Distribution
Docket
ORB #3
Local PDR
NRC PDR
VStello
BGrimes
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SSheppard
RBevan
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BJones (16)
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JMcGough
DEisenhut
ACRS (16)
OPA (CMiles)
DRoss
RDiggs
TBAbernathy
JRBuchanan
HSmith
PO'Connor

Docket Nos. 50-237
50-249
50-254
and 50-265

NOVEMBER 1 3 1978

Commonwealth Edison Company
ATTN: Mr. Cordell Reed
Assistant Vice President
P. O. Box 767
Chicago, Illinois 60690

Gentlemen:

The Commission has issued the enclosed Amendment Nos. 39, 37, 48 and 47 for the Dresden Station Units Nos. 2 and 3 and Quad Cities Station Units Nos. 1 and 2, respectively, in response to your requests of September 10, 1974 and May 17, 1976, as supplemented March 21, 1977 and March 13, 1978.

The amendments revise Technical Specifications to provide operating temperature and pressure limits in accordance with Appendix G, 10 CFR Part 50.

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

Sincerely,

Original signed by

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

7811300058

Enclosures:
1. Amendment Nos. 39, 37, 48 and 47 to DPR-19, DPR-25, DPR-29 and DPR-30
2. Safety Evaluation
3. Notice

Constant
CCP

cc w/enclosures: See page 2

OFFICE➤		DOR	DOR	OELD	ORB #3
SURNAME➤		Sheppard/Smith	Bevan/O'Connor		TIppolito
DATE➤		/ /78	/ /78 mjf	/ /78	/ /78

NRC FORM 318 (9-76) NRCM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1976 — 626-624

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Natural Resources Defense Council
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Washington, D. C. 20005

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Federal Activities Branch
Region V Office
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Chicago, Illinois 60604

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-237

DRESDEN STATION UNIT NO. 2

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 39
License No. DPR-19

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

   A. The applications for amendment by the Commonwealth Edison Company (the licensee) dated September 10, 1974 and May 17, 1976, as supplemented March 21, 1977 and March 13, 1978, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Provisional Operating License No. DPR-19 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 39, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

7811300064

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

TAIppolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 13, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 39

PROVISIONAL OPERATING LICENSE NO. DPR-19

DOCKET NO. 50-237

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

Remove	Insert
88	88
90	90
---	91f
92	92
93	93

3.6 LIMITING CONDITION FOR OPERATION

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 Curve C of Figure 3.6.1. Operation for hydrostatic or leakage tests, during heatup or cooldown, and with the core critical shall be conducted only when vessel temperature is equal to or above that shown in the appropriate curve of Fig. 3.6.1. Figure 3.6.1 is effective through 6 effective full power years. At least six months prior to 6 effective full power years 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 $\geq$100°F.

C. Coolant Chemistry

1. The reactor coolant system radioactivity concentration in water shall not exceed 20 microcuries of total iodine per ml of water

4.6 SURVEILLANCE REQUIREMENT

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. 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.

3. 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 as a minimum conform to ASTM E 185. The monitors and samples shall be removed and tested during the third refueling outage to experimentally verify the calculated values of integrated neutron flux that are used to determine the NDTT for Figure 4.6.1.

C. Coolant Chemistry

1. a. A sample of reactor coolant shall be taken at least every 96 hours and analyzed for radio-activity.

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

an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown condition within 24 hours.

2. The primary containment sump sampling system and an air sampling system shall be operable during power operation. If either a sump water sample or a containment air sample cannot be obtained for any reason, reactor operation is permissible only during the succeeding seven days unless the system is made operable during this period.

E. Safety and Relief Valves

1. During reactor power operating conditions and whenever the reactor coolant pressure is greater than 90 psig and temperature greater than 320°F, all eight of the safety valves shall be operable. The solenoid activated pressure valves shall be operable as required by Specification 3.5.D.

2. If Specification 3.6.E.1 is not met, an orderly shutdown shall be initiated and the reactor coolant pressure and temperature shall be $\geq$90 psig and $\geq$320°F within 24 hours.

2. The primary containment sump sampling and air sampling system operability will be observed daily as part of 4.6.D.2.

E. Safety and Relief Valves

A minimum of 1/2 of all safety valves shall be bench checked or replaced with a bench checked valve each refueling outages. The popping point of the safety valves shall be set as follows:

Number of Valves	Set Point (Psig
1	1125*
2	1240
2	1250
2	[illegible]
2	1250

The allowable set point error for each valve is ±1%

All relief valves shall be checked for set pressure each refueling outage. The set pressures shall be:

Number of Valves	Set Point (Psig
1	1125*
2	≤ 1130
2	≤ 1135

*Target Rock combination safety/relief valve

Minimum Temperature Requirements per Appendix G of 10 CFR 50

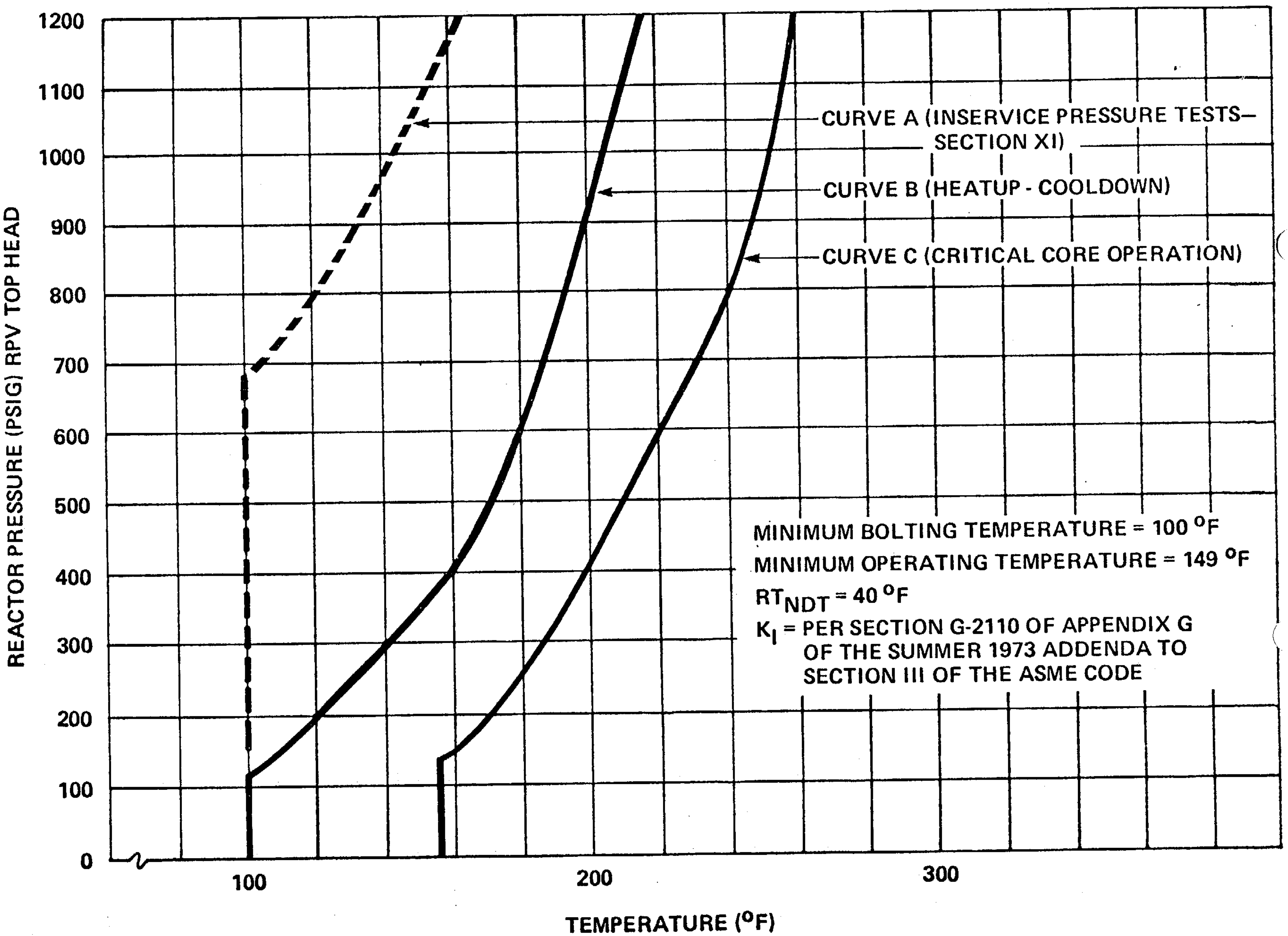

Fig. 3.6.1

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 per hour averaged over a period of one 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 per hour 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 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/hour 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 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 five 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. Specification 3.6.A.4 increases margin of safety for thermal-hydraulic stability and startup of recirculation pump.

Pressurization Temperature - The reactor coolant system 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, restrictions have been placed on the operating conditions to which it can be subjected. These restrictions on inservice hydrostatic testing, on heatup and cooldown, and on critical core operation shown in Figure 3.6.1, were established to be in conformance with Appendix G to 10 CFR 50.

In evaluating the adequacy of ferritic steels Sa302B it is necessary that the following be established:

a) The reference nil-ductility temperature (RT_{NDT}) for all vessel and adjoining materials,

b) the relationship between RT_{NDT} and and integrated neutron flux (fluence, at energies > 1 Mev), and
c) the fluence at the location of a postulated flow.

The initial RT_{NDT} of the main closure flange, 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. Reference Appendix F to the FSAR. The closure flanges and connecting shell materials are not subject to any appreciable neutron radiation exposure, nor are the vertical electroslag seams. The flange area is moderately stressed by tensioning the head bolts. Therefore, as is indicated in curves (a) and (b) of Figure 3.6.1, the minimum temperature of the vessel shell immediately below the vessel flange is established as 100°F below a pressure of 400 psig. (40°F + 60°F, where 40°F is the RT_{NDT} of the electroslag weld and 60°F is a conservatism required by the ASME Code). Above approximately 400 psig pressure, the stresses associated with pressurization are more limiting than the bolting stresses, a fact that is reflected in the non-linear portion of curves (a) and (b). Curve (c), which defines the temperature limitations for critical core operation, was established per Section IV 2.c. of Appendix G of 10CFR50. Each of the curves, (a), (b) and (c) define temperature limitations for unirradicated ferrectic steels. Provision has been made for the modification of these curves to account for the change in RT_{NDT} as a result of neutron embrittlement.

C. <u>Coolant Chemistry</u> - A radioactivity concentration limit of 20 μCi/ml total iodine can be reached if the gaseous effluents are near the limit as set forth in Specification 3.8.C.1 or there is a failure or a prolonged shutdown of the cleanup demineralizer. In the event of a steam line rupture, outside the drywell, the resultant radiological dose at the site boundary would be about 10 rem to the thyroid. This dose was calculated on the basis of a total iodine activity limit of 20 μCi/ml, meteorology corresponding

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-249

DRESDEN STATION UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 37
License No. DPR-25

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

   A. The applications for amendment by the Commonwealth Edison Company (the licensee) dated September 10, 1974 and May 17, 1976, as supplemented March 21, 1977 and March 13, 1978, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Provisional Operating License No. DPR-25 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 37 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

7811300067

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

T.A. Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 13, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 37

FACILITY OPERATING LICENSE NO. DPR-25

DOCKET NO. 50-249

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

Remove	Insert
88	88
90	90
---	91f
92	92
93	93

3.6 LIMITING CONDITION FOR OPERATION	4.6 SURVEILLANCE REQUIREMENT

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 Curve C of Figure 3.6.1. Operation for hydrostatic or leakage tests, during heatup or cooldown, and with the core critical shall be conducted only when vessel temperature is equal to or above that shown in the appropriate curve of Fig. 3.6.1. Figure 3.6.1 is effective through 6 effective full power years. At least six months prior to 6 effective full power years 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 reactor coolant system radioactivity concentration in water shall not exceed 20 microcuries of total iodine per ml 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. 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.

3. 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 as a minimum conform to ASTM E 185. The monitors and samples shall be removed and tested during the third refueling outage to experimentally verify the calculated values of integrated neutron flux that are used to determine the NDTT for Figure 4.6.1.

C. Coolant Chemistry

1. a. A sample of reactor coolant shall be taken at least every 96 hours and analyzed for radio-activity.

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

3.6 LIMITING CONDITION FOR OPERATION

an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown condition within 24 hours.

2. The primary containment sump sampling system and an air sampling system shall be operable during power operation. If either a sump water sample or a containment air sample cannot be obtained for any reason, reactor operation is permissible only during the succeeding seven days unless the system is made operable during this period.

E. Safety and Relief Valves

1. During reactor power operating conditions and whenever the reactor coolant pressure is greater than 90 psig and temperature greater than $320^{\circ}F$, all eight of the safety valves shall be operable. The solenoid activated pressure valves shall be operable as required by Specification 3.5.D.

2. If Specification 3.6.E.1 is not met, an orderly shutdown shall be initiated and the reactor coolant pressure and temperature shall be ≥ 90 psig and $\geq 320°F$ within 24 hours.

4.6 SURVEILLANCE REQUIREMENT

2. The primary containment sump sampling and air sampling system operability will be observed daily as part of 4.6.D.2.

E. Safety and Relief Valves

A minimum of 1/2 of all safety valves shall be bench checked or replaced with a bench checked valve each refueling outages. The popping point of the safety valves shall be set as follows:

Number of Valves	Set Point (Psig)
1	1125*
2	1240
2	1250
2	[illegible]
2	1260

The allowable set point error for each valve is ±1%

All relief valves shall be checked for set pressure each refueling outage. The set pressures shall be:

Number of Valves	Set Point (Psig)
1	1125*
2	≤ 1130
2	≤ 1135

*Target Rock combination safety/relief valve

Minimum Temperature Requirements per Appendix G of 10 CFR 50

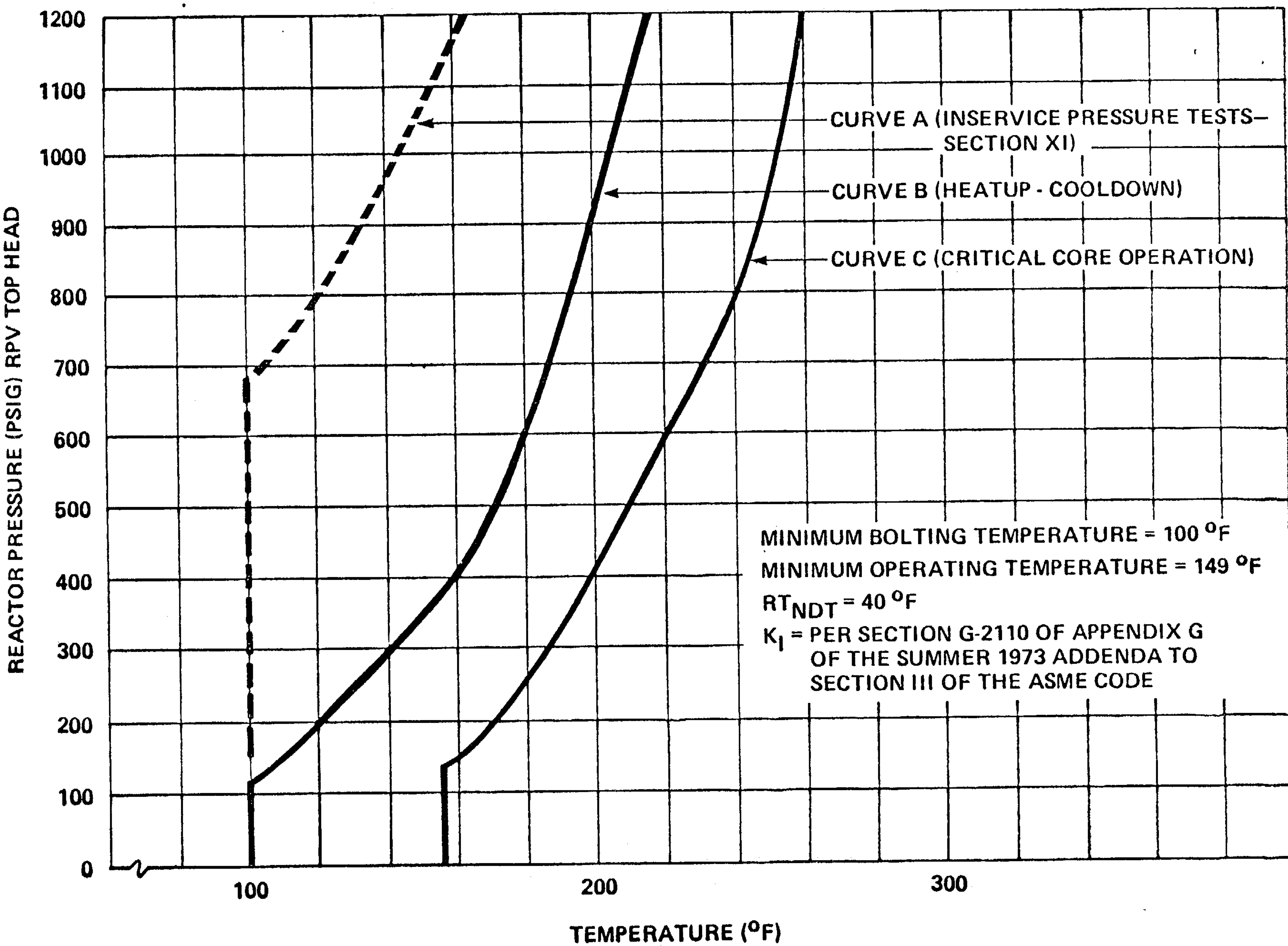

Fig. 3.6.1

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 per hour averaged over a period of one 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 per hour 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 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/hour 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 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 five 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. Specification 3.6.A.4 increases margin of safety for thermal-hydraulic stability and startup of recirculation pump.

Pressurization Temperature - The reactor coolant system 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, restrictions have been placed on the operating conditions to which it can be subjected. These restrictions on inservice hydrostatic testing, on heatup and cooldown, and on critical core operation shown in Figure 3.6.1, were established to be in conformance with Appendix G to 10 CFR 50.

In evaluating the adequacy of ferritic steels Sa302B it is necessary that the following be established:

a) The reference nil-ductility temperature (RT_{NDT}) for all vessel and adjoining materials,

b) the relationship between RT_{NDT} and and integrated neutron flux (fluence, at energies > 1 Mev), and

c) the fluence at the location of a postulated flow.

The initial RT_{NDT} of the main closure flange, 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. Reference Appendix F to the FSAR. The closure flanges and connecting shell materials are not subject to any appreciable neutron radiation exposure, nor are the vertical electroslag seams. The flange area is moderately stressed by tensioning the head bolts. Therefore, as is indicated in curves (a) and (b) of Figure 3.6.1, the minimum temperature of the vessel shell immediately below the vessel flange is established as 100°F below a pressure of 400 psig. (40°F + 60°F, where 40°F is the RT_{NDT} of the electroslag weld and 60°F is a conservatism required by the ASME Code). Above approximately 400 psig pressure, the stresses associated with pressurization are more limiting than the bolting stresses, a fact that is reflected in the non-linear portion of curves (a) and (b). Curve (c), which defines the temperature limitations for critical core operation, was established per Section IV 2.c. of Appendix G of 10CFR50. Each of the curves, (a), (b) and (c) define temperature limitations for unirradicated ferrectic steels. Provision has been made for the modification of these curves to account for the change in RT_{NDT} as a result of neutron embrittlement.

C. <u>Coolant Chemistry</u> - A radioactivity concentration limit of 20 μCi/ml total iodine can be reached if the gaseous effluents are near the limit as set forth in Specification 3.8.C.1 or there is a failure or a prolonged shutdown of the cleanup demineralizer. In the event of a steam line rupture, outside the drywell, the resultant radiological dose at the site boundary would be about 10 rem to the thyroid. This dose was calculated on the basis of a total iodine activity limit of 20 μCi/ml, meteorology corresponding

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

COMMONWEALTH EDISON COMPANY
AND
IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

DOCKET NO. 50-254

QUAD CITIES STATION UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 48
License No. DPR-29

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

   A. The applications for amendment by Commonwealth Edison Company (the licensee) dated September 10, 1974 and May 17, 1976, as supplemented March 21, 1977 and March 13, 1978, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility 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. 48, 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

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 13, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 48

FACILITY OPERATING LICENSE NO. DPR-29

DOCKET NO. 50-254

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by the captioned Amendment number and contain vertical lines indicating the area of change.

Remove	Insert
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

Add pages 3.6/4.6-2a and 3.6/4.6-15c

that shown in Figure 3.6-1. Operation for hydrostatic or leakage tests, during heatup or cooldown, and with the core critical shall be conducted only when vessel temperature is equal to or above that shown in the appropriate curve of Fig. 3.6.1. Figure 3.6.1 is effective through 6 EFPY. At least six months prior to 6 EFPY new curves will be submitted.

2. The reactor vessel heat bolting studs shall nor be under tension unless the temperature of the vessel shell immediately below the vessel flange is $\geq 100^{\circ}F$.

C. Coolant Chemistry

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

below $220^{\circ}F$ 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 as a minimum conform to ASTM E 185-66. The monitors and samples shall be removed and tested during the third refueling outage 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 uCi/sec. whichever is greater, during steady-state reactor operation, a reactor coolant sample 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 than 10% of Specification 3.6.C.1, a sample of reactor coolant shall be taken within 24 hours of any reactor startup and analyzed for radioactive iodines of I-131 through I-135.

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 coolant system 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, restrictions have been placed on the operating conditions to which it can be subjected. These restrictions on inservice hydrostatic testing, on heatup and cooldown, and on critical core operation, shown in Figure 3.6.1, were established to be in conformance with Appendix G to 10CFR50.

In evaluating the adequacy of ferritic steels Sa302B it is necessary that the following be established:

1. The reference nil-ductility temperature (RT_{NDT}) for all vessel and adjoining materials,

2. The relationship between RT_{NDT} and integrated neutron flux (fluence, at energies > 1 Mev), and

3. The fluence at the location of a postulated flaw.

The initial RT_{NDT} of the main closure flange, 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. Reference Appendix F to the Dresden FSAR. The closure flanges and connecting shell materials are not subject to any appreciable neutron radiation exposure, nor are the vertical electroslag seams. The flange area is moderately stressed by tensioning the head bolts. Therefore, as is indicated in curves (a) and (b) of Figure 3.6.1, the minimum temperature of the vessel shell immediately below the vessel flange is established as 100°F below a pressure of 400 psig. (40°F + 60°F, where 40°F is the RT_{NDT} of the electroslag weld and 60°F is a conservatism required by the ASME Code). Above approximately 400 psig pressure, the stresses associated with pressurization are more limiting than the bolting stresses, a fact that is reflected in the non-linear portion of curves (a) and (b). Curve (c), which defines the temperature limitations for critical core operation, was established per Section IV 2.c. of Appendix G of 10CFR50. Each of the curves, (a), (b) and (c) define temperature limitations for unirradiated ferritic steels. Provision has been made for the modification of these curves to account for the change in RT_{NDT} as a result of neutron embrittlement.

C. Coolant Chemistry

A steady-state radioiodine concentration limit of 5 uCi 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 of 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 uCi 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.

Minimum Temperature Requirements per Appendix G of 10 CFR 50

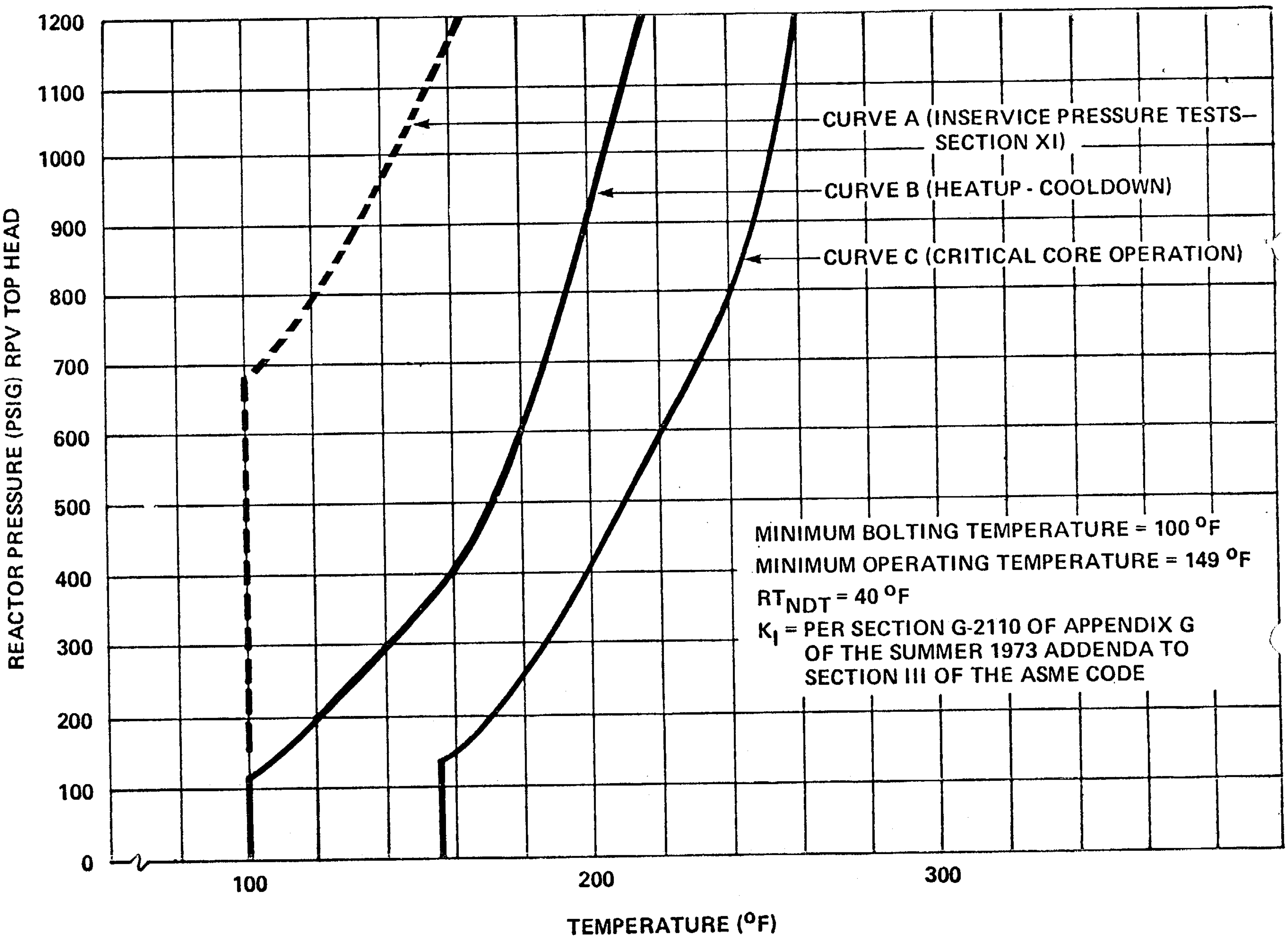

Fig. 3.6.1

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 STATION UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENCE

Amendment No. 47
License No. DPR-30

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

   A. The applications for amendment by Commonwealth Edison Company (the licensee) dated September 10, 1974 and May 17, 1976, as supplemented March 21, 1977 and March 13, 1978, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

   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. 47 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

T. A. Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 13, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 47

FACILITY OPERATING LICENSE NO. DPR-30

DOCKET NO. 50-265

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by the captioned Amendment number and contain vertical lines indicating the area of change.

Remove	Insert
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

Add pages 3.6/4.6-2a and 3.6/4.6-15c

that shown in Figure 3.6-1. Operation for hydrostatic or leakage tests, during heatup or cooldown, and with the core critical shall be conducted only when vessel temperature is equal to or above that shown in the appropriate curve of Fig. 3.6.1. Figure 3.6.1 is effective through 6 EFPY. At least six months prior to 6 EFPY new curves will be submitted.

2. The reactor vessel heat bolting studs shall nor be under tension unless the temperature of the vessel shell immediately below the vessel flange is $\geq 100^{\circ}F$.

C. Coolant Chemistry

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

below $220^{\circ}F$ 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 as a minimum conform to ASTM E 185-66. The monitors and samples shall be removed and tested during the third refueling outage 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 uCi/sec. whichever is greater, during steady-state reactor operation, a reactor coolant sample 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 than 10% of Specification 3.6.C.1, a sample of reactor coolant shall be taken within 24 hours of any reactor startup and analyzed for radioactive iodines of I-131 through I-135.

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 coolant system 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, restrictions have been placed on the operating conditions to which it can be subjected. These restrictions on inservice hydrostatic testing, on heatup and cooldown, and on critical core operation, shown in Figure 3.6.1, were established to be in conformance with Appendix G to 10CFR50.

In evaluating the adequacy of ferritic steels Sa302B it is necessary that the following be established:

1. The reference nil-ductility temperature (RT_{NDT}) for all vessel and adjoining materials,

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A steady-state radioiodine concentration limit of 5 uCi 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 of 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 uCi 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.

Minimum Temperature Requirements per Appendix G of 10 CFR 50

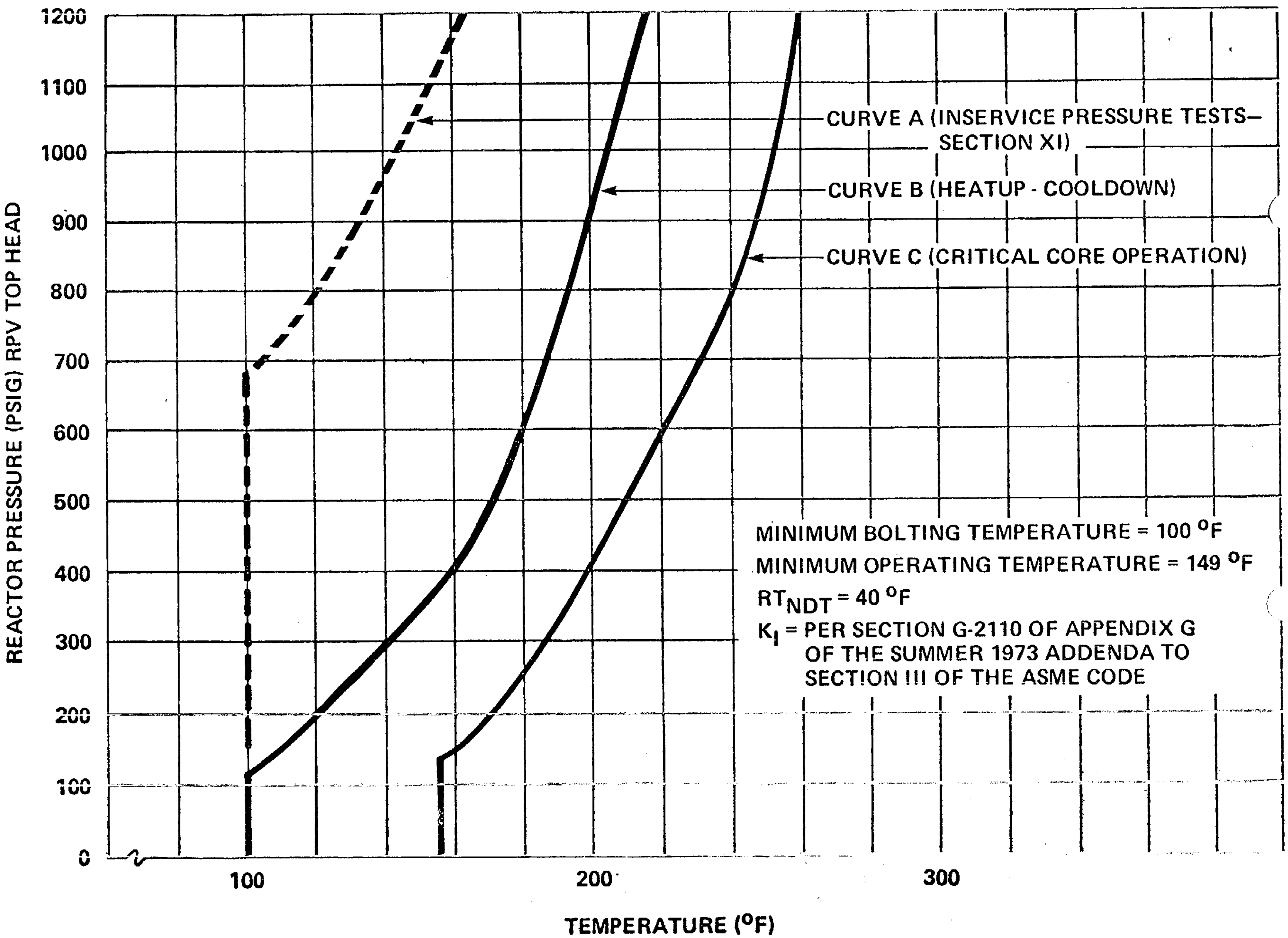

Fig. 3.6.1

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

AMENDMENT NOS. 39, 37, 48 AND 47

LICENSE NOS. DPR-19, DPR-25, DPR-29, AND

DRESDEN AND QUAD CITIES NUCLEAR POWER STATIONS

COMMONWEALTH EDISON COMPANY

DOCKET NOS. 50-237, 50-249, 50-254 AND 50-265

Introduction

By letters dated September 10, 1974 and May 17, 1976, as supplemented by letters dated March 21, 1977 and March 13, 1978, Commonwealth Edison (CE) proposed changes to the Technical Specifications appended to Operating Licenses DPR-19 and DPR-25 for Dresden Units 2 and 3, and Technical Specifications appended to Operating Licenses DPR-29 and DPR-30 for Quad Cities Units 1 and 2. The changes would modify the reactor coolant system thermal and pressurization limitations to account for irradiation induced increases in reactor vessel metal nil ductility temperature (RT_{NDT}). The CE submittals were based on the determination that certain changes were necessary to bring the reactor coolant system pressure-temperature limits into conformity with the requirements of Appendix G to 10 CFR Part 50.

Discussion

Title 10 CFR Part 50, Appendix G "Fracture Toughness Requirements", requires that pressure-temperature limits be established for reactor coolant system heatup and cooldown operations, inservice leak and hydrostatic tests, and reactor core operation. These limits are required to ensure that the stresses in the reactor vessel remain within acceptable limits. They are intended to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences.

The specific pressure-temperature limits which are initially established depend upon the metallurgical properties of the reactor vessel material and the design service conditions. However, the metallurgical properties vary over the lifetime of the reactor vessel because of the effects of neutron irradiation. One principal effect of the neutron irradiation is that it causes the reactor vessel nil ductility temperature (RT_{NDT}) to increase or shift with time. The practical results of the RT_{NDT} shift

[1]RT_{NDT} is the temperature associated with the transition from ductile to brittle fracture mode of failure.

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is that for any given value of reactor pressure, the reactor vessel metal temperature must be maintained at higher values during the heatup and cooldown process. By periodically revising the pressure-temperature limits to account for neutron irradiation induced increases in RT_{NDT}, the stresses in the reactor vessel are maintained within acceptable limits.

Evaluation

The CE submittals dated September 10, 1974 and May 17, 1976 included, for the Dresden Units and Quad Cities Units, respectively, pressure-temperature limits for hydrostatic testing, mechanical heatup and cooldown and minimum temperature for Core Operation (criticality). During our review of these submittals we determined that the radiation damage estimate curves, i.e., the effect of neutron fluence on RT_{NDT}, did not appear to be as conservative as that presented in Regulatory Guide 1.99.

In a letter dated February 15, 1978[1], CE stated that their radiation damage estimates were based on a "worst case" basis. Subsequent staff calculations made on "worst case" conditions[2] determined the end of life neutron fluence at the one-quarter thickness (1/4T) location to be 9×10^{17} n/cm^2.[3]

Branch Technical Position MTEB 5-2 "Fracture Toughness Requirements" requires that calculations be performed in regions of high stress unless the assumed RT_{NDT} of the beltline region is at least 50°F above the RT_{NDT} of all higher stressed regions. To satisfy this requirment, CE obtained stress intensities in regions of discontinuities by adjusting the results of a generic analysis made to account for differences between the design and materials of the Dresden and Quad Cities vessels and those of the reference plant. We have reviewed the licensee's submittal and determined that this is an acceptable procedure for calculating pressure-temperature operating limits. The operating limits for hydrostatic testing, mechanical heatup or cooldown, and minimum temperature for core operation (criticality) were calculated by CE and submitted on March 13, 1978.[4]

Based on the use of the previously discussed "worst case" damage estimates, an end of life neutron fluence of 9×10^{17} n/cm^2 at the 1/4T location, and the limiting curves submitted on March 13, 1978, we conclude that these limits and damage estimates are acceptable for operation through approximately 6 effective full power years. Accordingly, the staff added a Specification to the temperature and pressure limits to require that the figures for hydrostatic testing, mechanical heatup and cooldown, and minimum temperature for core operation (criticality) will be updated to account for radiation damage at least 6 months prior to 6 effective full power years. This additional requirement was discussed with and agreed to by the licensee.

We conclude that the pressure-temperature operating limits as amended by the staff are acceptable through 6 EFPY. For this operating period the proposed pressure-temperature operating limits are in accordance with Appendix G, 10 CFR Part 50. Compliance with Appendix G in establishing safe operating limitations will ensure adequate safety margins during operation, testing, maintenance and postulated accident conditions and constitute an acceptable basis for satisfying the requirements of NRC General Design Criterion 31, Appendix A, 10 CFR Part 50.

Environmental Consideration

We have determined that the amendments do not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendments involve an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of these amendments.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendments do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the amendments do not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Dated: November 13, 1978

References

1. Letter to Don Davis (NRC/ORB #2) from M. S. Turbak (CE); dtd February 15, 1978.

2. i.e., the upper limit line in Regulatory Guide 1.99, Revision 1.

3. Adjusted data from Table 2-2 of NEDO-21708 per instructions in table note. Results from this procedure yield more oonservative results than those supplied by CE.

4. Submitted as Figures 4, 5, and 6 in attachment to letter from M. S. Turback (CE) to George Lear (NRC/ORB #3); dtd March 13, 1978.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-237, 50-249, 50-254 AND 50-265

COMMONWEALTH EDISON COMPANY
AND
IOWA ILLINOIS GAS AND ELECTRIC COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued an amendment each to Facility Operating License Nos. DPR-19, DPR-25, DPR-29 and DPR-30, issued to Commonwealth Edison Company (and, in the matter of License Nos. DPR-29 and DPR-30, the Iowa-Illinois Gas and Electric Company), which revised Technical Specifications for operation of each of the Dresden and Quad Cities Nuclear Power Stations (collectively referred to as the facilities). The Dresden Station consists of Unit Nos. 1, 2, and 3 and is located in Grundy County, Illinois. However, the actions noticed herein relate to Dresden Station Units 2 and 3. The Quad Cities Station consists of Unit Nos. 1 and 2 and is located in Rock Island County, Illinois. These amendments are effective as of their dates of issuance.

The amendments revise Technical Specifications to provide operating temperature and pressure limits in accordance with Appendix G, 10 CFR Part 50.

The applications for the amendments comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments. Prior public notice of these amendments was not required since the amendments do not involve a significant hazards consideration.

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

For further details with respect to this action, see (1) the applications for amendments dated September 10, 1974 and May 17, 1976, as supplemented March 21, 1977 and March 13, 1978, (2) Amendment Nos. 39 and 37 to License Nos. DPR-19, and DPR-25, (3) Amendment Nos. 48 and 47 to License Nos. DPR-29 and DPR-30, and (4) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and for those items relating to Dresden Unit Nos. 2 and 3 at the Morris Public Library, 604 Liberty Street, Morris, Illinois 60450 and for those items relating to Quad Cities Unit Nos. 1 and 2 at the Moline Public Library, 504 -17th Street, Moline, Illinois 60625. A single copy of items (2), (3), and (4) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland this 13th day of November 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

TA Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors