Docket Nos.: 50-254
50-265

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OCT 2 3 1976

Commonwealth Edison Company
ATTN: Mr. R. L. Bolger
Assistant Vice President
Post Office Box 767
Chicago, Illinois 60690

Gentlemen:

In response to your requests dated October 13 and 19, 1976, the Commission has issued the enclosed Amendment Nos. 33 and 32 to Facility Operating License Nos. DPR-29 and DPR-30 for Unit Nos. 1 and 2 of the Quad Cities Nuclear Power Station, respectively.

These amendments require that the structural integrity of the primary coolant system boundary be maintained at the level required by Section XI of the 1974 Edition of the ASME Boiler and Pressure Vessel Code. This specification supersedes a previous requirement based upon Section III of the 1965 Edition of the ASME Code.

Copies of our related Safety Evaluation and the Notice of Issuance also are enclosed.

Sincerely,

Original Signed by:
Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

Enclosures:
1. Amendment No. 33 to License No. DPR-29
2. Amendment No. 32 to License No. DPR-30
3. Safety Evaluation
4. Notice

#33 for DPR-29
#32 for DPR-30

cc w/enclosures:
See next page

DRZ

OFFICE➤	ORB#2:DOR	ORB#2:DOR	OELD	DOR:ORB #2		
SURNAME➤	RDiggs	PO'Connor:rm	KARMAN	DLZiemann		
DATE➤	10/20/76	10/20/76	10/22/76	10/23/76		

Form AEC-318 (Rev. 9-53) AECM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1974-526-166

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cc w/enclosures and cy. of CECo
filings dtd. 10/13/76 and 10/19/76
Mr. Leroy Stratton
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Illinois Department of Public Health
Springfield, Illinois 62706

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 UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 33
License No. DPR-29

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

   A. The application for amendment by the Commonwealth Edison Company (the licensee) dated October 13, 1976, as supplemented by letter dated October 19, 1976, 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.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed by:
Dennis L. Ziemann
Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: OCT 2 5 1976

ATTACHMENT TO LICENSE AMENDMENT NO. 33

FACILITY OPERATING LICENSE NO. DPR-29

DOCKET NO. 50-254

Replace the following pages of the Technical Specifications contained in Appendix A of the above-indicated license with the attached pages bearing the same numbers except as otherwise indicated. The changed areas on the pages are reflected by a marginal line.

Remove Pages	Insert Pages
3.6/4.6-4	3.6/4.6-4
	3.6/4.6-4a (new)
	3.6/4.6-4b (new)
3.6/4.6-12	3.6/4.6-12
	3.6/4.6-12a (new)

2. Both the sump and air sampling systems shall be operable during reactor power operation. From and after the date that one of these systems is made or found to be inoperable for any reason, reactor power operation is permissible only during the succeeding 7 days.

3. If the conditions in 1 or 2 above cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

E. Safety and Relief Valves

1. Prior to reactor startup for power operation, during reactor power operating conditions, and whenever the reactor coolant pressure is greater than 90 psig and temperature greater than 320° F, all nine of the safety valves shall be operable. The solenoid-activated pressure valves shall be operable as required by Specification 3.5.D.

2. If Specification 3.6.E.1 is not met, the reactor shall remain shut down until the condition is corrected or, if in operation, an orderly shutdown shall be initiated and the reactor coolant pressure and temperature shall be below 90 psig and 320° F within 24 hours.

F. Structural Integrity

The structural integrity of the primary system boundary shall be maintained at the level required by the ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components", 1974 Edition, Summer 1975 Addenda. (ASME Code Section XI)

E. Safety and Relief Valves

A minimum of 1/2 of all safety valves shall be bench checked or replaced with a bench checked valve each refueling outage. The popping point of the safety valves shall be set as follows:

Number of Valves	*Setpoint (psig)*
1	1125[1]
2	1240
2	1250
4	1260

The allowable setpoint error for each valve is ± 1%.

All relief valves shall be checked for set pressure each refueling outage. The set pressures shall be:

Number of Valves	*Setpoint (psig)*
1	≤ 1125[1]
2	≤ 1130
2	≤ 1135

[1] Target Rock combination safety/relief valve.

F. Structural Integrity

The nondestructive inspections listed in Table 4.6.1 shall be performed as specified in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1971Edition, Summer 1971 Addenda. the results obtained from compliance with this spec-

F. STRUCTURAL INTEGRITY (CONT)

Components of the primary system boundary whose inservice examination reveals the absence of flaw indications or flaw indications not in excess of the allowable indication standards of this Code are acceptable for continued service. Plant operation with components which have inservice examination flaw indication(s) in excess of the allowable indication standards of the Code shall be subject to NRC approval.

a. Components whose inservice examination reveals flaw indication(s) in excess of the allowable indication standards of the ASME Code, Section XI, are unacceptable for continued service unless the following requirements are met:

(i) An analysis and evaluation of the detected flaw indication(s) shall be submitted to the NRC that demonstrate that the component structural integrity justifies continued service. The analysis and evaluation shall follow the procedures outlined in Appendix A, "Evaluation of Flaw Indications", of ASME Code, Section XI.

(ii) Prior to the resumption of service, the NRC shall review the analysis and evaluation and either approve resumption of plant operation with the affected component or require that the component be repaired or replaced.

F. STRUCTURAL INTEGRITY (CONT)

ification will be evaluated after 5 years and the conclusions will be reviewed with the NRC.

b. For components approved for continued service in accordance with paragraph a. above, reexamination of the area containing the flaw indication(s) shall be conducted during each scheduled successive inservice inspection. An analysis and evaluation shall be submitted to the NRC following each inservice inspection. The analysis and evaluation shall follow the procedures outlined in Appendix A, "Evaluation of Flaw Indications", of ASME Code, Section XI, and shall reference prior analyses submitted to the NRC to the extent applicable. Prior to resumption of service following each inservice inspection, the NRC shall review the analysis and evaluation and either approve resumption of plant operation with the affected component or require that the component be repaired or replaced.

c. Repair or replacement of components, including reexaminations, shall conform with the requirements of the ASME Code, Section XI. In the case of repairs, flaws shall be either removed or repaired to the extent necessary to meet the allowable indication standards specified in ASME Code, Section XI.

not exceeded. Solenoid-actuated relief valves are used to avoid activation of the safety valves. In view of the fact that the solenoid-activated relief valves are more complicated, it is prudent to test them at each refueling outage. The safety valves are required to be operable above the design pressure (90 psig) at which the core spray subsystems are not designed to deliver full flow.

F. Structural Integrity

A preservice inspection of the components listed in Table 4.6-1 will be conducted after site erection to assure the system is free of gross defects and as a reference base for later inspections.

Prior to operation, the reactor primary system will be free of gross defects. In addition, the facility has been designed such that gross defects should not occur throughout life. The inspection program given in Table 4.6.1 was based on Section XI of the ASME Boiler and Pressure Vessel Code, 1971 Edition, Summer 1971 Addenda, which was followed except where accessibility for inspection was not provided. This edition of the Code is suitable for detecting flaw indications but does not provide adequate guidance for the evaluation of ultrasonic reflectors. The requirement in the 1971 Edition of Section XI that the operator evaluate the reflector to determine the size, shape, and nature can best be satisfied by examination and evaluation of the flaw in accordance with the techniques presented in Appendix A to ASME Section XI in the 1974 Edition, Summer 1975 Addenda. It is the intent of this specification to require inservice inspection of the primary system boundary per Table 4.6.1 of this specification and the 1971 Edition of ASME Section XI including the Summer 1971 Addenda and to permit the evaluation of flaws in excess of the acceptance standards of that Edition and Addenda in accordance with the techniques of the 1974 version.

Table 4.6-1 was based on Section XI of the ASME Boiler and Pressure Vessel Code. Rules for Inservice Inspection of Nuclear Reactor Coolant Systems, which was followed except where accessibility for inspection was not provided. The Commonwealth Edison Company recognizes the importance of inspection of those areas which are presently not accessible and will study and implement, if practicable, new means to include those areas within the inspection program. This inspection provides further assurance that gross defects are not occurring after the system is in service. This inspection will reveal problem areas should they occur before a leak develops.

The special inspection of the main feed and steamlines is to provide added protection against pipe whip, in addition to the protective energy absorbing system to be installed inside the drywell as described in Amendment 27 to the SAR. The Group I welds are selected on the basis of an analysis that shows these welds are the highest stress welds and that due to their physical location, a break would result in the least interference and maximum energy upon impact with the drywell. These welds are the only ones which offer any significant risk and are therefore inspected 4 times as often as the other welds within the drywell.

Group II welds are selected because without regard for the operating stress levels and interfering equipment, they have sufficient theoretical energy to penetrate and would propel the pipe toward the containment. They are therefore included in first inspection. Upon consideration of impact angle, interfering equipment, and the distance pipe travels, no substantial risk is involved and no extra inspection is needed.

In addition, extensive visual inspection for leaks will be made periodically on critical systems. The inspection program specified encompasses the major areas of the vessel and piping systems within the drywell. The inspection period is based on the observed rate of growth of defects from fatigue studies sponsored by the NRC. These studies show that it requires thousands of stress cycles at stresses beyond any expected to occur in a reactor system to propagate a crack. The test frequency established is at intervals such that in comparison to study results only a small number of stress cycles at values below limits will occur. On this basis, it is considered that the test frequencies are adequate.

The type of inspection planned for each component depends on location, accessibility, and type of expected defect. Direct visual examination is proposed wherever possible since it is sensitive, fast, and reliable. Magnetic particle and liquid penetrant inspections are planned where practical and where added sensitivity is required. Ultrasonic testing and radiography shall be used where defects can occur on concealed surfaces.

After 5 years of operation, a program for inservice inspection of piping and components within the primary pressure boundary which are outside the downstream containment isolation valve shall be submitted to the NRC.

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 UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 32
License No. DPR-30

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

   A. The application for amendment by the Commonwealth Edison Company (the licensee) dated October 13, 1976, as supplemented by letter dated October 19, 1976, 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.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed by:
Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: [illegible] 1976

ATTACHMENT TO LICENSE NO. 32

FACILITY OPERATING LICENSE NO. DPR-30

DOCKET NO. 50-265

Replace the following pages of the Technical Specifications contained in Appendix A of the above-indicated license with the attached pages bearing the same numbers, except as otherwise indicated. The changed areas on the pages are reflected by a marginal line.

Remove Pages	Insert Pages
3.6/4.6-4	3.6/4.6-4
	3.6/4.6-4a (new)
	3.6/4.6-4b (new)
3.6/4.6-12	3.6/4.6-12
	3.6/4.6-12a (new)

2. Both the sump and air sampling systems shall be operable during reactor power operation. From and after the date that one of these systems is made or found to be inoperable for any reason, reactor power operation is permissible only during the succeeding 7 days.

3. If the conditions in 1 or 2 above cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

E. **Safety and Relief Valves**

1. Prior to reactor startup for power operation, during reactor power operating conditions, and whenever the reactor coolant pressure is greater than 90 psig and temperature greater than 320° F, all nine of the safety valves shall be operable. The solenoid-activated pressure valves shall be operable as required by Specification 3.5.D.

2. If Specification 3.6.E.1 is not met, the reactor shall remain shut down until the condition is corrected or, if in operation, an orderly shutdown shall be initiated and the reactor coolant pressure and temperature shall be below 90 psig and 320° F within 24 hours.

E. **Safety and Relief Valves**

A minimum of 1/2 of all safety valves shall be bench checked or replaced with a bench checked valve each refueling outage. The popping point of the safety valves shall be set as follows:

Number of Valves	*Setpoint (psig)*
1	1125(1)
2	1240
2	1250
4	1260

The allowable setpoint error for each valve is ± 1%.

All relief valves shall be checked for set pressure each refueling outage. The set pressures shall be:

Number of Valves	*Setpoint (psig)*
1	≤ 1125(1)
2	≤ 1130
2	≤ 1135

(1)Target Rock combination safety/relief valve.

F. **Structural Integrity**

The structural integrity of the primary system boundary shall be maintained at the level required by the ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components", 1974 Edition, Summer 1975 Addenda. (ASME Code Section XI)

F. **Structural Integrity**

The nondestructive inspections listed in Table 4.6.1 shall be performed as specified in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1971Edition, Summer 1971 Addenda. the results obtained from compliance with this spec-

F. STRUCTURAL INTEGRITY (CONT)

Components of the primary system boundary whose inservice examination reveals the absence of flaw indications or flaw indications not in excess of the allowable indication standards of this Code are acceptable for continued service. Plant operation with components which have inservice examination flaw indication(s) in excess of the allowable indication standards of the Code shall be subject to NRC approval.

a. Components whose inservice examination reveals flaw indication(s) in excess of the allowable indication standards of the ASME Code, Section XI, are unacceptable for continued service unless the following requirements are met:

(i) An analysis and evaluation of the detected flaw indication(s) shall be submitted to the NRC that demonstrate that the component structural integrity justifies continued service. The analysis and evaluation shall follow the procedures outlined in Appendix A, "Evaluation of Flaw Indications", of ASME Code, Section XI.

(ii) Prior to the resumption of service, the NRC shall review the analysis and evaluation and either approve resumption of plant operation with the affected component or require that the component be repaired or replaced.

F. STRUCTURAL INTEGRITY (CONT)

ification will be evaluated after 5 years and the conclusions will be reviewed with the NRC.

b. For components approved for continued service in accordance with paragraph a. above, reexamination of the area containing the flaw indication(s) shall be conducted during each scheduled successive inservice inspection. An analysis and evaluation shall be submitted to the NRC following each inservice inspection. The analysis and evaluation shall follow the procedures outlined in Appendix A, "Evaluation of Flaw Indications", of ASME Code, Section XI, and shall reference prior analyses submitted to the NRC to the extent applicable. Prior to resumption of service following each inservice inspection, the NRC shall review the analysis and evaluation and either approve resumption of plant operation with the affected component or require that the component be repaired or replaced.

c. Repair or replacement of components, including reexaminations, shall conform with the requirements of the ASME Code, Section XI. In the case of repairs, flaws shall be either removed or repaired to the extent necessary to meet the allowable indication standards specified in ASME Code, Section XI.

not exceeded. Solenoid-actuated relief valves are used to avoid activation of the safety valves. In view of the fact that the solenoid-activated relief valves are more complicated, it is prudent to test them at each refueling outage. The safety valves are required to be operable above the design pressure (90 psig) at which the core spray subsystems are not designed to deliver full flow.

F. Structural Integrity

A preservice inspection of the components listed in Table 4.6-1 will be conducted after site erection to assure the system is free of gross defects and as a reference base for later inspections.

Prior to operation, the reactor primary system will be free of gross defects. In addition, the facility has been designed such that gross defects should not occur throughout life. The inspection program given in Table 4.6.1 was based on Section XI of the ASME Boiler and Pressure Vessel Code, 1971 Edition, Summer 1971 Addenda, which was followed except where accessibility for inspection was not provided. This edition of the Code is suitable for detecting flaw indications but does not provide adequate guidance for the evaluation of ultrasonic reflectors. The requirement in the 1971 Edition of Section XI that the operator evaluate the reflector to determine the size, shape, and nature can best be satisfied by examination and evaluation of the flaw in accordance with the techniques presented in Appendix A to ASME Section XI in the 1974 Edition, Summer 1975 Addenda. It is the intent of this specification to require inservice inspection of the primary system boundary per Table 4.6.1 of this specification and the 1971 Edition of ASME Section XI including the Summer 1971 Addenda and to permit the evaluation of flaws in excess of the acceptance standards of that Edition and Addenda in accordance with the techniques of the 1974 version.

Table 4.6-1 was based on Section XI of the ASME Boiler and Pressure Vessel Code, Rules for Inservice Inspection of Nuclear Reactor Coolant Systems, which was followed except where accessibility for inspection was not provided. The Commonwealth Edison Company recognizes the importance of inspection of those areas which are presently not accessible and will study and implement, if practicable, new means to include those areas within the inspection program. This inspection provides further assurance that gross defects are not occurring after the system is in service. This inspection will reveal problem areas should they occur before a leak develops.

The special inspection of the main feed and steamlines is to provide added protection against pipe whip, in addition to the protective energy absorbing system to be installed inside the drywell as described in Amendment 27 to the SAR. The Group I welds are selected on the basis of an analysis that shows these welds are the highest stress welds and that due to their physical location, a break would result in the least interference and maximum energy upon impact with the drywell. These welds are the only ones which offer any significant risk and are therefore inspected 4 times as often as the other welds within the drywell.

Group II welds are selected because without regard for the operating stress levels and interfering equipment, they have sufficient theoretical energy to penetrate and would propel the pipe toward the containment. They are therefore included in first inspection. Upon consideration of impact angle, interfering equipment, and the distance pipe travels, no substantial risk is involved and no extra inspection is needed.

In addition, extensive visual inspection for leaks will be made periodically on critical systems. The inspection program specified encompasses the major areas of the vessel and piping systems within the drywell. The inspection period is based on the observed rate of growth of defects from fatigue studies sponsored by the NRC. These studies show that it requires thousands of stress cycles at stresses beyond any expected to occur in a reactor system to propagate a crack. The test frequency established is at intervals such that in comparison to study results only a small number of stress cycles at values below limits will occur. On this basis, it is considered that the test frequencies are adequate.

The type of inspection planned for each component depends on location, accessibility, and type of expected defect. Direct visual examination is proposed wherever possible since it is sensitive, fast, and reliable. Magnetic particle and liquid penetrant inspections are planned where practical and where added sensitivity is required. Ultrasonic testing and radiography shall be used where defects can occur on concealed surfaces.

After 5 years of operation, a program for inservice inspection of piping and components within the primary pressure boundary which are outside the downstream containment isolation valve shall be submitted to the NRC.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NOS. 33 AND 32 TO

FACILITY LICENSE NOS. DPR-29 and DPR-30

COMMONWEALTH EDISON COMPANY

AND

IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

QUAD CITIES UNIT NOS. 1 AND 2

DOCKET NOS. 50-254 AND 50-265

INTRODUCTION

By letter dated October 13, 1976, as supplemented by letter dated October 19, 1976, the Commonwealth Edison Company requested amendments to Facility License Nos. DPR-29 and DPR-30 for the Quad Cities Nuclear Power Station Unit Nos. 1 and 2. The proposed request involves revisions to the Technical Specifications with regard to updating the ASME Code requirements to which Commonwealth Edison maintains the structural integrity of the primary coolant system boundary.

DISCUSSION

The Technical Specifications for Quad Cities Unit Nos. 1 and 2 currently require that the structural integrity of the primary coolant system boundary be maintained at the level required by the original acceptance standards. For Quad Cities Unit Nos. 1 and 2, these original acceptance standards are contained in Section III of the 1965 Edition of the ASME Code. Subsequent to the issuance of the operating licenses for Quad Cities 1 and 2, the Technical Specifications were revised to require that Commonwealth Edison conduct an inservice inspection program at Quad Cities Station that is in accordance with the provisions of Section XI of the ASME Code, "Rules for Inservice Inspection of Nuclear Power Plant Components." Section XI also contains acceptance standards for the structural integrity of the components examined. The proposed technical specification revisions require that the acceptance standards prescribed in Section XI of the ASME Code be applied to the examinations made in accordance with that code.

EVALUATION

The NRC staff has reviewed Commonwealth Edison's proposed revision to the Quad Cities Unit Nos. 1 and 2 Technical Specifications which would require the primary coolant system structural integrity to be maintained at the level required by Section XI of the 1974 Edition of the ASME Boiler and Pressure Vessel Code, "Rules for Inservice Inspection of Nuclear Power Plant Components."

We conclude that because Section XI of the ASME Code contains the acceptance standards for evaluating the results of the inservice inspections required by the Quad Cities Unit Nos. 1 and 2 Technical Specifications, it should be specified as the acceptance standard for the structural integrity of these plants. This conclusion is based upon the following:

1. The original acceptance criteria of Section III of the ASME Code were fabrication criteria and were not intended to be used as inservice acceptance criteria.

2. The acceptance criteria of Section XI of the ASME Code have been explicitly developed for use as acceptance criteria for inservice inspections conducted on nuclear power plant components.

3. The use of the acceptance criteria of Section XI of the ASME Code is consistent with the Commission's current regulations contained in 10 CFR 50.55a which do not apply to Quad Cities Unit Nos. 1 and 2 only because construction permits were issued prior to January 1, 1971. The use of Section XI acceptance criteria is also consistent with the Commission's Regulatory position as expressed in its Standard Technical Specifications for Boiling Water Reactors. The application of the acceptance criteria of Section XI of the ASME Code to Quad Cities Unit Nos. 1 and 2 does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin.

ENVIRONMENTAL CONSIDERATION

We have determined that the amendments do not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendments involve an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR § 51.5(d)(4) that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of these amendments.

CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) because the amendments do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the amendments do not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) 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 or to the health and safety of the public.

Date: October 23, 1976

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKETS NOS. 50-254 AND 50-265

COMMONWEALTH EDISON COMPANY
AND
IOWA-ILLINOIS AND ELECTRIC COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendments Nos. 33 and 32 to Facility Operating License Nos. DPR-29 and DPR-30, issued to Commonwealth Edison Company (acting for itself and on behalf of the Iowa-Illinois Gas and Electric Company), which revised Technical Specifications for operation of the Quad Cities Station Units Nos. 1 and 2 (the facilities) located in Rock Island County, Illinois. The amendments are effective as of their date of issuance.

These license amendments revised the Technical Specifications for the facilities to require that the structural integrity of the primary coolant system boundary be maintained at the level required by Section XI of the 1974 Edition of the ASME Boiler and Pressure Vessel Code. This specification supersedes a previous requirement that was based upon Section III of the 1965 Edition of the ASME Code in accordance with the licensee's request dated October 13, 1976, and supplement thereto dated October 19, 1976.

The application, as supplemented, for the amendments complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments. Prior public notice of these amendments was not required since the amendments do not involve a significant hazards consideration.

The Commission has determined that the issuance of these amendments will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of these amendments.

For further details with respect to this action, see (1) the application for amendments dated October 13, 1976, and supplement thereto dated October 19, 1976, (2) Amendments Nos. 33 and 32 to License Nos. DPR-29 and DPR-30, and (3) the Commission's concurrently issued related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Moline Public Library, 504 - 17th Street, Moline, Illinois 60625. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this OCT 2 8 1976

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed by:
Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors