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

September 25, 1991

Posted
Amdt. 132 to DPR-29

Docket Nos. 50-254
and 50-265

Mr. Thomas J. Kovach
Nuclear Licensing Manager
Commonwealth Edison Company-Suite 300
OPUS West III
1400 OPUS Place
Downers Grove, Illinois 60515

Dear Mr. Kovach:

SUBJECT: ISSUANCE OF AMENDMENT AND EVALUATION OF RESPONSE TO GENERIC LETTER 88-11 (TAC NOS. 71538 AND 71539)

In response to Generic Letter (GL) 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations," you submitted letters dated November 28, 1988, June 26, 1989, October 23, 1989, March 23, 1990, and July 26, 1991.

As a result of utilizing the methodology described in Revision 2 of Regulatory Guide 1.99, which was recommended in the GL, your October 23, 1989, response proposed revisions to the Technical Specifications pressure/temperature operating limits for both Quad Cities units. Additional clarifying information was provided in your March 23, 1990 submittal and your July 26, 1991 submittal revised the Bases.

The NRC staff has reviewed your application and the Commission has issued the enclosed Amendment No. 132 to Facility Operating License No. DPR-29 and Amendment No. 127 to Facility Operating License No. DPR-30 for Quad Cities, Units 1 and 2, respectively. The amendments also include several minor corrections to the Table of Contents. Your application also proposed to remove the reactor vessel venting requirement; however, the staff, in a letter dated March 1, 1990, rejected this change because of the uncertainty regarding the basis for this requirement.

As a result of its review of your response to GL 88-11, the staff identified two open items that require resolution.

1. Section IV.B of Appendix G requires that the predicted Charpy upper shelf energy (USE) at end-of-life (EOL) be above 50 ft-lb. Since you could not provide unirradiated USE values for beltline materials for Quad Cities, Units 1 and 2, the staff could not determine whether the limiting USE at end-of-life satisfies the minimum of 50 ft-lb requirement. The surveillance capsule reports provided a few USE data, however, the data could not be used in calculating the limiting USE because it could not be correlated to a specific vessel material. The staff needs to know the unirradiated USE of all beltline materials in order to determine the limiting USE.

2. The reactor vessel materials surveillance program for Quad Cities, Units 1 and 2, does not comply with ASTM E 185 and 10 CFR 50, Appendix H, because the surveillance specimens could not be correlated with the reactor vessel beltline materials.

You are requested to provide a plan for resolving these two open items within 6 months of the issuance of this letter.

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

Sincerely,

Original Signed By:

Leonard N. Olshan, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 132 to DPR-29
2. Amendment No. 127 to DPR-30
3. Safety Evaluation

cc w/enclosures:
See next page

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Mr. Thomas J. Kovach
Commonwealth Edison Company

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Unit Nos. 1 and 2

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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. 132
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 28, 1988, June 26, 1989, October 23, 1989, March 23, 1990, and July 26, 1991, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B. of Facility Operating License No. DPR-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. 132, 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

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 25, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 132

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
ii	ii
v	v
vi	vi
vii	vii
3.6/4.6-2	3.6/4.6-2
3.6/4.6-16	3.6/4.6-16
3.4/4.6-17	3.6/4.6-17
---	3.6/4.6-17a
Figure 3.6-1	Figure 3.6-1
Figure 3.6-2	---
3.6/4.6-35	3.6/4.6-35

TABLE OF CONTENTS (Cont'd)

	Page
3.4/4.4 STANDBY LIQUID CONTROL SYSTEM	3.4/4.4-1
A. Normal Operation	3.4/4.4-1
B. Operation with Inoperable Components	3.4/4.4-2
C. Liquid Poison Tank-Boron Concentration	3.4/4.4-2
3.4 Limiting Conditions for Operation Bases	3.4/4.4-4
3.5/4.5 CORE CONTAINMENT COOLING SYSTEMS	3.5/4.5-1
A. Core Spray Subsystems and the LPCI Mode of the RHR System	3.5/4.5-1
B. Containment Cooling Mode of the RHR System	3.5/4.5-3
C. HPCI Subsystem	3.5/4.5-5
D. Automatic Pressure Relief Subsystems	3.5/4.5-6
E. Reactor Core Isolation Cooling System	3.5/4.5-7
F. Minimum Core and Containment Cooling System Availability	3.5/4.5-8
G. Maintenance of Filled Discharge Pipe	3.5/4.5-10
H. Condensate Pump Room Flood Protection	3.5/4.5-11
I. Average Planar Linear Heat Generation Rate (APLHGR)	3.5/4.5-13
J. Local LHGR	3.5/4.5-13
K. Minimum Critical Power Ratio (MCPR)	3.5/4.5-14
3.5 Limiting Conditions for Operation Bases	3.5/4.5-15
4.5 Surveillance Requirements Bases	3.5/4.5-23
3.6/4.6 PRIMARY SYSTEM BOUNDARY	3.6/4.6-1
A. Thermal Limitations	3.6/4.6-1
B. Pressurization Temperature	3.6/4.6-2
C. Coolant Chemistry	3.6/4.6-3
D. Coolant Leakage	3.6/4.6-5
E. Safety and Relief Valves	3.6/4.6-6
F. Structural Integrity	3.6/4.6-6
G. Jet Pumps	3.6/4.6-9
H. Recirculation Pump Flow Limitations	3.6/4.6-11
I. Shock Suppressors (Snubbers)	3.6/4.6-12
3.6 Limiting Conditions for Operation Bases	3.6/4.6-16
3.7/4.7 CONTAINMENT SYSTEMS	3.7/4.7-1
A. Primary Containment	3.7/4.7-1
B. Standby Gas Treatment System	3.7/4.7-12
C. Secondary Containment	3.7/4.7-17
D. Primary Containment Isolation Valves	3.7/4.7-18

QUAD CITIES
DPR-29

TECHNICAL SPECIFICATIONS

APPENDIX A

LIST OF TABLES

Number	Title	Page
3.1-1	Reactor Protection System (Scram) Instrumentation Requirements-Refuel Mode	3.1/4.1-11
3.1-2	Reactor Protection System (Scram) Instrumentation Requirements-Startup/Hot Standby Mode	3.1/4.1-12
3.1-3	Reactor Protection System (Scram) Instrumentation Requirements-Run Mode	3.1/4.1-13
3.1-4	Notes for Tables 3.1-1, 3.1-2, and 3.1-3	3.1/4.1-14
4.1-1	Scram Instrumentation and Logic Systems Functional Test-Minimum Functional Test Frequencies for Safety Instrumentation, Logic Systems, and Control Circuits	3.1/4.1-15
4.1-2	Scram Instrument Calibration-Minimum Calibration Frequencies for Reactor Protection Instrument Channels	3.1/4.1-17
3.2-1	Instrumentation that Initiates Primary Containment Isolation Functions	3.2/4.2-15
3.2-2	Instrumentation that Initiates or Controls the Core and Containment Cooling Systems	3.2/4.2-17
3.2-3	Instrumentation that Initiates Rod Block	3.2/4.2-19
3.2-4	Postaccident Monitoring Instrumentation Requirements	3.2/4.2-21
3.2-5	Radioactive Liquid Effluent Monitoring Instrumentation	3.2/4.2-24
3.2-6	Radioactive Gaseous Effluent Monitoring Instrumentation	3.2/4.2-25
4.2-1	Minimum Test and Calibration Frequency for Core and Containment Cooling Systems Instrumentation, Rod Blocks and Isolation	3.2/4.2-27
4.2-2	Postaccident Monitoring Instrumentation Surveillance Requirements	3.2/4.2-30
4.2-3	Radioactive Liquid Effluent Monitoring Instrumentation Surveillance Requirements	3.2/4.2-32
4.2-4	Radioactive Gaseous Effluent Monitoring Instrumentation Surveillance Requirements	3.2/4.2-33
4.6-1	Inservice Inspection Requirements for Quad-Cities	3.6/4.6-26
4.6-2	Revised Withdrawal Schedule	3.6/4.6-35
3.7-1	Primary Containment Isolation	3.7/4.7-33
3.7-2	Primary Containment Leakage Test Penetrations	3.7/4.7-39
4.7-1	Temporarily Untested Pathways	3.7/4.7-40
4.8-1	Radioactive Gaseous Waste Sampling and Analysis Program	3.8/4.8-27

LIST OF TABLES (Cont'd)

Number	Title	Page
4.8-2	Maximum Permissible Concentration of Dissolved or Entrained Noble Gases Released from the Site to Unrestricted Areas in Liquid Waste	3.8/4.8-29
4.8-3	Radioactive Liquid Waste Sampling and Analysis Program	3.8/4.8-30
4.8-4	Radiological Environmental Monitoring Program	3.8/4.8-32
4.8-5	Reporting Levels for Radioactivity Concentrations in Environmental Samples	3.8/4.8-35
4.8-6	Practical Lower Limits of Detection (LLD) for Standard Environmental Radiological Monitoring Program	3.8/4.8-36
4.11-1	Surveillance Requirements for High Energy Piping Outside Containment	3.11/4.11-3
3.12-1	Fire Detection Instruments	3.12/4.12-11
3.12-2	Sprinkler Systems	3.12/4.12-12
3.12-3	CO_2 Systems	3.12/4.12-12
3.12-4	Fire Hose Stations	3.12/4.12-13
6.6-1	Special Reports	6.6-5

QUAD CITIES
DPR-29

TECHNICAL SPECIFICATIONS

APPENDIX A

LIST OF FIGURES

Number	Title
2.1-1	APRM Flow Reference Scram and APRM Rod Block Settings
2.1-2	Deleted
2.1-3	APRM Flow Bias Scram Relationship to Normal Operating Conditions
4.1-1	Graphical Aid in the Selection of an Adequate Interval Between Tests
4.2-1	Test Interval vs. System Unavailability
3.4-1	Deleted
3.4-2	Sodium Pentaborate Solution Temperature Requirements
3.5-1	Deleted
3.5-2	Deleted
3.6-1	Pressure Limit as a Function of Vessel Metal Temperature
3.6-2	Deleted
4.6-1	Chloride Stress Corrosion Test Results at 500°F
4.8-1	Locations of Fixed Environmental Radiological Monitoring Stress
6.1-1	Deleted
6.1-2	Deleted
6.1-3	Minimum Shift Manning Chart

4. The recirculation pump in an idle recirculation loop shall not be started unless the coolant in that loop is within 50°F of the operating loop coolant temperature.

B. Pressurization Temperature

1. The reactor vessel shall be vented and power operation shall not be conducted unless the reactor vessel temperature is equal to or greater than that shown in Figure 3.6-1. Operation for hydrostatic or leakage tests (Curve A), during heatup or cooldown (Curve B), or with the core critical (Curve C) shall be conducted only when vessel temperature is equal to or above that shown in the appropriate curve of Figure 3.6-1. Figure 3.6-1 is effective through 16 EFPY. At least six months prior to 16 EFPY new curves will be submitted.

2. The reactor vessel head bolting studs shall not be under tension unless the temperature of the vessel shell immediately below the vessel flange is ≥ 100°F.

B. Pressurization Temperature

1. Reactor vessel shell temperature and reactor coolant pressure shall be permanently recorded at 15-minute intervals whenever the shell temperature is below 220°F and the reactor vessel is not vented.

2. Neutron flux monitors and samples shall be installed in the reactor vessel adjacent to the vessel wall at the core midplane level. The monitor and sample program shall conform to ASTM E 185-66. The monitors and samples shall be removed and tested in accordance with the guidelines set forth in 10CFR50 Appendix H to experimentally verify the calculated values of integrated neutron flux that are used to determine the NDTT for Figure 3.6-1.

3.6 LIMITING CONDITIONS FOR OPERATION BASES

A. Thermal Limitations

The reactor vessel design specification requires that the reactor vessel be designed for a maximum heatup and cooldown rate of the contained fluid (water) of 100°F/hour averaged over a period of 1 hour. This rate has been chosen based on past experience with operating power plants. The associated time periods for heatup and cooldown cycles when the 100°F/hr rate is limiting provides for efficient but safe plant operation.

The reactor vessel manufacturer has designed the vessel to the above temperature criterion. In the course of completing the design, the manufacturer performed a detailed stress analysis. This analysis includes more severe thermal conditions than those which would be encountered during normal heating and cooling operations.

Specific analyses were made based on a heating and cooling rate of 100°F/hr applied continuously over a temperature range of 100°F to 550°F. Because of the slow temperature-time response of the massive flanges relative to the adjacent head and shell sections, calculated temperatures obtained were 500°F (shell) and 360°F (flange) (140°F differential).

Both axial and radial thermal stresses were considered to act concurrently with full primary loadings. Calculated stresses were within ASME Boiler and Pressure Vessel Code Section III stress intensity and fatigue limits even at the flange area where maximum stress occurs.

The flange metal temperature differential of 140°F occurred as a result of sluggish temperature response and the fact that the heating rate continued over a 450°F coolant temperature range.

The uncontrolled cooldown rate of 240°F/hr was based on the maximum expected transient over the lifetime of the reactor vessel. This maximum expected transient is the injection of cold water into the vessel by the high-pressure coolant injection subsystem. This transient was considered in the design of the pressure vessel, and ten such cycles were considered in the design. Detailed stress analyses were conducted to assure that the injection of cold water into the vessel by the HPCI would not exceed ASME stress code limitations.

B. Pressurization Temperature

The reactor vessel is a primary barrier against the release of fission products to the environs. In order to provide assurance that this barrier is maintained at a high degree of integrity, pressure-temperature limits have been established for the operating conditions to which the reactor vessel can be subjected. Figure 3.6-1 presents the pressure-temperature curves for those operating conditions; Inservice Hydrostatic Testing (Curve A), Non-Nuclear Heatup/Cooldown (Curve B), and Core Critical Operation (Curve C). These curves have been established to be in conformance

with Appendix G to 10 CFR 50 and Regulatory Guide 1.99, Revision 2, and take into account the change in reference nil-ductility transition temperature (RT_{NDT}) as a result of neutron embrittlement. The adjusted reference temperature (ART) of the limiting vessel material is used to account for irradiation effects.

Three vessel regions are considered for the development of the pressure-temperature curves: 1) the core beltline region; 2) the non-beltline region (other than the closure flange region); and 3) the closure flange region. The beltline region is defined as that region of the reactor vessel that directly surrounds the effective height of the reactor core (between the bottom and top of active fuel), and is subject to an RT_{NDT} adjustment to account for irradiation embrittlement. The non-beltline and closure flange regions receive insufficient fluence to necessitate an RT_{NDT} adjustment.

These regions contain components which include; the reactor vessel nozzles, closure flanges, top and bottom head plates, control rod drive penetrations, and shell plates that do not directly surround the reactor core. Although the closure flange region is a non-beltline region, it (the closure flange region) is treated separately for the development of the pressure-temperature curves to address 10 CFR 50 Appendix G requirements.

In evaluating the adequacy of the steel which comprises the reactor vessel, it is necessary that the following be established: 1) the RT_{NDT} for all vessel and adjoining materials; 2) the relationship between RT_{NDT} and integrated neutron flux (fluence, at energies greater than one Mev); and 3) the fluence at the location of a postulated flaw.

Boltup Temperature

The initial RT_{NDT} of the main closure flanges, the shell and head materials connecting to these flanges, and connecting welds is 10°F; however, the vertical electroslag welds which terminate immediately below the vessel flange have an RT_{NDT} of 40°F. Therefore, the minimum allowable boltup temperature is established as 100°F (RT_{NDT} + 60°F), which includes a 60°F conservatism required by the original ASME Code of construction.

Curve A - Hydrotesting

As indicated in Curve A of Figure 3.6-1 for system hydrotesting, the minimum metal temperature of the reactor vessel shell is 100°F for reactor pressures less than 312 psig. This 100°F minimum boltup temperature is based on an RT_{NDT} of 40°F for the electroslag weld immediately below the vessel flange and a 60°F conservatism required by the original ASME Code of construction.

At reactor pressures greater than 312 psig, the minimum vessel metal temperature is established as 130°F. The 130°F minimum temperature is based on a closure flange region RT_{NDT} of 40°F and a 90°F conservatism

required by 10 CFR 50 Appendix G for pressure in excess of 20% of the preservice hydrostatic test pressure (1563 psig).

At approximately 650 psig the effects of pressurization are more limiting than the boltup stresses at the closure flange region, hence a family of non-linear curves intersect the 130°F vertical line. Beltline as well as non-beltline curves have been provided to allow separate monitoring of the two regions. Beltline curves as a function of vessel exposure for 12, 14 and 16 effective full power years (EFPY) are presented to allow the use of the appropriate curve up to 16 EFPY of operation.

Curve B - Non-Nuclear Heatup/Cooldown

Curve B of Figure 3.6-1 applies during heatups with non-nuclear heat (e.g., recirculation pump heat) and during cooldowns when the reactor is not critical (e.g., following a scram). The curve provides the minimum reactor vessel metal temperatures based on the most limiting vessel stress.

As indicated by the vertical 100°F line, the boltup stresses at the closure flange region are most limiting for reactor pressures below approximately 110 psig. For reactor pressures greater than approximately 110 psig, pressurization and thermal stresses become more limiting than the boltup stresses, which is reflected by the non-linear portion of Curve B. The non-linear portion of the curve is dependent on non-beltline and beltline regions, with the beltine region temperature limits having been adjusted to account for vessel irradiation (up to a vessel exposure of 16 EFPY). The non-beltline region is limiting between approximately 110 psig and 830 psig. Above approximately 830 psig, the beltline region becomes limiting.

Curve C - Core Critical Operation

Curve C, the core critical operation curve shown in Figure 3.6-1, is generated in accordance with 10 CFR 50 Appendix G which requires core critical pressure-temperature limits to be 40°F above any Curve A or B limits. Since Curve B is more limiting, Curve C is Curve B plus 40°F.

The withdrawal schedule in Table 4.6-2 is based on the three capsule surveillance program as defined in Section II.C of 10 CFR 50 Appendix H. The accelerated capsule (Near Core Top Guide) are not required by Appendix H.

This surveillance program conforms to ASTM E 185-73 "Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels" with one exception. The base metal specimens of the vessel were made with their longitudinal axes parallel to the principle rolling direction of the vessel plate.

QUAD CITIES

DPR-29

PRESSURE LIMIT AS A FUNCTION OF VESSEL METAL TEMPERATURE

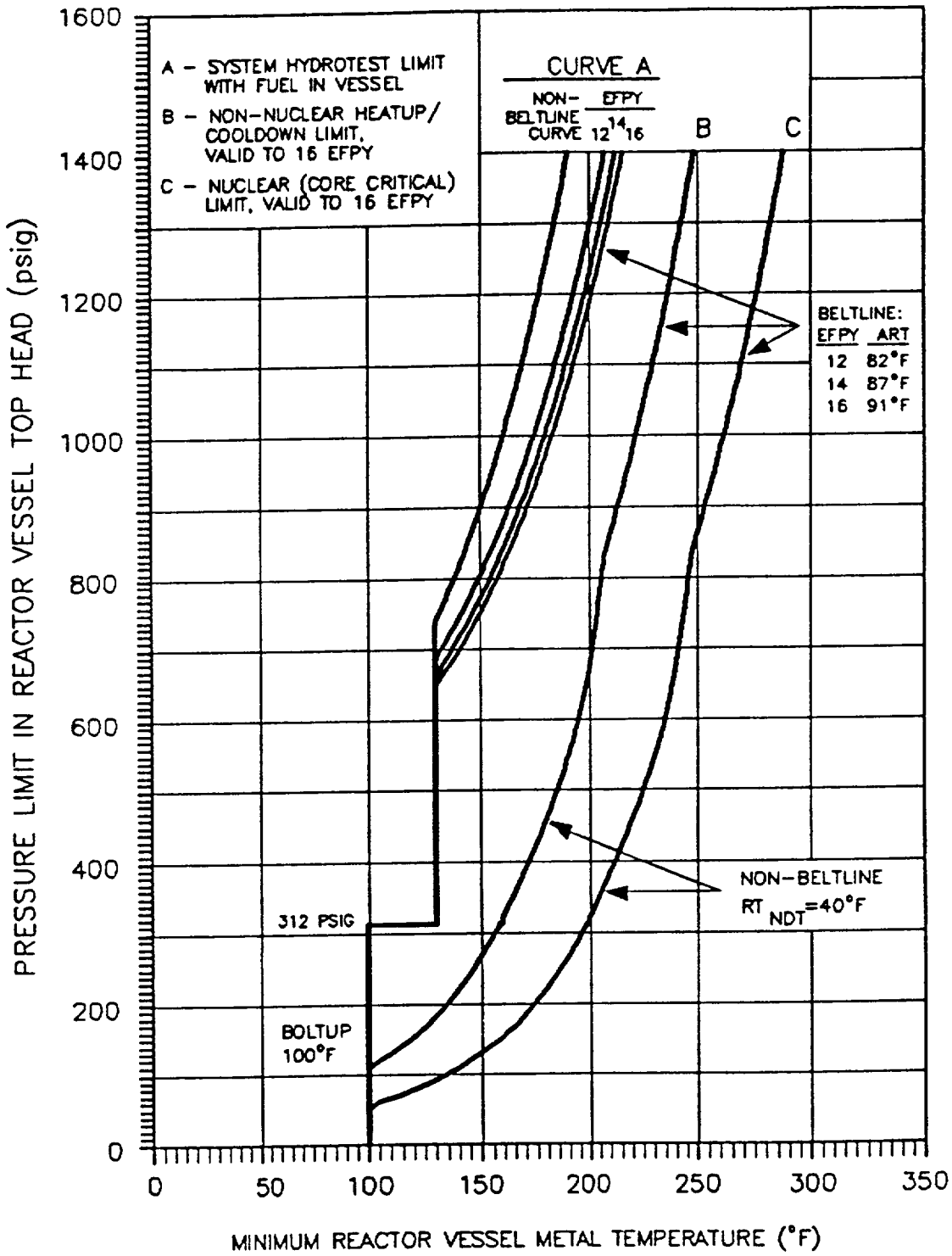

FIGURE 3.6-1

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
1979	4	Near Core Top Guide - 90°	
1982	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. 127
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 28, 1988, June 26, 1989, October 23, 1989, March 23, 1990, and July 26, 1991, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 127 , 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

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 25, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 127

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
i	i
ii	ii
iii	iii
iv	iv
v	v
vi	vi
---	vii
3.6/4.6-1	3.6/4.6-1
3.6/4.6-2	3.6/4.6-2
3.6/4.6-8	3.6/4.6-8
3.6/4.6-9	3.6/4.6-9
3.6/4.6-9a	3.6/4.6-9a
Figure 3.6-1	Figure 3.6-1
3.6/4.6-21A	3.6/4.6-21A

QUAD CITIES
DPR-30

TECHNICAL SPECIFICATIONS

APPENDIX A

TABLE OF CONTENTS

		Page
1.0 DEFINITIONS		1.0-1
1.1/2.1	FUEL CLADDING	1.1/2.1-1
1.1	Safety Limit Bases	1.1/2.1-4
2.1	Limiting Safety System Setting Bases	1.1/2.1-7
1.2/2.2	REACTOR COOLANT SYSTEM	1.2/2.2-1
1.2	Safety Limit Bases	1.2/2.2-2
2.2	Limiting Safety System Setting Bases	1.2/2.2-3
3.0/4.0	LIMITING CONDITIONS FOR OPERATION (GENERAL)	3.0/4.0-1
3.1/4.1	REACTOR PROTECTION SYSTEM	3.1/4.1-1
3.1	Limiting Conditions for Operation Bases	3.1/4.1-2
4.1	Surveillance Requirements Bases	3.1/4.1-5
3.2/4.2	PROTECTIVE INSTRUMENTATION	3.2/4.2-1
	A. Primary Containment Isolation Functions	3.2/4.2-1
	B. Core and Containment Cooling Systems-Initiation and Control	3.2/4.2-1
	C. Control Rod Block Actuation	3.2/4.2-1
	D. Refueling Floor Radiation Monitors	3.2/4.2-2
	E. Postaccident Instrumentation	3.2/4.2-2
	F. Control Room Ventilation System Isolation	3.2/4.2-3
	G. Radioactive Liquid Effluent Instrumentation	3.2/4.2-3
	H. Radioactive Gaseous Effluent Instrumentation	3.2/4.2-4
3.2	Limiting Conditions for Operation Bases	3.2/4.2-5a
4.2	Surveillance Requirements Bases	3.2/4.2-9
3.3/4.3	REACTIVITY CONTROL	3.3/4.3-1
	A. Reactivity Limitiations	3.3/4.3-1
	B. Control Rods	3.3/4.3-2
	C. Scram Insertion Times	3.3/4.3-4
	D. Control Rod Accumulators	3.3/4.3-5
	E. Reactivity Anomalies	3.3/4.3-5
	F. Economic Generation Control System	3.3/4.3-5
3.3	Limiting Conditions for Operation Bases	3.3/4.3-7

TABLE OF CONTENTS (Cont'd)

		Page
3.4/4.4	STANDBY LIQUID CONTROL SYSTEM	3.4/4.4-1
	A. Normal Operation	3.4/4.4-1
	B. Operation with Inoperable Components	3.4/4.4-2
	C. Liquid Poison Tank-Boron Concentration	3.4/4.4-2 |
3.4	Limiting Conditions for Operation Bases	3.4/4.4-3
3.5/4.5	CORE CONTAINMENT COOLING SYSTEMS	3.5/4.5-1
	A. Core Spray Subsystems and the LPCI Mode of the RHR System	3.5/4.5-1
	B. Containment Cooling Mode of the RHR System	3.5/4.5-3
	C. HPCI Subsystem	3.5/4.5-4a |
	D. Automatic Pressure Relief Subsystems	3.5/4.5-5
	E. Reactor Core Isolation Cooling System	3.5/4.5-6
	F. Minimum Core and Containment Cooling System Availability	3.5/4.5-6a |
	G. Maintenance of Filled Discharge Pipe	3.5/4.5-7
	H. Condensate Pump Room Flood Protection	3.5/4.5-8
	I. Average Planar Linear Heat Generation Rate (APLHGR)	3.5/4.5-9
	J. Local LHGR	3.5/4.5-10 |
	K. Minimum Critical Power Ratio (MCPR)	3.5/4.5-10
3.5	Limiting Conditions for Operation Bases	3.5/4.5-11
4.5	Surveillance Requirements Bases	3.5/4.5-15 |
3.6/4.6	PRIMARY SYSTEM BOUNDARY	3.6/4.6-1
	A. Thermal Limitations	3.6/4.6-1
	B. Pressurization Temperature	3.6/4.6-2 |
	C. Coolant Chemistry	3.6/4.6-2
	D. Coolant Leakage	3.6/4.6-3
	E. Safety and Relief Valves	3.6/4.6-4
	F. Structural Integrity	3.6/4.6-4
	G. Jet Pumps	3.6/4.6-5
	H. Recirculation Pump Flow Limitations	3.6/4.6-5a
	I. Shock Suppressors (Snubbers)	3.6/4.6-5c
3.6	Limiting Conditions for Operation Bases	3.6/4.6-8
3.7/4.7	CONTAINMENT SYSTEMS	3.7/4.7-1
	A. Primary Containment	3.7/4.7-1
	B. Standby Gas Treatment System	3.7/4.7-7
	C. Secondary Containment	3.7/4.7-8
	D. Primary Containment Isolation Valves	3.7/4.7-9
3.7	Limiting Conditions for Operation Bases	3.7/4.7-11
4.7	Surveillance Requirements Bases	3.7/4.7-15

TABLE OF CONTENTS (Cont'd)

		Page
3.8/4.8	RADIOACTIVE EFFLUENTS	3.8/4.8-1
	A. Gaseous Effluents	3.8/4.8-1
	B. Liquid Effluents	3.8/4.8-6a
	C. Mechanical Vacuum Pump	3.8/4.8-9
	D. Environmental Monitoring Program	3.8/4.8-10
	E. Solid Radioactive Waste	3.8/4.8-13
	F. Miscellaneous Radioactive Materials Sources	3.8/4.8-14
	H. Control Room Emergency Filtration System	3.8/4.8-14a
3.8/4.8	Limiting Conditions for Operation and Surveillance Req. Bases	3.8/4.8-15
3.9/4.9	AUXILIARY ELECTRICAL SYSTEMS	3.9/4.9-1
	A. Normal and Emergency A-C Auxiliary Power	3.9/4.9-1
	B. Station Batteries	3.9/4.9-2
	C. Electric Power Availability	3.9/4.9-2
	D. Diesel Fuel	3.9/4.9-3
	E. Diesel-Generator Operability	3.9/4.9-3
	F. Reactor Protection Bus Power Monitoring System	3.9/4.9-4
3.9	Limiting Conditions for Operation Bases	3.9/4.9-5
4.9	Surveillance Requirements Bases	3.9/4.9-6
3.10/4.10	REFUELING	3.10/4.10-1
	A. Refueling Interlocks	3.10/4.10-1
	B. Core Monitoring	3.10/4.10-2
	C. Fuel Storage Pool Water Level	3.10/4.10-2
	D. Control Rod and Control Rod Drive Maintenance	3.10/4.10-2
	E. Extended Core Maintenance	3.10/4.10-3
	F. Spent Fuel Cask Handling	3.10/4.10-3
3.10	Limiting Conditions for Operation Bases	3.10/4.10-4
4.10	Surveillance Requirements Bases	3.10/4.10-6
3.11/4.11	HIGH ENERGY PIPING INTEGRITY (Outside Containment)	3.11/4.11-1
3.11/4.11	Bases	3.11/4.11-2
3.12/4.12	FIRE PROTECTION SYSTEMS	3.12/4.12-1
	A. Fire Detection Instrumentation	3.12/4.12-1
	B. Fire Suppression Water System	3.12/4.12-2
	C. Sprinkler Systems	3.12/4.12-3
	D. CO_2 Systems	3.12/4.12-4
	E. Fire Hose Station	3.12/4.12-4
	F. Penetration Fire Barriers	3.12/4.12-4
	G. Fire Pump Diesel Engine	3.12/4.12-5

TABLE OF CONTENTS (Cont'd)

	Page
3.12/4.12 Limiting Conditions for Operations and Surveillance Requirements Bases	3.12/4.12-6
5.0 DESIGN FEATURES	5.0-1
6.0 ADMINISTRATIVE CONTROLS	6.1-1
6.1 Organization, Review, Investigation, and Audit	6.1-1
6.2 Plant Operation Procedures	6.2-1
6.3 Action to be Taken in the Event of a Reportable Occurrence in Plant Operation	6.3-1
6.4 Action to be Taken in the Event a Safety Limit is Exceeded	6.4-1
6.5 Plant Operating Records	6.5-1
6.6 Reporting Requirements	6.6-1
6.7 Environmental Qualification	6.7-1
6.8 Offsite Dose Calculation Manual (ODCM)	6.8-1
6.9 Process Control Program (PCP)	6.9-1
6.10 Major Changes to Radioactive Waste Treatment Systems (Liquid, Gaseous, Solid)	6.10-1

TECHNICAL SPECIFICATIONS

APPENDIX A

LIST OF TABLES

Number	Title	Page
3.1-1	Reactor Protection System (Scram) Instrumentation Requirements-Refuel Mode	3.1/4.1-8
3.1-2	Reactor Protection System (Scram) Instrumentation Requirements-Startup/Hot Standby Mode	3.1/4.1-9
3.1-3	Reactor Protection System (Scram) Instrumentation Requirements-Run Mode	3.1/4.1-10
3.1-4	Notes for Tables 3.1-1, 3.1-2, and 3.1-3	3.1/4.1-11
4.1-1	Scram Instrumentation and Logic Systems Functional Test-Minimal Functional Test Frequencies for Safety Instrumentation, Logic Systems, and Control Circuits	3.1/4.1-12
4.1-2	Scram Instrument Calibration-Minimum Calibration Frequencies for Reactor Protection Instrument Channels	3.1/4.1-14
3.2-1	Instrumentation that Initiates Primary Containment Isolation Functions	3.2/4.2-11
3.2-2	Instrumentation that Initiates or Controls the Core and Containment Cooling Systems	3.2/4.2-12
3.2-3	Instrumentation that Initiates Rod Block	3.2/4.2-14
3.2-4	Postaccident Monitoring Instrumentation Requirements	3.2/4.2-15
3.2-5	Radioactive Liquid Effluent Monitoring Instrumentation	3.2/4.2-15b
3.2-6	Radioactive Gaseous Effluent Monitoring Instrumentation	3.2/4.2-15c
4.2-1	Minimum Test and Calibration Frequency for Core and Containment Cooling Systems Instrumentation, Rod Blocks and Isolation	3.2/4.2-16
4.2-2	Postaccident Monitoring Instrumentaion Surveillance Requirements	3.2/4.2-18
4.2-3	Radioactive Liquid Effluent Monitoring Instrumentation Surveillance Requirements	3.2/4.2-19
4.2-4	Radioactive Gaseous Effluent Monitoring Instrumentation Surveillance Requirements	3.2/4.2-20
4.6-1	Inservice Inspection Requirements for Quad-Cities	3.6/4.6-16
4.6-2	Revised Withdrawl Schedule	3.6/4.6-21A
3.7-1	Primary Containment Isolation	3.7/4.7-20
3.7-2	Primary Containment Leakage Test Penetrations	3.7/4.7-23
4.8-1	Radioactive Gaseous Waste Sampling and Analysis Program	3.8/4.8-20

LIST OF TABLES (Cont.)

Number	Title	Page
4.8-2	Maximum Permissible Concentration of Dissolved or Entrained Noble Gases Released from the Site to Unrestricted Areas in Liquid Waste.	3.8/4.8-21a
4.8-3	Radioactive Areas in Liquid Waste Sampling and Analysis Program	3.8/4.8-22
4.8-4	Radiological Environmental Monitoring Program	3.8/4.8-24
4.8-5	Reporting Levels for Radioactivity Concentrations in Environmental Samples	3.8/4.8-27
4.8-6	Practical Lower Limits of Detection (LLD) for Standard Environmental Monitoring Program	3.8/4.8-28
4.11-1	Surveillance Requirements for High Energy Piping Outside Containment	3.11/4.11-3
3.12-1	Fire Detection Instruments	3.12/4.12-7
3.12-2	Sprinkler Systems	3.12/4.12-8
3.12-3	CO_2 Systems	3.12/4.12-8
3.12-4	Fire Hose Stations	3.12/4.12-9
6.6-1	Special Reports	6.6-5

TECHNICAL SPECIFICATIONS

APPENDIX A

LIST OF FIGURES

Number	Title
2.1-1	APRM Flow Reference Scram and APRM Rod Block Settings
2.1-2	Deleted
2.1-3	APRM Flow Bias Scram Relationship to Normal Operating Conditions
4.1-1	Graphical Aid in the Selection of and Adequate Interval Between Tests
4.2-1	Test Interval vs. System Unavailability
3.4-1	Deleted
3.4-2	Sodium Pentaborate Solution Temperature Requirements
3.5-1	Deleted
3.5-2	Deleted
3.6-1	Pressure Limit as a Function of Vessel Metal Temperature
4.6-1	Chloride Stress Corrosion Test Results at 500°F
4.8-1	Locations of Fixed Environmental Radiological Monitoring Stress
6.1-1	Deleted
6.1-2	Deleted
6.1-3	Minimum Shift Manning Chart

3.6/4.6 PRIMARY SYSTEM BOUNDARY

LIMITING CONDITIONS FOR OPERATION

Applicability:

Applies to the operating status of the reactor coolant system.

Objective:

To assure the integrity and safe operation of the reactor coolant system.

SURVEILLANCE REQUIREMENTS

Applicability:

Applies to the periodic examination and testing requirements for the reactor coolant system.

Objective:

To determine the condition of the reactor coolant system and the operation of the safety devices related to it.

SPECIFICATIONS

A. Thermal Limitations

1. Except as indicated in Specification 3.6.A.2 below, the average rate of reactor coolant temperature change during normal heatup or cooldown shall not exceed 100°F/hr when averaged over a 1-hour period.

2. A step reduction in reactor coolant temperature of 240°F is permissible so long as the limit in Specification 3.6.A.3 below is met.

3. At all times, the shell flange to shell temperature differential shall not exceed 140°F.

4. The recirculation pump in an idle recirculation loop shall not be started unless the coolant in that loop is within 50°F of the operating loop coolant temperature.

A. Thermal Limitations

1. During heatups and cooldowns the following temperatures shall be permanently recorded at 15-minute intervals:

   a. reactor vessel shell,

   b. reactor vessel shell flange, and

   c. recirculation loops A and B.

2. The temperatures listed in Specification 4.6.A.1 shall be permanently recorded subsequent to a heatup or cooldown at 15-minute intervals until three consecutive readings at each given location are within 5 degrees of each other.

B. Pressurization Temperature

1. The reactor vessel shall be vented and power operation shall not be conducted unless the reactor vessel temperature is equal to or greater than that shown in Figure 3.6-1.

   Operation for hydrostatic or leakage tests (Curve A), during heatup or cooldown (Curve B), or with the core critical (Curve C) shall be conducted only when the reactor vessel temperature is equal to or above that shown in the appropriate curve of Figure 3.6-1. Figure 3.6-1 is effective through 16 EFPY. At least six months prior to 16 EFPY new curves will be submitted.

2. The reactor vessel head bolting studs shall not be under tension unless the temperature of the vessel shell immediately below the vessel flange is ≥ 100°F.

C. Coolant Chemistry

1. The steady-state radioiodine concentration in the reactor coolant shall not exceed 5 μCi of I-131 dose equivalent per gram of water.

B. Pressurization Temperature

1. Reactor vessel shell temperature and reactor coolant pressure shall be permanently recorded at 15-minute intervals whenever the shell temperature is below 220°F and the reactor vessel is not vented.

2. Neutron flux monitors and samples shall be installed in the reactor vessel adjacent to the vessel wall at the core midplane level. The monitor and sample program shall conform to ASTM E 185-66. The monitors and samples shall be removed and tested in accordance with the guidelines set forth in 10CFR50 Appendix H to experimentally verify the calculated values of integrated neutron flux that are used to determine the NDTT for Figure 3.6-1.

3. When the reactor vessel head bolting studs are tightened or loosened, the reactor vessel shell temperature immediately below the head flange shall be permanently recorded.

C. Coolant Chemistry

1. a. A sample of reactor coolant shall be taken at least every 96 hours and analyzed for radioactive iodines of I-131 through I-135 during power operation. In addition, when chimney monitors indicate an increase in radioactive gaseous effluents of 25% or 5000 μCi/sec. whichever is greater, during steady-state reactor operation, a reactor coolant sample shall be taken and analyzed for radioactive iodines.

   b. An isotopic analysis of a reactor coolant sample shall be made at least once per month.

   c. Whenever the steady-state radioiodine concentration of prior operation is greater than 1% but less

3.6 LIMITING CONDITIONS FOR OPERATION BASES

A. Thermal Limitations

The reactor vessel design specification requires that the reactor vessel be designed for a maximum heatup and cooldown rate of the contained fluid (water) of 100° F/hour averaged over a period of 1 hour. This rate has been chosen based on past experience with operating power plants. The associated time periods for heatup and cooldown cycles when the 100° F/hr rate is limiting provides for efficient but safe plant operation.

The reactor vessel manufacturer has designed the vessel to the above temperature criterion. In the course of completing the design, the manufacturer performed a detailed stress analysis. This analysis includes more severe thermal conditions than those which would be encountered during normal heating and cooling operations.

Specific analysis were made based on a heating and cooling rate of 100° F/hr applied continuously over a temperature range of 100° F to 550° F. Because of the slow temperature-time response of the massive flanges relative to the adjacent head and shell sections, calculated temperatures obtained were 500° F (shell) and 360° F (flange) (140° F differential).

Both axial and radial thermal stresses were considered to act concurrently with full primary loadings. Calculated stresses were within ASME Boiler and Pressure Vessel Code Section III stress intensity and fatigue limits even at the flange area where maximum stress occurs.

The flange metal temperature differential of 140° F occurred as a result of sluggish temperature response and the fact that the heating rate continued over a 450° F coolant temperature range.

The uncontrolled cooldown rate of 240° F/hr was based on the maximum expected transient over the lifetime of the reactor vessel. This maximum expected transient is the injection of cold water into the vessel by the high-pressure coolant injection subsystem. This transient was considered in the design of the pressure vessel, and ten such cycles were considered in the design. Detailed stress analyses were conducted to assure that the injection of cold water into the vessel by the HPCI would not exceed ASME stress code limitations.

B. Pressurization Temperature

The reactor vessel is a primary barrier against the release of fission products to the environs. In order to provide assurance that this barrier is maintained at a high degree of integrity, pressure-temperature limits have been established for the operating conditions to which the reactor vessel can be subjected. Figure 3.6-1 presents the pressure-temperature curves for those operating conditions; Inservice Hydrostatic Testing (Curve A), Non-Nuclear Heatup/Cooldown (Curve B), and Core Critical Operation (Curve C). These curves have been established to be in conformance with Appendix G to 10 CFR 50 and Regulatory Guide 1.99, Revision 2, and take into account the change in reference nil-ductility transition temperature (RT_{NDT}) as a result of neutron embrittlement. The adjusted reference temperature (ART) of the limiting vessel material is used to account for irradiation effects.

Three vessel regions are considered for the development of the pressure-temperature curves: 1) the core beltline region; 2) the non-beltline region (other than the closure flange region); and 3) the closure flange region. The beltline region is defined as that region of the reactor vessel that directly surrounds the effective height of the reactor core (between the bottom and top of active fuel), and is subject to an RT_{NDT} adjustment to account for irradiation embrittlement. The non-beltline and closure flange regions receive insufficient fluence to necessitate an RT_{NDT} adjustment. These regions contain components which include; the reactor vessel nozzles, closure flanges, top and bottom head plates, control rod drive penetrations, and shell plates that do not directly surround the reactor core. Although the closure flange region is a non-beltline region, it (the closure flange region) is treated separately for the development of the pressure-temperature curves to address 10 CFR 50 Appendix G requirements.

In evaluating the adequacy of the steel which comprises the reactor vessel, it is necessary that the following be established: 1) the RT_{NDT} for all vessel and adjoining materials; 2) the relationship between RT_{NDT} and integrated neutron flux (fluence, at energies greater than one Mev); and 3) the fluence at the location of a postulated flaw.

Boltup Temperature

The initial RT_{NDT} of the main closure flanges, the shell and head materials connecting to these flanges, and connecting welds is 10°F; however, the vertical electroslag welds which terminate immediately below the vessel flange have an RT_{NDT} of 40°F. Therefore, the minimum allowable boltup temperature is established as 100°F (RT_{NDT} + 60°F), which includes a 60°F conservatism required by the original ASME Code of construction.

Curve A - Hydrotesting

As indicated in Curve A of Figure 3.6-1 for system hydrotesting, the minimum metal temperature of the reactor vessel shell is 100°F for reactor pressures less than 312 psig. This 100°F minimum boltup temperature is based on a RT_{NDT} of 40°F for the electroslag weld immediately below the vessel flange and a 60°F conservatism required by the original ASME Code of construction.

At reactor pressures greater than 312 psig the minimum vessel metal temperature is established as 130°F. The 130°F minimum temperature is based on a closure flange region RT_{NDT} of 40°F and a 90°F conservatism required by 10 CFR 50 Appendix G for pressure in excess of 20% of the preservice hydrostatic test pressure (1563 psig).

At approximately 650 psig the effects of pressurization are more limiting than the boltup stresses at the closure flange region, hence a family of non-linear curves intersect the 130°F vertical line. Beltline as well as non-beltline curves have been provided to allow separate monitoring of the two regions. Beltline curves as a function of vessel exposure for 12, 14 and 16 effective full power years (EFPY) are presented to allow the use of the appropriate curve up to 16 EFPY of operation.

Curve B - Non-Nuclear Heatup/Cooldown

Curve B of Figure 3.6-1 applies during heatups with non-nuclear heat (e.g., recirculation pump heat) and during cooldowns when the reactor is not critical (e.g., following a scram). The curve provides the minimum reactor vessel metal temperatures based on the most limiting vessel stress.

As indicated by the vertical 100°F line, the boltup stresses at the closure flange region are most limiting for reactor pressures below approximately 110 psig. For reactor pressures greater than approximately 110 psig, pressurization and thermal stresses become more limiting than the boltup stresses, which is reflected by the non-linear portion of Curve B. The non-linear portion of the curve is dependent on non-beltline and beltline regions, with the beltline region temperature limits having been adjusted to account for vessel irradiation (up to a vessel exposure of 16 EFPY). The non-beltline region is limiting between approximately 110 psig and 830 psig. Above approximately 830 psig, the beltline region becomes limiting.

Curve C - Core Critical Operation

Curve C, the core critical operation curve shown in Figure 3.6-1, is generated in accordance with 10 CFR 50 Appendix G which requires core critical pressure-temperature limits to be 40°F above any Curve A or B limits. Since Curve B is more limiting, Curve C is Curve B plus 40°F.

The withdrawal schedule in Table 4.6-2 is based on the three capsule surveillance program as defined in Section II.C of 10 CFR 50 Appendix H. The accelerated capsule (Near Core Top Guide) are not required by Appendix H.

This surveillance program conforms to ASTM E 185-73 "Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels" with one exception. The base metal specimens of the vessel were made with their longitudinal axes parallel to the principle rolling direction of the vessel plate.

C. Coolant Chemistry

A steady-state radioiodine concentration limit of 5 µCi of I-131 dose equivalent per gram of water in the reactor coolant system can be reached if the gross radioactivity in the gaseous effluents are near the limit as set forth in Specification 3.8.C.1 or there is a failure or prolonged shutdown of the cleanup demineralizer. In the event of a steamline rupture outside the drywell, the NRC staff calculations show the resultant radiological dose at the site boundary to be less than 30 rem to the thyroid. This dose was calculated on the basis of the radioiodine concentration limit of 5 µCi of I-131 dose equivalent per gram of water, atmospheric diffusion from an elevated release at 30 meters under fumigation conditions for Pasquill Type F, 1 meter per second wind speed, and a steamline isolation valve closure time of 5 seconds.

QUAD CITIES

DPR-30

PRESSURE LIMIT AS A FUNCTION OF VESSEL METAL TEMPERATURE

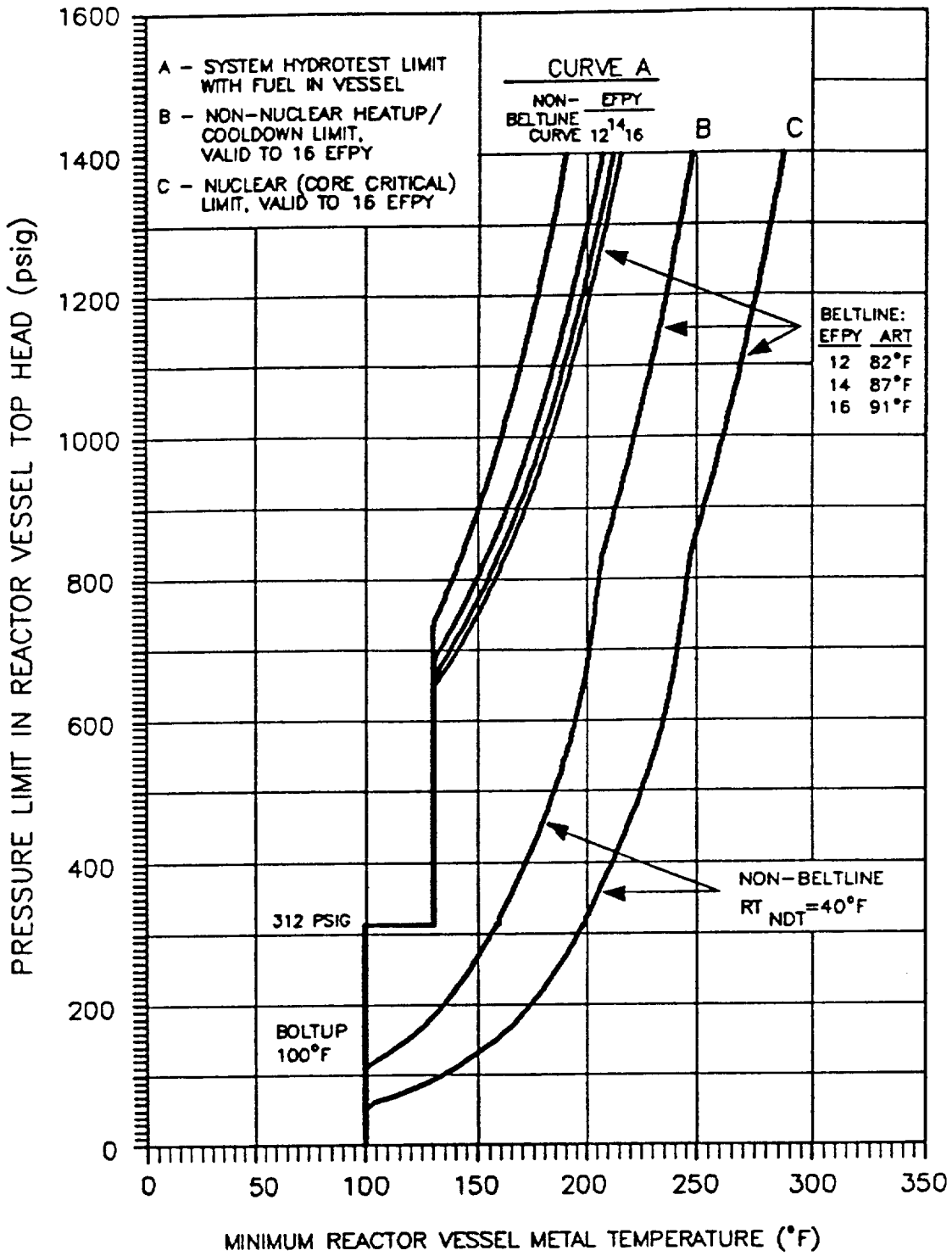

FIGURE 3.6-1

TABLE 4.6-2

REVISED WITHDRAWAL SCHEDULE FOR QUAD-CITIES UNIT 2

Withdrawal Year	Part No.	Location	Comments
1981	18	Wall - 215°	
2002	17	Wall - 95°	
	19	Wall - 245°	Standby
	15	Wall - 65°	Standby
	20	Wall - 275°	Standby
1979	14	Near Core Top Guide - 90°	
1981	16	Near Core Top Guide - 180°	

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 132 TO FACILITY OPERATING LICENSE NO. DPR-29

AND AMENDMENT NO. 127 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 AND 50-265

1.0 INTRODUCTION

In response to Generic Letter (GL) 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations," the Commonwealth Edison Company (the licensee) requested permission to revise the pressure/temperature (P/T) limits in the Quad Cities Station, Units 1 and 2, Technical Specifications, Section 3.6. The licensee's request was dated November 28, 1988, with supplements dated October 23, 1989, March 23, 1990, and July 26, 1991. The request changes the P/T limits from 6 effective full power years (EFPY) to 16 EFPY. GL 88-11 requires that the nil-ductility transition reference temperature in the P/T limits be based on Regulatory Guide (RG) 1.99, Revision 2.

To evaluate the P/T limits, the staff uses the following NRC regulations and guidance: Appendices G and H of 10 CFR Part 50; the American Society for Testing and Materials Standards (ASTM) and the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) which are referenced in Appendices G and H; 10 CFR 50.36(c)(2); RG 1.99, Revision 2; Standard Review Plan (SRP) Section 5.3.2; and GL 88-11.

Each licensee authorized to operate a nuclear power reactor is required by 10 CFR 50.36 to provide Technical Specifications for the operation of the plant. In particular, 10 CFR 50.36(c)(2) requires that limiting conditions for operation be included in the Technical Specifications. The P/T limits are among the limiting conditions for operation in the Technical Specifications for all commercial nuclear plants in the United States. Appendices G and H of 10 CFR Part 50 describe specific requirements for fracture toughness and reactor vessel material surveillance that must be considered in setting P/T limits. An acceptable method for constructing the P/T limits is described in SRP Section 5.3.2.

Appendix G of 10 CFR Part 50 specifies fracture toughness and testing requirements for reactor vessel materials in accordance with the ASME Code and, in particular, that the beltline materials in the surveillance capsules

be tested in accordance with Appendix H of 10 CFR Part 50. Appendix H, in turn, refers to ASTM Standards. These tests define the extent of vessel embrittlement at the time of capsule withdrawal in terms of the increase in reference temperature. Appendix G also requires the licensee to predict the effects of neutron irradiation on vessel embrittlement by calculating the adjusted reference temperature (ART) and Charpy upper shelf energy (USE). GL 88-11 requested that licensees and permittees use the methods in RG 1.99, Revision 2, to predict the effect of neutron irradiation on reactor vessel materials. This guide defines the ART as the sum of unirradiated reference temperature, the increase in reference temperature resulting from neutron irradiation, and a margin to account for uncertainties in the prediction method.

Appendix H of 10 CFR Part 50 requires the licensee to establish a surveillance program to periodically withdraw surveillance capsules from the reactor vessel. Appendix H refers to the ASTM Standards which, in turn, require that the capsules be installed in the vessel before startup and that they contain test specimens made from plate, weld, and heat-affected-zone (HAZ) materials of the reactor beltline.

2.0 EVALUATION

The staff evaluated the effect of neutron irradiation embrittlement on each beltline material in the reactor vessels of Quad Cities 1 and 2. The amount of irradiation embrittlement was calculated in accordance with RG 1.99, Revision 2. The staff has determined that the material with the highest ART at 16 EFPY was the lower intermediate to lower shell longitudinal weld PQ-1300 with 0.3% copper (Cu), 0.33% nickel (Ni), and an initial RT_{ndt} of 40°F.

The licensee has removed four surveillance capsules from Quad Cities 1. The results from capsules 1 and 2 were published in Battelle Columbus Report BMI-0275. The results from capsules 3 and 8 were published in Westinghouse Report WCAP-9920 and Southwest Research Report SWRI-7857, respectively. The surveillance capsules contained Charpy impact specimens and tensile specimens; however, the specimens were not identified well enough to determine with which beltline materials they corresponded. This discrepancy does not comply with ASTM E 185 and 10 CFR 50, Appendix H.

The licensee has removed four surveillance capsules from Quad Cities 2. The results from capsules 12 and 13 were published in Batelle Columbus Report BMI-0975/QC2. The results from capsules G14 and 18 were published in Westinghouse Report WCAP-10064 and Southwest Research Report SWRI-7804-002/1, respectively. All surveillance capsules contained Charpy impact specimens and tensile specimens; however, the specimens also were not identified well enough to determine with which beltline materials they corresponded. This discrepancy also does not comply with ASTM E 185 and 10 CFR 50, Appendix H.

For the limiting beltline material in both Quad Cities 1 and 2, weld PQ-1300, the staff calculated the ART at 16 EFPY to be 90.8°F at 1/4T (T = reactor vessel beltline thickness) and 71.2°F at 3/4T. The staff used a neutron

fluence of 1.7E17 n/cm² at 1/4T and 8E16 n/cm² at 3/4T. The ART was determined by Section 1 of RG 1.99, Revision 2, and the surveillance data were not used because the surveillance capsule specimens were not identified well enough to determine with which beltline materials they corresponded.

The licensee used the method in RG 1.99, Revision 2, to calculate an ART at 16 EFPY at 1/4T of 90.5°F for the same limiting weld (PQ-1300). The staff judges that a difference of 0.3°F between the licensee's ART of 90.5°F and the staff's ART of 90.8°F is acceptable. Substituting the ART of 90.8°F into equations in SRP 5.3.2, the staff verified that the proposed P/T limits for heatup, cooldown, and hydrotest meet the beltline material requirements in Appendix G of 10 CFR Part 50.

In addition to beltline materials, Appendix G of 10 CFR Part 50 also imposes P/T limits based on the reference temperature for the reactor vessel closure flange materials. Section IV.2 of Appendix G states that when the pressure exceeds 20% of the preservice system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120°F for normal operation and by 90°F for hydrostatic pressure tests and leak tests. Paragraph IV.A.3 of Appendix G states, "An exception may be made for boiling water reactor vessels when water level is within the normal range for power operation and the pressure is less than 20 percent of the preservice system hydrostatic test pressure." In this case the minimum permissible temperature is 60°F (33°C) above the reference temperature of the closure flange regions that are highly stressed by the bolt preload." Based on the flange reference temperature of 10°F, the staff has determined that the proposed P/T limits satisfy Section IV.2 of Appendix G for both units.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Illinois State official was notified of the proposed issuance of the amendments. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendments change 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 NRC 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 (55 FR 10530). 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

The staff concludes that the proposed P/T limits for Units 1 and 2 for heatup, cooldown, leak test, and criticality operation of the reactor coolant system are valid through 16 EFPY because the limits conform to the requirements of Appendix G of 10 CFR Part 50. The limits also satisfy Generic Letter 88-11 because the licensee used the method in RG 1.99, Revision 2 to calculate the ART. Hence, the proposed P/T limits may be incorporated into the Quad Cities 1 and 2 Technical Specifications.

The Commission 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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