June 27, 1990

Docket No. 50-254

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Mr. Thomas J. Kovach
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1400 OPUS Place
Downers Grove, Illinois 60515

Dear Mr. Kovach:

SUBJECT: ISSUANCE OF EMERGENCY TECHNICAL SPECIFICATIONS AMENDMENT (TAC NO. 76818)

The Commission has issued the enclosed Amendment No. 125 to Facility Operating License No. DPR-29 for the Quad Cities Nuclear Power Station, Unit 1. The amendment is in response to your application dated May 19, 1990.

The amendment excludes eight containment leak pathways from Appendix J, Type C testing, until the next refueling outage scheduled for October 1990. A temporary waiver of compliance was approved orally on May 18, 1990, and in writing on May 22, 1990. The waiver allowed the staff sufficient time to process this emergency technical specification amendment.

A copy of the related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

/s/

Leonard N. Olshan, Project Manager
Project Directorate III-2
Division of Reactor Projects - III, IV, V and Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 125 to License No. DPR-29
2. Safety Evaluation

cc w/enclosures:
See next page

DOCUMENT NAME: 76818 AMD

| Office: | LA/PDIII-2 | PM/PDIII-2 | (A)PD/PDIII-2 | OGC-WF1 | BC/PRPB | BC/SPLB |
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| Date: | 6/14/90 | 6/6/90 | 6/6/90 | 6/15/90 | 6/11/90 | 6/25/90 |

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

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2. Safety Evaluation

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See next page

Mr. Thomas J. Kovach
Commonwealth Edison Company

Quad Cities Nuclear Power Station
Unit Nos. 1 and 2

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

COMMONWEALTH EDISON COMPANY

AND

IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

DOCKET NO. 50-254

QUAD CITIES NUCLEAR POWER STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 125
License No. DPR-29

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated May 19, 1990, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B. of Facility Operating License No. DPR-29 is hereby amended to read as follows:

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B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 125 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Richard J. Barrett, Director
Project Directorate III-2
Division of Reactor Projects - III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 27, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 125

FACILITY OPERATING LICENSE NO. DPR-29

DOCKET NO. 50-254

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change.

| REMOVE | INSERT |
|---|---|
| v | v |
| 3.7/4.7-3 | 3.7/4.7-3 |
| 3.7/4.7-4 | 3.7/4.7-4 |
| 3.7/4.7-5 | 3.7/4.7-5 |
| | 3.7/4.7-40 |

QUAD CITIES
DPR-29

TECHNICAL SPECIFICATIONS

APPENDIX A

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| 3.1-3 | Reactor Protection System (Scram) Instrumentation Requirements-Run Mode | 3.1/4.1-13 |
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| 4.8-1 | Radioactive Gaseous Waste Sampling and Analysis Program | 3.8/4.8-27 |

2. Primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel except while performing low power physics tests at atmospheric pressure at power levels not to exceed 5 MWt.

   a. When primary containment integrity is required, primary containment leakage rates shall be limited to:

      1) An overall integrated leakage rate of:

         a) $\leq L_a$, 1.0 percent by weight of the containment air per 24 hours at $P_a$, 48 psig, or

         b) $\leq L_t$, 1.0 percent by weight of the containment air per 24 hours at a reduced pressure of $P_t$, 25 psig.

      2) A combined leakage rate of $\leq 0.60\ L_a$ for all penetrations and valves, except the main steam isolation valves, and the M01-220-1 valve (while it is inoperable), subject to Type B and C tests when pressurized to $P_a$. The M01-220-1 valve will be restored to operability no later than startup after the first Cold Shutdown following a 30-day period after December 21, 1982. Also temporarily excluded are those pathways specified in Section 4.2.d.5).

2. The containment leakage rates shall be demonstrated at the following test schedule, and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR 50 using the methods and provisions of ANSI N45.4 (1972).

   a. Three Type A tests (Overall Integrated Containment Leakage Rate) shall be conducted at 40 ±10 month intervals during shutdown at either $P_a$, 48 psig, or at $P_t$, 25 psig during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection.

3) 11.5 scf per hour for any one main steam isolation valve when tested at 25 psig.

b. With the measured overall integrated containment leakage rate exceeding 0.75 $L_a$ or 0.75 $L_t$, as applicable, restore the overall integrated leakage rate(s) to $\leq$ 0.75 $L_a$ or $\leq$ 0.75 $L_t$, as applicable. Temporarily excluded are those pathways specified in Section 4.2.d.5).

b. If any periodic Type A test fails to meet either 0.75 $L_a$ or 0.75 $L_t$, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet either 0.75 $L_a$ or 0.75 $L_t$, a Type A test shall be performed at each shutdown for refueling or approximately every 18 months, whichever occurs first, until two consecutive Type A tests meet either 0.75 $L_a$ or 0.75 $L_t$, at which time the above test schedule may be resumed.

c. With the measured combined leakage rate for all penetrations and valves, except for main steam isolation valves, subject to Type B and C tests exceeding 0.60 $L_a$, restore the combined leakage rate for all penetrations and valves, except for main steam isolation valves, subject to Type B and C tests to 0.60 $L_a$. Also temporarily excluded are those pathways specified in Section 4.2.d.5).

c. The accuracy of each Type A test shall be verified by a supplemental test which;

1) Confirms the accuracy of the test by verifying that the difference between the supplemental data and the Type A test data is within 0.25 $L_a$ or 0.25 $L_t$.

2) Has a duration sufficient to establish accurately the change in leakage rate between the Type A test and the supplemental test.

3) Requires the quantity of gas injected into the containment or bled from the containment during the supplemental test to be equivalent to at least 25 percent of the total measured leakage at $P_a$, 48 psig, or $P_t$, 25 psig.

d. Leakage shall be limited to a leakage rate of less than or equal to 3.75 percent of $L_a$ for any one air lock when pressurized to 10 psig.

d. Type B and C tests shall be conducted at $P_a$, 48 psig, at intervals no greater than 24 months except for tests involving:

1) Air locks which shall be tested at 10 psig at least once per 18 months, and

2) Main steam isolation valves which shall be leak tested at least once per operating cycle, of a frequency not to exceed 24 months, at a pressure of 25 psig.

3) Bolted double-gasketed seals which shall be tested at a pressure of 48 psig whenever the seal is closed after being opened and each operating cycle.

4) While valve MO1-220-1 is inoperable, valves MO1-220-2, MO1-220-3, and MO1-220-4 shall be VERIFIED closed after each closure.

5) The pathways identified in Table 4.7-1, which will not be tested until the end of cycle 11 refueling outage.

TABLE 4.7-1

TEMPORARILY UNTESTED PATHWAYS*

| System | Penetration No. | Line No. |
|---|---|---|
| Instr. Air | X216 | 47138-1/2" |
| Instr. Air | X22 | 47200-2" |
| RBCCW | X23 | 3769-10" |
| RBCCW | X24 | 3747-10" |
| Core Spray | X16B | 1404-10" |
| Core Spray | X16A | 1403-10" |
| SBLC | X47 | 1102-1 1/2" |
| Clean Demin. Water | X20 | 4309-3" |

*Local Leak Rate Testing not required until End of Cycle 11 refueling outage.

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

# SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

## SUPPORTING AMENDMENT NO. 125 TO FACILITY OPERATING LICENSE NO. DPR-29

## COMMONWEALTH EDISON COMPANY

## AND

## IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

## QUAD CITIES NUCLEAR POWER STATION, UNIT 1

## DOCKET NO. 50-254

## 1.0 INTRODUCTION

By letter dated May 19, 1990, Commonwealth Edison Company (the licensee) submitted a request for a temporary waiver of compliance and emergency Technical Specification change for Quad Cities Nuclear Power Station, Unit 1. The proposed change to the Technical Specification excludes eight containment pathways from the requirement to perform Type C local leak rate tests in accordance with Appendix J to 10 CFR Part 50 until the next refueling outage scheduled for October 1990. This proposed change was verbally requested on May 18, 1990 and verbally granted by the staff later on that same day. By letter dated May 22, 1990, the staff provided written approval of the temporary waiver of compliance.

## 2.0 DISCUSSION AND EVALUATION

In addition to not being Type C tested, the valves in the eight containment pathways were not properly included as part of the tested containment boundary during the last containment integrated leak rate test (Type A test). Thus, neither the sum of local leak rates (Type B and Type C) nor the overall integrated containment leak rate (Type A) is known. Because the valves cannot be tested with the plant at power, nor can the lines be isolated by other barriers with the plant at power, the requested change to the Technical Specifications was needed to allow continued plant operation.

Even though the subject valves are untested, there are other valves, piping systems, and other components which could act as barriers to containment leakage through the eight containment pathways under consideration. Although these components and systems may not be safety-grade or may only satisfy some but not all of the staff's standards for safety-grade equipment, they nevertheless provide some assurance that, during the limited time period until the next refueling outage, potential leakage through the eight pathways would be limited or eliminated during an accident. These considerations, which are described below, provide sufficient assurance such that the staff's engineering judgement is that the plant may operate until the next refueling outage without undue risk to the public health and safety.

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The eight untested pathways are associated with five systems: clean demineralized water, core spray, standby liquid control, instrument air, and reactor building closed cooling water. Each is discussed below.

(1) Clean Demineralized Water, Penetration X-20

This pathway is a single three-inch line that penetrates the containment. The containment isolation valves are a check valve and a locked-closed manual valve in series outside of containment. In addition to these two containment isolation valves, there are two normally closed, manual valves in series inside containment. There is also a closed piping system outside containment. The entire system is pressurized with water at about 100 psig during unit operation. This water could serve to seal any potential leakage through the valves during an accident and it also continuously demonstrates the integrity of the piping system. The system is supplied by multiple pumps feeding a common header taking suction from a 100,000 gallon storage tank. With these barriers in place, the water pressure would have to fail and containment atmosphere would have to leak past all four valves and then escape the closed system outside containment in order to reach the environment.

(2) Core Spray System, Penetration X-16 A & B

The Core Spray System is a low pressure emergency core cooling system which provides reactor coolant in the event of a loss of coolant accident (LOCA). The system is designed to be pressurized with high pressure water (relative to Pa, peak containment accident pressure) which acts as a water seal for the containment isolation valves during post-accident conditions. The 10-inch injection lines which pass through the subject penetrations are each equipped with a check valve inside containment in addition to two containment isolation valves (remotely-operated gate valves) in series outside containment. This check valve is subject to reactor pressure during normal operation which gives an indication of its leak-tightness. There is also a closed piping system outside containment which would either contain containment leakage or route it to the suppression pool. As with the clean demineralized water system, core spray injection would have to stop and water pressure be lost, containment atmosphere would have to leak past three valves, and then it would have to escape the closed system outside containment to reach the environment.

(3) Standby Liquid Control (SBLC) System, Penetration X-47

The 1½-inch SBLC line which penetrates containment contains a closed valve outside containment in series with two containment isolation check valves (one inside containment and one outside). The closed valve is an explosive

squib valve which consists of a solid metal cap which blocks the pathway unless actuated. The potential of a seat or packing leak, therefore, does not exist. The SBLC system is an Engineered Safety Feature and the squib valves are only actuated in the event that the control rod scram function fails and reactor power cannot be reduced using normal methods. The squib charge is then exploded, shearing off the metal cap and opening the valve. The valve, therefore, would not be actuated during a design basis LOCA. There is also a closed system outside containment. Thus, containment atmosphere would have to leak past two valves, through a solid piece of metal, and out of a closed system in order to reach the environment.

(4) Instrument Air to the Drywell and Torus: Penetration X-216 & X-22

The instrument air system penetrates containment by two lines. The line which penetrates the drywell is a one inch line and that which penetrates the torus is one-half inch. Containment isolation is achieved in each line by one check valve inside containment and one check valve outside containment. The penetrating lines are connected inside of containment to a closed piping system that does not interface with the drywell atmosphere. Outside of containment, the lines are connected to a closed piping system that does not interface with the Reactor Building atmosphere. During normal operation, the lines are pressurized with nitrogen at a pressure of approximately two times Pa. This pressurization may serve as a valve sealing system in the event of a leak.

During the last Integrated Leak Rate Test (ILRT), these lines were properly depressurized and vented outside of containment. The closed piping inside of containment, however, was not vented to the containment; therefore, the containment isolation valves were not adequately challenged. The ILRT was successfully completed which provides assurance that either the inside piping system or the containment isolation valves were not leaking excessively. With these provisions in place during a LOCA, the lines would have to depressurize and containment atmosphere leak past two valves and penetrate two closed systems to reach the environment.

(5) Reactor Building Closed Cooling System (RBCCW): Penetration X-23 & X-24

The RBCCW system consists of two eight-inch lines that penetrate containment. The supply line containment isolation valves are a check valve inside and a remotely-operated manual gate valve outside containment. The return line containment isolation valves are two remotely operated valves, one inside and one outside of containment.

In addition to the two containment isolation valves on each line, other barriers exist. Inside of the containment, the piping forms a closed loop. Outside of containment, the piping is configured such that loop water seals are created. The system is filled with pressurized water during normal operation. The water serves as a seal for potentially leaky valves and as a system leakage detection system. Any through-wall water leaks would be detected either inside or outside of the containment through operational indicators (sump levels, system pressures, tank levels, etc.).

The piping outside of containment is connected to a vented surge tank. This tank receives makeup that is supplied by multiple pumps connected to a common header, which provides suction from a 100,000 gallon storage tank. This configuration provides substantial assurance that the system would remain water-filled in post accident conditions. Containment leakage, then, would have to enter the closed system inside containment, pass through two valves and a loop seal, and the system would have to be depressurized, before leakage could enter the environment.

In addition to the above, the licensee has performed a probabilistic risk assessment (PRA) to calculate the probability of the occurrence, in the next five months, of a loss-of-coolant accident in conjunction with certain additional failures (such as pipe ruptures outside containment) that the licensee states would allow containment leakage to be released to the environment through any of the eight containment pathways. The licensee has calculated this probability to be less than $1 \times 10^{-7}$. The licensee believes that this is sufficiently low so that the plant can be operated for the next five months without significant increase in risk to the public health and safety during that period.

The staff has evaluated the licensee's submittal and concludes that operation until the next refueling outage with the subject valves untested will not add significantly to the risk to public health and safety, and therefore, the proposed Technical Specification change is acceptable.

## 3.0 FINDINGS OF EMERGENCY WARRANTING AN AMENDMENT WITHOUT NOTICE

In the fourth quarter of 1989, the licensee assessed the leak rate testing program at Quad Cities and identified 29 containment pathways that were not being local leak rate tested. The licensee determined the Appendix J, Type C testing was not required for these 29 pathways. However, the NRC staff questioned this determination and on May 18, 1990, concluded that these 29 pathways were, in fact, subject to Appendix J, Type C testing. Once informed of the NRC's position on these pathways, the licensee requested a temporary waiver of compliance. The NRC staff orally granted this waiver on May 18, 1990. By May 19, 1990, the licensee had tested those pathways that could be tested with the plant at power (21 pathways) and submitted a written request for a temporary waiver of compliance and an emergency technical specification amendment for the remaining eight pathways. The licensee requested that the testing of these eight pathways be deferred until the next refueling outage.

The NRC staff finds that the licensee acted as quickly as possible once informed of the requirement to test these pathways. Furthermore, the staff finds that failure to grant the proposed changes in a timely manner would have required a shutdown of Quad Cities, Unit 1. Accordingly, the staff concludes that the licensee has satisfied the requirements of 10 CFR 50.91(a)(5), and that a valid emergency exists.

## 4.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The NRC staff reviewed the licensee's amendment application and determined, in accordance with the criteria of 50.92(c), that operation of Quad Cities, Unit 1, according to the proposed amendment will not:

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

With respect to an increase in the probability of previously evaluated accidents, leakage through the associated valves does not alter or change initiating aspects of the events. With regard to the consequences of an accident previously evaluated, the continued operation in the existing configuration does not present a significant increase in the probability of a larger release of radioactivity than described in the FSAR. For those credible events, the normal as-built designs function to inhibit potential release paths either as a sealed water-filled system or normally manually isolated system. In addition, it has been calculated that for less credible events, the probability of increased risk over an additional five months of continued operation is not significant (less than 1.0E-07). The potential for an adverse change in consequences is not supported for accidents of reasonable probabilities but would require accident scenarios of very low probabilities. Thus, it is concluded that continued operation will not pose a significant increase in risk with regards to accident probabilities or consequences.

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

The proposed amendment does not result in any physical plant changes during the period of interest. Potential leakage of the valves in question would, at worst, affect the severity and not the type of accident.

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

As described in the Technical Specification Bases, dose calculations suggest that the accident leak rate could be allowed to increase to about 2.6% a day before the guideline thyroid dose value given in 10 CFR 100 would be exceeded. However, 1.0% a day provides an adequate margin of safety to assure the health and safety of the general public. Additional margin is achieved by establishing the allowable operational leak rate at 0.75 of the maximum allowable leak rate. As described in the attached documentation and despite the lack of leak testing, substantial barriers to fission product release are provided by the intact system piping and associated valves. These barriers provide mitigating capability such that the potential impact on the margin of safety is insignificant. The Probabilistic Risk Assessment performed demonstrates that the probability of containment function failure coincident with LOCA conditions is also acceptably small.

Accordingly, the Commission has determined that this request does not involve a significant hazards consideration.

## 5.0 STATE CONSULTATION

The State of Illinois was informed by telephone June 7, 1990, of the staff's final no significant hazards consideration determination and intent to issue a license amendment. The State contact had no comment.

## 6.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change to 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 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 previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this 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 this amendment.

## 7.0 CONCLUSION

The staff 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, and (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: J. Pulsipher, L. Olshan

Dated: June 27, 1990