

January 24, 1984

Docket No. 50-265

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

Dear Mr. Farrar:

The Commission has issued the enclosed Amendment No. 80 to Facility Operating License No. DPR-30 for Quad Cities Nuclear Power Station, Unit 2. This amendment is in response to your letters dated July 5 and July 15, 1983.

This amendment (1) reduces the current minimum critical power ratio (MCPR) operating limit to provide additional operating margin during the upcoming operating cycle, (2) explicitly applies a maximum average planar linear heat generation rate (MAPLHGR) curve to a fuel type to be used in the upcoming operating cycle, and (3) changes the pressure setpoints for three safety-relief valves as a result of Mark I containment modifications.

A copy of the Safety Evaluation is also enclosed.

Sincerely,



Roby B. Bevan, Project Manager  
Operating Reactors Branch #2  
Division of Licensing

Enclosures:

1. Amendment No. 80 to License No. DPR-30
2. Safety Evaluation

cc w/enclosures:  
See next page

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RDiggs	NSIC	Gray	Extra - 5		

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Mr. Dennis L. Farrar  
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UNITED STATES  
NUCLEAR REGULATORY COMMISSION  
WASHINGTON, D. C. 20555

COMMONWEALTH EDISON COMPANY

AND

IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

DOCKET NO. 50-265

QUAD CITIES NUCLEAR POWER STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 80  
License No. DPR-30

1. The Nuclear Regulatory Commission (the Commission) has found that:
  - A. The applications for amendment by Commonwealth Edison Company (the licensee) dated July 5 and July 15, 1983, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;
  - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
  - C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;
  - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; 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 2.C.(2) of Facility Operating License No. DPR-30 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 80, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION



Vernon L. Rooney, Acting Chief  
Operating Reactors Branch #2  
Division of Licensing

Attachment:  
Changes to the Technical  
Specifications

Date of Issuance: January 24, 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 80

FACILITY OPERATING LICENSE NO. DPR-30

DOCKET NO. 50-265

Revise the Appendix "A" Technical Specifications as follows:

Remove

1.2/2.2-1

3.5/4.5-10

Fig. 3.5-1, Sheet 3

3.6/4.6-4

Replace

1.2/2.2-1

3.5/4.5-10

Fig. 3.5-1, Sheet 3

3.6/4.6-4

QUAD-CITIES  
DPR-30

1.2/2.1 REACTOR COOLANT SYSTEM

**SAFETY LIMIT**

**Applicability:**

Applies to limits on reactor coolant system pressure.

**Objective:**

To establish a limit below which the integrity of the reactor coolant system is not threatened due to an overpressure condition.

**LIMITING SAFETY SYSTEM SETTING**

**Applicability:**

Applies to trip settings of the instruments and devices which are provided to prevent the reactor system safety limits from being exceeded.

**Objective:**

To define the level of the process variables at which automatic protective action is initiated to prevent the safety limits from being exceeded.

**SPECIFICATIONS**

A. The reactor coolant system pressure as measured by the vessel steam space pressure indicator shall not exceed 1345 psig at any time when irradiated fuel is present in the reactor vessel.

A. Reactor coolant high-pressure scram shall be  $\leq 1060$  psig.

B. Primary system safety valve nominal settings shall be as follows:

- 1 valve at 1135 psig
- 2 valves at 1240 psig
- 2 valves at 1250 psig
- 4 valves at 1260 psig

"Target Rock combination safety/relief valve

The allowable setpoint error for each valve shall be  $\pm 1\%$ .

Quad Cities  
DPR-30

within the prescribed limit 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. Maximum allowable LHGR for all 8X8 fuel types is 13.4 KW/ft.\*

K. Minimum Critical Power Ratio (MCPR)

During steady-state operation at rated core flow, MCPR shall be greater than or equal to:

$$1.34 \text{ for } T_{ave} \leq 0.73 \text{ secs}$$

$$1.39 \text{ for } T_{ave} \geq 0.86 \text{ secs}$$

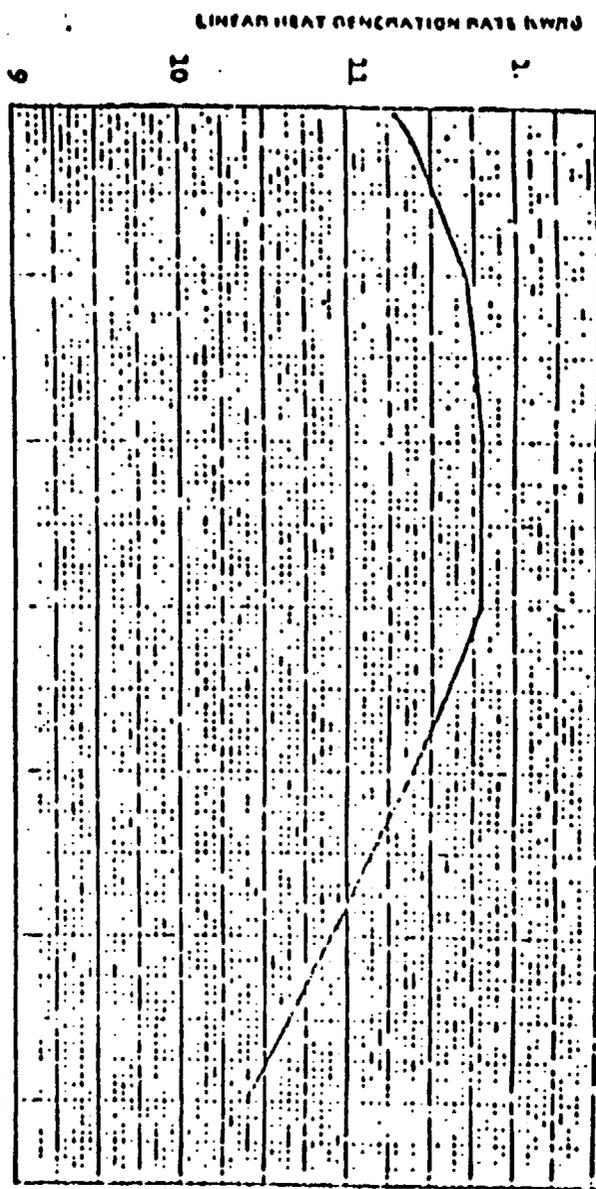
$$0.385 T_{ave} + 1.059 \\ \text{for } 0.73 < T_{ave} < 0.86 \text{ secs}$$

where  $T_{ave}$  = mean 20% scram insertion time for all surveillance data from Specification 4.3.C which has been generated in the current cycle.

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. If 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 the purpose of the end-of-cycle 6 Barrier Fuel Ramp Test, the steady-state LHGR for the Barrier Ramp Cell fuel may exceed the maximum allowable LHGR identified in Technical Specification 3.5.J by no more than 10 percent (14.7 KW/ft), effective from initiation of the test until the end of operation Cycle 6 shutdown.

23  
 ER-30  
 Grid Codes Unit 2  
 Fuel Type PLUO203



MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (kW/m) VERSUS PLANAR AVERAGE EXPOSURE

Grid Codes Unit 2  
 Fuel Type PLUO203  
 and BP0DRB265H

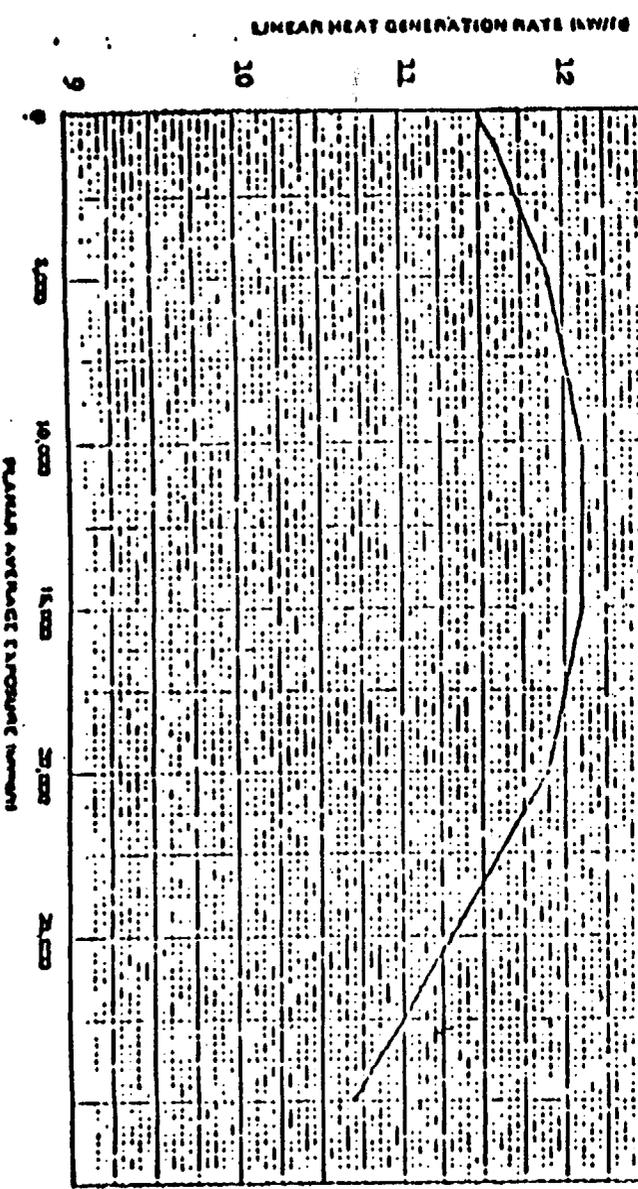


Figure 3.5-1  
 (Sheet 3 of 5)  
 Maximum Average Planar Linear Heat Generation Rate (kW/m) vs. Planar Average Exposure

2. Both the sump and air sampling systems shall be operable during reactor power operation. From and after the date that one of these systems is made or found to be inoperable for any reason, reactor power operation is permissible only during the succeeding 7 days.
3. If the conditions in 1 or 2 above cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

**E. Safety and Relief Valves**

1. Prior to reactor startup for power operation, during reactor power operating conditions, and whenever the reactor coolant pressure is greater than 90 psig and temperature greater than 320° F, all nine of the safety valves shall be operable. The solenoid-activated pressure valves shall be operable as required by Specification 3.5.D.
2. If Specification 3.6.E.1 is not met, the reactor shall remain shut down until the condition is corrected or, if in operation, an orderly shutdown shall be initiated and the reactor coolant pressure and temperature shall be below 90 psig and 320° F within 24 hours.

**F. Structural Integrity**

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

**E. Safety and Relief Valves**

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

Number of Valves	Setpoint (psig)
1	1135 <sup>(1)</sup>
2	1240
2	1250
4	1260

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

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

Number of Valves	Setpoint (psig)
1	≤ 1135 <sup>(1)</sup>
2	≤ 1115
2	≤ 1135

<sup>(1)</sup>Target Rock combination safety/relief valve.

**F. Structural Integrity**

The nondestructive inspections listed in Table 4.6-1 shall be performed as specified in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1971 Edition, Summer 1971 Addenda. The results obtained from compliance with this specification will be evaluated after 5 years and the conclusions will be reviewed with the NRC.



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 80 TO FACILITY LICENSE NO. DPR-30

COMMONWEALTH EDISON COMPANY

IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

QUAD CITIES NUCLEAR POWER STATION, UNIT 2

DOCKET NO. 50-265

1.0 Introduction

By letters dated July 5 and July 15, 1983, Commonwealth Edison Company (CECo, the licensee) proposed changes to the Technical Specifications (TS) for Quad Cities Unit 2 (see References 1 and 2). Although a previously approved amendment (Reference 3) sufficiently bounded the upcoming operating Cycle 7, additional changes are requested in order to (1) reduce the current minimum critical power ratio (MCPR) operating limit to provide additional operating margin, (2) explicitly apply a maximum average planar linear heat generation rate (MAPLGHR) curve to a fuel type to be used in the upcoming operating cycle, and (3) change the pressure setpoints for three safety-relief valves as a result of Mark I containment modifications.

2.0 Evaluation

2.1 Operating Limit Minimum Critical Power Ratio

The operating limit minimum critical power ratio (OL-MCPR) for Quad Cities Unit 2 uses the computer code ODYN Option B approach, taking credit for actual scram time performance compared to the scram time TS surveillance limit  $\tau_B$  (Reference 6). Under Option B, when the actual scram time exceeds the TS surveillance limit  $\tau_B$ , a MCPR penalty is applied in the form of a linearly increasing value from the Option B value (currently 1.37) to the Option A limiting value of 1.42.

The licensee has requested a change in the Technical Specification OL-MCPR for the upcoming operating Cycle 7. The transient analyses performed by General Electric (GE) for Cycle 7 show the most limiting transient to be the turbine trip without bypass event. For this event, the analysis yields an operating limit MCPR of 1.32. This is a less restrictive limit than the present limit of 1.37, and comes about from the reduced time-of-transfer of the recirculation pump motor/generator sets to offsite power during this transient. The shorter transfer time precludes circulation pump runup during the limiting operational transient, resulting in a reduced operating limit MCPR.

The licensee's request is for an amendment to reduce the TS limit MCPR from the current value of 1.37 to 1.34. This new value is more conservative than the value of 1.32 resulting from the analysis of the most limiting transient for Cycle 7, but still provides the desired improved operating margin.

We have reviewed the licensee's submittal (Reference 1) and the appropriate parts of the reference document (Reference 5). The supporting analysis was performed by GE using previously approved methodology and models (Reference 4). We find the proposed change in operating limit MCPR to be acceptable.

## 2.2 MAPLHGR Limits

The reference document containing the ECCS analysis for Quad Cities Units 1 and 2 has previously been approved by the NRC staff and continues to be the basis for MAPLHGR limits for fuel types used in these units (Reference 5). This document has been updated as appropriate for other fuel types by issuance of Errata and Addenda to Reference 5. Barrier fuel type BP8DRB265H will be used in the core for the upcoming operating cycle. This fuel is of the same nuclear design as non-barrier fuel type P8DRB265L, which has previously been approved for use in Quad Cities Unit 2. Since the two fuel types are of the same nuclear design, the MAPLHGR curve for the previously approved fuel type P8DRB265L applies also to barrier fuel type BP8DRB265H, as documented by GE in Errata and Addenda No. 10 to Reference 5, and the licensee's request to incorporate this into their Technical Specifications is acceptable.

## 2.3 Safety/Relief Valve Setpoint Changes

In analyses associated with the Mark I containment program, it was discovered that the torus could be subjected to excessive loads if a relief valve actuation occurs shortly after closure. This loading is the result of a water leg entrapped in the relief valve discharge line from the vacuum caused by the condensed steam in this line. To prevent such loadings, a modification to the electromatic relief (EMR) valve logic is currently being installed which will delay automatic opening of two EMR valves up to ten seconds from the last closure of the valve. In order to maintain very similar overall Target Rock and EMR valve performance with the logic change and prevent excessive loading, the two affected EMR valves TS pressure setpoints must be lowered so that they are the first to actuate and the setpoint of one valve (Target Rock) must be raised.

For the limiting transient (load rejection w/o bypass) the pressurization is estimated by GE to be milder because there is a net relief valve setpoint decrease, thus slightly lowering the peak pressure and power for the transient.

The ASME overpressurization event (no credit for EMR valve actuation) is estimated by GE to have slightly increased peak pressure (no more than 5 psi) because the Target Rock SRV setpoint is slightly increased. This peak pressure increase is insignificant compared to the calculated margin to 1375 psig of 50 psi.

The delay in actuation between successive valve openings is required because of the possibility of an automatic depressurization immediately following opening of a valve for pressure relief. The calculated minimum acceptable delay time reported by the Licensee is 6.2 seconds. This compares conservatively with the ten-second delay proposed for the TS, with ample margin.

We have reviewed the proposed changes in the SRV setpoints and the proposed delay for successive actuations and find the change to have minimal effect on safety limits, and therefore, we find the proposed changes to be acceptable.

### 3.0 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.

### 4.0 Conclusion

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: Roby Bevan

Dated: January 24, 1984

References

1. Letter, Ryback (CECo) to Denton (NRC), dated July 5, 1983.
2. Letter, Ryback (CECo) to Denton (NRC), dated July 15, 1983.
3. Letter, Ippolito (NRC) to DeGeorge (CECo), dated December 23, 1983.
4. NEDO-24011-A-1, "General Electric BWR Generic Reload Fuel Application", July 1979.
5. NEDO-24146A, "LOCA Report for Dresden 2 and 3 and Quad Cities 1 and 2 Nuclear Power Stations", Revision 1, April 1979, as modified Errata and Addenda Nos. 1 through 10.
6. Letter, Bucholz (GE) to Check (NRC) "ODYN Adjustment Methods for Determination of Operating Limits", dated January 19, 1981.
7. Letter, Bevan (NRC) to DeGeorge (CECo), dated December 15, 1982.