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. 130
License No. DPR-30

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

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated June 28, 1991, 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. of Facility Operating License No. DPR-30 is hereby amended to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 130 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective immediately, to be implemented during the eleventh refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

Richard J. Barrett, Director
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: February 21, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 130

FACILITY OPERATING LICENSE NO. DPR-30

DOCKET NO. 50-265

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change.

REMOVE	INSERT
3.2/4.2-6a	3.2/4.2-6a
3.2/4.2-7	3.2/4.2-7
3.2/4.2-8	3.2/4.2-8
3.2/4.2-11	3.2/4.2-11
---	3.2/4.2-11a
3.7/4.7-21	3.7/4.7-21
---	3.7/4.7-21a
3.7/4.7-22	3.7/4.7-22

The RCIC and the HPCI high flow and temperature instrumentation are provided to detect a break in their respective piping. A trip of this instrumentation results in closure of the RCIC or HPCI steam supply isolation valves. The trip logic for this function is similar to that for the main steamline isolation valves, thus all sensors are required to be operable or in a tripped condition to meet single-failure criteria. The trip settings of 170°F and 300% of design flow and valve closure time are such that core uncovery is prevented and fission product release is within limits. In addition, the steam supply valves for each system are closed on low steamline pressure to provide primary containment isolation when the reactor pressure, as sensed in the system steamlines, is below the required pressure for turbine operation.

Operation of the HPCI turbine will continue as long as reactor pressure is above 150 psig. When the reactor pressure falls below 150 psig, the speed of the turbine-pump unit will decrease and would gradually be slowed due to stop friction and windage losses at low reactor pressures. The low reactor pressure isolation setpoint was developed in accordance with NEDC-31336, "General Electric Instrument Setpoint Methodology," dated October, 1986. The trip setpoint of greater than or equal to 100 psig was calculated such that the isolation will occur on decreasing reactor pressure to provide primary containment isolation when the reactor pressure, as sensed in the system steamlines, is below the required pressure for turbine operation. The external vacuum breaker line for the HPCI turbine will isolate on low steamline pressure concurrent with high drywell pressure signals. The instrumentation and controls ensure the proper HPCI and primary containment response to a HPCI steamline break (isolation of the steamline supply valves only), a large break inside the containment (closure of the steam supply and vacuum relief isolation valves) and a small or intermediate size break inside of containment (steam supply and vacuum breaker isolation valves remain open for HPCI operation).

The instrumentation which initiates ECCS action is arranged in a one-out-of-two taken twice logic circuit. Unlike the reactor scram circuits, however, there is one trip system associated with each function rather than the two trip systems in the reactor protection system. The single-failure criteria are met by virtue of the fact that redundant core cooling functions are provided, e.g., sprays and automatic blowdown and high-pressure coolant injection. The specification requires that if a trip system becomes inoperable, the system which it activates is declared inoperable. For example, if the trip system for core spray A becomes inoperable, core spray A is declared inoperable and the out-of-service specifications of Specification 3.5 govern. This specification preserves the effectiveness of the system with respect to the single-failure criteria even during periods when maintenance or testing is being performed.

The control rod block functions are provided to prevent excessive control rod withdrawal so that MCPR does not go below the MCPR Fuel Cladding Integrity Safety Limit. The trip logic for this function is one out of n; e.g., any trip on one of the six APRM's, eight IRM's, four SRM's will result in a rod block. The minimum instrument channel requirements assure sufficient instrumentation to assure that the single-failure criteria are met. The minimum instrument channel requirements for the RBM may be reduced by one for a short period of time to allow for maintenance, testing, or calibration. This time period is only ~3% of the operating time in a month and does not significantly increase the risk of preventing an inadvertent control rod withdrawal.

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 withdrawal of control rods in the normal withdrawal sequence.

In the refuel and startup/hot standby modes, the APRM rod block function is set at 12% of rated power. 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., 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 is analyzed for each reload to assure that, with the specific trip settings, rod withdrawal is blocked before the MCPR reaches the fuel cladding integrity safety limit.

Below 30% power, the worst-case withdrawal of a single control rod without rod block action will not violate the fuel cladding integrity safety limit. Thus the RBM rod block function 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 the MCPR fuel cladding integrity safety limit.

A downscale indication on an APRM 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 $\leq$ 100 CPS and the detector not full 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 block provide 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 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 $\leq$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 Trains "A" and "B" is 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 streamline 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. Manual isolation capability is also provided. Isolation from high toxic chemical concentration has been added as a result of the "Control Room Habitability Study" submitted to the NRC in December 1981 in response to NUREG-0737 Item III D.3.4. As explained in Section 3 of this study, ammonia, chlorine, and sulphur dioxide detection capability has been provided. The setpoints chosen for the control room ventilation isolation are based on early detection in the outside air supply at the odor threshold, so that the toxic chemical will not achieve toxicity limit concentrations in the Control Room.

The radioactivie liquid and gaseous effluent instrumentation is provided to monitor the release of radioactive materials in liquid and gaseous effluents during releases. The alarm setpoints for the instruments are provided to ensure that the alarms will occur prior to exceeding the limits of 10 CFR 20.

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]	≤2.5 psig [3]	A
16	High flow main steamline[5]	≤140% of rated steam flow	B
16	High temperature main steamline tunnel	≤200° F	B
4	High radiation main steamline tunnel[6]	≤15 x normal rated power background (without hydrogen addition)	B
4	Low main steam pressure[4]	≥825 psig	B
2	High flow RCIC steamline	≤300 % of rated steam flow(7)	C
4	RCIC turbine area high temperature	≤170° F	C
2	High flow HPCI steamline	≤300% of rated steam flow(7)	D
4	HPCI area high temperature	≤170° F	D
4	HPCI Steamline pressure	≥100 psig	D

TABLE 3.2-1
(Cont.)

INSTRUMENTATION THAT INITIATES PRIMARY CONTAINMENT ISOLATION FUNCTIONS

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. The instrumentation also isolates the control room ventilation system.

6. This signal also automatically closes the mechanical vacuum pump discharge line isolation valves.

7. Includes a time delay of $3 \leq t \leq 9$ seconds.

*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 3.7-1 (Cont'd)

Isolation Group	Valve Identification	Valve Number for Units 1 and 2	Number of Power-Operated Valves Inboard	Outboard	Maximum Operating Time (sec)	Normal Operating Position	Action on Initiating Signal
	Radwaste						
2	Drywell floor drain discharge	AO-2001-3	1		≤20	0	GC
2	Drywell floor drain discharge	AO-2001-4		1	≤20	0	GC
2	Drywell equipment drain discharge	AO-2001-15	1		≤20	0	GC
2	Drywell equipment drain discharge	AO-2001-16		1	≤20	0	GC
	Note: Valve can be reopened after isolation for sampling						
	Oxygen Analyzer						
2	Oxygen analyzer valve	AO-8801-A, B,C,D	4		≤10	0	GC
2	Oxygen analyzer valve	AO-8802-A, B,C,D		4	≤10	0	GC
2	Oxygen analyzer valve	AO-8803A	1		≤10	0	GC
2	Oxygen analyzer valve	AO-8803B		1	≤10	0	GC

TABLE 3.7-1 (Cont'd)

Isolation Group	Valve Identification	Valve Number for Units 1 and 2	Number of Power-Operated Valves Inboard	Outboard	Maximum Operating Time (sec)	Normal Operating Position	Action on Initiating Signal
	Traversing Incore Probe						
2	On isolation signal, the TIP detector is withdrawn if in use: five ball valves and one nitrogen purge are closed.	Tip Ball Valve 700-733					
		Tip Purge Valve Assembly 700-743					
	Reactor Water Cleanup						
3	Pump suction isolation valve	MO-1201-2	1		≤30	0	GC
3	Pump suction isolation valve	MO-1201-5		1	≤30	0	GC
	HPCI						
4	Steam isolation valve	MO-2301-4	1		≤50	0	GC
4	Steam isolation valve	MO-2301-5		1	≤50	0	GC
4	Vacuum breaker isolation	MO-2399-40	1		≤50	0	GC
4	Vacuum breaker isolation	MO-2399-41		1	≤50	0	GC
	RCIC						
5	Turbine steam supply	MO-1301-16	1		≤25	0	GC
5	Turbine steam supply	MO-1301-17		1	≤25	0	GC

TABLE 3.7-1 (Cont'd)

Key: O: open
C: closed
SC: stays closed
GC: goes closed

Note: Isolation groupings are as follows:

Group 1: The valves in Group 1 are closed upon any one of the following conditions:

1. Reactor low-low water level
2. Main steamline high radiation
3. Main steamline high flow
4. Main steamline tunnel high temperature
5. Main steamline low pressure

Group 2: The actions in Group 2 are initiated by any one of the following conditions:

1. Reactor low water level
2. High drywell pressure

Group 3: Reactor low water level alone initiates the following:

1. Cleanup demineralizer system isolation

Group 4: The steam supply isolation valves in the high pressure coolant injection system (HPCI) are closed upon any one of the following signals:

1. HPCI steamline high flow
2. High temperature in the vicinity of the HPCI steamline
3. Low reactor pressure

The turbine exhaust vacuum breaker isolation valves close when both of the following signals are present (simultaneously):

1. High drywell pressure
2. Low reactor pressure

Group 5: Isolation valves in the reactor core isolation cooling system (RCIC) are closed upon any one of the following signals:

1. RCIC steamline high flow
2. High temperature in the vicinity of the RCIC steamline
3. Low reactor pressure