UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 139 TO FACILITY OPERATING LICENSE NO. DPR-30

COMMONWEALTH EDISON COMPANY

IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

QUAD CITIES NUCLEAR POSER STATION, UNIT 2

DOCKET NO. 50-265

1.0 INTRODUCTION

By letter dated October 29, 1993, Commonwealth Edison Company (CECo or the licensee) submitted an amendment for resolving two unreviewed safety questions (USQs) dealing with modifications at Dresden Units 2 and 3, and Quad Cities Units 1 and 2, to eliminate possible errors in reactor vessel water level indication in accordance with NRC Bulletin 93-03, "Resolution of Issues Related to Reactor Vessel Water Level Instrumentation in BWRs." The modifications involve connecting the control rod drive (CRD) water header into the existing reactor vessel water level instrumentation reference leg. This connection provides a continuous backfill of deaerated water through the reference leg to prevent the accumulation of dissolved gases.

During review of the USQs, the staff discovered that the design of the modifications is such that the closure of the root valve in the reference leg piping would result in a major plant transient. On November 16, 1993, at the request of the staff, CECo made a presentation of the design modifications to meet Bulletin 93-03 requirements and the scenario of the root valve closure and resulting plant transients. CECo indicated in the meeting that preventing inadvertent closure of the root valve would be based only on strict administrative controls.

By letter dated November 26, 1993, the staff informed the licensee that the reliance on administrative controls alone as described in its plant-specific USQ is not an acceptable long-term approach to ensure that these isolation valves are not inadvertently closed. The staff also suggested that prior to its final decision on the USQs, CECo should reconsider its position to rely solely on administrative controls to prevent inadvertent closure of the root valve and factor into the USQ any lessons learned from the meeting. In addition, the staff issued Information Notice 93-39, "Potential Problems with BWR Level Instrumentation," providing additional information on the subject.

By letter dated December 22, 1993, the licensee informed the staff that it had elected to develop and install an alternative design which eliminates reliance upon administrative controls to address the concerns discussed in

Bulletin 93-03. This eliminated the need for review of the unreviewed safety questions for Quad Cities Unit 1 and Dresden Units 2 and 3. However, the licensee proposed to install an interim modification at Quad Cities Unit 2 until a final modification is completed in the 13th refueling outage for Unit 2, which is scheduled for fall 1994. Thus, this evaluation is applicable to Quad Cities Unit 2 only.

Finally, on January 14, 1994, the licensee requested that the Commission grant an emergency license amendment for resolving the unreviewed safety questions associated with the interim modifications.

2.0 DISCUSSION

NRC Bulletin 93-03 was issued on May 28, 1993, to notify all holders of operating licenses or construction permits for boiling water reactors (BWRs) except Millstone Unit 1 and Big Rock Point, about new information concerning reactor vessel water level indication errors which may occur during plant depressurization. The basic safety issue addressed in the Bulletin arises from the concern that noncondensable gases may become dissolved in the reference legs of the BWR reactor vessel water level instrumentation systems (RVLIS) during normal operation and later lead to a false high water level indication either after a rapid depressurization event or during a slow depressurization. The Bulletin requested that affected licensees take certain actions, including short-term compensatory actions and hardware modifications to ensure that the level instrumentation system is of high functional reliability for long-term operations. Licensees were required by the Bulletin to report if the requested actions would be taken. The Bulletin requested that the hardware modifications be implemented prior to startup from the next cold shutdown occurring after July 30, 1993.

The RVLIS backfill modification is being installed at Quad Cities Unit 2 in response to the Bulletin. The actual physical routing of the design is similar to the design that has previously been installed within the industry. The proposed backfill subsystem will resolve the concern of inaccurate reactor pressure vessel (RPV) level indication due to the presence of noncondensable gases in the RVLIS reference legs after a depressurization of the vessel. This modification to RVLIS includes the connection of a low flow, high pressure water supply to four reference legs to provide a continuous backfill through the reference leg, condensate pot and the reactor vessel. The source of the water supply is the CRD drivewater header, which operates at a pressure that is approximately 300 psi above reactor pressure. The new backfill subsystem provides deaerated water to the reference leg to prevent the accumulation of dissolved gases that can later come out of solution during reactor vessel depressurization.

The new subsystem prevents degraded level indications commonly appearing as "notches" by (1) forming a barrier of degassed water that will prevent gases from dissolving in the condensate pot and being transported down the reference leg, (2) purging the reference leg with deaerated water to sweep dissolved gases from the reference leg, and (3) providing a continuous fill of the reference leg in case noncondensable gases prevent adequate condensation in the condensate pot to keep the reference leg full of water.

The water supply from the CRD drivewater header flows to each of two instrument racks, 2202-5 and 2202-6 at Quad Cities Unit 2. Near these racks is a new panel with two flow stations and a water filter which acts as a pressure snubber. Each flow station consists of: needle valves for system startup and shutdown; metering valves for flow regulation; local flow indicators for setting flow rates through the backfill line for each reference leg; multiple check valves for safety-related to nonsafety-related system separation; instrument taps for testing components; a vent connection for purging air from the lines and isolation valves to isolate components for maintenance. A simplified drawing of the preliminary design is provided in CECo's letter dated October 29, 1993.

The licensee stated that the proposed modification increases the probability of a previously analyzed accident due to the potential for inadvertent closure of the reference leg root valve and subsequent pressurization of the RPV level and containment pressure instrumentation. Therefore, the licensee concluded that the installation of the modification with this configuration represents an Unreviewed Safety Question, and requires NRC review and approval prior to its implementation.

The non-safety and non-seismic CRD system will be actively connected to each division of reactor pressure vessel instrumentation. The connection of the non-safety-related backfill piping to the safety-related vessel instrumentation line requires that a safety-related isolation boundary be established. The isolation boundary will ensure that the vessel reference leg piping remains filled in the event of challenges to the piping integrity or depressurization of the CRD system piping. This isolation boundary is provided by two safety-related check valves in series. The check valves allow flow to the vessel instrumentation reference leg piping and prevent flow out of the reference leg piping. However, the licensee concluded that the installation of the modification with this configuration represents an Unreviewed Safety Question and requires NRC review and concurrence prior to its implementation.

The backfill lines are connected in such a manner that they do not have an adverse effect on the capability of the connected instruments to perform their function. The design of the backfill system satisfies the redundancy, independence and testability requirements of the reactor protection system. The safety-related portions of the backfill lines are designed to the same level of quality as the existing instrument lines; the check valves will not close accidentally during normal operation, but will close if instrument line integrity is challenged during normal or accident conditions.

3.0 EVALUATION

The modifications described above are being installed in response to an issue identified in Bulletin 93-03. The installation of these modifications will enhance plant safety by ensuring that the degassing phenomenon described in Bulletin 93-03 will not be encountered at Quad Cities Unit 2. The modifications are similar in design to the modifications that have been installed at other plants. The staff's evaluation of the two Unreviewed Safety Questions regarding the proposed modifications of reactor vessel water level instrumentation at Quad Cities Unit 2 is discussed below.

3.1 Inadvertent Closure of Root Valve

If a reference leg root valve were inadvertently closed in the current unmodified configuration, the instruments using that reference leg would be inoperable. The pressure instruments would indicate the pressure existing at the time of isolation or show a declining pressure if there is leakage from the reference leg. Level instruments would also lose accuracy but would not immediately cause engineered safety features (ESF) actuations. The following event is possible, though very unlikely, in the modified RVLIS system, if (1) the unit is at power, (2) the reference leg root valve is inadvertently closed, and (3) the backfill system has not been isolated. The backfill system will continue injecting CRD drivewater into the reference leg to pressurize it to approximately 1300 psig if a normal reactor vessel pressure of 1000 psig exists. This event causes the affected pressure instruments to indicate a false high reactor pressure and the level instruments to indicate or trip on a false low level indication.

The root valve(s) referred to above are installed so that the instrument lines can be isolated. Isolation of the instrument lines is required when the excess flow check valves in those lines are repaired, tested, or when the instrument lines are taken out of service for other reasons. The testing and/or repair of the instrument lines almost exclusively occur when the reactor is not in an operational mode to which the phenomenon described in Bulletin 93-03 is applicable. The station procedural controls governing the out-of-service process lessens the possibility of a valve manipulation error. The out-of-service process also ensures that the valves are properly returned to service. The licensee indicated that this administrative control is performed in conjunction with the usage of valve checklists that are performed prior to a unit startup. In addition, the status of the safety lock that will be installed on the valves will be checked at the end of each refueling outage prior to a unit startup.

The primary concern is a mismanipulation of the valves while the unit is at power. The plausible consequences of inadvertent closing of each reference leg root valve without first isolating the backfill subsystem vary depending on which valve is closed. The detailed discussion about the effect of each root valve closure is provided in the licensee's October 29, 1993 letter. The inadvertent closure of the root valve on the reference leg from 12A condensate pot causes the relief valves to immediately open, causing a loss of coolant. The emergency core cooling systems (ECCS) and reactor core isolation cooling (RCIC) start and inject into the vessel in a manner similar to a loss-of-coolant accident (LOCA). If the reference leg from the 13A or 13B condensate pot is isolated and that loop is being used for feedwater level control, the response of the plant is similar to that for a failed open feedwater regulating valve. The feedwater pump runout trip would not occur, but the high water level trip of the feedwater pumps and turbine would still be operable. The potential valve manipulation errors, therefore, can result in increased probability of occurrence or consequences of an accident previously analyzed in the safety analysis report.

The licensee suggested that these events are unlikely because these valves are located at the penetration where there are no normally operated valves. In addition, to minimize the errors, the licensee proposed the following features addressing administrative controls and training.

1. The valves will be locked in the open position with a lock, the keys for which will be administratively controlled. In addition, labels that clearly identify the valves will be provided at the valve location indicating that operation of the valves will result in a plant transient and that they are not to be operated without permission of shift supervision.

2. The operators and instrument maintenance technicians will be trained on the location and purpose of the valves and on the consequences of closing the valves without first taking the backfill system out of service. Also, the processes of taking the backfill system out of service and returning it to service will be administratively controlled by station procedures.

3. Training will be provided to the control room operators as part of the modification. This training will include directions concerning how to recognize the indications that the root valves have been mispositioned, and what actions to take to control a possible resultant transient.

After having a detailed discussion in a meeting with the licensee on November 16, 1993, the staff determined that the licensee should not solely rely on administrative controls to prevent inadvertent closure of the root valves. Therefore, by letter dated November 26, 1993, the staff informed the licensee that the reliance on administrative controls alone is not an acceptable long-term approach to ensuring that these isolation valves are not inadvertently closed. The staff suggested that the licensee should reconsider its position on relying solely on administrative controls to prevent inadvertent closure of the root valves prior to the staff making its final determinations on the USQ.

By letter dated December 22, 1993, the licensee informed the staff that it had elected to develop and install an alternative design which eliminates reliance upon administrative controls to prevent inadvertent closure of the root valves. The licensee stated that it will fully install the alternative design of the modification during the 13th refueling outage for Quad Cities Unit 2 expected to begin in September 1994. However, due to extended period of time until the start of this outage, the licensee indicated that it will complete the installation of an interim modification at Quad Cities Unit 2 during the current maintenance outage. The interim modification satisfies the concerns addressed in the Bulletin, but relies upon administrative controls until the upcoming refueling outage (Q2R13). However, the administrative controls associated with the modification were revised to include the installation of a welded "collar lock" on the root valve stem, which will be installed prior to declaring the backfill modification operable. The collar lock prevents the valve from closing, and removal of the collar lock would require mechanical removal in accordance with the approved station work control procedure. In addition to the collar lock, the licensee will utilize a valve specific lock/chain and install a physical cage around the root valve to further reduce the potential for inadvertent closure during the operating cycle.

The staff has reviewed the licensee's proposed interim modifications including the revised administrative controls for Quad Cities Unit 2 and determined that the licensee has proposed proper precautionary measures to avoid mismanipulation of root valves while the reactor is operating at power. The modification will enhance plant safety by ensuring that the degassing phenomenon described in the Bulletin will not be encountered at Quad Cities Unit 2. The benefit achieved from the interim modification outweighs the disadvantage of small increases in the probability of an accident previously analyzed in the safety analysis report. Therefore the interim modification is acceptable.

3.2 Challenges to RVLIS Accuracy

The licensee also indicated that the proposed modification connects the non-safety-related CRD system to each division of RPV instrumentation. The failure of CRD piping or loss of CRD system pressure could result in challenges to RPV instrumentation due to reference leg leakage. However, the licensee indicated that the isolation action of the redundant safety-related reference leg backfill instrument check valves limit the consequences associated with this malfunction. The licensee has established a test leakage rate of 3.0 cc/hr for RVLIS backfill check valves. This criterion was conservatively established to ensure that instrument accuracy will be maintained during a pipe break in the non-safety-related piping or a loss of the CRD system pressure. The licensee also indicated that the RVLIS backfill instrument check valves will be periodically tested as part of the inservice testing program.

Based on the review of critical seat leakage rate for the RVLIs backfill instrument check valves and the testing criteria established by the licensee the staff agrees with the licensee's conclusion that potential failures in non-safety related piping will result in a leakage rate less than those previously found acceptable for the present RVLIS design configuration. Therefore, the staff finds the design to be acceptable.

In summary, the staff finds that the licensee's proposed interim modification to address the requirements of the Bulletin will enhance the overall safety of Quad Cities Unit 2 until the final modification is completed during the next refueling outage in fall 1994. The proposed interim modification is expected to meet the applicable standards and it is similar to the design installed at other plants in the industry. Therefore, the proposed interim modification is acceptable.

4.0 <u>DISCUSSION OF EMERGENCY SITUATION</u>

10 CFR 50.91(a)(5) provides the necessary requirements for issuing an amendment when the Commission finds that an emergency situation exists and failure to act in a timely way would result in derating or shutdown of a nuclear plant, or in prevention of resumption of operation. The Commission expects its licensees to: apply for a license amendment in timely fashion; not abuse the emergency provisions by failing to make a timely application for the amendment and thus itself creating the emergency; provide an explanation as to why the emergency situation occurred; and why it could not have been avoided.

As discussed before, on October 29, 1993, the licensee originally submitted proposed license amendments for the resolution of two Unreviewed Safety Questions (USQ) for Quad Cities Units 1 and 2, and Dresden Units 2 and 3. In this submittal the licensee requested NRC review and approval for Quad Cities Unit 1 on an Exigent Basis to support the installation of the modification during a planned maintenance outage in November 1993. On November 9, 1993, the staff issued a Notice of Consideration of Amendment for Quad Cities Unit 1 in the <u>Federal Register</u> (58 FR [illegible]495). However, no notice was issued for Quad Cities Unit 2 and Dresden Units 2 and 3 because of an administrative error by the staff.

The October 29, 1993, submittal was supplemented by the licensee by letter of December 22, 1993. In this submittal, the licensee requested the staff's review and approval of the proposed interim modification for Quad Cities Unit 2 which was planned for installation and operation prior to startup following the current maintenance outage. The unit was expected to startup on January 15, 1994.

On January 13, 1994, the licensee identified that the license amendment for Quad Cities Unit 2, which was originally submitted on October 29, 1993, had not been noticed in the <u>Federal Register</u>. On January 14, 1994, the staff informed the licensee that the proposed license amendment for Quad Cities Unit 2 could not be approved prior to startup unless CECo submitted an emergency license amendment request. Therefore, to support startup of Quad Cities Unit 2 with the interim RVLIS modification operable, by letter dated January 14, 1994, the licensee requested that the NRC grant an emergency license amendment for review and approval of two Unreviewed Safety Questions associated with the interim modification. The Federal Register Notice on the proposed license amendment for Quad Cities Unit 1 is identical in technical content to the license amendment for Quad Cities Unit 2.

Based on the above circumstances, the staff has determined that the licensee has not abused the emergency provision of 10 CFR 50.91(a)(5), and failure of the modifications outlined in Bulletin 93-03. Therefore, the request should be processed under the emergency provisions of 10 CFR 50.91(a)(5).

5.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The Commission's regulations in 10 CFR 50.92 state that the Commission may make a final determination that a license amendment involves no significant hazards considerations, if operation of the facility, in accordance with the amendment would not:

(1) Involve a significant increase in the probability or consequences of any accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

This amendment has been evaluated against the standards in 10 CFR 50.92. It does not involve a significant hazards consideration because the changes would not:

1. Involve a significant increase in the probability or consequences of any accident previously evaluated.

   The addition of the backfill instrumentation piping does not significantly increase the probability of an accident previously evaluated due to the low probability of the inadvertent closure of the root valve(s). CECo has evaluated the estimated frequency of the inadvertent closure of the root valve(s) at approximately 1E-08 per reactor year given the implementation of administrative controls. The resulting condition (valve mismanipulation) places the reactor pressure vessel through a transient similar to that of a plant LOCA (i.e., simulates LOCA conditions). The current (pre-modification) LOCA initiation frequency is predicted to be approximately 1E-04 per reactor year. Therefore, the proposed modifications do not significantly increase the probability of any previously evaluated accident.

   The consequences of any previously evaluated accident are not increased by the proposed modifications. For example, the consequence of closing the root valve for the reference leg from condensing chamber 12A, without first isolating the backfill injection, is the inadvertent pressurization of the reference leg resulting in the opening of the SRV and all electromatic reliefs. This is equivalent to an inadvertent actuation of the automatic depressurization system (ADS); an event that is not analyzed in the safety analysis as an initiating event. However, the event is bounded by the recirculation line break analysis in terms of the RPV response. Because this event would release reactor inventory to the suppression pool, it has less significant consequence than other events previously analyzed for Quad Cities Unit 2.

2. Create the possibility of a new or different kind of accident from any accident previously evaluated.

   For Quad Cities Unit 2, a spectrum of Loss-of-Coolant Accidents has previously been evaluated. The accident in question associated with the proposed modifications can be categorized as a LOCA due to the resultant plant response following the initiating conditions. The previously analyzed LOCA analyses bound the conditions introduced by the proposed modifications. As such, the proposed amendment request for Quad Cities Unit 2 does not introduce any new or different kinds of accidents.

   The proposed modification connects the non-safety-related CRD system to each division of RPV instrumentation. The failure of the CRD piping may result in instrument line leakage. However, this event is mitigated by the isolation action of the reference leg backfill instrument check valves. Although the proposed modifications may introduce the potential for a malfunction of equipment of a different type than previously evaluated in the safety analysis report, the proposed amendment request for Quad Cities Unit 2 does not introduce any new or different kinds of accidents.

3. Involve a significant reduction in a margin of safety.

   The previously analyzed LOCA consequences bound the consequences introduced by the inadvertent closure of the root valve(s) and subsequent LOCA conditions. As such, the previously approved safety margin remains unchanged. Therefore, the proposed modifications do not significantly reduce the margin of safety for Quad Cities Station Unit 2.

   The proposed amendment request does not involve a significant relaxation of the criteria used to establish safety limits, a significant relaxation of the bases for the limiting safety system setting, or a significant relaxation of the bases for the limiting conditions for operations.

Accordingly, the Commission has determined that this amendment involves no significant hazards consideration.

6.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Illinois State official was notified of the proposed issuance of the amendment. The State official had no comments.

7.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has made a final no significant hazards consideration determination with respect to this amendment. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

8.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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