Docket File
TSC3
50-454/455

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

April 15, 1997

Ms. Irene Johnson, Acting Manager
Nuclear Regulatory Services
Commonwealth Edison Company
Executive Towers West III
1400 Opus Place, Suite 500
Downers Grove, IL 60515

SUBJECT: ISSUANCE OF AMENDMENTS (TAC NOS. M95715, M95716, M95717 AND M95718)

Dear Ms. Johnson:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 87 to Facility Operating License No. NPF-37 and Amendment No. 87 to Facility Operating License No. NPF-66 for the Byron Station, Unit Nos. 1 and 2, respectively, and Amendment No. 79 to Facility Operating License No. NPF-72 and Amendment No. 79 to Facility Operating License No. NPF-77 for the Braidwood Station, Unit Nos. 1 and 2, respectively. The amendments are in response to your application dated April 29, 1996, as supplemented on January 21 and March 25, 1997.

The amendments would:

1. Revise Technical Specification (TS) 3.7.1.1, Action a., to require the unit to be in hot shutdown, rather than cold shutdown, for consistency with NUREG-1431, "Standard Technical Specifications for Westinghouse Plants," and add a new Action b. to clarify the shutdown requirements when there are more than three inoperable main steam line American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code) safety valves on any one steam generator.

2. Revise TS Surveillance Requirement 4.7.1.1 to clarify that Specification 4.0.4 does not apply for entry into Mode 3 for Byron and Braidwood and for Braidwood only, delete the one-time requirements for Unit 1, Cycle 5 and Unit 2 after outage A2F27.

3. Revise the maximum allowable power range neutron flux high trip setpoints in Table 3.7-1.

4. Revise Table 3.7-2 to increase the as-found main steam safety valve (MSSV) lift setpoint tolerance to ±3 percent, provide an as-left setpoint tolerance of ±1 percent, and change a table notation.

5. Delete the orifice size column from Table 3.7-2.

6. Revise the Bases for TS 3.7.1.1 to be consistent with the proposed changes to TS 3.7.1.1.

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 DF01

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

Sincerely,

Original signed by:

George F. Dick, Senior Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket Nos. STN 50-454, STN 50-455,
STN 50-456 and STN 50-457

Enclosures:
1. Amendment No. 87 to NPF-37
2. Amendment No. 87 to NPF-66
3. Amendment No. 79 to NPF-72
4. Amendment No. 79 to NPF-77
5. Safety Evaluation

cc w/encl: see next page

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R. Jones, 08E23	~~R. Wessman, 07E23~~

*concurrence provided by memo dated 2/10/97; no major changes

DOCUMENT NAME: G:\CMNTJR\BRAID-BY\BB95715.AMD

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-454

BYRON STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 87
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 April 29, 1996, as supplemented on January 21 and March 25, 1997, 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:

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PDR ADOCK 05000454
P PDR

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 87 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 and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

George F. Dick

George F. Dick, Senior Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 15, 1997

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-455

BYRON STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 87
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 April 29, 1996, as supplemented on January 21 and March 25, 1997, 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-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. 87 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 and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

George F. Dick

George F. Dick, Senior Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 15, 1997

ATTACHMENT TO LICENSE AMENDMENT NOS. 87 AND 87

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.

Remove Pages	Insert Pages
X	X
3/4 7-1	3/4 7-1
3/4 7-2	3/4 7-2
3/4 7-3	3/4 7-3
B 3/4 7-1	B 3/4 7-1
B 3/4 7-2	B 3/4 7-2

3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line Code safety valves associated with each steam generator shall be OPERABLE with lift settings as specified in Table 3.7-2.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With up to 3 inoperable main steam line Code safety valves on any one steam generator, within 4 hours, either restore the inoperable valves to OPERABLE status, or reduce the Power Range Neutron Flux High Trip Setpoints per Table 3.7-1; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With more than 3 inoperable main steam line Code safety valves on any one steam generator, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.1 No additional requirements other than those required by Specification 4.0.5. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES

MAXIMUM NUMBER OF INOPERABLE SAFETY VALVES ON ANY OPERATING STEAM GENERATOR	MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT (PERCENT OF RATED THERMAL POWER)
1	60
2	43
3	25

TABLE 3.7-2

STEAM LINE SAFETY VALVES PER LOOP

VALVE NUMBER	LIFT SETTING (±3%)*#
MS013(A-D)	1235 psig
MS014(A-D)	1220 psig
MS015(A-D)	1205 psig
MS016(A-D)	1190 psig
MS017(A-D)	1175 psig

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

#All tested valves shall be set to ±1% tolerance.

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line Code safety valves ensures that the Secondary Coolant System pressure will be limited to within 110% (1320 psia) of its design pressure of 1200 psia during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 102% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam dumps to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The requirement that the main steam line Code safety valves be set to within ±1% of the appropriate setpoint is consistent with Section III of the ASME Boiler and Pressure Vessel Code. The allowed operating tolerance of ±3% is supported by the Commonwealth Edison Company, Byron/Braidwood Unit 1 & 2 Overpressure Protection Report. The total relieving capacity for all valves on all of the steam lines is 17.958×10^6 lbs/h which is 119% of the total secondary steam flow of 15.135×10^6 lbs/h at 100% RATED THERMAL POWER. A minimum of two OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-1.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in Secondary Coolant System steam flow and THERMAL POWER required by the reduced Reactor trip settings of the Power Range Neutron Flux channels. The Reactor Trip Setpoint reductions are derived on the following bases:

$$\text{High}\Phi = \frac{100}{Q}\left(\frac{w_s h_{fg} N}{K}\right)$$

Where:

HighΦ = Safety Analysis power range high neutron flux setpoint, in percent.

Q = Nominal NSSS power rating of the plant (including reactor coolant pump heat), in Mwt (=3427.6 MWt).

K = Conversion factor = 947.82 (BTU/sec.)/MWt.

w_s = minimum total steam flow rate capability of the operable MSSVs on any one steam generator at the highest MSSV opening pressure including tolerance and accumulation, as appropriate, in lbm/sec.

3/4.7.1.1 SAFETY VALVES (Continued)

h_{fg} = Heat of vaporization for steam at the highest MSSV opening pressure including tolerance and accumulation, as appropriate, in BTU/lbm.

N = Number of loops in the plant (=4).

The values calculated from this algorithm were adjusted lower for use in Technical Specification 3.7.1.1 to account for instrument and channel uncertainties (9% power).

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

The OPERABILITY of the Auxiliary Feedwater System ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total loss-of-offsite power.

The motor-driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 740 gpm at a pressure of 1450 psig to the entrance of the steam generators. The diesel-driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 740 gpm at a pressure of 1450 psig to the entrance of the steam generators. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F when the RHR System may be placed into operation.

3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank with the minimum water level of 40% ensures that sufficient water (200,000 gallons) is available to maintain the RCS at HOT STANDBY conditions for 9 hours with steam discharge to the atmosphere concurrent with total loss-of-offsite power. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

3/4.7.1.4 SPECIFIC ACTIVITY

The limitations on Secondary Coolant System specific activity ensure that the resultant offsite radiation dose will be limited to a small fraction of 10 CFR Part 100 dose guideline values in the event of a steam line break. This dose also includes the effects of a coincident 1 gpm reactor to secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the safety analyses.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-456

BRAIDWOOD STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 79
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 April 29, 1996, as supplemented on January 21 and March 25, 1997, 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. 79 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 and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

George F. Dick

George F. Dick, Senior Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 15, 1997

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-457

BRAIDWOOD STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 79
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 April 29, 1996, as supplemented on January 21 and March 25, 1997, 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. 79 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 if its issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

George F. Dick

George F. Dick, Senior Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 15, 1997

ATTACHMENT TO LICENSE AMENDMENT NOS. 79 AND 79

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.

Remove Pages	Insert Pages
X	X
3/4 7-1	3/4 7-1
3/4 7-2	3/4 7-2
3/4 7-3	3/4 7-3
B 3/4 7-1	B 3/4 7-1
B 3/4 7-2	B 3/4 7-2

SECTION		PAGE
3/4.7 PLANT SYSTEMS		

3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line Code safety valves associated with each steam generator shall be OPERABLE with lift settings as specified in Table 3.7-2.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With up to 3 inoperable main steam line Code safety valves on any one steam generator, within 4 hours, either restore the inoperable valves to OPERABLE status, or reduce the Power Range Neutron Flux High Trip Setpoints per Table 3.7-1; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With more than 3 inoperable main steam line Code safety valves on any one steam generator, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.1 No additional requirements other than those required by Specification 4.0.5. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

TABLE 3.7-1

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TABLE 3.7-2

STEAM LINE SAFETY VALVES PER LOOP

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MS016(A-D)	1190 psig
MS017(A-D)	1175 psig

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

#All tested valves shall be set to ±1% tolerance.

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line Code safety valves ensures that the Secondary Coolant System pressure will be limited to within 110% (1320 psia) of its design pressure of 1200 psia during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 102% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam dumps to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The requirement that the main steam line Code safety valves be set to within ±1% of the appropriate setpoint is consistent with Section III of the ASME Boiler and Pressure Vessel Code. The allowed operating tolerance of ±3% is supported by the Commonwealth Edison Company, Byron/Braidwood Unit 1 & 2 Overpressure Protection Report. The total relieving capacity for all valves on all of the steam lines is 17.958×10^6 lbs/h which is 119% of the total secondary steam flow of 15.135×10^6 lbs/h at 100% RATED THERMAL POWER. A minimum of two OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-1.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in Secondary Coolant System steam flow and THERMAL POWER required by the reduced Reactor trip settings of the Power Range Neutron Flux channels. The Reactor Trip Setpoint reductions are derived on the following bases:

$$\text{High}\Phi = \frac{100}{Q}\left(\frac{w_s h_{fg} N}{K}\right)$$

Where:

HighΦ = Safety Analysis power range high neutron flux setpoint, in percent.

Q = Nominal NSSS power rating of the plant (including reactor coolant pump heat), in Mwt (=3427.6 MWt).

K = Conversion factor = 947.82 (BTU/sec.)/MWt.

w_s = minimum total steam flow rate capability of the operable MSSVs on any one steam generator at the highest MSSV opening pressure including tolerance and accumulation, as appropriate, in lbm/sec.

3/4.7.1.1 SAFETY VALVES (Continued)

h_{fg} = Heat of vaporization for steam at the highest MSSV opening pressure including tolerance and accumulation, as appropriate, in BTU/lbm.

N = Number of loops in the plant (=4).

The values calculated from this algorithm were adjusted lower for use in Technical Specification 3.7.1.1 to account for instrument and channel uncertainties (9% power).

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

The OPERABILITY of the Auxiliary Feedwater System ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total loss-of-offsite power.

The motor-driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 740 gpm at a pressure of 1450 psig to the entrance of the steam generators. The diesel-driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 740 gpm at a pressure of 1450 psig to the entrance of the steam generators. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F when the RHR System may be placed into operation.

3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank with the minimum water level of 40% ensures that sufficient water (200,000 gallons) is available to maintain the RCS at HOT STANDBY conditions for 9 hours with steam discharge to the atmosphere concurrent with total loss-of-offsite power. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

3/4.7.1.4 SPECIFIC ACTIVITY

The limitations on Secondary Coolant System specific activity ensure that the resultant offsite radiation dose will be limited to a small fraction of 10 CFR Part 100 dose guideline values in the event of a steam line break. This dose also includes the effects of a coincident 1 gpm reactor to secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the safety analyses.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 87 TO FACILITY OPERATING LICENSE NO. NPF-37,

AMENDMENT NO. 87 TO FACILITY OPERATING LICENSE NO. NPF-66,

AMENDMENT NO. 79 TO FACILITY OPERATING LICENSE NO. NPF-72,

AND AMENDMENT NO. 79 TO FACILITY OPERATING LICENSE NO. NPF-77

COMMONWEALTH EDISON COMPANY

BYRON STATION, UNIT NOS. 1 AND 2

BRAIDWOOD STATION, UNIT NOS. 1 AND 2

DOCKET NOS. STN 50-454, STN 50-455, STN 50-456 AND STN 50-457

1.0 INTRODUCTION

By letter dated April 29, 1996, as supplemented on January 21 and March 25, 1997, Commonwealth Edison Company (ComEd, the licensee) requested an amendment to Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, which would change the Technical Specification (TS) Section 3/4.7.1, "Turbine Cycle Safety Valves" and the associated Bases. Additional information was provided in a letter from ComEd to the NRC dated March 25, 1997, and in a Westinghouse letter to the NRC dated January 28, 1994. The March 25, 1997, submittal provided additional information that did not change the initial proposed no significant hazards consideration determination.

The proposed TS would:

1. Revise TS 3.7.1.1, Action a., to require the unit to be in hot shutdown, rather than cold shutdown, for consistency with Revision 1 of NUREG-1431, "Standard Technical Specifications for Westinghouse Plants," and adding a new Action b. to clarify the shutdown requirements when there are more than three inoperable main steam line American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code) safety valves on any one steam generator.

2. Revise TS Surveillance Requirement (SR) 4.7.1.1 to clarify that Specification 4.0.4 does not apply for entry into Mode 3 for Byron and Braidwood and for Braidwood only, deleting the one-time requirements for Unit 1, Cycle 5 and Unit 2 after outage A2F27 (during Cycle 5).

3. Revise the maximum allowable power range neutron flux high trip setpoints in Table 3.7-1.

4. Revise Table 3.7-2 to increase the as-found main steam safety valve (MSSV) lift setpoint to ±3 percent, provide as-left setpoint tolerance of ±1 percent, and change a table notation.

5. Delete the orifice size column from Table 3.7-2.

6. Revise the Bases for TS 3.7.1.1 to be consistent with the proposed changes to TS 3.7.1.1.

2.0 EVALUATION

The evaluations of each of the six proposed changes listed above in Section 1.0 are given below.

2.1 Technical Specification 3.7.1.1

a. Description

ComEd proposes to revise TS 3.7.1.1, Action a., to delete the phrase "four reactor coolant loops and associated steam generators in operation and with." The final mode requirements are changed to require the unit to be in hot shutdown within 6 hours of reaching hot standby, rather than cold shutdown within 30 hours of reaching hot standby. The action statement adds words to clarify that the action applies when there are up to three inoperable main steam line Code safety valves on any one steam generator.

ComEd proposes to add a new Action b. to clarify the shutdown requirements when there are more than three inoperable main steam line Code safety valves on any one steam generator; i.e., there is no provision to restore the inoperable valves to operable status within 4 hours before the 6 hour clock to hot standby begins. The current Action b. becomes Action c.

ComEd also proposes to revise the title to Table 3.7-1 to delete "during four loop operation." The Table of Contents is also revised to reflect the title change.

b. Impact of the Changes

The proposed change by ComEd to require the final mode to be hot shutdown rather than cold shutdown is consistent with the applicability section of the specification, which does not require the main steam safety valves (MSSVs) to be operable in hot shutdown. There are no credible transients requiring the MSSVs in Modes 4 and 5. The steam generators are not normally used for heat removal in Modes 5 and 6 and, thus, can not be overpressurized. NUREG-1431 does not include requirements for the MSSVs to be operable in these modes. The change will also eliminate the unnecessary transient that had been imposed on the unit by forcing entry into cold shutdown. Therefore, these proposed changes have no significant negative impact on any system or operating mode, and provide a benefit of eliminating unnecessary plant transients.

The new Action b. provides clarification to the current Action a. and is consistent with the requirements in NUREG-1431. The shutdown requirements are clarified based on the number of inoperable valves. There are no technical changes to these times.

The title change to TS 3.7-1 is an editorial change.

The staff has found the above changes to be acceptable as they have no significant negative impact and they are consistent with NUREG-1431.

2.2 Surveillance Requirement 4.7.1.1

a. Description

ComEd proposes to revise TS SR 4.7.1.1 to clarify that TS 4.0.4 does not apply for entry into Mode 3 for Byron and Braidwood; i.e., entry into Mode 3 would be allowed prior to completing surveillance testing on the MSSVs. The proposed revision also deletes the one-time requirements for Braidwood, Unit 1, Cycle 5 and Braidwood, Unit 2, after outage A2F27 (during Cycle 5) that were added by Amendment Nos. 49 and 51.

The current practice for testing the MSSVs is to enter Action a. for TS 3.7.1.1 while testing the valves, since the ASME Code, 1983 Edition through summer 1983 Addenda requires the plant to be at normal operating temperature and pressure (Mode 3) and TS SR 4.0.4 would require the testing to be completed prior to entering Mode 3, which places severe time restriction on the valve testing.

b. Impact of the Changes

Changing TS SR 4.7.1.1 to delete the one-time requirements imposed by previous Braidwood amendments and allow entry into Mode 3 for MSSV testing for Byron and Braidwood will permit testing of the MSSVs at normal operating pressures and temperatures in accordance with the applicable codes while allowing a reasonable amount of time for completion of the surveillance.

The staff finds these changes to be acceptable as they have no significant negative impact and they are consistent with NUREG-1431.

The one-time requirements for Braidwood are no longer applicable and, therefore, deletion of them is acceptable.

2.3 Changes to Maximum Allowable Power Range Neutron Flux High Setpoints

a. Description

ComEd proposes to revise the power range neutron flux high setpoints in the event of MSSVs inoperability based on a revision to the method for calculating the setpoints that was provided in a Westinghouse Nuclear Safety Advisory Letter, (NSAL-94-001), which was sent to the NRC on January 28, 1994. The existing equation for the Reactor Trip Setpoint reductions for four-loop operation in TS Bases 3/4 7-1 has been replaced by:

$$\text{High}\Phi = \frac{100}{Q}\left(\frac{w_s h_{fg} N}{K}\right)$$

Where:

HighΦ = Safety Analysis power range high neutron flux setpoint, in percent.

Q = Nominal NSSS power rating of the plant (including reactor coolant pump heat), in Mwt (=3427.6 MWt).

K = Conversion factor = 947.82 (BTU/sec.)/MWt.

w_s = minimum total steam flow rate capability of the operable MSSVs on any one steam generator at the highest MSSV opening pressure including tolerance and accumulation, as appropriate, in lbm/sec.

h_{fg} = Heat of vaporization for steam at the highest MSSV opening pressure including tolerance and accumulation, as appropriate, in BTU/lbm.

N = Number of loops in the plant (=4).

The TS Bases clarify that the values from the above algorithm must be adjusted lower for use in TS 7.1.1.1 to account for instrument and channel uncertainties (typically 9 percent power) as specified in NSAL-94-001. This reduction has been included in the proposed values in the TS Table 3.7-1.

The proposed changes in TS Table 3.7-1 to the maximum allowable power range neutron flux high setpoint are as follows:

Maximum Number of Inoperable Safety Valves on any Steam Generator	Maximum allowable power range neutron flux high setpoint (percent of rated thermal power) Current TS Value	Proposed Value
1	87	60
2	65	43
3	43	25

b. Impact of the Changes

Westinghouse has determined (NSAL-94-001) that the method used to calculate the current setpoints may not be valid under certain conditions. That method assumes a linear relationship between the maximum allowable initial power level and available MSSV relief capacity. In particular, a loss of load/

turbine trip (LOL/TT) transient from a reduced power condition may result in overpressurization of the main steam system. With fully operational MSSVs, it can be demonstrated that overpressure protection is provided at all initial power levels. However, TS 3.7.1.1 allows operation with a reduced number of operable MSSVs at a reduced power level. In certain LOL/TT scenarios from low initial power levels with pressure control available, the reactor trip may be delayed until low-low steam generator level is reached. By this time, using the allowed reactor power levels of the current setpoints in TS 3.7.1.1 with one or more safety valves inoperable, the secondary side pressure may exceed the acceptance criterion of 110 percent pressure. As a result, Westinghouse recommended that the maximum allowable power range neutron flux high trip setpoints of TS 3.7.1.1 be lowered.

The staff has found these proposed changes to be acceptable as they are more conservative than the current TS requirements and are based on calculations performed by Westinghouse. There is no operational impact on Braidwood or Byron since both stations have already administratively incorporated these reduced power range neutron flux high trip setpoints as a result of NSAL-94-001.

2.4 Main Steam Safety Valve Setpoint Tolerance

a. Description

ComEd proposes to revise Table 3.7-2 to allow MSSV as-found setpoints to be within ±3 percent of the lift settings. The obsolete note is deleted and replaced with a note that requires all tested valves to be set to ±1 percent tolerance. The new note maintains the current as-left setpoint tolerance requirement and removes the reference to the outdated provision.

The change is consistent with the ANSI/ASME OM-1-1981 Code, "Requirements for Inservice Performance Testing of Nuclear Power Plant Pressure Relief Devices" which is a more recent code endorsed by the NRC. ComEd has determined that, over an operating cycle, the setpoint of the MSSVs often change by more than 1 percent from the original set-pressure. As a result, when valves are tested and one or more are found outside the current ±1 percent criterion, the plant is placed in an action statement. Changing the as-found setpoint to ±3 percent of the lift settings is expected to preclude frequently entering the action statement. Also, the conditions requiring the one-time note in Amendment No. 49 for Braidwood Station and Amendment No. 51 for Byron Station to address calculational errors in a vendor MSSV calibration procedure have been corrected.

b. Impact of Changes

The effects of increasing the as-found lift setpoint tolerance on the MSSVs were examined by the licensee. The various evaluations were made by Westinghouse (letter from ComEd to NRC dated March 25, 1997) and included non-loss-of-coolant accident and loss-of-coolant (LOCA) related evaluations. The non-LOCA evaluations included ΔT protection and departure from nuclear boiling (DNB) events. The ΔT protection events included the affect of the increase in MSSV lift tolerance on the overtemperature ΔT and overpower ΔT setpoints.

The DNB events reviewed included the following areas (Updated Final Safety Analysis Report (UFSAR) sections given): Feedwater system malfunction-reduction in temperature (15.1.1); Feedwater system malfunction-increase in feedwater flow (15.1.2); Excessive increase in secondary steam flow (15.1.3); Inadvertent opening of a steam generator relief or safety valve (15.1.4); Steam system piping failure (15.1.5); Partial loss of forced reactor coolant flow (15.3.1); Complete loss of forced reactor coolant flow (15.3.2); Reactor coolant pump shaft seizure (15.3.3); Reactor coolant pump shaft break (15.3.4); Uncontrolled Rod Cluster Control Assembly (RCCA) bank withdrawal from a subcritical condition (15.4.1); Uncontrolled RCCA bank withdrawal at power (15.4.2); RCCA misalignment (15.4.3); Inadvertent operation of the Emergency Core Cooling System (ECCS) (15.5.1); Inadvertent opening of a pressurizer safety or relief valve (15.4.4); Startup of an inactive reactor coolant pump (15.4.4); and Chemical Volume Control System (CVCS) malfunction (boron dilution) (15.4.6). It was found that the above non-LOCA DNB transients are not adversely impacted by the proposed change, and the results and conclusions presented in the UFSAR remain valid.

The following long-term heat removal events were reviewed: Loss of non-emergency AC power to plant auxiliaries (15.2.6); Loss of normal feedwater (15.2.7); and Feedwater system pipe break (15.2.8). It was found that the proposed change does not impact the long-term cooling overpressurization requirements.

For the -3 percent tolerance, the secondary steam releases generated for the offsite dose calculations for the following non-LOCA transients were examined: steam system piping failure (Table 15.1-3), the loss of external load (Table 15.2-4), and the RCP shaft seizure (locked rotor - Table 15.3-3). It was found that the current releases remain valid.

The loss of external load/turbine trip is the limiting non-LOCA event for overpressurization. It was determined that this is the only UFSAR transient that is impacted such that a new analysis must be performed in order to address effects of the MSSV lift setpoint increase from ±1 to ±3 percent. The loss of external load/turbine trip event was analyzed by Westinghouse in order to quantify the impact of the setpoint tolerance relaxation. The evaluation by Westinghouse (letter from ComEd to USNRC, dated April 29, 1996) concluded that all applicable acceptance criteria for this event remain satisfied and demonstrated that the conclusion presented in the FSAR remains valid.

Regarding the LOCA and LOCA-related evaluations, the following accidents were reviewed for the licensee by Westinghouse (letter from ComEd to NRC dated March 25, 1997) for the effects of increasing the safety valve setpoint tolerance from ±1 to ±3 percent: Large Break LOCA (15.6.5); Small Break LOCA (15.6.5); LOCA blowdown reactor vessel and RCS loop forces (3.9); LOCA mass and energy released for containment integrity analyses (6.2); Steam generator tube rupture (15.6.3); Hot leg switchover of the ECCS to prevent potential boron precipitation (6.3.2.5); and Post-LOCA long-term core cooling (15.6.5). The effect of an increase in the MSSV lift setpoint increase from ±1 to ±3 percent on the FSAR LOCA was evaluated. In each case, the applicable regulatory or design limit was satisfied. Specific analyses were performed for small break LOCA assuming the current MSSV TS set pressures plus the proposed additional 3 percent uncertainty. The calculated peak cladding temperatures were well below the 10 CFR 50.46 2200 degrees Fahrenheit limit.

In summary, the departure from nucleate boiling design basis, primary and secondary pressure limits and dose release limits continue to be met. For the LOCA, the peak cladding temperatures remain well below the limits specified in 10 CFR 50.46. Neither the mass and energy release to the containment following a postulated LOCA, nor the analysis of containment response following a LOCA, credit the MSSV in mitigating the consequences of an accident. Therefore, changing the MSSV lift setpoint tolerances would have no impact on the containment integrity analysis. In addition, based on the conclusion of the transient analysis, the change to the MSSV tolerance will not affect the calculated steam line break mass and energy releases inside containment. The conclusions for the current accident analyses for LOCAs and non-LOCAs as described in the UFSAR remain valid. Therefore, the staff finds the change to allow the MSSV as-found setpoints to be within ±3 percent of the lift settings to be acceptable.

2.5 Change to Delete the Orifice Size Column from Table 3.7-2

a. Description

ComEd proposes to delete the orifice size column from Table 3.7-2. This information is not used by the operator and the operators have no control of MSSV orifice size. MSSV capacity is discussed in UFSAR Section 10.3, Main Steam Supply System, and in UFSAR Chapter 15, Accident Analyses. The information is not appropriate for inclusion in the TSs and is, therefore, proposed to be deleted.

b. Impact of Changes

The staff finds this change to be acceptable as the proposed change does not introduce any new equipment, equipment modifications, or any new or different modes of plant operation. This change will not affect the operational characteristics of any equipment or systems. The MSSVs are described in detail in the UFSAR and the information is not required in the TSs.

2.6 Technical Specification Bases 3/4.7.1.1

a. Description

The Bases for TS 3/4.7.1.1 contain information on secondary system pressure limits and how operability of the MSSV maintains design pressure during transients. The applicable ASME Code (1971 Edition) is also referenced when discussing MSSV design flow rates and setpoints. The bases also contain a discussion on thermal power limitations based on the total number of safety valves inoperable.

ComEd proposed to add new information to the bases section. One addition states that the requirement to set the steam line safety valves to within ±1 percent of the appropriate setpoint is consistent with Section III of the ASME Code. The change also states that the allowed operating tolerance of ±3 percent is supported by the Commonwealth Edison Company, Byron/Braidwood Unit 1 & 2 Overpressure Protection Report. The equation that provides the bases for the reactor trip setpoint reductions is revised. The changes to the Bases are proposed to address and support the changes in TS 3.7.1.1 and Tables 3.7-1 and 3.7-2.

b. <u>Impact of Changes</u>

No component or system operating characteristics will be affected by these revisions. The proposed setpoints in Table 3.7-1 are more limiting than those currently allowed in Specification 3.7.1.1. Westinghouse has determined that the current setpoints are non-conservative for some combinations of reduced MSSV availability and reactor power levels. The reactor trip settings were calculated using a revised methodology to account for the non-linear relationship of reactor trip setpoints and reduced MSSV availability. The revised equation, as proposed in the Bases, is used to calculate the reduced reactor trip setpoints. By reducing the setpoints, the original design margin of safety is maintained.

The staff has found the revisions to Bases 3/4.7.1.1 to be acceptable. Increasing the as-found valve setpoint tolerance from ±1 percent to ±3 percent does not have a significant impact on any accident. The peak primary and secondary pressures remain below 110 percent of design at all times. The MSSVs are actuated after accident initiation to protect the secondary systems from overpressuriztion. Increasing the as-found setpoint tolerance will not result in any hardware modification to the MSSVs. Therefore, there is not an increase in the probability of the spurious opening of an MSSV. Sufficient margin exists between the normal steam system operating pressure and the valve setpoint with the increased tolerance to preclude an increase in the probability of inadvertently actuating the valves.

3.0 <u>SUMMARY</u>

The impacts of the proposed TS changes have been revised by the staff as discussed in Section 2. The proposed changes are acceptable.

4.0 <u>STATE CONSULTATION</u>

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (62 FR 11486). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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