January 21, 1987

Docket Nos. 50-254/265

Mr. Dennis L. Farrar
Director of Nuclear Licensing
Commonwealth Edison Company
Post Office Box 767
Chicago, Illinois 60690

Dear Mr. Farrar:

SUBJECT: EMERGENCY DIESEL GENERATOR SURVEILLANCE TESTING (MPA D-19; TAC 55894, 55895, 59232, 59324)

Re: Quad Cities Nuclear Power Station, Units 1 and 2

The Commission has issued the enclosed Amendment Nos. 99 and 96 to Facility Operating License Nos. DPR-29 and DPR-30 for the Quad Cities Nuclear Power Station, Units 1 and 2. The amendments are in response to your application dated June 28, 1985. A portion of your requested changes for each unit to delete the requirement for testing the operable Emergency Diesel Generator (EDG) immediately and daily thereafter when it is determined that either unit EDG or the shared EDG is inoperable was found to be unacceptable and has been denied. While NRC staff agrees that deletion of EDG testing daily thereafter is consistent with the intent of Generic Letter 84-15, it is not NRC staff's intent to eliminate completely the testing of remaining EDGs when an EDG is inoperable. A copy of the Notice of Denial is enclosed. This notice has been forwarded to the Office of the Federal Register for publication.

The amendments modify the Technical Specifications to delete certain EDG testing requirements in response to NRC staff concerns regarding excessive EDG testing as established in Generic Letter 84-15.

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

Sincerely,

Original signed by
John A. Zwolinski, Director
BWR Project Directorate #1
Division of BWR Licensing

Enclosures:
1. Amendment No. 99 to License No. DPR-29
2. Amendment No. 96 to License No. DPR-30
3. Safety Evaluation
4. Notice of Denial

cc w/enclosures:
See next page

DISTRIBUTION

Docket File	CJamerson	BGrimes	OPA
NRC PDR	TRotella	TBarnhart (8)	RDiggs
Local PDR	OGC-BETH	WJones	JPartlow
BWD#1 Reading	LJHarmon	DVassallo	EButcher
RBernero	ELJordan	ACRS (10)	NThompson
		OChopra	JZwolinski

SEE PREVIOUS CONCURRENCE*

DBL:PD#1	DBL:PD#1	OGC-BETH	DBL:PD#1
CJamerson*	TRotella:ac*	RPirfo*	JZwolinski
01/07/87	01/08/87	01/14/87	01/21/87

Docket Nos. 50-254/265

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Re: Quad Cities Nuclear Power Station, Units 1 and 2

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CJamerson	TRotella:ac		JZwolinski
01/07/87	01/08/87	01/ /87	01/ /87

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

COMMONWEALTH EDISON COMPANY

AND

IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

DOCKET NO. 50-254

QUAD CITIES NUCLEAR POWER STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 99
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 June 28, 1985 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. 99, 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

John A. Zwolinski, Director
BWR Project Directorate #1
Division of BWR Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 21, 1987

ATTACHMENT TO LICENSE AMENDMENT NO. 99

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
Bases 3.5/4.5-11a	Bases 3.5/4.5-11a

e. Core spray header Δ p 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. When it is determined that one core spray subsystem is inoperable, the operable core spray subsystem and the LPCI mode of the RHR system shall be demonstrated to be operable immediately. The operable core spray subsystem shall be demonstrated to be operable daily thereafter.

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.

3. 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.

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

4. When it is determined that one of the RHR pumps is inoperable, the remaining active components of the LPCI mode of the RHR, containment cooling mode of the RHR, and both core spray subsystems shall be demonstrated to be operable immediately and the operable RHR pumps daily thereafter.

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 July 1, 1982, the "A" loop of the containment cooling mode of the RHR system for each reactor may share the Unit 2 "A" and "B" RHR service water pumps using cross tie line 1/2-10124-16"-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.

containment cooling mode of the RHR shall be demonstrated to be operable immediately and daily thereafter.

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.

2. When it is determined that one RHR service water pump is inoperable, the remaining components of that loop and the other containment cooling loop of the RHR system shall be demonstrated to be operable immediately and daily thereafter.

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.

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 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 the HPCI subsystem 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,

3. When one loop of the containment cooling mode of the RHR system becomes inoperable, the operable loop shall be demonstrated to be operable immediately, and daily thereafter.

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 follows:

1. HPCI subsystem testing shall be as specified in Specification 4.5.A.1.a.b, c, and d, except that the HPCI pump shall deliver at least 5000 gpm against a system head corresponding to a reactor vessel pressure of 1150 psig to 150 psig, and a logic system functional test shall be performed during each refueling outage.

2. When it is determined that the HPCI subsystem is inoperable, the LPCI mode of the RHR system, both core spray subsystems, the automatic pressure relief subsystem, and the RCIC system shall be demonstrated to be

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. For multiple failures, a shorter interval is specified; to improve the assurance that the remaining systems will function, a daily test is called for. 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 LCPI mode of the RHR system are available should the need for core cooling arise. To assure that the remaining core spray and the LPCI mode of the RHR system are available, they are demonstrated to be operable immediately. This demonstration includes a manual initiation of the pumps and associated valves. Based on judgments of the reliability of the remaining systems, i.e., the core spray and LPCI, a 7-day repair period was obtained.

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

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. 96
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 June 28, 1985, 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 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. 96 , 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

John A. Zwolinski, Director
BWR Project Directorate #1
Division of BWR Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 21, 1987

ATTACHMENT TO LICENSE AMENDMENT NO. 96

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
3.5/4.5-3	3.5/4.5-3
3.5/4.5-4	3.5/4.5-4
Bases 3.5/4.5-11	Bases 3.5/4.5-11

e. Core spray header Δ p 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. When it is determined that one core spray subsystem is inoperable, the operable core spray subsystem and the LPCI mode of the RHR system shall be demonstrated to be operable immediately. The operable core spray subsystem shall be demonstrated to be operable daily thereafter.

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.

3. 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.

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

4. When it is determined that one of the RHR pumps is inoperable, the remaining active components of the LPCI mode of the RHR, containment cooling mode of the RHR and both core spray subsystems shall be demonstrated to be operable immediately and the operable RHR pumps daily thereafter.

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.

containment cooling mode of the RHR shall be demonstrated to be operable immediately and daily thereafter.

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 July 1, 1982, the "A" loop of the containment cooling mode of the RHR system for each reactor may share the Unit 2 "A" and "B" RHR service water pumps using cross tie line 1/2-10124-16"-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.

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.

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 mode operable, provided that during such 30 days all other active components of the containment cooling mode of the RHR system are operable.

2. When it is determined that one RHR service water pump is inoperable, the remaining components of that loop and the other containment cooling loop of the RHR system shall be demonstrated to be operable immediately and daily thereafter.

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.

3. When one loop of the containment cooling mode of the RHR system becomes inoperable, the operable loop shall be demonstrated to be operable immediately and daily thereafter.

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.

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 peak 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. For multiple failures, a shorter interval is specified; to improve the assurance that the remaining systems will function, a daily test is called for. 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. To assure that the remaining core spray and the LPCI mode of the RHR system are available, they are demonstrated to be operable immediately. This demonstration includes a manual initiation of the pumps and associated valves. 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, which will be demonstrated to be operable, 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 RHR5 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 "B" loop on each unit contains 2 RHR Service Water Pumps. During the period from November 24, 1981, to July 1, 1982, the "A" loop on each unit may utilize the "A" and "B" RHR Service Water Pumps from Unit 2 via a cross-tie line. After July 1, 1982, each "A" 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. The operable system is demonstrated to be operable each day when the above condition occurs.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 99 TO FACILITY OPERATING LICENSE NO. DPR-29

AND AMENDMENT NO. 96 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. 50-254/265

1.0 INTRODUCTION

The reliability of Emergency Diesel Generators (EDG) is one of the main factors affecting the risk from station blackout and thus the improvement of EDG reliability can reduce the risk of core damage from station blackout events. Further, the staff has concluded that excessive testing results in degradation of diesel engines and potential for reduced reliability. The staff was concerned with the number of unnecessary EDG tests for the earlier licensed operating plants which are required under their current Technical Specifications (TS) to perform frequent testing. No such TS exists for the recently licensed plants as established in Standard Technical Specifications. In an effort to reduce excessive testing of EDGs in these earlier plants and amend their TS to reflect comparable testing with that of Standard Technical Specifications, Generic Letter 84-15 (D. Eisenhut to All Licensees, dated July 2, 1984) recommended that the surveillance requirements for testing EDGs, because of inoperability of emergency core cooling systems, be deleted from plant unique TS.

2.0 EVALUATION

By submittal dated June 28, 1985, Commonwealth Edison Company (CECo) requested to amend EDG TS for Quad Cities Units 1 and 2. The TS and bases which the licensee proposed to amend for both Quad Cities Units 1 and 2 are as follows:

Surveillance/Bases	Page	Inoperable ECCS Equipment
1. 3.5.A.2	3.5/4.5-2	One core spray system
2. 3.5.A.4	3.5/4.5-2	One of the RHR pumps
3. 3.5.A.5	3.5/4.5-3	LPCI mode of RHR
4. 3.5.B.3	3.5/4.5-4	One containment cooling subsystem
5. Bases 3.5.A	3.5/4.5-11	LCO Bases
6. 3.9.E.1	3.9/4.9-3	EDG inoperable

We have reviewed all of the above proposed TS changes and find the changes to be consistent with the intent of NRC staff actions (Generic Letter 84-15) to improve and maintain EDG reliability by reducing excessive EDG testing, with the exception noted below. Therefore, NRC staff finds all of the proposed TS changes to be acceptable with the exception noted below.

To reduce further the number of EDG starts, the CECo also proposed to change TS 3.9.E.1 on page 3.9/4.9-3. In the current Quad Cities TS, the operable EDGs are required to be tested immediately and daily thereafter when either the unit or shared EDG is declared inoperable. The proposed amendment would delete the initial and daily thereafter testing of operable EDG. While NRC staff agrees that deletion of EDG testing daily thereafter is consistent with the intent of Generic Letter 84-15, it is not NRC staff's intent to eliminate completely the testing of remaining EDGs when an EDG is inoperable. Therefore, the licensee's proposed change to TS 3.9.E.1 on page 3.9/4.9-3 is unacceptable and has been denied.

3.0 ENVIRONMENTAL CONSIDERATION

These amendments involve change to a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes to the surveillance requirements. The 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 these amendments involve no significant hazards consideration and there has been no public comment on such finding. Accordingly, these 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 nor environmental assessment need be prepared in connection with the issuance of these amendments.

4.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security nor to the health and safety of the public.

Principal Contributor: O. Chopra, T. Rotella

Dated: January 21, 1987

UNITED STATES NUCLEAR REGULATORY COMMISSION

COMMONWEALTH EDISON COMPANY

DOCKET NOS. 50-254 AND 50-265

DENIAL OF AMENDMENT TO FACILITY OPERATING LICENSE

AND OPPORTUNTIY FOR HEARING

The U.S. Nuclear Regulatory Commission (the Commission) has denied in part a request by the licensee for an amendment to Facility Operating License Nos. DPR-29 and DPR-30 issued to the Commonwealth Edison Company for operation of the Quad Cities Nuclear Power Station, Units 1 and 2 in Rock Island County, Illinois. Notice of consideration of issuance of these amendments was published in the FEDERAL REGISTER on August 14, 1985 (50 FR 32790).

The amendments, as proposed by the licensee, would change Technical Specification 3.9.E.1, for both Units 1 and 2, deleting the requirement for testing the operable Emergency Diesel Generator (EDG) immediately and daily thereafter when it is determined that either unit EDG or the shared EDG is inoperable. Standard Technical Specifications require, as a minimum, testing the operable EDG within 24 hours and 72 hours thereafter when the other unit diesel is inoperable.

Although NRC staff has determined that reducing EDG cold fast starts would improve EDG reliability and therefore, the risk of core damage from station blackout events, it was not NRC staff's intent to eliminate completely the testing of the remaining EDG when an EDG is declared inoperable. NRC staff intent was only to reduce testing. Therefore, the Commonwealth Edison Company proposed changes were denied.

All other proposed changes to the Technical Specifications, for both Units 1 and 2, have been approved by Amendment Nos. 99 , and 96 . Notice of issuance of Amendment Nos. 99 and 96 , will be published in the Commission's biweekly FEDERAL REGISTER Notices.

The licensee was notified of the Commission's denial of the proposed technical specification changes by letter dated January 21 , 1987.

By March 2 , 1987 the licensee may demand a hearing with respect to the denial described above and any person whose interest may be affected by this proceeding may file a written petition for leave to intervene.

A request for a hearing or petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Docketing and Service Branch, or may be delivered to the Commission's Public Document Room, 1717 H Street, NW, Washington, D.C., by the above date.

A copy of any petitions should also be sent to the Office of the General Counsel-Bethesda, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, and to M. I. Miller; Isham, Lincoln & Beale, Three First National Plaza, Suite 5200, Chicago, Illinois 60602.

For further details with respect to this action, see (1) the application for amendment dated June 28, 1985, and (2) the Commission's Safety Evaluation issued with Amendment No. 99 to DPR-29 and Amendment No. 96 to DPR-30, which are available for public inspection at the Commission's Public Document Room, 1717 H Street, NW, Washington, D.C., and at the Moline Public Library, 504 17th Street, Moline, Illinois 61265. A copy of item (2) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of BWR Licensing.

Dated at Bethesda, Maryland, this 21st day of January, 1987.

FOR THE NUCLEAR REGULATORY COMMISSION

John A. Zwolinski, Director
BWR Project Directorate #1
Division of BWR Licensing

January 21, 1987

MEMORANDUM FOR: Sholly Coordinator

FROM: John A. Zwolinski, Director
BWR Directorate #1, DBL

SUBJECT: REQUEST FOR PUBLICATION IN BIWEEKLY FR NOTICE - NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSES (TAC 55894, 55895, 59323,59324)

Commonwealth Edison Company, Docket Nos. 50-254 and 50-265, Quad Cities Nuclear Power Station, Units 1 and 2, Rock Island County, Illinois

Date of application for amendments: June 28, 1985

Brief description of amendments: The amendments change the technical specifications to delete certain emergency diesel generator surveillance testing requirements. A portion of the amendment request has been denied by the Commission and a separate Notice of Denial of Amendment has been forwarded to the Office of the Federal Register for publication.

Date of issuance: January 21, 1987

Effective date: January 21, 1987

Amendment Nos.: 99 and 96

Facility Operating License Nos. DPR-29 and DPR-30. Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: August 14, 1985 (50 FR 32790).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 21, 1987 .

No significant hazards consideration comments received: No.

Local Public Document Room location: Moline Public Library, 504 - 17th Street, Moline, Illinois 61265.

Original signed by
John A. Zwolinski, Director
BWR Project Directorate #1, DBL

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