March 8, 1994

Docket Nos. 50-254
and 50-265

DISTRIBUTION:

Docket File	NRC & Local PDRs
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ACRS (10)	OPA
OC/LFDCB	PDIII-2 r/f
T. Dunning	T. Bergman

Mr. D. L. Farrar
Nuclear Regulatory Services
Commonwealth Edison Company
Executive Towers West III, Suite 500
1400 OPUS Place
Downers Grove, Illinois 60515

Dear Mr. Farrar:

SUBJECT: ISSUANCE OF AMENDMENTS (TAC NOS. M88047 AND M88048)

The Commission has issued the enclosed Amendment No. 144 to Facility Operating License No. DPR-29 and Amendment No. 140 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 October 21, 1993.

The amendments delete the requirements for demonstrating the operability of redundant equipment when emergency core cooling system equipment is found to be inoperable, or made inoperable for maintenance. The changes are consistent with the guidance provided by the NRC staff in Generic Letter 93-05, dated September 27, 1993.

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,

Original Signed By:

Chandu P. Patel, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 144 to DPR-29
2. Amendment No. 140 to DPR-30
3. Safety Evaluation

cc w/enclosures:
See next page

OFC	LA:PDIII-2	PM:PDIII-2	D:PDIII-2	OGC		
NAME	TCLARK	CPATEL	JDYER			
DATE	2/28/94	2/28/94	2/28/94	3/7/94	/ /94	/ /94
COPY	YES/NO	YES/NO	YES/NO	YES/NO	YES/NO	YES/NO

DFol

March 8, 1994

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Chandu P. Patel, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

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See next page

OFC	LA:PDIII-2	PM:PDIII-2	D:PDIII-2	OGC		
NAME	TCLARK	CPATEL	JDYER			
DATE	2/28/94	2/28/94	2/28/94	3/7/94	/ /94	/ /94
COPY	YES/NO	YES/NO	YES/NO	YES/NO	YES/NO	YES/NO

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

March 8, 1994

Docket Nos. 50-254
and 50-265

Mr. D. L. Farrar, Manager
Nuclear Regulatory Services
Commonwealth Edison Company
Executive Towers West III, Suite 500
1400 OPUS Place
Downers Grove, Illinois 60515

Dear Mr. Farrar:

SUBJECT: ISSUANCE OF AMENDMENTS (TAC NOS. M88047 AND M88048)

The Commission has issued the enclosed Amendment No. 144 to Facility Operating License No. DPR-29 and Amendment No. 140 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 October 21, 1993.

The amendments delete the requirements for demonstrating the operability of redundant equipment when emergency core cooling system equipment is found to be inoperable, or made inoperable for maintenance. The changes are consistent with the guidance provided by the NRC staff in Generic Letter 93-05, dated September 27, 1993.

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,

Chandu P. Patel

Chandu P. Patel, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 144 to DPR-29
2. Amendment No. 140 to DPR-30
3. Safety Evaluation

cc w/enclosures:
See next page

Mr. D. L. Farrar
Commonwealth Edison Company

Quad Cities Nuclear Power Station
Unit Nos. 1 and 2

cc:

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

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. 144
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 October 21, 1993, 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. <u>Technical Specifications</u>

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 144 , 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

James E. Dyer

James E. Dyer, 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, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 144

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
3.5/4.5-2	3.5/4.5-2
3.5/4.5-3	3.5/4.5-3
3.5/4.5-4	3.5/4.5-4
3.5/4.5-5	3.5/4.5-5
3.5/4.5-15	3.5/4.5-15
3.5/4.5-16	3.5/4.5-16
3.5/4.5-17	3.5/4.5-17
3.5/4.5-23	3.5/4.5-23

e. Core spray header Δp instrumentation

check	Once/day
calibrate	Once/3 months
test	Once/3 months

f. Logic system functional test — Once/Each refueling outage

2. From and after the date that one of the core spray subsystems is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding 7 days unless such subsystem is sooner made operable, provided that during such 7 days all active components of the other core spray subsystem and the LPCI mode of the RHR system and the diesel generators required for operation of such components if no external source of power were available shall be operable.

3. The LPCI mode of the RHR system shall be operable whenever irradiated fuel is in the reactor vessel and prior to reactor startup from a cold condition.

4. From and after the date that one of the RHR pumps is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding 30 days unless such pump is sooner made operable, provided that during such 30 days the remaining active components of the LPCI mode of the RHR, containment cooling

2. LPCI mode of the RHR system testing shall be as specified in Specifications 4.5.A.1.a, b, c, d, and f, except that each LPCI division (two RHR pumps per division) shall deliver at least 9000 gpm against a system head corresponding to a reactor vessel pressure of 20 psig, with a minimum flow valve open.

mode of the RHR, all active components of both core spray subsystems, and the diesel generators required for operation of such components if no external source of power were available shall be operable.

5. From and after the date that the LPCI mode of the RHR system is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding 7 days unless it is sooner made operable, provided that during such 7 days all active components of both core spray subsystems, the containment cooling mode of the RHR (including two RHR pumps), and the diesel generators required for operation of such components if no external source of power were available shall be operable.

6. If the requirements of Specification 3.5.A cannot be met, an orderly shutdown of the reactor shall be initiated, and the reactor shall be in the cold shutdown condition within 24 hours.

B. Containment Cooling Mode of the RHR System

1. a. Both loops of the containment cooling mode of the RHR system, as defined in the bases for Specification 3.5.B, shall be operable whenever irradiated fuel is in the reactor vessel and prior to reactor startup from a cold condition.

B. Containment Cooling Mode of the RHR System

Surveillance of the containment cooling mode of the RHR system shall be performed as follows:

1. RHR service water subsystem testing:

	Item	Frequency
a.	Pump and valve operability	Once/3 months

1. b. From the effective date of this amendment until November 1, 1989, the "B" loop of the containment cooling mode of the RHR system for each reactor may share the Unit 1 "C" and "D" RHR service water pumps using cross tie line 1/2-10509-16"-D. Consequently, the requirements of Specifications 3.5.B.2 and 3.5.B.3 will impose the corresponding surveillance testing of equipment associated with both reactors if the shared RHR service water pump or pumps, or the cross tie line, are made or found to be inoperable.

2. From and after the date that one of the RHR service water pumps is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding 30 days unless such pump is sooner made operable, provided that during such 30 days all other active components of the containment cooling mode of the RHR system are operable.

3. From and after the date that one loop of the containment cooling mode of the RHR system is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding 7 days unless such subsystem is sooner made operable, provided that all active components of the other loop of the containment cooling mode of the RHR system, both core spray subsystems, and both diesel generators required for operation of such components if no external source of power were available, shall be operable.

b.	Flow rate test - each RHR service water pump shall deliver at least 3500 gpm against a pressure of 198	After pump maintenance and every 3 months
c.	A logic system functional test	Each refueling outage

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.

2. 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

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.a 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 limitation 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 provision of Specification 3.5.C.4 shall be implemented.

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 ≥ 1150 psig when steam is being supplied to the turbine at 920 to 1005 psig.	Every 92 days

3.5 LIMITING CONDITIONS FOR OPERATION BASES

A. Core Spray and LPCI Mode of the RHR System

This specification assures that adequate emergency cooling capability is available whenever irradiated fuel is in the reactor vessel.

Based on the loss-of-coolant analytical methods described in General Electric Topical Report NEDC-31345P core cooling systems provide sufficient cooling to the core to dissipate the energy associated with the loss-of-coolant accident, to limit calculated fuel cladding temperature to less than 2200°F, to assure that core geometry remains intact, to limit cladding metal-water reaction to less than 1%, and to limit the calculated local metal-water reaction to less than 17%.

The limiting conditions of operation in Specifications 3.5.A.1 through 3.5.A.6 specify the combinations of operable subsystems to assure the availability of the minimum cooling systems noted above.

Core spray distribution has been show, in full-scale tests of systems similar in design to that of Quad-Cities 1 and 2, to exceed the minimum requirements by at least 25%. In addition, cooling effectiveness has been demonstrated at less than half the rated flow in simulated fuel assemblies with heater rods to duplicate the decay heat characteristics of irradiated fuel. The accident analysis is additionally conservative in that no credit is taken for spray cooling of the reactor core before the internal pressure has fallen to 90 psig.

The LPCI mode of the RHR system is designed to provide emergency cooling to the core by flooding in the event of a loss-of-coolant accident. This system functions in combination with the core spray system to prevent excessive fuel cladding temperature. The LPCI mode of the RHR system in combination with the core spray subsystem provides adequate cooling for break areas of approximately 0.05 ft^2 up to and including 4.26 ft^2, the latter being the double-ended recirculation line break with the equalizer line between the recirculation loops closed without assistance from the high-pressure emergency core cooling subsystems.

The allowable repair times are established so that the average risk rate for repair would be no greater than the basis risk rate. The method and concept are described in Reference 3. Using the results developed in this reference, the repair period is found to be less than half the test interval. This assumes that the core spray subsystems and LPCI constitute a one-out-of-two system; however, the combined effect of the two systems to limit excessive cladding temperature must also be considered. The test interval specified in Specification 4.5 was 3 months. Therefore, an allowable repair period which maintains the basic risk considering single failures should be less than 30 days, and this specification is within this period. Although it is recognized that the information given in Reference 1 provides a quantitative method to estimate allowable repair times, the lack of operating data to support the analytical approach prevents complete acceptance of this method at this time. Therefore, the times stated in the specific items were established with due regard to judgment.

Should one core spray subsystem become inoperable, the remaining core spray subsystem and the entire LPCI mode of the RHR system are available should the need for core cooling arise. Based on judgments of the reliability of the remaining systems, i.e., the core spray and LPCI, a 7-day repair period was obtained.

Should the loss of one RHR pump occur, a nearly full complement of core and containment cooling equipment is available. Three RHR pumps in conjunction with the core spray subsystem will perform the core cooling function. Because of the availability of the majority of the core cooling equipment, a 30-day repair period is justified. If the LPCI mode of the RHR system is not available, at least two RHR pumps must be available to fulfill the containment cooling function. The 7-day repair period is set on this basis.

B. RHR Service Water

The containment cooling mode of the RHR system is provided to remove heat energy from the containment in the event of a loss-of-coolant accident. For the flow specified, the containment long-term pressure is limited to less than 8 psig and is therefore more than ample to provide the required heat-removal capability (reference SAR Section 5.2.3.2).

The Containment Cooling mode of the RHR System consists of two loops. Each loop consists of 1 Heat Exchanger, 2 RHR Pumps, and the associated valves, piping, electrical equipment, and instrumentation. The "A" loop on each unit contains 2 RHR Service Water Pumps. Until November 1, 1989, the "B" loop on each unit may utilize the "C" and "D" RHR Service Water Pumps from Unit 1 via a cross-tie line. After November 1, 1989, each "B" loop will contain 2 RHR Service Water Pumps. Either set of equipment is capable of performing the containment cooling function. Loss of one RHR service water pump does not seriously jeopardize the containment cooling capability, as any one of the remaining three pumps can satisfy the cooling requirements. Since there is some redundancy left, a 30-day repair period is adequate. Loss of one loop of the containment cooling mode of the RHR system leaves one remaining system to perform the containment cooling function. Based on the fact that when one loop of the containment cooling mode of the RHR system becomes inoperable, only one system remains, a 7-day repair period was specified.

C. High-Pressure Coolant Injection

The high-pressure coolant injection subsystem is provided to adequately cool the core for all pipe breaks smaller than those for which the LPCI mode of the RHR system or core spray subsystems can protect the core.

The HPCI meets this requirement without the use of offsite electrical power. For the pipe breaks for which the HPCI is intended to function, the core never uncovers and is continuously cooled, thus no cladding damage occurs (reference SAR Section 6.2.5.3). The repair times for the limiting conditions of operation were set considering the use of the HPCI as part of the isolation cooling system.

D. Automatic Pressure Relief

The relief valves of the automatic pressure relief subsystem are a backup to the HPCI subsystem. They enable the core spray subsystem and LPCI mode of the RHR system to provide protection against the small pipe break in the event of HPCI failure by depressurizing the reactor vessel rapidly enough to actuate the core spray subsystem and the LPCI mode of the RHR system. The core spray subsystem and the LPCI mode of the RHR system provide sufficient flow of coolant to limit fuel cladding temperatures to less than 2200°F, to assure that core geometry remains intact, to limit the core wide clad metal-water reaction to less than 1%, and to limit the calculated local metal-water reaction to less than 17%.

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.

With a system, subsystem, loop, or equipment out-of-service, overall core and containment cooling reliability is maintained by verifying the operability of the remaining systems, subsystems, loops, or equipment. 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.

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.

The surveillance requirements bases described in this paragraph apply only to the RCIC and HPCI systems. 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 pressure at which it is

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

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. 140
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 October 21, 1993 , 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. 140 , 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

James E. Dyer

James E. Dyer, 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, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 140

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
3.5/4.5-2	3.5/4.5-2
3.5/4.5-2a	-
3.5/4.5-3	3.5/4.5-3
3.5/4.5-4	3.5/4.5-4
3.5/4.5-11	3.5/4.5-11
3.5/4.5-12	3.5/4.5-12
3.5/4.5-15	3.5/4.5-15

e. Core spray header Δp instrumentation

check	Once/day
calibrate	Once/3 months
test	Once/3 months

f. Logic system functional test — Once/each refueling outage

2. From and after the date that one of the core spray subsystems is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding 7 days unless such subsystem is sooner made operable, provided that during such 7 days all active components of the other core spray subsystem and the LPCI mode of the RHR system and the diesel generators required for operation of such components if no external source of power were available shall be operable.

2. LPCI mode of the RHR system testing shall be as specified in Specifications 4.5.A.1.a, b, c, d, and f except that each LPCI division (two RHR pumps per division) shall deliver at least 9000 gpm against a system head corresponding to a reactor vessel pressure of 20 psig, with a minimum flow valve open.

3. The LPCI mode of the RHR system shall be operable whenever irradiated fuel is in the reactor vessel and prior to reactor startup from a cold condition.

4. From and after the date that one of the RHR pumps is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding 30 days unless such pump is sooner made operable, provided that during such 30 days the remaining active components of the LPCI mode of the RHR, containment cooling mode of the RHR, all active components of both core spray subsystems, and the diesel generators required for operation of such components if no external source of power were available shall be operable.

5. From and after the date that the LPCI mode of the RHR system is made or found to be inoperable for any reason,

continued reactor operation is permissible only during the succeeding 7 days unless it is sooner made operable, provided that during such 7 days all active components of both core spray subsystems, the containment cooling mode of the RHR (including two RHR pumps), and the diesel generators required for operation of such components if no external source of power were available shall be operable.

6. If the requirements of Specification 3.5.A cannot be met, an orderly shutdown of the reactor shall be initiated, and the reactor shall be in the cold shutdown condition within 24 hours.

B. Containment Cooling Mode of the RHR System

1. a. Both loops of the containment cooling mode of the RHR system, as defined in the bases for Specification 3.5.B, shall be operable whenever irradiated fuel is in the reactor vessel and prior to reactor startup from a cold condition.

1. b. From the effective date of this amendment until Nov. 1, 1989, the "B" loop of the containment cooling mode of the RHR system for each reactor may share the Unit 1 "C" and "D" RHR service water pumps using cross tie line 1/2-10509-16"-D. Consequently, the requirements of Specifications 3.5.B.2 and 3.5.B.3 will impose the corresponding surveillance testing of equipment associated with both reactors if the shared RHR service water pump or pumps, or the cross tie line, are made or found to be inoperable.

2. From and after the date that one of the RHR service water pumps is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding 30 days unless such pump is sooner made operable, provided that during such 30 days all other active components of the containment cooling mode of

B. Containment Cooling Mode of the RHR System

Surveillance of the containment cooling mode of the RHR system shall be performed as follows:

1. RHR service water subsystem testing:

	Item	Frequency
a.	Pump and valve operability	Once/3 months
b.	Flow rate test - each RHR service water pump shall deliver at least 3500 gpm against a pressure of 198 psig	After pump maintenance and every 3 months
c.	A logic system functional test	Each refueling outage

the RHR system are operable.

3. From and after the date that one loop of the containment cooling mode of the RHR system is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding 7 days unless such subsystem is sooner made operable, provided that all active components of the other loop of the containment cooling mode of the RHR system, both core spray subsystems, and both diesel generators required for operation of such components if no external source of power were available, shall be operable.

During the time period from April 17, 1978 through April 30, 1978 while the 2A Containment Cooling Loop of the RHR System is made inoperable for heat exchanger repair, continued reactor operation is permissible beyond the above 7-day limitation, unless such loop is sooner made operable, provided that during the time the 7-day limit is exceeded, a visual inspection is performed daily to assure that proper valve alignment and system integrity is maintained in the "B" RHR loop.

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.

2. 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.

3.5 LIMITING CONDITIONS FOR OPERATION BASES

A. Core Spray and LPCI Mode of the RHR System

This specification assures that adequate emergency cooling capability is available.

Based on the loss-of-coolant analyses included in References 1 and 2 and in accordance with 10 CFR 50.46 and Appendix K, core cooling systems provide sufficient cooling to the core to dissipate the energy associated with the loss-of-coolant accident, to limit the calculated fuel cladding temperature to less than 2200°F, to assure that core geometry remains intact to limit the corewide cladding metal-water reaction to less than 1% and to limit the calculated local metal-water reaction to less than 17%.

The allowable repair times are established so that the average risk rate for repair would be no greater than the basic risk rate. The method and concept are described in Reference 3. Using the results developed in this reference, the repair period is found to be less than half the test interval. This assumes that the core spray subsystems and LPCI constitute a one-out-of-two system; however, the combined effect of the two systems to limit excessive cladding temperature must also be considered. The test interval specified in Specification 4.5 was 3 months. Therefore, an allowable repair period which maintains the basic risk considering single failures should be less than 30 days, and this specification is within this period. Although it is recognized that the information given in Reference 3 provides a quantitative method to estimate allowable repair times, the lack of operating data to support the analytical approach prevents complete acceptance of this method at this time. Therefore, the times stated in the specific items were established with due regard to judgment.

Should one core spray subsystem become inoperable, the remaining core spray subsystem and the entire LPCI mode of the RHR system are available should the need for core cooling arise. Based on judgments of the reliability of the remaining systems, i.e., the core spray and LPCI, a 7-day repair period was obtained.

Should the loss of one RHR pump occur, a nearly full complement of core and containment cooling equipment is available. Three RHR pumps in conjunction with the core spray subsystem will perform the core cooling function. Because of the availability of the majority of the core cooling equipment, a 30-day repair period is justified. If the LPCI mode of the RHR system is not available, at least two RHR pumps must be available to fulfill the containment cooling function. The 7-day repair period is set on this basis.

B. RHR Service Water

The containment cooling mode of the RHR system is provided to remove heat energy from the containment in the event of a loss-of-coolant accident. For the flow specified, the containment long-term pressure is limited to less than 8 psig and is therefore more than ample to provide the required heat-removal capability (reference SAR Section 5.2.3.2).

The Containment Cooling mode of the RHR System consists of two loops. Each loop consists of 1 Heat Exchanger, 2 RHR Pumps, and the associated valves, piping, electrical equipment, and instrumentation. The "A" loop on each unit contains 2 RHR Service Water Pumps. Until Nov. 1, 1989, the "B" loop on each unit may utilize the "C" and "D" RHR Service Water Pumps from Unit 1 via a cross-tie line. After Nov. 1, 1989, each "B" loop will contain 2 RHR Service Water Pumps. Either set of equipment is capable of performing the containment cooling function. Loss of one RHR service water pump does not seriously jeopardize the containment cooling capability, as any one of the remaining three pumps can satisfy the cooling requirements. Since there is some redundancy left, a 30-day repair period is adequate. Loss of one loop of the containment cooling mode of the RHR system leaves one remaining system to perform the containment cooling function. Based on the fact that when one system of the containment cooling mode of the RHR system becomes inoperable, only one system remains, a 7-day repair period was specified.

C. High-Pressure Coolant Injection

The high-pressure coolant injection subsystem is provided to adequately cool the core for all pipe breaks smaller than those for which the LPCI mode of the RHR system or core spray subsystems can protect the core.

The HPCI meets this requirement without the use of offsite electrical power. For the pipe breaks for which the HPCI is intended to function, the core never uncovers and is continuously cooled, thus no cladding damage occurs (reference SAR Section 6.2.5.3). The repair times for the limiting conditions of operation were set considering the use of the HPCI as part of the isolation cooling system.

D. Automatic Pressure Relief

The relief valves of the automatic pressure relief subsystems are a backup to the HPCI subsystem. They enable the core spray subsystem and LPCI mode of the RHR system to provide protection against the small pipe break in the event of HPCI failure by depressurizing the reactor vessel rapidly enough to actuate the core spray subsystems and LPCI mode of the RHR system. The core spray subsystem and/or the LPCI mode of the RHR system provide sufficient flow of coolant to limit fuel cladding temperatures to less than 2200°F, to assure that core geometry remains intact, to limit the core wide clad metal-water reaction to less than 1%, and to limit the calculated local metal-water reaction to less than 17%.

Analyses have shown that only four of the five valves in the automatic depressurization system are required to operate. Loss of one of the relief valves does not significantly affect the pressure relieving capability, therefore continued operation is acceptable. Loss of two relief valves significantly reduces the pressure relief capability of the ADS: thus, a 7 day repair period is specified with the HPCI available, and a 24 hour repair period with the HPCI unavailable.

E. RCIC

The RCIC system is provided to supply continuous makeup water to the reactor core when the reactor is isolated from the turbine and when the feedwater system is not available. Under these conditions the pumping capacity of the RCIC system is sufficient to maintain the water level above the core without any other water system in operation. If the water level in the reactor vessel decreases to the RCIC initiation level, the system automatically starts. The system may also be manually initiated at any time.

The HPCI system provides an alternate method of supplying makeup water to the reactor should the normal feedwater become unavailable. Therfore, the specification calls for an operability check of the HPCI system should the RCIC system be found to be inoperable.

F. Emergency Cooling Availability

The purpose of Specification 3.5.F is to assure a minimum of core cooling equipment is available at all times. If, for example, one core spray were out of service and the diesel which powered the opposite core spray were out of service, only two RHR pumps would be available. Likewise, if two RHR pumps were out of service and two RHR service water pumps on the opposite side were also out of service no containment cooling would be available. It is during the refueling outages that major maintenance is performed and during such time that all low-pressure core cooling systems may be out of service. This specification provides that should this occur, no work will be performed on the primary system which could lead to draining the vessel. This work would include work on certain control rod drive components and recirculation systems. Thus, the specification precludes the events which could require core cooling. Specification 3.9 must also be consulted to determine other requirements for the diesel generators.

Quad Cities Units 1 and 2 share certain process systems such as the makeup demineralizers and the radwaste system and also some safety systems such as the standby gas treatment system, batteries, and

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.

With a system, subsystem, loop or equipment out of service, overall core and containment cooling reliability is maintained by verifying the operability of the remaining systems, subsystems, loops or equipment. 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.

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.

The surveillance requirements bases described in this paragraph apply only to the RCIC and HPCI systems. 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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 144 TO FACILITY OPERATING LICENSE NO. DPR-29

AND AMENDMENT NO. 140 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 of October 21, 1993, Commonwealth Edison Company (CECo or the licensee) requested an amendment to Facility Operating Licenses, DPR-29 and DPR-30 for Quad Cities Units 1 and 2. The proposed amendment would modify the Technical Specifications (TS) to incorporate the line-item TS improvements that were identified by the staff of the U.S. Nuclear Regulatory Commission (NRC) as reported in NUREG-1366, "Improvements to Technical Specification Surveillance Requirements," December 1992. The TS improvements were based on an NRC study of surveillance requirements and included information provided by licensee personnel that plan, manage, and perform surveillances. The study included insights from a qualitative risk assessment of surveillance requirements based on the standard technical specifications for Westinghouse plants and the TS for the Edwin I. Hatch Nuclear Plant, Unit 2. The staff examined operational data from licensee event reports, the nuclear plant reliability data system (NPRDS), and other sources to assess the effect of TS surveillance requirements on plant operation. The staff evaluated the effect of longer surveillance intervals to reduce the possibility for plant transients, wear on equipment, personnel radiation exposure, and burden on personnel resources. Finally, the staff considered surveillance activities for which the safety benefits are small and not justified when compared to the effects of these activities on the safety of personnel and the plant. The NRC staff issued guidance on the proposed TS changes to all holders of operating licenses or construction permits for nuclear power reactors in Generic Letter (GL) 93-05, dated September 27, 1993.

2.0 EVALUATION

The licensee proposed the modifications to the TS surveillance requirements as discussed below.

The current Quad Cities, Units 1 and 2 TS Section 3.5/4.5 requires immediate and daily operation of the redundant equipment when a Core Spray subsystem; Containment Cooling subsystem or pump; or the Low Pressure Coolant Injection (LPCI) mode of the Residual Heat Removal (RHR) system or a pump in the LPCI mode of the RHR system; are found to be inoperable. The Containment Cooling subsystem includes the residual heat removal service water (RHRSW) and RHR pumps. Therefore, the maintenance activities being performed on the RHRSW pumps require the licensee to start and operate each of the remaining RHRSW pumps and the RHR pumps on a daily basis. This daily starting and operation of the pumps during an RHRSW pump outage provides unnecessary challenges to the pumps and pump seals. To eliminate unnecessary testing, the licensee proposed a revision to the present TSs. The proposed revision would remove the requirements for performing the test to demonstrate the operability of alternate trains, systems, or subsystems when one train, system, or subsystem is inoperable. The bases section would also change to reflect the removal of operability requirements.

The proposed TS modifications are consistent with the guidance provided in GL 93-05. This guidance is based on the NRC staff findings and recommendations stated in NUREG-1366. The staff concludes that the proposed TS changes do not adversely affect plant safety and will result in a net benefit to the safe operation of the facility, and, therefore, are acceptable.

3.0 STATE CONSULTATION

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

4.0 ENVIRONMENTAL CONSIDERATION

The amendments change 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 and change surveillance requirements. The NRC staff has determined that the amendments involve 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 the amendments involve no significant hazards consideration, and there has been no public comment on such finding (58 FR 59747). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

5.0 CONCLUSION

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

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