Commonwealth Edison
One First National Plaza, Chicago, Illinois
Address Reply to: Post Office Box 767
Chicago, Illinois 60690

February 14, 1979

Director of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Quad-Cities Station Unit 1
Proposed Amendment to License
and Appendix A, Technical
Specifications, for Facility
Operating License DPR-29
NRC Docket No. 50-254

Reference (a): C. Reed letter to Director of
NRR dated November 20, 1978

Dear Sir:

Reference (a) transmitted a proposed amendment to the License and Appendix A, Technical Specifications, to Facility Operating License DPR-29 to support core reload No. 4 at Quad-Cities Station Unit 1.

Subsequent to that transmittal and in telephone conversations with the NRC Staff, additional changes to support core reload No. 4 have been identified. The changes include corrections to water level references which reflect the longer fuel length of the 8x8R retrofit fuel and revision of Paragraph 3.C on Page 4 of the License to limit operation in the coastdown mode to 40% power. These changes are identified in Enclosure I and have received on-site and off-site review and approval.

Since these changes are additions to or revisions of a previous submittal currently under review, Commonwealth Edison has determined that an additional fee remittance in accordance with 10 CFR 120 is not required.

Please address any questions concerning this matter to this office.

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Three (3) signed originals and thirty-seven (37) copies of this transmittal are provided for your use.

Very truly yours,

Robert F. Janecek
for Cordell Reed
Assistant Vice-President

enclosure

SUBSCRIBED and SWORN to
before me this 14th, day
of February, 1979.

Notary Public

ENCLOSURE I

QUAD-CITIES UNIT 1

NRC DOCKET NO. 50-254

3. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

   A. Maximum Power Level

   Commonwealth Edison is authorized to operate Quad-Cities Unit No. 1 at power levels not in excess of 2511 megawatts (thermal).

   B Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 49, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

   C. Restrictions

   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. 4 licensing submittal for Quad Cities Unit No. 1 (NEDO 24145) Plant operation shall be limited to the operating plan described therein. Subsequent operation in the coastdown mode is permitted to 40% power.

   D. Equalizer Valve Restriction

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

4. This license is effective as of the date of issuance, and shall expire at midnight, February 15, 2007.

FOR THE ATOMIC ENERGY COMMISSION

A Giambusso

A. Giambusso, Deputy Director for Reactor Projects
Directorate of Licensing

Enclosures: Appendices A and B--
Technical Specifications

Date of Issuance: December 14, 1972

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.

*Top of active fuel is defined to be 360 inches above vessel zero (see Bases 3.2).

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 ≥ 144 inches above the top of the active fuel* at normal operating conditions.

D. Reactor low water level ECCS initiation shall be 84 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.

settings which maintain equivalent safety margins when the total peak factor (TPF) exceeds the LTPF. Specification 3.5.J established the LHGR maximum which cannot be exceeded under steady power operation.

B. Core Thermal Power Limit (Reactor Pressure<800 psia)

At pressures below 800 psia, the core elevation pressure drop (0 power, 0 flow) is greater than 4.56 psi. At low powers and flows this pressure differential is maintained in the bypass region of the core. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low powers and flows will always be greater than 4.56 psi. Analyses show that with a flow of 28 x 10^3 lb/hr bundle flow, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus the bundle flow with a 4.56-psi driving head will be greater than 28 x 10^3 lb/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. At 25% of rated thermal power, the peak powered bundle would have to be operating at 3.86 times the average powered bundle in order to achieve this bundle power. Thus, a core thermal power limit of 25% for reactor pressures below 800 psia is conservative.

C. Power Transient

During transient operation the heat flux (thermal power-to-water) would lag behind the neutron flux due to the inherent heat transfer time constant of the fuel, which is 8 to 9 seconds. Also, the limiting safety system scram settings are at values which will not allow the reactor to be operated above the safety limit during normal operation or during other plant operating situations which have been analyzed in detail. In addition, control rod scrams are such that for normal operating transients, the neutron flux transient is terminated before a significant increase in surface heat flux occurs. Scram times of each control rod are checked each refueling outage, and at least every 32 weeks, 50% are checked to assure adequate inseration times. Exceeding a neutron flux scram setting and a failure of the control rods to reduce flux to less than the scram setting within 1.5 seconds does not necessarily imply that fuel is damaged; however, for this specification, a safety limit violation will be assumed any time a neutron flux scram setting is exceeded for longer than 1.5 seconds.

If the scram occurs such that the neutron flux dwell time above the limiting safety system setting is less than 1.7 seconds, the safety limit will not be exceeded for normal turbine or generator trips, which are the most severe normal operating transients expected. These analyses show that even if the bypass system fails to operate, the design limit of MCPR = 1.06 is not exceeded. Thus, use of a 1.5-second limit provides additional margin.

The computer provided has a sequence annunciation program which will indicate the sequence in which scrams occur, such as neutron flux, pressure, etc. This program also indicates when the scram setpoint is cleared. This will provide information on how long a scram condition exists and thus provide some measure of the energy added during a transient. Thus, computer information normally will be available for analyzing scrams; however, if the computer information should not be available for any scram analysis, Specification 1.1.C.2 will be relied on to determine if a safety limit has been violated.

During periods when the reactor is shut down, consideration must also be given to water level requirements due to the effect of decay heat. If reactor water level should drop below the top of the active fuel during this time, the ability to cool the core is reduced. This reduction in core-cooling capability could lead to elevated cladding temperatures and cladding perforation. The core will be cooled sufficiently to prevent cladding melting should the water level be reduced to two-thirds the core height. Establishment of the safety limit at 12 inches above the top of the fuel* provides adequate margin. This level will be continuously monitored whenever the recirculation pumps are not operating.

*Top of active fuel is defined to be 360 inches above vessel zero (see Bases 3.2).

TABLE 3.1-4

NOTES FOR TABLES 3.1-1, 3.1-2, AND 3.1-3

1. There shall be two operable trip systems or one operable and one tripped system for each function.

2. If the first column cannot be met for one of the trip systems, that trip system shall be tripped. If the first column cannot be met for both trip systems, the appropriate actions listed below shall be taken:

   A. Initiate insertion of operable rods and complete insertion of all operable rods within 4 hours.

   B. Reduce power level to IRM range and place mode switch in the Startup/Hot Standby position within 8 hours.

   C. Reduce turbine load and close main steamline isolation valves within 8 hours.

3. An APRM will be considered inoperable if there are fewer than 2 LPRM inputs per level or there are less than 50% of the normal complement of LPRM's to an APRM.

4. Permissible to bypass, with control rod block for reactor protection system reset in refuel and shutdown positions of the reactor mode switch.

5. Not required to be operable when primary containment integrity is not required.

6. The design permits closure of any one line without a scram being initiated.

7. Automatically bypassed when reactor pressure is <1060 psig.

8. The +8-inch trip point is the water level as measured by the instrumentation outside the shroud. The water level inside the shroud will decrease as power is increased to 100% in comparison to the level outside the shroud to a maximum of 7 inches. This is due to the pressure drop across the steam dryer. Therefore, at 100% power, an indication of +8-inch water level will actually be +1 inch inside the shroud. 1 inch on the water level instrumentation is ≥504" above vessel zero. (see Bases 3.2)

9. Permissible to bypass when first stage turbine pressure is less than that which corresponds to 45% rated steam flow. (<400 psi)

10. Trips upon actuation of the fast-closure solenoid which trips the turbine control valves.

11. The APRM downscale trip function is automatically bypassed when the IRM instrumentation is operable and not high.

12. Channel shared by the reactor protection and containment isolation system.

3.2 LIMITING CONDITIONS FOR OPERATION BASES

In addition to reactor protection instrumentation which initiates a reactor scram, protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the operator's ability to control, or terminates operator errors before they result in serious consequences. This set of specifications provides the limiting conditions of operation for the primary system isolation function, initiation of the emergency core cooling system, control rod block, and standby gas treatment systems. The objectives of the specifications are (1) to assure the effectiveness of the protective instrumentation when required by preserving its capability to tolerate a single failure of any component of such systems even during periods when portions of such systems are out of service for maintenance, and (2) to prescribe the trip settings required to assure adequate performance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations. Some of the settings on the instrumentation that initiates or controls core and containment cooling have tolerances explicitly stated where the high and low values are both critical and may have a substantial effect on safety. It should be noted that the setpoints of other instrumentation, where only the high or low end of the setting has a direct bearing on safety, are chosen at a level away from the normal operating range to prevent inadvertent actuation of the safety system involved and exposure to abnormal situations.

Isolation valves are installed in those lines that penetrate the primary containment and must be isolated during a loss-of-coolant accident so that the radiation dose limits are not exceeded during an accident condition. Actuation of these valves is initiated by the protective instrumentation which senses the conditions for which isolation is required (this instrumentation is shown in Table 3.2-1). Such instrumentation must be available whenever primary containment integrity is required. The objective is to isolate the primary containment so that the guidelines of 10 CFR 100 are not exceeded during an accident.

The instrumentation which initiates primary system isolation is connected in a dual bus arrangement. Thus the discussion given in the bases for Specification 3.1 is applicable here.

The low-reactor water level instrumentation is set to trip at ≥8 inches on the level instrument (top of active fuel is defined to be 360 inches above vessel zero) after allowing for the full power pressure drop across the steam dryer the low level trip is at 504 inches above vessel zero, or 144 inches above top of active fuel. Retrofit 8x8 fuel has an active fuel length 1.24 inches longer than earlier fuel designs, however, present trip setpoints were used in the LOCA analysis (NEDO 24146). This trip initiates closure of Group 2 and 3 primary containment isolation valves but does not trip the recirculation pumps (reference SAR, Section 7.7.2). For a trip setting of 504 inches above vessel zero and a 60-second valve closure time, the valves will be closed before perforation of the cladding occurs even for the maximum break; the setting is, therefore, adequate.

The low-low reactor level instrumentation is set to trip when reactor water level is 444 inches above vessel zero (with top of active fuel defined as 360 inches above vessel zero, -59" is 84 inches above the top of active fuel). This trip initiates closure of Group 1 primary containment isolation valves (reference SAR Section 7.7.2.2) and also activates the ECC subsystems, starts the emergency diesel generator, and trips the recirculation pumps. This trip setting level was chosen to be high enough to prevent spurious operation but low enough to initiate ECCS operation and primary system isolation so that no melting of the fuel cladding will occur and so that postaccident cooling can be accomplished and the guidelines of 10 CFR 100 will not be exceeded. For the complete circumferential break of a 28-inch recirculation line and with the trip setting given above, ECCS initiation and primary system isolation are initiated and in time to meet the above criteria (reference SAR Sections 6.2.7.1 and 14.2.4.2). The instrumentation also covers the full spectrum of breaks and meets the above criteria (reference SAR Section 6.2.7.1).

The high-drywell pressure instrumentation is a backup to the water level instrumentation and, in addition to initiating ECCS, it causes isolation of Group 2 isolation valves. For the breaks discussed above, this instrumentation will initiate ECCS operation at about the same time as the low low water level instrumentation; thus the results given above are applicable here also. Group 2 isolation valves include the drywell vent, purge, and sump isolation valves. High-drywell pressure activates only these valves because high drywell pressure could occur as the result of non-safety-related causes such as not purging the drywell air during startup. Total system isolation is not desirable for these conditions, and only the valves in Group 2 are required to close. The low low water level instrumentation initiates protection for the full spectrum of loss-of-coolant accidents and causes a trip of Group 1 primary system isolation valves.

TABLE 3.2-1

INSTRUMENTATION THAT INITIATES PRIMARY CONTAINMENT ISOLATION FUNCTIONS

Minimum Number of Operable or Tripped Instrument Channels[1]	Instruments	Trip Level Setting	Action[2]
4	Reactor low water[5]	>144 inches above top of active fuel*	A
4	Reactor low low water	≥84 inches above top of active fuel *	A
4	High drywell pressure[5]	≤2 psig[3]	A
16	High flow main steamline[5]	≤120% of rated steam flow	B
16	High temperature main steamline tunnel	≤200° F	B
4	High radiation main steamline tunnel[6]	≤7 x normal rated power background	B
4	Low main steam pressure[4]	≥850 psig	B
4	High flow RCIC steamline	≤300% of rated steam flow	C
16	RCIC turbine area high temperature	≤200° F	C
4	High flow HPCI steamline	≤300% of rated steam flow	D
16	HPCI area high temperature	≤200° F	D

Notes

1. Whenever primary containment integrity is required, there shall be two operable or tripped systems for each function, except for low-pressure main steamline which only need be available in the Run position.
2. Action: If the first column cannot be met for one of the trip systems, that trip system shall be tripped.

   If the first column cannot be met for both trip systems, the appropriate actions listed below shall be taken:

   A. Initiate an orderly shutdown and have the reactor in Cold Shutdown condition in 24 hours.

   B. Initiate an orderly load reduction and have reactor in Hot Standby within 8 hours.

   C. Close isolation valves in RCIC system.

   D. Close isolation valves in HPCI subsystem.
3. Need not be operable when primary containment integrity is not required.
4. The isolation trip signal is bypassed when the mode switch is in Refuel or Startup/Hot Shutdown.
5. This instrumentation also isolates the control room ventilation system.
6. This signal also automatically closes the mechanical vacuum pump discharge line isolation valves.

*Top of active fuel is defined as 360" above vessel zero for all water levels used in the LOCA analysis (see Bases 3.2).

TABLE 3.2-2

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum Number of Operable or Tripped Instrument Channels[1]	Trip Function	Trip Level Setting	Remarks
4	Reactor low low water level	≥84 inches (+4 inches/-0 inch) above top of active fuel *	1. In conjunction with low-reactor pressure initiates core spray and LPCI. 2. In conjunction with high-drywell pressure 120-second time delay and low-pressure core cooling interlock initiates auto blowdown. 3. Initiates HPCI and RCIC. 4. Initiates starting of diesel generators.
4[4]	High-drywell pressure[2],[3]	≤2 psig	1. Initiates core spray, LPCI, HPCI, and SGTS. 2. In conjunction with low low water level, 120-second time delay, and low-pressure core cooling interlock initiates auto blowdown. 3. Initiates starting of diesel generators. 4. Initiates isolation of control room ventilation.
2	Reactor low pressure	300 psig≤p≤350 psig	1. Permissive for opening core spray and LPCI admission valves. 2. In conjunction with low low reactor water level initiates core spray and LPCI.
	Containment spray interlock		Prevents inadvertent operation of containment spray during accident conditions.
2[3]	2/3 core height	≥2/3 core height	
4[3]	containment high pressure	0.5 psig≤p≤1.5 psig	
2	Timer auto blowdown	≤120 seconds	In conjunction with low low reactor water level, high-drywell pressure, and low-pressure core cooling interlock initiates auto blow-down.
4	Low-pressure core cooling pump discharge pressure	75 psig≤p≤100 psig	Defers APR actuation pending confirmation of low-pressure core cooling system operation.
2	Undervoltage on emergency buses	N/A	1. Initiates starting of diesel generators. 2. Permissive for starting ECCS pumps. 3. Removes nonessential loads from buses.

*Top of active fuel is defined as 360" above vessel zero for all water levels used in the LOCA analysis.