U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-254/89027(DRP)

Docket No. 50-254 License No. DRP-29

Licensee: Commonwealth Edison Company
P. O. Box 767
Chicago, IL 60609

Facility Name: Quad Cities Nuclear Power Station, Unit-1

Inspection At: Quad Cities Site, Cordova, Illinois

Inspection Conducted: December 15, 1989

Inspectors: W. D. Shafer, Team Leader
R. D. Lanksbury
M. E. Parker
R. L. Higgins

Approved By: W D Shafer
W. D. Shafer, Chief
Reactor Projects Branch 1

1/5/90
Date

Inspection Summary

Inspection on December 15, 1989 (Report No. 50-254/89027(DRP))
Areas Inspected: Special reactive team inspection conducted in response to the Unit 1 turbine trip and subsequent 150°F reduction in reactor feedwater temperature event of December 14, 1989. The inspection included a review of the adequacy of the licensee's analysis of the reactor feedwater transient on the Minimum Critical Power Ratio safety limit; a determination of the root cause of the unexpected turbine trip that initiated the event; the reason the annunciator for Reactor Feedwater Pump (RFP)/Turbine Trip failed to actuate; and an evaluation of the Operations Department's response to the event, including the appropriateness of the procedures used.
Results: No violations or deviations were identified. The team concluded that the feedwater temperature reduction was a normal expected response to the turbine trip and the licensee's actions during and subsequent to the event were conservative and appropriate. These actions ultimately resulted in a decision not to scram the reactor during the transient, thereby avoiding an additional transient on the reactor. A major contributor to this event was inadequate control by the licensee of work activities performed by contract workers during the last Unit 1 refueling outage conducted in late 1989.

DETAILS

1. Persons Contacted

   Commonwealth Edison Company

   *R. L. Bax, Station Manager
   *G. F. Spedl, Production Superintendent
   R. H. Thompson, Shift Engineer
   *A. L. Misak, Lead Nuclear Engineer

   The inspectors also contacted other technical personnel during the course of the inspection.

   *Denotes those attending the exit meeting on December 15, 1989.

2. Introduction

   a. Description of Event

      At 9:20 p.m. (CST) on December 12, 1989, a radiation technician reported to the control room that water was dripping from the Unit 1 Yarway Level Instrument No. 1-263-59A. Supervisors from the Maintenance and Operations Departments could not determine the source of the leak after being dispatched to the location. Senior station management made a decision to monitor the leak hourly and wait for the day shift to prepare a work package to repair the leaking level switch.

      On December 13, 1989, the licensee completed their work package and at 11:20 p.m. started dropping load on Unit 1 to less than 45% power to prevent a scram in case the turbine tripped during the repair. The Reactor Feedwater Pump (RFP) high level trip was also disabled to prevent an inadvertent pump trip during the repair.

      At 3:16 a.m. on December 14, 1989, while instrument mechanics were working on the Yarway Level Switch No. 1-263-59A, Unit 1 received a turbine trip when the switch was placed in the tripped position. The turbine trip was not expected since only one-half of the trip logic was actuated to perform the maintenance work activity. As a result of the turbine trip, the reactor feedwater inlet temperature started dropping due to the loss of turbine extraction steam to the reactor feedwater heaters. The feedwater temperature at the start of the event was approximately 215°F, and when the event terminated the temperature was approximately 65°F, a total drop of approximately 150°F.

      As the reactor feedwater temperature was dropping (the elapsed time for the 150°F drop was approximately 15 minutes) the control room operator reportedly informed the Shift Engineer that the reactor must be scrammed (shutdown) if the reactor feedwater temperature decreased more than 140°F as specified in procedure QOA 3500-1, Revision 6, Loss of Feedwater Heaters. The Shift Engineer did not agree with the procedure requirement and placed a telephone call to the Production Superintendent and the Lead Nuclear Engineer to discuss the requirement to scram the reactor. As a result of this conference call, a decision was made to not scram the reactor; however, log records showed that the reactor feedwater temperature exceeded the 140°F limit prior to the final decision to not scram the reactor.

During the above described event, two other incidents occurred that were not directly related to the event but were caused by the trip of the turbine. At the time of the turbine trip there was an automatic transfer of electrical power to the Auxiliary Power Transformer. The Technical Advisor log indicated that during this transfer the Unit 1 Diesel Generator received a start signal. The Diesel Generator Run Light came on, but the diesel engine did not start. As this incident was not related to the event of interest, the failure of the diesel generator to start is an open item and will be followed by the Senior Resident Inspector (Open Item 254/89027-01(DRP)). The second incident involved a temporary loss of the security perimeter lighting. Reportedly, this is not unusual during an electric power transfer.

b. Licensee Immediate Action

At the time of the turbine trip event, the operator immediately entered QOA 5600-4, Revision 2, Loss of Turbine Generator, and confirmed the automatic and immediate actions described in the procedure. As the procedure does not specify any actions when the reactor power is less than 40%, a decision was made to insert rods in sequence to obtain a reactor power sufficient to maintain five turbine bypass valves fully open (there are a total of nine turbine bypass valves).

When the automatic transfer to Auxiliary Power occurred the control room logged an automatic start of the DG. After noting the DG had not run, the start switch was placed in stop when the licensee determined the start signal was erroneous.

c. Licensee Followup Actions

On December 14, at approximately 10:59 a.m., the licensee made a courtesy Emergency Notification System (ENS) notification that the feedwater temperature decrease appeared to exceed the 145°F decrease assumed in the accident analysis and that the plant may have operated outside the design bases. The licensee's initial evaluation showed that no Minimum Critical Power Ratio (MCPR) or other safety limits were exceeded during the transient. However, after discussion with their corporate management, the licensee determined that a MCPR concern could exist if the transient had occurred at a higher power level with the unit operating closer to the MCPR limit. Base on this a unit shutdown was commenced while the analysis of the transient continued.

At 1:30 p.m. on December 14, 1989, with Unit 1 at approximately 11% power, the decision to complete the shutdown was terminated based on a General Electric (GE) evaluation that the event was bounded by the design analysis.

At approximately 4:30 p.m. on December 14, 1989, station management conducted a conference call with the NRC (NRR and Region III), GE representatives, and Commonwealth Edison Corporate representatives to discuss the feedwater transient. GE reported that the maximum feedwater temperature drop (145°F) is used to calculate the delta-Critical Power Ratio margin at rated power to ensure that the safety limit is met during the time the unit is close to the rated power limits in the event the single worst failure should occur. The maximum temperature drop is also used to ensure that the thermal and mechanical overpower criteria are met. GE concluded that since the unit was below 40% power at the time of the transient, the safety limits had not been approached. This evaluation was documented by GE on December 14, 1989 (Attachment 1).

On December 14, 1989, the licensee established an Onsite Review Committee to evaluate the event and to determine what actions should be taken prior to resuming power operations. The following immediate corrective actions were recommended:

(1) Repair the Yarway Level Switch 1-263-59A.

(2) Verify the operability of all annunciators important to plant operations.

(3) Test the annunciator for the RFP/Turbine trip from both level switches (1-263-59 A and B).

(4) Revise QOA 3500-1 to provide guidance to the operator as to when the feedwater temperature drop of 140°F is in effect and train each crew before assuming shift duties.

In addition to the above, corporate management decided to conduct a corporate review of the event.

3. Formation of Special Inspection Team

a. Assignment

On December 15, 1989, the NRC Region III office established a Special Inspection Team consisting of the Branch Chief, Reactor Projects Branch 1, Division of Reactor Projects, and the Senior Resident Inspectors from LaSalle, Duane Arnold, and Quad Cities. The team was instructed to inspect and determine the following:

(1.0) Determine the adequacy of the licensee's analysis of the reactor feedwater transient on the Minimum Critical Power Ratio safety limit.

(2.0) Determine the root cause of the unexpected turbine trip that initiated the event.

(3.0) Determine the reason the annunciator (RFP/Turbine Trip) failed to actuate on the 1-263-59B switch signal.

(4.0) Determine and evaluate the Operations Department response to the event including the appropriateness of the procedures used.

b. Inspection Team Review

(1.0) Adequacy of the Licensee's Analysis of the Reactor Feedwater Transient on the Minimum Critical Power Ration (MCPR) Safety Limit

The inspectors participated in the licensee's conference call previously described and had no questions regarding the GE conclusions. The team further determined that there was no significant effect on MCPR as a result of this event. Prior to the turbine trip and subsequent reduction in feedwater temperature MCPR was at approximately 51.9% of the operating limit. After the event, MCPR was calculated to be roughly at 48% of the operating limit (the decrease was due to operator actions of driving control rods). The decrease in feedwater temperature alone would tend to reduce the margin to the MCPR limit. For this particular event, with reactor power below 45% and with a turbine trip with the bypass valves functioning, the margin to the MCPR operating limit is fairly large and therefore is not a concern.

(2.0) Root Cause of the Unexpected Turbine Trip

On December 14, 1989, with reactor power at 40%, the licensee received a turbine trip without a reactor scram while technicians were working on Yarway Level Instrument No. 1-263-59A to repair a leak that had developed on the level switch.

During the last fuel cycle, the licensee had determined that Yarway Level Switch 1-263-59B was operating erratically and decided to replace the switch during the next scheduled refuel outage. Switch 59B was replaced in the last refuel outage as planned. When the switch was replaced the installation instructions specified to verify "like for like." The switch that was provided was specified based on the part number in the vendor manual. This part number is for a switch that would normally be open (at normal reactor water levels) and would close at a water level of 48". The required switch for this application is a switch that would be normally closed and would open at 48" reactor water level. The exact reason why the vendor manual specified a switch not appropriate to this application is unknown at this time. The likely reason is that the vendor manual specifies an arrangement that they viewed as the most widely used, but for this application it was purchased by the licensee without specifying the switch to fit their specific needs.

During the installation of the new switch it appeared that the licensee failed to verify a "like for like" replacement. The testing instructions were inadequate in that they only required the Instrument Mechanic (IM) to verify that the switch changed state on increasing reactor water level. They did not require him to verify in which direction the change occurred. The appropriateness of the testing instructions provided for the installation and testing of the level switch is considered an unresolved item (URI 254/89027-02(DRP)) and will be followed up for possible enforcement action in a future inspection.

The root cause of the unexpected turbine trip was the installation of the wrong switch and the apparent inadequate post-installation testing prior to placing the switch in operation.

With the incorrect B switch in place one-half of the turbine trip signal existed at all times when the reactor water level remained below 48". However, the leads to the annunciator that would have activated, making the operator aware of this condition, had been lifted (i.e., one-half the signal was defeated). Assuming the worse case (both switches 1-263-59A and 59B were set exactly at 48"), had the reactor water level reached the 48" level, the 59A switch would have tripped providing one-half the turbine trip signal, and the 59B switch would have closed or cleared, providing no signal to the trip logic. As a result, with the reactor water level above 48", the turbine trip signal and reactor feedwater pump trip signal would have been defeated.

During the attempted repair of the Yarway Level Switch 1-263-59A, at normal reactor water level, the IM tripped the 59A switch. With the 59B switch already tripped (unknown to the operator), the tripping of the 59A switch by the IM completed the two-out-of-two trip signals and caused the turbine trip.

(3.0) Reason the Annunciator (RFP/Turbine Trip) Failed to Actuate in the Control Room

During the 1989 refuel outage, the licensee had scheduled a portion of a modification to provide a re-flash capability to the control room annunciators. The work analyst (a contract employee) had developed the work package from the design drawings and identified that there were wires that affected another annunciator panel, other than the one he was working on. At this point he had already completed the work package. He wrote a memo to the appropriate organizations to identify this discrepancy. Subsequent to this, and prior to work package issuance, he received verbal instructions to change the work package to reflect not moving the wires for the other panels. A Field Change Request (FCR) was to follow. Two of the wires affected were the wires that were found lifted on the annunciator of concern. The work analyst proceeded to make the changes to the work packages per his verbal instructions to not lift leads connected to other panels, however, he failed to remove the instructions on the completed work package for lifting the two leads for annunciator F-11 on panel 901-6. Subsequently, the FCR was issued and the work package issued. The FCR did not address the original error but assumed that all leads had been returned as had been instructed verbally. The work package was completed, including Quality Control verification of each step. The as built drawing used by the test engineer showed the leads as having been untouched so the leads were not included in the panel testing.

The root cause of the lifted lead was a personnel error by the work analyst compounded by inadequate control of the design process by the licensee. The lifted leads resulted in a turbine trip because the annunciator was not lit indicating that half a trip signal was already present when the licensee took the second trip channel out-of-service for maintenance work.

(4.0) Adequacy of the Operations Department Response to the Event

As a result of the turbine trip, the control room operator entered procedure QOA 5600-4, Revision 2, Loss of Turbine Generator. All actions (automatic and immediate operator action) are predicated on reactor power being greater than 40%. Subsequent operator actions required that if the reactor had not scrammed, the operator should insert rods in sequence until the turbine bypass valves were closed. The operators took action to insert rods to keep the power level constant and until five turbine bypass valves remained open.

The inspectors determined by review of QOA 5600-4, that no reference was made to the loss of extraction steam to the reactor feedwater heaters which had the same effect as a loss of feedwater heaters. However, the control room operator referenced procedure QOA 3500-1, Revision 6, Loss of Feedwater Heaters, and alerted the Shift Engineer to the requirement to scram the reactor if the reactor feedwater temperature dropped greater than 140°F. Records revealed that the Shift Engineer did not agree with the scram requirement and initiated a conference call with the Production Superintendent and Lead Nuclear Engineer to discuss the requirement. Records also revealed that the 140°F feedwater temperature drop had been exceeded before an official determination not to scram the reactor was made.

The Inspection Team conducted interviews with the Production Superintendent, Shift Engineer, and Lead Nuclear Engineer to determine the accuracy of the available records and to understand why the operator was instructed to not scram the reactor as required by procedure QOA 3500-1.

Separate interviews with each individual revealed that each individual ultimately agreed that the reactor should not be scrammed because the reactor was not at a condition where the 140°F feedwater temperature drop was applicable. It must be noted that these interviews were conducted after the GE analysis was received by the licensee. The inspection team also confirmed that the decision not to scram the reactor was made after the 140°F feedwater temperature drop was exceeded.

With regard to procedures, the inspection team determined that the control room operator had entered the appropriate procedure for a turbine trip and should have entered procedure QOA 3500-1 on loss of extraction steam to the feedwater heater. There were sufficient symptoms at the time of the event to cause the operator to enter procedure QOA 3500-1. However, both procedures were determined to be inadequate for the following reasons:

- QOA 5600-4, Revision 2, Loss of Turbine Generator, was inadequate in that guidance was not provided to the operator on what actions must be taken when the reactor power is less than 40 percent, including guidance regarding the loss of extraction steam, which has the same effect as a loss of feedwater heaters. The intent of the procedure was to provide direction to the operator on loss of the turbine generator at reactor powers greater than 40 percent and was determined to be acceptable because monitoring requirements were appropriate for the conditions that existed.

- QOA 3500-1, Revision 6, Loss of Feedwater Heaters, was an appropriate procedure for the operator to enter due to the symptoms (alarms on the control panel and other plant conditions) provided to the control room operator. However, the procedure was inadequate in that a requirement was imposed on the operator to scram the reactor if the reactor feedwater temperature drop exceeded 140°F. There were no instructions provided to describe when this limiting temperature drop should be in effect. The intervention of the Shift Engineer instructing the operator to not scram the reactor was appropriate and prevented an unnecessary transient on the reactor.

Reportedly, the licensee has recognized a need to upgrade existing procedures in all departmental areas due to third party assessments, Corporate QA and NRC concerns. In November 1988, the licensee developed a program to implement the procedural upgrade. The implementation of this effort began in February 1989, and to date the licensee has 350 out of a total of 4300 procedures in the process of being upgraded.

Pending completion of the licensee's procedure upgrade, the inadequacy of procedures QOA 5600-4 and QOA 3500-1 is considered an open item (Open Item 254/89027-03(DRP)).

4. Conclusions

a. Safety Impact of the 150°F Feedwater Transient on the Minimum Critical Power Ratio Safety Limit

The team concluded that the feedwater transient did not adversely affect the Minimum Critical Power Ratio safety limit, at the time of the event, due to the reactor power Unit 1 was at during the incident and the fact that the bypass valves functioned per design.

b. Safety Impact of Operating the Reactor With an Incorrect Switch

The switch in question is used only for the Yarway Level Switches which provide the RFP/Turbine Trip signal on each unit. The inspectors determined that upon recognition that an incorrect switch had been installed in instrument 59B, the licensee verified that both 59A and B switches on Unit 2 were correct and concluded that the Unit 1 59A switch was correct because a trip signal was correctly provided by the switch during the event and based upon verification of its part number.

The switches provide a two-out-of-two trip signal to the turbine, to protect the turbine blades from excessive carry over (moisture content of the steam), which can occur when the reactor water level is too high (greater than 48").

In addition the switches provide a reactor feedwater pump trip when the reactor water level is too high in order to stop the increase in reactor water level. The inspectors determined that the defeat of the turbine protection signal was mitigated by administrative controls requiring operator action.

Procedure QOA 201-8, Revision 5, High Reactor Water Level, requires the operator to manually trip the reactor feedwater pumps and the main turbine if the equipment did not trip automatically at a reactor water level of 48". This procedure is entered when the reactor water level reaches or exceeds 44".

The team concluded that the safety impact of operating the plant with an incorrect switch was minimized by the existence of administrative controls to serve as a backup in the event of a failure of the automatic RFP/Turbine Trip circuitry.

c. Significance of the Lifted Leads on the RFP/Turbine Trip Annunciators

The significance of the lifted leads on the RFP/Turbine Trip annunciator was considered important and resulted from a personnel error that was further compounded by the licensee's inadequate control of the design process when verbal instructions were issued that resulted in a personnel error made during the last refueling outage. The team notes that during a management meeting with the licensee on November 9, 1989, considerable concern was expressed by NRC Region III regarding the licensee's lack of control of contract employees. While some contract work activities were stopped by the licensee during the outage, this particular modification was given an early work release because the work was reportedly under the direct supervision of a Commonwealth Edison supervisor. This licensee overview was apparently not sufficient to prevent this problem.

d. Appropriateness of the Operations Department Actions and Adequacy of Procedures Used

The team concluded that the decision to not scram the unit, while contrary to the procedural guidance, was appropriate for the plant conditions that existed at the time of the event. While the management involved in this decision believed at the time that the procedure change was made in accordance with the TS, they later learned that a fourth signature was required to approve the change due to a typographical error in the technical specification. The team is concerned, however, that the decision to violate the procedural requirement may impact negatively on personnel by causing them to not promptly initiate an immediate action described by procedure. The licensee should ensure, by familiarizing personnel with the circumstances surrounding the event, and by additional training that this event will not have a negative effect on future operations.

The operators used the correct procedures for the circumstances, however, additional clarification of these procedures is appropriate. The licensee's procedure upgrade program should resolve this problem, however, this effort is not expected to be completed until early 1996 per the present schedule.

5. Open Items

Open items are matters which will be reviewed further by the NRC and which may involve some action on the part of the licensee, NRC, or both. Open items disclosed during this inspection are discussed in Paragraphs 2.a. and 3.b.(4.0).

6. Unresolved Items

Unresolved items are matters about which more information is required in order to ascertain whether they are acceptable items, open items, deviations or violations. An unresolved item disclosed during this inspection is discussed in Paragraph 3.b.(2.0).

7. Exit Interview

The inspectors met with licensee representatives (denoted in Paragraph 1) at the conclusion of the inspection on December 15, 1989, and summarized the scope and findings of the team's activities. The licensee acknowledged these findings. The inspectors also discussed the likely informational contents of the inspection report with regard to documents or process reviewed by the team during the inspection. The licensee did not identify any documents or processes as proprietary.

ATTACHMENT

TURBINE TRIP AT LOW POWER AT QUAD CITIES UNIT 1

EVENT

At about 3:15 AM on December 14, 1989 a turbine trip occurred at Quad Cities 1. At the time the turbine tripped the reactor was below 40% power. At such low powers there is no scram on turbine trip. With the turbine off-line extraction steam was no longer available for feedwater heating. As a result the feedwater temperature dropped roughly 150 degrees Fahrenheit. The site staff took actions to insert rods to keep the power level constant. The entire transient took about 15 minutes. No scram occurred.

CONCERN

CECo has raised two concerns. These are:
1) did the plant exceed the fuel safety limits during the actual event
2) does the fact that they had a 150 degree change in feedwater temperature invalidate their license as the LFWH analysis is performed with a 145 degree drop in feedwater temperature

CONDITIONS DURING THE EVENT:

This event occurred at low power. At such a low power MFLCPR and MFLPD have ample margin to their limiting values. As the power level was maintained constant the changes to MFLCPR and MFLPD would be small. The safety limits were not exceeded.

TEMPERATURE DROP AND THE LOSS OF FEEDWATER HEATER (LFWH) ANALYSIS

The LFWH is licensed for a 145 degree drop in temperature. This event showed a 150 degree drop. This does not mean that the licensing basis has been invalidated.

The 145 degree LFWH analysis performed for licensing is done at rated power. It determines the necessary delta-CPR margin to insure that the safety limit is met ... if the plant is close to the rated power limits. It also is used to ensure that thermal and mechanical overpower criteria is met. As noted above the conditions during the event were such that these limits were not approached.

More importantly the LFWH analysis does not apply to this event. A feedwater heater was not lost. A turbine trip occurred and the system behaved as expected. The change in feedwater temperature was due to the loss of the extraction steam when the turbine tripped. Such a loss of extraction steam is typical for a turbine trip. This event was a turbine trip at low power with the bypass operational. Such an event is bounded by the licensing analysis of turbine trip without bypass at rated power.