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

January 18, 1989

Docket Nos.: 50-254
and 50-265

Mr. Henry E. Bliss
Nuclear Licensing Manager
Commonwealth Edison Company
Post Office Box 767
Chicago, Illinois 60690

Dear Mr. Bliss:

SUBJECT: HYDROGEN WATER CHEMISTRY CONTROL SYSTEM AND TECHNICAL SPECIFICAITON AMENDMENT (TAC NOS. 69422 AND 69423)

Re: Quad Cities Nuclear Power Station, Units 1 and 2

The Commission has issued the enclosed Amendment Nos. 112 and 108 to Facility Operating License Nos. DPR-29 and DPR-30 for the Quad Cities Nuclear Power Station (QCNPS), Units 1 and 2. These amendments are in response to Commonwealth Edison Company's (CECo) applications dated September 16 and November 18, 1988. As per CECo's request, the main steam line radiation monitors trip setpoint for the reactor protection system was changed to provide for implementation of hydrogen water chemistry (HWC).

By letter dated September 28, 1988, CECo also submitted additional information describing their plans to initiate a HWC program at QCNPS. Based on this and subsequent information provided in response to our questions, we concluded that the proposed permanent HWC system is in accordance with the BWR Owners Group "Guidelines for Permanent BWR Hydrogen Water Chemistry Installations - 1987 Revision."

It is our intention to visit QCNPS when the HWC system is completely installed. The purpose of this visit is to verify compliance with the BWR Owners Group Guidelines.

We will work with the Nuclear Licensing Administrator for QCNPS and the station management to coordinate an appropriate time.

c/p-1 DF02
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A copy of our related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly Federal Register notices.

Sincerely,

Thierry Ross

Thierry Ross, Project Manager
Project Directorate III-2
Division of Reactor Projects - III,
IV, V and Special Projects

Enclosures:
1. Amendment No. 112 to
   License No. DPR-29
2. Amendment No. 108 to
   License No. DPR-30
3. Safety Evaluation

cc w/enclosures:
See next page

cc:

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

COMMONWEALTH EDISON COMPANY

AND

IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

DOCKET NO. 50-254

QUAD CITIES NUCLEAR POWER STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 112
License No. DPR-29

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

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated September 16 and November 18, 1988, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B. of Facility Operating License No. DPR-29 is hereby amended to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 112 , 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

Byron L Siegel for

Daniel R. Muller, Director
Project Directorate III-2
Division of Reactor Projects - III,
IV, V and Special Projects

Attachment:
Changes to the Technical Specifications

Date of Issuance: January 18, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 112

FACILITY OPERATING LICENSE NO. DPR-29

DOCKET NO. 50-254

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

REMOVE	INSERT
3.1/4.1-3	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

Loss of condensate 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 fifteen times normal background (without hydrogen addition). 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 Section 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 (per bank)	High water level in scram discharge volume(4)	≤ 40 gallons per bank	A
2	High reactor pressure	≤ 1060 psig	A
2	High drywell pressure (5)	≤ 2.5 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)	≤ 15 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)	≤ 2.5 psig	A
2	Reactor low water level	≥ 8 inches(8)	A
2 (per bank)	High water level in scram discharge volume(4)	≤ 40 gallons per bank	A
2	Turbine condenser low vacuum (7)	≥ 21 inches Hg vacuum	A
2	Main steamline high radiation (12)	≤ 15 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	≤ 2.5 psig	A
2	Reactor low water level	≥ 8 inches(8)	A
2 (per bank)	High-water level in scram discharge volume	≤ 40 gallons per bank	A
2	Turbine condenser low vacuum	≥ 21 inches Hg vacuum	A or C
2	Main Steamline high radiation (12)	≤ 15 X normal full power background (without hydrogen addition)	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 1 isolation valves. The primary function of the instrumentation is to detect a break in the main steamline, thus only Group 1 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 1 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 1 valves, the only valves required to close for this accident. With the established setting of 15 times normal background (without hydrogen addition) 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 14.2.1.7).

Pressure instrumentation is provided which trips when main steamline pressure drops below 825 psig. A trip of this instrumentation results in closure of Group 1 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 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]	≤2.5 psig [3]	A
16	High flow main steamline[5]	≤140% of rated steam flow	B
16	High temperature main steamline tunnel	≤200° F	B
4	High radiation main steamline tunnel[6]	≤15 x normal rated power background (without hydrogen addition)	B
4	Low main steam pressure[4]	≥825 psig	B
2	High flow RCIC steamline	≤300 % of rated steam flow(7)	C
16	RCIC turbine area high temperature	≤200° F	C
2	High flow HPCI steamline	≤300% of rated steam flow(7)	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. The instrumentation also isolates the control room ventilation system.

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

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

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

UNITED STATES
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WASHINGTON, D. C. 20555

COMMONWEALTH EDISON COMPANY

AND

IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

DOCKET NO. 50-265

QUAD CITIES NUCLEAR POWER STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 108
License No. DPR-30

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

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated September 16 and November 18, 1988, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

   C. There is reasonable assurance (i) that the activities authorized and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

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

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

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

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 108 , 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

Byron L. Siegel for

Daniel R. Muller, Director
Project Directorate III-2
Division of Reactor Projects - III,
IV, V and Special Projects

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 18, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 108

FACILITY OPERATING LICENSE NO. DPR-30

DOCKET NO. 50-265

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

REMOVE	INSERT
3.1/4.1-3	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

Loss of condensate 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 fifteen times normal background (without hydrogen addition). 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 Section 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 (per bank)	High water level in scram discharge volume(4)	≤ 40 gallons per bank	A
2	High-reactor pressure	≤ 1060 psig	A
2	High-drywell pressure(5)	≤ 2.5 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)	≤ 15 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)	≤ 2.5 psig	A
2	Reactor low water level	≥ 8 inches(8)	A
2 (per bank)	High water level in scram discharge volume(4)	≤ 40 gallons per bank	A
2	Turbine condenser low vacuum(7)	≥ 21 inches Hg vacuum	A
2	Main steamline high radiation(12)	≤15 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	≤ 2.5 psig	A
2	Reactor low water level	≥ 8 inches(8)	A
2 (per bank)	High-water level in scram discharge volume	≤ 40 gallons per bank	A
2	Turbine condenser low vacuum	≥ 21 inches Hg vacuum	A or C
2	Main Steamline high radiation (12)	≤ 15 X normal full power background (without hydrogen addition)	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 1 isolation valves. The primary function of the instrumentation is to detect a break in the main steamline, thus only Group 1 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 1 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 1 valves, the only valves required to close for this accident. With the established setting of 15 times normal background (without hydrogen addition) 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 14.2.1.7).

Pressure instrumentation is provided which trips when main steamline pressure drops below 825 psig. A trip of this instrumentation results in closure of Group 1 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 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]	$\geq$144 inches above top of active fuel*	A
4	Reactor low low water	$\geq$84 inches above top of active fuel*	A
4	High drywell pressure[5]	$\leq$2.5 psig [3]	A
16	High flow main steamline[5]	$\leq$140% of rated steam flow	B
16	High temperature main steamline tunnel	$\leq$200° F	B
4	High radiation main steamline tunnel[6]	$\leq$15 x normal rated power background (without hydrogen addition)	B
4	Low main steam pressure[4]	$\geq$825 psig	B
2	High flow RCIC steamline	$\leq$300 % of rated steam flow(7)	C
16	RCIC turbine area high temperature	$\leq$200° F	C
2	High flow HPCI steamline	$\leq$300% of rated steam flow(7)	D
16	HPCI area high temperature	$\leq$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. The instrumentation also isolates the control room ventilation system.

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

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

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 112 TO FACILITY OPERATING LICENSE NO. DPR-29

AND AMENDMENT NO. 108 TO FACILITY OPERATING LICENSE NO. DPR-30

COMMONWEALTH EDISON COMPANY

AND

IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2

DOCKET NOS. 50-254/265

1.0 INTRODUCTION

By letters dated September 16 and November 18, 1988, the licensee requested to amend Quad Cities Station Units 1 and 2 Operating Licenses DPR-29 and DPR-30 to change the setpoint of the main steam line radiation monitors (MSLRMs), to correct typographical errors and to make changes in the Technical Specifications. The requested change involves increasing the setpoint of MSLRMs from seven times Normal Full Power Background (NFPB) to fifteen times NFPB to allow for implementation of Hydrogen Water Chemistry (HWC) which is expected to mitigate the effects of Intergranular Stress Corrosion Cracking (IGSCC). The MSLRM setpoint change is necessary since the injection of hydrogen into the feedwater lowers the oxidizing potential in the reactor coolant which in turn converts more N-16 to a volatile species and results in an increase in steam line radiation level. As a consequence, the steam activity during hydrogen addition can increase up to a factor of approximately five.

By letter dated September 28, 1988 the licensee provided additional information to support the implementation of HWC. The additional information included:

(1) Report titled, "HWC Installation Report for Amendment to the Facility Operating License" dated May 16, 1988.

(2) Report titled "HWC Installation Compliance with Electric Power Research Institute (ERPI) Guidelines for Permanent BWR Hydrogen Water Chemistry Installations - 1987 Revision."

(3) Draft Copy of Proposed Changes to Updated FSAR as a Result of HWC Addition at Quad Cities Station.

The changes will be included in the June 30, 1989 update to the FSAR.

2.0 EVALUATION

The MSLRMs provide reactor scram and main steam line isolation signals when high-activity levels are detected in the main stream lines. Additionally, these monitors serve to limit radioactivity releases in the event of fuel failures. Technical Specification (TS) changes are needed to accommodate the expected main steam line radiation levels (from increased N-16 activity levels in the steam phase) as a result of hydrogen injection into the reactor coolant system.

The licensee has requested TS changes involving raising the MSLRM set points from the current seven times NFPB to fifteen times NFPB. The licensee proposes a single set point for the MSLRMs which is an exception to the EPRI "Guidelines for Permanent BWR Hydrogen Water Chemistry Installation - 1987 Revision" (hereafter referred to as the Guidelines). The Guidelines recommend a dual MSLRM set point: (1) For reactor power less than 20% of rated, when hydrogen should not be injected, the setpoint is maintained at the current TS factor above NFPB, and (2) For reactor power greater than 20% of rated, the setpoint is readjusted to the same TS factor above NFPB with hydrogen addition.

The only design basis event in which the Quad Cities Station takes credit for the MSLRM is the Control Rod Drop Accident (CRDA). In the event of a CRDA, the MSLRMs detect high radiation levels in the main steam lines and provide signals for reactor scram and Main Steam Line Isolation Valve (MSIV) closure to reduce the release of fission products to the environment. For the proposed MSLRM set point of fifteen times NFPB, the calculated dose rate at the MSLRM is 1.5 R/hr. For a CRDA the dose rate at the MSLRM is 8 R/hr. Since the MSLRM dose rate from the CRDA is over five times the proposed increased MSLRM set point, the high radiation signal caused by the CRDA will still scram the reactor and isolate the MSIVs.

Raising the MSLRM trip set point from the current 0.7 R/hr to 1.5 R/hr will not result in a significant increase in the radiological consequences of a CRDA. The time to reach the proposed MSLRM trip set point following a CRDA will be increased the less than 1/4 second. The Quad Cities TS permits five seconds for MSIV closure. The increase in time-to-closure due to the proposed MSLRM set point is only 5% of the current time-to-closure. Since the calculated dose from the CRDA is only 12 mrem, the minor increase in MSIV isolation will have an insignificant effect on the total activity release and the resulting dose to the general public.

In the event of an incident causing minor fuel damage such that radiation levels will not exceed the proposed MSLRM set point of fifteen times NFPB, the downstream steam jet air ejectors radiation detectors would be alarmed. These detectors have a greater sensitivity than the MSLRMs for noble gases because of the holdup period (delay between MSLRM and steam jet air ejector radiation detectors) which allows for significant decay of N-16 (7.1 second half-life). Since steam jet air ejector radiation detectors are in the Quad Cities Unit 1 and 2 TS, the proposed MSLRM set point change will not result in offsite doses in excess of established release limits.

Therefore, the proposed TS changes are acceptable.

2.2 RADIATION PROTECTION

The staff has reviewed the licensee's submittal regarding the radiological implications due to the increased dose rate associated with increased N-16 activity levels during hydrogen injections into the reactor system. The licensee is committed to designing, installing, operating, and maintaining the HWC System in accordance with Regulatory Guides 8.8 and 8.10 to assure that occupational radiation exposures and doses to the general public will be As Low As Reasonably Achievable (ALARA). A preliminary radiological survey has been completed at the Quad Cities Station to identify areas of the station which may experience increased dose rates due to HWC. When HWC is implemented, the results of the preliminary survey will be confirmed and additional measurements will be made, if required. Based on the preliminary survey and experience from the Dresden Unit 2 (implemented HWC in March 1983), additional shielding appears to be unnecessary. Again, when HWC is implemented, these results will be confirmed and additional shielding will be provided, if required.

Plant procedures will address access control of radiation areas that are affected by HWC. Guidelines will be established for any additional controls needed for area posting and monitoring due to HWC. The existing radiological surveillance program (Section 8.4 of Offsite Dose Calculational Manual) assures compliance with regulatory requirements for offsite doses to the public.

Radiation protection practices implemented for HWC will insure ALARA in accordance with Regulatory Guide 8.8 and is, therefore, acceptable.

2.3 HYDROGEN AND OXYGEN STORAGE FACILITIES

The licensee will utilize an interim storage facility for gaseous hydrogen until a long-term liquid hydrogen storage facility is completed. After the liquid hydrogen facility is installed, the gaseous hydrogen facility will be used as a backup supply. The gaseous hydrogen supply will consist of two tractor trailers each containing a bank of compressed hydrogen gas tubes (total capacity 50,000 - 70,000 scf, each tube capacity 8300 scf, maximum pressure 2400 psig). The pressure control station has two parallel full flow pressure reducing regulators. An excess flow check valve is installed downstream of the interim tube trailer and long-term liquid hydrogen storage facility. An additional excess flow check valve is installed in the hydrogen gas supply line near the west wall of the Unit 1 turbine building. Each excess flow check valve has a stop-flow-setpoint of 200 scfm (plant's hydrogen flow requirements are 140 scfm).

The long-term liquid hydrogen storage facility will consist of a 20,000 gallon tank constructed in accordance with Section VIII, Division 1 of the ASME Code for Unfired Pressure Vessels. The hydrogen storage facility (compressed gas and liquid) is located 1500 feet from the nearest safety-related structure. This distance meets the Guidelines which requires 140 and 962 feet separation distance in the event of an explosion of a gaseous hydrogen storage tube and liquid hydrogen tank respectively.

The hydrogen supply facility provides the gaseous hydrogen requirements for turbine generator cooling/purging as well as HWC for Units 1 and 2.

The liquid oxygen storage tank, with a maximum capacity of 11,000 gallons, is located 1000 feet away from the nearest safety-related air intake which meets the Guidelines.

The hydrogen and oxygen storage facilities meet the Guidelines.

2.3 HYDROGEN AND OXYGEN INJECTION SYSTEM

The hydrogen piping is run underground from the storage facility to the outer wall of the Unit 1 turbine building. The piping is covered with protective coating to protect against corrosion and is electrically grounded. The hydrogen injection lines for each Unit are equipped with check valves and solenoid isolation valves, which are interlocked with the condensate pump. The individual solenoid isolation valves provide hydrogen flow isolation if the associated condensate pump is shut down and for all hydrogen injection trips. The hydrogen is injected into the condensate pump discharge to provide adequate dissolving and mixing and to avoid gas pockets at high points.

The HWC system is tripped by the following signals:

- Reactor scram,
- Low residual off-gas oxygen concentration,
- High hydrogen flow,
- Low hydrogen flow,
- Area hydrogen concentration high,
- Operator manual,
- Hydrogen storage facility trouble, and
- Low reactor power level.

The area hydrogen monitors are in eight locations in the vicinity of hydrogen injection system components that may leak. The sensors feed to a monitor panel which trips the system at 20% of the lower explosive limit.

Oxygen is injected into the off-gas system to insure that all excess hydrogen in the off-gas stream is recombined.

The hydrogen and oxygen injection system meet the Guidelines.

3.0 CONCLUSION

On the basis of the above evaluation, we find that the proposed Technical Specification changes required for implementation of HWC at Quad Cities Station Units 1 and 2 are acceptable. The proposed increased single set point, versus a dual power dependent set point, for the MSLRMs is an exception to the BWR Owners Group "Guidelines for Permanent Hydrogen Water Chemistry Installation - 1987 Revision". This exception is justified on the basis that the CRDA dose rate is already limiting at five times the new set point. Thus, it will not affect the safety of the plant or the general public.

3.0 ENVIRONMENTAL CONSIDERATION

These amendments involve a change to a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that these amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that these amendments involve no significant hazards consideration and there has been no public comment on such finding. Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and 10 CFR 51.22(c)(10). Pursuant to 10 CFR 51.22(b) no environmental impact statement nor environmental assessment need be prepared in connection with the issuance of these amendments.

4.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security nor to the health and safety of the public.

Principal Contributor: Frank Witt

Dated: January 18, 1989

A copy of our related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly Federal Register notices.

Sincerely,

/S/

Thierry Ross, Project Manager
Project Directorate III-2
Division of Reactor Projects - III,
IV, V and Special Projects

Enclosures:
1. Amendment No. 112 to License No. DPR-29
2. Amendment No. 108 to License No. DPR-30
3. Safety Evaluation

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See next page

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