February 20, 1987

Docket Nos. 50-254/265

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

Dear Mr. Farrar:

SUBJECT: CONTAINMENT PRESSURE SETPOINTS AND MSIV SURVEILLANCE (TAC 56507, 56508)

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

The Commission has issued the enclosed Amendment Nos. 101 and 98 to Facility Operating License Nos. DPR-29 and DPR-30 for the Quad Cities Nuclear Power Station, Units 1 and 2, respectively. These amendments consist of changes to the Technical Specifications in response to your letter of November 27, 1984 as supplemented by submittal dated July 22, 1986.

The amendments revise the Technical Specifications to change the containment pressure trip setpoints and the main steam isolation valves surveillance requirements.

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

Sincerely,

Original signed by

Rajender Auluck, Acting Director
BWR Project Directorate #1
Division of BWR Licensing

Enclosures:
1. Amendment No. 101 to License No. DPR-29
2. Amendment No. 98 to License No. DPR-30
3. Safety Evaluation

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cc w/enclosures:
See next page

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Mr. Dennis L. Farrar
Commonwealth Edison Company

Quad Cities Nuclear Power Station
Units 1 and 2

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. 101
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 November 27, 1984 as supplemented July 22, 1986, 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. 101, 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

Rajender Auluck

Rajender Auluck, Acting Director
BWR Project Directorate #1
Division of BWR Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 20, 1987

ATTACHMENT TO LICENSE AMENDMENT NO. 101

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
3.1/4.1-8	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
3.2/4.2-11	3.2/4.2-11
3.2/4.2-12	3.2/4.2-12
3.7/4.7-10	3.7/3.7-10
3.7/4.7-18	3.7/4.7-18
3.7/4.7-19	3.7/4.7-19

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	$\leq$ 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)	$\leq$ 40 gallons per bank	A
2	High reactor pressure	$\leq$ 1060 psig	A
2	High drywell pressure (5)	$\leq$ 2.5 psig	A
2	Reactor low water level	$\geq$ 8 inches(8)	A
2	Turbine condenser low vacuum (7)	$\geq$ 21 inches Hg vacuum	A
2	Main steamline high radiation (12)	$\leq$ 7 X normal full power background	A
4	Main steamline isolation valve closure (7)	$\leq$ 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)	≤ 7 X normal full power background	A
4	Main steamline isolation valve closure (7)	≤ 10% valve closure	A

Amendment No. ~~80~~, ~~90~~, 101

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)	≤ 7 X normal full power 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

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]	≤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(7)	C
16	RCIC turbine area high temperature	≤200° F	C
4	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 ≤ t ≤ 10 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).

 Amendment No. ~~66~~, ~~88~~, 101

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]	≤2.5 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-presure core cooling system operation.
2/BUS [5]	Undervoltage on emergency buses	3045 ± 5% volts	1. Initiates starting of diesel generators. 2. Permissive for starting ECCS pumps. 3. Removes nonessential loads from buses 4. Bypasses degraded voltage timer.

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

reopened.

2) The main steamline isolation valves (one at a time) shall be verified for closure time.

2. In the event any isolation valve specified in Table 3.7-1 becomes inoperable, reactor power operation may continue provided at least one valve in each line having an inoperable valve is in the mode corresponding to the isolated condition.

2. When an isolation valve listed in Table 3.7-1 is inoperable, the position of at least one other valve in each line having an inoperable valve shall be recorded daily.

3. If Specifications 3.7.D.1 and 3.7.D.2 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

4. The temperature of the main steamline air pilot valves shall be less than 170°F except as specified in Specifications 3.7.D.5 and 3.7.D.6 below.

5. From and after the date that the temperature of any main steamline air pilot valve is found to be greater than 170°F reactor operation is permissible only during the succeeding 7 days unless the temperature of such valve is sooner reduced to less than 170°F provided the main steamline isolation valves are operable.

6. If Specification 3.7.D.5 cannot be met, the main steamline isolation valve shall be considered inoperable and action taken in accordance with Specification 3.7.D.2.

will be replaced with filters qualified pursuant to regulatory guide position C.3.d. of Regulatory Guide 1.52 Revision 1 (June 1976). Once per operating cycle demonstration of HEPA filter pressure drop, operability of inlet heaters at rated power, air distribution to each HEPA filter, and automatic initiation of each standby gas treatment system circuit is necessary to assure system performance capability). Note: bases within parentheses will not be applicable until about December 31, 1976, when equipment modifications are completed to allow increased testing.

D. Primary Containment Isolation Valves

Those large pipes comprising a portion of the reactor coolant system, whose failure could result in uncovering the reactor core, are supplied with automatic isolation valves (except those lines needed for emergency core cooling system operation or containment cooling). The closure times specified herein are adequate to prevent loss of more coolant from the circumferential rupture of any of these lines outside the containment than from a steamline rupture. Therefore, this isolation valve closure time is sufficient to prevent uncovering the core.

In order to assure that the doses that may result from a steamline break do no exceed the 10 CFR 100 guidelines, it is necessary that no fuel rod perforation resulting from the accident occur prior to closure of the main steam line isolation valves. Analyses indicate that fuel rod cladding perforations would be avoided for main steam valve closure times, including instrument delay, as long as 10.5 seconds. However, for added margin, the technical specifications require a valve close time of not greater than 5 seconds.

For reactor coolant system temperature less than 212°F, the containment could not become pressurized due to a loss-of-coolant accident. The 212°F limit is based on preventing pressurization of the reactor building and rupture of the blowout panels. These valves are highly reliable, have low service requirement, and most are normally closed. The initiating sensors and associated trip channels are also checked to demonstrate the capability for automatic isolation (reference SAR Section 5.2.2 and Table 5.2.4).

The test interval at once per operating cycle for automatic initiation results in a failure probability of 1.1×10^{-7} that a line will not isolate. More frequent testing for valve operability results in a more reliable system.

The containment is penetrated by a large number of small diameter instrument lines which contact the primary coolant system. A program for periodic testing and examination of the flow check valves in these lines is performed by blowing down the instrument line during a vessel hydro and observing the following conditions, which will verify that the flow check valve is operable:

1. a distinctive 'click' when the poppet valve seats, and
2. an instrumentation high flow that quickly reduces to a slight trickle.

References

1. Quad-Cities Special Report Number 4.
2. R.E. Adams and W. E. Browning Jr., ORNL 3726, 'Iodine Vapor Adsorption Studies for the NS 'Savannah' Project, February 1965.

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

COMMONWEALTH EDISON COMPANY
AND
IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

DOCKET NO. 50-265

QUAD CITIES NUCLEAR POWER STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 98
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 November 27, 1984 as supplemented July 22, 1986, 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. 98 , 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

Rajender Auluck

Rajender Auluck, Acting Director
BWR Project Directorate #1
Division of BWR Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 20, 1987

ATTACHMENT TO LICENSE AMENDMENT NO. 98

FACILITY OPERATING LICENSE NO. DPR-30

DOCKET NO. 50-265

Revise the Appendix A Technical Specifications by removing the page 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
3.1/4.1-8	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
3.2/4.2-11	3.2/4.2-11
3.2/4.2-12	3.2/4.2-12
3.7/4.7-10	3.7/3.7-10
3.7/4.7-18	3.7/4.7-18
3.7/4.7-19	3.7/4.7-19

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	$\leq$ 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)	$\leq$ 40 gallons per bank	A
2	High-reactor pressure	$\leq$ 1060 psig	A
2	High-drywell pressure(5)	$\leq$ 2.5 psig	A
2	Reactor low water level	$\geq$ 8 inches(8)	A
2	Turbine condenser low vacuum(7)	$\geq$ 21 inches Hg vacuum	A
2	Main steamline high radiation(12)	$\leq$ 7 X normal full power background	A
4	Main steamline isolation valve closure(7)	$\leq$ 10% valve closure	A

Amendment No. ~~60~~, ~~86~~, 98

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)	≤ 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	≤ 2.5 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)	≤ 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

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]	≤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(7)	C
16	RCIC turbine area high temperature	≤200° F	C
4	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 ≤ t ≤ 10 seconds.

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

TABLE 3.2-2

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]	≤2.5 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] 4[3]	2/3 core height containment high pressure	≥2/3 core height 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-presure core cooling system operation.
2/BUS [5]	Undervoltage on emergency buses	3045 ± 5% volts	1. Initiates starting of diesel generators. 2. Permissive for starting ECCS pumps. 3. Removes nonessential loads from buses 4. Bypasses degraded voltage timer.

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

reopened.

2) The main steamline isolation valves (one at a time) shall be verified for closure time.

2. In the event any isolation valve specified in Table 3.7-1 becomes inoperable, reactor power operation may continue provided at least one valve in each line having an inoperable valve is in the mode corresponding to the isolated condition.

3. If Specifications 3.7.D.1 and 3.7.D.2 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

4. The temperature of the main steamline air pilot valves shall be less than 170°F except as specified in Specifications 3.7.D.5 and 3.7.D.6 below.

5. From and after the date that the temperature of any main steamline air pilot valve is found to be greater than 170°F reactor operation is permissible only during the succeeding 7 days unless the temperature of such valve is sooner reduced to less than 170°F provided the main steamline isolation valves are operable.

6. If Specification 3.7.D.5 cannot be met, the main steamline isolation valve shall be considered inoperable and action taken in accordance with Specification 3.7.D.2.

2. When an isolation valve listed in Table 3.7-1 is inoperable, the position of at least one other valve in each line having an inoperable valve shall be recorded daily.

will be replaced with filters qualified pursuant to regulatory guide position C.3.d. of Regulatory Guide 1.52 Revision 1 (June 1976). Once per operating cycle demonstration of HEPA filter pressure drop, operability of inlet heaters at rated power, air distribution to each HEPA filter, and automatic initiation of each standby gas treatment system circuit is necessary to assure system performance capability). Note: bases within parentheses will not be applicable until about December 31, 1976, when equipment modifications are completed to allow increased testing.

D. Primary Containment Isolation Valves

Those large pipes comprising a portion of the reactor coolant system, whose failure could result in uncovering the reactor core, are supplied with automatic isolation valves (except those lines needed for emergency core cooling system operation or containment cooling). The closure times specified herein are adequate to prevent loss of more coolant from the circumferential rupture of any of these lines outside the containment than from a steamline rupture. Therefore, this isolation valve closure time is sufficient to prevent uncovering the core.

In order to assure that the doses that may result from a steamline break do no exceed the 10 CFR 100 guidelines, it is necessary that no fuel rod perforation resulting from the accident occur prior to closure of the main steam line isolation valves. Analyses indicate that fuel rod cladding perforations would be avoided for main steam valve closure times, including instrument delay, as long as 10.5 seconds. However, for added margin, the technical specifications require a valve close time of not greater than 5 seconds.

For reactor coolant system temperature less than 212°F, the containment could not become pressurized due to a loss-of-coolant accident. The 212°F limit is based on preventing pressurization of the reactor building and rupture of the blowout panels. These valves are highly reliable, have low service requirement, and most are normally closed. The initiating sensors and associated trip channels are also checked to demonstrate the capability for automatic isolation (reference SAR Section 5.2.2 and Table 5.2.4).

The test interval at once per operating cycle for automatic initiation results in a failure probability of 1.1×10^{-7} that a line will not isolate. More frequent testing for valve operability results in a more reliable system.

The containment is penetrated by a large number of small diameter instrument lines which contact the primary coolant system. A program for periodic testing and examination of the flow check valves in these lines is performed by blowing down the instrument line during a vessel hydro and observing the following conditions, which will verify that the flow check valve is operable:

1. a distinctive 'click' when the poppet valve seats, and
2. an instrumentation high flow that quickly reduces to a slight trickle.

References

1. Quad-Cities Special Report Number 4.
2. R.E. Adams and W. E. Browning Jr., ORNL 3726, 'Iodine Vapor Adsorption Studies for the NS 'Savannah' Project, February 1965.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 101 TO FACILITY OPERATING LICENSE NO. DPR-29

AND AMENDMENT NO. 98 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 letter dated November 27, 1984 as supplemented July 22, 1986, Commonwealth Edison Company (CECo, the licensee) proposed amendments to Appendix A, the Technical Specifications (TS), for Operating License Nos. DPR-29 and DPR-30. The proposed changes would raise the drywell high pressure trip setpoint from 2.0 psig to 2.5 psig, delete the existing biweekly main steam isolation valve surveillance, and delete a note on page 3.7/4.7-10 in Appendix A to DPR-29. The July 22, 1986 supplemental submittal provided new proposed TS pages as a number of administrative typographical errors were identified in the November 27, 1984 submittal.

This Safety Evaluation is a review of the requested changes and their impact on the operation and administration of plant activities.

2.0 SUMMARY OF EVALUATION

The changes proposed by the licensee to the high drywell pressure trip setpoint will reduce the frequency of spurious Engineered Safeguards Features (ESF) actuation without measurably impacting existing safety margins. The change proposed to the frequency of Main Steam Isolation Valve (MSIV) testing is reflective of the demonstrated reliability of the MSIVs and is consistent with Standard Technical Specifications (STS) requirements. Deletion of the footnote on page 3.7/4.7-10 of the Unit 1 TS is reflective of restored equipment operability.

The staff agrees with the changes as described in the proposed amendment.

3.0 EVALUATION

3.1 Proposed Changes to Unit 1 and Unit 2 Technical Specification Tables 3.1-1, 2, 3 and 3.2.1, 2

These tables are revised to reflect the increase in the high drywell pressure trip setpoint from 2.0 to 2.5 psig. The high drywell pressure trip signal is used to initiate certain ESF systems in response to a loss-of-coolant accident (LOCA). An increase in the high drywell pressure trip setpoint from 2.0 to 2.5 psig must be evaluated for its effect on the safety systems to which it provides an actuation signal and the potential for an increase in peak containment accident pressure.

An increase in the high drywell pressure trip setpoint could impact plant response to accident conditions in the following ways:

a. An increased trip setpoint would allow plant operation with a higher nominal drywell pressure. Hence, initial pressure at the beginning of an accident could be higher. This impacts directly the drywell pressure throughout the transient, the volume of water in the torus, primary system blowdown rate, and Emergency Core Cooling System (ECCS) flow rates.

b. The time required to reach the trip setpoint during an accident will be longer at the higher trip point. Consequently, ESF actuation in response to high drywell pressure may be delayed.

With respect to Item a above, the licensee currently controls drywell pressure at a nominal 1.3 psig by approved procedures. As confirmed by discussions with the licensee, no change has been made or is contemplated to existing procedurally controlled drywell pressure limits. However, if drywell pressure were to be maintained at an elevated value, the effect on plant transient behavior would be within analyzed limits for the following reasons:

a. A post accident peak drywell pressure of 47 psig has been calculated for the Quad Cities drywell. This value assumes that the drywell is at atmospheric pressure prior to the accident. Using the conservative assumption that the peak drywell pressure value of 47 psig is a pressure rise due to accident conditions, with a 2.5 psig initial drywell pressure the peak value achieved would be approximately 50 psig, well below the 57 psig drywell design pressure. Thus, drywell integrity will not be compromised.

b. Technical Specifications require that a specified minimum volume of water be maintained in the torus to satisfy energy absorption requirements. If drywell nominal pressure is increased, water will be forced from the drywell through torus vents into the torus. This will, in turn, cause indicated water level in the torus to increase. Operator action to restore indicated torus water level would reduce the torus water inventory. This effect has been analyzed and procedural limits are in place to maintain sufficient water in the torus to satisfy energy absorbing requirements.

c. An increase in drywell nominal pressure at the time of the accident would not appreciably affect primary system blowdown rates as the blowdown occurs under choked flow conditions where flow is essentially independent of downstream pressure.

d. ECCS pump performance would be essentially unchanged in the case of a slightly elevated drywell pressure for the following reasons:

(1) Reactor coolant system pressure versus time is essentially unchanged by a small increase in drywell pressure owing to the choked conditions of break flow. Hence, ECCS pump discharge pressure requirements for those pumps injecting into the reactor coolant system remain unchanged. Pump capabilities would, in fact, be slightly improved owing to the increase in available suction head resulting from increased drywell pressure.

(2) Those pumps discharging directly into the drywell free air space (containment spray pumps) would experience an increased resistance to flow due to higher drywell pressure. This would be partially offset by an increase in available pump suction head resulting from increased drywell pressure. However, as noted in Figure 5.2.17 of the Quad Cities Final Safety Analysis Report, containment pressure remains below design values even in the absence of containment spray flow.

With respect to Item b above, the time required to achieve high drywell pressure ESF actuation is a function of the difference in pressure between drywell ambient and the trip setting. The original Quad Cities analysis assumed that this difference was 2 psig based on the drywell being at atmospheric pressure (0 psig). Mark I Containment studies and resultant analysis have resulted in the current Technical Specification (TS) requirement of normal minimum drywell pressurization of 1.20 psig above the containment wetwell which is normally atmospheric (0.0 psig) or 1.20 psid.

As stated above the original Quad Cities analysis assumed a difference of 2.0 psig between the ESF actuation trip setting and drywell ambient. Also, as stated above, the current TS require normal drywell minimum pressure of 1.20 psig. Therefore, since the current trip setting is 2.0 psig, the current actual pressure differential to actuation is only 0.8 psig (2.0 psig trip - 1.2 psig ambient) even though the current analysis supports a differential as large as 2.0 psig. The time to ESF actuation is larger for 2.0 psig and, therefore, bounding.

Therefore, if the trip setting is increase to 2.5 psig, assuming no change in drywell pressure control requirements, the differential pressure between the new trip setting and drywell ambient would be 1.3 psig (2.5 psig - 1.2 psig ambient = 1.3 psig). Since 1.3 psig is less than 2.0 psig, the original analysis bounds this change; therefore, the staff finds this change acceptable.

3.2 Proposed Changes to Unit 1 and Unit 2 Technical Specifications Surveillance Requirements 4.7.D.1.d and Surveillance Requirement Basis 4.7.D

This surveillance requirement and basis are revised to reflect the deletion of the biweekly Main Steam Isolation Valve surveillance. The purpose of the biweekly MSIV surveillance test was to ensure that the MSIVs would close when required to do so. This test was a result of several instances in early 1971 where one or more MSIVs did not isolate when required. Continued testing has demonstrated the MSIV closure function to be highly reliable.

The General Electric STS, NUREG-0123, and Quad Cities Technical Specifications require certain containment isolation valves, including MSIVs, be tested for operability on a quarterly basis. The STS do not require that the MSIVs undergo a biweekly partial closure test. Thus, the proposed change is consistent with STS requirements.

On these bases, the deletion of the biweekly MSIV surveillance test and corresponding change to the TS basis are acceptable to the staff and do not adversely affect the safety of the plant or the health and safety of the general public.

3.3 Proposed Change to Unit 1 Technical Specifications 3.7/4.7 Page 10

This Technical Specification change deletes an obsolete note at the bottom of the page which required valves MO1-220-2, 3 and 4 be closed during Operating Cycle 7 or until valve MO-220-1 was restored to operable status. Since valve MO-220-1 is now operable, this change does not adversely affect the safety of the plant or the health and safety of the public, and is acceptable to the staff.

4.0 ENVIRONMENTAL CONSIDERATIONS

The amendments involve a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes in surveillance requirements. The staff has determined that the 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 the amendments involve no significant hazards consideration and there has been no public comment on such finding. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

5.0 CONCLUSION

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

Principal Contributors: S. Hare, W. G. Guldemond, T. Rotella

Dated: February 20, 1987

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

February 20, 1987

MEMORANDUM FOR: Sholly Coordinator

FROM: Rajender Auluck, Acting Director
BWR Directorate #1, DPL

SUBJECT: REQUEST FOR PUBLICATION IN BIWEEKLY FR NOTICE - NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES (TAC 56507, 56508)

Commonwealth Edison Company, Docket Nos. 50-254 and 50-265, Quad Cities Nuclear Power Station, Units 1 and 2, Rock Island County, Illinois

Date of application for amendments: November 27, 1984 as supplemented July 22, 1986.

Brief description of amendments: The amendments revise the Technical Specifications to change the containment pressure trip setpoints and the main steam isolation valves surveillance requirements.

Date of issuance: February 20, 1987

Effective date: February 20, 1987

Amendment Nos.: 101 and 98

Facility Operating License Nos. DPR-29 and DPR-30. Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: February 27, 1985 (50 FR 7982). Since the initial notice, the licensee supplemented the application with a submittal dated July 27, 1986. This submittal corrected a number of administrative/typographical errors which did not affect the initial no significant hazards consideration determination nor the substance of the amendment. Therefore, renoticing of the application was not warranted. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 20, 1987.

No significant hazards consideration comments received: No.

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Original signed by
Rajender Auluck, Acting Director
BWR Project Directorate #1, DBL

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