February 22, 1989

Docket Nos.: 50-254
and 50-265

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

Dear Mr. Bliss:

SUBJECT: DELETION OF SNUBBER LISTING FROM TECHNICAL SPECIFICATIONS (TAC NOS. 71179 AND 71180)

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

The Commission has issued the enclosed Amendment Nos. 115and 111to Facility Operating License Nos. DPR-29 and DPR-30 for the Quad Cities Nuclear Power Station, Units 1 and 2. These amendments are in response to your application dated November 15, 1988. In accordance with Generic Letter 84-13, the snubber tables were deleted from Technical Specifications (TS). Also, per your request, all reference to hydraulic snubbers were removed and identified typographical errors were corrected.

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. 115 to
License No. DPR-29
2. Amendment No. 111 to
License No. DPR-30
3. Safety Evaluation

cc w/enclosures:
See next page

Subject to Change

DF01
1/1

PDIII-2:PM	PDIII-2:LA	OGC/WF	PDIII-2:PD
TRoss:dmj	LLuther		DMuller
1/26/89	1/26/89	2/ /89	1/27/89

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

February 22, 1989

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Mr. Henry E. Bliss
Nuclear Licensing Manager
Commonwealth Edison Company
Post Office Box 767
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Dear Mr. Bliss:

SUBJECT: DELETION OF SNUBBER LISTING FROM TECHNICAL SPECIFICATIONS (TAC NOS. 71179 AND 71180)

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

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

Enclosures:
1. Amendment No. 115 to License No. DPR-29
2. Amendment No. 111 to License No. DPR-30
3. Safety Evaluation

cc w/enclosures:
See next page

Mr. Henry E. Bliss
Commonwealth Edison Company

Quad Cities Nuclear Power Station
Units 1 and 2

cc:

Mr. Stephen E. Shelton
Vice President
Iowa-Illinois Gas and
Electric Company
P. O. Box 4350
Davenport, Iowa 52808

Michael I. Miller, Esq.
Sidley and Austin
One First National Plaza
Chicago, Illinois 60603

Mr. Richard Bax
Station Manager
Quad Cities Nuclear Power Station
22710 206th Avenue North
Cordova, Illinois 61242

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

Chairman
Rock Island County Board
of Supervisors
1504 3rd Avenue
Rock Island County Office Bldg.
Rock Island, Illinois 61201

Mr. Michael E. Parker, Chief
Division of Engineering
Illinois Department of Nuclear Safety
1035 Outer Park Drive,
Springfield, Illinois 62704

Regional Administrator, Region III
U. S. Nuclear Regulatory Commission
799 Roosevelt Road, Bldg. #4
Glen Ellyn, Illinois 60137

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. 115
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 15, 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. 115, 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

Daniel R. Muller

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: February 22, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 115

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
v	v
3.6/4.6-12	3.6/4.6-12
3.6/4.6-13	3.6/4.6-13
3.6/4.6-14	3.6/4.6-14
3.6/4.6-15	3.6/4.6-15
3.6/4.6-25	3.6/4.6-25
3.6/4.6-26	3.6/4.6-26
3.6/4.6-27	3.6/4.6-27
3.6/4.6-28	3.6/4.6-28
3.6/4.6-29	3.6/4.6-29
3.6/4.6-30	3.6/4.6-30
3.6/4.6-31	3.6/4.6-31
3.6/4.6-32	3.6/4.6-32
3.6/4.6-33	3.6/4.6-33
3.6/4.6-34	3.6/4.6-34
3.6/4.6-35	3.6/4.6-35
3.6/4.6-36	
3.6/4.6-37	
3.6/4.6-38	
3.6/4.6-39	
3.6/4.6-40	
3.6/4.6-41	
3.6/4.6-42	

TECHNICAL SPECIFICATIONS

APPENDIX A

LIST OF TABLES

I. Shock Suppressors (Snubbers)

1. During all modes of operation except Shutdown and Refuel, all snubbers on safety-related piping systems shall be operable except as noted in 3.6.I.2 following.

I. Shock Suppressors (Snubbers)

The following surveillance requirements apply to all snubbers on safety-related piping systems.

1. Visual inspections shall be performed in accordance with the following schedule utilizing the acceptance criteria given by Specification 4.6.I.2.

Number of Snubbers Found Inoperable During Inspection or During Inspection Interval	Next Required Inspection Interval
0	18 months ±25%
1	12 months ±25%
2	6 months ±25%
3,4	124 days ±25%
5,6,7	62 days ±25%
≥8	31 days ±25%

The required inspection interval shall not be lengthened more than one step at a time.

Snubbers may be categorized in two groups, 'accessible' or 'inaccessible' based on their accessibility for inspection during reactor operation. These two groups may be inspected independently according to the above schedule.

1. (Cont'd)

   Snubber service life monitoring shall be followed by the snubber surveillance inspection records and maintenance history records. The above record retention method shall be used to prevent the snubbers from exceeding a service life.

2. From and after the time that a snubber is determined to be inoperable, continued reactor operation is permissible during the succeeding 72 hours only if the snubber is sooner made operable.

2. Visual inspections shall verify:

   a. There are no visible indications of damage or impaired operability, and

   b. Attachments to the foundation or supporting structure are secure.

3. If the requirements of 3.6.I.1 and 3.6.I.2 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 36 hours.

3. Once each refueling cycle a representative sample of 10% of the total of each type of snubber in use in the plant shall be functionally tested either in place or in a bench test. For each snubber that does not meet the functional test criteria, an additional 10% of that type of snubber shall be functionally tested.

4. If a snubber is determined to be inoperable while the reactor is in the Shutdown or Refuel mode, the snubber shall be made operable prior to reactor startup.

4. The mechanical snubber functional tests shall verify:

   a. That the breakaway force that initiates free movement of the snubber rod in either tension or compression is less than the specified maximum force.

   b. That the activation (restraining action) is achieved within the specified range of acceleration in both tension and compression.

5. When a snubber is deemed inoperable, a review shall be conducted to determine the mode of failure and to decide if an engineering evaluation should be performed. If the engineering evaluation is deemed necessary, it will determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the supported component or system.

6. If any snubber selected for functional testing either fails to lockup or fails to move, i.e., frozen in place, the cause will be evaluated and if determined to be generically deficient all snubbers of the same design, subject to the same defect shall be functionally tested.

7. In addition to the regular sample, snubbers which failed the previous functional test shall be retested during the next test period. If a spare snubber has been installed in place of a failed snubber, then both the failed snubber (if it is repaired and installed in another position) and the spare snubber shall be retested. Test results of these snubbers may not be included for the resampling.

Snubber service life monitoring will be followed by the existing snubber surveillance inspection records and maintenance history records. The above record retention method should be used to prevent mechanical snubbers from exceeding a service life of 40 years.

When the cause of the rejection of a snubber is clearly established and remedied for that snubber and for any other snubbers that may be generically susceptible, and verified by inservice functional testing, that snubber may be exempted from being counted as inoperable. Generically susceptible snubbers are those which are of a specific make or model and have the same design features directly related to rejection of the snubber by visual inspection, or are similarly located or exposed to the same environmental conditions such as temperature, radiation, and vibration.

To provide assurance of snubber functional reliability, a representative sample of the installed snubbers will be functionally tested during plant shutdowns at refueling cycle intervals. Functional testing of the mechanical snubber will consist of verification that the force that initiates free movement of the snubber in either tension or compression is less than the maximum breakaway friction force and verification that the activation (restraining action) is achieved within the specified range of acceleration in both tension and compression.

When a snubber is found inoperable, a review shall be performed to determine the snubber mode of failure. Results of the review shall be used to determine if an engineering evaluation of the safety-related system or component is necessary. The engineering evaluation shall determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the support component or system.

Observed failures of these sample snubbers shall require functional testing of additional units.

TABLE 4.6-1

INSERVICE INSPECTION REQUIREMENTS FOR QUAD-CITIES

Category	Component Parts to be Examined	Examination Method	Frequency of Examination	Examinations[1]
A	Longitudinal and circumferential shell welds in core region			Note: Not applicable with present plant design.
B	Longitudinal and circumferential welds in shell (other than those of categories A and C) and meridional and circumferential seam welds in bottom head and closure head (other than those of Category C)	Volumetric	During each 10-year inspection interval (for 10% of each longitudinal and meridional 5% circumferential length seam)	Accessible top 10 feet of vertical vessel weld in two places (100% inspected in 10 years for approximately 2 feet each refueling outage). 10% of meridional seam welds in vessel closure head and 5% of circumferential welds in vessel closure head Note: Bottom head closure not applicable with present plant design.
C	Vessel-to-flange and head-to-flange circumferential welds	Volumetric	Cumulative 100% coverage at end of 10-year interval	Equivalent to 10% of vessel-to-flange and head-to-flange circumferential weld area each refueling outage
D	Primary nozzle-to-vessel and nozzle-to-head welds and nozzle-to-vessel, nozzle-to head inside radiused section	Volumetric	Cumulative 100% coverage at end of 10-year interval	Nozzle welds: Recirculation outlet(2): once every 5 years Recirculation inlet(10): at least once each refueling outage

TABLE 4.6-1 (Cont'd)

Category	Component Parts to be Examined	Examination Method	Frequency of Examination	Examinations[1]
				Core spray inlet(2): once every 5 years Control rod drive return(1): once every 10 years Standby liquid control(1): once every 10 years Head instrumentation(2): once every 5 years Head spray inlet(1): once every 10 years
E	Partial penetration welds including control rod drive penetrations and vessel instrumentation nozzles	Visual	The examinations performed during each inspection interval shall cover at least 25% of each group of penetrations of comparable size and function	The area surrounding each penetration shall be examined for evidence of leakage during pressure testing
F	Primary nozzles to safe-end welds	Visual, surface, and volumetric	Cumulative 100% coverage at end of 10-year interval	Safe-ended nozzles: Recirculation outlet(2): once every 5 years Recirculation inlet(10): at least once each refueling outage Core spray inlet(2): once every 5 years Control rod drive(1): once every 10 years Standby liquid control(1): once every 10 years Head instrumentation(2): once every 5 years Head spray inlet(1): once every 10 years

TABLE 4.6-1 (Cont'd)

Category	Component Parts to be Examined	Examination Method	Frequency of Examination	Examinations[1]
G-1	Closure studs and nuts	Volumetric and visual or surface	Cumulative 100% coverage at end of 10-year interval	100% of vessel studs and nuts will be inspected each refueling outage
	Ligaments between threaded stud holes	Volumetric	Cumulative 100% coverage at end of 10-year interval	Equivalent to 10% of ligaments each refueling outage. Examination of bushings, threads, and ligaments in base material of flanges may be performed from the face of the flange and are required to be examined only when the connection is disassembled
	Closure washers, bushings	Visual	Cumulative 100% coverage at end of 10-year interval	Equivalent to 10% of washers each refueling outage, bushings not applicable with present design.
	Pressure-retaining bolting $\geq$ 2 inch diameter	Visual and volumetric	Cumulative 100% coverage at end of 10-year interval	Equivalent to 10% of recirculating pump bolts each refueling outage.
G-2	Pressure-retaining bolting $\leq$ 2 inch diameter	Visual	Cumulative 100% of coverage at end of 10-year interval	Bolting will be examined when bolting is removed or when the bolted connected is broken or disassembled. For bolting which is not removed or where the bolted connection is not broken, the inspection will consist of a visual examination to detect signs of distress or evidence of leaking.

TABLE 4.6-1 (Cont'd)

Category	Component Parts to be Examined	Examination Method	Frequency of Examination	Examinations[1]
H	Integrally welded vessel supports	Volumetric	During 10-year interval	10% (approximately 8 ft) of lineal feet of vessel support skirt welding in 10th year.
I	Closure head cladding	Visual and surface or volumetric	During 10-year interval	During the 10-year interval, at least six patches (each 36 in^2) evenly distributed in the closure head.
	Vessel cladding	Visual	During 10-year interval	6 patches (each 36 in^2) evenly distributed in the accessible sections of the vessel shell shall be examined.

TABLE 4.6-1 (Cont'd)

Category	Component Parts to be Examined	Examination Method	Frequency of Examination	Examinations[1] System	Pipe Sizes	Unit 1 Total Welds
J	Circumferential and longitudinal pipe welds (Refer to Note 2 at the end of this table for a breakdown of these welds.)	Visual and volumetric	Cumulative 25% of all weld joints (selectively distributed among the higher stress joints in entire system) every 10 years. Group I and Group II welds (See Note 1 for location breakdown) on main feedlines and main steamlines shall be inspected in 10 years during the first period. At least 25% of the welds shall be inspected at approximately each 2-1/2-year interval. Group I welds shall be inspected during each 10-year period thereafter.	Shutdown cooling	20-in.	17
				RCIC	3-in., 4-in.	33
				Reactor water cleanup	4-in., 6-in.	27
				CRD hydraulic system	3-in., 4-in.	18
				RHR	16-in.	29
				Head spray	4-in.	28
				Core spray piping	10-in.	32
				HPCI	10-in., 14-in.	22
				Feed piping	4-in., 12-in., 18-in.	96
				Recirculation	4-in., 12-in., 22-in., 22-in., 28-in.	133
				Main Steam	3-in., 20-in.	119
K-1	Integrally-welded external support attachments for piping, valves, and pumps	Visual and volumetric	100% cumulative in first 10 years 25% cumulative in each following 10-year inspection interval	Welds to the pressure-containing boundary, the base metal beneath the weld zone, and along the support attachment member for a distance of two base metal thicknesses.		

TABLE 4.6-1 (Cont'd)

Category	Component Parts to be Examined	Examination Method	Frequency of Examination	Examinations[1]
K-2	Support members and structures for piping, valves, and pumps whose structural integrity is relied upon to withstand design loads and seismic-induced displacements.	Visual	100% cumulative during each 10-year inspection interval	Support settings of constant and variable spring type hangers, snubbers, and shock absorbers shall be inspected to verify proper distribution of design loads among the associated support components.
L-1	Pump casing welds	Visual and volumetric	One pump of each type during 10-year interval	Not applicable with present plant design.
L-2	Pump casings	Visual	One pump of each type during 10-year interval if disassembled	One recirculating pump in 10 years.
M-1	Welds in valve bodies 3 inches and above	Visual and volumetric	One valve of each type during 10-year interval	Not applicable with present plant design
M-2	Valve bodies 3 inches and above	Visual	One valve of each type during 10-year interval if disassembled	One disassembled valve (with or without welds and 3 inches over in normal size) in each category and type shall be subject to visual examination. Individual examination shall cover 100% of the pressure boundary welds and may be performed at or near the end of the 10-year interval.

TABLE 4.6-1 (Cont'd)

Category	Component Parts to be Examined	Examination Method	Frequency of Examination	Examinations[1]
N	Interior surfaces and internals and integrally welded internal supports of the reactor vessel, including core spray spargers, core spray nozzles, and upper portions of jet pumps	Visual (not Inservice Inspection Code)	During first refueling outage and during subsequent refueling outages at approximately 3-year intervals	Interior surfaces and internal components of the reactor vessel, including the space at the bottom head and internal attachments which are welded to the vessel made accessible by the removal of components during normal refueling operations. All internal attachments whose failure may adversely affect core integrity shall be examined.
O	Control rod drive housing pressure-retaining welds.	Volumetric	The examinations performed during each inspection interval shall include the welds in 10% of the peripheral control rod drive housings.	The areas shall include the weld metal and base metal for one weld thickness beyond the edge of the weld.

NOTES:

[1] Extent of Examinations

Examinations which reveal unacceptable structural defects in a category shall be extended to include an additional number (or areas) of system components or piping in the same category approximately equal to those initially examined. In the event further unacceptable structural defects are revealed, all remaining system components or piping in the category shall be examined to the extent specified in that examination category.

TABLE 4.6-1 (Cont'd)

[2] Category J Weld Breakdown

Main Steamline - Group I Welds

Line	Weld Identification Unit 1
3001A-20-in.	30A-S11
3001B-20-in.	30B-S10
3001C-20-in.	30C-S10
3001D-20-9n.	30D-S10

Group II Welds

Line	Weld Identification Unit 1
3001A-20-in.	30A-S22
	30A-F23
	30A-F24
3001B-20-in.	30B-S25
	30B-F26
	30B-F27
3001C-20-in.	30C-S21
	30C-F22
	30C-F23
3001D-20-in.	30D-S20
	30D-F21
	30D-F22

Feedwater Line - Group I Welds

Line	Weld Identification Unit 1
3204A-18-in.	32A-S5
3204B-18-in.	32B-S4

Group II Welds

Line	Weld Identification Unit 1
3204A-18-in.	32A-S1
	32A-F3
	32A-F7
3204B-18-in.	32B-S1
	32B-F6
3204C-12-in.	32C-S4
3204D-12-in.	32D-S4
	32D-S8
3204E-12-in.	32E-F9
	32D-S2
3204F-12-in.	32F-S2
	32F-F6

[3] Supplemental Inspection Program for First and Second Refueling Outages

a. The following critical and sensitized components shall be nondestructively examined by the methods indicated:

Component	Examination Method
Bimetallic welds of field-replaced safe-ends	PT and (UT or RT)

TABLE 4.6-1 (Cont'd)

b. The areas subject to examination shall include 100% of the exterior surfaces of the welds in Item 1. Weld areas to be examined shall include the base material for at least one wall thickness beyond the edge of the weld.

c. All examinations shall be conducted in accord with the examination techniques and procedures and meet the acceptance standards specified in the ASME Section XI Inservice Inspection Code and supplemented where necessary by special techniques with demonstrated capability to detect stress-corrosion cracking.

d. The examination frequency shall conform to the following schedule:

Bimetallic welds of field-replaced safe-ends

1) 25% at or within the first refueling outage 2) 25% at or within the second refueling outage

e. In the event any of the examinations for Item 4 reveal indications of structural defects which upon evaluation require repairs or replacements, the specified examination frequency shall be subject to review by the NRC.

TABLE 4.6-2

REVISED WITHDRAWAL SCHEDULE FOR QUAD-CITIES UNIT 1

Withdrawal Year	Part No.	Location	Comments
1982	8	Wall - 215°	
2002	7	Wall - 95°	
	9	Wall - 245°	Standby
	5	Wall - 65°	Standby
	10	Wall - 275°	Standby
1981	4	Near Core Top Guide - 90°	
1984	6	Near Core Top Guide - 180°	

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. 111
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 15, 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-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. 111, 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

Daniel R. Muller

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: February 22, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 111

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
iv	iv
3.6/4.6-5b	3.6/4.6-5b
3.6/4.6-6	3.6/4.6-6
3.6/4.6-7	3.6/4.6-7
3.6/4.6-14	3.6/4.6-14
3.6/4.6-14a	3.6/4.6-14a
3.6/4.6-15	3.6/4.6-15
3.6/4.6-15a	
3.6/4.6-15b	
3.6/4.6-15c	
3.6/4.6-15d	
3.6/4.6-15e	

TECHNICAL SPECIFICATIONS

APPENDIX A

LIST OF TABLES

I. Shock Suppressors (Snubbers)

1. During all modes of operation except Shutdown and Refuel, all snubbers on safety related piping systems shall be operable except as noted in 3.6.I.2 following.

2. From and after the time that a snubber is determined to be inoperable, continued reactor operation is permissible during the succeeding 72 hours only if the snubber is sooner made operable.

3. If the requirements of 3.6.I.1 and 3.6.I.2 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 36 hours.

4. If a snubber is determined to be inoperable while the reactor is in the Shutdown or Refuel mode, the snubber shall be made operable prior to reactor startup.

I. Shock Suppressors (Snubbers)

The following surveillance requirements apply to all snubbers on safety related piping systems.

1. Visual inspections shall be performed in accordance with the following schedule utilizing the acceptance criteria given by Specification 4.6.I.2.

Number of Snubbers Found Inoperable During Inspection or During Inspection Interval	Next Required Inspection Interval
0	18 months $\pm 25\%$
1	12 months $\pm 25\%$
2	6 months $\pm 25\%$
3,4	124 days $\pm 25\%$
5,6,7	62 days $\pm 25\%$
≥ 8	31 days $\pm 25\%$

The required inspection interval shall not be lengthened more than one step at a time.

Snubbers may be categorized in two groups, 'accessible' or 'inaccessible' based on their accessibility for inspection during reactor operation. These two groups may be inspected independently according to the above schedule.

Snubber service life monitoring shall be followed by the snubber surveillance inspection records and maintenance history records. The above record retention method shall be used to prevent the snubbers from exceeding a service life.

2. Visual inspections shall verify:

   a. There are no visible indications of damage or impaired operability, and

   b. Attachments to the foundation or supporting structure are secure.

3. Once each refueling cycle a representative sample of 10% of the total of each type of snubber in use in the plant shall be functionally tested either in place or in a bench test. For each snubber that does not meet the functional test criteria, an additional 10% of that type of snubber shall be functionally tested.

4. The mechanical snubber functional tests shall verify:

   a. That the breakaway force that initiates free movement of the snubber rod in either tension or compression is less than the specified maximum force.

   b. That the activation (restraining action) is achieved within the specified range of acceleration in both tension and compression.

5. When a snubber is deemed inoperable, a review shall be conducted to determine the mode of failure and to decide if an engineering evaluation should be performed. If the engineering evaluation is deemed necessary, it will determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the supported component or system.

6. If any snubber selected for functional testing either fails to lockup or fails to move, i.e., frozen in place, the cause will be evaluated and if determined to be generically deficient all snubbers of the same design, subject to the same defect shall be functionally tested.

7. In addition to the regular sample, snubbers which failed the previous functional test shall be retested during the next test period. If a spare snubber has been installed in place of a failed snubber, then both the failed snubber (if it is repaired and installed in another position) and the spare snubber shall be retested. Test results of these snubbers may not be included for the resampling.

I. Snubbers

All snubbers are required OPERABLE to ensure that the structural integrity of the reactor coolant system and all other safety-related systems is maintained during and following a seismic or other event initiating dynamic loads. Snubbers excluded from this inspection program are those installed on non-safety-related systems and then only if their failure or failure of the system on which they are installed, would have no adverse effect on any safety-related system.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to systems. Therefore, the required inspection interval varies inversely with the observed snubber failures and is determined by the number of inoperable snubbers found during an inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

Snubber service life monitoring will be followed by the existing snubber surveillance inspection records and maintenance history records. The above record retention method should be used to prevent the mechanical snubbers from exceeding a service life of 40 years.

When the cause of the rejection of a snubber is clearly established and remedied for that snubber and for any other snubbers that may be generically susceptible, and verified by inservice functional testing, that snubber may be exempted from being counted as inoperable. Generically susceptible snubbers are those which are of a specific make or model and have the same design features directly related to rejection of the snubber by visual inspection, or are similarly located or exposed to the same environmental conditions such as temperature, radiation, and vibration.

To provide assurance of snubber reliability, a representative sample of the installed snubbers will be functionally tested during plant shutdowns at refueling cycle intervals. Functional testing of the mechanical snubber will consist of verification that the force that initiates free movement of the snubber in either tension or compression is less than the maximum breakaway friction force and verification that the activation (restraining action) is achieved within the specified range of acceleration in both tension and compression.

When a snubber is found inoperable, a review shall be performed to determine the snubber mode of failure. Results of the review shall be used to determine if an engineering evaluation of the safety-related system or component is necessary. The engineering evaluation shall determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the support component or system.

Observed failures of these sample snubbers shall require functional testing of additional units.

Minimum Temperature Requirements per Appendix G of 10 CFR 50

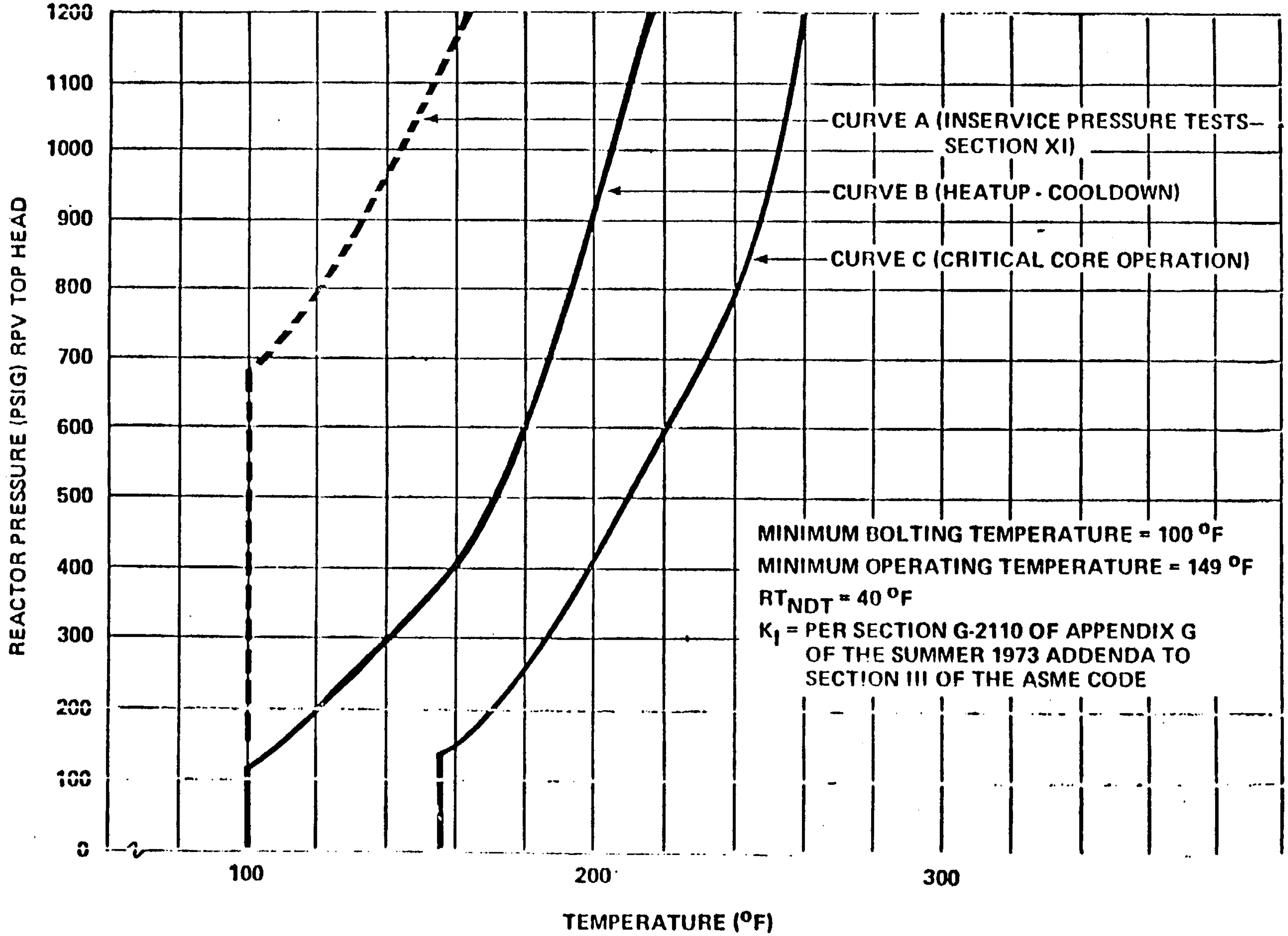

Fig. 3.6.1

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 115 TO FACILITY OPERATING LICENSE NO. DPR-29

AND AMENDMENT NO. 111 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 15, 1988, Commonwealth Edison Company (CECo, the licensee) submitted an application to amend the Quad Cities Nuclear Power Station (QCNPS) Technical Specifications (TS). The proposed amendments to DPR-29 and DPR-30 would change TS to: 1) delete tabular listings of individual safety-related snubbers, 2) delete all references made to hydraulic snubbers, and 3) correct editorial and typographical errors.

2.0 EVALUATION

Deleting the tables of safety-related snubbers from TS was endorsed by the NRC in Generic Letter (GL) 84-13 dated May 3, 1984. According to GL 84-13, as long as all safety-related snubbers were properly identified and tracked, they need not be listed in TS. The licensee only needs to specify in TS that snubbers are required to be operable for supporting safety-related systems. Removing snubber tables from the TS, was intended to eliminate the large number of unnecessary amendments by licensees whenever snubbers were added, deleted, or modified. However, GL 84-13 did not intend for licensees to revise the operational, surveillance, or record keeping requirements of safety-related snubbers. Furthermore, even though they would no longer be in TS, any subsequent changes in snubber quantities, types, or locations would be considered a change to the facility, and such changes are subject to the provisions of 10 CFR 50.59.

CECo proposed to delete Table 3.6.-1, "Safety Related Shock Suppressors (Snubbers)," and replace all references to this table with the statement that specifies "all snubbers on safety-related systems." The record of installed safety-related snubbers will be maintained within the control of QCNPS administrative procedures in order to comply with the record keeping requirements of TS 4.6.1.1. CECo did not significantly revise any of the operational, surveillance, or record keeping requirements of TS. The NRC staff concludes their proposed deletion of TS Table 3.6-1 and associated reference changes are consistent with the guidance provided in GL 84-13, and therefore acceptable, since the operational, surveillance and recordkeeping requirements for safety-related snubbers will remain essentially unchanged.

The test requirements of hydraulic snubbers are no longer needed by QCNPS as the total population of safety-related snubbers consist of mechanical snubbers. Therefore, the proposal to delete all reference to hydraulic snubbers is acceptable to the NRC staff.

Several typographical and editorial corrections were proposed by CECo in their amendment. The NRC staff considers these to be acceptable.

CECo should ensure the QCNPS FSAR is properly updated to reflect any and all of the aforementioned changes.

3.0 ENVIRONMENTAL CONSIDERATION

These amendments involve changes to a requirements related to the use and surveillance of facility components 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 comments 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 (3) 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: Thierry Ross

Dated: February 22, 1989