Commonwealth Edison
1400 Opus Place
Downers Grove, Illinois 60515

March 6, 1994

Mr. John B. Martin
Regional Administrator
U.S. Nuclear Regulatory Commission
Region III
801 Warrenville Road
Lisle, IL 60532-4351

Subject: Quad Cities Nuclear Power Station Unit 1
Request for Regional Enforcement Discretion Regarding Facility Operating License DPR-29, Appendix A, Technical Specification 3.7.D
NRC Docket No. 50-254

Reference: Teleconferences on March 5, 1994 between CECo (R. Pleniewicz, et al) and the NRC (B. Clayton, et al)

Dear Mr. Martin:

This letter documents the results of teleconferences held on March 5, 1994, between Commonwealth Edison (CECo) and the NRC Staff, in which Commonwealth Edison requested a Notice of Enforcement Discretion from Technical Specification 3.7.D for Quad Cities Unit 1.

On March 4, 1994, Quad Cities declared the inboard isolation valve on the Reactor Core Isolation Cooling (RCIC) system **(MO 1-1301-16)** inoperable due to the results of an EPRI test concerning the valve factor. In accordance with Technical Specification 3.7.D, the outboard isolation valve on the RCIC steam supply line **(MO 1-1301-17)** was closed, rendering RCIC inoperable, and placing the station in a 14-day time clock (Technical Specification 3.5.E).

In the referenced teleconferences, CECo requested that Quad Cities Unit 1 be allowed to operate with the inboard and outboard isolation valves on the RCIC steam supply line open, and the inboard isolation valve inoperable. This would allow RCIC to be operable and available for accident mitigation. A Notice of Enforcement Discretion was verbally approved by Region III on March 5, 1994, during the referenced teleconferences.

The basis of the request is provided in Attachment 1 and includes:

- The Technical Specification for which a Notice of Enforcement Discretion is requested;
- The circumstances surrounding the condition, including the need for prompt action;

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- The safety basis for the request that enforcement discretion be exercised, including an evaluation of the safety significance and potential consequences of the proposed course of action;

- The proposed compensatory measures;

- The justification for the duration of the request;

- The basis for the conclusion that the request will not have a potential adverse impact on the public health and safety and that a significant safety hazard is not involved;

- The basis for the conclusion that the request will not involve adverse consequences to the environment.

CECo will shut down Quad Cities Station Unit 1 on March 13, 1994 for the start of the 13th refuel outage. During that outage, the valve will be made operable.

This request does not affect Unit 2, as the HPCI and RCIC steam supply valves on that unit, as well as the HPCI valves and outboard isolation valve on RCIC for Unit 1, have been shown to be operable.

This request for Enforcement Discretion has been reviewed and approved by the Quad Cities Station Plant Operations Review Committee per station policy and procedure.

CECo appreciates the NRC staff's effort and participation in the review of this request. Please direct any questions or comments to John Schrage, Nuclear Licensing Administrator, at (708) 663-7283.

Very truly yours,

John L. Schrage
Nuclear Licensing Administrator

Attachment

cc: Document Control Desk - NRC Docket No. 50-254
B. Clayton, Branch Chief - Region III
C. Miller, Senior Resident Inspector - Quad Cities
C. Patel, Project Manager - NRR
Office of Nuclear Facility Safety - IDNS

REQUEST FOR ENFORCEMENT DISCRETION
QUAD CITIES STATION
UNIT 1
MARCH 6, 1994

1. TECHNICAL SPECIFICATION OR LICENSING CONDITION WHICH REQUIRES ENFORCEMENT DISCRETION

Technical Specification Section 3.7.D provides the requirements for Primary Containment Isolation Valves. Specification 3.7.D.1 requires that during power operating conditions, all isolation valves listed in Table 3.7-1 which contact the primary coolant system be operable except as specified in Specification 3.7.D.2. Specification 3.7.D.2 allows for continued power operation in the event any isolation valves specified in Table 3.7-1 become inoperable provided that at least one valve in each line having an inoperable valve is in the mode corresponding to the isolated condition. Specification 3.7.D.3 requires the initiation of an orderly shutdown of the reactor, in order to achieve the cold shutdown condition within 24 hours, if Specifications 3.7.D.1 and 3.7.D.2 cannot be met.

On March 4, 1994, Commonwealth Edison Company (CECo) determined that the Reactor Core Isolation Cooling (RCIC) steam line inboard isolation valve, **MO 1-1301-16**, would not perform its required design isolation function. As a result, the outboard isolation valve, **MO 1-1301-17**, was placed in the isolated condition (closed) in accordance with Specification 3.7.D.2, and the Unit 1 RCIC System was declared inoperable. In order to restore the availability of RCIC until the upcoming Unit 1 refueling outage (Q1R13), which is scheduled to commence on March 13, 1994, CECo is requesting Enforcement Discretion from Technical Specification Section 3.7.D for **MO 1-1301-16**, for one time only, until March 13,1994. This Enforcement Discretion will allow the reopening of the **MO 1-1301-17** valve, and thus restore the availability of the Unit 1 RCIC System.

2. CIRCUMSTANCES SURROUNDING THE SITUATION

On December 12, 1993, the NRC issued Information Notice 93-88, "Status of Motor-Operated Valve Performance Prediction Program by the Electric Power Research Institute," which described EPRI's performance prediction testing program for MOVs and presented some preliminary results. The preliminary EPRI results were received by CECo at an industry meeting in Dallas, Texas on December 1, 1993. On December 14, 1993, EPRI issued the formal test results, specifications, and drawings for EPRI flow loop tested valves to the utilities which sponsored the testing. CECo began evaluating the test results for applicability. The CECO evaluation encompassed discussions with Crane and EPRI to determine which blowdown valves were similar to the EPRI-tested 6" Crane valves. At Quad Cities, those valves were determined to be the 1(2)-1301-16 & 17 (RCIC Steam Line Isolation) and the 1(2)-2301-4 & 5 (HPCI Steam Line Isolation), which are Crane carbon steel 783U gate valves.

On March 3, 1994, results from the vendor and EPRI feedback were evaluated by corporate and site engineering. On the basis of this evaluation, all Crane carbon steel 783U blowdown valves were determined to be susceptible to the damage

mechanism identified by the EPRI testing. An operability evaluation was initiated, and it was concluded that the **MO 1-1301-16** valve was inoperable (on 3/4/94). All other susceptible blowdown valves (2-1301-16, 1(2)-1301-17, and 1(2)-2301-4 & 5) were determined to be operable.

Currently, the Unit 1 RCIC System is inoperable, with the **MO 1-1301-17** va... closed. Prompt action is necessary to re-establish the operability of the RCIC system during operation of Unit 1 until March 13, 1994.

3. EVALUATION OF SAFETY SIGNIFICANCE AND CONSEQUENCES

The RCIC Steam Supply Containment Isolation valves (see attached Figure, "Quad Cities 1 Reactor Core Isolation Cooling System") have the following safety functions:

- They form part of the reactor coolant pressure boundary.
- They isolate the RCIC steam line, and therefore the reactor vessel, in case of a line break in the RCIC steam line.

MO 1-1301-16 is capable of isolating the steam line at a maximum dP of 1084 psid. The valve is required to close across a maximum dP of 1146 psid.

MO 1-1301-17 is capable of isolating the steam line in the event of a RCIC steam line break at a dP of 1146 psid. This valve receives an isolation signal at the same time as **MO 1-1301-16**. Assuming that both valves have power to close, the total differential pressure between the reactor and the break would be shared across the two valves. The reduced pressure load will increase the likelihood of **MO 1-1301-16** closing.

PROBABILITY RISK ASSESSMENT REVIEW

A core damage frequency comparison between leaving RCIC inoperable and having a RCIC steam line break without isolation was performed. The result of this evaluation was that the estimated core damage frequency increase for having RCIC inoperable was 24 times higher than for the RCIC steam line rupture without isolation. CECo has concluded that the plant would be in a safer condition by opening **MO 1-1301-17**.

LIKELIHOOD OF A LINE BREAK

There is a probability of 4.4×10^{-7} that the RCIC steam line will break in the next nine days. This is based on a piping failure rate of 8.6×10^{-11}/Hr-section with 20 piping sections in this line, plus a valve failure rate of 1.5×10^{-10}/Hr-component with 2 valves in the line.

Additionally, as described in the CECo Safety Assessment submitted i... ...ponse to Generic Letter 89-10 Supplement 3 (D. Taylor to USNRC letter dated Ma...n 11,

1991), the Leak Before Break (LBB) concept is based on the fact that reactor piping is fabricated from tough ductile materials which can tolerate large through-wall cracks without fracture under service loadings. By monitoring the leak rate from through-wall cracks and setting conservative limits on the leakage, cracks in piping can be detected well before the margin to rupture is challenged.

In NUREG 1061, Volume 3, the NRC Piping Review Committee outlined the limitations and general technical guidance on LBB analysis to justify mechanistically that breaks in high-energy fluid system piping need not be postulated. In General Design Criterion (GDC) 4, the NRC has formalized the use of the LBB approach to justify the elimination of pipe-whip restraint and jet impingement barriers as design requirements for hypothetical Double Ended Guillotine Break (DEGB) in high energy reactor piping systems.

As detailed in the GL 89-10, Supplement 3 Safety Assessment, the LBB approach is used to illustrate the concept of detecting and mitigating leakage prior to a pipe rupture. Leakage from a through-wall crack with a length up to, but less than, the critical crack length would be large enough to be readily detected. Thus isolations can be achieved well before the crack grows to critical length and well below the point where design basis flows and pressures are established.

Based on the evaluation results, it is concluded that the RCIC steam supply line is expected to develop a detectable leak prior to reaching the point of incipient break. Thus, a DEGB in these lines is highly unlikely.

The crack and leak will be detected prior to the DEGB. This allows sufficient time for the RCIC isolation valves to be closed prior to a condition developing that would create a dP greater than 1084 psid. In this event either the **MO 1-1301-16** or the **MO 1-1301-17** valve will close.

EFFECTS OF A RCIC LINE BREAK

If the RCIC Room temperature is raised to the high temperature isolation setting (170°F), or if the RCIC steam line reaches 300% flow, **MO 1-1301-16** and **MO 1-1301-17** will receive a signal to isolate and a control room annunciator will alarm.

If this line breaks in the drywell, the containment isolation valves will close since the differential pressure across the valves would be less than 1084 psid. The effects of this event would be bounded by the recirculation line break accident scenario described in the FSAR.

Realistically, there may exist an obstruction from the downstream piping system if the line break was not a guillotine type break. This obstruction may reduce the differential pressure across the valve. If the line break does not occur at the valve, the affects of pipe elbows and line losses may also lower the differential pressure across the isolation valve. A reduced differential pressure across the valve will require less thrust for valve closure during a line break event.

The steam flow out the RCIC line break is reduced in proportion to the reactor pressure during depressurization. After full reactor depressurization, steam flow is assumed to be minimal. Therefore, inventory loss will be terminated after full depressurization.

This break scenario is anticipated to result in liquid inventory losses less than the design basis MSLB event. This break would occur inside the secondary containment, with additional removal of radionuclides accomplished by the Standby Gas Treatment (SBGT) System, as well as deposition within the reactor building. Therefore this scenario does not present unacceptable consequences beyond previously licensed events.

The water necessary to makeup the inventory loss for this Loss of Coolant Accident (LOCA) is well within the capabilities of the station's water storage systems. The two condensate storage tanks have a capacity of 500,000 gallons. The HPCI suction is normally from these tanks. The low pressure ECCS pumps can draw suction from the Torus (1,000,000 gallons) that can also restore the lost reactor coolant inventory.

The RCIC isolation valves are qualified for the harsh environment that would be created by a guillotine rupture. The ECCS pumps and motors are qualified to a steam environment and are expected to perform their design functions in the event of a line break.

The divisionalized ECCS equipment is located in rooms adjacent to the Torus. The walls of these rooms are flood barriers equipped with submarine doors. The motors for the pumps are elevated so that flooding in the individual rooms will not necessarily render the equipment inoperable. The ECCS equipment rooms all communicate with the Torus area through pipe chases. These wall openings are located above the maximum flood height. Emergency Operating Procedures for Secondary Containment control provide for mitigation of flooding. The Torus basement sumps will receive a high level alarm, which may provide useful information to the control room operator. These sumps isolate on low reactor water level, high drywell pressure or high drywell radiation. In summary, the existing flood protection features are expected to maintain the ECCS equipment operable during and after the line break event.

The release of large quantities of steam to the reactor building will result in an increase to on-site dose. However, it is postulated that most of the release will be confined to the reactor building and processed by Radwaste and SBGT. Area Radiation Monitors would alert the operators and personnel that radiological conditions were changing.

The on-site dose consequences of a line break are expected to be less severe than a Main Steam Line break (MSLB) scenario, which releases large quantities of steam to the turbine building. A turbine building release is expected to have a greater effect on control room doses than the unisolable RCIC steam line break.

The quantities of steam released by the postulated unisolable RCIC steam line breaks are less than or equal to that postulated for a MSLB scenario. This scenario,

which is evaluated in the FSAR, results in a release well below the limits of 10 CFR 100. Therefore, the unisolable RCIC line break will not result in off-site doses in excess of 10 CFR 100 limits.

In the unlikely event of an unisolable pipe break, the operators, following the direction provided in the Emergency Operating Procedures, would SCRAM the unit and depressurize the vessel. This action would minimize the force driving reactor coolant out the line break. Any emergency core cooling pump is capable of restoring the lost reactor coolant inventory. The ECCS pumps are environmentally qualified and protected by flood barriers. A line break of the RCIC system is bounded by the release due to the Main Steam Line Break described in the FSAR. The MSLB at full power results in 40% of the allowable values of 10 CFR 100. Therefore, the on-site and off-site dose consequences of an unisolable RCIC pipe break will be below the requirements of 10 CFR 100.

4. COMPENSATORY ACTIONS

Compensatory measures, as indicated below, will be implemented to support this request for Enforcement Discretion:

The operating crews will be briefed on this issue as they come on shift. They will be given direction to attempt to close **MO 1-1301-17** and **MO 1-1301-61** in the event of an unisolated RCIC steam line break. This may reduce the dP across **MO 1-1301-16** enough to allow the valve to isolate the break. In the event the line can not be isolated, the operators have guidance in the Emergency Operating Procedures to depressurize the reactor when an area reaches Maximum Safe Operating Temperature or Radiation level.

A caution card will be placed on the control switch for the RCIC inboard steam supply isolation valve, explaining that the valve may not close during a full dP blowdown through a break in the RCIC steam supply line.

No unnecessary preventative maintenance or system outages will be performed.

No work that increases the potential for a line break will be performed.

5. JUSTIFICATION FOR THE DURATION OF THE REQUEST

The requested duration for the enforcement discretion is until March 13, 1994, at which time Unit 1 is scheduled to shutdown for the 13th refueling outage (Q1R13). This timeframe is justified by the following items:

The power provided by Unit 1 will be important to the CECo system during the next week. The Commonwealth Edison Company has six coal and four nuclear units down next week.

Compounding the situation is the fact that a railroad strike has reduced the quantity of coal to the Midwest (inventories at all of CECo's coal Stations are quite low). There presently exists a court injunction to halt the strike; however restoration of coal inventories is not assured.

Iowa-Illinois Gas and Electric is also in a coal conservation mode, due to the strike, and has shut down the Louisa plant. This strike and the potential for flooding of the Mississippi River have compounded their coal delivery problems and forced the utility to curtail electric generation.

Repair of the **MO 1-1301-16** would require a plant shutdown, and would not be completed prior to the refuel outage. Although work on this valve was planned for the refuel outage, this work scope will be enhanced to assure valve operability after the refuel outage.

The likelihood of a break in the steam supply piping for the RCIC system occurring during the requested time frame is extremely low.

6. EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATION

Commonwealth Edison has evaluated this request for Discretionary Enforcement and determined that it does not represent a significant hazards consideration. Based on the criteria for defining a significant hazards consideration established in 10 CFR 50.92, operation of Quad Cities Station Unit 1 in accordance with the proposed request will not:

Involve a significant increase in the probability or consequences of an accident previously evaluated because:

The proposed request will allow operation of Unit 1 with the inboard RCIC steam supply valve open and inoperable, while the outboard isolation valve is open and operable. This does not change the probability of any previously analyzed accident, nor does it significantly increase the consequences of a previously analyzed accident.

The steam supply isolation valve for RCIC does not affect any accident precursors or initiators, and therefore, does not affect the probability of a previously analyzed accident.

The consequences of a steam line break on the RCIC system are not significantly increased because of the items noted in section 3, and listed below.

a) There is a high likelihood that the pipe will leak such that it can be isolated before it breaks;

b) There is a limited likelihood for the occurrence of a break in the RCIC steam supply line outside containment during the short time frame for which the request is being made;

c) There is a high likelihood that the break will not be a full guillotine rupture;

d) The radiological affects and inventory loss are bounded by the MSLB analysis. The on-site and off-site dose consequences of the MSLB are below 10 CFR 100 limits;

e) There is a high likelihood that the operable outboard valve will close, isolating the line. This will decrease the differential pressure that the inboard valve will have to close against.

The radiological consequences of a steam supply line break on the RCIC system are not significantly increased, as supported by the items listed above. Therefore, the consequences of a previously analyzed accident are not significantly increased.

Create the possibility of a new or different kind of accident from any accident previously evaluated because:

The proposed request will allow operation of Unit 1 with the inboard RCIC steam supply valve open and inoperable, while the outboard isolation valve is open and operable. This will allow RCIC to be operable and available for accident mitigation. A steam line break is an analyzed accident. This mode of operation does not create the possibility of a new or different kind of accident.

Involve a significant reduction in the margin of safety because:

Allowing the Unit to be operated with an open and inoperable RCIC steam supply valve does not significantly reduce the margin of safety. The low likelihood of a steam line break during the short time frame for which the request is being made, the high likelihood of identifying a leak with time to isolate the leak, the high likelihood that the other isolation valve will operate as designed (and the positive effect this may have on the ability of the inboard valve to operate), and the bounding analysis for the MSL break, support that the margin to 10 CFR 100 limits as a result of having one inoperable RCIC steam supply valve for the time period requested, and with the noted compensatory actions in place, is not significantly reduced.

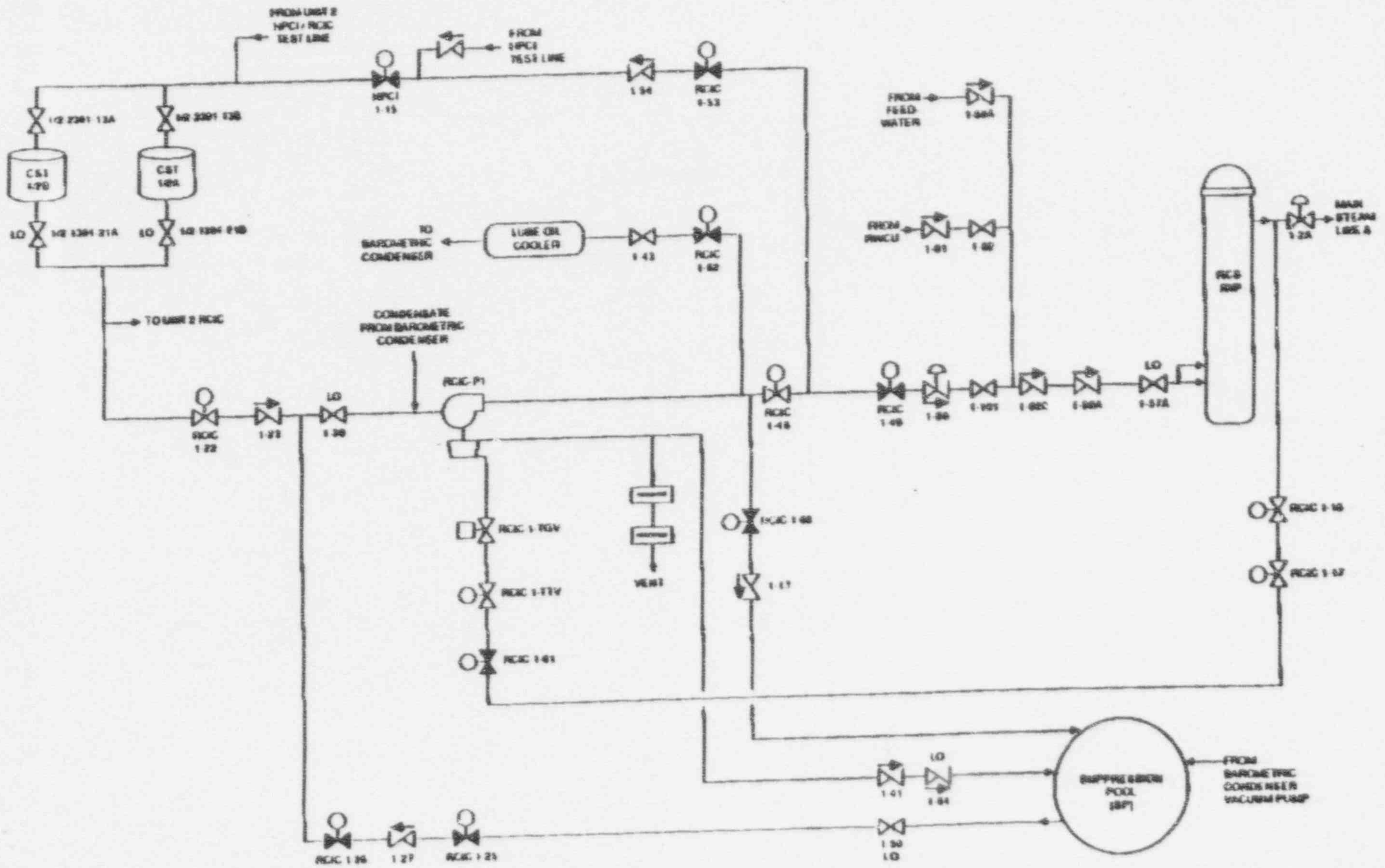
7. ENVIRONMENTAL ASSESSMENT

Commonwealth Edison has evaluated the proposed discretionary enforcement request against the criteria for the identification of licensing and regulatory actions requiring an environmental assessment in accordance with 10 CFR 51.21. It has been determined that the proposed change meets the criteria for a categorical exclusion as provided in 10 CFR 51.22(c)(9). This conclusion has been determined because the proposed changes do not pose a significant hazards consideration and do not involve a significant increase in the amounts or changes in the types of effluents released offsite. The proposed change does not involve a significant increase in individual or cumulative occupational radiation exposure. The quantities

of steam released by the postulated unisolable RCIC steam line breaks are less than or equal to that postulated for a MSLB scenario. This scenario, which is evaluated in the FSAR, results in a release well below the limits of 10 CFR 100. Therefore, the unisolable RCIC line break will not result in off-site doses in excess of 10 CFR 100 limits.

8. APPROVAL BY STATION REVIEW

The request has been reviewed and approved by the Plant Operating Review Committee.



Quad Cities 1 Reactor Core Isolation Cooling System