



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

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-454

BYRON STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 44  
License No. NPF-37

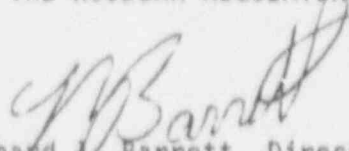
1. The Nuclear Regulatory Commission (the Commission) has found that:
  - A. The application for amendment by Commonwealth Edison Company (the licensee) dated June 28, 1991, as supplemented August 27, 1991, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the 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 2.C.(2) of Facility Operating License No. NPF-37 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 44 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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: November 18, 1991



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

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-455

BYRON STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 44  
License No. NPF-66

1. The Nuclear Regulatory Commission (the Commission) has found that:
  - A. The application for amendment by Commonwealth Edison Company (the licensee) dated June 28, 1991, as supplemented August 27, 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 1;
  - 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
  - F. 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 2.C.(2) of Facility Operating License No. NPF-66 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A (NUREG-1113), as revised through Amendment No. 44 and revised by Attachment 2 to NPF-66, and the Environmental Protection Plan contained in Appendix B, both of which were attached to License No. NPF-37, dated February 14, 1985, are hereby incorporated into this license. Attachment 2 contains a revision to Appendix A which is hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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: November 18, 1991

ATTACHMENT TO LICENSE AMENDMENT NOS. 44 AND 44  
FACILITY OPERATING LICENSE NOS. NPF-37 AND NPF-66  
DOCKET NOS. STN 50-454 AND STN 50-455

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. Pages indicated with an asterisk are provided for convenience.

<u>Remove Pages</u>	<u>Insert Pages</u>
* 3/4 4-11	* 3/4 4-11
3/4 4-12	3/4 4-12
3/4 4-39	3/4 4-39
3/4 4-41	3/4 4-41
* 3/4 4-42	* 3/4 4-42
*B 3/4 4-1	*B 3/4 4-1
B 3/4 4-2	B 3/4 4-2
	B 3/4 4-2a
B 3/4 4-16	B 3/4 4-16

## REACTOR COOLANT SYSTEM

### 3/4.4.3 PRESSURIZER

#### LIMITING CONDITION FOR OPERATION

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3.4.3 The pressurizer shall be OPERABLE with at least two groups of pressurizer heaters each having a capacity of at least 150 kW and a water level of less than or equal to 92%.

APPLICABILITY: MODES 1, 2, and 3.

#### ACTION:

- a. With less than two groups of pressurizer heaters OPERABLE, restore at least two groups of pressurizer heaters to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.
- b. With the pressurizer otherwise inoperable, be in at least HOT STANDBY with the Reactor trip breakers open within 6 hours and in HOT SHUTDOWN within the following 6 hours.

#### SURVEILLANCE REQUIREMENTS

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4.4.3.1 The pressurizer water level shall be determined to be within its limit at least once per 12 hours.

4.4.3.2 The capacity of each of the above required groups of pressurizer heaters shall be verified by energizing the heaters and measuring circuit current at least once per 92 days.

4.4.3.3 The cross-tie for the pressurizer heaters to the ESF power supply shall be demonstrated OPERABLE at least once per 18 months by energizing the heaters.

## REACTOR COOLANT SYSTEM

### 3/4.4.4 RELIEF VALVES

#### LIMITING CONDITION FOR OPERATION

3.4.4 Both power-operated relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

#### ACTION:

- a. With one or more PORV(s) inoperable because of excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) with power maintained to the block valve(s); otherwise be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.
- b. With one PORV inoperable due to causes other than excessive seat leakage, within 1 hour either restore the PORV to OPERABLE status or close the associated block valve and remove power from the block valve; restore the PORV to OPERABLE status within the following 72 hours or be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.
- c. With both PORVs inoperable due to causes other than excessive seat leakage, within 1 hour either restore at least one PORV to OPERABLE status or close its associated block valve and remove power from the block valve and be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.
- d. With one or more block valves inoperable, within 1 hour restore the block valve(s) to OPERABLE status or place its associated PORV in manual control. Restore at least one block valve to OPERABLE status within the next hour if both block valves are inoperable; restore any remaining inoperable block valve to OPERABLE status within 72 hours; otherwise, be in at least HOT STANDBY within the next 6 hours and HOT SHUTDOWN within the following 6 hours.
- e. The provisions of Specification 3.0.4 are not applicable.

#### SURVEILLANCE REQUIREMENTS

4.4.4.1 In addition to the requirements of Specification 4.0.5, each PORV shall be demonstrated OPERABLE at least once per 18 months by:

- a. Performance of a CHANNEL CALIBRATION of the actuation instrumentation, and
- b. Operating solenoid air control and check valves on associated air accumulators in the PORV control system through one complete cycle of full travel, and
- c. Operating the valve through one complete cycle of full travel during MODES 3 or 4.

4.4.4.2 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel unless the block valve is closed with power removed in order to meet the requirements of ACTION b. or c. of Specification 3.4.4.

## REACTOR COOLANT SYSTEM

### OVERPRESSURE PROTECTION SYSTEMS

#### LIMITING CONDITION FOR OPERATION

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3.4.9.3 At least two overpressure protection devices shall be OPERABLE, and each device shall be either:

- a. A residual heat removal (RHR) suction relief valve with a lift setting of less than or equal to 450 psig, or
- b. A power operated relief valve (PORV) with a lift setpoint that varies with RCS temperature which does not exceed the limit established in Figure 3.4-4.

APPLICABILITY: MODES 4, 5, and 6 with the reactor vessel head on.

#### ACTION:

- a. With one of the two required overpressure protection devices inoperable in MODE 4, restore two overpressure protection devices to OPERABLE status within 7 days or depressurize and vent the RCS through at least a 2 square inch vent within the next 8 hours.
- b. With one of the two required overpressure protection devices inoperable in MODES 5 or 6, restore two overpressure protection devices to OPERABLE status within 24 hours or vent the RCS through at least a 2 square inch vent within the next 8 hours.
- c. With both of the required overpressure protection devices inoperable, depressurize and vent the RCS through at least a 2 square inch vent within 8 hours.
- d. With the RCS vented per ACTIONS a, b, or c, verify the vent pathway at least once per 31 days when the pathway is provided by a valve(s) that is locked, sealed, or otherwise secured in the open position; otherwise, verify the vent pathway every 12 hours.
- e. In the event either the PORVs, RHR suction relief valves, or the RCS vents are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs, RHR suction relief valves, or RCS vents on the transient, and any corrective action necessary to prevent recurrence.
- f. The provisions of Specification 3.0.4 are not applicable.



## REACTOR COOLANT SYSTEM

### SURVEILLANCE REQUIREMENTS

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4.4.9.3.1 Each PORV shall be demonstrated OPERABLE when the PORVs are being used for cold overpressure protection by:

- a. Performance of an ANALOG CHANNEL OPERATIONAL TEST on the PORV actuation channel, but excluding valve operation, at least once per 31 days when the PORV is required OPERABLE; and
- b. Performance of a CHANNEL CALIBRATION on the PORV actuation channel at least once per 18 months; and
- c. Verifying the PORV isolation valve is open at least once per 72 hours.

4.4.9.3.2 Each RHR suction relief valve shall be demonstrated OPERABLE when the RHR suction relief valves are being used for cold overpressure protection as follows:

- a. For RHR suction relief valve RH8708B verify at least once per 72 hours that valves RH8702A and RH8702B are open.
- b. For RHR suction relief valve RH8708A verify at least once per 72 hours that valves RH8701A and RH8701B are open.
- c. Testing pursuant to Specification 4.0.5.

## REACTOR COOLANT SYSTEM

### 3/4.4.10 STRUCTURAL INTEGRITY

#### LIMITING CONDITION FOR OPERATION

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3.4.10 The structural integrity of ASME Code Class 1, 2, and 3 components shall be maintained in accordance with Specification 4.4.10.

APPLICABILITY: All MODES.

#### ACTION:

- a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.
- b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.
- c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) from service.
- d. The provisions of Specification 3.0.4 are not applicable.

#### SURVEILLANCE REQUIREMENTS

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4.4.10 In addition to the requirements of Specification 4.0.5, each reactor coolant pump flywheel shall be inspected as follows:

- a. Volumetric examination of the areas of higher stress concentration at the bore and keyways will be performed each 40 month period during refueling or maintenance shutdowns coinciding with the service inspection schedule as required by Section XI of the ASME Code.
- b. Visual examination of all exposed surfaces will be performed and a surface examination of the bore and keyway surfaces will be performed whenever the flywheels are removed for maintenance purposes, but not more frequently than once each 10 year interval.

## 3/4.4 REACTOR COOLANT SYSTEM

### BASES

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#### 3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with all reactor coolant loops in operation and maintain DNBR above the applicable design bases DNBR during all normal operations and anticipated transients. In MODES 1 and 2 with one reactor coolant loop not in operation this specification requires that the plant be in at least HOT STANDBY within 6 hours.

In MODE 3, two reactor coolant loops provide sufficient heat removal capability for removing decay heat even in the event of a bank withdrawal accident; however, a single reactor coolant loop provides sufficient heat removal if a bank withdrawal accident can be prevented, i.e., by opening the Reactor Trip System breakers. Single failure considerations require that two loops be OPERABLE at all times.

In MODE 4, and in MODE 5 with reactor coolant loops filled, a single reactor coolant loop or RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops (either RHR or RCS) be OPERABLE.

In MODE 5 with reactor coolant loops not filled, a single RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations, and the unavailability of the steam generators as a heat removing component, require that at least two RHR loops be OPERABLE.

The operation of one reactor coolant pump (RCP) or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a reactor coolant pump with one or more RCS cold legs less than or equal to 350°F are provided to prevent RCS pressure transients, caused by energy additions from the Secondary Coolant System, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

The requirement to maintain the boron concentration of an isolated loop greater than or equal to the boron concentration of the operating loops ensures that no reactivity addition to the core could occur during startup of an isolated loop. Verification of the boron concentration in an idle loop prior to opening the stop valves provides a reassurance of the adequacy of the boron concentration in the isolated loop.

Startup of an idle loop will inject cool water from the loop into the core. The reactivity transient resulting from this cool water injection is minimized by delaying isolated loop startup until its temperature is within 20°F of the operating loops.

## REACTOR COOLANT SYSTEM

### BASES

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#### 3/4.4.2 SAFETY VALVES

The pressurizer Code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 420,000 lbs per hour of saturated steam at the valve Setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization. In addition, the Overpressure Protection System provides a diverse means of protection against RCS overpressurization at low temperatures.

During operation, all pressurizer Code safety valves must be OPERABLE to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss-of-load assuming no Reactor trip and also assuming no operation of the power-operated relief valves or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code.

#### 3/4.4.3 PRESSURIZER

The limit on the maximum water volume (1656 cubic feet) in the pressurizer assures that the parameter is maintained within the normal steady-state envelope of operation assumed in the SAR. The limit is consistent with the initial SAR assumptions. The 12-hour periodic surveillance is sufficient to ensure that the parameter is restored to within its limit following expected transient operation. The maximum water volume also ensures that a steam bubble is formed and thus the RCS is not a hydraulically solid system. The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability of the plant to control Reactor Coolant System pressure and establish natural circulation.

#### 3/4.4.4 RELIEF VALVES

The power-operated relief valves (PORVs) and steam bubble function to relieve RCS pressure during all design transients up to and including the design step load decrease with steam dump. Operation of the PORVs minimizes the undesirable opening of the spring-loaded pressurizer Code safety valves. Each PORV has a remotely operated block valve to provide a positive shutoff capability should a relief valve become inoperable.

## REACTOR COOLANT SYSTEM

### BASES

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#### 3/4.4.4 RELIEF VALVES (Continued)

The OPERABILITY of the PORVs and block valves is determined on the basis of their being capable of performing the following functions:

- A. Manual control of PORVs to control reactor coolant system pressure. This is a function that is used for the steam generator tube rupture accident and for plant shutdown. This function has been classified as safety related for more recent plant designs.
- B. Maintaining the integrity of the reactor coolant pressure boundary. This is a function that is related to controlling identified leakage and ensuring the ability to detect unidentified reactor coolant pressure boundary leakage.
- C. Manual control of the block valve to: (1) unblock an isolated PORV to allow it to be used for manual control of reactor coolant system pressure (Item A), and (2) isolate a PORV with excessive seat leakage (Item B).
- D. Manual control of a block valve to isolate a stuck-open PORV.

Surveillance Requirements provide the assurance that the PORVs and block valves can perform their functions. The block valves are exempt from the surveillance requirements to cycle the valves when they have been closed to comply with the ACTION requirements. This precludes the need to cycle the valves with full system differential pressure or when maintenance is being performed to restore an inoperable PORV to operable status.

Surveillance requirement 4.4.4.1.b has been added to include testing of the mechanical and electrical aspects of control systems for air-operated PORVs.

Testing of PORVs in HOT STANDBY or HOT SHUTDOWN is required in order to simulate the temperature and pressure environmental effects on PORVs. In many PORV designs, testing at COLD SHUTDOWN is not considered to be a representative test for assessing PORV performance under normal plant operating conditions.

The PORVs are equipped with automatic actuation circuitry and manual control capability. Because no credit for PORV operation is taken in the FSAR analyses for Mode 1, 2 and 3 transients, the PORVs are considered OPERABLE in either the manual or automatic mode. It should be noted that the automatic mode is the preferred configuration, as this provides pressure relieving capability without reliance on operator action.

## REACTOR COOLANT SYSTEM

### BASES

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#### PRESSURE/TEMPERATURE LIMITS (Continued)

The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion.

Finally, the composite curves for the heatup rate data and the cooldown rate data are adjusted for possible errors in the pressure and temperature sensing instruments by the values indicated on the respective curves.

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of nonductile failure, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of two PORVs, or two RHR suction relief valves, or one PORV and one RHR suction relief valve, or an RCS vent opening of at least 2 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 350°F. Either PORV has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either: (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 50°F above the RCS cold leg temperatures, or (2) the start of a centrifugal charging pump and its injection into a water solid RCS.

These two scenarios are analyzed to determine the resulting overshoots assuming a single PORV actuation with a stroke time of 2.0 seconds from full closed to full open. Figure 3.4-4 is based upon this analysis and represents the maximum allowable PORV variable setpoint such that, for the two overpressurization transients noted, the resulting pressure will not exceed the Appendix G reactor vessel NDT limits (nominal 10 effective full power years for Unit 1 only).

#### 3/4.4.10 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i).



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

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-456

BRAIDWOOD STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 33  
License No. NPF-72

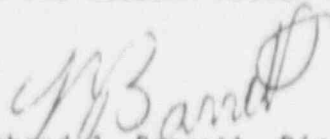
1. The Nuclear Regulatory Commission (the Commission) has found that:
  - A. The application for amendment by Commonwealth Edison Company (the licensee) dated June 28, 1991, as supplemented August 27, 1991, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the 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 2.C.(2) of Facility Operating License No. NPF-72 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 33 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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: November 18, 1991





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

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-457

BRAIDWOOD STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 33  
License No. NPF-77

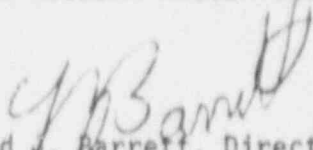
1. The Nuclear Regulatory Commission (the Commission) has found that:
  - A. The application for amendment by Commonwealth Edison Company (the licensee) dated June 28, 1991, as supplemented August 27, 1991, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter 1;
  - 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 2.C.(2) of Facility Operating License No. NPF-77 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 33 and the Environmental Protection Plan contained in Appendix B, both of which were attached to License No. NPF-72, dated July 2, 1987, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

  
Richard V. 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: November 18, 1991

ATTACHMENT TO LICENSE AMENDMENT NOS. 33 AND 33  
FACILITY OPERATING LICENSE NOS. NPF-72 AND NPF-77  
DOCKET NOS. STN 50-456 AND STN 50-457

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change. Pages indicated with an asterisk are provided for convenience.

<u>Remove Pages</u>	<u>Insert Pages</u>
* 3/4 4-11	* 3/4 4-11
3/4 4-12	3/4 4-12
3/4 4-39	3/4 4-39
3/4 4-41	3/4 4-41
* 3/4 4-42	* 3/4 4-42
*B 3/4 4-1	*B 3/4 4-1
B 3/4 4-2	B 3/4 4-2
	B 3/4 4-2a
B 3/4 4-16	B 3/4 4-16

## REACTOR COOLANT SYSTEM

### 3/4.4.3 PRESSURIZER

#### LIMITING CONDITION FOR OPERATION

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3.4.3 The pressurizer shall be OPERABLE with at least two groups of pressurizer heaters each having a capacity of at least 150 kW and a water level of less than or equal to 92%.

APPLICABILITY: MODES 1, 2, and 3.

#### ACTION:

- a. With less than two groups of pressurizer heaters OPERABLE, restore at least two groups of pressurizer heaters to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.
- b. With the pressurizer otherwise inoperable, be in at least HOT STANDBY with the Reactor trip breakers open within 6 hours and in HOT SHUTDOWN within the following 6 hours.

#### SURVEILLANCE REQUIREMENTS

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4.4.3.1 The pressurizer water level shall be determined to be within its limit at least once per 12 hours.

4.4.3.2 The capacity of each of the above required groups of pressurizer heaters shall be verified by energizing the heaters and measuring circuit current at least once per 92 days.

4.4.3.3 The cross-tie for the pressurizer heaters to the ESF power supply shall be demonstrated OPERABLE at least once per 18 months by energizing the heaters.

## REACTOR COOLANT SYSTEM

### 3.4.4 RELIEF VALVES

#### 3.4.4.1 ACTION FOR OPERATION

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3.4.4.1.1 All power-operated relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

- a. With one or more PORV(s) inoperable because of excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) with power maintained to the block valve(s); otherwise be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.
- b. With one PORV inoperable due to causes other than excessive seat leakage, within 1 hour either restore the PORV to OPERABLE status or close the associated block valve and remove power from the block valve, restore the PORV to OPERABLE status within the following 72 hours or be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.
- c. With both PORVs inoperable due to causes other than excessive seat leakage, within 1 hour either restore at least one PORV to OPERABLE status or close its associated block valve and remove power from the block valve and be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.
- d. With one or more block valves inoperable, within 1 hour restore the block valve(s) to OPERABLE status or place its associated PORV in manual control. Restore at least one block valve to OPERABLE status within the next hour if both block valves are inoperable; restore any remaining inoperable block valve to OPERABLE status within 72 hours; otherwise, be in at least HOT STANDBY within the next 6 hours and HOT SHUTDOWN within the following 6 hours.
- e. The provisions of Specification 3.4.4 are not applicable.

#### SURVEILLANCE REQUIREMENTS

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4.4.4.1 In addition to the requirements of Specification 4.0.5, each PORV shall be demonstrated OPERABLE at least once per 18 months by:

- a. Performance of CHANNEL CALIBRATION of the actuation instrumentation, and
- b. Operating solenoid air control and check valves on associated air accumulators in the PORV control system through one complete cycle of full travel, and
- c. Operating the valve through one complete cycle of full travel during MODES 3 or 4.

4.4.4.2 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel unless the block valve is closed with power removed in order to meet the requirements of ACTION b. or c. of Specification 3.4.4.

## REACTOR COOLANT SYSTEM

### OVERPRESSURE PROTECTION SYSTEMS

#### LIMITING CONDITION FOR OPERATION

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3.4.9.3 At least two overpressure protection devices shall be OPERABLE, and each device shall be either:

- a. A residual heat removal (RHR) suction relief valve with a lift setting of less than or equal to 450 psig, or
- b. A power operated relief valve (PORV) with a lift setpoint that varies with RCS temperature which does not exceed the limit established in Figure 3.4-4.

APPLICABILITY: MODES 4, 5, and 6 with the reactor vessel head on.

#### ACTION:

- a. With one of the two required overpressure protection devices inoperable in MODE 4, restore two overpressure protection devices to OPERABLE status within 7 days or depressurize and vent the RCS through at least a 2 square inch vent within the next 8 hours.
- b. With one of the two required overpressure protection devices inoperable in MODES 5 or 6, restore two overpressure protection devices to OPERABLE status within 24 hours or vent the RCS through at least a 2 square inch vent within the next 8 hours.
- c. With both of the required overpressure protection devices inoperable, depressurize and vent the RCS through at least a 2 square inch vent within 8 hours.
- d. With the RCS vented per ACTIONS a, b, or c, verify the vent pathway at least once per 31 days when the pathway is provided by a valve(s) that is locked, sealed, or otherwise secured in the open position; otherwise, verify the vent pathway every 12 hours.
- e. In the event either the PORVs, RHR suction relief valves, or the RCS vents are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs, RHR suction relief valves, or RCS vents on the transient, and any corrective action necessary to prevent recurrence.
- f. The provisions of Specification 3.0.4 are not applicable.

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS

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4.4.9.3.1 Each PORV shall be demonstrated OPERABLE when the PORVs are being used for cold overpressure protection by:

- a. Performance of an ANALOG CHANNEL OPERATIONAL TEST on the PORV actuation channel, but excluding valve operation, at least once per 31 days when the PORV is required OPERABLE; and
- b. Performance of a CHANNEL CALIBRATION on the PORV actuation channel at least once per 18 months; and
- c. Verifying the PORV isolation valve is open at least once per 72 hours.

4.4.9.3.2 Each RHR suction relief valve shall be demonstrated OPERABLE when the RHR suction relief valves are being used for cold overpressure protection as follows:

- a. For RHR suction relief valve RH8708B verify at least once per 72 hour that valves RH8702A and RH8702B are open.
- b. For RHR suction relief valve RH8708A verify at least once per 72 hours that valves RH8701A and RH8701B are open.
- c. Testing pursuant to Specification 4.0.5.

## REACTOR COOLANT SYSTEM

### 3/4.4.10 STRUCTURAL INTEGRITY

#### LIMITING CONDITION FOR OPERATION

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3.4.10 The structural integrity of ASME Code Class 1, 2, and 3 components shall be maintained in accordance with Specification 4.4.10.

APPLICABILITY: All MODES.

#### ACTION:

- a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.
- b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.
- c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) from service.
- d. The provisions of Specification 3.0.4 are not applicable.

#### SURVEILLANCE REQUIREMENTS

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4.4.10 In addition to the requirements of Specification 4.0.5, each reactor coolant pump flywheel shall be inspected as follows:

- a. Volumetric examination of the areas of higher stress concentration at the bore and keyways will be performed each 40 month period during refueling or maintenance shutdowns coinciding with the service inspection schedule as required by Section XI of the ASME Code.
- b. Visual examination of all exposed surfaces will be performed and a surface examination of the bore and keyway surfaces will be performed whenever the flywheels are removed for maintenance purposes, but not more frequently than once each 10 year interval.



## 3/4.4 REACTOR COOLANT SYSTEM

### BASES

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#### 3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with all reactor coolant loops in operation and maintain DNBR above the applicable design bases DNBR during all normal operations and anticipated transients. In MODES 1 and 2 with one reactor coolant loop not in operation this specification requires that the plant be in at least HOT STANDBY within 6 hours.

In MODE 3, two reactor coolant loops provide sufficient heat removal capability for removing decay heat even in the event of a bank withdrawal accident; however, a single reactor coolant loop provides sufficient heat removal if a bank withdrawal accident can be prevented, i.e., by opening the Reactor Trip System breakers. Single failure considerations require that two loops be OPERABLE at all times.

In MODE 4, and in MODE 5 with reactor coolant loops filled, a single reactor coolant loop or RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops (either RHR or RCS) be OPERABLE.

In MODE 5 with reactor coolant loops not filled, a single RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations, and the unavailability of the steam generators as a heat removing component, require that at least two RHR loops be OPERABLE.

The operation of one reactor coolant pump (RCP) or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a reactor coolant pump with one or more RCS cold legs less than or equal to 350°F are provided to prevent RCS pressure transients, caused by energy additions from the Secondary Coolant System, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

The requirement to maintain the boron concentration of an isolated loop greater than or equal to the boron concentration of the operating loops ensures that no reactivity addition to the core could occur during startup of an isolated loop. Verification of the boron concentration in an idle loop prior to opening the stop valves provides a reassurance of the adequacy of the boron concentration in the isolated loop.

Startup of an idle loop will inject cool water from the loop into the core. The reactivity transient resulting from this cool water injection is minimized by delaying isolated loop startup until its temperature is within 20°F of the operating loops.

## REACTOR COOLANT SYSTEM

### BASES

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#### 3/4.4.2 SAFETY VALVES

The pressurizer Code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 420,000 lbs per hour of saturated steam at the valve Setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization. In addition, the Overpressure Protection System provides a diverse means of protection against RCS overpressurization at low temperatures.

During operation, all pressurizer Code safety valves must be OPERABLE to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss-of-load assuming no Reactor trip and also assuming no operation of the power-operated relief valves or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code.

#### 3/4.4.3 PRESSURIZER

The limit on the maximum water volume (1656 cubic feet) in the pressurizer assures that the parameter is maintained within the normal steady-state envelope of operation assumed in the SAR. The limit is consistent with the initial SAR assumptions. The 12-hour periodic surveillance is sufficient to ensure that the parameter is restored to within its limit following expected transient operation. The maximum water volume also ensures that a steam bubble is formed and thus the RCS is not a hydraulically solid system. The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability of the plant to control Reactor Coolant System pressure and establish natural circulation.

#### 3/4.4.4 RELIEF VALVES

The power-operated relief valves (PORVs) and steam bubble function to relieve RCS pressure during all design transients up to and including the design step load decrease with steam dump. Operation of the PORVs minimizes the undesirable opening of the spring-loaded pressurizer Code safety valves. Each PORV has a remotely operated block valve to provide a positive shutoff capability should a relief valve become inoperable.

## REACTOR COOLANT SYSTEM

### BASES

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#### 3/4.4.4 RELIEF VALVES (Continued)

The OPERABILITY of the PORVs and block valves is determined on the basis of their being capable of performing the following functions:

- A. Manual control of PORVs to control reactor coolant system pressure. This is a function that is used for the steam generator tube rupture accident and for plant shutdown. This function has been classified as safety related for more recent plant designs.
- B. Maintaining the integrity of the reactor coolant pressure boundary. This is a function that is related to controlling identified leakage and ensuring the ability to detect unidentified reactor coolant pressure boundary leakage.
- C. Manual control of the block valve to: (1) unblock an isolated PORV to allow it to be used for manual control of reactor coolant system pressure (Item A), and (2) isolate a PORV with excessive seat leakage (Item B).
- D. Manual control of a block valve to isolate a stuck-open PORV.

Surveillance Requirements provide the assurance that the PORVs and block valves can perform their functions. The block valves are exempt from the surveillance requirements to cycle the valves when they have been closed to comply with the ACTION requirements. This precludes the need to cycle the valves with full system differential pressure or when maintenance is being performed to restore an inoperable PORV to operable status.

Surveillance requirement 4.4.4.1.b has been added to include testing of the mechanical and electrical aspects of control systems for air-operated PORVs.

Testing of PORVs in HOT STANDBY or HOT SHUTDOWN is required in order to simulate the temperature and pressure environmental effects on PORVs. In many PORV designs, testing at COLD SHUTDOWN is not considered to be a representative test for assessing PORV performance under normal plant operating conditions.

The PORVs are equipped with automatic actuation circuitry and manual control capability. Because no credit for PORV operation is taken in the FSAR analyses for Mode 2 and 3 transients, the PORVs are considered OPERABLE in either the manual or automatic mode. It should be noted that the automatic mode is the preferred configuration, as this provides pressure relieving capability without reliance on operator action.

## REACTOR COOLANT SYSTEM

### BASES

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#### PRESSURE/TEMPERATURE LIMITS (Continued)

comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration.

The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion.

Finally, the composite curves for the heatup rate data and the cooldown rate data are adjusted for possible errors in the pressure and temperature sensing instruments by the values indicated on the respective curves.

Although the pressurizer operates in temperature ranges above those in which there is reason for concern of nonductile failure, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of two PORVs, or two RHR suction relief valves, or one PORV and one RHR suction relief valve, or an RCS vent opening of at least 2 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 350°F. Either PORV has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either: (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 50°F above the RCS cold leg temperatures, or (2) the start of a centrifugal charging pump and its injection into a water solid RCS.

These two scenarios are analyzed to determine the resulting overshoots assuming a single PORV actuation with a stroke time of 2.0 seconds from full closed to full open. Figure 3.4-4a (3.4-4b) are based upon this analysis and represents the maximum allowable PORV variable setpoint such that, for the two overpressurization transients noted, the resulting pressure will not exceed the Appendix G reactor vessel NDT limits.

#### 3/4.4.10 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i).