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MAR 12 1976

Docket No. 50-254

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 July 1, 1975 and December 23, 1975 and supplements thereto dated July 7, 1975, September 19, 1975, November 6, 1975, February 4, 1976, February 6, 1976, and March 5, 1976, the Commission has issued the enclosed Amendment No. 25 to Facility Operating License No. DPR-29 for the Quad Cities Nuclear Power Station Unit 1.

This amendment (1) authorizes operation with additional 8 x 8 fuel assemblies and (2) incorporates operating limits in the Technical Specifications for the facility based on an acceptable evaluation model that conforms with the requirements of Section 50.46 of 10 CFR Part 50.

Copies of the related Safety Evaluation, the Negative Declaration, Environmental Impact Appraisal, and the Federal Register Notice are also enclosed.

Sincerely,

Original signed by
Richard D. Silver for
Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

Enclosures:
1. Amendment No. 25 to DPR-29
2. Safety Evaluation
3. Negative Declaration with Supporting Environmental Impact Appraisal
4. Federal Register Notice

I notified CECo (Gary Abrell) at 4:45 pm on March 12, 1975

OFFICE ➤	OR:ORB #2	OR:ORB #2	OELD	OR:ORB #2	OR:AD/OR
SURNAME ➤	RMDiggs	PO'Connor:ro		DLZiemann	KRGoller
DATE ➤	3/12/76	3/12/76	3/12/76	3/12/76	3/12/76

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

MAR 1 2 1976

cc w/enclosures:
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President and Chairman
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Mr. Robert W. Watts, Chairman
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Rock Island Country Courthouse
Rock Island, Illinois 61201

cc w/enclosures and cy of CECo
filings dtd. 7/7/75, 9/19/75,
11/6/75, 12/23/75, 2/4&6/76 & 3/5/76:
Mr. Leroy Stratton
Bureau of Radiological Health
Illinois Department of Public Health
Springfield, Illinois 62706

COMMONWEALTH EDISON COMPANY
AND
IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

DOCKET NO. 50-254

QUAD CITIES UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 25
License No. DPR-29

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

   A. The applications for amendment by the Commonwealth Edison Company (the licensee) dated July 1, 1975 and December 23, 1975 and supplements thereto dated July 7, 1975, September 19, 1975, November 6, 1975, February 4, 1976, February 6, 1976, and March 5, 1976, comply 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;

   E. The Negative Declaration and Environmental Impact Appraisal are necessary and have been prepared in connection with the issuance of this amendment.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment and paragraphs 3.C and 3.D of Facility License No. DPR-29 are hereby amended and added (respectively) to read as follows:

OFFICE ➤						
SURNAME ➤						
DATE ➤						

3.C Restrictions

Beyond the point in the fuel cycle at which the reactivity reduction rate during a scram is less than that of Curve B in Figure 1 of "Supplement B to Dresden Station Special Report 29," dated March 29, 1974, operation of the reactor shall not exceed the core thermal power versus flow conditions defined by the "Nominal Expected 90% Flow Control Line" on Figure 2.1-3 of the Commonwealth Edison letter (J. S. Abel to Benard C. Rusche) dated June 24, 1975.

Beyond the point in the fuel cycle at which the reactivity reduction rate during a scram is less than that of end-of-cycle curve on Figure 1-1 of the Commonwealth Edison letter (J. S. Abel to D. L. Ziemann) dated February 27, 1975, operation of the reactor is not authorized.

D. Equalizer Valve Restriction

The valves in the equalizer piping between the recirculation loops shall be closed at all times during reactor operation.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by:
Karl R. Goller

Karl R. Goller, Assistant Director
for Operating Reactors
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: MAR 12 1976

OFFICE ➤						
SURNAME ➤						
DATE ➤						

ATTACHMENT TO LICENSE AMENDMENT NO. 25

FACILITY OPERATING LICENSE NO. DPR-29

DOCKET NO. 50-254

The following changes relate to the Appendix A portion of the Quad Cities Technical Specifications. Changed areas on the revised pages are shown by a marginal line.

Remove Pages	Insert Pages
100 and 101	100, 100A, 101, and 101A
105	105 and 105A
105A and 105A-1	105A-1, 105A-2, and 105A-3
105B	105B,and 105B-1 and 105B-2
105C	105C and 105C-1

Note: Pages 100, 101, 105, 105B and 105C have been reissued to reflect that they now only apply to Unit 2. There are no changes in the Unit 2 Specifications as a result of the issuance of this amendment for Unit 1.

OFFICE ➤						
SURNAME ➤						
DATE ➤						

3.5 LIMITING CONDITION FOR OPERATION

D. Automatic Pressure Relief Subsystems

1. The Automatic Pressure Relief Subsystem shall be operable whenever the reactor pressure is greater than 90 psig, irradiated fuel is in the reactor vessel and prior to reactor startup from a Cold Condition.

17 & 24 2. From and after the date that one of the five relief valves of the automatic pressure relief subsystem is made or found to be inoperable when the reactor is pressurized above 90 psig with irradiated fuel in the reactor vessel, continued reactor operation is permissible only during the succeeding thirty days unless repairs are made and provided that during such time the HPCI subsystem is operable.

4.5 SURVEILLANCE REQUIREMENT

D. Automatic Pressure Relief Subsystems

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

1. During each operating cycle the following shall be performed:

   a. A simulated automatic initiation which opens all pilot valves, and

   b. With the reactor at low pressure each relief valve shall be manually opened until thermocouples downstream of the valve indicate fluid is flowing from the valve.

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

2. When it is determined that one relief valve of the automatic pressure relief subsystem is inoperable, the HPCI shall be demonstrated to be operable immediately and weekly thereafter.

3.5 LIMITING CONDITION FOR OPERATION

D. Automatic Pressure Relief Subsystems

1. The Automatic Pressure Relief Subsystem shall be operable whenever the reactor pressure is greater than 90 psig, irradiated fuel is in the reactor vessel and prior to reactor startup from a Cold Condition.

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

4.5 SURVEILLANCE REQUIREMENT

D. Automatic Pressure Relief Subsystems

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

1. During each operating cycle the following shall be performed:

   a. A simulated automatic initiation which opens all pilot valves, and

   b. With the reactor at low pressure each relief valve shall be manually opened until thermocouples downstream of the valve indicate fluid is flowing from the valve.

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

2. When it is determined that one relief valve of the automatic pressure relief subsystem is inoperable, the HPCI shall be demonstrated to be operable immediately and weekly thereafter.

3.5 LIMITING CONDITION FOR OPERATION	4.5 SURVEILLANCE REQUIREMENT
3. From and after the date that more than one of five electromatic relief valves of the automatic pressure relief subsystem are made or found to be inoperable when the reactor is pressurized above 90 psig with irradiated fuel in the reactor vessel, continued reactor operation is permissible only during the succeeding 24 hours unless repairs are made and provided that during such time, the HPCI subsystem is operable.	3. When it is determined that more than one electromatic relief valve of the automatic pressure relief subsystem is inoperable, the HPCI subsystem shall be demonstrated to be operable immediately.
3 4. If the requirements of Specification 3.5.D cannot be met, an orderly shutdown shall be initiated and the reactor pressure shall be reduced to 90 psig within 24 hours.	
E. Reactor Core Isolation Cooling System	4 E. Reactor Core Isolation Cooling System Surveillance of the RCIC System shall be performed as follows:
1. The RCIC system shall be operable whenever the reactor pressure is greater than 150 psig, irradiated fuel is in the reactor vessel, and prior to startup from a Cold Condition.	1. RCIC system testing shall be as specified in Specification 4.5.A.1.a, b, c, and d, except that the RCIC pump shall deliver at least 400 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 run during each refueling outage.

3.5 LIMITING CONDITION FOR OPERATION	4.5 SURVEILLANCE REQUIREMENT
3. If the requirements of Specification 3.5.D cannot be met, an orderly shutdown shall be initiated and the reactor pressure shall be reduced to 90 psig within 24 hours.	
E. Reactor Core Isolation Cooling System	E. Reactor Core Isolation Cooling System
	Surveillance of the RCIC System shall be performed as follows:
1. The RCIC system shall be operable whenever the reactor pressure is greater than 150 psig, irradiated fuel is in the reactor vessel, and prior to startup from a Cold Condition.	1. RCIC system testing shall be as specified in Specification 4.5.A.1.a, b, c, and d, except that the RCIC pump shall deliver at least 400 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 run during each refueling outage.

3.5 LIMITING CONDITION FOR OPERATION

3. If Specification 3.5.H.1 and 2 cannot be met, reactor startup shall not commence or if operating, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

I. Average Planar LHGR

During steady state power operation, the average linear heat generation rate (LHGR) of all the rods in any fuel assembly, as a function of average planar exposure, at any axial location, shall not exceed the maximum average planar LHGR shown in Figure 3.5.1 and 3.5.1A.

4.5 SURVEILLANCE REQUIREMENT

c. The RHR service water pump and diesel generator cooling water pump bed plate drains shall be checked during each operating cycle by assuring that water can be run through the drain lines and actuating the air operated valves by operation of the following sensors:

   i. loss of air

   ii. equipment drain sump high level

   iii. vault high level

d. The condenser pit 5 foot trip circuits for each channel shall be checked once a month. A logic system functional test shall be performed during each refueling outage.

I. Average Planar LHGR

Daily during reactor power operation, the average planar LHGR shall be checked.

3.5 LIMITING CONDITION FOR OPERATION	4.5 SURVEILLANCE REQUIREMENT

3. If Specification 3.5.H.1 and 2 cannot be met, reactor startup shall not commence or if operating, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

c. The RHR service water pump and diesel generator cooling water pump bed plate drains shall be checked during each operating cycle by assuring that water can be run through the drain lines and actuating the air operated valves by operation of the following sensors:

i. loss of air
ii. equipment drain sump high level
iii. vault high level

d. The condenser pit 5 foot trip circuits for each channel shall be checked once a month. A logic system functional test shall be performed during each refueling outage.

I. Average Planar Linear Heat Generation Rate (APLHGR)

During steady state power operation, the average planar linear heat generation rate (APLHGR) of all the rods in any fuel assembly, as a function of average planar exposure, at any axial location, shall not exceed the maximum average planar LHGR shown in Figures 3.5.1, 3.5.1A and 3.5.1B. If at any time during operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

I. Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR for each type of fuel as a function of average planar exposure shall be determined daily during reactor operation at $\geq$ 25% rated thermal power.

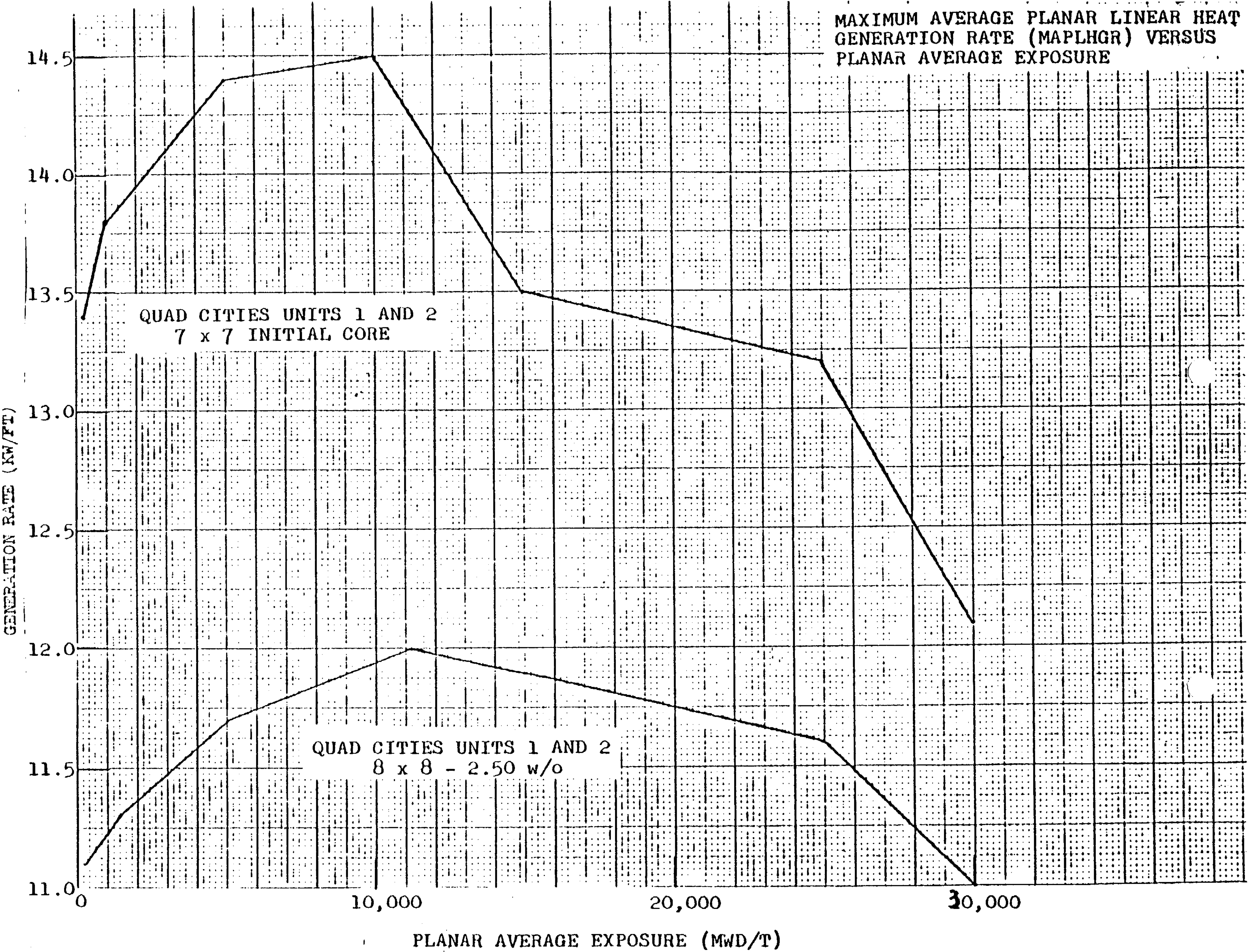
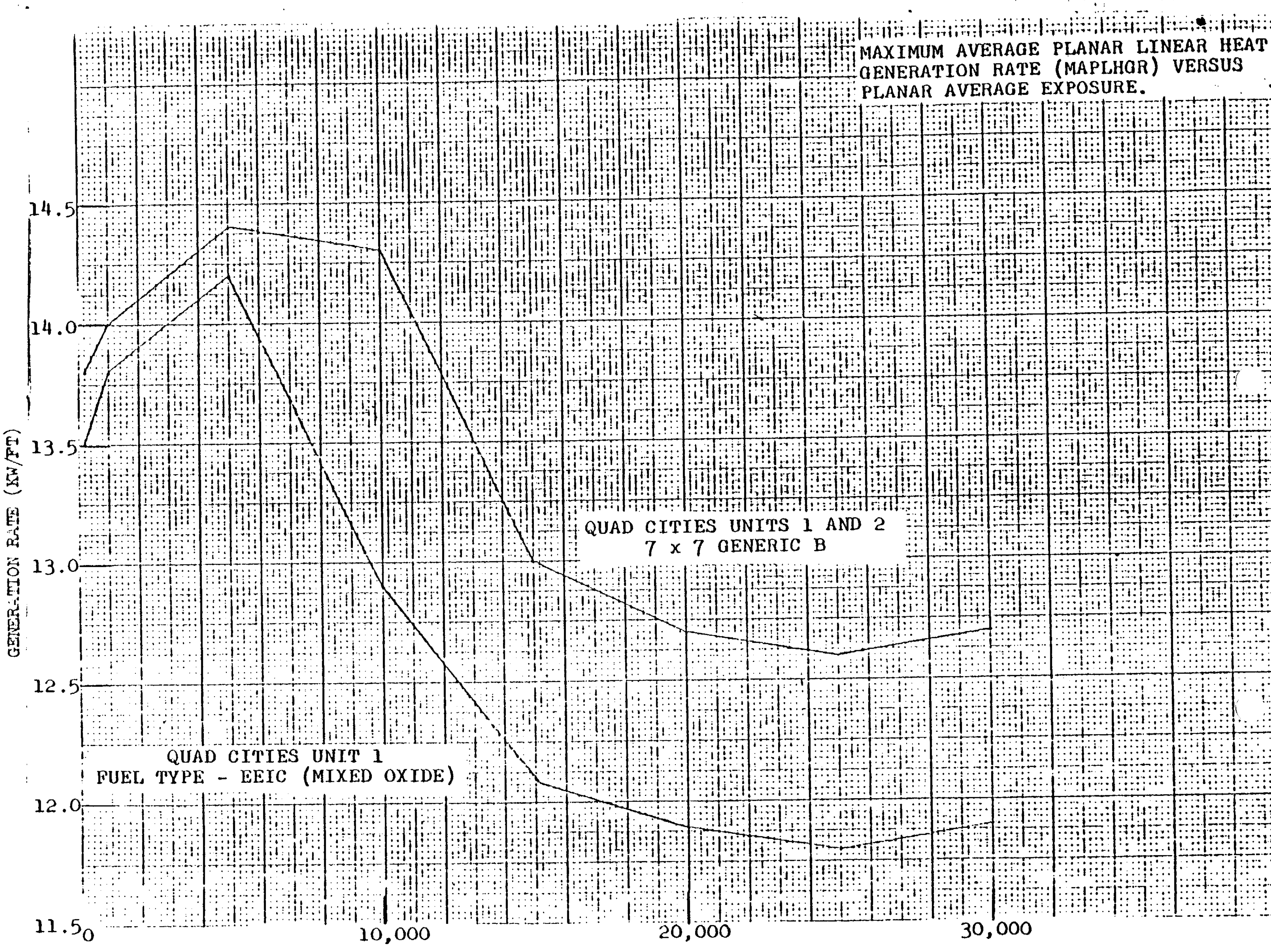
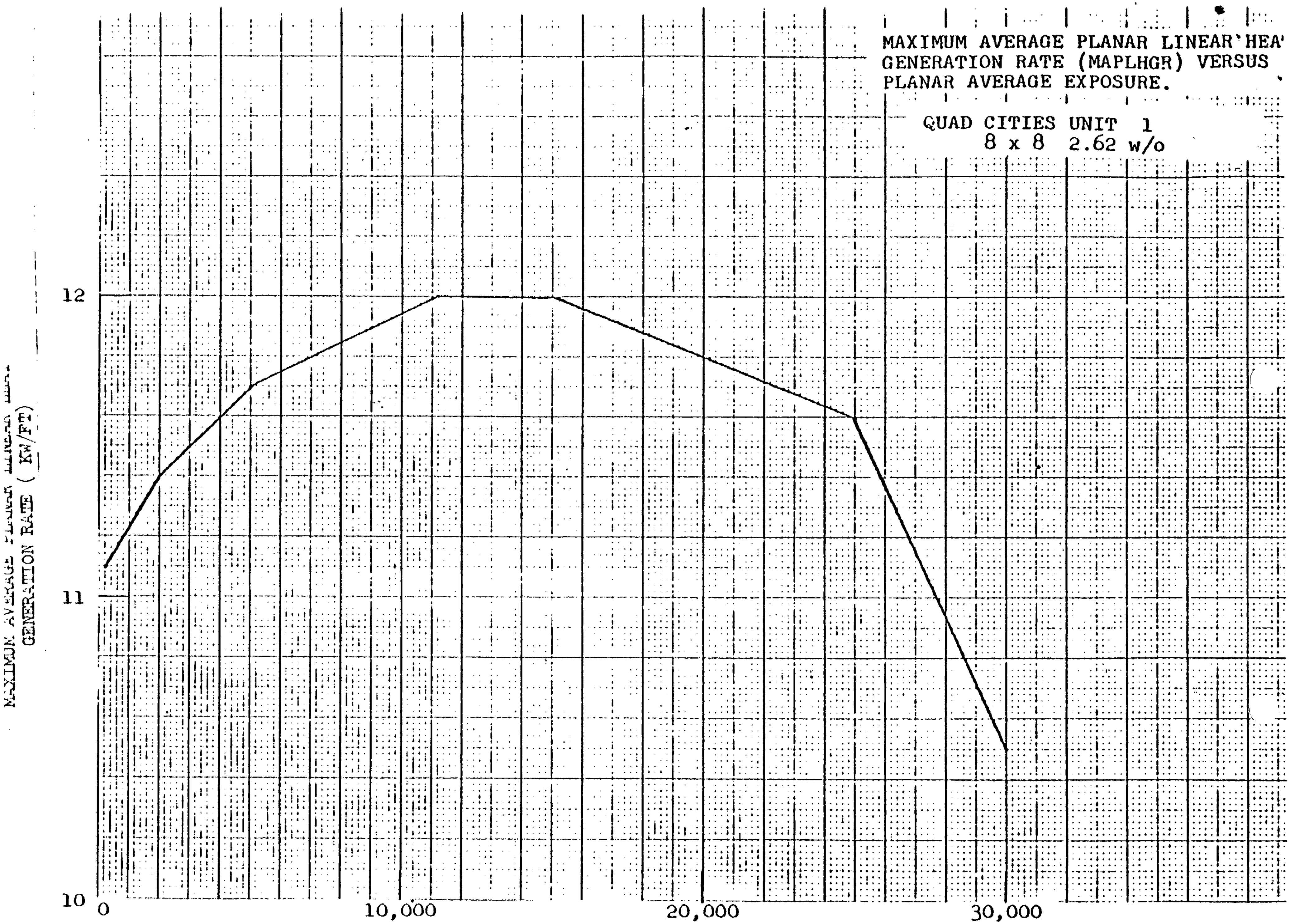

MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS PLANAR AVERAGE EXPOSURE
QUAD CITIES UNITS 1 AND 2
7 x 7 INITIAL CORE
QUAD CITIES UNITS 1 AND 2
8 x 8 - 2.50 w/o
GENERATION RATE (KW/FT)
14.5
14.0
13.5
13.0
12.5
12.0
11.5
11.0
0
10,000
20,000
30,000
PLANAR AVERAGE EXPOSURE (MWD/T)

FIGURE 3.5.1



MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS PLANAR AVERAGE EXPOSURE.
QUAD CITIES UNITS 1 AND 2
7 x 7 GENERIC B
QUAD CITIES UNIT 1
FUEL TYPE - EEIC (MIXED OXIDE)
GENERATION RATE (KW/FT)
14.5
14.0
13.5
13.0
12.5
12.0
11.5
0
10,000
20,000
30,000
PLANAR AVERAGE EXPOSURE (MWD/T)

FIGURE 3.5.1.A



PLANAR AVERAGE EXPOSURE (MWD/T)

FIGURE 3.5.1.B

3.5 LIMITING CONDITIONS FOR OPERATION

J. Local LHGR

During steady state power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed the maximum allowable LHGR as calculated by the following equation.

$$LHGR_{max} \leq LHGR_d \left[1 - \left(\frac{\Delta P}{P}\right)_{max} \left(\frac{L}{L_T}\right)\right]$$

$LHGR_d$ = Design LHGR

= 17.5 kw/ft, 7X7 fuel assemblies

= 13.4 kw/ft, 8X8 fuel assemblies

$\left(\frac{\Delta P}{P}\right)_{max}$ = Maximum power spiking penalty

= .035 initial core fuel

= .029 reload 1, 7X7 fuel

26 = .022 reload, 8X8 fuel

= .028 reload 1, mixed oxide fuel

L_T = Total Core Length

= 12 ft

L = Axial distance from bottom of core

4.5 SURVEILLANCE REQUIREMENTS

J. Local LHGR

26 Daily during steady state power operation above 25 per cent of rated thermal power, the Local LHGR shall be checked.

J. Local LHGR

During steady state power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed the maximum allowable LHGR as calculated by the following equation. If at any time during operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

$$LHGR_{max} \leq LHGR_d \left[1 - \left(\frac{\Delta P}{P}\right)_{max} \left(\frac{L}{L_T}\right)\right]$$

$LHGR_d$ = Design LHGR

= 17.5 kw/ft, 7X7 fuel assemblies

= 13.4 kw/ft, 8X8 fuel assemblies

$\left(\frac{\Delta P}{P}\right)_{max}$ = Maximum power spiking penalty

= .035 initial core fuel

= .029 reload 1, 7X7 fuel

= .022 reload, 8X8 fuel

= .028 reload 1, mixed oxide fuel

L_T = Total Core Length

= 12 ft

L = Axial distance from bottom of core

J. Local LHGR

Daily during steady state power operation above 25 per cent of rated thermal power, the Local LHGR shall be checked.

3.5 LIMITING CONDITION FOR OPERATION	4.5 SURVEILLANCE REQUIREMENTS
K. <u>Minimum Critical Power Ratio (MCPR)</u> During steady state operation MCPR shall be greater than or equal to - 1.29 (7X7 fuel) 1.35 (8X8 fuel) at rated power and flow. For core flows other than rated, these nominal values of MCPR shall be increased by a factor of K_f, where K_f is as shown in Figure 3.5-2.	K. <u>Minimum Critical Power Ratio (MCPR)</u> The MCPR shall be determined daily during steady state power operation above 25% of rated thermal power. 26

3.5 LIMITING CONDITION FOR OPERATION

K. Minimum Critical Power Ratio (MCPR)

During steady state operation MCPR shall be greater than or equal to -

1.29 (7X7 fuel)
1.35 (8X8 fuel)

at rated power and flow. If at any time during operation it is determined by normal surveillance that the limiting value for MCPR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the steady state MCPR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits. For core flows other than rated, these nominal values of MCPR shall be increased by a factor of K_f, where K_f is as shown in Figure 3.5-2.

4.5 SURVEILLANCE REQUIREMENTS

K. Minimum Critical Power Ratio (MCPR)

The MCPR shall be determined daily during steady state power operation above 25% of rated thermal power. 26

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 25 TO FACILITY LICENSE NO. DPR-29

COMMONWEALTH EDISON COMPANY
AND
IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

QUAD CITIES NUCLEAR POWER STATION UNIT 1

DOCKET NO. 50-254

1.0 INTRODUCTION

Commonwealth Edison has proposed to operate Quad Cities Unit 1:

(1) with additional 8 x 8 fuel assemblies (Reload-2), as requested in their application dated December 23, 1975, and supplements dated February 4, 1976, February 6, 1976, and March 5, 1976; and

(2) using modified operating limits based on an acceptable emergency core cooling system evaluation model that conforms with Section 50.46 of 10 CFR Part 50, as requested in their application dated July 1, 1975, and supplements dated July 7, 1975, September 19, 1975, and November 6, 1975.

2.0 RELOAD

2.1 DISCUSSION

The reference core loading for Quad Cities 1 Reload-2 consists of 500 initial 7 x 7 fuel assemblies, 22 Reload-1 7 x 7 assemblies, 36 Reload-1 8 x 8 fuel assemblies, 1 segmented test assembly, 5 mixed oxide fuel assemblies, and 160 Reload-2 8 x 8 fuel assemblies. The reload assemblies are scatter loaded throughout the core. The acceptability of the neutronic, thermal-hydraulic, and mechanical design of 8 x 8 fuel assemblies during normal operation, operational transients and postulated accidents was evaluated by the NRC staff in a previous report[1/]. The use of

1/ Technical Report on the General Electric Company 8 x 8 Fuel Assembly, dated February 5, 1974, by the Directorate of Licensing.

8 x 8 fuel assemblies for reloads was also reviewed by the Advisory Committee on Reactor Safeguards and discussed in its report dated February 12, 1974[2/]. The use of 8 x 8 reload fuel assemblies in Quad Cities 1 was evaluated and approved by Amendment No. 10 to Facility Operating License No. DPR-29 dated June 5, 1974.

With two exceptions, the evaluations of the acceptability of the reload fuel for the Quad Cities Unit 1 Reload-1 core are applicable to the Reload-2 fuel. A design change for this reload 8 x 8 fuel is the use of leaf springs to minimize the bypass flow area between the fuel assembly shroud and the lower end fitting. Another change is the use of fuel with a slightly higher enrichment for 8 x 8 fuel than previously evaluated for Quad Cities 1.

Our safety evaluation of this reload (Reload No. 2) for the Quad Cities Unit 1 core is based on the licensee's application as amended, and on information contained in a GE topical report, NEDO-20360[3/] referred to in the application. The NEDO-20360 report is still being evaluated by the staff for use as a topical. Our use of that report in this analysis was limited to considerations applicable to Quad Cities 1 and does not imply acceptability of its use for other facilities.

2.2 EVALUATION

2.2.1. NUCLEAR CHARACTERISTICS

The information presented in the licensing submittal for the reconstituted core[4/][5/] closely follows the guidelines of Appendix A of Reference 3. Up to 160 8 x 8 reload fuel bundles will be loaded throughout the core. As many as 108 of these reload fuel bundles will have an average enrichment of 2.50% by weight of the uranium-235 isotope while the remainder, as

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2/ Report on General Electric 8 x 8 Fuel Design for Reload Use, Advisory Committee on Reactor Safeguards, February 12, 1974.

3/ General Electric Boiling Water Reactor Generic Reload Application for 8 x 8 Fuel, NEDO-20360 Supplement 2 (May 1975).

4/ General Electric BWR Reload-2 Licensing Submittal for Quad Cities Unit 1 Nuclear Power Station - NEDO 20974, December 1975.

5/ Quad Cities 1 Reload-2 Licensing Submittal (NEDO 20974), Supplement A, February 6, 1976.

many as 52 fuel bundles, will have an average enrichment of 2.62%. The core contains a total of 724 fuel bundles. Thus, about 22 percent of the fuel bundles are being replaced for this reload. Previously, for Reload-1 36 (2.50% average enrichment 8 x 8 fuel bundles) had been loaded. The Reload-2 loading pattern may be described as follows: (1) the two rows and columns of fuel bundles which intersect at the center of the core will not contain any Reload-2 fuel, (2) the lower enrichment reload bundles are loaded in the interior of the core while the higher enrichment reload bundles are loaded near the outer periphery of the core, (3) in the core interior only one fuel bundle in a four bundle array surrounding a control rod will be replaced, (4) near the core periphery two diagonally located fuel bundles of the four bundle array surrounding a control rod will be replaced, and (5) some initial fuel bundles will be shuffled. The 8 x 8 reload fuel assemblies in the Reload-2 core are, therefore, basically scatter loaded. The data in Reference 1 indicate that the nuclear characteristics of the Reload-2 8 x 8 fuel bundles are similar to those previously loaded. Thus, the temperature, void dependent behavior of the reconstituted core and the total control system worth will not differ significantly from those values which were previously analyzed and approved for Quad Cities Unit 1.

The shutdown margin of the reconstituted core meets the Technical Specification requirement that the core be at least 0.25% Δ k subcritical in the most reactive operating state with the largest worth control rod fully withdrawn and with all other control rods fully inserted. A minimum shutdown margin of 1.0% Δ k, with one rod fully withdrawn, exists for the Reload-2 cycle. This shutdown margin was calculated for a core average exposure of 10,600 MWd/t at the end of the cycle 2.

The information presented in the application indicates that a boron concentration of 600 ppm in the moderator will make the reactor subcritical by at least 0.03 Δ k at 20°C, xenon free. Therefore, the alternate shutdown requirement of the General Design Criteria is met by the Standby Liquid Control System.

The Technical Specification requirement for the storage of fuel for Quad Cities Unit 1 is that the effective multiplication factor, k_{eff}, of the fuel as stored in the fuel storage rack is equal to or less than 0.90 for normal conditions. This is achieved if the uncontrolled k_∞ of a single fuel bundle is less than 1.30[3] at 65°C. The 8 x 8 8D250 and 8D262 fuel bundles, at both the zero exposure and the peak reactivity point, have a k_∞ less than 1.26 and, therefore, meet the fuel storage requirement for Quad Cities Unit 1 and are acceptable.

The full power scram reactivity curves used for the Reload-2 cycle are the GE generic "B" curve and the end-of-cycle curve shown in Figure 1[6]. This end-of-cycle curve in Reference 6 is slightly more conservative than the predicted end-of-cycle curve for Reload-2. The "B" scram curve is applicable to the Reload-2 cycle for the first 2200 MWd/t of exposure while the end-of-cycle curve of Reference 6 is applicable for the remainder of the cycle. These scram curves are multiplied by a design conservatism factor of 0.8 for use in the anticipated transient analyses.

The void and Doppler coefficients of reactivity for the Reload-2 cycle are given in Table 5-1 of Reference 4. The void coefficient of reactivity at the core average void fraction of 33 percent varies from -10.4 to -11.4×10^{-4} $\Delta k/k/\Delta\%V$. The Doppler coefficient of reactivity at a fuel temperature of 650°C varies from -1.168 to -1.229×10^{-5} $\Delta k/k/\Delta T$. Also the effective delayed neutron fraction varies from 0.00547 to 0.00608 over the fuel cycle.

2.2.1.1. CONCLUSION

Thus, based on our review of the information presented in the Quad Cities Unit 1 licensing submittal, and the generic 8 x 8 reload report (Reference 3), we conclude that the nuclear characteristics (e.g., scram reactivity, void coefficient of reactivity and Doppler coefficient of reactivity) and performance of the reconstituted core for the Reload-2 cycle will not differ significantly from previously analyzed and approved Quad Cities Unit 1 fuel cycles and are acceptable.

6/ Commonwealth Edison Letter (Abel) to NRC (Skovholt), "Dresden Station Special Report No. 29, Supplement B - Dresden Station Unit 3 Transient Analyses for Cycle 3 and Quad Cities Unit 1 Cycle 2, NRC Dockets 50-249 and 50-254," dated March 29, 1974.

2.2.2. MECHANICAL DESIGN

Mechanical and operating parameters for the 8 x 8 assemblies are compared to the 7 x 7 assemblies in Table I. The small diameter rods, with lower linear heat generation rate and increased cladding thickness/diameter ratio for the 8 x 8 fuel design as compared to the 7 x 7 fuel assemblies, result in increased safety margins with respect to maximum design linear heat generation rate. In addition, the 8 x 8 Reload-2 fuel incorporates finger springs in 12 bundles for controlling moderator/coolant bypass flow at the interface of the channel and fuel bundle lower tie plate. This device has been used satisfactorily in General Electric's initial core and reload fuel for all BWR-4/5 plants, and for several BWR-3 plants. The finger springs employed in Quad Cities 1 Reload-2 fuel are identical in design to those that have been used previously on Dresden Units 2 and 3, and Quad Cities Unit 2 which are similar plants. Inspection of more than 900 fuel assemblies in operating plants employing finger springs has not revealed any problems related to their use.

2.2.2.1. CONCLUSION

On the basis of our review of the generic 8 x 8 reload report, current operating experience with the 8 x 8 reload design in similar plants, and our review of Commonwealth Edison's Reload-2 licensing submittal, we conclude that the Quad Cities Unit 1 Reload-2 mechanical design is acceptable.

2.2.3. THERMAL-HYDRAULICS

The GE generic 8 x 8 fuel reload topical report[3/] and GETAB report[7/] are referenced to provide the description of the thermal-hydraulic methods which were used to calculate the thermal margins. Application of GETAB involves:

1) establishing the fuel damage safety limit,

2) establishing limiting conditions of operation such that the safety limit is not exceeded for normal operation and anticipated transients, and

7/ "General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application," "NEDO - 10958, 73NED9, Class I, November 1973.

TABLE 1

COMPARISON OF PARAMETERS FOR 8 x 8 AND 7 x 7
ROD FUEL ASSEMBLY DESIGN

	7 x 7	8 x 8
Pellet Outside Diameter (in.)	0.477	0.416
Rod Outside Diameter (in.)	0.563	0.493
Rod-to-Rod Pitch (in.)	0.738	0.640
Water-Fuel Ratio (cold)	2.53	2.60
U Bundle Weight (pounds)	412.8	404.6
Cladding Thickness (mils)	37	34
Active Fuel Length (in.)	144	144

3) establishing limiting conditions for operation such that the initial conditions assumed in the accident analyses are satisfied.

We have evaluated and report herein the Quad Cities Unit 1 Cycle 3 (Reload-2) developed thermal margins based on the GETAB report[7] and plant specific input information provided by the licensee.

2.2.3.1. FUEL CLADDING INTEGRITY SAFETY LIMIT MINIMUM CRITICAL POWER RATIO (MCPR)

A critical power ratio (CPR) is defined as the ratio of that assembly power which causes some point in the assembly to experience transition boiling to the assembly power at the reactor condition of interest. The minimum critical power ratio (MCPR) is the critical power ratio corresponding to the most limiting fuel assembly in the core.

The fuel cladding integrity safety limit MCPR is 1.06. It is based on the GETAB statistical analysis which assures that 99.9% of the fuel rods in the core are not expected to experience boiling transition during abnormal operational transients. The uncertainties in the core and system operating parameters and the GEXL correlation (Table 4-1 of the licensee submittal[4]), combined with the relative bundle power distribution in the core form the basis for the GETAB statistical determination of the safety limit MCPR. In comparing the tabulated list of uncertainties for Quad Cities 1 and those reported in the GETAB[7/8] analyses we have found only one difference. The standard deviation for the TIP readings uncertainty for the subject reload is 8.7% whereas the GETAB NEDO - 10958 report shows 6.3%. The increase in uncertainty for the subject reload is a consequence of the increase in uncertainty in the measurement of power in a reload core. A TIP uncertainty of 6.3% would be applicable if this were the initial core. In both cases the TIP reading uncertainties are based on a symmetrical planar power distribution.

The generic core selected for the GETAB statistical analysis is a typical 251/764 core while the Quad Cities Unit 1 is a 251/724 core. The generic GETAB statistical analysis results are conservative since the bundle power distribution used for the GETAB application has more high power bundles than the distribution expected during the third cycle of operation of the Quad Cities Unit 1 reactor. This results in a conservative value of the MCPR which meets the 99.9% criterion. We conclude that the proposed fuel integrity safety limit, a MCPR of 1.06, is acceptable for Quad Cities Unit 1 Fuel Cycle 3 (Reload-2).

8/ General Electric letter (Hinds) to AEC (Butler) "Responses to the Third Set of AEC Questions on the General Electric Licensing Topical Reports, NEDO - 10558 and NEDO - 10958, "General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application," dated July 24, 1974.

2.2.3.2. OPERATING LIMIT MCPR

Various transient events will reduce the MCPR below the operating MCPR. To assure that the fuel cladding integrity safety limit (MCPR of 1.06) is not exceeded during anticipated abnormal operational transients, the most limiting transients have been analyzed to determine which one results in the largest reduction in critical power ratio (Δ CPR). The licensee has submitted the results of analyses of those transients which produce a significant decrease in MCPR (Reference 4 and 5). The types of transients evaluated were overpressure, feedwater temperature decrease, coolant flow increase, etc. The most limiting transients in these categories were the turbine trip without bypass assuming end of cycle (EOC) scram reactivity insertion rates (90% of rated power, 100% of rated flow). The turbine trip transient results in ΔMCPR's of 0.23 (7 x 7 fuel) and 0.29 (8 x 8 fuel). Addition of these ΔMCPR's to the safety limit MCPR of 1.06 gives the minimum operating limit MCPR for each fuel type required to avoid violation of the safety limit, should this limiting transient occur. Therefore, the operating limit MCPR's are 1.29 for 7 x 7 fuel and 1.35 for 8 x 8 fuel.

The calculated change in MCPR for the second most severe abnormal transient, the loss of feedwater heating, is 0.15 for 7 x 7 fuel and 0.17 for 8 x 8 fuel.

The transient analyses were evaluated with scram reactivity insertion rates that included a design conservatism factor of 0.80. The design conservatism factor for the void coefficient used was 1.33 and the design conservatism factor for Doppler coefficient was 0.90. The initial conditions[4/] and the design conservatism factors used for the worst operational transient are acceptable. The initial MCPR assumed in the transient analyses was equal to or greater than the established operating limit MCPR of 1.29 and 1.35 for 7 x 7 and 8 x 8 fuel assemblies respectively. This results in a conservative ΔMCPR and is acceptable.

A GE study[7/] has shown that the required operating MCPR varies with the axial and local power peaking distribution. Axial peaking in the middle or upper portion of the core results in higher required MCPR's than peaking in the lower portion of the core. In the analyses the axial power peak was assumed to be representative of beginning-of-cycle conditions and to be located in the upper portion of the core.

The R-factors, which are a function of the local power peaking, assumed in the analyses are also representative of beginning-of-cycle conditions. The values assumed are 1.075 for 7 x 7 fuel and 1.102 for 8 x 8 fuel. During the cycle the local peaking and therefore the R-factor is reduced while the peak in the axial shape moves toward the bottom of the core. Although the operating limit MCPR would be increased by approximately 1% by the reduced end-of-cycle R-factor, this is offset by the reduction in MCPR resulting from the relocation of the axial peak to below the midplane.

Conservatism was applied in the determination of the required operating limit MCPR because the assumed axial and local peaking were representative of the beginning of the fuel cycle. This is the worst consistent set of axial and local peaking.

It is concluded from the analyses of the limiting pressure transient, a generator load rejection with bypass failure, that Quad Cities Unit 1, Reload-2 can operate at 100% power until that point in the fuel cycle (approximately 2200 MWd/t into the cycle) when the scram reactivity is less than that of the B curve in Figure 1 of Reference 5. The power will then be limited to 90% of rated power at 100% of rated flow for the remainder of the cycle. (The flow control line is shown on the power/flow map appearing in Figure 1 of Reference 9). Since the transient and safety analyses with a reduced scram reactivity insertion rate are based on the power/flow line defined by the 90% power/100% flow, operation above this line could result in calculated transients that violate the MCPR and pressure safety limits. Therefore in accordance with the licensee's proposal, Reference 4, operation is restricted to power/flow conditions along or below this derated flow control line which is consistent with the rod patterns necessary to give the derated power levels at 100% flow.

Analyses have shown that the operating limit MCPR's of 1.29 for 7 x 7 fuel and 1.35 for 8 x 8 fuel assure that the fuel cladding integrity safety limit is not exceeded during anticipated abnormal operational transients. Hence we conclude that the operating limit MCPR's of 1.29 for 7 x 7 fuel and 1.35 for 8 x 8 fuel are acceptable.

9/ Commonwealth Edison letter (Abel) to NRC (Rusche) "Quad Cities Units 1 and 2 Proposed Change to Facility Operating Licenses DPR-29 and 30. NRC Docket 50-254 and 265," dated June 24, 1975.

2.2.3.3. ROD WITHDRAWAL ERROR

The rod withdrawal error transient is discussed in Reference 4 in terms of worst case conditions. Assumptions and descriptions of the rod withdrawal event are given in Reference 3. The information in these two references indicates that the local power range monitor subsystem (LPRM's) will detect high local powers and alarm. However, if the operator ignores the LPRM alarm, the rod block monitor subsystem (RBM) will stop the rod withdrawal while the critical power ratio is still equal to or greater than the 1.06 MCPR safety limit and the cladding is under the one percent plastic strain limit. This rod withdrawal error transient is not limiting for the Quad Cities Unit 1 Reload-2 cycle with the RBM setting at 110% of its initial level. We conclude that the analysis performed for this localized transient and the consequences of this localized transient are acceptable.

2.2.3.4. OPERATING MCPR LIMITS FOR LESS THAN RATED POWER AND FLOW

For the limiting transient of recirculation pump speed control failure at lower than rated power and flow condition, the licensee will conform to Technical Specification limiting conditions for operation; (Figure 3.5-2 - page 105 D of the Technical Specifications). This requires that for core flows less than the rated flow, the licensee maintain the MCPR greater than the operating minimum values (1.29 for 7 x 7 fuel and 1.35 for 8 x 8 fuel). The minimum MCPR values for less than rated flow are the rated flow values multiplied by the respective K_f factors appearing in Figure 3.5-2 of the Technical Specifications. The K_f factor curves were generically derived and assure that the most limiting transient occurring at less than rated flow will not exceed the safety limit MCPR of 1.06. We conclude that the calculated consequences of the anticipated abnormal transients initiated at less than rated flow and power do not violate the thermal and plastic strain limits of the fuel or the pressure limits of the reactor coolant boundary.

2.2.3.5. CONCLUSION

Based upon the above, we conclude that the analyses and operating limits based upon the use of the General Electric Thermal Analysis Basis have been conservatively applied to Reload-2 (Cycle 3) and are acceptable.

3.0 ACCIDENT ANALYSIS

3.1. ECCS APPENDIX K ANALYSIS

On December 27, 1974, the Atomic Energy Commission issued an Order for Modification of License implementing the requirements of 10 CFR 50.46 "Acceptance Criteria and Emergency Core Cooling Systems for Light Water Nuclear Power Reactors." One of the requirements of the Order was that prior to any license amendment authorizing any core reloading "...the licensee shall submit a re-evaluation of ECCS cooling performance calculated in accordance with an acceptable evaluation model which conforms to the provisions of 10 CFR Part 50, 50.46." The order also required that the evaluation shall be accompanied by such proposed changes in Technical Specifications or license amendments as may be necessary to implement the evaluation results.

On February 4, 1976[10/], Commonwealth Edison proposed an amendment to the facility operating license, requesting changes to the Technical Specifications for Quad Cities Unit 1 to implement the results of their evaluation of the ECCS (References 11, 12 13). These analyses showed compliance to the 10 CFR 50.46 criteria and Appendix K to 10 CFR Part 50.

The Order for Modification of License issued December 27, 1974, stated that evaluation of ECCS cooling performance may be based on the vendor's evaluation model as modified in accordance with the changes described in the staff Safety Evaluation Report of Quad Cities Unit 1 dated December 27, 1974.

The background of the staff review of the General Electric ECCS model and its application to Quad Cities Unit 1 is described in the December 27, 1974 SER. The bases for acceptance of the principal portions of the evaluation model are set forth in the staff's Status Report of October 1974 which are referenced in the December 27, 1974 SER. The December 27, 1974 SER also describes the various changes required in the earlier GE evaluation model.

10/ Commonwealth Edison Letter (Bolger) to NRC (Rusche), "Quad Cities Station Unit 1 Proposed Amendment to Facility Operating License DPR-29, NRC Docket No. 50-254," dated February 4, 1976.

11/ Commonwealth Edison Letter (Abel) to NRC (Ziemann), "Quad Cities Station Unit 2, Special Report No. 15, Supplement C, NRC Docket No. 50-265," dated April 18, 1975.

12/ Commonwealth Edison Letter (Abel) to NRC (Ziemann), "Quad Cities Station Unit 2, Special Report No. 15, Supplement C, NRC Docket No. 50-265," (Revision), dated April 21, 1975.

13/ Commonwealth Edison Letter (Bolger) to NRC (Rusche), "Quad Cities Station Unit 1 Proposed Amendment to Facility Operating License No. DPR-29 and Quad Cities Station Special Report No. 15, Supplement D. NRC Docket No. 50-254," dated July 1, 1975.

Together, the December 27, 1974 SER and the Status Report and its Supplement, describe an acceptable ECCS evaluation model and the basis for the staff's acceptance of the model. The Quad Cities Unit 1 evaluation which is covered by this SER properly conforms to the accepted model.

With respect to reflood and refill computation, the Quad Cities Unit 1 analysis was based on a modified version of the SAFE computer code, with explicit consideration of the staff recommended limitations. These are described on pages 7 and 8 of the December 27, 1974 SER. The Quad Cities Unit 1 evaluation did not attempt to include any further credit for other potential changes which the December 27, 1974 SER indicated were under consideration by GE at that time.

During the course of our review, we concluded that additional individual break sizes should be analyzed to substantiate the break spectrum curves submitted in connection with the evaluation provided in August 1974. We also requested that other break locations be studied to substantiate that the limiting break location was the recirculation line.

The additional analyses supported the earlier submittal which concluded that the worst break was the complete severence of the recirculation line. These additional calculations provided further details with regard to the limiting location and size of break as well as worst single failure for the Quad Cities Unit 1 design. The limiting break which is the design basis accident is the complete severence of the recirculation suction line assuming a failure of the LPCI injection valve.

3.1.1. CONCLUSION

We have reviewed the evaluation of ECCS performance submitted by Commonwealth Edison for Quad Cities Unit 1 and conclude that the evaluation has been performed wholly in conformance with the requirements of 10 CFR 50.46(a). Therefore, operation of the reactor would meet the requirements of 10 CFR 50.46 provided that operation is limited to the maximum average planar linear heat generation rates (MAPLHGR) of figures 3.5.1, 3.5.1A and 3.5.1B of Commonwealth Edison's February 4, 1976 submittal (Reference 10) and to a minimum critical power ratio (MCPR) greater than 1.18.

However, certain changes must be made to the proposed technical specifications to conform with the evaluation of ECCS performance. The largest recirculation break area assumed in the evaluation was 4.2 square feet. This break size is based on operation with a closed valve in the equalizer line between the two recirculation loops. Therefore, the license is being amended to prohibit reactor operation unless the valve in the equalizer line is closed.

The ECCS performance analysis assumed that reactor operation will be limited to a MCPR of 1.18. However, the operating MCPR limits will be more limiting.

An evaluation was not provided for ECCS performance during reactor operation with one recirculation loop out of service. Therefore, the Quad Cities Technical Specifications have been changed[14/] to prohibit reactor operation under such conditions until the necessary analyses have been performed, evaluated and determined acceptable.

The LOCA analysis assumed all ADS valves operated for small line breaks with HPCI failure. Since the licensee did not provide a LOCA analysis with one ADS valve out of service for small line breaks, the Technical Specifications are being modified so as not to allow continuous operation with any ADS valve out of service.

3.2. STEAMLINE BREAK ACCIDENT

The steamline break accident analysis as presented by the licensee is acceptable based on our generic review of NEDO-20360[3/].

3.3. FUEL LOADING ERROR

Fuel loading errors are discussed in Reference 4 for an 8 x 8 fuel bundle placed in an improper location or rotated 180 degrees in a location near the center of the core. The information in Reference 4 indicates that a fuel loading error results in a peak linear heat generation rate (LHGR) of 15.7 kW/ft and a minimum critical power ratio (MCPR) of 1.08 in the misplaced 8 x 8 (2.62% enrichment) fuel bundle during steady state operation. The peak LHGR is less than that required to cause a 1% plastic strain

14/ Amendment No. 23 to Facility License No. DPR-29, dated February 26, 1976.

in the cladding. The MCPR of 1.08 in the misplaced bundle is higher than the core wide fuel integrity safety limit of 1.06, and consequently, the misplaced fuel bundle will avoid boiling transition. Fuel bundles adjacent to a misplaced fuel bundle will be negligibly affected. We conclude that the consequences of a fuel loading error are acceptable.

3.4. CONTROL ROD DROP ACCIDENT

The control rod drop accident for the Quad Cities Unit 1 reloaded core is within the bounding analysis presented in Reference 3. The Doppler coefficient of reactivity, the accident reactivity shape and magnitude function, and the rod drop scram reactivity functions are compared with the technical bases presented in Reference 3. This analysis is performed for Doppler coefficients of reactivity at the beginning of the Reload-2 fuel cycle, zero void fraction, and at both cold (20°C) and hot (286°) startup conditions. It is shown by Figures 6-1, 6-2, 6-3, 6-4 and 6-5 of Reference 3 that the maximum values of the parameters for this reloaded core will not exceed the bounding values.

Therefore, we conclude that the consequences of a control rod drop accident from any insequence control rod during startup will be below the design limit of 280 cal/gm.

3.5. FUEL HANDLING ACCIDENT

With respect to fuel handling accidents, in Reference 4, the applicant noted that the general conclusions reached in the generic 8 x 8 reload report (Reference 3) are applicable to this reload: i.e., The total activity released to the environment and the radiological exposures for the 8 x 8 fuel will be less than those values presented in the FSAR for the 7 x 7 core. As identified in the FSAR the radiological exposures for this accident with 7 x 7 fuel are well below the guidelines set forth in 10 CFR 100. Therefore, it is concluded that the consequences of this accident for the 8 x 8 fuel will also be well below the 10 CFR 100 guidelines.

3.6. OVERPRESSURE ANALYSIS

The licensee performed an overpressure analysis in order to demonstrate that an adequate margin exists below the ASME code allowable vessel pressure of 110% of vessel design pressure. The transient analyzed was the closure of all main steam isolation valves with high neutron flux scram. The analysis was performed for 100% power with the end of cycle scram reactivity insertion rate curve. The scram was initiated by high neutron flux and the void reactivity applicable to this reload was used. No credit for relief function of safety/relief valves was assumed and the failure of one safety valve to operate was assumed. This analysis (Reference 15) utilized input parameters which were equal to or more severe than those which will be experienced during this fuel cycle. The results of the analysis indicate that the peak pressure at the vessel bottom was calculated to be 1327 psig yielding a 48 psi margin below the code allowable, which is acceptable to the staff.

3.7. CONCLUSION

We have concluded that the accident analyses for Reload-2 have been performed in accordance with methods acceptable to the NRC staff and demonstrate that the consequences of postulated accidents are acceptable.

4.0. TECHNICAL SPECIFICATION AND LICENSE CHANGES

The proposed Technical Specification changes based on GETAB for Quad Cities Unit 1 identify the same Fuel Cladding Integrity Safety Limit MCPR of 1.06, but different operating limit MCPR's for the fuel types. We accept the incorporation of the Operating Limit MCPR's of Reference 4 into the Technical Specification for Quad Cities Unit 1.

15/ Commonwealth Edison Letter (Abel) to NRC (Ziemann), Quad Cities Station Unit 2 Reload No. 1 Licensing Submittal Supplement E, NRC Docket No. 50-265 dated April 16, 1975.

The proposed Technical Specification Limiting Conditions of Operation present two limitations on power distribution related to the LOCA analysis. These are the limiting assembly maximum average planar power density, MAPLHGR, and the minimum power ratio limit related to boiling crisis, MCPR. The MCPR value used in the LOCA analysis was 1.18 and this value is less than the value determined from the transient analysis which will be incorporate in the proposed Technical Specifications. The bases for establishing the limiting value of MAPLHGR are indicated in Section 3.3 of this analysis.

The licensee did not include the equalizer line area in the LOCA analysis, therefore, the license has been modified to require that the equalizer line valves remain closed at all times during reactor operation. The LOCA analysis did not address one loop operation, therefore, the Technical Specifications should not allow continuous operation with one loop out of service. The reactor may operate for periods up to 24 hours with one recirculation loop out of service. This short period of time permits corrective action to be taken and reduces the number of unnecessary shutdowns which is consistent with other Technical Specifications. During this period the reactor will be operated within the restrictions of the thermal analysis and will be protected from fuel damage resulting from anticipated transients. The restriction on operation with one loop out of service has already been placed in the Quad Cities 1 Technical Specifications as a result of a review associated with Quad Cities 2.

The LOCA analysis assumed all ADS valves operated for small line breaks with HPCI failure. Since the licensee did not provide a LOCA analysis with one ADS valve out of service for small line breaks,we have modified the Technical Specifications so as not to allow continuous operation with any ADS valve out of service; except one valve may be out of service for seven days, with HPCI tested daily. The modified specification reduces the period of time that one ADS valve may be out of service from 30 days to 7 days, and eliminates a provision that permitted two ADS valves to be out of service for 7 days.

During our review of the proposed amendments, we have identified certain changes that were necessary to conform to the NRC staff's requirements. These changes have been discussed with and agreed to by representatives of Commonwealth Edison, and they have been made.

4.1. CONCLUSION

We conclude that the Technical Specifications as modified are consistent with the evaluations and are acceptable.

5.0 ENVIRONMENTAL CONSIDERATIONS

The Commission's staff has evaluated the potential for environmental impact associated with operation of Quad Cities Unit 2 in the proposed manner. From this evaluation, the staff had determined that there will be no change in effluent types or total amounts, no change in authorized power level and no significant environmental impact attributable to the proposed action. Having made this determination, the Commission has further concluded pursuant to 10 CFR Section 51.5(c)(1) that no environmental impact statement need be prepared for this action. A Negative Declaration and supporting Environmental Impact Appraisal are being issued with this amendment to the license. As required by Part 51, the Negative Declaration is being filed with the Office of the Federal Register for publication.

6.0 CONCLUSION

Based on our evaluation of reactor operation with Reload-2 fuel, we have concluded that because this change 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, the change does not involve a significant hazards consideration and that there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner. Based on our evaluation of operating limits based upon GETAB and on an acceptable ECCS evaluation model, we have concluded that there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner. We have also concluded, based on the considerations discussed in this evaluation that all of the activities discussed herein will be conducted in compliance with the Commission's regulations and that the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: March 12, 1976

NEGATIVE DECLARATION

REGARDING PROPOSED CHANGES TO THE

TECHNICAL SPECIFICATIONS OF LICENSE NO. DPR-29,

QUAD CITIES NUCLEAR POWER STATION UNIT 1,

DOCKET NO. 50-254

The Nuclear Regulatory Commission (the Commission) has considered the issuance of changes to the Technical Specifications of Facility Operating License No. DPR-29. These changes would authorize the Commonwealth Edison Company (the licensee) to operate the Quad Cities Nuclear Power Station Unit 1 (located in Rock Island County, Illinois) with changes to the limiting conditions for operation associated with fuel assembly specific power (average planar linear heat generation rate) resulting from application of the Acceptance Criteria for Emergency Core Cooling System (ECCS). This change is being made in conjunction with refueling with additional 8 x 8 fuel.

The U. S. Nuclear Regulatory Commission, Division of Operating Reactors, has prepared an environmental impact appraisal for the proposed changes to the Technical Specifications of License No. DPR-29, Quad Cities Unit 1, described above. On the basis of this appraisal, the Commission has concluded that an environmental impact statement for this particular action is not warranted because there will be no environmental impact attributable to the proposed action other than that which has already been predicted and described in the Commission's Final Environmental Statement for Quad Cities Nuclear Power Station Units 1 and 2 published in September 1972.

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The environmental impact appraisal is 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 61265.

Dated at Bethesda, Maryland, this 10th day of March 1976.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by Richard D. Silver

Richard D. Silver, Acting Chief
~~Dennis L. Ziemann, Chief~~
Operating Reactors Branch #2
Division of Operating Reactors

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ENVIRONMENTAL IMPACT APPRAISAL BY

THE DIVISION OF OPERATING REACTORS

SUPPORTING AMENDMENT NO. 25 TO DPR-29

COMMONWEALTH EDISON COMPANY

QUAD CITIES NUCLEAR POWER STATION UNIT-1

ENVIRONMENTAL IMPACT APPRAISAL

1. Description of Proposed Action

By letters dated July 1, 1975 and December 23, 1975 and supplements thereto dated July 7, 1975, September 19, 1975, November 6, 1975, February 4, 1976, February 6, 1976 and March 5, 1976, the Commonwealth Edison Company (CECo) submitted proposed changes to the Technical Specifications Appendix A to License No. DPR-29. The proposed changes were requested to incorporate limiting conditions for operation associated with fuel assembly specific power (average planar linear heat generation rate) resulting from the application of the Acceptance Criteria for Emergency Core Cooling System (ECCS) in conjunction with refueling using additional 8 x 8 fuel assemblies. The NRC staff has reviewed this proposed action to determine whether any environmental impact is associated with these proposed changes and the conclusions are set forth below.

The licensee is presently licensed to possess and operate Quad Cities Nuclear Power Station Unit 1 located in the State of Illinois, County of Rock Island, at power levels up to 2,511 megawatt thermal (MWt) using a core consisting of 7 x 7 and 8 x 8 fuel assemblies (containing U-235). The proposed change to incorporate the ECCS Acceptance Criteria in conjunction with refueling using additional 8 x 8 fuel does not result in an increase or decrease in power levels of the unit. The restrictions on heat generation rates will require careful control of fuel operating history. However, there should be no reduction on total burnup resulting from the revised ECCS evaluation methods. Since neither power level nor fuel burnup is affected by the action, the action does not affect the benefits of electric power production considered for the captioned facility in the Commission's Final Environmental Statement (FES) for Quad Cities Nuclear Power Station, Docket Nos. 50-254 and 50-265 dated September 1972.

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2. Environmental Impacts of Proposed Action

Potential environmental impacts associated with the proposed action are those which may be associated with incorporation of the ECCS Acceptance Criteria and utilization of nuclear fuel for this facility.

It is particularly noted that in the absence of any significant change in power levels, there will be no change in cooling water requirements and consequently no increase in environmental impact from radioactive effluents and thermal effluents for normal operation or post-accident conditions which in turn could not lead to significant increases in radiation doses or thermal stress to the public or to biota in the environment.

For normal operating conditions, no environmental impact other than as described in the Commission's Final Environmental Statement (FES) for Quad Cities Nuclear Power Station, Docket Nos. 50-254 and 50-265 dated September 1972, can be predicted for the proposed action. The Commission's calculated releases for radioactive effluents, both gaseous and liquid, are based on expected release rates to the environment and are quantified on the basis of the total quantity of nuclear fuel within the reactor. The estimates of radionuclides and release rates will not be affected by the proposed action, and since the total quantity of nuclear fuel is unchanged, no increase in the calculated release of radioactive effluents is predicted. Consequently, no increases in radiation doses to man or other biota are predicted.

3. Conclusion and Basis for Negative Declaration

On the basis of the foregoing analysis, it is concluded that there will be no environmental impact attributable to the proposed action other than has already been predicted and described in the Commission's FES for Quad Cities Nuclear Power Station Units 1 and 2. Having made this conclusion, the Commission has further concluded that no environmental impact statement for the proposed action need be prepared and that a negative declaration to this effect is appropriate.

Date: 3/10/76

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UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-254

COMMONWEALTH EDISON COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

Notice is hereby given that the U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 25 to Facility Operating License No. DPR-29, 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 Nuclear Power Station Unit 1 located in Rock Island County, Illinois. The amendment is effective as of its date of issuance.

The amendment (1) authorizes operation with additional 8 x 8 fuel assemblies and (2) incorporates operating limits in the Technical Specifications for the facility based on an acceptable evaluation model that conforms with the requirements of Section 50.46 of 10 CFR Part 50.

The applications for the amendment comply 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 amendment. Notice of Proposed Issuance of Amendment to Facility Operating License in connection with item (2) above was published in the FEDERAL REGISTER on September 5, 1975 (40 FR 41197). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action on item (2) above. Prior public notice of item (1) above is not required since the amendment does not involve a significant hazards consideration.

In connection with the issuance of this amendment, the Commission has issued a Negative Declaration and Environmental Impact Appraisal.

For further details with respect to this action, see (1) the applications for amendments dated July 1, 1975 and December 23, 1975 and supplements thereto dated July 7, 1975, September 19, 1975, November 6, 1975, February 4, 1976, February 6, 1976 and March 5, 1976, (2) Amendment No. 25 to License No. DPR-29, (3) the Commission's concurrently issued related Safety Evaluation, and (4) the Commission's Negative Declaration dated (which is also being published in the FEDERAL REGISTER), and associated Environmental Impact Appraisal. 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 at 504 - 17th Street, Moline, Illinois 61265. A single copy of items (2), (3) and (4) 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 12th day of March, 1976.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by Richard D. Silver

Richard D. Silver, Acting Chief

~~Dennis L. Ziemann, Chief~~

Operating Reactors Branch #2

Division of Operating Reactors

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DATE ➤	3/12/76	3/[illegible]/76	3/11/76	3/12/76	3/12/76