwater pumps were also operable in Mode 3. In the Updated Final Safety Analysis Report the automatic actuation of AFW is credited for several design basis accidents. A list of the accidents and the analyzed AFW actuation signals are tabulated below.

| Design Basis Accident | AFW Actuation Signal |
|---|---|
| Loss of Normal Feedwater Flow | Low-Low Steam Generator Level |
| Loss of All Non-Emergency AC Power | Low-Low Steam Generator Level |
| Steam Generator Tube Rupture | Safety Injection |
| Main Steam Line Break | Safety Injection |
| Loss of Coolant Accident | Safety Injection |

The Loss of Normal Feedwater Flow accident analysis assumes that the AFW system automatically actuates on a low-low Steam Generator level signal. The absence of this signal does not preclude operation of AFW during those conditions that can cause a loss of Feedwater event. For example, a loss of normal Feedwater can be caused by failure of a Feedwater pump(s), or by a failure of the Feedwater control system.

The 3A Main Feedwater Pump was being used for decay heat removal in Mode 3 on March 1, 2003. Loss of this pump, the last running Feedwater pump would have provided the requisite automatic AFW actuation signal such that there would not have been a loss of decay heat removal function. A loss of normal Feedwater event caused by a failure of the Feedwater control system would result in a mismatch between steam flow and Feedwater flow for the affected Steam Generator(s). Compliance with plant procedures would instruct the operator to manually initiate AFW under such conditions, preserving the decay heat removal function.

The Loss of All Non-Emergency AC Power accident analysis also assumes that the AFW system automatically actuates on a low-low Steam Generator level signal. Automatic actuation of the AFW system under these conditions would have occurred on a bus stripping signal if the low-low Steam Generator level signal were not available. Again, the decay heat removal function would be accomplished by the AFW system. All of the remaining accident analyses assume that the AFW system automatically actuates on a safety injection signal. Actuation on a Unit 3 safety injection signal was not affected by the as-found condition of breakers 3D01-40 and 3D23-08. Hence, they were not impacted by the subject event. The health and safety of the public were not affected by this event.

## CORRECTIVE ACTIONS

1. Engineering will update and revise the Vital AC / DC Load Lists to specifically include more detailed information on the components of RPS or ESF functions and generically to address other potential description deficiencies.

2. Operations will add procedural steps to not allow the Reactor Protection Racks to be de-energized until less than 350 degrees (Mode 4).

3. This event will be incorporated into Pre-outage training for the upcoming Unit 4 Refueling Outage.

4. Operations will review all master electrical clearances in the Turkey Point database for similar impact and add appropriate limitations and plant conditions. Historical clearance zones will be verified prior to use until review is complete.

5. Engineering will review scheduled clearances for vital AC/DC loads until completion of corrective actions # 1 and # 4.

## ADDITIONAL INFORMATION

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

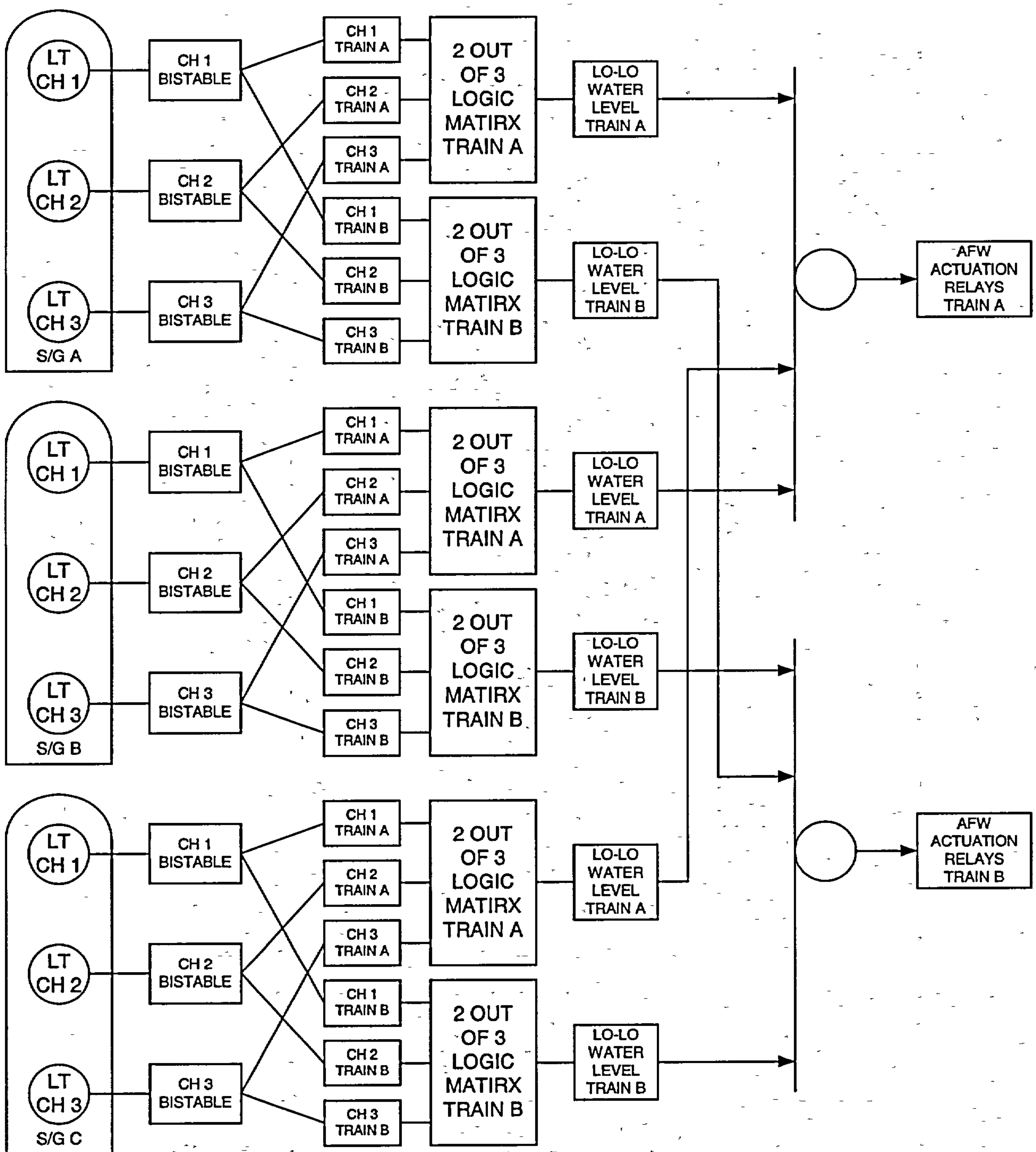


**Figure 1**
**STEAM GENERATOR LOW-LOW WATER LEVEL AUXILIARY FEEDWATER ACTUATION**
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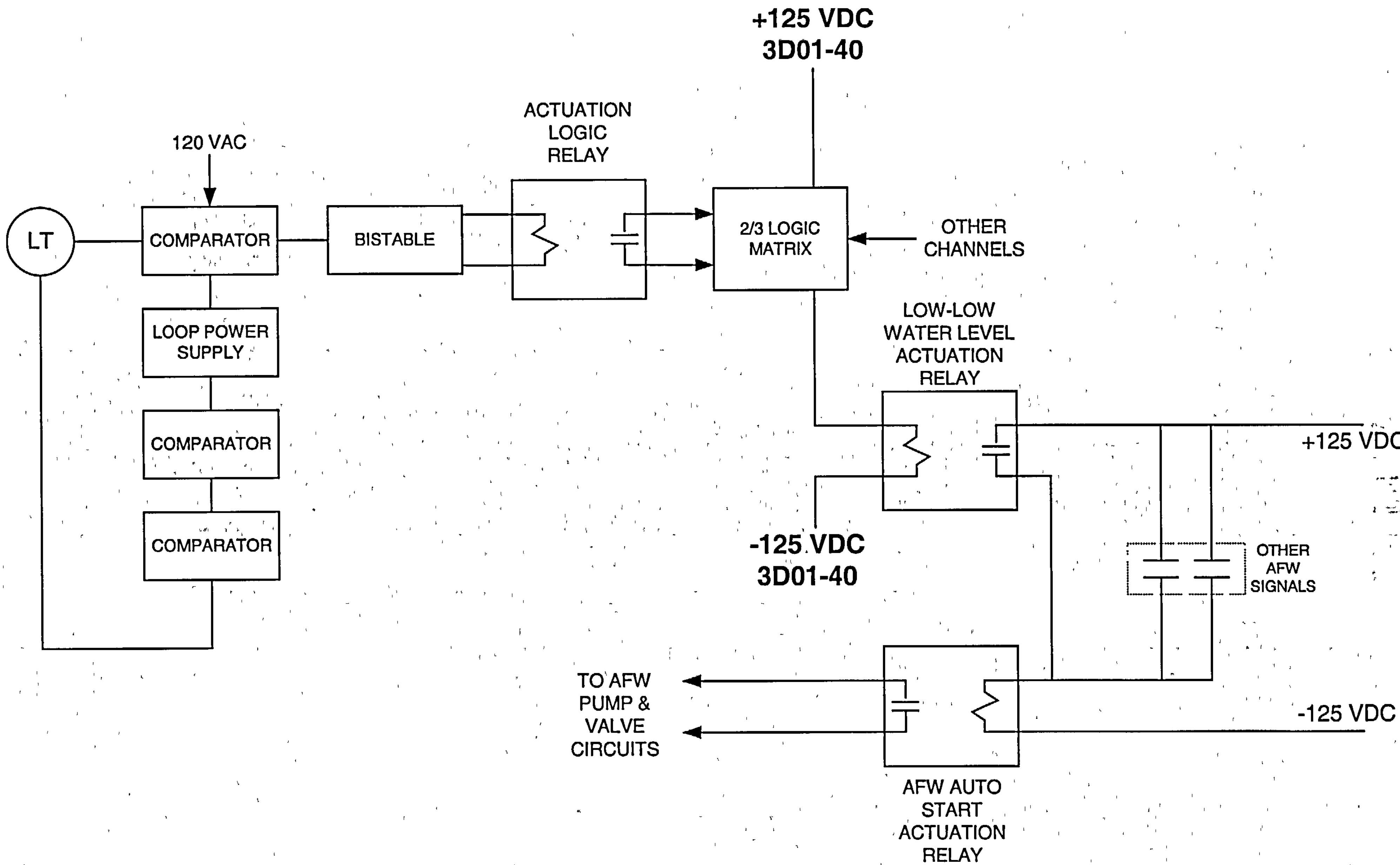


**Figure 2**

**STEAM GENERATOR LOW-LOW WATER LEVEL AUXILIARY FEEDWATER ACTUATION**