Docket No. 50-265

OCT 1976

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

Gentlemen:

In response to your request dated August 6, 1976, and a supplement thereto dated October 15, 1976, the Commission has issued the enclosed Amendment No. 33 to Facility Operating License No. DPR-30 for Unit No. 2 of the Quad Cities Nuclear Power Station.

This amendment (1) authorizes operation with additional 8 x 8 fuel assemblies, (2) incorporates revised MAPLHGR and MCPR limits in response to the plant specific analysis for reload 2 and (3) modifies License Condition 3.C to reflect End-of-Cycle scram reactivity conditions for reload 2.

Copies of the related Safety Evaluation and the Notice of Issuance are also 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-30
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

DOR: RSB
R.L. Baer
10/22/76

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DATE	10/20/76	10/ /76	10/26/76	10/23/76	10/23/76	

OCT 1976

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ATTN: Chief, Division of
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OFFICE ➤						
SURNAME ➤						
DATE ➤						

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. 33
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 August 6, 1976, as supplemented on October 15, 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.

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2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 3.C of Facility License No. DPR-30 is hereby amended to read as follows:

   3.C <u>Restrictions</u>

   Reactor power level shall be limited to maintain pressure margin to the safety valve set points during the worst case pressurization transient. The magnitude of the power limitation, if any, and the point in the cycle at which it shall be applied is specified in the Reload No. 2 licensing submittal for Quad-Cities Unit 2 (NEDO-21313). Plant operation shall be limited to the operating plant described therein.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Original [illegible]
Dennis L. [illegible]

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: OCT 28 1976

OFFICE ➤						
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DATE ➤						

ATTACHMENT TO LICENSE AMENDMENT NO. 33

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. The changed areas on the revised pages are reflected by a marginal line.

Remove Pages	Insert Pages
1.2/2.2-3	1.2/2.2-3
3.5/4.5-5	3.5/4.5-5
3.5/4.5-6	3.5/4.5-6
3.5/4.5-9	3.5/4.5-9
3.5/4.5-10	3.5/4.5-10
3.5/4.5-12	3.5/4.5-12
3.5/4.5-14	3.5/4.5-14
Figure 3.5-1 (Sheets 1 and 2)	Figure 3.5-1 (Sheets 1 and 2)

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ATTACHMENT TO LICENSE AMENDMENT NO. 33

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. The changed areas on the revised pages are reflected by a marginal line.

Remove Pages	Insert Pages
1.2/2.2-3	1.2/2.2-3
3.5/4.5-5	3.5/4.5-5
3.5/4.5-6	3.5/4.5-6
3.5/4.5-9	3.5/4.5-9
3.5/4.5-10	3.5/4.5-10
3.5/4.5-12	3.5/4.5-12
3.5/4.5-14	3.5/4.5-14
Figure 3.5-1 (Sheets 1 and 2)	Figure 3.5-1 (Sheets 1 and 2)

2.2 LIMITING SAFETY SYSTEM SETTING BASES

In compliance with Section III of the ASME Code, the safety valves must be set to open at no higher than 103% of design pressure, and they must limit the reactor pressure to no more than 110% of design pressure. Both the high-pressure scram and safety valve actuation are required to prevent overpressurizing the reactor pressure vessel and thus exceeding the pressure safety limit. The pressure scram is actually a backup protection to the high flux scram which was analyzed (Reference 1) in Section 4.4.3 of the SAR and reexamined for Unit 1 fuel cycle 2 in 'Dresden Station Special Report No. 29 Supplement B.' If the high flux scram were to fail during a maximum pressure transient (also assuming failure of the turbine stop valve closure scram, failure of the bypass system to actuate, and failure of the relief valves to open), the pressure would rise rapidly due to void reduction in the core. A high-pressure scram would occur at 1060 psig.

The pressure at the bottom of the vessel is about 1232 psig when the first safety valve opens and about 1272 psig when the last valve opens. Both values are clearly within the code requirements.

Vessel dome pressure reaches about 1298 psig, with the peak at the bottom of the vessel near 1327 psig. Therefore, the neutron flux scram and safety valve actuation provide adequate margin below the peak allowable vessel pressure of 1375 psig.

References

1. 'Quad-Cities/Nuclear Power Station First Reload License Submittal,' Section 6.2.4.2, February 1974.

provided that during such 7 days all active components of the automatic pressure relief subsystems, the core spray subsystems, LPCI mode of the RHR system, and the RCIC system are operable.

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

operable immediately. The automatic pressure relief and RCIC systems shall be demonstrated to be operable daily thereafter.

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 7 days unless repairs are made and provided that during such time the HPCI subsystem is operable.

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

D. **Automatic Pressure Relief Subsystems**

Surveillance of the automatic pressure relief 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.

   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.

E. Reactor Core Isolation Cooling System

1. The RCIC system will 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.
2. From and after the date that the RCIC system is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding 7 days unless such system is sooner made operable, provided that during such 7 days all active components of the HPCI system are operable.
3. If the requirements of Specification 3.5.E.1 and 3.5.E.2 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

Surveillance of the RCIC system shall be performed as follows:

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.
2. When it is determined that the RCIC system is inoperable, the HPCI system shall be demonstrated to be operable immediately and daily thereafter.

F. Minimum Core and Containment Cooling System Availability

1. Any combination of inoperable components in the core and containment cooling systems shall not defeat the capability of the remaining operable components to fulfill the core and containment cooling functions.
2. When irradiated fuel is in the reactor vessel and the reactor is in the cold shutdown condition, all low-pressure core and containment cooling systems may be inoperable provided no work

F. Minimum Core and Containment Cooling System Availability

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

cycle by assuring that water can be run through the drain lines and actuating the air-operated valves by operation of the following sensors:

1) loss of air

2) equipment drain sump high level

3) 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 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 Figure 3.5-1 (3 sheets). 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 in within the prescribed limits within 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 ≥ 25% rated thermal power.

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

J. **Local LHGR**

Daily during steady-state power operation above 25% of rated thermal power, the local LHGR shall be checked.

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.

$$LHGR_{max} \quad LHGR_d \ 1 \ -(\Delta P/P)_{max}(L/L_T)$$

$LHGR_d$ = design LHGR

where:

= 17.5 kW/ft, 7 x 7 fuel assemblies

= 13.4 kW/ft, 8 x 8 fuel assemblies

$(\Delta P/P)_{max}$ = maximum power spiking penalty

= .035 initial core fuel

= .029 reload 1, 7 x 7 fuel

= .022 reload, 8 x 8 fuel

= .028 reload 1, mixed oxide fuel

L_T = total core length

= 12 feet

L = Axial distance from bottom of core

K. Minimum Critical Power Ratio (MCPR)

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

1.31 (7 x 7 fuel)
1.33 (8 x 8 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 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.

K. Minimum Critical Power Ratio (MCPR)

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

Based on the fact that when one loop of the containment cooling mode of the RHR system becomes inoperable, only one system remains, which is tested daily, a 7-day repair period was specified.

C. **High-Pressure Coolant Injection**

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

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

D. **Automatic Pressure Relief**

The relief valves of the automatic pressure relief subsystem are a backup to the HPCI subsystem. They enable the core spray subsystem or LPCI mode of the RHR system to provide protection against the small pipe break in the event of HPCI failure by depressurizing the reactor vessel rapidly enough to actuate the core spray subsystems or LPCI mode of the RHR system. The core spray subsystem and/or the LPCI mode of the RHR system provide sufficient flow of coolant to limit fuel cladding temperatures to well below cladding melt and to assure that core geometry remains intact.

Redundancy has been provided in the automatic pressure relief function in that only four of the five valves are required to operate. Loss of one of the relief valves does not materially affect the pressure-relieving capability, therefore a 24-hour repair period is specified based on the HPCI system availability during this period.

E. **RCIC**

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

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

F. **Emergency Cooling Availability**

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

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

Amendment No. 33

shown on Figure 3.5-1 as limits because conformance calculations have not been performed to justify operation at LHGR's in excess of those shown.

J. **Local LHGR**

This specification assures that the maximum linear heat-generation rate in any rod is less than the design linear heat-generation rate even if fuel pellet densification is postulated. The power spike penalty specified is based on that presented in Reference 4 and assumes a linearly increasing variation in axial gaps between core bottom and top and assures with 95% confidence that no more than one fuel rod exceeds the design LHGR due to power spiking. An irradiation growth factor of 0.25% was used as the basis for determining $\Delta P/P$ in accordance with References 5 and 6.

K. **Minimum Critical Power Ratio (MCPR)**

The steady state values for MCPR specified in this specification were selected to provide margin to accommodate transients and uncertainties in monitoring the core operating state as well as uncertainties in the critical power correlation itself. These values also assure that operation will be such that the initial condition assumed for the LOCA analysis, an MCPR of 1.18, is satisfied. For any of the special set of transients or disturbances caused by single operator error or single equipment malfunction, it is required that design analyses initialized at this steady-state operating limit yield a MCPR of not less than that specified in Specification 1.1.A at any time during the transient, assuming instrument trip settings given in Specification 2.1. For analysis of the thermal consequences of these transients, the limiting value of MCPR stated in this specification is conservatively assumed to exist prior to the initiation of the transients. The results apply with increased conservatism while operating with MCPR's greater than specified. The limiting transient which determines the required steady-state MCPR limits is the rod withdrawal error which assumes that the operator ignores all alarms during the course of the transient.

For core flow rates less than rated, the steady state MCPR is increased by the formula given in the specification. This assures that the MCPR will be maintained greater than that specified in Specification 1.1.A even in the event that the motor-generator set speed controller causes the scoop tube positioner for the fluid coupler to move to the maximum speed position.

References

1. Quad-Cities Station Special Report No. 15, Supplement B, 'Unit 1 and 2 Loss of Coolant Accident Analyses in Conformation with 10 CFR 50, Appendix K.'
2. GE Topical Report NEDO-20566, 'General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50 Appendix K.'
3. I. M. Jacobs and P. W. Marriott, GE Topical Report APED 5736, 'Guidelines for Determining Safe Test Intervals and Repair Times for Engineered Safeguards,' April 1969.
4. 'Fuel Densification Effects on General Electric Boiling Water Reactor Fuel,' Section 3.2.1, Supplement 6, August 1973.
5. J. A. Hinds, GE, Letter to V. A. Moore, USAEC, 'Plant Evaluation with GE GEGAP-III,' December 1973.
6. USAEC Report, 'Supplement 1 to the Technical Report on Densification of General Electric Reactor Fuels,' December 14, 1973.

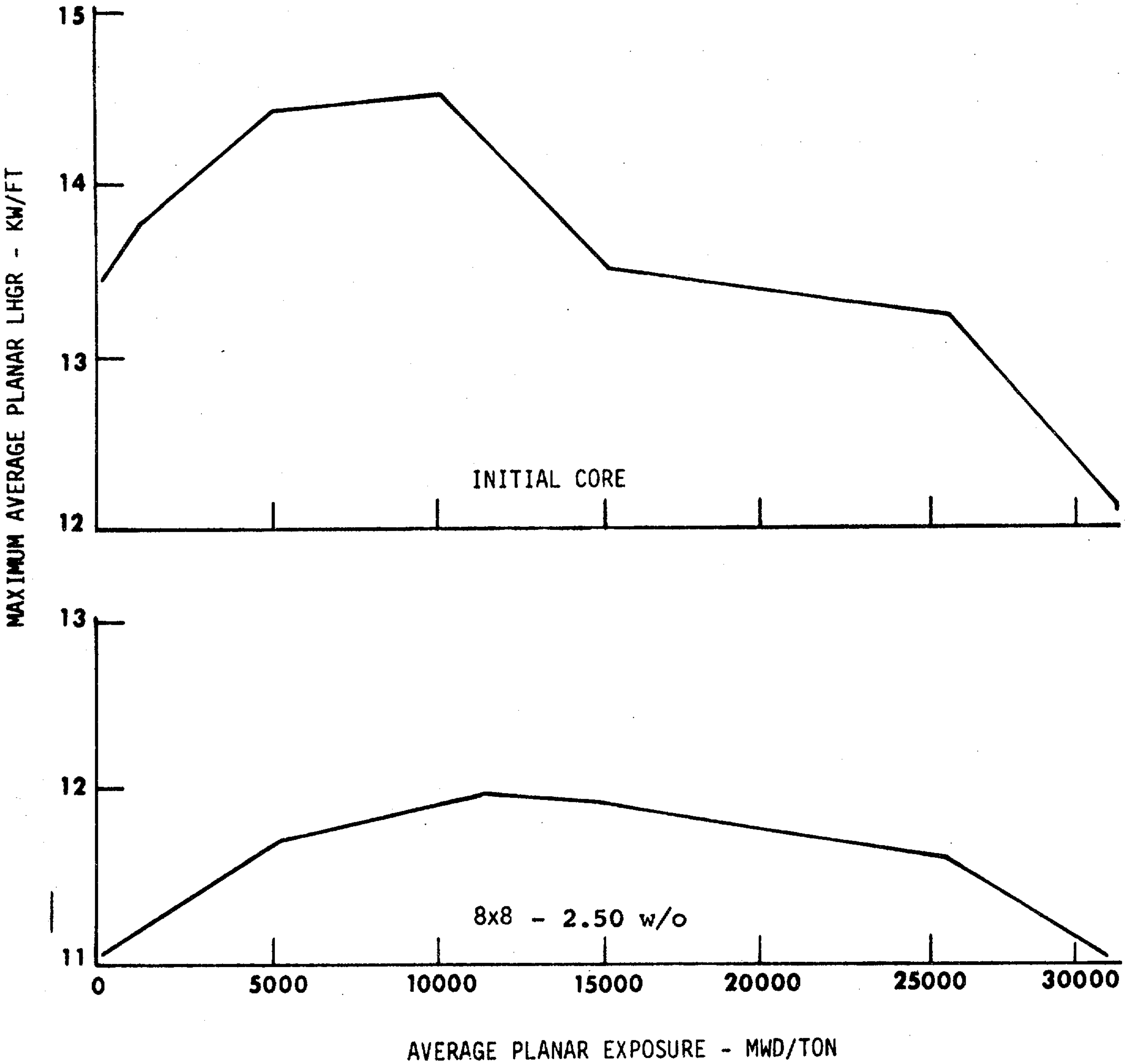

FIGURE 3.5-1 (Sheet 1 of 2)

MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VS. PLANAR AVERAGE EXPOSURE

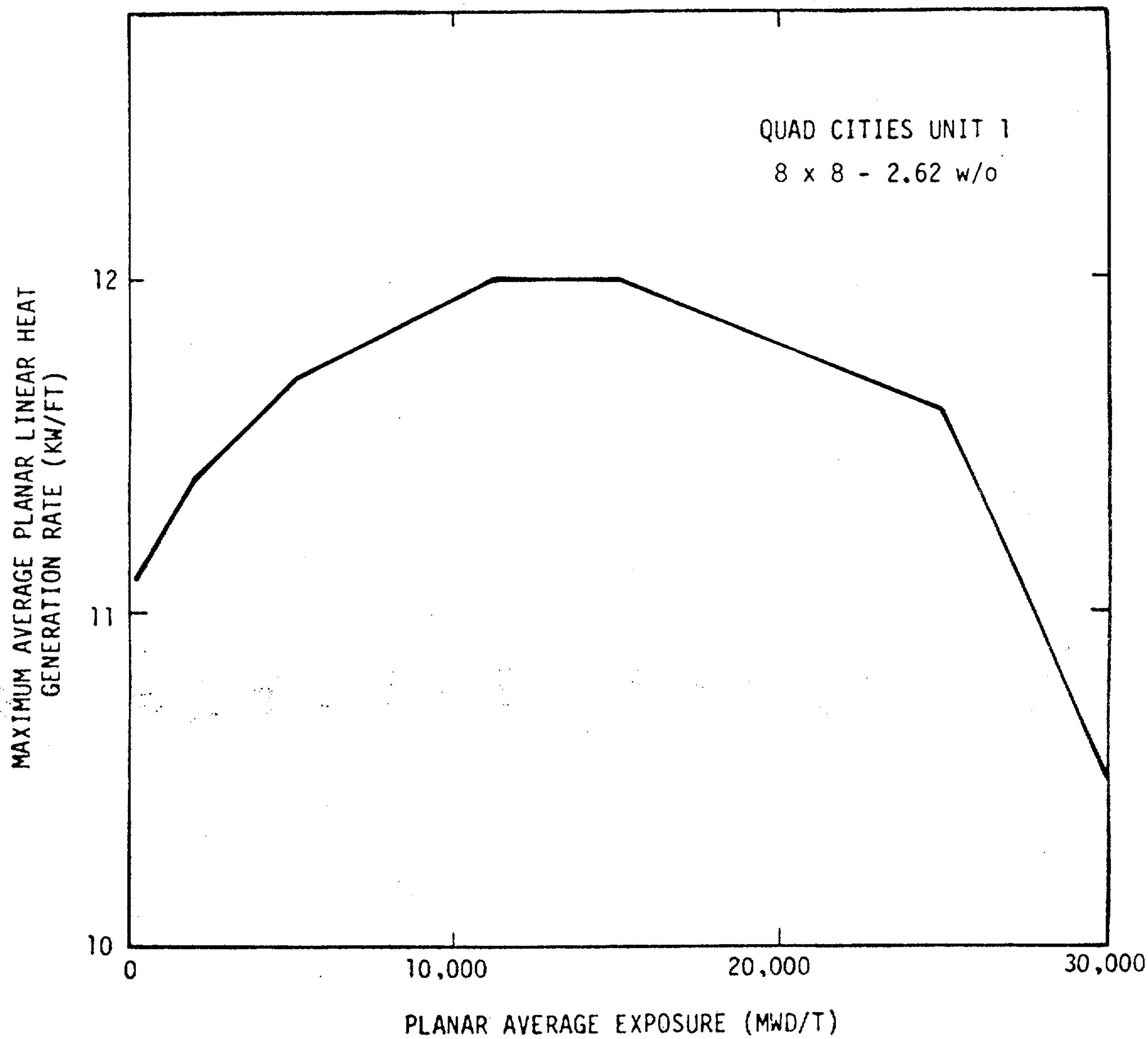

FIGURE 3.5-1 (Sheet 2 of 2)

MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VS. PLANAR AVERAGE EXPOSURE

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 33 TO FACILITY LICENSE NO. DPR-30

COMMONWEALTH EDISON COMPANY
AND
IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

QUAD CITIES NUCLEAR POWER STATION UNIT 2

DOCKET NO. 50-265

INTRODUCTION

By letter dated August 6, 1976 and a supplement thereto dated October 15, 1976, Commonwealth Edision (CECo) requested an amendment to Facility Operating License No. DPR-30. The amendment would modify the license and technical specifications for Quad Cities Station Unit No. 2 to permit operation:

(1) with additional 8 x 8 fuel assemblies (Reload-2), as requested in their application dated August 6, 1976;

(2) incorporating revised MAPLHGR and MCPR limits in response to the plant specific analysis for Reload 2; and

(3) with License Condition 3.C modified to reflect end-of-cycle scram reactivity conditions for Reload 2.

During our review of the proposed technical specifications we determined that certain changes were necessary to conform with Regulatory Requirements. These changes have been accepted by CECo.

DISCUSSION

The reference core loading for Quad Cities 2, Reload-2 consists of 412 initial 7 x 7 fuel assemblies, 148 Reload-1 8 x 8 fuel assemblies and 164 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 8 x 8 fuel assemblies for reloads was also

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

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 2 was evaluated and approved by Amendment No. 9 to Facility Operating License No. DPR 30 dated April 21, 1975.

With two exceptions, the evaluations of the acceptability of the reload fuel for the Quad Cities Unit 2 Reload-1 core are applicable to the Reload-2 fuel. A design change for this reload 8 x 8 fuel is the use of the improved water rod design[3/] and the use of fuel with slightly higher enrichment for 8 x 8 bundles than previously evaluated for Quad Cities 2.

Our safety evaluation of this reload (Reload No. 2) for the Quad Cities Unit 2 core is based on the licensee's application as amended, and on information contained in a GE topical report, NEDO-20360[3/] referenced in the application.

EVALUATION

NUCLEAR CHARACTERISTICS

The information presented in the licensing submittal for the reconstituted core[4/] closely follows the guidelines of Appendix A of Reference 3. Up to 164 8 x 8 reload fuel bundles will be loaded throughout the core. As many as 112 of these reload fuel bundles will have an average enrichment of 2.50% by weight of the uranium-235 isotope while the remainder, as 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 during this reload. The new loading pattern may be described as follows: (1) the two rows and two columns of fuel bundles intersecting at the center of the core will not contain any Reload-2 fuel, (2) the higher enrichment

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, Revision 1, Supplement 4 (April 1976).

4/ General Electric BWR Reload-2 Licensing Submittal for Quad Cities Unit 2 Nuclear Power Station - NEDO-21313, June 1976.

reload bundles will be loaded in the interior of the core and the lower enrichment reload bundles will be loaded near the outer periphery of the core, (3) some of the initial and Reload-1 fuel bundles will be relocated. 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 2.

The shutdown margin of the reconstituted core meets the Technical Specification requirement that the core be at least 0.25% Δk subcritical in the most reactive operating state with the control rod of highest reactivity worth fully withdrawn with all other control rods fully inserted. A minimum shutdown margin of 2.5% Δk, with one rod fully withdrawn, exists for the Reload-2 cycle. This shutdown margin was calculated for a core average exposure of 11,939 MWd/t at the end of the cycle 3.

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. 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 the peak reactivity point, have a k_∞ less than 1.25. Therefore, the fuel storage requirement for Quad Cities Unit 2 is acceptable.

The full power scram reactivity curves for Cycle 3 are compared with the technical bases in Figures 6-4 and 6-5 of Reference 4. The maximum value of the reactivity after Reload-2 will be well below the bounding value corresponding to the peak fuel enthalpy of 280 cal/gm. This will be true for both cold and hot startup operating states.

Based on our review of the information presented in the Quad Cities Unit 2 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 2 fuel cycles and are acceptable.

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 14 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 2 Reload-2 fuel are identical in design to those that have been used previously on Dresden Units 2 and 3, and Quad Cities Unit 1 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. The Reload-2 incorporates also the improved water rod design described in Reference 3.

The staff has reviewed the water rod desiqn on a generic basis in Reference 1 and deemed it acceptable.

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 2 Reload-2 mechanical design is acceptable.

TABLE 1

COMPARISON OF PARAMETERS FOR 8x8 AND 7x7
ROD FUEL ASSEMBLY DESIGN

	7x7	8x8
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

THERMAL-HYDRAULICS

The GE generic 8 x 8 fuel reload topical report[3/] and GETAB report[5/] 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

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

Our evaluation of the Quad Cities Unit 2 Cycle 3 (Reload-2) developed thermal margins based on the GETAB report[5/] and plant specific input information provided by the licensee.

We have evaluated and report herein the Quad Cities Unit 2 Cycle 3 (Reload-2) developed thermal margins based on the GETAB report[5/] and plant specific input information provided by the licensee.

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, combined with the relative bundle power distribution in the core, form the basis for the GETAB statistical determination of the safety limit MCPR.

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

The generic core selected for the GETAB statistical analysis is a typical 251/764 core while the Quad Cities Unit 2 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 2 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 2 Fuel Cycle 3 (Reload-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 (ΔCPR). The licensee has submitted the results of analyses of those transients which produce a significant decrease in MCPR (Reference 4).

The most limiting transient was the rod withdrawal error. This transient results in ΔCPR's of 0.20 (7x7 fuel) and 0.22 (8x8 fuel). Addition of these ΔCPR'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 would be 1.26 for 7 x 7 fuel and 1.28 for 8 x 8 fuel for the most limiting transient. However, these limits would be further increased to 1.31 and 1.33 to compensate for the reduction in CPR that could result from a fuel loading error as discussed later.

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 1.25 and the design conservatism factor for Doppler coefficient was 0.95. 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. This results in a conservative ΔMCPR and is acceptable.

The R-factors, which are a function of the local power peaking, assumed in the analyses are representative of beginning-of-cycle conditions. The values assumed are 1.079 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 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 which according to the GE[6/] study would reduce the required MCPR.

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.

The above analyses have shown that the operating limit MCPR's of 1.31 for 7 x 7 fuel and 1.33 for 8 x 8 fuel assure that the fuel cladding integrity safety limit is not exceeded during anticipated abnormal operational transients. It can be concluded, therefore, that these values of the operating limit MCPR's are acceptable.

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. We conclude that the analysis performed for this localized transient and the consequences of this localized transient are acceptable.

TURBINE TRIP WITH FAILURE OF BYPASS VALVES

The turbine trip with failure of the bypass valves is the most severe pressure transient. The primary characteristic of this transient is a pressure increase due to obstruction of steam flow by the turbine stop valves. The analyses of the transient have indicated that Quad Cities, Unit 2, Reload 2 can operate at 100 percent of power until the point in the fuel cycle corresponding to 1500 MWd/t before the end of Cycle 3 (EOC-3). At that time, in order to maintain pressure margin to the safety valve setpoints the plant should coast down to 98 percent of full power. The license is being amended to incorporate this restriction.

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 conditon, the licensee will conform to Technical Specification limiting conditions for operation.

6/ Amendment No. 12 to Facility License No. DPR-30, dated April 21, 1975.

This requires that for core flows less than the rated flow, the licensee maintain the MCPR greater than the operating minimum values (1.26 for 7 x 7 fuel and 1.28 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. 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.

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.

The analyses of Loss-of-Coolant accident for the Quad Cities 2, Reload 2, type 8D262 fuel were performed by the licensee using the analytical models and associated assumptions given in Reference 3 which was approved by the NRC. The results of these analyses, presented in Reference 4, are in conformance with the requirements of 10 CFR 50.46. The type 8D250 reload fuel for Cycle 3 is the same as previously licensed for operation in Quad Cities 2. The LOCA analysis for this type of fuel has been previously approved by the staff[6/].

Based upon the review of the analyses presented above we conclude that the operation of the reactor would meet the requirement of 10 CFR 50.46 provided that operation is limited to the maximum average planar linear heat generation rates (MAPLHGR) of Figure 3.5-1 of the Quad Cities 2 Technical Specifications for the initial core fuel and for the fuel type 8D250 and of Figure 6-7 of Reference 4 for the fuel type 8D262. It should also meet the requirement of a minimum critical power ratio (MCPR) greater than 1.18.

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

The LOCA analysis assumed all Automatic Depressurization System (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 should be modified to prohibit continuous operation with any ADS valve out of service.

STEAMLINE BREAK ACCIDENT

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

FUEL LOADING ERROR ACCIDENT

Fuel loading errors are discussed in Reference 4 for a fuel bundle inserted in an improper location or rotated 180 degrees in a location near the center of the core and the error not discovered in the subsequent core verification. The information in Reference 4 indicates that this fuel loading error results in a peak linear heat generation rate (LHGR) of 16.7 kw/ft and a MCPR of 1.01 in the misplaced reload bundle. Fuel elements adjacent to a misplaced bundle will not be significantly affected by the presence of a misplaced bundle.

The fuel loading accident is being generically reviewed by the NRC staff and generic resolution is anticipated. In the interim the licensee has agreed to increase the operating limit MCPR to 1.31 and 1.33 for 7 x 7 and 8 x 8 fuel assemblies respectively. thereby assuring that the safety limit of 1.06 will not be violated by this accident. This will keep the rod bundle from boiling transition during steady state operation even if the worst misloading error should occur. We have concluded that this is acceptable.

CONTROL ROD DROP ACCIDENT

The control rod drop accident for the Quad Cities Unit 2 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 (260°C) startup conditions. It is shown by Figures 6-1, 6-2, 6-3, 6-4 and 6-5 of Reference 4 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.

FUEL HANDLING ACCIDENT

With respect to fuel handling accidents, the applicant noted in Reference 4, 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, we have concluded that the consequences of this accident for the 8 x 8 fuel will also be well below the 10 CFR 100 guidelines.

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 a high neutron flux scram.

The analysis was performed for full power, full flow conditions at the end of cycle and at 1500 MWd/t before the end of cycle. In both these cases no credit for relief function of safety/relief valves was taken. Although the analysis assumed all safety valves functioning a sensitivity study was provided (Reference 7) indicating how much the pressure would increase if one safety valve fails.

The results of the analysis indicate that the peak pressure at the bottom of the vessel to be 1295 psig, providing an 80 psig margin to the vessel code limit of 1375 psig. We consider this to be an acceptable margin.

7/ Letter I. F. Stuart (GE) to V. Stello (NRC), Code Overpressure Protection Analysis - Sensitivity of Peak Vessel Pressure to Valve Operability, dated December 23, 1975.

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.

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. On the basis of the above evaluation of operating limit MCPR, the MCPR's of Reference 4, as modified, are acceptable.

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. A curve specifying the MAPLHGR for 8D262 type fuel will be added to the Technical Specifications. The value of MCPR used in the LOCA analysis was 1.18. This value is less than the value determined from the transient analysis which would be incorporated in the proposed Technical Specifications for an operating limit.

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 proposed Technical Specifications would be modified to prohibit 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.

ENVIRONMENTAL CONSIDERATION

We have determined that the amendment does 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 amendment involves 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 this amendment.

CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) because the amendment 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 amendment does 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 the amendment 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

DOCKET NO. 50-265

COMMONWEALTH EDISON COMPANY
AND
IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY
OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 33 to Facility Operating License No. 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 Nuclear Power Station Unit No. 2 (the facility) located in Rock Island County, Illinois. The amendment is effective as of its date of issuance.

The amendment (1) authorized operation with additional 8 x 8 fuel assemblies, (2) incorporated revised MAPLHGR and MCPR limits in response to the plant specific analysis for reload 2, and (3) modified License Condition 3.C to reflect end-of-cycle scram reactivity conditions for reload 2.

The application for the amendment 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 amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

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The Commission has determined that the issuance of this amendment 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 this amendment.

For further details with respect to this action, see (1) the application for amendment dated August 6, 1976 and a supplement thereto dated October 15, 1976, (2) Amendment No. 33 to License No. DPR-30, and (3) the Commission's 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 23rd day of October, 1976

FOR THE NUCLEAR REGULATORY COMMISSION

Original S[illegible] by
Dennis L. [illegible]

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

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