October 15, 1991

Docket No. 50-412
Serial No. BV-91-039

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Mr. J. D. Sieber, Vice President
Nuclear Group
Duquesne Light Company
Post Office Box 4
Shippingport, Pennsylvania 15077

Dear Mr. Sieber:

SUBJECT: ISSUANCE OF AMENDMENT (TAC NO. M76478)

The Commission has issued the enclosed Amendment No. 39 to Facility Operating License No. NPF-73 for the Beaver Valley Power Station, Unit 2, in response to your application dated April 2, 1990, Change Request No. 34).

The amendment modifies the Appendix A Technical Specifications for the pressurizer safety valves. Specifically, the amendment adds additional actions to be taken if a pressurizer safety valve has discharged liquid water due to an overpressure event.

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

Sincerely,

/s/

Albert W. De Agazio, Sr. Project Manager
Project Directorate I-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 39 to NPF-73
2. Safety Evaluation

cc w/enclosures:
See next page

OFC	PDI-4:LA	PDI-4:PM	PDI-4:D	OGC
NAME	SNorris	ADe Agazio/Bah	JStolz	
DATE	9/11/91	9/26/91	9/26/91	10/4/91

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Document Name: AMEND 76478

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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-73 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 39 , and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto are hereby incorporated in the license. DLCO 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, to be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Director
Project Directorate I-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 15, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 39

FACILITY OPERATING LICENSE NO. NPF-73

DOCKET NO. 50-412

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages as indicated. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

Remove	Insert
3/4 4-8	3/4 4-8
3/4 4-9	3/4 4-9
B 3/4 4-2	B 3/4 4-2

Mr. J. D. Sieber
Duquesne Light Company
cc:

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Beaver Valley Power Station
Units 1 & 2

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

DUQUESNE LIGHT COMPANY

OHIO EDISON COMPANY

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

THE TOLEDO EDISON COMPANY

DOCKET NO. 50-412

BEAVER VALLEY POWER STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 39
License No. NPF-73

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

   A. The application for amendment by Duquesne Light Company, et al. (the licensee) dated April 2, 1990, 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.

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REACTOR COOLANT PUMP-STARTUP

LIMITING CONDITION FOR OPERATION

3.4.1.6 An idle reactor coolant pump in a non-isolated loop shall not be started, unless the secondary water temperature* of each steam generator is less than 50°F above each of the inservice RCS cold leg temperatures.

APPLICABILITY: When the temperature of one or more of the non-isolated loop cold legs is ≤ 350°F.

ACTION:

With the temperature of the steam generator in the loop associated with the reactor coolant pump being started greater than 50°F above the cold leg temperature of the other non-isolated loops, suspend the startup of the reactor coolant pump.

SURVEILLANCE REQUIREMENTS

4.4.1.6.1 The secondary water temperature of the non-isolated steam generators shall be determined within 10 minutes prior to starting a reactor coolant pump.

*The secondary water temperature is to be verified by direct measurement of the fluid temperature, or contact temperature readings on the steam generator secondary, or blowdown piping after purging of stagnant water within the piping.

3/4.4.2 SAFETY VALVES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2 A minimum of one pressurizer code safety valve shall be OPERABLE with a lift setting* of 2485 psig + 1% - 3%.**

APPLICABILITY: MODES 4 and 5

ACTION:

a. With no pressurizer code safety valve OPERABLE, immediately suspend all operations involving positive reactivity changes and place an OPERABLE RHR loop into operation in the shutdown cooling mode.

b. After any pressurizer code safety lift, as indicated by the safety valve position indicator, involving loop seal or water discharge; borate to a SHUTDOWN MARGIN equivalent to at least 1% delta K/K at 200 deg F within the next 24 hours. Inspect the valve for potential damage, initiate corrective action to return the valve to OPERABLE status prior to increasing RCS temperature and document the inspection results in the Annual Report pursuant to Specification 6.9.1.5.b.

SURVEILLANCE REQUIREMENTS

4.4.2 No additional requirements other than those required by Specification 4.0.5.

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

**Within ± 1% following pressurizer code safety valve testing.

3/4.4.3 SAFETY VALVES - OPERATING

LIMITING CONDITION FOR OPERATION

3.4.3 All pressurizer code safety valves shall be OPERABLE with a lift setting* of 2485 psig + 1% - 3%.**

APPLICABILITY: MODES 1, 2, and 3

ACTION:

a. With one pressurizer code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in HOT SHUTDOWN within 12 hours.

b. After any pressurizer code safety valve lift, as indicated by the safety valve position indicator, involving loop seal or water discharge; be in at least HOT STANDBY within the next 6 hours, and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.3 No additional requirements other than those required by Specification 4.0.5.

*The lift setting shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.
**Within ± 1% following pressurizer code safety valve testing.

3/4.4.4 PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.4 The pressurizer shall be OPERABLE with at least 150 kW of pressurizer heaters and with a steam bubble.

APPLICABILITY: MODES 1, 2, and 3

ACTION:

With the pressurizer inoperable due to less than 150 kW of heaters supplied by an emergency bus, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 12 hours. With the pressurizer otherwise inoperable, be in at least HOT STANDBY with the reactor trip breakers open within 6 hours and in the HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.4.1 The power supply for the pressurizer heaters shall be demonstrated OPERABLE at least once per 18 months by energizing the heaters supplied by the emergency bus.

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 1.30 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, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, due to the initial conditions assumed in the analysis for the control rod bank withdrawal from a subcritical condition, two operating coolant loops are required to meet the DNB design basis for this Condition II event when the rod control system is capable of control bank rod withdrawal.

In MODES 4 and 5, a single reactor coolant loop or RHR subsystem provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops be OPERABLE. Thus, if the reactor coolant loops are not OPERABLE, this specification requires two RHR loops to be OPERABLE.

The operation of one Reactor Coolant Pump 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 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.

3/4.4.2 and 3/4.4.3 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 345,000 lbs. per hour of saturated steam at the valve set point. 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.

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 until the first Reactor Protective System trip set point is reached (i.e., no credit is taken for a direct reactor trip on the loss of load) 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.

Safety valves similar to the pressurizer code safety valves were tested under an Electric Power Research Institute (EPRI) program to determine if the valves would operate stably under feedwater line break accident conditions. The test results indicated the need for inspection and maintenance of the safety valves to determine the potential damage that may have occurred after a safety valve has lifted and either discharged the loop seal or discharged water through the valve. Additional action statements require safety valve inspection to determine the extent of the corrective actions required to ensure the valves will be capable of performing their intended function in the future.

3/4.4.4 PRESSURIZER

The requirement that 150 kw of pressurizer heaters and their associated controls and emergency bus provides assurance that these heaters can be energized during a loss of offsite power condition to maintain natural circulation at HOT STANDBY.

3/4.4.5 STEAM GENERATORS

One OPERABLE steam generator in a non-isolated reactor coolant loop provides sufficient heat removal capability to remove decay heat after a reactor shutdown. The requirement for two OPERABLE steam generators, combined with other requirements of the Limiting Conditions for Operation ensures adequate

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 39 TO FACILITY OPERATING LICENSE NO. NPF-73

DUQUESNE LIGHT COMPANY

OHIO EDISON COMPANY

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

THE TOLEDO EDISON COMPANY

BEAVER VALLEY POWER STATION, UNIT NO. 2

DOCKET NO. 50-412

1.0 INTRODUCTION

By letter dated April 2, 1991, Duquesne Light Company (DLC/the licensee) proposed a revision to the Limiting Conditions for Operation applicable to the Reactor Coolant System as specified in the Beaver Valley Power Station, Unit 2 (BVPS-2) Technical Specifications (TSs), Sections 3.4.2 and 3.4.3. The proposed revision would add additional actions to be taken in the event a pressurizer code safety lift involving the discharge of water or the loop seal.

2.0 DISCUSSION AND EVALUATION

On September 13, 1989, the NRC documented the completion of the review of the open issues associated with NUREG-0737, Item II.D.1, Relief and Safety Valve Testing. The staff concluded that most of the requirements of Item II.D.1 were met, and DLC could proceed to complete the remaining actions without further NRC review. Our letter of September 13, 1989, also provided our consultant's Technical Evaluation Report (TER) (EGG-NTA-8480) documenting the results of the EG&G Idaho, Inc., review.

One of the open actions identified in the TER related to the reliability of the BVPS-2 safety valves following a lift involving discharge of water or the loop seal. The safety valve testing program results suggested that valve inspection and maintenance are important to the continued operability of the safety valves. The TER noted that in certain tests, the safety valve reopened after closure and chattered before the test was terminated. After refurbishment and/or replacement of the galled surfaces and damaged parts, the test valve would first perform satisfactory, but closing chatter would recur in subsequent tests. It was hypothesized that the closing chatter was the result of repeated valve actuation in loop seal and water discharge tests. Because of this, it was recommended that the safety valves should be inspected after each lift involving loop seal or water discharge, and that the inspection requirement should be incorporated into plant procedures or the Technical Specifications.

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In response to this recommendation, DLC has proposed to incorporate pressurizer safety valve inspection and maintenance requirements in the BVPS-2 Technical Specifications. The proposed amendment would require cold shutdown of the unit following any pressurizer safety valve lift that involved loop seal or water discharge. In addition, inspection of the safety valve and necessary corrective action to return the valve to operable status would be required prior to increasing reactor coolant system temperature. These requirements are intended to improve the reliability of the safety valves and are identical to the requirement incorporated into the Beaver Valley Power Station, Unit 1 Technical Specifications by Amendment 115 (September 8, 1987).

The staff has reviewed the proposed changes and concludes that the changes will improve the reliability of the pressurizer safety valves by eliminating or reducing the potential for safety valve chatter upon valve closure. Therefore, the staff finds the proposed changes acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, good faith attempts were made to contact the Pennsylvania State official regarding proposed issuance of the amendment.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves 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 amendment involves no significant hazards consideration, and there has been no public comment on such finding (55 FR 21969). Accordingly, the amendment meets 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 amendment.

5.0 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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: Albert W. De Agazio

Date: October 15, 1991