March 8, 1991

Docket Nos. 50-254
and 50-265

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Mr. Thomas J. Kovach
Nuclear Licensing Manager
Commonwealth Edison Company-Suite 300
OPUS West III
1400 OPUS Place
Downers Grove, Illinois 60515

Dear Mr. Kovach:

SUBJECT: ISSUANCE OF AMENDMENT (TAC NOS. 72813 AND 72814)

The Commission has issued the enclosed Amendment No. 130 to Facility Operating License No. DPR-29 and Amendment No. 124 to Facility Operating License No. DPR-30 for the Quad Cities Nuclear Power Station, Units 1 and 2, respectively. The amendments are in response to your application dated May 25, 1989, as supplemented January 25, 1991.

The amendments revise the Technical Specifications associated with the High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) systems to make them more consistent with the Standard Technical Specifications.

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

Sincerely,

Original Signed By:

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

Enclosures:
1. Amendment No. 130 to License No. DPR-29
2. Amendment No. 124 to License No. DPR-30
3. Safety Evaluation

cc w/enclosures:
See next page

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1/1

OFFICIAL RECORD COPY
DOCUMENT NAME: [AMENDMENT 72813/14]

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Surname:	CMoore	LOlshan:ta	RBarrett		JCalvo
Date:	2/2/91	2/25/91	2/25/91	3/4/91	3/7/91

Mr. Thomas J. Kovach
Commonwealth Edison Company

Quad Cities Nuclear Power Station
Unit Nos. 1 and 2

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COMMONWEALTH EDISON COMPANY

AND

IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

DOCKET NO. 50-265

QUAD CITIES NUCLEAR POWER STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 130
License No. DPR-30

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

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated May 25, 1989, as supplemented January 25, 1991, 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-30 is hereby amended to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 130 , 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

J. N. Dyer

For

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 8, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 130

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
3.5/4.5-5	3.5/4.5-5
3.5/4.5-6	3.5/4.5-6
3.5/4.5-7	3.5/4.5-7
3.5/4.5-8	3.5/4.5-8
3.5/4.5-10	3.5/4.5-10
3.5/4.5-23	3.5/4.5-23
3.5/4.5-24	3.5/4.5-24
3.5/4.5-25	3.5/4.5-25
3.5/4.5-26	3.5/4.5-26
	3.5/4.5-27

4. Containment cooling spray loops are required to be operable when the reactor water temperature is greater than 212°F and prior to reactor startup from a cold condition. Continued reactor operation is permitted provided that a maximum of one drywell spray loop may be inoperable for 30 days when the reactor water temperature is greater than 212°F.

5. If the requirements of 3.5.B cannot be met, an orderly shutdown shall be initiated, and the reactor shall be in a cold shutdown condition within 24 hours.

C. HPCI Subsystem

1. The HPCI subsystem shall be operable whenever the reactor pressure is greater than 150 psig and fuel is in the reactor vessel.

2. During startup following a refuel outage or an outage in which work was performed that directly affects HPCI system operability, if the testing requirements of 4.5.C.3 cannot be met, continued reactor startup is not permitted. The HPCI subsystem shall be declared inoperable, and the provisions of Specification 3.5.C.4 shall be implemented.

3. Except for the limitations of 3.5.C.2, if the HPCI subsystem is made or found to be inoperable, continued reactor operation is permissible only during the succeeding 14 days unless such subsystem is sooner made operable, provided that during such 14 days the automatic pressure relief subsystems, the core spray subsystems, LPCI mode of the RHR system, and the RCIC system are operable. Otherwise, the provisions of Specification 3.5.C.4 shall be implemented.

4. During each 5-year period, an air test shall be performed on the drywell spray headers and nozzles and a water spray test performed on the torus spray header and nozzles.

C. HPCI Subsystem

Surveillance of HPCI subsystem shall be performed as specified below with the following limitations. For item 4.5.C.3, the plant is allowed 12 hours in which to successfully complete the test once reactor vessel pressure is adequate to perform each test. In addition, the testing required by item 4.5.C.3.a shall be completed prior to exceeding 325 psig reactor vessel pressure. If HPCI is made inoperable to perform overspeed testing, 24 hours is allowed to complete the tests before exceeding 325 psig.

	Item	Frequency
1.	Valve Position	Every 31 days
2.	Flow Rate Test- HPCI Pump shall deliver at least 5000 gpm against a system head corresponding to a reactor vessel pressure of $\geq$ 1150 psig when steam is being supplied to the turbine at 920 to 1005 psig.	Every 92 days

3. Flow Rate Test-HPCI pump shall deliver at least 5000 gpm against a system head corresponding to a reactor vessel pressure of: — During startup following a refuel outage or an outage in which work was performed that directly affects HPCI system operability.

   a. $\geq$ 300 psig when steam is being supplied to the turbine at 250 to 325 psig, and

   b. $\geq$ 1150 psig when steam is being supplied to the turbine at 920 to 1005 psig.

4. If the requirements of Specification 3.5.C.1, 3.5.C.2 or 3.5.C.3 cannot be met, an orderly shutdown shall be initiated, and the reactor pressure shall be reduced to < 150 psig within 24 hours.

4. Simulated Automatic Actuation Test — Each refueling outage

5. Logic System Functional Test — Each refueling outage.

D. Automatic Pressure Relief Subsystems

1. The automatic pressure relief subsystem shall be operable whenever the reactor pressure is greater than 90 psig, irradiated fuel is in the reactor vessel and prior to reactor startup from a cold condition.

2. From and after the date that two of the five relief valves of the automatic pressure relief subsystem are made or found to be inoperable

D. Automatic Pressure Relief Subsystems

Surveillance of the automatic pressure relief subsystem shall be performed as follows:

1. The following surveillance shall be carried out on a six-month surveillance interval:

   a. With the reactor at pressure each relief valve shall be manually opened. Relief valve opening shall be verified by a compensating turbine bypass valve or control valve closure.

2. A logic system functional test shall be performed each refueling outage.

when the reactor is pressurized above 90 psig with irradiated fuel in the reactor vessel, reactor operation is permissible only during the succeeding 7 days unless repairs are made and provided that during such time the HPCI subsystem is operable.

3. If the requirements of Specification 3.5.D cannot be met, an orderly shutdown shall be initiated and the reactor pressure shall be reduced to 90 psig within 24 hours.

3. A simulated automatic initiation which opens all pilot valves shall be performed each refueling outage.

4. When it is determined that two valves of the automatic pressure relief subsystem are inoperable, the HPCI shall be demonstrated to be operable immediately.

E. Reactor Core Isolation Cooling System

1. The RCIC system will be operable whenever the reactor pressure is greater than 150 psig and fuel is in the reactor vessel.

2. During startup following a refuel outage or an outage in which work was performed that directly affects the RCIC system operability, if the testing requirements of 4.5.E.3 cannot be met, continued reactor startup is not permitted. The RCIC system shall be declared inoperable, and the provisions of Specification 3.5.E.4 shall be implemented.

3. Except for the limitations of 3.5.E.2, if the RCIC system is made or found to be inoperable, continued reactor operation is permissible only during the succeeding 14 days unless such system is sooner made operable, provided that during such 14 days the HPCI system is operable. Otherwise, the provisions of Specification 3.5.E.4 shall be implemented.

E. Reactor Core Isolation Cooling System

Surveillance of the RCIC system shall be performed as specified below with the following limitations. For item 4.5.E.3, the plant is allowed 12 hours in which to successfully complete the test once reactor vessel pressure is adequate to perform each test. In addition, the testing required by item 4.5.E.3.a shall be completed prior to exceeding 325 psig reactor vessel pressure. If RCIC is made inoperable to perform overspeed testing, 24 hours is allowed to complete the tests before exceeding 325 psig.

	Item	Frequency
1.	Valve Position	Every 31 days
2.	Flow Rate Test - RCIC Pump shall deliver at least 400 gpm against a system head corresponding to a reactor vessel pressure of $\geq$ 1150 psig when steam is	Every 92 days

being supplied to the turbine at 920 to 1005 psig.

3. Flow Rate Test-RCIC pump shall deliver at least 400 gpm against a system head corresponding to a reactor vessel pressure of: — During startup following a refuel outage or an outage in which work was performed that directly affects RCIC system operability.

    a. $\geq$ 300 psig when steam is being supplied to the turbine at 250 to 325 psig, and

    b. $\geq$ 1150 psig when steam is being supplied to the turbine at 920 to 1005 psig.

4. If the requirements of Specification 3.5.E.1, 3.5.E.2, or 3.5.E.3 cannot be met, an orderly shutdown shall be initiated and the reactor pressure shall be reduced to < 150 psig within 24 hours.

4. Simulated Automatic Actuation Test — Each refueling outage

5. Logic System Functional Test — Each refueling outage

F. Minimum Core and Containment Cooling System Availability

1. Any combination of inoperable components in the core and containment cooling systems shall not defeat the capability of the remaining operable components to fulfill the core and containment cooling functions.

2. When irradiated fuel is in the reactor vessel and the reactor is in the cold shutdown condition, all low-pressure core and containment cooling systems may be inoperable provided no work is being done which has the potential for draining the reactor vessel.

F. Minimum Core and Containment Cooling System Availability

Surveillance requirements to assure that minimum core and containment cooling systems are available have been specified in Specification 4.2.B.

G. Maintenance of Filled Discharge Pipe

1. Whenever core spray, LPCI mode of the RHR, HPCI, or RCIC are required to be operable, the discharge piping from the pump discharge of these systems to the last check valves shall be filled.

2. The discharge pipe pressure for Core Spray and LPCI mode of RHR shall be maintained at greater than 40 psig and less than 90 psig. If pressure in any of these systems is less than 40 psig or greater than 90 psig, this condition shall be alarmed in the control room and immediate corrective action taken. If the discharge pipe pressure is not within these limits in 12 hours after the occurrence, an orderly shutdown shall be initiated, and the reactor shall be in a cold shutdown condition within 24 hours after initiation.

3. Filled discharge piping for HPCI and RCIC systems is ensured by maintaining the level in the Contaminated Condensate Storage Tanks (CCST's) at or above 9.5 feet. If the CCST level falls below 9.5 feet, restore the level within 12 hours or line up both HPCI and RCIC to take a suction from the torus per 4.5.G.3.

G. Maintenance of Filled Discharge Pipe

The following surveillance requirements shall be adhered to to assure that the discharge piping of the core spray, LPCI mode of the RHR, HPCI, and RCIC are filled:

1. Every month the LPCI mode of the RHR, core spray ECCS, HPCI and RCIC discharge piping shall be vented from the high point and water flow observed.

2. Following any period where HPCI, RCIC, LPCI mode of the RHR or core spray have been out of service and drained for maintenance, the discharge piping of the inoperable system shall be vented from the high point prior to the return of the system to service.

3. Whenever the HPCI or RCIC system is lined up to take suction from the torus, the discharge piping of the HPCI and RCIC shall be vented from the high point of the system and water flow observed every 24 hours.

4.5 SURVEILLANCE REQUIREMENTS BASES

The testing interval for the core and containment cooling systems is based on a quantitative reliability analysis, judgment, and practicality. The core cooling systems have not been designed to be fully testable during operation. For example, the core spray final admission valves do not open until reactor pressure has fallen to 350 psig. Thus, during operation, even if high drywell pressure were simulated, the final valves would not open. In the case of the HPCI, automatic initiation during power operation would result in pumping cold water into the reactor vessel which is not desirable.

The surveillance requirements bases described in this paragraph apply to all core and containment cooling systems except HPCI and RCIC. The systems can be automatically actuated during a refueling outage and this will be done. To increase the availability of the individual components of the core and containment cooling systems, the components which make up the system, i.e., instrumentation, pumps, valve operators, etc., are tested more frequently. The instrumentation is functionally tested each month. Likewise the pumps and motor-operated valves are also tested each month to assure their operability. The combination of a yearly simulated automatic actuation test and monthly tests of the pumps and valve operators is deemed to be adequate testing of these systems. With components or subsystems out of service, overall core and containment cooling reliability is maintained by demonstrating the operability of the remaining cooling equipment. The degree of operability to be demonstrated depends on the nature of the reason for the out-of-service equipment. For routine out-of-service periods caused by preventative maintenance, etc., the pump and valve operability checks will be performed to demonstrate operability of the remaining components. However, if a failure, design deficiency, etc., causes the out-of-service period, then the demonstration of operability should be thorough enough to assure that a similar problem does not exist on the remaining components. For example, if an out-of-service period is caused by failure of a pump to deliver rated capacity due to a design deficiency, the other pumps of this type might be subjected to a flow rate test in addition to the operability checks.

The surveillance requirements bases described in this paragraph apply only to the RCIC and HPCI systems. With a cooling system out of service, overall core and containment cooling reliability is maintained by verifying the operability of the remaining cooling systems. The verification of operability, as used in this context, for the remaining cooling systems means to administratively check by examining logs or other information to verify that the remaining systems are not out-of-service for maintenance or other reasons. It does not mean to perform the surveillance requirements needed to demonstrate the operability of the remaining systems. However, if a failure, design deficiency, etc., causes the out-of-service period, then the verification of operability should be thorough enough to assure that a similar problem does not exist on the remaining systems. For example, if an out-of-service period is caused by failure of a pump to deliver rated capacity due to a design deficiency, the other pumps of this type might be subjected to a flow rate test. Following a refueling outage or an outage in which work was performed that directly affects system operability, the HPCI and RCIC pumps are flow rate tested prior to exceeding 325 psig and again at rated reactor steam pressure. This combination of testing provides adequate assurance of pump performance throughout the range of reactor pressures at which it is required to operate. The low pressure limit is selected to allow testing at a point of stable plant operation and also to provide overlap with low pressure ECC systems. A time limit is provided in which to perform the required tests during start-up. This time limit is considered adequate to allow stable plant conditions to be achieved and the required tests to be performed. Flow rate testing of the HPCI and RCIC pumps is also conducted every 92 days at rated reactor pressure to demonstrate system operability in accordance with the LCO provisions and to meet inservice testing requirements for the HPCI system. Applicable valves are tested in accordance with the provisions of the inservice testing program. In addition, monthly checks are made on the position of each manual, power operated or automatic valve installed in the direct flowpath of the suction or discharge of the pump or turbine that is not locked, sealed, or otherwise secured in position. At each refueling outage, a logic system functional test and a simulated automatic actuation test is performed on the HPCI and RCIC systems. The tests and checks described above are considered adequate to assure system operability.

The verification of the main steam relief valve operability during manual actuation surveillance testing must be made independent of temperatures indicated by thermocouples downstream of the relief valves. It has been found that a temperature increase may result with the valve still closed. This is due to steam being vented through the pilot valves during the surveillance test. By first opening a turbine bypass valve, and then observing its closure response during relief valve actuation, positive verification can be made for the relief valve opening and passing steam flow. Closure response of the turbine control valves during relief valve manual actuation would likewise serve as an adequate verification for the relief valve opening. This test method may be performed over a wide range of reactor pressures greater than 150 psig. Valve operation below 150 psig is limited by the spring tension exhibited by the relief valves.

The surveillance requirements to ensure that the discharge piping of the core spray, LPCI mode of the RHR, HPCI, and RCIC systems is filled provides for a visual observation that water flows from a high point vent. This ensures that the line is in a full condition.

Instrumentation has been provided on core spray and LPCI mode of RHR to monitor the pressure of water in the discharge piping between the monthly intervals at which the lines are vented and alarm the control room if the pressure is inadequate. This instrumentation will be calibrated on the same frequency as the safety system instrumentation and the alarm system tested monthly. This testing ensures that, during the interval between the monthly venting checks, the status of the discharge piping is monitored on a continuous basis. An alarm point of 40 psig for the low pressure of the fill system has been chosen because, due to elevations of piping within the plant, 39 psig is required to keep the lines full. The shutoff head of the fill system pumps is less than 90 psig and therefore will not defeat the low-pressure cooling pump discharge pressure interlock of 100 psig as shown in Table 3.2-2. A margin of 10 psig is provided by the high pressure alarm point of 90 psig.

HPCI and RCIC systems normally take a suction from the Contaminated Condensate Storage Tanks (CCSTs). The level in the CCST's is maintained at or above

9.5 feet. This level corresponds to an elevation which is greater than the elevation of the last check valves in the discharge pipes of either the HPCI or RCIC systems. Therefore, filled discharge piping of HPCI or RCIC systems is ensured when lined up to the CCST and tank level is at or above 9.5 feet.

The watertight bulkhead and submarine doors and the penetration seals for pipes and cables penetrating the vault walls and ceilings have been designed to withstand the maximum flood conditions. To assure that their installation is adequate for maximum flood conditions, a method of testing each seal has been devised.

In order to test an electrical penetration or pipe seal, compressed air is supplied to a test connection and the space between the fittings is pressurized to approximately 15 psig. The outer faces are then tested for leaks using a soap bubble solution.

In order to test the submarine doors, a test frame must be installed around each door. The frame is then pumped to a pressure of approximately 15 psig and held to test for leaktightness. The watertight bulkhead doors are tested by pressurizing the volume between the double-gasket seals to approximately 15 psig. The gasket seal area is inspected using a soap bubble solution. Each RHR service water vault contains a sump, which will collect any floor or equipment leakage inside the vault. A sump pump will automatically start on high level in the sump, and will pump the water out of the vault, via 2 discharge check valves outside the vault to the service water discharge pipe. A composite sampler is located on the sump discharge line. A radiation monitor is also located on the service water discharge. The sump discharge water is not expected to be contaminated, and any in-leakage to the vault is prevented by 2 check valves. Surveillance of these check valves is performed each operating cycle to assure their integrity. The previously installed bedplate drains to the turbine building equipment drain sump have been capped off permanently.

A level switch set at a water level of 6 inches is located inside each vault. Upon actuation, the switch alarms in the control room to notify the operator of trouble in the vault. The operator will also be aware of problems in the vaults/condensate pump room if the high-level alarm on the equipment drain sump is not terminated in a reasonable amount of time.

A system of level switches has been installed in the condenser pit to indicate and control flooding of the condenser area. The following switches are installed:

	Level	Function
a.	1 foot (one switch)	alarm, low water level
b.	3 feet (one switch)	alarm, high water level
c.	5 feet (two redundant switch pairs)	alarm and circulating water pump trip

Level (a) indicates water in the condenser pit from either the hotwell or the circulating water system. Level (b) is above the hotwell capacity and indicates a probable circulating water failure.

Should the switches at levels (a) and (b) fail or the operator fails to trip the circulating water pumps on alarm at level (b), the actuation of either level switch pair at level (c) shall trip the circulating water pumps automatically and alarm in the control room. These redundant level switch pairs at level (c) are designed and installed to IEEE-279, "Criteria for Nuclear Power Plant Protection Systems." As the circulating water pumps are tripped, either manually or automatically at level (c) of 5 feet, the maximum water level reached in the condenser pit due to pumping will be at elevation 568 feet 6 inches elevation (10 feet above condenser pit floor elevation 558 feet 6 inches; 5 feet plus an additional 5 feet attributed to pump coastdown).

In order to prevent the RHR service water pump motors and diesel generator cooling water pump motors from overheating a vault cooler is supplied for each pump. Each vault cooler is designed to maintain the vault at a maximum 105°F temperature during operation of its respective pump. For example, if diesel generator cooling water pump 1/2-3903 starts, its cooler also starts and maintains the vault at 105°F by removing heat supplied to the vault by the motor of pump 1/2-3903. If, at the same time that pump 1/2-3903 is in operation, RHR service water pump 1C starts, its cooler will also start and compensate for the added heat supplied to the vault by the 1C pump motor keeping the vault at 105°F.

Each of the coolers is supplied with cooling water from its respective pump's discharge line. After the water has been passed through the cooler it returns to its respective pump's suction line. The cooling water quantity needed for each cooler is approximately 1% to 5% of the design flow of the pumps so that the recirculation of this small amount of heated water will not affect pump or cooler operation.

Operation of the fans and coolers is required during shutdown and thus additional surveillance is not required.

Verification that access doors to each vault are closed following entrance by personnel is covered by station operating procedures.

The LHGR shall be checked daily to determine if fuel burnup or control rod movement has caused changes in power distribution. Since changes due to burnup are slow and only a few control rods are moved daily, a daily check of power distribution is adequate.

Average Planar LHGR

At core thermal power levels less than or equal to 25%, operating plant experience and thermal hydraulic analyses indicate that the resulting average planar LHGR is below the maximum average planar LHGR by a considerable margin; therefore, evaluation of the average planar LHGR below this power level is not necessary. The daily requirement for calculating average planar LHGR above 25% rated thermal power is sufficient, since power distribution shifts are slow when there have not been significant power or control rod changes.

Local LHGR

The LHGR as a function of core height shall be checked daily during reactor operation at greater than or equal to 25% power to determine if fuel burnup or control rod movement has caused changes in power distribution. A limiting LHGR value is precluded by considerable margin when employing any permissible control rod pattern below 25% rated thermal power.

Minimum Critical Power Ratio (MCPR)

At core thermal power levels less than or equal to 25%, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience and thermal hydraulic analysis indicate that the resulting MCPR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR.

The daily requirement for calculating MCPR above 25% rated thermal power is sufficient, since power distribution shifts are very slow when there have not been significant power or control rod changes. In addition, the K_f correction, as specified in the CORE OPERATING LIMITS REPORT, applied to the LCO provides margin for flow increases from low flows.

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. 124
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 25, 1989, as supplemented January 25, 1991, 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:

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 124, 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

J. L. Dyer
for
Richard J. Barrett, Director
Project Directorate III-2
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 8, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 124

FACILITY OPERATING LICENSE NO. DPR-30

DOCKET NO. 50-265

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
3.5/4.5-4a	3.5/4.5-4a
3.5/4.5-5	3.5/4.5-5
3.5/4.5-6	3.5/4.5-6
3.5/4.5-6a	3.5/4.5-6a
3.5/4.5-7	3.5/4.5-7
3.5/4.5-15	3.5/4.5-15
3.5/4.5-15a	3.5/4.5-15a

C. HPCI Subsystem

1. The HPCI subsystem shall be operable whenever the reactor pressure is greater than 150 psig and fuel is in the reactor vessel.

2. During startup following a refuel outage or an outage in which work was performed that directly affects HPCI system operability, if the testing requirements of 4.5.C.3 cannot be met, continued reactor startup is not permitted. The HPCI subsystem shall be declared inoperable, and the provisions of Specification 3.5.C.4 shall be implemented.

3. Except for the limitations of 3.5.C.2, if the HPCI subsystem is made or found to be inoperable, continued reactor operation is permissible only during the succeeding 14 days unless such subsystem is sooner made operable, provided that during such 14 days the automatic pressure relief subsystem, the core spray subsystems, LPCI mode of the RHR system, and the RCIC system are operable. Otherwise, the provisions of Specification 3.5.C.4 shall be implemented.

4. If the requirements of Specification 3.5.C.1, 3.5.C.2 or 3.5.C.3 cannot be met, an orderly shutdown shall be initiated, and the reactor pressure shall be reduced to <150 psig within 24 hours.

C. HPCI Subsystem

Surveillance of the HPCI subsystem shall be performed as specified below with the following limitations. For item 4.5.C.3, the plant is allowed 12 hours in which to successfully complete the test once reactor pressure is adequate to perform each test. In addition, the testing required by item 4.5.C.3.a shall be completed prior to exceeding 325 psig reactor vessel pressure. If HPCI is made inoperable to perform overspeed testing, 24 hours is allowed to complete the tests before exceeding 325 psig.

	Item	Frequency
1.	Valve Position	Every 31 days
2.	Flow Rate Test - HPCI pump shall deliver at least 5000 gpm against a system head corresponding to a reactor vessel pressure of $\geq$1150 psig when steam is being supplied to the turbine at 920 to 1005 psig.	Every 92 days
3.	Flow Rate Test - HPCI pump shall deliver at least 5000 gpm against a system head corresponding to a reactor vessel pressure of: a. $\geq$ 300 psig when steam is being supplied to the turbine at 250 to 325 psig, and b. $\geq$ 1150 psig when steam is being supplied to the turbine at 920 to 1005 psig.	During startup following a refuel outage or an outage in which work was performed that directly affects HPCI system operability.
4.	Simulated Automatic Actuation Test	Each refueling outage
5.	Logic System Functional Test	Each refueling outage

Amendment No. 124

D. Automatic Pressure Relief Subsystems

1. The automatic pressure relief subsystem shall be operable whenever the reactor pressure is greater than 90 psig, irradiated fuel is in the reactor vessel and prior to reactor startup from a cold condition.

2. From and after the date that two of the five relief valves of the automatic pressure relief subsystem is made or found to be inoperable when the reactor is pressurized above 90 psig with irradiated fuel in the reactor vessel, reactor operation is permissible only during the succeeding 7 days unless repairs are made and provided that during such time the HPCI subsystem is operable.

3. If the requirements of Specification 3.5.D cannot be met, an orderly shutdown shall be initiated and the reactor pressure shall be reduced to 90 psig within 24 hours.

D. Automatic Pressure Relief Subsystems

Surveillance of the automatic pressure relief subsystem shall be performed as follows:

1. The following surveillance shall be carried out on a six-month surveillance interval:

   a. With the reactor at pressure each relief valve shall be manually opened. Relief valve opening shall be verified by a compensating turbine bypass valve or control valve closure.

2. A logic system functional test shall be performed each refueling outage.

3. A simulated automatic initiation which opens all pilot valves shall be performed each refueling outage.

4. When it is determined that two relief valves of the automatic pressure relief subsystem are inoperable, the HPCI shall be demonstrated to be operable immediately.

E. Reactor Core Isolation Cooling System

1. The RCIC system will be operable whenever the reactor pressure is greater than 150 psig and fuel is in the reactor vessel.

2. During startup following a refuel outage or an outage in which work was performed that directly affects RCIC system operability, if the testing requirements of 4.5.E.3 cannot be met, continued reactor startup is not permitted. The RCIC system shall be declared inoperable, and the provisions of Specification 3.5.E.4 shall be implemented.

3. Except for the limitations of 3.5.E.2, if the RCIC system is made or found to be inoperable, continued reactor operation is permitted only during the succeeding 14 days unless such system is sooner made operable, provided that during such 14 days the HPCI system is operable. Otherwise, the provisions of Specification 3.5.E.4 shall be implemented.

4. If the requirements of Specification 3.5.E.1, 3.5.E.2 or 3.5.E.3 cannot be met, an orderly shutdown shall be initiated and the reactor pressure shall be reduced to <150 psig within 24 hours.

E. Reactor Core Isolation Cooling System

Surveillance of the RCIC system shall be performed as specified below with the following limitations. For item 4.5.E.3, the plant is allowed 12 hours in which to successfully complete the test once reactor vessel pressure is adequate to perform each test. In addition, the testing required by item 4.5.E.3.a shall be completed prior to exceeding 325 psig reactor vessel pressure. If RCIC is made inoperable to perform overspeed testing, 24 hours is allowed to complete the tests before exceeding 325 psig.

	<u>Item</u>	<u>Frequency</u>
1.	Valve Position	Every 31 days
2.	Flow Rate Test - RCIC pump shall deliver at least 400 gpm against a system head corresponding to a reactor vessel pressure of $\geq$1150 psig when steam is being supplied to the turbine at 920 to 1005 psig.	Every 92 days
3.	Flow Rate Test - RCIC pump shall deliver at least 400 gpm against a system head corresponding to a reactor vessel pressure of: a. $\geq$ 300 psig when steam is being supplied to the turbine at 250 to 325 psig, and b. $\geq$ 1150 psig when steam is being supplied to the turbine at 920 to 1005 psig.	During startup following a refuel outage or an outage in which work was performed that directly affects RCIC system operability.
4.	Simulated Automatic Actuation Test	Each refueling outage
5.	Logic System Functional Test	Each refueling outage

Amendment No. 124

F. Minimum Core and Containment Cooling System Availability

1. Any combination of inoperable components in the core and containment cooling systems shall not defeat the capability of the remaining operable components to fulfill the core and containment cooling functions.

2. When irradiated fuel is in the reactor vessel and the reactor is in the cold shutdown condition, all low-pressure core and containment cooling systems may be inoperable provided no work

F. Minimum Core and Containment Cooling System Availability

Surveillance requirements to assure that minimum core and containment cooling systems are available have been specified in Specification 4.2.B.

is being done which has the potential for draining the reactor vessel.

3. When irradiated fuel is in the reactor and the vessel head is removed, the suppression chamber may be drained completely and no more than one control rod drive housing opened at any one time provided that the spent fuel pool gate is open and the fuel pool water level is maintained at a level of greater than 33 feet above the bottom of the pool. Additionally, a minimum condensate storage reserve of 230,000 gallons shall be maintained, no work shall be performed in the reactor vessel while a control rod drive housing is blanked following removal of the control rod drive, and a special flange shall be available which can be used to blank an open housing in the event of a leak.

4. When irradiated fuel is in the reactor and the vessel head is removed, work that has the potential for draining the vessel may be carried on with less than 112,200 ft^3 of water in the suppression pool, provided that: (1) the total volume of water in the suppression pool, refueling cavity, and the fuel storage pool above the bottom of the fuel pool gate is greater than 112,200 ft^3; (2) the fuel storage pool gate is removed; (3) the low-pressure core and containment cooling systems are operable; and (4) the automatic mode of the drywell sump pumps is disabled.

G. Maintenance of Filled Discharge Pipe

1. Whenever core spray, LPCI mode of the RHR, HPCI, or RCIC are required to be operable, the discharge piping from the pump discharge of these systems to the last check valves shall be filled.

G. Maintenance of Filled Discharge Pipe

The following surveillance requirements shall be adhered to to assure that the discharge piping of the core spray, LPCI mode of the RHR, HPCI, and RCIC are filled:

1. Every month the LPCI mode of the RHR, Core Spray ECCS, HPCI, and RCIC discharge piping shall be vented from the high point and water flow observed.

4.5 SURVEILLANCE REQUIREMENTS BASES

The testing interval for the core and containment cooling systems is based on a quantitative reliability analysis, judgment, and practicality. The core cooling systems have not been designed to be fully testable during operation. For example, the core spray final admission valves do not open until reactor pressure has fallen to 350 psig. Thus, during operation, even if high drywell pressure were simulated, the final valves would not open. In the case of the HPCI, automatic initiation during power operation would result in pumping cold water into the reactor vessel which is not desirable.

The surveillance requirements bases in this paragraph apply to all core and containment cooling systems except RCIC and HPCI. The systems can be automatically actuated during a refueling outage and this will be done. To increase the availability of the individual components of the core and containment cooling systems, the components which make up the system, i.e., instrumentation, pumps, valve operators, etc., are tested more frequently. The instrumentation is functionally tested each month. Likewise the pumps and motor-operated valves are also tested each month to assure their operability. The combination of a yearly simulated automatic actuation test and monthly tests of the pumps and valve operators is deemed to be adequate testing of these systems. With components or subsystems out of service, overall core and containment cooling reliability is maintained by demonstrating the operability of the remaining cooling equipment. The degree of operability to be demonstrated depends on the nature of the reason for the out-of-service equipment. For routine out-of-service periods caused by preventative maintenance, etc., the pump and valve operability checks will be performed to demonstrate operability of the remaining components. However, if a failure, design deficiency, etc., causes the out-of-service period, then the demonstration of operability should be thorough enough to assure that a similar problem does not exist on the remaining components. For example, if an out-of-service period caused by failure of a pump to deliver rated capacity due to a design deficiency, the other pumps of this type might be subjected to a flow rate test in addition to the operability checks.

The surveillance requirements bases described in this paragraph apply only to the RCIC and HPCI systems. With a cooling system out of service, overall core and containment cooling reliability is maintained by verifying the operability of the remaining cooling systems. The verification of operability, as used in this context, for the remaining cooling systems means to administratively check by examining logs or other information to verify that the remaining systems are not out-of-service for maintenance or other reasons. It does not mean to perform the surveillance requirements needed to demonstrate the operability of the remaining systems. However, if a failure, design deficiency, etc., causes the out-of-service period, then the verification of operability should be thorough enough to assure that a similar problem does not exist on the remaining systems. For example, if an out-of-service period is caused by failure of a pump to deliver rated capacity due to a design deficiency, the other pumps of this type might be subjected to a flow rate test. Following a refueling outage or an outage in which work was performed that directly affects system operability, the HPCI and RCIC pumps are flow rate tested prior to exceeding 325 psig and again at rated reactor steam pressure. This combination of testing provides adequate assurance of pump performance throughout the range of reactor pressures at which it is required to operate. The low pressure limit is selected to allow testing at a point of stable plant operation and also to provide overlap with low pressure ECC systems. A time limit is provided in which to perform the required tests during startup. This time limit is considered adequate to allow stable plant conditions to be achieved and the required tests to be performed. Flow rate testing of the HPCI and RCIC pumps is also conducted every 92 days at rated reactor pressure to demonstrate system operability in accordance with the LCO provisions and to meet inservice testing requirements for the HPCI system. Applicable valves are tested in accordance with the provisions of the inservice testing program. In addition, monthly checks are made on the position of each manual, power operated or automatic valve installed in the direct flowpath of the suction or discharge of the pump or turbine that is not locked, sealed, or otherwise secured in position. At each refueling outage, a logic system functional test and a simulated automatic actuation test is performed on the HPCI and RCIC systems. The tests and checks described above are considered adequate to assure system operability.

The verification of the main steam relief valve operability during manual actuation surveillance testing must be made independent of temperatures indicated by thermocouples downstream of the relief valves. It has been found that a temperature increase may result with the valve still closed. This is due to steam being vented through the pilot valves during the surveillance test. By first opening a turbine bypass valve, and then observing its closure response during relief valve actuation, positive verification can be made for the relief valve opening and passing steam flow. Closure response of the turbine control valves during relief valve manual actuation would likewise serve as an adequate verification for the relief valve opening. This test method may be performed over a wide range of reactor pressures greater than 150 psig. Valve operation below 150 psig is limited by the spring tension exhibited by the relief valves.

The surveillance requirements to ensure that the discharge piping of the core spray, LPCI mode of the RHR, HPCI, and RCIC systems is filled provides for a visual observation that water flows from a high point vent. This ensures that the line is in a full condition.

Instrumentation has been provided on core spray and LPCI mode of RHR to monitor the pressure of water in the discharge piping between the monthly intervals at which the lines are vented and alarm the control room if the pressure is inadequate. This instrumentation will be calibrated on the same frequency as the safety system instrumentation and the alarm system tested monthly. This testing ensures that, during the interval between the monthly venting checks, the status of the discharge piping is monitored on a continuous basis. An alarm point of $\geq$40 psig for the low pressure of the fill system has been chosen because, due to elavations of piping within the plant, 39 psig is required to keep the lines full. The shutoff head of the fill system pumps is less than 90 psig and therefore will not defeat the low-pressure cooling pump discharge press interlock 100 psig as shown in Table 3.2-2. A margin of 10 psig is provided by the high pressure alarm point of 90 psig.

HPCI and RCIC systems normally take a suction from the Contaminated Condensate Storage Tanks (CCST's). The level in the CCST's is maintained at or above 9.5 feet. This level corresponds to an elevation which is greater than the elevation of the last check valves in the discharge pipes of either the HPCI or RCIC systems. Therefore, filled discharge piping of HPCI or RCIC systems is ensured when lined up to the CCST and tank level is at or above 9.5 feet.

The watertight bulkhead and submarine doors and the penetration seals for pipes and cables penetrating the vault walls and ceilings have been designed to withstand the maximum flood conditions. To assure that their installation is adequate for maximum flood conditions, a method of testing each seal has been devised.

In order to test an electrical penetration or pipe seal, compressed air is supplied to a test connection and the space between the fittings is pressurized to approximately 15 psig. The outer faces are then tested for leaks using a soap bubble solution.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 130 TO FACILITY OPERATING LICENSE NO. DPR-29

AND AMENDMENT NO. 124 TO FACILITY OPERATING LICENSE NO. DPR-30

COMMONWEALTH EDISON COMPANY

AND

IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2

DOCKET NOS. 50-254 AND 50-265

1.0 INTRODUCTION

By letter dated May 25, 1989, as supplemented January 25, 1991, Commonwealth Edison Company (CECo) proposed to amend Appendix A of the Quad Cities Facility Operating Licenses, DPR-29 and DPR-30. These amendments revise certain Limiting Conditions for Operation (LCO) and surveillance requirements associated with the High Pressure Core Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) systems. The changes are consistent with similar Technical Specifications (TSs) approved for more recently licensed BWRs and the BWR Standard Technical Specifications (STS). The January 25, 1991 letter provided an additional surveillance requirement that did not significantly alter the proposed action or change the initial proposed no significant hazards consideration determination published in the Federal Register on August 9, 1989.

2.0 EVALUATION

The current Quad Cities, Units 1 and 2, TSs require that other Emergency Core Cooling System (ECCS) subsystems be demonstrated to be operable whenever HPCI or RCIC are inoperable. The main purpose of the proposed amendments is to remove the requirement to demonstrate operability while still maintaining adequate assurance of system operability.

The requirement for demonstrating operability of other ECCS subsystems was originally needed because there was a lack of plant operating history and equipment failure data. However, plant operating history now shows that testing of other ECCS subsystems when one subsystem is inoperable is not necessary to provide adequate assurance of system operability. In fact, taking the other subsystem out of service for testing creates the risk of the second system also failing; in some instances, it has been observed that subsystem failures are related to the test itself and not an indication that the subsystem would have failed should it have been needed to actually mitigate an accident. Operability of these subsystems can be shown by checking records to verify that valve lineup, electrical lineups and instrumentation requirements have not been changed since the last time the subsystem was verified to be operable.

The current BWR STS and the TSs approved for more recently licensed BWRs accept the philosophy that testing subsystems to demonstrate operability is not required when another subsystem is inoperable (except for diesel generator testing). Instead, operability is based on satisfactory performance of monthly, quarterly, refueling interval, post-maintenance or other specified performance tests.

Therefore, based on the risk of the other subsystem failing, past operational experience, and the similarity to the BWR STS and other BWR TSs, we conclude that it is acceptable to eliminate the requirements to test other ECCS subsystems when either HPCI or RCIC is inoperable. (A similar change was approved for HPCI for Dresden on August 10, 1989. Dresden does not have a RCIC system.)

The proposed amendments also increase the required reactor pressure for HPCI operability from 90 psig to 150 psig. Since the HPCI system is designed to pump 5000 gpm into the reactor vessel within a reactor pressure range of about 1150 psig to 150 psig, the operability of the HPCI system cannot be tested at 90 psig (at pressures below 150 psig). Since the HPCI system is isolated below a steam line pressure of 100 psig, the present LCO requirement of 90 psig for operability is impractical. Therefore, because this change corrects inconsistencies in the current TSs and does not decrease safety, we find the increase in HPCI operability from 90 psig to 150 psig to be acceptable. (The same change was approved for Dresden on August 10, 1989.)

The amendments also increase the allowable outage time for HPCI and RCIC from 7 days to 14 days. This is consistent with the BWR STS and more recently licensed BWRs and reflects the availability of low and high pressure core cooling systems for mitigating an accident.

The amendments delete the requirement for HPCI and RCIC to be operable prior to reactor startup. These systems cannot be considered operable until reactor pressure is adequate for system operation. Thus, we find this change acceptable. (This is consistent with the BWR STS, more recently licensed BWRs, and the Dresden TSs for HPCI.)

In reviewing the licensee's May 25, 1989 amendment request, we compared the proposal to the BWR STS, NUREG-0123, Revision 3. It is our policy that when a licensee wishes to adopt provisions of the STS for a particular subsystem or TS section (in this case, HPCI and RCIC) then the licensee must adopt all provisions of the STS for that subsystem or section unless there is technical justification for not doing so. We noted that the licensee's May 25, 1989 submittal had not included the STS requirement to vent from the high point every month to verify that the system piping is filled with water. We informed the licensee of this omission and, by letter dated January 25, 1991, the licensee proposed the appropriate TS changes to include this provision.

Therefore, we find the amendment request submitted on May 25, 1989, as supplemented January 25, 1991, to be acceptable.

3.0 STATE CONSULTATION

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

4.0 ENVIRONMENTAL CONSIDERATION

This 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 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 nor environmental assessment need be prepared in connection with the issuance of this amendment.

5.0 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, (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 nor to the health and safety of the public.

Principal Contributor: L. Olshan

Date: March 8, 1991