March 20, 1987

Docket Nos. STN 50-454
and STN 50-455

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Mr. Dennis L. Farrar
Director of Nuclear Licensing
Post Office Box 767
Chicago, Illinois 60690

Dear Mr. Farrar:

The Commission has issued the enclosed Amendment Nos. 6 to Facility Operating License Nos. NPF-37 and NPF-66 for the Byron Station, Unit Nos. 1 and 2, respectively. The amendments consist of changes to the Technical Specifications in response to your application dated November 26, 1985, supplemented June 16, 1986 and November 24, 1986.

This amendment revises Section 3/4.5.2 to allow closure of either of the Residual Heat Removal Pump discharge to Reactor Coolant System (RCS) cold leg isolation valves (SI8809A and SI8809B), along with the Safety Injection Pump discharge to RCS cold leg valve (SI8835), in order to perform certain check valve back leakage testing.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular bi-weekly Federal Register notice.

Sincerely,

Leonard N. Olshan, Project Manager
Project Directorate #3
Division of PWR Licensing-A

Enclosures:
1. Amendment No. 6 to NPF-37
2. Amendment No. 6 to NPF-66
3. Safety Evaluation

cc: w/enclosures
See next page

No legal objection, With changes noted

PD#3	PD#3	OGC	PD#3
CVogan	LOlshan	SHLewis	SVarga
2/26/87	2/26/87	3/6/87	3/20/87
3-19-87			

Mr. Dennis L. Farrar
Commonwealth Edison Company

Byron Station
Units 1 and 2

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-454

BYRON STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 6
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 November 26, 1985, supplemented June 16, 1986 and November 24, 1986, 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. 6, and the Environment 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

Steven A. Varga

Steven A. Varga, Director
Project Directorate #3
Division of PWR Licensing-A

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 20, 1987

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-455

BYRON STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 6
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 November 26, 1985, supplemented June 16, 1986 and November 24, 1986, 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-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. 6, and revised by Attachment 2 to NPF-60, 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

Steven A. Varga, Director
Project Directorate #3
Division of PWR Licensing-A

Attachment:
Changes to the Technical Specifications

Date of Issuance: March 20, 1987

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

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 6 FACILITY OPERATING LICENSE NOS. NPF-37 and NPF-66

DOCKET NOS. STN-50-454 AND STN 50-455

Revise Appendix A as follows:

Remove Pages	Insert Pages
3/4 5-3	3/4 5-3
3/4 5-4	3/4 5-4
B 3/4 5-1*	B 3/4 5-1*
B 3/4 5-2	B 3/4 5-2

* Overleaf page added for convenience

3/4.5.2 ECCS SUBSYSTEMS - $T_{avg} \geq 350°F$

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent Emergency Core Cooling System (ECCS) subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE centrifugal charging pump,

b. One OPERABLE Safety Injection pump,

c. One OPERABLE RHR heat exchanger,

d. One OPERABLE RHR pump, and

e. An OPERABLE flow path* capable of taking suction from the refueling water storage tank on a Safety Injection signal and automatic opening of the containment sump suction valves.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem 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. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected Safety Injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

*During MODE 3, the discharge paths of both Safety Injection pumps may be isolated by closing SI 8835 and a portion of the discharge paths of both RHR pumps may be isolated by closing either SI8809A or SI8809B for a period of up to 2 hours to perform surveillance testing as required by Specification 4.4.6.2.2. When either SI8809A or SI8809B is closed and pressurizer pressure is below 1000 psig, the accumulators shall be OPERABLE with their isolation valves either closed, but energized, or open.

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the following valves are in the indicated positions with power to the valve operators removed:

Valve Number	Valve Function	Valve Position
MOV SI8806	Suction to the SI Pumps	Open
MOV SI8835	SI Pump Discharge To RCS Cold Legs	Open*
MOV SI8813	SI Pump Recirculation To The RWST	Open
MOV SI8809A	RHR Pump Discharge to RCS Cold Legs	Open*
MOV SI8809B	RHR Pump Discharge to RCS Cold Legs	Open*
MOV SI8840	RHR Pump Discharge to RCS Hot Legs	Closed
MOV SI8802A	SI Pump Discharge to RCS Hot Legs	Closed
MOV SI8802B	SI Pump Discharge to RCS Hot Legs	Closed

b. At least once per 31 days by:

1) Verifying that the ECCS piping is full of water by venting the ECCS pump casings and accessible discharge piping high points, and

2) Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

c. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump and cause restriction of the pump suctions during LOCA conditions. This visual inspection shall be performed:

*Valves may be realigned for testing pursuant to Specification 4.4.6.2.2.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

BASES

3/4.5.1 ACCUMULATORS

The OPERABILITY of each Reactor Coolant System (RCS) accumulator ensures that a sufficient volume of borated water will be immediately forced into the core through each of the cold legs in the event the RCS pressure falls below the pressure of the accumulators. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on accumulator volume, boron concentration and pressure ensure that the assumptions used for accumulator injection in the safety analysis are met. A contained borated water level between 31% and 63% ensures a volume of greater than or equal to 6995 gallons but less than or equal to 7217 gallons.

The accumulator power operated isolation valves are considered to be "operating bypasses" in the context of IEEE Std. 279-1971, which requires that bypasses of a protective function be removed automatically whenever permissive conditions are not met. In addition, as these accumulator isolation valves fail to meet single failure criteria, removal of power to the valves is required.

The limits for operation with an accumulator inoperable for any reason except an isolation valve closed minimizes the time exposure of the plant to a LOCA event occurring concurrent with failure of an additional accumulator which may result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be immediately opened, the full capability of one accumulator is not available and prompt action is required to place the reactor in a mode where this capability is not required.

The requirement to verify accumulator isolation valves shut with power removed from the valve operator when the pressurizer is solid ensures the accumulators will not inject water and cause a pressure transient when the Reactor Coolant System is on solid plant pressure control.

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of two independent ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the accumulators is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long-term core cooling capability in the recirculation mode during the accident recovery period.

With the RCS temperature below 350°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

ECCS SUBSYSTEMS (Continued)

The limitation for a maximum of one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps and Safety Injection pumps except the required OPERABLE charging pump to be inoperable below 330°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensures that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. Surveillance Requirements for throttle valve position stops and flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses. The Surveillance Requirements for leakage testing of ECCS check valves ensures that a failure of one valve will not cause an intersystem LOCA. In Mode 3, with pressurizer pressure below 1000 psig, the accumulators will be available with their isolation valves either closed but energized, or open, whenever a SI8809 valve is closed to perform check valve leakage testing.

3/4.5.4 REFUELING WATER STORAGE TANK

The OPERABILITY of the refueling water storage tank (RWST) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWST minimum volume and boron concentration ensure that: (1) sufficient water is available within containment to permit recirculation cooling flow to the core, and (2) the reactor will remain subcritical in the cold condition following mixing of the RWST and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics. A minimum contained borated water level of 89% ensures a volume of greater than or equal to 395,000 gallons.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 8.5 and 11.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 6 TO FACILITY OPERATING LICENSE NO. ~~DPR-39~~ NPF-37

AND AMENDMENT NO. 6 TO FACILITY OPERATING LICENSE NO. ~~DPR-48~~ NPF-66

COMMONWEALTH EDISON COMPANY

BYRON STATION, UNITS 1 AND 2

DOCKET NOS. STN 50-454 AND STN-50-455

INTRODUCTION

In a letter dated November 26, 1985, the licensee requested a change to Technical Specification (TS) 3/4 5.2 to allow certain valves in the safety injection (SI)/residual heat removal (RHR) systems to be temporarily closed during check valve leakage tests required in T.S. 4.4.6.2.2 Specifically T.S. 4.4.6.2.2 requires check valves SI 8818 A, B, C & D, located in the cold leg injection lines, to be checked for back leakage into the RHR system during Mode 3 at certain intervals, or after a valve malfunction or other unusual occurrence. The requested change would allow closure of either RHR injection valve SI 8809 A or B in order to perform leakage tests for the above check valves, since performance of these tests with the corresponding RHR injection valve in the open position could result in false readings, could significantly increase the time required to perform this surveillance, and could make it more difficult to determine which of several check valves is leaking.

Closure of one RHR injection valve (SI 8809 A or B) isolates injection flow from the RHR pumps into two RCS cold legs. The FSAR accident analysis assumes injection flow into all 4 cold legs during a large break LOCA. The licensee submitted additional analyses which would indicate that sufficient flow from the accumulators, charging, SI & RHR pumps would be available with the proposed system line-up to maintain the downcomer level completely full for the large break LOCA, and therefore little or no penalty in calculated peak clad temperature (PCT) would occur.

In a conference call of January 7, 1986, the staff requested that the licensee provide assurance that the accumulators would be available during performance of these tests, since the accumulators would normally be isolated in the pressure range in which the tests would be performed (800 to 1000 psi). In a conference call of March 11, 1986, the licensee agreed to this but further informed the staff that the common SI pump discharge line isolation valve

SI-8835 also has to be closed during performance of these tests. Thus, the SI pumps would not be available in the event of a LOCA during check valve leak test performance and the large break LOCA analysis in the November 26, 1985 letter was therefore invalid.

In a letter dated June 16, 1986, the licensee submitted a modified amendment, including a new LOCA analysis which reflected the unavailability of the SI pumps due to valve SI-8835 closure during the leak tests. It was assumed that the accumulators would be available with their isolation valves either open or closed with the valve motors energized. In this condition, the accumulators would be automatically aligned to the RCS in the event of an SI signal. The licensee further indicated that the 3.4 psig containment high-1 SI actuation setpoint would be exceeded during a large break LOCA in mode 3, thus automatically actuating SI, "based on detailed Westinghouse calculations on a similar plant", and that the PCT would be bounded by the FSAR analysis of LOCA at full power.

In subsequent conference calls, the staff requested the following additional information from the licensee: identification and a comparison of Byron with the plant referenced in the Westinghouse calculations; what indications and alarms would alert the operator to manually initiate SI in the event of a small break LOCA for which the containment pressure would not rise high enough to automatically actuate SI; and whether the core would stay covered in the event of a small break LOCA since the RCS pressure would not necessarily decrease sufficently for accumulator and RHR injection and the SI pumps would be isolated. The licensee identified the referenced plant as Millstone 3, which has a slightly smaller containment than Byron.

In a letter dated November 24, 1986 the licensee indicated that the probability of a large break LOCA is lower at shutdown than at operating conditions because of lower temperatures and pressures. The licensee provided the results of estimates for breaks less than 6 inches during mode 3 two hours after shutdown. For breaks up to 3 inches at least 20 minutes would be available to initiate flow from one of the charging pumps, which is expected to limit PCT to less than the design case, although some core uncovery might result. For breaks between 3 & 6 inches operator action to start one charging pump would be required in about 10 minutes. Additional operator action may be required, depending on break size, within one hour after LOCA initiation to start an additional charging or SI pump, or depressurize the RCS using the steam generators and to start an RHR pump. The licensee provided a list of alarms and indications that would alert the operator regarding occurrence of a small break LOCA. The licensee also indicated that the check valve surveillance test would probably be performed during startup after a refueling outage or after the plant has been in Mode 5 for longer than 3 days. Thus, the decay heat would be lower than assumed in the above estimates. The licensee also informally proposed the following footnote to the accumulator TS: "When either SI 8809 A or SI 8809 B is closed and pressurizer pressure is below 1000 psig, the accumulators shall be OPERABLE with their isolation valves either closed but energized or open."

In the "Significant Hazards Consideration" the licensee indicated that implementation of the proposed TS changes would result in an overall increase in the margin of safety since it would provide increased assurance of the integrity of the RHR injection check valves, thus reducing the probability of an intersystem LOCA. The licensee has also provided reasonable assurance that a LOCA occurring during these check valve leakage tests can be mitigated. The staff concurs with the licensee's conclusions and finds the proposed T.S. changes, namely closure of SI 8835 and either SI 8809 A or B during leak checks of SI 8818 A, B, C & D, acceptable providing the above footnote to the accumulator T.S. is also added and procedures are available to mitigate the LOCA which may occur while the valves are closed.

The licensee's submittal of November 24, 1986 was made as a result of NRC staff request to clarify the original submittal dated November 26, 1985, supplemented June 16, 1986.

ENVIRONMENTAL CONSIDERATION

These amendments involve a change in the installation or use of the facilities components located within the restricted areas as defined in 10 CFR 20. The staff has determined that these amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that these amendments involve no significant hazards consideration and there has been no public comment on such finding. Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR Sec 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 these amendments.

CONCLUSION

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

Dated: March 20, 1987

PRINCIPAL CONTRIBUTORS:

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L. Olshan