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Docket Nos. 50-254
and 50-265

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April 16, 1981

Mr. J. S. Abel
Director of Nuclear Licensing
Commonwealth Edison Company
P. O. Box 767
Chicago, Illinois 60690

Dear Mr. Abel:

In response to your requests of November 7, 1976 and February 21, 1978, with supplements of May 31, 1978, April 25, 1979 and February 14, 1979, the Commission has issued the enclosed amendment Nos. 66 and 60 to Facility Operating License Nos. DPR-29 and DPR-30 for Quad Cities Station Units 1 and 2.

These amendments consist of changes in the Technical Specifications for each of the two units which change setpoints for certain system settings. These changed setpoints are for (1) turbine condenser low vacuum scram, (2) main steamline low pressure isolation, (3) main steamline high flow isolation, (4) ECCS-ADS interlock and (5) ECCS fill system high pressure alarm. These changes in instrument and system setpoints have been made to reduce the number of nuisance alarms and spurious trips caused by setpoint drift. The changes will not adversely affect safety margins defined in the Technical Specifications for each unit.

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

Sincerely,

ORIGINAL SIGNED BY
Vernon L. Rooney
for
Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing

Enclosures:

1. Amendment No. 66 to DPR-29
2. Amendment No. 60 to DPR-30
3. Safety Evaluation
4. Notice

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cc w/encl:
See next page

no legal objection to form of notice & amend

OFFICE	DL:ORB#2	DL:ORB#2	DL:ORB#2	DL:OR	OELD		
SURNAME	SNorris	RBevan	TAIppolito	TMNovak	Olmstead		
DATE	4/10/81	4/10/81	4/11/81	4/13/81	4/14/81		

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

April 16, 1981

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and 50-265

Mr. J. S. Abel
Director of Nuclear Licensing
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cc w/encl:
See next page

Mr. J. S. Abel
Commonwealth Edison Company

cc:

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

Resident Inspector
U. S. Nuclear Regulatory Commission
22712 206th Avenue N.
Cordova, Illinois 61242

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

Illinois Department of Nuclear Safety
1035 Outer Park Drive
5th Floor
Springfield, Illinois 62704

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

Director, Criteria and Standards
Division
Office of Radiation Programs (ANR-460)
U. S. Environmental Protection Agency
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

Susan N. Sekuler
Assistant Attorney General
Environmental Control Division
188 W. Randolph Street
Suite 2315
Chicago, Illinois 60601

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

COMMONWEALTH EDISON COMPANY

AND

IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

DOCKET NO. 50-254

QUAD CITIES STATION UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 66
License No. DPR-29

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

   A. The applications for amendment by the Commonwealth Edison Company (the licensee) dated November 7, 1976, February 21, 1978, as supplemented May 31, 1978, April 25, 1979 and February 14, 1979, 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 (1) 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-29 is hereby amended to read as follows:

8105010517

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 66 , 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

TKooney for

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 16, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 66

FACILITY OPERATING LICENSE NO. DPR-29

DOCKET NO. 50-254

Revise the Appendix "A" Technical Specifications as follows:

Remove	Add
1.0-2	1.0-2
1.1/2.1-2a	1.1/2.1-2a
1.1/2.1-3	1.1/2.1-3
1.1/2.1-8	1.1/2.1-8
1.1/2.1-10	1.1/2.1-10
3.1/4.1-3	3.1/4.1-3
3.1/4.1-8	3.1/4.1-8
3.1/4.1-9	3.1/4.1-9
3.1/4.1-10	3.1/4.1-10
3.2/4/2-6	3.2/4.2-6
3.2/4.2-11	3.2/4.2-11
3.2/4.2-12	3.2/4.2-12
3.5/4.5-8	3.5/4.5-8
3.5/4.5-16	3.5/4.5-16

H. Limiting Conditions for Operation (LCO) - The limiting conditions for operation specify the minimum acceptable levels of system performance necessary to assure safe startup and operation of the facility. When these conditions are met, the plant can be operated safely and abnormal situations can be safely controlled.

I. Limiting Safety System Setting (LSSS) - The limiting safety system settings are settings on instrumentation which initiate the automatic protective action at a level such that the safety limits will not be exceeded. The region between the safety limit and these settings represents margin, with normal operation lying below these settings. The margin has been established so that with proper operation of the instrumentation, the safety limits will never be exceeded.

K. Logic System Functional Test - A logic system functional test means a test of all relays and contacts of a logic circuit from sensor to activated device to ensure all components are operable per design intent. Where possible, action will go to completion; i.e., pumps will be started and valves opened.

L. Modes of Operation - A reactor mode switch selects the proper interlocking for the operating or shutdown condition of the plant. Following are the modes and interlocks provided:

1. Shutdown - In this position, a reactor scram is initiated, power to the control rod drives is removed, and the reactor protection trip systems have been deenergized for 10 seconds prior to permissive for manual reset.

2. Refuel - In this position, interlocks are established so that one control rod only may be withdrawn when flux amplifiers are set at the proper sensitivity level and the refueling crane is not over the reactor. Also, the trips from the turbine control valves, turbine stop valves, main steam isolation valves, and condenser vacuum are bypassed. If the refueling crane is over the reactor, all rods must be fully inserted and none can be withdrawn.

3. Startup/Hot Standby - In this position, the reactor protection scram trips, initiated by condenser low vacuum and main steamline isolation valve closure, are bypassed, the low pressure main steamline isolation valve closure trip is bypassed, and the reactor protection system is energized, with IRM and APRM neutron monitoring system trips and control rod withdrawal interlocks in service.

4. Run - In this position the reactor system pressure is at or above 825 psig, and the reactor protection system is energized, with APRM protection and RMB interlocks in service (excluding the 15% high flux scram).

M. Operable - A system or component shall be considered operable when it is capable of performing its intended function in its required manner.

N. Operating - Operating means that a system or component is performing its intended functions in its required manner.

O. Operating Cycle - Interval between the end of one refueling outage for a particular unit and the end of the next subsequent refueling outage for the same unit.

P. Primary Containment Integrity - Primary containment integrity means that the drywell and pressure suppression chamber are intact and all of the following conditions are satisfied:

1. All manual containment isolation valves on lines connecting to the reactor coolant system or containment which are not required to be open during accident conditions are closed.

The definitions used above for the APRM scram trip apply. In the event of operation with a maximum fraction limiting power density (MFLPD) greater than the fraction of rated power (FRP), the setting shall be modified as follows:

$$S \leq (.65W_D + 43) \cdot \frac{FRP}{MFLPD}$$

The definitions used above for the APRM scram trip apply.

The ratio of FRP to MFLPD shall be set equal to 1.0 unless the actual operating value is less than 1.0, in which case the actual operating value will be used.

This may also be performed by increasing the APRM gain by the inverse ratio, MFLPD/FRP, which accomplishes the same degree of protection as reducing the trip setting by FRP/MFLPD.

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 ≥ 825 psig.

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

I. Turbine EHC control fluid low-pressure scram on loss of control oil pressure shall be set at greater than or equal to 900 psig.

J. Condenser low vacuum scram shall be set at $\geq$ 21 inches Hg vacuum.

An increase in the APRM scram trip setting would decrease the margin present before the fuel cladding integrity safety limit is reached. The APRM scram trip setting was determined by an analysis of margins required to provide a reasonable range for maneuvering during operation. Reducing this operating margin would increase the frequency of spurious scrams, which have an adverse effect on reactor safety because of the resulting thermal stresses. Thus, the APRM scram trip setting was selected because it provides adequate margin for the fuel cladding integrity safety limit yet allows operating margin that reduces the possibility of unnecessary scrams.

The scram trip setting must be adjusted to ensure that the LHGR transient peak is not increased for any combination of maximum fraction of limiting power density (MFLPD) and reactor core thermal power. The scram setting is adjusted in accordance with the formula in Specification 2.1.A.1. when the MFLPD is greater than the fraction of rated power (FRP). The adjustment may be accomplished by increasing the APRM gain by the reciprocal of FRP/MFLPD. This provides the same degree of protection as reducing the trip setting by FRP/MFLPD by raising the initial APRM readings closer to the trip settings such that a scram would be received at the same point in a transient as if the trip settings had been reduced by $\frac{FRP}{MFLPD}$.

2. APRM Flux Scram Trip Setting (Refuel or Startup/Hot Standby Mode)

For operation in the Startup mode while the reactor is at low pressure, the APRM scram setting of 15% of rated power provides adequate thermal margin between the setpoint and the safety limit, 25% of rated. The margin is adequate to accommodate anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures backed up by the rod worth minimizer. Of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated power, the rate of power rise is very slow. Generally, the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than 5% of rated power per minute, and the APRM system would be more than adequate to assure a scram before the power could exceed the safety limit. The 15% APRM scram remains active until the mode switch is placed in the Run position. This switch occurs when reactor pressure is greater than 825 psig.

3. IRM Flux Scram Trip Setting

The IRM system consists of eight chambers, four in each of the reactor protection system logic channels. The IRM is a 5-decade instrument which covers the range of power level between that covered by the SRM and the APRM. The 5 decades are broken down into 10 ranges, each being one-half a decade in size.

The IRM scram trip setting of 120 divisions is active in each range of the IRM. For example, if the instrument were on Range 1, the scram setting would be 120 divisions for that range; likewise, if the instrument were on Range 5, the scram would be 120 divisions on that range. Thus, as the IRM is ranged up to accommodate the increase in power level, the scram trip setting is also ranged up.

The most significant sources of reactivity change during the power increase are due to control rod withdrawal. In order to ensure that the IRM provides adequate protection against the single rod withdrawal error, a range of rod withdrawal accidents was analyzed. This analysis included starting the accident at various power levels. The most severe case involves an initial condition in which the reactor is just subcritical and the IRM system is not yet on scale.

Additional conservatism was taken in this analysis by assuming that the IRM channel closest to the withdrawn rod is bypassed. The results of this analysis show that the reactor is scrammed and peak power limited to 1% of rated power, thus maintaining MCPR above the fuel cladding integrity safety limit. Based on the above analysis, the IRM provides protection against local control rod withdrawal errors and continuous withdrawal of control rods in sequence provides backup protection for the APRM.

G. Reactor Coolant Low Pressure Initiates Main Steam Isolation Valve Closure

The low-pressure isolation at 825 psig was provided to give protection against fast reactor depressurization and the resulting rapid cooldown of the vessel. Advantage was taken of the scram feature which occurs in the Run mode when the main steamline isolation valves are closed to provide for reactor shutdown so that operation at pressures lower than those specified in the thermal hydraulic safety limit does not occur, although operation at a pressure lower than 825 psig would not necessarily constitue an unsafe condition.

H. Main Steamline Isolation to Valve Closure Scram

The low-pressure isolation of the main steamlines at 825 psig was provided to give protection against rapid reactor depressurization and the resulting rapid cooldown of the vessel. Advantage was taken of the scram feature in the Run mode which occurs when the main steamline isolation valves are closed to provide for reactor shutdown so that high power operation at low reactor pressures does not occur, thus providing protection for the fuel cladding integrity safety limit. Operation of the reactor at pressures lower than 825 psig requires that the reactor mode switch be in the Startup position, where protection of the fuel cladding integrity safety limit is provided by the IRM and APRM high neutron flux scrams. Thus, the combination of main steamline low-pressure isolation and isolation valve closure scram in the Run mode assures the availability of neutron flux scram protection over the entire range of applicability of the fuel cladding integrity safety limit. In addition, the isolation valve closure scram in the Run mode anticipates the pressure and flux transients which occur during normal or inadvertent isolation valve closure. With the scrams set at 10% valve closure in the Run mode, there is no increase in neutron flux.

I. Turbine EHC Control Fluid Low-Pressure Scram

The turbine EHC control system operates using high-pressure oil. There are several points in this oil system where a lost of oil pressure could result in a fast closure of the turbine control valves. This fast closure of the turbine control valves is not protected by the turbine control valve fast closure scram, since failure of the oil system would not result in the fast closure solenoid valves being actuated. For a turbine control valve fast closure, the core would be protected by the APRM and high-reactor pressure scrams. However, to provide the same margins as provided for the generator load rejection on fast closure of the turbine control valves, a scram has been added to the reactor protection system which senses failure of control oil pressure to the turbine control system. This is an anticipatory scram and results in reactor shutdown before any significant increase in neutron flux occurs. The transient response is very similar to that resulting from the turbine control valve fast closure scram. The scram setpoint of 900 psig is set high enough to provide the necessary anticipatory function and low enough to minimize the number of spurious scrams. Normal operating pressure for this system is 1250 psig. Finally, the control valves will not start until the fluid pressure is 600 psig. Therefore, the scram occurs well before valve closure begins.

J. Condenser Low Vacuum Scram

Loss of condenser vacuum occurs when the condenser can no longer handle the heat input. Loss of condenser vacuum initiates a closure of the turbine stop valves and turbine bypass valves which eliminates the heat input to the condenser. Closure of the turbine stop and bypass valves causes a pressure transient, neutron flux rise, and an increase in surface heat flux. To prevent the cladding safety limit from being exceeded if this occurs, a reactor scram occurs on turbine stop valve closure in the Run mode. The turbine stop valve closure scram function alone is adequate to prevent the cladding safety limit from being exceeded in the event of a turbine trip transient with bypass closure.

The condenser low vacuum scram is anticipatory to the stop valve closure scram and causes a scram before the stop valves are closed and thus the resulting transient is less severe. Scram occurs in the Run mode at 21-inch Hg vacuum stop valve closure occurs at 20-inch Hg vacuum, and bypass closure at 7-inch Hg vacuum.

gallons. As indicated above, there is sufficient volume in the piping to accommodate the scram without impairment of the scram times or amount of insertion of the control rods. This function shuts the reactor down while sufficient volume remains to accommodate the discharged water and precludes the situation in which a scram would be required but not be able to perform its function adequately.

Loss of condenser vacuum occurs when the condenser can no longer handle heat input. Loss of condenser vacuum initiates a closure of the turbine stop valves and turbine bypass valves, which eliminates the heat input to the condenser. Closure of the turbine stop and bypass valves causes a pressure transient, neutron flux rise, and an increase in surface heat flux. To prevent the cladding safety limit from being exceeded if this occurs, a reactor scram occurs on turbine stop valve closure. The turbine stop valve closure scram function alone is adequate to prevent the cladding safety limit from being exceeded in the event of a turbine trip transient with bypass closure. The condenser low-vacuum scram is a backup to the stop valve closure scram and causes a scram before the stop valves are closed, thus the resulting transient is less severe. Scram occurs at 21 inches Hg vacuum, stop valve closure occurs at 20 inches Hg vacuum, and bypass closure at 7 inches Hg vacuum.

High radiation levels in the main steamline tunnel above that due to the normal nitrogen and oxygen radioactivity are an indication of leaking fuel. A scram is initiated whenever such radiation level exceeds seven times normal background. The purpose of this scram is to reduce the source of such radiation to the extent necessary to prevent excessive turbine contamination. Discharge of excessive amounts of radioactivity to the site environs is prevented by the air ejector off-gas monitors, which cause an isolation of the main condenser off-gas line provided the limit specified in Specification 3.8 is exceeded.

The main steamline isolation valve closure scram is set to scram when the isolation valves are 10% closed from full open. This scram anticipates the pressure and flux transient which would occur when the valves close. By scramming at this setting, the resultant transient is insignificant.

A reactor mode switch is provided which actuates or bypasses the various scram functions appropriate to the particular plant operating status (reference SAR Section 7.7.1.2). Whenever the reactor mode switch is in the Refuel or Startup/Hot Standby position, the turbine condenser low-vacuum scram and main steamline isolation valve closure scram are bypassed. This bypass has been provided for flexibility during startup and to allow repairs to be made to the turbine condenser. While this bypass is in effect, protection is provided against pressure or flux increases by the high-pressure scram and APRM 15% scram, respectively, which are effective in this mode.

If the reactor were brought to a hot standby condition for repairs to the turbine condenser, the main steamline isolation valves would be closed. No hypothesized single failure or single operator action in this mode of operation can result in an unreviewed radiological release.

The manual scram function is active in all modes, thus providing for a manual means of rapidly inserting control rods during all modes of reactor operation.

The IRM system provides protection against excessive power levels and short reactor periods in the startup and intermediate power ranges (reference SAR Sections 7.4.4.2 and 7.4.4.3). A source range monitor (SRM) system is also provided to supply additional neutron level information during startup but has no scram functions (reference SAR Section 7.4.3.2). Thus the IRM is required in the Refuel and Startup/Hot Standby modes. In addition, protection is provided in this range by the APRM 15% scram as discussed in the bases for Specification 2.1. In the power range, the APRM system provides required protection (reference SAR Section 7.4.5.2). Thus, the IRM system is not required in the Run mode, the APRM's cover only the intermediate and power range; the IRM's provide adequate coverage in the startup and intermediate range.

The high-reactor pressure, high-drywell pressure, reactor low water level, and scram discharge volume high level scrams are required for the Startup/Hot Standby and Run modes of plant operation. They are therefore required to be operational for these modes of reactor operation.

The turbine condenser low-vacuum scram is required only during power operation and must be bypassed to start up the unit.

TABLE 3.1-1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENTS REFUEL MODE

Minimum Number of Operable or Tripped Instrument Channels per Trip System[1]	Trip Function	Trip Level Setting	Action[2]
1	Mode switch in shutdown		A
1	Manual scram		A
	IRM		
3	High flux	≤120/125 of full scale	A
3	Inoperative		
	APRM[3]		
2	High flux (15% scram)	Specification 2.1.A.2	A
2	Inoperative		A
2	High water level in scram discharge volume[4]	≤50 gallons	A
2	High reactor pressure	≤1060 psig	A
2	High drywell pressure[5]	≤2 psig	A
2	Reactor low water level	≥8 inches[8]	A
2	Turbine condenser low vacuum [7]	≥21 inches Hg vacuum	A
2	Main steamline high radiation[12]	≤7 X normal full power background	A
4	Main steamline isolation valve closure[7]	≤10% valve closure	A

TABLE 3.1-2

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENTS STARTUP/HOT STANDBY MODE

Minimum Number of Operable or Tripped Instrument Channels per Trip System[1]	Trip Function	Trip Level Setting	Action[2]
1	Mode switch in shutdown		A
1	Manual scram		A
	IRM		
3	High flux	≤120/125 of full scale	A
3	Inoperative		A
	APRM[3]		
2	High flux (15% scram)	Specification 2.1.A.2	A
2	Inoperative		A
2	High-reactor pressure	≤1060 psig	A
2	High-drywell pressure[5]	≤2 psig	A
2	Reactor low water level	≥8 inches[8]	A
2	High water level in scram discharge volume[4]	≤50 gallons	A
2	Turbine condenser low vacuum[7]	≥ 21 inches Hg vacuum	A
2	Main steamline high radiation[12]	≤7 X normal full power background	A
4	Main steamline isolation valve closure[7]	≤10% valve closure	A

TABLE 3.1-3

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENTS RUN MODE

Minimum Number of Operable or Tripped Instrument Channels per Trip System[1]	Trip Function	Trip Level Setting	Action[2]
1	Mode switch in shutdown		A
1	Manual scram		A
	APRM[3]		
2	High flux (flow biased)	Specification 2.1.A.1	A or B
2	Inoperative		A or B
2	Downscale[11]	≥3/125 of full scale	A or B
2	High-reactor pressure	<1060 psig	A
2	High-drywell pressure	≤2 psig	A
2	Reactor low water level	≥8 inches [8]	A
2	High-water level in scram discharge volume	≤50 gallons	A
2	Turbine condenser low vacuum	≥21 inches Hg vacuum	A or C
2	Main steamline high radiation[12]	≤7 X normal full power background	A or C
4	Main steamline isolation valve closure[6]	≤10% valve closure	A or C
2	Turbine control valve fast closure[9]	≥40% turbine/generator load mismatch[10]	A or C
2	Turbine stop valve closure[9]	≤10% valve closure	A or C
2	Turbine EHC control fluid low pressure[9]	≥900 psig	A or C

Venturi tubes are provided in the main steamlines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steamline break accident. In addition to monitoring steam flow, instrumentation is provided which causes a trip of Group I isolation valves. The primary function of the instrumentation is to detect a break in the main steamline, thus only Group I valves are closed. For the worst-case accident, main steamline break outside the drywell, this trip setting of 140% of rated steam flow, in conjunction with the flow limiters and main steamline valve closure, limits the mass inventory loss such that fuel is not uncovered, fuel temperatures remain less than 1500° F, and release of radioactivity to the environs is well below 10 CFR 100 guidelines (reference SAR Sections 14.2.3.9 and 14.2.3.10).

Temperature-monitoring instrumentation is provided in the main steamline tunnel to detect leaks in this area. Trips are provided on this instrumentation and when exceeded cause closure of Group I isolation valves. Its setting of 200° F is low enough to detect leaks of the order of 5 to 10 gpm; thus it is capable of covering the entire spectrum of breaks. For large breaks, it is a backup to high-steam flow instrumentation discussed above, and for small breaks with the resulting small release of radioactivity, gives isolation before the guidelines of 10 CFR 100 are exceeded.

High-radiation monitors in the main steamline tunnel have been provided to detect gross fuel failure. This instrumentation causes closure of Group I valves, the only valves required to close for this accident. With the established setting of 7 times normal background and main steamline isolation valve closure, fission product release is limited so that 10 CFR 100 guidelines are not exceeded for this accident (reference SAR Section 12.2.1.7).

Pressure instrumentation is provided which trips when main steamline pressure drops below 825 psig. A trip of this instrumentation results in closure of Group I isolation valves. In the Refuel and Startup/Hot Standby modes this trip function is bypassed. This function is provided primarily to provide protection against a pressure regulator malfunction which would cause the control and/or bypass valve to open. With the trip set at 825 psig, inventory loss is limited so that fuel is not uncovered and peak cladding temperatures are much less than 1500° F; thus, there are no fission products available for release other than those in the reactor water (reference SAR Section 11.2.3).

The RCIC and the HPCI high flow and temperature instrumentation are provided to detect a break in their respective piping. Tripping of this instrumentation results in actuation of the RCIC or of HPCI isolation valves. Tripping logic for this function is the same as that for the main steamline isolation valves, thus all sensors are required to be operable or in a tripped condition to meet the single-failure criteria. The trip settings of 200° F and 300% of design flow and valve closure time are such that core uncovery is prevented and fission product release is within limits.

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.

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]	≤140% 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]	≥825 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 blowdown.
4	Low-pressure core cooling pump discharge pressure	100 psig≤p≤150 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.

2. The discharge pipe pressure for the systems in Specification 3.5.G.1 shall be maintained at greater than 40 psig and less than 74 psig. If pressure in any of these systems is less than 40 psig or greater than 74 psig, this condition shall be alarmed in the control room and immediate corrective action taken. If the discharge pipe pressure is not within these limits in 12 hours after the occurrence, an orderly shutdown shall be initiated, and the reactor shall be in a cold shutdown condition within 24 hours after initiation.

2. Following any period where the LPCI mode of the RHR or core spray ECCS have been out of service and drained for maintenance, the discharge piping of the inoperable system shall be vented from the high point prior to the return of the system to service.

3. Whenever the HPCI or RCIC system is lined up to take suction from the torus, the discharge piping of the HPCI and RCIC shall be vented from the high point of the system and water flow observed on a monthly basis.

4. The pressure switches which monitor the discharge lines and the discharge of the fill system pump to ensure that they are full shall be functionally tested every month and calibrated every 3 months. The pressure switches shall be set to alarm at a decreasing pressure of ≥ 40 psig and an increasing pressure of ≤ 90 psig.

H. Condensate Pump Room Flood Protection

1. The systems installed to prevent or mitigate the consequences of flooding of the condensate pump room shall be operable prior to startup of the reactor.

2. The condenser pit water level switches shall trip the condenser circulating water pumps and alarm in the control room if water level in the condenser pit exceeds a level of 5 feet above the pit floor. If a failure occurs in one of these trip and alarm circuits, the failed circuit shall be immediately placed in a trip condition and reactor operation shall be permissible for the following 7 days unless the circuit is sooner made operable.

3. If Specification 3.5.H.1 and 2 cannot be met, reactor startup shall not commence or if operating, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

H. Condensate Pump Room Flood Protection

1. The following surveillance requirements shall be observed to assure that the condensate pump room flood protection is operable.

   a. The piping and electrical penetrations and bulkhead doors for the vaults containing the RHR service water pumps and diesel-generator cooling pumps shall be checked during each operating cycle by pressurizing to 15 ± 2 psig and checking for leaks using a soap bubble solution. The criteria for acceptance shall be no visible leakage through the soap bubble solution.

   b. The floor drains from the vaults shall be checked during each operating cycle by removing the end cap and assuring that water can be run through the drain lines.

   c. The RHR service water pump and diesel generator cooling water pump bed plate drains shall be checked during each operating

4.5 SURVEILLANCE REQUIREMENTS BASES

The testing interval for the core and containment cooling systems is based on a quantitative reliability analysis, judgment, and practicality. The core cooling systems have not been designed to be fully testable during operation. For example, the core spray final admission valves do not open until reactor pressure has fallen to 350 psig. Thus, during operation, even if high drywell pressure were simulated, the final valves would not open. In the case of the HPCI, automatic initiation during power operation would result in pumping cold water into the reactor vessel which is not desirable.

The systems can be automatically actuated during a refueling outage and this will be done. To increase the availability of the individual components of the core and containment cooling systems, the components which make up the system, i.e., instrumentation, pumps, valve operators, etc., are tested more frequently. The instrumentation is functionally tested each month. Likewise the pumps and motor-operated valves are also tested each month to assure their operability. The combination of a yearly simulated automatic actuation test and monthly tests of the pumps and valve operators is deemed to be adequate testing of these systems.

With components or subsystems out of service, overall core and containment cooling reliability is maintained by demonstrating the operability of the remaining cooling equipment. The degree of operability to be demonstrated depends on the nature of the reason for the out-of-service equipment. For routine out-of-service periods caused by preventative maintenance, etc., the pump and valve operability checks will be performed to demonstrate operability of the remaining components. However, if a failure, design deficiency, etc., causes the out-of-service period, then the demonstration of operability should be thorough enough to assure that a similar problem does not exist on the remaining components. For example, if an out-of-service period caused by failure of a pump to deliver rated capacity due to a design deficiency, the other pumps of this type might be subjected to a flow rate test in addition to the operability checks.

The verification of the main steam relief valve operability during manual actuation surveillance testing must be made independent of temperatures indicated by thermocouples downstream of the relief valves. It has been found that a temperature increase may result with the valve still closed. This is due to steam being vented through the pilot valves during the surveillance test. By first opening a turbine bypass valve, and then observing its closure response during relief valve actuation, positive verification can be made for the relief valve opening and passing steam flow. Closure response of the turbine control valves during relief valve manual actuation would likewise serve as an adequate verification for the relief valve opening. This test method may be performed over a wide range of reactor pressures greater than 150 psig. Valve operation below 150 psig is limited by the spring tension exhibited by the relief valves.

The surveillance requirements to ensure that the discharge piping of the core spray, LPCI mode of the RHR, HPCI, and RCIC systems is filled provides for a visual observation that water flows from a high point vent. This ensures that the line is in a full condition. Instrumentation has been provided to monitor the pressure of water in the discharge piping between the monthly intervals at which the lines are vented and alarm the control room if the pressure is inadequate. This instrumentation will be calibrated on the same frequency as the safety system instrumentation and the alarm system tested monthly. This testing ensures that, during the interval between the monthly venting checks, the status of the discharge piping is monitored on a continuous basis.

An alarm point of 40 psig for the low pressure of the fill system has been chosen because, due to elevations of piping within the plant, 39 psig is required to keep the lines full. The shutoff head of the fill system pumps is less than 90 psig and therefore will not defeat the low-pressure cooling pump discharge press interlock 100 psig as shown in Table 3.2-2. A margin of 10 psig is provided by the high pressure alarm point of 90 psig.

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. 60
License No. DPR-30

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

   A. The applications for amendment by the Commonwealth Edison Company (the licensee) dated November 7, 1976, February 21, 1978, as supplemented May 31, 1978, April 25, 1979 and February 14, 1979, 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 3.B of Facility License No. DPR-30 is hereby amended to read as follows:

3.B Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 60 , 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

T. Ippolito for

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 16, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 60

FACILITY OPERATING LICENSE NO. DPR-30

DOCKET NO. 50-265

Revise the Appendix "A" Technical Specifications as follows:

Remove	Add
1.0-2	1.0-2
	1.1/2.1-2a
1.1/2.1-3	1.1/2.1-3
1.1/2.1-8	1.1/2.1-8
1.1/2.1-10	1.1/2.1-10
3.1/4.1-3	3.1/4.1-3
3.1/4.1-8	3.1/4.1-8
3.1/4.1-9	3.1/4.1-9
3.1/4.1-10	3.1/4.1-10
3.2/4.2-6	3.2/4.2-6
3.2/4.2-11	3.2/4.2-11
3.2/4.2-12	3.2/4.2-12
3.5/4.5-7	3.5/4.5-7
3.5/4.5-8	3.5/4.5-8
3.5/4.5-15	3.5/4.5-15

H. Limiting Conditions for Operation (LCO) - The limiting conditions for operation specify the minimum acceptable levels of system performance necessary to assure safe startup and operation of the facility. When these conditions are met, the plant can be operated safely and abnormal situations can be safely controlled.

I. Limiting Safety System Setting (LSSS) - The limiting safety system settings are settings on instrumentation which initiate the automatic protective action at a level such that the safety limits will not be exceeded. The region between the safety limit and these settings represents margin, with normal operation lying below these settings. The margin has been established so that with proper operation of the instrumentation, the safety limits will never be exceeded.

K. Logic System Functional Test - A logic system functional test means a test of all relays and contacts of a logic circuit from sensor to activated device to ensure all components are operable per design intent. Where possible, action will go to completion; i.e., pumps will be started and valves opened.

L. Modes of Operation - A reactor mode switch selects the proper interlocking for the operating or shutdown condition of the plant. Following are the modes and interlocks provided:

1. Shutdown - In this position, a reactor scram is initiated, power to the control rod drives is removed, and the reactor protection trip systems have been deenergized for 10 seconds prior to permissive for manual reset.

2. Refuel - In this position, interlocks are established so that one control rod only may be withdrawn when flux amplifiers are set at the proper sensitivity level and the refueling crane is not over the reactor. Also, the trips from the turbine control valves, turbine stop valves, main steam isolation valves, and condenser vacuum are bypassed. If the refueling crane is over the reactor, all rods must be fully inserted and none can be withdrawn.

3. Startup/Hot Standby - In this position, the reactor protection scram trips, initiated by condenser low vacuum and main steamline isolation valve closure, are bypassed, the low pressure main steamline isolation valve closure trip is bypassed, and the reactor protection system is energized, with IRM and APRM neutron monitoring system trips and control rod withdrawal interlocks in service.

4. Run - In this position the reactor system pressure is at or above 825 psig, and the reactor protection system is energized, with APRM protection and RMB interlocks in service (excluding the 15% high flux scram).

M. Operable - A system or component shall be considered operable when it is capable of performing its intended function in its required manner.

N. Operating - Operating means that a system or component is performing its intended functions in its required manner.

O. Operating Cycle - Interval between the end of one refueling outage for a particular unit and the end of the next subsequent refueling outage for the same unit.

P. Primary Containment Integrity - Primary containment integrity means that the drywell and pressure suppression chamber are intact and all of the following conditions are satisfied:

1. All manual containment isolation valves on lines connecting to the reactor coolant system or containment which are not required to be open during accident conditions are closed.

The definitions used above for the APRM scram trip apply. In the event of operation with a maximum fraction limiting power density (MFLPD) greater than the fraction of rated power (FRP), the setting shall be modified as follows:

$$S \leq (.65W_D + 43) \frac{FRP}{MFLPD}$$

The definitions used above for the APRM scram trip apply.

The ratio of FRP to MFLPD shall be set equal to 1.0 unless the actual operating value is less than 1.0, in which case the actual operating value will be used.

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 ≥ 825 psig.

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

I. Turbine EHC control fluid low-pressure scram on loss of control oil pressure shall be set at greater than or equal to 900 psig.

J. Condenser low vacuum scram shall be set at $\geq$ 21 inches Hg vacuum.

An increase in the APRM scram trip setting would decrease the margin present before the fuel cladding integrity safety limit is reached. The APRM scram trip setting was determined by an analysis of margins required to provide a reasonable range for maneuvering during operation. Reducing this operating margin would increase the frequency of spurious scrams, which have an adverse effect on reactor safety because of the resulting thermal stresses. Thus, the APRM scram trip setting was selected because it provides adequate margin for the fuel cladding integrity safety limit yet allows operating margin that reduces the possibility of unnecessary scrams.

The scram trip setting must be adjusted to ensure that the LHGR transient peak is not increased for any combination of maximum fraction of limiting power density (MFLPD) and reactor core thermal power. The scram setting is adjusted in accordance with the formula in Specification 2.1.A.1, when the MFLPD is greater than the fraction of rated power (FRP).

2. APRM Flux Scram Trip Setting (Refuel or Startup/Hot Standby Mode)

For operation in the Startup mode while the reactor is at low pressure, the APRM scram setting of 15% of rated power provides adequate thermal margin between the setpoint and the safety limit, 25% of rated. The margin is adequate to accommodate anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures backed up by the rod worth minimizer. Of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated power, the rate of power rise is very slow. Generally, the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than 5% of rated power per minute, and the APRM system would be more than adequate to assure a scram before the power could exceed the safety limit. The 15% APRM scram remains active until the mode switch is placed in the Run position. This switch occurs when reactor pressure is greater than 825 psig.

3. IRM Flux Scram Trip Setting

The IRM system consists of eight chambers, four in each of the reactor protection system logic channels. The IRM is a 5-decade instrument which covers the range of power level between that covered by the SRM and the APRM. The 5 decades are broken down into 10 ranges, each being one-half a decade in size.

The IRM scram trip setting of 120 divisions is active in each range of the IRM. For example, if the instrument were on Range 1, the scram setting would be 120 divisions for that range; likewise, if the instrument were on Range 5, the scram would be 120 divisions on that range. Thus, as the IRM is ranged up to accommodate the increase in power level, the scram trip setting is also ranged up.

The most significant sources of reactivity change during the power increase are due to control rod withdrawl. In order to ensure that the IRM provides adequate protection against the single rod withdrawal error, a range of rod withdrawal accidents was analyzed. This analysis included starting the accident at various power levels. The most severe case involves an initial condition in which the reactor is just subcritical and the IRM system is not yet on scale.

Additional conservatism was taken in this analysis by assuming that the IRM channel closest to the withdrawn rod is bypassed. The results of this analysis show that the reactor is scrammed and peak power limited to 1% of rated power, thus maintaining MCPR above the fuel cladding integrity safety limit. Based on the above analysis, the IRM provides protection against local control rod withdrawal errors and continuous withdrawal of control rods in sequence and provides backup protection for the APRM.

G. Reactor Coolant Low Pressure Initiates Main Steam Isolation Valve Closure

The low-pressure isolation at 825 psig was provided to give protection against fast reactor depressurization and the resulting rapid cooldown of the vessel. Advantage was taken of the scram feature which occurs in the Run mode when the main steamline isolation valves are closed to provide for reactor shutdown so that operation at pressures lower than those specified in the thermal hydraulic safety limit does not occur, although operation at a pressure lower than 825 psig would not necessarily constitue an unsafe condition.

H. Main Steamline Isolation to Valve Closure Scram

The low-pressure isolation of the main steamlines at 825 psig was provided to give protection against rapid reactor depressurization and the resulting rapid cooldown of the vessel. Advantage was taken of the scram feature in the Run mode which occurs when the main steamline isolation valves are closed to provide for reactor shutdown so that high power operation at low reactor pressures does not occur, thus providing protection for the fuel cladding integrity safety limit. Operation of the reactor at pressures lower than 825 psig requires that the reactor mode switch be in the Startup position, where protection of the fuel cladding integrity safety limit is provided by the IRM and APRM high neutron flux scrams. Thus, the combination of main steamline low-pressure isolation and isolation valve closure scram in the Run mode assures the availability of neutron flux scram protection over the entire range of applicability of the fuel cladding integrity safety limit. In addition, the isolation valve closure scram in the Run mode anticipates the pressure and flux transients which occur during normal or inadvertent isolation valve closure. With the scrams set at 10% valve closure in the Run mode, there is no increase in neutron flux.

I. Turbine EHC Control Fluid Low-Pressure Scram

The turbine EHC control system operates using high-pressure oil. There are several points in this oil system where a lost of oil pressure could result in a fast closure of the turbine control valves. This fast closure of the turbine control valves is not protected by the turbine control valve fast closure scram, since failure of the oil system would not result in the fast closure solenoid valves being actuated. For a turbine control valve fast closure, the core would be protected by the APRM and high-reactor pressure scrams. However, to provide the same margins as provided for the generator load rejection on fast closure of the turbine control valves, a scram has been added to the reactor protection system which senses failure of control oil pressure to the turbine control system. This is an anticipatory scram and results in reactor shutdown before any significant increase in neutron flux occurs. The transient response is very similar to that resulting from the turbine control valve fast closure scram. The scram setpoint of 900 psig is set high enough to provide the necessary anticipatory function and low enough to minimize the number of spurious scrams. Normal operating pressure for this system is 1250 psig. Finally, the control valves will not start until the fluid pressure is 600 psig. Therefore, the scram occurs well before valve closure begins.

J. Condenser Low Vacuum Scram

Loss of condenser vacuum occurs when the condenser can no longer handle the heat input. Loss of condenser vacuum initiates a closure of the turbine stop valves and turbine bypass valves which eliminates the heat input to the condenser. Closure of the turbine stop and bypass valves causes a pressure transient, neutron flux rise, and an increase in surface heat flux. To prevent the cladding safety limit from being exceeded if this occurs, a reactor scram occurs on turbine stop valve closure in the Run mode. The turbine stop valve closure scram function alone is adequate to prevent the cladding safety limit from being exceeded in the event of a turbine trip transient with bypass closure.

The condenser low vacuum scram is anticipatory to the stop valve closure scram and causes a scram before the stop valves are closed and thus the resulting transient is less severe. Scram occurs in the Run mode at 21-inch Hg vacuum stop valve closure occurs at 20-inch Hg vacuum, and bypass closure at 7-inch Hg vacuum.

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gallons. As indicated above, there is sufficient volume in the piping to accommodate the scram without impairment of the scram times or amount of insertion of the control rods. This function shuts the reactor down while sufficient volume remains to accommodate the discharged water and precludes the situation in which a scram would be required but not be able to perform its function adequately.

Loss of condenser vacuum occurs when the condenser can no longer handle heat input. Loss of condenser vacuum initiates a closure of the turbine stop valves and turbine bypass valves, which eliminates the heat input to the condenser. Closure of the turbine stop and bypass valves causes a pressure transient, neutron flux rise, and an increase in surface heat flux. To prevent the cladding safety limit from being exceeded if this occurs, a reactor scram occurs on turbine stop valve closure. The turbine stop valve closure scram function alone is adequate to prevent the cladding safety limit from being exceeded in the event of a turbine trip transient with bypass closure. The condenser low-vacuum scram is a backup to the stop valve closure scram and causes a scram before the stop valves are closed, thus the resulting transient is less severe. Scram occurs at 21 inches Hg vacuum, stop valve closure occurs at 20 inches Hg vacuum, and bypass closure at 7 inches Hg vacuum.

High radiation levels in the main steamline tunnel above that due to the normal nitrogen and oxygen radioactivity are an indication of leaking fuel. A scram is initiated whenever such radiation level exceeds seven times normal background. The purpose of this scram is to reduce the source of such radiation to the extent necessary to prevent excessive turbine contamination. Discharge of excessive amounts of radioactivity to the site environs is prevented by the air ejector off-gas monitors, which cause an isolation of the main condenser off-gas line provided the limit specified in Specification 3.8 is exceeded.

The main steamline isolation valve closure scram is set to scram when the isolation valves are 10% closed from full open. This scram anticipates the pressure and flux transient which would occur when the valves close. By scramming at this setting, the resultant transient is insignificant.

A reactor mode switch is provided which actuates or bypasses the various scram functions appropriate to the particular plant operating status (reference SAR Section 7.7.1.2). Whenever the reactor mode switch is in the Refuel or Startup/Hot Standby position, the turbine condenser low-vacuum scram and main steamline isolation valve closure scram are bypassed. This bypass has been provided for flexibility during startup and to allow repairs to be made to the turbine condenser. While this bypass is in effect, protection is provided against pressure or flux increases by the high-pressure scram and APRM 15% scram, respectively, which are effective in this mode.

If the reactor were brought to a hot standby condition for repairs to the turbine condenser, the main steamline isolation valves would be closed. No hypothesized single failure or single operator action in this mode of operation can result in an unreviewed radiological release.

The manual scram function is active in all modes, thus providing for a manual means of rapidly inserting control rods during all modes of reactor operation.

The IRM system provides protection against excessive power levels and short reactor periods in the startup and intermediate power ranges (reference SAR Sections 7.4.4.2 and 7.4.4.3). A source range monitor (SRM) system is also provided to supply additional neutron level information during startup but has no scram functions (reference SAR Section 7.4.3.2). Thus the IRM is required in the Refuel and Startup/Hot Standby modes. In addition, protection is provided in this range by the APRM 15% scram as discussed in the bases for Specification 2.1. In the power range, the APRM system provides required protection (reference SAR Section 7.4.5.2). Thus, the IRM system is not required in the Run mode. the APRM's cover only the intermediate and power range; the IRM's provide adequate coverage in the startup and intermediate range.

The high-reactor pressure, high-drywell pressure, reactor low water level, and scram discharge volume high level scrams are required for the Startup/Hot Standby and Run modes of plant operation. They are therefore required to be operational for these modes of reactor operation.

The turbine condenser low-vacuum scram is required only during power operation and must be bypassed to start up the unit.

TABLE 3.1-1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENTS REFUEL MODE

Minimum Number of Operable or Tripped Instrument Channels per Trip System(1)	Trip Function	Trip Level Setting	Action(2)
1	Mode switch in shutdown		A
1	Manual scram		A
	IRM		
3	High flux	≤120/125 of full scale	A
3	Inoperative		
	APRM(3)		
2	High flux (15% scram)	Specification 2.1.A.2	A
2	Inoperative		A
2	High water level in scram discharge volume(4)	≤50 gallons	A
2	High reactor pressure	≤1060 psig	A
2	High drywell pressure(5)	≤2 psig	A
2	Reactor low water level	≥8 inches(8)	A
2	Turbine condenser low vacuum (7)	≥21 inches Hg vacuum	A
2	Main steamline high radiation(12)	≤7 X normal full power background	A
4	Main steamline isolation valve closure(7)	≤10% valve closure	A

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TABLE 3.1-2

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENTS STARTUP/HOT STANDBY MODE

Minimum Number of Operable or Tripped Instrument Channels per Trip System(1)	Trip Function	Trip Level Setting	Action(2)
1	Mode switch in shutdown		A
1	Manual scram		A
	IRM		
3	High flux	≤120/125 of full scale	A
3	Inoperative		A
	APRM(3)		
2	High flux (15% scram)	Specification 2.1.A.2	A
2	Inoperative		A
2	High-reactor pressure	≤1060 psig	A
2	High-drywell pressure(5)	≤2 psig	A
2	Reactor low water level	≥8 inches(8)	A
2	High water level in scram discharge volume(4)	≤50 gallons	A
2	Turbine condenser low vacuum(7)	≥21 inches Hg vacuum	A
2	Main steamline high radiation(12)	≤7 X normal full power background	A
4	Main steamline isolation valve closure(7)	≤10% valve closure	A

Amendment No. 60

TABLE 3.1-3

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENTS RUN MODE

Minimum Number of Operable or Tripped Instrument Channels per Trip System[1]	Trip Function	Trip Level Setting	Action[2]
1	Mode switch in shutdown		A
1	Manual scram		A
	APRM[3]		
2	High flux (flow biased)	Specification 2.1.A.1	A or B
2	Inoperative		A or B
2	Downscale[11]	≥3/125 of full scale	A or B
2	High-reactor pressure	<1060 psig	A
2	High-drywell pressure	≤2 psig	A
2	Reactor low water level	≥8 inches [8]	A
2	High-water level in scram discharge volume	≤50 gallons	A
2	Turbine condenser low vacuum	≥ 21 inches Hg vacuum	A or C
2	Main steamline high radiation[12]	≤7 X normal full power background	A or C
4	Main steamline isolation valve closure[6]	≤10% valve closure	A or C
2	Turbine control valve fast closure[9]	≥40% turbine/generator load mismatch[10]	A or C
2	Turbine stop valve closure[9]	≤10% valve closure	A or C
2	Turbine EHC control fluid low pressure[9]	≥900 psig	A or C

Amendment No. 60

Venturi tubes are provided in the main steamlines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steamline break accident. In addition to monitoring steam flow, instrumentation is provided which causes a trip of Group I isolation valves. The primary function of the instrumentation is to detect a break in the main steamline, thus only Group I valves are closed. For the worst-case accident, main steamline break outside the drywell, this trip setting of 140% of rated steam flow, in conjunction with the flow limiters and main steamline valve closure, limits the mass inventory loss such that fuel is not uncovered, fuel temperatures remain less than 1500° F, and release of radioactivity to the environs is well below 10 CFR 100 guidelines (reference SAR Sections 14.2.3.9 and 14.2.3.10).

Temperature-monitoring instrumentation is provided in the main steamline tunnel to detect leaks in this area. Trips are provided on this instrumentation and when exceeded cause closure of Group I isolation valves. Its setting of 200° F is low enough to detect leaks of the order of 5 to 10 gpm; thus it is capable of covering the entire spectrum of breaks. For large breaks, it is a backup to high-steam flow instrumentation discussed above, and for small breaks with the resulting small release of radioactivity, gives isolation before the guidelines of 10 CFR 100 are exceeded.

High-radiation monitors in the main steamline tunnel have been provided to detect gross fuel failure. This instrumentation causes closure of Group I valves, the only valves required to close for this accident. With the established setting of 7 times normal background and main steamline isolation valve closure, fission product release is limited so that 10 CFR 100 guidelines are not exceeded for this accident (reference SAR Section 12.2.1.7).

Pressure instrumentation is provided which trips when main steamline pressure drops below 825 psig. A trip of this instrumentation results in closure of Group I isolation valves. In the Refuel and Startup/Hot Standby modes this trip function is bypassed. This function is provided primarily to provide protection against a pressure regulator malfunction which would cause the control and/or bypass valve to open. With the trip set at 825 psig, inventory loss is limited so that fuel is not uncovered and peak cladding temperatures are much less than 1500° F; thus, there are no fission products available for release other than those in the reactor water (reference SAR Section 11.2.3).

The RCIC and the HPCI high flow and temperature instrumentation are provided to detect a break in their respective piping. Tripping of this instrumentation results in actuation of the RCIC or of HPCI isolation valves. Tripping logic for this function is the same as that for the main steamline isolation valves, thus all sensors are required to be operable or in a tripped condition to meet the single-failure criteria. The trip settings of 200° F and 300% of design flow and valve closure time are such that core uncovery is prevented and fission product release is within limits.

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.

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)	≤140% 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)	≥825 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	100 psig≤p≤150 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.

is being done which has the potential for draining the reactor vessel.

3. When irradiated fuel is in the reactor and the vessel head is removed, the suppression chamber may be drained completely and no more than one control rod drive housing opened at any one time provided that the spent fuel pool gate is open and the fuel pool water level is maintained at a level of greater than 33 feet above the bottom of the pool. Additionally, a minimum condensate storage reserve of 230,000 gallons shall be maintained, no work shall be performed in the reactor vessel while a control rod drive housing is blanked following removal of the control rod drive, and a special flange shall be available which can be used to blank an open housing in the event of a leak.

4. When irradiated fuel is in the reactor and the vessel head is removed, work that has the potential for draining the vessel may be carried on with less than 112,200 ft^3 of water in the suppression pool, provided that: (1) the total volume of water in the suppression pool, refueling cavity, and the fuel storage pool above the bottom of the fuel pool gate is greater than 112,200 ft^3; (2) the fuel storage pool gate is removed; (3) the low-pressure core and containment cooling systems are operable; and (4) the automatic mode of the drywell sump pumps is disabled.

G. Maintenance of Filled Discharge Pipe

1. Whenever core spray, LPCI mode of the RHR, HPCI, or RCIC are required to be operable, the discharge piping from the pump discharge of these systems to the last check valves shall be filled.

G. Maintenance of Filled Discharge Pipe

The following surveillance requirements shall be adhered to to assure that the discharge piping of the core spray, LPCI mode of the RHR, HPCI, and RCIC are filled:

1. Every month prior to the testing of the LPCI mode of the RHR and core spray ECCS, the discharge piping of these systems shall be vented from the high point and water flow observed.

2. The discharge pipe pressure for the systems in Specification 3.5.G.1 shall be maintained at greater than 40 psig and less than 74 psig. If pressure in any of these systems is less than 40 psig or greater than 74 psig, this condition shall be alarmed in the control room and immediate corrective action taken. If the discharge pipe pressure is not within these limits in 12 hours after the occurrence, an orderly shutdown shall be initiated, and the reactor shall be in a cold shutdown condition within 24 hours after initiation.

2. Following any period where the LPCI mode of the RHR or core spray ECCS have been out of service and drained for maintenance, the discharge piping of the inoperable system shall be vented from the high point prior to the return of the system to service.

3. Whenever the HPCI or RCIC system is lined up to take suction from the torus, the discharge piping of the HPCI and RCIC shall be vented from the high point of the system and water flow observed on a monthly basis.

4. The pressure switches which monitor the discharge lines and the discharge of the fill system pump to ensure that they are full shall be functionally tested every month and calibrated every 3 months. The pressure switches shall be set to alarm at a decreasing pressure of ≥ 40 psig and an increasing pressure of ≤ 90 psig.

H. Condensate Pump Room Flood Protection

1. The systems installed to prevent or mitigate the consequences of flooding of the condensate pump room shall be operable prior to startup of the reactor.

2. The condenser pit water level switches shall trip the condenser circulating water pumps and alarm in the control room if water level in the condenser pit exceeds a level of 5 feet above the pit floor. If a failure occurs in one of these trip and alarm circuits, the failed circuit shall be immediately placed in a trip condition and reactor operation shall be permissible for the following 7 days unless the circuit is sooner made operable.

3. If Specification 3.5.H.1 and 2 cannot be met, reactor startup shall not commence or if operating, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

H. Condensate Pump Room Flood Protection

1. The following surveillance requirements shall be observed to assure that the condensate pump room flood protection is operable.

   a. The piping and electrical penetrations and bulkhead doors for the vaults containing the RHR service water pumps and diesel-generator cooling pumps shall be checked during each operating cycle by pressurizing to 15 ± 2 psig and checking for leaks using a soap bubble solution. The criteria for acceptance shall be no visible leakage through the soap bubble solution.

   b. The floor drains from the vaults shall be checked during each operating cycle by removing the end cap and assuring that water can be run through the drain lines.

   c. The RHR service water pump and diesel generator cooling water pump bed plate drains shall be checked during each operating

4.5 SURVEILLANCE REQUIREMENTS BASES

The testing interval for the core and containment cooling systems is based on a quantitative reliability analysis, judgment, and practicality. The core cooling systems have not been designed to be fully testable during operation. For example, the core spray final admission valves do not open until reactor pressure has fallen to 350 psig. Thus, during operation, even if high drywell pressure were simulated, the final valves would not open. In the case of the HPCI, automatic initiation during power operation would result in pumping cold water into the reactor vessel which is not desirable.

The systems can be automatically actuated during a refueling outage and this will be done. To increase the availability of the individual components of the core and containment cooling systems, the components which make up the system, i.e., instrumentation, pumps, valve operators, etc., are tested more frequently. The instrumentation is functionally tested each month. Likewise the pumps and motor-operated valves are also tested each month to assure their operability. The combination of a yearly simulated automatic actuation test and monthly tests of the pumps and valve operators is deemed to be adequate testing of these systems.

With components or subsystems out of service, overall core and containment cooling reliability is maintained by demonstrating the operability of the remaining cooling equipment. The degree of operability to be demonstrated depends on the nature of the reason for the out-of-service equipment. For routine out-of-service periods caused by preventative maintenance, etc., the pump and valve operability checks will be performed to demonstrate operability of the remaining components. However, if a failure, design deficiency, etc., causes the out-of-service period, then the demonstration of operability should be thorough enough to assure that a similar problem does not exist on the remaining components. For example, if an out-of-service period caused by failure of a pump to deliver rated capacity due to a design deficiency, the other pumps of this type might be subjected to a flow rate test in addition to the operability checks.

The verification of the main steam relief valve operability during manual actuation surveillance testing must be made independent of temperatures indicated by thermocouples downstream of the relief valves. It has been found that a temperature increase may result with the valve still closed. This is due to steam being vented through the pilot valves during the surveillance test. By first opening a turbine bypass valve, and then observing its closure response during relief valve actuation, positive verification can be made for the relief valve opening and passing steam flow. Closure response of the turbine control valves during relief valve manual actuation would likewise serve as an adequate verification for the relief valve opening. This test method may be performed over a wide range of reactor pressures greater than 150 psig. Valve operation below 150 psig is limited by the spring tension exhibited by the relief valves.

The surveillance requirements to ensure that the discharge piping of the core spray, LPCI mode of the RHR, HPCI, and RCIC systems is filled provides for a visual observation that water flows from a high point vent. This ensures that the line is in a full condition. Instrumentation has been provided to monitor the pressure of water in the discharge piping between the monthly intervals at which the lines are vented and alarm the control room if the pressure is inadequate. This instrumentation will be calibrated on the same frequency as the safety system instrumentation and the alarm system tested monthly. This testing ensures that, during the interval between the monthly venting checks, the status of the discharge piping is monitored on a continuous basis.

An alarm point of ≥ 40 psig for the low pressure of the fill system has been chosen because, due to elevations of piping within the plant, 39 psig is required to keep the lines full. The shutoff head of the fill system pumps is less than 90 psig and therefore will not defeat the low-pressure cooling pump discharge press interlock 100 psig as shown in Table 3.2-2. A margin of 10 psig is provided by the high pressure alarm point of 90 psig.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO.66 TO FACILITY OPERATING LICENSE NO. DPR-29,

AND AMENDMENT NO. 60 TO FACILITY OPERATING LICENSE NO. DPR-30

COMMONWEALTH EDISON COMPANY

AND

IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

QUAD CITIES STATION UNIT NO. 1

QUAD CITIES STATION UNIT NO. 2

DOCKET NOS. 50-254, 50-265

1.0 Introduction

By letters dated November 7, 1976 and Fberuary 21, 1978, Commonwealth Edison Company (the licensee) requested amendments to the Technical Specifications for Quad Cities Units 1 and 2. Additional information was provided by licensees letters dated May 31, 1978, and February 14 and April 25, 1979.

The requested amendments to the Quad Cities Units 1 and 2 Technical Specifications involve five proposed changes to instrument setpoints associated with the main steam, condensate and emergency core cooling systems. The setpoint changes involve the following four trip functions and alarm:

Turbine Condenser-Low Vacuum Scram Setpoint
Main Steamline-Low Pressure Isolation Setpoint
Main Steamline-High Flow Isolation Setpoint
ECCS-ADS Interlock Setpoint
ECCS-High Pressure Alarm Setpoint

The licensee has proposed changes to the above instrument setpoints to reduce the number of nuisance alarms and spurious trips caused by drift of the present instrument settings. This has resulted in reportable occurrences having no safety significance.

2.0 Evaluation

The staff has evaluated the effects of the proposed changes on core and system transient performance and on postulated accident consequences. We also have evaluated the setpoint changes relative to Regulatory Guide 1.105, "Instrument Setpoints." This provides guidelines in the selection of instrument setpoints, considering instrument accuracy, drift, margin, range, adjustment mechanism, and assumptions used in selecting setpoints.

2.1 Turbine Condenser-Low Vacuum Scram Setpoint

The licensee has proposed that the turbine condenser low vacuum scram setpoint be lowered from >23 inches to >21 inches Hg vacuum. Loss of main condenser vacuum results in closure of the turbine stop valves and disables the turbine bypass function. Closure of the stop valves without bypass in turn causes a pressure transient, neutron flux rise and increase in fuel cladding surface heat flux. However, closure of the turbine stop valves results in a reactor scram signal as soon as the valve position reaches 90% open. Thus for a loss of condenser vacuum, the condenser low vacuum scram anticipates the subsequent stop valve closure and thereby results in a less severe reactor pressurization and fuel thermal heat flux rise.

An instantaneous and total loss of vacuum represents the most severe loss of condenser vacuum which can be postulated. Such an event would simultaneously cause fast closure of the turbine stop valves and prevent bypass function while minimizing the beneficial effects of the anticipatory low condenser vacuum scram function. The consequences of this postulated event (instantaneous loss of condenser vacuum), are bounded by the turbine trip without bypass (TT w/o BP) transient, which assumes no credit for the anticipatory (low condenser vacuum) scram. For the TT w/o BP analysis, credit is taken for the direct stop valve closure scram. Thus, the consequences of the loss of condenser vacuum event is not explicitly analyzed since it is bounded by the more limiting TT w/o BP event. Since the more severe licensing basis TT w/o BP event does not take credit for the subject low vacuum scram, a change of the scram setpoint value will not affect the consequences for this limiting event. Thus, although the consequences of the (slower) loss of vacuum could be expected to be somewhat more severe because of a later scram, the safety margins associated with the licensing basis event (i.e., TT w/o BP)will not change.

The proposed vacuum scram setpoint change has also been reviewed for consistency with the quidelines in Regulatory Guide 1.105. The range, accuracy, drift, margin and span of the four turbine condenser low vacuum pressure switches satisfy the recommendations of the applicable guide.

In view of the foregoing considerations, the proposed low vacuum setpoint changes are acceptable.

2.2 Main Steamline-Low Pressure Isolation Setpoint

The licensee has proposed to lower the main steamline low pressure isolation setpoint from >850 psig to >825 psig. The purpose of the low pressure isolation setpoint is to give protection against an excessive reactor depressurization which would result in rapid cooldown of the vessel and to assure that reactor power operation at pressures lower than that specified in the basis for the thermal-hydraulic safety limit does not occur.

The most limiting event, which takes credit for the main steamline low pressure isolation setpoint is the pressure regulator failure transient. For this event the regulator is assumed to fail in the fully open position. Vessel pressure drops rapidly until steamline pressure falls to the low pressure isolation setpoint, which initiates closure of the main steamline isolation valves. The resulting pressurization and power increase transient is quickly terminated when the MSIVs reach 10 percent closed position causing a reactor scram.

Lowering the setpoint from 850 psig to 825 psig will permit a somewhat lower pressure to be attained during the subject transient which will result in increased voiding prior to the effects of repressurization resulting from MSIV closure. The incremental increase on negative reactivity caused by the somewhat larger void content will effectively result in the pressurization phase of the transient initiating at a lower power level. Since sensitivity studies show pressurization transients are milder when initiated at lower power levels the proposed change will effectively lessen the decrease in Critical Power Ratio (CPR) caused by a pressure regulator failure. No other potentially limiting transients or accidents take credit for this safety setting, and the reduced setting provides adequate protection against violation of the lowest pressure specified in the basis for the thermal-hydraulic safety limit.

The proposed main steamline low pressure isolation setpoint changes have been reviewed for consistency with guidance in Regulatory Guide 1.105. The accuracy, drift, margin, range and span of the four main steamline low pressure switches satisfy the recommendations of the applicable guidance.

On the basis of the foregoing considerations regarding the proposed main steamline low pressure isolation setpoint changes, the proposed changes are acceptable.

2.3 Main Steamline-High Flow Isolation Setpoint

The licensee has proposed to increase the main steamline high flow isolation setpoint from $<120\%$ of rated steam flow to $<140\%$ of rated steam flow. The licensee states that the higher value should allow the reactor to operate at full power during MSIV closure testing without initiating a total isolation which would result in a reactor scram. That is, the remaining three steam lines will be able to pass 133% of rated steam flow without initiating a high flow isolation.

The purpose of the main steamline high flow isolation function is to provide protection against pipe breaks in the main steamline outside the drywell. The licensee states that the consequences of a main steamline break, as evaluated in the FSAR, will remain unchanged with the high flow setpoint increased from 120% to 140% of rated steam flow. The basis for this conclusion is that for a complete severence of one main steamline, steam flow almost instantaneously increases to a maximum of 200% of rated steam flow as limited by the flow restrictors. Thus, the present and proposed setpoint would be attained virtually at the same time. The licensee further states that for this reason the FSAR does not explicitely consider a particular high flow isolation setpoint.

We agree with the licensee's conclusion that for a complete severence break the consequences will remain unchanged. However, we requested that the licensee show that the proposed setpoint change would not alter the conclusion that the complete severence break is the worst break within the spectrum of postulated steamline breaks occurring outside of containment.

In response to our request the licensee reports that the plant would have to be operated at 140 percent steam flow conditions for several minutes before the doses associated for the DBA steamline break would be equaled. However, the licensee further states that the plant would operate no longer than 10 seconds in this mode. This conclusion is based on the results of the pressure regulator failure transient analysis which shows that at a steam flow rate of only 115%, vessel pressure would drop by about 100 psi in the first 10 seconds. Thus closure of the MSIV would be rapidly initiated. For a steam flow rate of 140% MSIV closure would initiate even sooner due to the faster depressurization. Thus even though the steam flow associated with a somewhat larger steam line break could avoid being terminated by the raised high flow isolation setpoint, the break would rapidly be isolated by a low reactor pressure condition.

The proposed main steamline high flow isolation setpoint changes have been reviewed also for consistency with guidance in Regulatory Guide 1.105. The sixteen differential pressure flow switches used for the main steamline high flow isolation function are found to satisfy the recommendations of the applicable regulatory guide.

On the basis of the foregoing considerations regarding the proposed main steamline high flow isolation setpoint changes, the proposed changes are acceptable.

2.4 ECCS-ADS Interlock Setpoint and ECCS-High Pressure Alarm

The licensee has proposed to increase the Low Pressure Coolant Injection (LPCI) pump discharge pressure interlock setpoint of the Emergency Core Cooling System - Automatic Depressurization System (ECCS-ADS) from its present range of 75 psig to 100 psig to a new range of 100 psig to 150 psig. Additionally, the licensee has proposed that the high pressure alarm setpoint of the ECCS fill system be increased from < 74 psig to < 90 psig.

The purpose of the ECCS-ADS interlock is to ensure that, prior to ADS initiation, following a small break LOCA in which the high pressure core spray system fails, the LPCI pumps are running with sufficient discharge pressure to adequately reflood the core. The licensee has requested the interlock setpoint pressure range be increased to ensure that the low end of the pressure range cannot be satisfied by the ECCS fill system pump.

The shutoff head of the low pressure coolant injection pumps is approximately 350 psi. Thus the increased ECCS-ADS interlock pressure can easily be achieved and exceeded by any normally running LPCI pump. In response to our request, the licensee provided an acceptable basis for their position that the LPCI pumps could reach the higher interlock pressure within the time period assumed in the most recent LOCA-ECCS analysis involving ADS initiation. The maximum allowable time for the LPCI pumps to achieve full speed (discharge pressure) after receiving the ECCS actuation signal is 43 seconds. In addition, ADS blowdown initiates after a 120 second time period has elapsed from a coincident high drywell pressure and low water level signals with LPCI discharge side pressure satisfying the ECCS-ADS interlock pressure. Thus at least a 77 second margin is available for ADS operation. Additionally the startup sequence for the LPCI pumps given in Section 6.2.7.3 of the FSAR indicates that all LPCI pumps would be expected to reach full speed (discharge pressure) 20 seconds after high drywell pressure is reached. Thus the staff agrees that there is no additional delay in ADS initiation which would affect peak cladding temperatures for LOCA-ECCS analyses involving ADS initiation.

The proposed ECCS-ADS interlock and ECCS-high pressure alarm setpoint changes have been reviewed also for consistency with Regulatory Guide 1.105. The ECCS-ADS interlock Technical Specification limit and setpoint changes include coordinating and changing eight RHR pressure switches, four core spray switches and one ECCS fill system high pressure alarm switch. The thirteen pressure switches affected by the proposed changes satisfy the recommendations of Regulatory Guide 1.105.

On the basis of the foregoing considerations regarding the proposed ECCS-ADS system interlock setpoint changes, the proposed changes are acceptable.

3.0 Environmental Consideration

We have determined that these amendments do 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 these amendments involve an action which is insignificant from the standpoint of environmental impact, and pursuant to 10 CFR Section 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 these amendments.

4.0 Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendments do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the amendments do 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 these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Dated: April 16, 1981

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-254 AND 50-265

COMMONWEALTH EDISON COMPANY

AND

IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO

OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No.66 to Facility Operating License No. DPR-29, and Amendment No.60 to Facility Operating License No. DPR-30, issued to Commonwealth Edison Company and Iowa-Illinois Gas and Electric Company, which revised the Technical Specifications for operation of the Quad-Cities Nuclear Power Station, Unit Nos. 1 and 2, located in Rock Island County, Illinois. The amendments are effective as of the date of issuance.

The amendments revise the technical specifications to change setpoints for certain system settings which include (1) turbine condenser low vacuum scram, (2) main steamline low pressure isolation, (3) main steamline high flow isolation, (4) ECCS-ADS interlock and (5) ECCS fill system high pressure alarm.

The applications for the amendments comply 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 amendments. Prior

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public notice of these amendments was not required since the amendments do not involve a significant hazards consideration.

The Commission has determined that the issuance of these amendments will not result in any significant environmental impact and that pursuant to 10 CFR Section 51.5(d)(4) an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of these amendments.

For further details with respect to this action, see (1) the application for amendments dated November 7, 1976 and February 21, 1978, with supplements of May 31, 1978, April 25, 1979 and February 14, 1979, (2) Amendment No. 66 to License No. DPR-29, and Amendment No. 60 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, NW., Washington, D. C., and at the Moline Public Library, 504 - 17th Street, Moline, Illinois. 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 Licensing.

Dated at Bethesda, Maryland, this 16th day of April 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

Vernon L. Rooney

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