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MARCH 8 1978

Docket No. 50-265

Commonwealth Edison Company
ATTN: Mr. Cordell Reed
Assistant Vice President
Post Office Box 767
Chicago, Illinois 60690

Gentlemen:

The Commission has issued the enclosed Amendment No. 43 to Facility License No. DPR-30 for the Quad Cities Nuclear Power Station, Unit No. 2. This amendment consists of changes to the License and appended Technical Specifications in response to your request dated December 2, 1977, as supplemented February 28 and March 1, 1978.

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

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

Sincerely,

Original signed by

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosures:
1. Amendment No. 43 to DPR-30
2. Safety Evaluation
3. Notice

cc w/enclosures: See page 2

Notified N Kalivianakis QC Plant superintendent that this amendment was signed. PWO'Connor 3/7/78 3:15PM

OFFICE	ORB #2 PWO	~~ORB #3~~	~~OT~~	OELD	ORB #3
SURNAME	PO'Connor	RBevan:mjf		Karman	GLear
DATE	3/7/78	3/7/78	~~3/ /78~~	3/7/78	3/7/78

NRC FORM 318 (9-76) NRCM 0240

U. S. GOVERNMENT PRINTING OFFICE: 1976 — 626-624

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

March 8, 1978

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cc w/enclosures: See page 2

cc w/enclosures:
Mr. D. R. Stichnoth
President
Iowa-Illinois Gas and
Electric Company
206 East Second Avenue
Davenport, Iowa 52801

Mr. John W. Rowe
Isham, Lincoln & Beale
Counselors at Law
One First National Plaza, 42nd Floor
Chicago, Illinois 60603

Mr. Nick Kalivianakas
Plant Superintendent
Quad-Cities Nuclear Power Station
22710 - 206th Avenue - North
Cordova, Illinois 61242

Anthony Z. Roisman
Natural Resources Defense Council
917 15th Street, N. W.
Washington, D. C. 20005

Moline Public Library
504 - 17th Street
Moline, Illinois 61265

Illinois Department of Public Health
ATTN: Chief, Division of Nuclear
Safety
535 West Jefferson
Springfield, Illinois 62761

Mr. Marcel DeJaegher, Chairman
Rock Island County Board
of Supervisors
Rock Island County Court House
Rock Island, Illinois 61201

Chief, Energy Systems Analyses
Branch (AW-459)
Office of Radiation Programs
U. S. Environmental Protection Agency
Room 645, East Tower
401 M Street, S. W.
Washington, D. C. 20460

U. S. Environmental Protection Agency
Federal Activities Branch
Region V Office
ATTN: EIS COORDINATOR
230 South Dearborn Street
Chicago, Illinois 60604

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

COMMONWEALTH EDISON COMPANY
AND
IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

DOCKET NO. 50-265

QUAD CITIES UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 43
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 December 2, 1977, as supplemented on February 28 and March 1, 1978, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

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

3.B <u>Technical Specifications</u>

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 43, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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. 3 licensing submittal for Quad Cities Unit 2 (NEDO 24063). Plant operation shall be limited to the operating plan described therein. Subsequent operation in the coastdown mode is permitted to 40% power.

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

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 8, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 43

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

Remove Pages	Insert Pages
1.1/2.1-2	1.1/2.1-2
1.2/2.2-2	1.1/2.2-2
1.2/2.2-3	1.2/2.2-3
3.2/4.2-7	3.2/4.2-7
3.2/4.2-8	3.2/4.2-8
3.2/4.2-14	3.2/4.2-14
3.3/4.3-3	3.3/4.3-3
3.5/4.5-14	3.5/4.51-4
	3.5/4.5-14A (added)
3.5/4.5-10	3.5/4.5-10

D. **Reactor Water Level (Shutdown Condition)**

Whenever the reactor is in the shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be less than that corresponding to 12 inches above the top of the active fuel when it is seated in the core.

curve in Figure 2.1-2, at which point the actual peaking factor value shall be used.

LTPF = 3.06 (7 x 7 fuel assemblies)
3.03 (8 x 8 fuel assemblies)

2. APRM Flux Scram Trip Setting (Refueling or Startup and Hot Standby Mode)

   When the reactor mode switch is in the Refuel or Startup Hot Standby position, the APRM scram shall be set at less than or equal to 15% of rated neutron flux.

3. IRM Flux Scram Trip Setting

   The IRM flux scram setting shall be set at less than or equal to 120/125 of full scale.

4. When the reactor mode switch is in the startup or run position, the reactor shall not be operated in the natural circulation flow mode.

B. **APRM Rod Block Setting**

The APRM rod block setting shall be as shown in Figure 2.1-1 and shall be:

$$S \leq (.65W + 43)(LTPF/TPF)$$

The definitions used above for the APRM scram trip apply.

C. Reactor low water level scram setting shall be ≥ 143 inches above the top of the active fuel at normal operating conditions.

D. Reactor low water level ECCS initiation shall be 83 inches (+4 inches/-0 inch) above the top of the active fuel at normal operating conditions.

E. Turbine stop valve scram shall be ≤ 10% valve closure from full open.

F. Turbine control valve fast closure scram shall initiate upon actuation of the fast closure solenoid valves which trip the turbine control valves.

G. Main steamline isolation valve closure scram shall be ≤ 10% valve closure from full open.

H. Main steamline low-pressure initiation of main steamline isolation valve closure shall be ≥ 850 psig.

1.2 SAFETY LIMIT BASES

The reactor coolant system integrity is an important barrier in the prevention of uncontrolled release of fission products. It is essential that the integrity of this system be protected by establishing a pressure limit to be observed for all operating conditions and whenever there is irradiated fuel in the reactor vessel.

The pressure safety limit of 1325 psig as measured by the vessel steam space pressure indicator is equivalent to 1375 psig at the lowest elevation of the reactor coolant system. The 1375 psig value is derived from the design pressures of the reactor pressure vessel and coolant system piping. The respective design pressures are 1250 psig at 575° F and 1175 psig at 560° F. The pressure safety limit was chosen as the lower of the pressure transients permitted by the applicable design codes: ASME Boiler and Pressure Vessel Code Section III for the pressure vessel. and USASI B31.1 Code for the reactor coolant system piping. The ASME Boiler and Pressure Vessel Code permits pressure transients up to 10% over design pressure (110% x 1250 = 1375 psig). and the USASI Code permits pressure transients up to 20% over the design pressure (120% x 1175 = 1410 psig). The safety limit pressure of 1375 psig is referenced to the lowest elevation of the primary coolant system.

The design basis for the reactor pressure vessel makes evident the substantial margin of protection against failure. at the safety pressure limit of 1375 psig. The vessel has been designed for a general membrane stress no greater than 26,700 psi at an internal pressure of 1250 psig: this is a factor of 1.5 below the yield strength of 40,100 psi at 575° F. At the pressure limit of 1375 psig. the general membrane stress will only be 29,400 psi. still safely below the yield strength.

The relationships of stress levels to yield strength are comparable for the primary system piping and provide a similar margin of protection at the established safety pressure limit.

The normal operating pressure of the reactor coolant system is 1000 psig. For the turbine trip or loss of electrical load transients. the turbine trip scram or generator load rejection scram together with the turbine bypass system limits the pressure to approximately 1100 psig (References 1. 2. and 3). In addition. pressure relief valves have been provided to reduce the probability of the safety valves operating in the event that the turbine bypass should fail. These valves and the neutron flux scram limit the reactor pressure to a value (References 4. 5. 6. and 7) which is at least 25 psi below the setting of the first safety valve. Finally. the safety valves are sized to keep the reactor coolant system pressure below 1375 psig with no credit taken for relief valves during the postulated full closure of all MSIVs without direct (valve position switch) scram. Credit is taken for the neutron flux scram, however. The pressure at the bottom of the vessel peaks at less than 1325 psig. The indirect flux scram and safety valve actuation, therefore, provide adequate margin below the peak allowable vessel pressure of 1375 psig.

Reactor pressure is continuously monitored in the control room during operation on a 1500 psi full-scale pressure recorder.

References

1. SAR, Section 11.2.2.
2. 'Quad-Cities' 1 Nuclear Power Station First Reload License Submittal.' Section 6.2.4.2. February 1974.
3. GE Topical Report NEDO-20693. 'General Electric Boiling Water Reactor Reload No. 1 Licensing Submittal for Quad-Cities Nuclear Power Station Unit 2.' December 1974.
4. SAR Section 4.4.3.
5. Dresden 3 Special Report No. 29. 'Transient Analysis for Cycle 2'.
6. Letter to D. J. Skovholt from J. S. Abel. October 18, 1973. Subject: Scram Reactivity Limitations for Dresden Units 2 and 3 and Quad-Cities Units 1 and 2.
7. Dresden Station Special Report No. 29. Supplement B.

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 neutron flux 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 available as backup protection to the high flux scram which was analyzed (Reference 1) in Section 4.4.3 of the SAR, reexamined in Reference 2 and the reload license submittal for each subsequent cycle. If the high flux scram were to fail, a high-pressure scram would occur at 1060 psig.

References

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

2. Dresden Station Special Report No. 29, Supplement B.

The APRM rod block function is flow biased and prevents a significant reduction in MCPR, especially during operation at reduced flow. The APRM provides gross core protection, i.e., limits the gross core control rods in the normal withdrawal sequence. The trips are set so that MCPR is maintained greater than 1.06.

The APRM rod block function, which is set at 12% of rated power, is functional in the Refuel and Startup/Hot Standby modes. This control rod block provides the same type of protection in the Refuel and Startup/Hot Standby modes as the APRM flow-biased rod block does in the Run mode, i.e., it prevents MCPR from decreasing below 1.06 during control rod withdrawals and prevents control rod withdrawal before a scram is reached.

The RBM rod block function provides local protection of the core, i.e., the prevention of transition boiling in a local region of the core for a single rod withdrawal error from a limiting control rod pattern. The trip point is flow biased. The worst-case single control rod withdrawal error has been analyzed, and the results show that with the specified trip settings, rod withdrawal is blocked before the MCPR reaches 1.06, thus allowing adequate margin (Reference 1).

Below 30% power, the worst-case withdrawal of a single control rod results in a MCPR greater than 1.06 without rod block action. Thus it is not required below this power level.

The IRM block function provides local as well as gross core protection. The scaling arrangement is such that the trip setting is less than a factor of 10 above the indicated level. Analysis of the worst-case accident results in rod block action before MCPR approaches 1.06.

A downscale indication on an APRM or IRM is an indication the instrument has failed or is not sensitive enough. In either case the instrument will not respond to changes in control rod motion, and the control rod motion is thus prevented. The downscale trips are set at 3/125 of full scale.

The SRM rod block with ≤ 100 CPS and the detector not fully inserted assures that the SRM's are not withdrawn from the core prior to commencing rod withdrawal for startup. The scram discharge volume high water level rod block provides annunciation for operator action. The alarm setpoint has been selected to provide adequate time to allow determination of the cause of level increase and corrective action prior to automatic scram initiation.

For effective emergency core cooling for small pipe breaks, the HPCI system must function, since reactor pressure does not decrease rapidly enough to allow either core spray or LPCI to operate in time. The automatic pressure relief function is provided as a backup to the HPCI in the event the HPCI does not operate. The arrangement of the tripping contacts is such as to provide this function when necessary and minimize spurious operation. The trip settings given in the specification are adequate to assure the above criteria are met (reference SAR Section 6.2.6.3). The specification preserves the effectiveness of the system during periods of maintenance, testing, or calibration and also minimizes the risk of inadvertent operation, i.e., only one instrument channel out of service.

Two air ejector off-gas monitors are provided and, when their trip point is reached, cause an isolation of the air ejector off-gas line. Isolation is initiated when both instruments reach their high trip point or one has an upscale trip and the other a downscale trip. There is a 15-minute delay before the air ejector off-gas isolation valve is closed. This delay is accounted for by the 30-minute holdup time of the off-gas before it is released to the chimney.

Both instruments are required for trip, but the instruments are so designed that any instrument failure gives a downscale trip. The trip settings of the instruments are set so that the chimney release rate limit given in Specification 3.8.A.2 is not exceeded.

Four radiation monitors are provided in the reactor building ventilation ducts which initiate isolation of the reactor building and operation of the standby gas treatment system. The monitors are located in the reactor building ventilation duct. The trip logic is a one-out-of-two for each set, and each set can initiate a trip independent of the other set. Any upscale trip will cause the desired action. Trip settings of 2 mR/hr for monitors in the ventilation duct are based upon initiating normal ventilation isolation and standby gas treatment system operation so that the ventilation stack release rate limit given in Specification 3.8.A.3 is not exceeded. Two radiation monitors are provided on the refueling floor which initiate isolation of the reactor building and operation of the standby gas treatment systems. The trip logic is one-out-of-two. Trip settings of 100 mR/hr for the monitors on the refueling floor are based upon initiating normal ventilation isolation and standby gas treatment system operation so that none of the activity released during the refueling accident leaves the reactor building via the normal ventilation stack but that all the activity is processed by the standby gas treatment system.

The instrumentation which is provided to monitor the postaccident condition is listed in Table 3.2-4. The instrumentation listed and the limiting conditions for operation on these systems ensure adequate monitoring of the containment following a loss-of-coolant accident. Information from this instrumentation will provide the operator with a detailed knowledge of the conditions resulting from the accident; based on this information he can make logical decisions regarding postaccident recovery.

The specifications allow for postaccident instrumentation to be out of service for a period of 7 days. This period is based on the fact that several diverse instruments are available for guiding the operator should an accident occur, on the low probability of an instrument being out of service and an accident occurring in the 7-day period, and on engineering judgment.

The normal supply of air for the control room ventilation system comes from outside the service building. In the event of an accident, this source of air may be required to be shut down to prevent high doses of radiation in the control room. Rather than provide this isolation function on a radiation monitor installed in the intake air duct, signals which indicate an accident. i.e., high drywell pressure, low water level, main steamline high flow, or high radiation in the reactor building ventilation duct, will cause isolation of the intake air to the control room. The above trip signals result in immediate isolation of the control room ventilation system and thus minimize any radiation dose.

References

1. GE Topical Report NEDO-24063 'General Electric Boiling Water Reactor Reload No. 3 Licensing Submittal for Quad-Cities Nuclear Power Station (Unit 2)', Section 6.3.3.2, September, 1977

TABLE 3.2-3

INSTRUMENTATION THAT INITIATES ROD BLOCK

Minimum Number of Operable or Tripped Instrument Channels per Trip System[1]	Instrument	Trip Level Setting
2	APRM upscale (flow bias)[7]	≤0.650W + 43[2]
2	APRM upscale (Refuel and Startup/Hot Standby mode)	≤12/125 full scale
2	APRM downscale[7]	≥3/125 full scale
1	Rod block monitor upscale (flow bias)[7]	≤0.650W + 42[2]
1	Rod block monitor downscale[7]	≥3/125 full scale
3	IRM downscale [3] [8]	≥3/125 full scale
3	IRM upscale[8]	≤108/125 full scale
2[5]	SRM detector not in Startup position[4]	≥2 feet below core center-line
3	IRM detector not in Startup position[8]	≥2 feet below core center-line
2[5] [6]	SRM upscale	≤10^5 counts/sec
2[5]	SRM downscale[9]	≥10^2 counts/sec
1	High water level in scram discharge volume	≤25 gallons

Notes

1. For the Startup/Hot Standby and Run positions of the reactor mode selector switch, there shall be two operable or tripped trip systems for each function except the SRM rod blocks. IRM upscale and IRM downscale need not be operable in the Run position. APRM downscale, APRM upscale (flow biased), RBM upscale, and RBM downscale need not be operable in the Startup/Hot Standby mode. If the first column cannot be met for one of the two trip systems, this condition may exist for up to 7 days provided that during that time the operable system is functionally tested immediately and daily thereafter; if this condition lasts longer than 7 days the system shall be tripped. If the first column cannot be met for both trip systems, the systems shall be tripped.
2. W is the reactor recirculation loop flow in percent. Trip level setting is in percent of rated power (2511 MWt).
3. IRM downscale may be bypassed when it is on its lowest range.
4. This function is bypassed when the count rate is ≥100 CPS.
5. One of the four SRM inputs may be bypassed.
6. This SRM function may be bypassed in the higher IRM ranges (ranges 8, 9, and 10) when the IRM upscale rod block is operable.
7. Not required to be operable while performing low power physics tests at atmospheric pressure during or after refueling at power levels not to exceed 5 MWt.
8. This IRM function occurs when the reactor mode switch is in the Refuel or Startup/Hot Standby position.
9. This trip is bypassed when the SRM is fully inserted.

3. The control rod drive housing support system shall be in place during reactor power operation and when the reactor coolant system is pressurized above atmospheric pressure with fuel in the reactor vessel, unless all control rods are fully inserted and Specification 3.3.A.1 is met.

   a. Control rod withdrawal sequences shall be established so that maximum reactivity that could be added by dropout of any increment of any one control blade would not make the core more than 0.013 Δk supercritical.

   b. Whenever the reactor is in the Startup/Hot Standby or Run mode below 20% rated thermal power, the rod worth minimizer shall be operable. A second operator or qualified technical person may be used as a substitute for an inoperable rod worth minimizer which fails after withdrawal of at least 12 control rods to the fully withdrawn position. The rod worth minimizer may also be bypassed for low power physics testing to demonstrate the shutdown margin requirements of Specification 3.3.A if a nuclear engineer is present and verifies the step-by-step rod movements of the test procedure.

4. Control rods shall not be withdrawn for startup or refueling unless at least two source range channels have an observed count rate equal to or greater than three counts per second and these SRM's are fully inserted.

5. During operation with limiting control rod patterns, as determined by the nuclear engineer, either:

   a. both RBM channels shall be operable,

   b. control rod withdrawal shall be blocked; or

3. The correctness of the control rod withdrawal sequence input to the RWM computer shall be verified after loading the sequence.

   Prior to the start of control rod withdrawal towards criticality, the capability of the rod worth minimizer to properly fulfill its function shall be verified by the following checks:

   a. The RWM computer online diagnostic test shall be successfully performed.

   b. Proper annunciation of the selection error of one out-of-sequence control rod shall be verified.

   c. The rod block function of the RWM shall be verified by withdrawing the first rod as an out-of-sequence control rod no more than to the block point.

4. Prior to control rod withdrawal for startup or during refueling, verify that at least two source range channels have an observed count rate of at least three counts per second.

5. When a limiting control rod pattern exists, an instrument functional test of the RBM shall be performed prior to withdrawal of the designated rod(s) and daily thereafter.

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.

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

where:

$LHGR_d$ = design LHGR

= 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

1.35 (7 x 7 fuel)
1.35 (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.

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 most limiting transients with respect to MCPR are generally:

a) Rod withdrawal error

b) Load rejection or Turbine Trip without bypass

c) Loss of feedwater heater

Several factors influence which of these transients results in the largest reduction in critical power ratio such as the specific fuel loading, exposure, and fuel type. The current cycles reload licensing submittal specifies the limiting transients for a given exposure increment for each fuel type. The values specified as the Limiting Condition of Operation are conservatively chosen as the most restrictive over the entire cycle for each fuel type.

For Cycle 4, the operating limit has been increased by 0.04 over the limit based on transient analyses to assure that boiling transition would not occur in a misloaded fuel bundle during steady state operation.

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 specifiec 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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 43 TO FACILITY OPERATING LICENSE NO. DPR-30

COMMONWEALTH EDISON COMPANY

QUAD-CITIES STATION UNIT NO. 2

DOCKET NO. 50-265

1.0 Introduction

By letter dated December 2, 1977 and supplements thereto dated February 28 and March 1, 1978, Commonwealth Edison (the licensee) proposed changes to the Technical Specifications of Facility Operating License DPR-30 for Quad-Cities Station Unit No. 2. The proposed changes related to the replacement of 180 fuel assemblies constituting refueling of the core for fourth cycle operation at power levels up to 2511 MWt (100% power).

In support of the reload application the licensee has provided the GE BWR Reload 3 licensing submittal for Quad-Cities Unit 2 (Reference 1) and proposed Technical Specification changes (Reference 2).

The information presented in the licensing submittal closely follows the guidelines of Appendix A of NEDO-20360 (Reference 3). Although later supplements to this report are undergoing review by the staff, portions of this topical have been found applicable for reactors containing 8x8 reload fuel and are acceptable to the staff when supplemented with information required by our status report (Reference 4). The supplemental information provided by the licensee and the staff's evaluation thereof are summarized below.

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 the licensee.

2.0 Evaluation

2.1 Nuclear Characteristics

For Cycle 4 operation of Quad-Cities Unit 2, 180 fresh 8x8 fuel bundles (8D262) with an enrichment of 2.62% U235 by weight will be loaded into the core. In addition, 232 7x7 assemblies from the initial core loading, 148 8D250 assemblies from Reload 1, and 112 8D250 and 52 8D262 assemblies from Reload 2 will remain in the core. Thus, for Cycle 4 (Reload 3) approximately 25% of the 724 fuel bundles will be fresh fuel.

As indicated by the loading diagram presented in Reference 1, the fresh fuel will be distributed symmetrically throughout the core.

The nuclear characteristics of the reload 8D262 fuel bundles are identical to those previously loaded in the core. The licensee therefore states that the total control system worth, and the temperature and void dependent behavior of the reconstituted core will not differ significantly from those values previously reported 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 condition throughout the cycle when the highest worth control rod is fully withdrawn and all other rods are fully inserted. For Cycle 4 the licensee has calculated the minimum shutdown margin to be 0.017 Δk. This occurs at the beginning of cycle. The effect of settling of B_4C in inverted poison tubes in control rods will not have a significant effect on the Cycle 4 shutdown margin.

The information presented in Reference 1 indicates that a boron concentration of 600 ppm in the moderator will bring the reactor subcritical by at least 0.040 Δk at 20°C, xenon free. Therefore, the alternate shutdown requirement of the General Design Criteria is met by the Standby Liquid Control System.

The full power scram reactivity curves used for the Reload 3 cycle are shown in Figure 6.6 of Reference 1. The scram curves used in the anticipated transient analyses include a design conservatism factor of 0.8 which is acceptable to the staff (Reference 4).

Based on our review of the information presented in the Ouad-Cities Unit 2 licensing submittal (Reference 1) as supplemented by applicable portions of the generic 8x8 reload report (Reference 3) and the staff's acceptance thereof (Reference 4), we have determined that the nuclear characteristics and expected performance of the reconstituted core for Cycle 4 are acceptable.

2.2 Mechanical Design

The Reload 3 fuel has the same mechanical configuration and fuel bundle enrichments as the 8D262 assemblies described in the 8x8 generic reload report (Reference 3). The improved water rod design described in Section 3 of Reference 3 has been adopted.

The generic 8x8 reload report (Reference 3), supplements of which are under review, has been found acceptable for use for reactors containing 8x8 reload fuel, when supplemented with information required by our status report (Reference 4) on the GE generic report evaluation. On the basis of our review of the generic 8x8 reload report and the reload submittal we conclude that the mechanical design of the Quad-Cities Unit 2 Reload 3 fuel is acceptable.

2.3 Thermal-Hydraulics

The GE generic 8x8 fuel reload topical report (Reference 3) and the General Electric Thermal Analysis Basis (GETAB) (Reference 5) are referenced to provide the description of the thermal-hydraulic methods which were used to calculate the thermal margins. Application of the GETAB establishes:

(1) the fuel damage safety limit Minimum Critical Power Ratio (MCPR),

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

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

We have evaluated the Quad-Cities Unit 2 Cycle 4 thermal margins based on the GETAB report and plant specific input information provided by the licensee. The staff evaluation of these margins is reported in the following subsections.

2.3.1 Fuel Cladding Integrity Safety Limit MCPR

The fuel cladding safety limit MCPR of 1.06 has been established, based on the GETAB (Reference 5) statistical analysis, to assure that 99.9% of the fuel rods in the core will not experience boiling transition during abnormal operational transients (Reference 6). This limit is applied for both core-wide and localized transients or perturbations to the expected CPR distribution.

The uncertainties in core and system operating parameters and the GEXL correlation uncertainties expected for Cycle 4 operation of Quad-Cities Unit 2 are the same as those used for the original statistical analysis on which the fuel cladding safety limit MCPR is based. The bundle power distribution for Cycle 4 is expected to include fewer high power bundles than the distribution assumed for the original statistical analysis as is indicated by comparing Figure 4-3 with Figures 4-4.1 through 4-4.4 of Reference 3. Therefore, it is conservative to apply the fuel cladding safety limit MCPR of 1.06 to Cycle 4 operation of Quad-Cities Unit 2.

We conclude that the proposed fuel integrity safety limit MCPR of 1.06 is acceptable for both the 7x7 and 8x8 fuel in the Quad-Cities Unit 2 reactor core during Cycle 4 (Reload 3).

2.3.2 Operating Limit MCPR

Various transients or perturbations to the CPR distribution could reduce the MCPR below the intended operating limit during Cycle 4 operation of Quad-Cities Unit 2. The most limiting operational transients and the fuel loading error have been analyzed by the licensee to determine which could potentially induce the largest reduction in MCPR.

The transients evaluated were the generator load rejection without bypass, the turbine trip with failure of the bypass valves, loss of 145°F of feedwater heating, and the control rod withdrawal error. Initial conditions and transient input parameters as specified in Table 4-2, Table 6-1 and Figure 6-6 of Reference 1 were assumed.

The input to the transient calculations and the application of the analysis methods of Reference 3 have been reviewed and determined to provide appropriate conservatism for determination of the operating limit MCPR for Quad-Cities Unit 2 during Cycle 4.

The calculated reductions in CPR during each of the operational transients have been tabulated in Reference 1. For Cycle 4 operation, the ΔCPR for the rod withdrawal error is the largest. Addition of these ΔCPR's to the safety limit MCPR of 1.06 would give the operating limit MCPR's for each fuel type which would protect against boiling transition during plant transients. The licensee has also analyzed fuel loading errors. The worst error is one in which a fresh 8x8 bundle is placed in an exposed 8x8 bundle location. Should this occur, an even higher operating limit MCPR for the 8x8 fuel would be required to ensure that for this localized perturbation the CPR at the misloading site would not be below the safety limit of 1.06 during steady state operation and that the fuel rods in the misloaded fuel assembly would not experience transition boiling.

On this basis the licensee has calculated that an operating limit MCPR of 1.34 for the 8x8 fuel and 1.32 for the 7x7 fuel is sufficient so that in the event of a fuel loading error the MCPR will not be below the 1.06 safety limit during steady state operation. Furthermore, should there be no fuel loading error, then with the proposed operating limit MCPR, 99.9% of the fuel rods will avoid transition boiling by an extra margin during any operational transient. We have reviewed these analyses and find the MCPR values acceptable.

2.3.3 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 the limiting conditions for operation stated in 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. The MCPR values for less than rated flow are the rated flow values of 1.34 and 1.32 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 operational transients do not violate the thermal limits of the fuel or the pressure limits of the reactor coolant boundary.

2.4 Accident Analysis

2.4.1 ECCS Appendix K Analysis

In December of 1976 the NRC staff was informed that certain input errors and computer code errors had been made in the evaluations that were provided under the requirements described above. An Order was issued to the Commonwealth Edison Company on March 11, 1977 (Reference 7), requiring that corrected, revised calculations fully conforming to the requirements of 10 CFR 50.46 be provided for Quad-Cities Unit 2 as soon as possible. Such corrected analyses were provided for the present reload in Reference 8. The corrected analyses included correction of all input errors previously made and correction of all computer code errors. The corrected analyses were performed using a calculational model which contains several model changes approved by the NRC staff in a Safety Evaluation issued April 12, 1977 (Reference 9).

We have reviewed the corrected analyses submitted for the Reload in Reference 10. We conclude that the Quad-Cities Unit 2 plant will be in conformance with all requirements of 10 CFR 50.46 and Appendix K to 10 CFR 50.46 when: 1) it is operated in accordance with the "MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE" values given in Appendix A of Reference 7, and 2) when it is operated at a Minimum Critical Power Ratio (MCPR) equal to or greater than 1.20 (more restrictive MCPR limits are currently required for reasons not connected with the Loss-of-Coolant-Accident, as described elsewhere in this SER). These requirements are satisfied by the Quad-Cities Unit 2 proposed Technical Specifications. A more detailed discussion of the consideration in this review is contained in Reference 11.

2.4.2 Main Steam Line Break Accident

Steam line break accidents which are postulated to occur inside containment are covered by the ECCS analysis discussed in Section 2.4.1. The analysis of steam line break accidents occurring outside containment as presented by the licensee is acceptable based on our generic review of NEDO-20360 (References 3 and 4).

2.4.3 <u>Fuel Loading Error</u>

Fuel loading errors are discussed in Reference 1 for a fuel bundle placed in an improper location or rotated 180°. For Quad-Cities Unit No. 2 the worst potential fuel loading error for Cycle 4 was analyzed from an initial MCPR of 1.30. This resulted in a MCPR of 1.02 and a peak linear heat generation rate of 17.5 KW/ft. Commonwealth Edison has proposed to increase the MCPR by at least 0.04. However, it has been observed that initiating the analysis at a higher MCPR results in a greater ΔCPR. By letter dated February 28, 1978, the licensee has submitted an analysis which shows that an operating limit MCPR of 1.35 is necessary to ensure that the safety limit MCPR will not be violated for the fuel loading error. This is in accordance with the staff's guidance in this area and the licensee has proposed that the operating limit MCPR be revised to 1.35. The implications of the MCPR have been discussed previously. The peak LHGR associated with a fuel loading error is not large enough to cause fuel damage. Therefore, we find this analysis and proposed Technical Specifications acceptable.

2.4.4 <u>Control Rod Drop Accident</u>

The control rod drop accident is defined as a power excursion caused by accidental removal of a control rod from the core at a more rapid rate than can be achieved by the use of the contol rod drive mechanism. In the control rod drop accident, a fully inserted control rod is assumed to fall out of the core after becoming disconnected from its drive and after the drive has been removed to the fully withdrawn position. There are many design safeguards which minimize the risk of this accident, e.g., the control rod design minimizes probability of sticking in the core and separation from a control rod drive, rod coupling is verified by changes in neutron flux during criticality and a rod bottoming out indication before criticality, the rod velocity limiter limits rod drop velocity, and the control rod worth minimizer interlock system consists of a computer program which monitors the control rod withdrawal sequence and actuates interlocks to prevent abnormal control rod patterns and high rod worths.

For reloads, the significant parameters of the control rod drop accident are compared to values used in a bounding analysis. If the reload specific parameters are conservatively compared to the bounding analysis values, the consequences of the control rod drop are less severe than those of the bounding analysis. In Figures 6-1 through 6-3 of Reference 1 the licensee has shown that for Cycle 4 operation of Quad Cities Unit 2 the Doppler Coefficient as a function of fuel temperature is more negative and the reactivity insertion due to a dropped in-sequence control rod versus rod position is smaller than bounding curves of these quantities presented in Reference 3. Furthermore, the scram reactivity as a function of time at 286°C (Figure 6-5 of Reference 1) will be greater than the corresponding bounding function presented in Reference 3. At 20°C the specific scram reactivity function (Figure 6-4 of Reference 1) is slightly greater than the corresponding bounding analysis curve. Thus, all the significant parameters for the control rod drop accident are within the values established for the bounding analysis.

The previous Quad Cities Unit 2 Technical Specifications require that the rod worth minimizer be operable for all powers, less than 10% of rated. This requirement is necessary in order to minimize control rod worths at the lower powers. The lower power levels result in greater fuel enthalpy rises during the rod drop accidents. Previously the accident analyses were performed at <10% of rated power. Since then this basis (analyses) has changed to ≤20% of rated power. Therefore, the technical specification on rod worth minimizer operability has been changed to ≤20% of rated.

Based on the analysis presented in Reference 3 and the discussion above, it is concluded that no in-sequence rod drop accident will lead to peak fuel enthalpies greater than the 280 cal/gm design basis for Quad-Cities Unit 2 Cycle 4.

2.4.5 Fuel Handling Accident

The fuel handling accident was addressed in the staff's Safety Evaluation Report (SER) on the FSAR dated August 25, 1971 (Reference 16) and in the staff's evaluation of the topicals on the generic reload for 8x8 fuel (Reference 3). In the generic reload evaluation, the staff stated that the mechanical analysis of the fuel handling accident should be better justified. However, the conclusions drawn in the staff's evaluation of the generic reload that the amount of fission products released from 8x8 assemblies in a refueling accident would not be significantly greater than from the 7x7 assemblies is not changed by this reload; and the conclusions of the SER (Reference 16) that the dose consequences of a fuel handling accident are appropriately within 10 CFR Part 100 guidelines are not changed.

2.5 Overpressure Analysis

The licensee has presented an analysis to demonstrate that during the most severe overpressure event an adequate margin (62 psi) exists between the peak vessel pressure and the ASME code allowable vessel pressure which is 110% of the vessel design pressure (Reference 1). The event analyzed was the closure of all main steam line isolation valves with indirect (high flux) scram.

The input to the calculations is listed in Table 6-1 of Reference 1 at end of cycle conditions for void coefficient, Doppler coefficient and scram characteristics.

The licensee referenced a sensitivity study (Reference 12) which demonstrates that should the transient be initiated at the maximum pressure permitted by the high pressure trip point rather than that assumed for the analysis there would be a reduction in the margin to the pressure limit of approximately 20 psi. It has also been shown that the increase in peak vessel pressure during an MSIV closure due to a failed safety valve would not reduce the margin to the limit by more than approximately 15 psi (Reference 13).

Furthermore, it has been demonstrated that should the MSIV transient be initiated at a value of reactor power slightly above the value assumed for the analysis (because of uncertainties in monitoring of power) there would not be a significant reduction in margin (approximately 10 psi at 102% power) (Reference 14).

Based on the analysis and the sensitivity studies submitted, the overpressure analysis for Quad-Cities Unit 2 for Cycle 4 has been found acceptable.

2.6 Thermal Hydraulic Stability Analysis

The thermal hydraulic stability analyses and results are described in References 3 and 1, respectively. The results of the Cycle 4 analysis show that for both the 7x7 and 8x8 fuel the channel hydrodynamic stability, at either rated power and flow conditions or at the low end of the flow control range, is within the General Electric Company's operational design guide in terms of decay ratio. Calculations were also performed by the licensee to assess the reactor power dynamic response at the two afore-mentioned reactor operating conditions. The results of this analysis showed that the reactor core decay ratios at both conditions are well within the operational design guide decay ratio. These results are acceptable to the NRC staff.

The NRC staff has expressed generic concerns regarding the least stable reactor condition allowed by Technical Specifications. The concerns are motivated by increasing decay ratios as equilibrium fuel cycles are approached and as fuel designs improve. The staff concerns relate to both the consequences of operating at an ultimate decay ratio and the capacity of analytical methods to accurately predict decay ratios. The General Electric Company is addressing the staff concerns through meetings, topical reports and a test program.

Until this issue has been resolved generically, the staff has imposed a requirement on Quad-Cities Unit 2 which will restrict planned operations in the natural circulation flow mode. The licensee has agreed to this Technical Specification limitation. The restriction will provide a significant increase in the reactor core stability margins during Cycle 4. On the basis of the foregoing, the NRC staff considers the thermal-hydraulic stability of Quad-Cities Unit 2 to be acceptable.

2.7 Recirculation Pump Startup From The Natural Circulation Operational Mode

During a recent BWR reload review (Reference 15), the question of recirculation pump startup from the natural circulation operational mode was raised. This pump startup could increase flow, collapse moderator voids, and subsequently result in a reactivity insertion transient. The consequences of such an accident sequence have not been previously evaluated. Therefore, authorization to operate in this fashion would require additional analyses as to this accident sequence and its consequences. In the absence of this information, the Technical Specifications have been amended to eliminate the potential for such an accident. We find this measure to be acceptable.

3.0 Physics Startup Testing

As documented in Reference 17, the licensee will conduct physics startup tests which in addition to verifying the predicted shutdown margin, will provide assurance that the incore monitoring instrumentation is functioning properly, that the process computer is programmed correctly, and that the core is loaded as intended. These tests will provide additional assurance that the Cycle 4 core as loaded is consistent with the physics input submittal (Reference 1). The results of the tests will be submitted to the staff within 45 days of startup. The staff finds the licensee's plan for confirmatory testing and documentation acceptable.

4.0 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.

5.0 Conclusions

Based on our evaluation of the reload application and available information, we conclude that it is acceptable for the licensee to proceed with Cycle 4 operation of Quad-Cities Unit 2 in the manner proposed.

We have reviewed the proposed changes to the Technical Specifications and find them acceptable.

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 change 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 this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: March 8, 1978

REFERENCES

1. "General Electric Boiling Water Reactor Reload 3 Licensing Submittal for Quad-Cities Nuclear Power Station Unit 2," NEDO-24063, September 1977.

2. Letter from R. L. Bolger, CECo, to E. G. Case, NRC, dated December 2, 1977.

3. "General Electric Boiling Water Reactor Generic Reload Application For 8x8 Fuel," NEDO-20360, Rev. 1, Supp. 4, April 1, 1976.

4. Status Report on the Licensing Topical Report, "General Electric Boiling Water Reactor Generic Reload Application for 8x8 Fuel," NEDO-20360, Revision 1 and Supplement 1 by Division of Technical Review, Office of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission, April 1975.

5. "General Electric Thermal Analysis Basis (GETAB): Data Correlation and Design Application," NEDO-10958, November 1973.

6. Letter from J. A. Hinds, GE, to W. Butler, AEC, transmitting Responses to the Third Set of AEC Questions on the General Electric Licensing Topical Reports, NEDO-10958 and NEDE-10958, dated July 24, 1974.

7. Letter from D. L. Ziemann, NRC, to R. L. Bolger, CECo, transmitting Order for Modification of License, dated March 11, 1977.

8. Letter from K. R. Goller, NRC, to G. G. Sherwood, GE, transmitting "Safety Evaluation for General Electric ECCS Evaluation Model Modifications," dated April 12, 1977.

9. Letter from D. G. Eisenhut, NRC, to E. D. Fuller, GE, "Documentation of the Reanalysis Results for the Loss-of-Coolant Accident (LOCA) of Lead and Non-Lead Plants," dated June 30, 1977.

10. Letter from R. L. Bolger, CECo, to E. G. Case, NRC, dated October 3, 1977 transmitting "Loss-of-Coolant Accident Analysis Report For Dresden Units 2, 3 and Quad-Cities Units 1,2 Nuclear Power Stations (Lead Plant)", NEDO-24046, August 1977.

11. Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment No. DPR-19 to Facility Operating License No. DPR-19, Dresden Unit 2, dated December 2, 1977.

12. Letter from M. S. Turbak, CECo, to D. K. Davis, NRC, dated April 25, 1977.

13. Letter from I. F. Stuart, GE, to V. Stello, NRC, dated December 23, 1975.

14. Letter from R. L. Gridley, GE, to D. G. Eisenhut, NRC, dated September 12, 1977.

15. Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment No. 33 to License No. DPR-49, Duane Arnold Energy Center (Docket No. 50-331), dated May 6, 1977.

16. "Safety Evaluation by Division of Reactor Licensing, U.S. AEC in the matter of Commonwealth Edison Company's Quad Cities Units 1 & 2," August 25, 1971.

17. Letter from R. L. Bolger, CECo, to E. G. Case, NRC, dated March 1, 1978.

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. 43 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 the license and Technical Specifications appended thereto 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 repsonse to the plant specific analysis for reload 3, and (3) modified License Condition 3.C to reflect end-of-cycle scram reactivity conditions for reload 3.

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.

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 December 2, 1977, as supplemented February 28 and March 1, 1978, (2) Amendment No. 43 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 61265. 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 8th day of March 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors