50-334/412

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

December 10, 1997

Mr. J. E. Cross
President-Generation Group
Duquesne Light Company
Post Office Box 4
Shippingport, PA 15077

SUBJECT: BEAVER VALLEY POWER STATION, UNIT NOS. 1 AND 2 (TAC NOS. M98222 AND M98223)

Dear Mr. Cross:

The Commission has issued the enclosed Amendment No. 209 to Facility Operating License No. DPR-66 and Amendment No. 87 to Facility Operating License No. NPF-73 for the Beaver Valley Power Station, Unit Nos. 1 and 2 (BVPS-1 and BVPS-2). These amendments consist of changes to the Technical Specifications (TSs) in response to your application dated March 14, 1997, as supplemented July 29, 1997, and August 13, 1997, which submitted Proposed Operating License Change Request Nos. 1A-236 and 2A-110.

These amendments relocate certain administrative control TSs from the BVPS-1 and BVPS-2 TSs to the licensee's operational quality assurance program description, which is presented in Section 17.2 of the BVPS-2 Updated Final Safety Analysis Report (UFSAR). Section 17.2 of the BVPS-2 UFSAR contains the quality assurance program description for both BVPS-1 and BVPS-2. The following TSs are being relocated to the quality assurance program description.

BVPS-2 TS 6.2.3 (Independent Safety Evaluation Group)

BVPS-1 and BVPS-2 TS 6.5.1 (Onsite Safety Committee)

BVPS-1 and BVPS-2 TS 6.5.2 (Offsite Review Committee)

BVPS-1 and BVPS-2 TS 6.8.2 (Procedures, Review and Approval)

BVPS-1 and BVPS-2 TS 6.8.3 (Temporary Procedure Changes, Review and Approval)

BVPS-1 and BVPS-2 TS 6.10.1 (Records Retention, At least 5 Years)

BVPS-1 and BVPS-2 TS 6.10.2 (Records Retention, Duration of Operating License)

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

Sincerely,

Donald S. Brinkman

Donald S. Brinkman, Senior Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-334 and 50-412

Enclosures:
1. Amendment No. 209 to License No. DPR-66
2. Amendment No. 87 to License No. NPF-73
3. Safety Evaluation

cc w/encls: See next page

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

Sincerely,

/s/

Donald S. Brinkman, Senior Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-334 and 50-412

Enclosures:
1. Amendment No. 209 to License No. DPR-66
2. Amendment No. 87 to License No. NPF-73
3. Safety Evaluation

cc w/encls: See next page

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J. E. Cross
Duquesne Light Company

Beaver Valley Power Station
Units 1 & 2

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

DUQUESNE LIGHT COMPANY

OHIO EDISON COMPANY

PENNSYLVANIA POWER COMPANY

DOCKET NO. 50-334

BEAVER VALLEY POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 209
License No. DPR-66

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

   A. The application for amendment by Duquesne Light Company, et al. (the licensee) dated March 14, 1997, as supplemented July 29, 1997, and August 13, 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. DPR-66 is hereby amended to read as follows:

(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No. 209, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

In addition, the license is amended by changes to paragraph 2.C.(10) to the Facility Operating License No. DPR-66 as follows:

(10) <u>Additional Conditions</u>

The Additional Conditions contained in Appendix C, as revised through Amendment No. 209, are hereby incorporated into this license. Duquesne Light Company shall operate the facility in accordance with the Additional Conditions.

3. This license amendment is effective as of the date of its issuance, to be implemented within 60 days. Implementation of this amendment shall include the relocation of these technical specification requirements to the appropriate documents, as described in the licensee's application dated March 14, 1997, as supplemented July 29 and August 13, 1997, and evaluated in the staff's safety evaluation attached to this amendment.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Director
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachments: 1. Page 1 to Appendix C of License* DPR-66
2. Changes to the Technical Specifications

Date of Issuance: December 10, 1997

* Page 1 of Appendix C is attached, for convenience, for the composite license to reflect this change.

ATTACHMENT TO LICENSE AMENDMENT NO. 209

FACILITY OPERATING LICENSE NO. DPR-66

DOCKET NO. 50-334

1. Revise Appendix C of the License as follows:

Remove Page	Insert Page
1	1

2. Replace the following pages of 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
XIV	XIV
XV	XV
XVI	XVI
XVII	XVII
XVIII	XVIII
XIX	XIX
XX	---
3/4 7-30	3/4 7-30
6-5	6-5
6-6	6-6
6-7	6-7
6-8	---
6-9	---
6-10	---
6-11	---
6-12	---
6-13	---
6-21	6-21
6-22	---
6-24	6-24
6-25	6-25

APPENDIX C

ADDITIONAL CONDITIONS
OPERATING LICENSE NO. DPR-66

Duquesne Light Company, Ohio Edison Company, and Pennsylvania Power Company shall comply with the following conditions on the schedules noted below:

Amendment Number	Additional Condition	Implementation Date
202	The licensee is authorized to relocate certain Technical Specification requirements to licensee-controlled documents. Implementation of this amendment shall include the relocation of these technical specification requirements to the appropriate documents, as described in the licensee's application dated September 9, 1996, and evaluated in the staff's safety evaluation attached to this amendment.	The amendment shall be implemented within 60 days from April 14, 1997
208	The licensee commits to perform the post weld heat treatment of sleeve welds and the NRC-recommended inspections for repaired tubes as described in the licensee's application dated March 10, 1997, as supplemented July 28 and September 17, 1997, and evaluated in the staff's safety evaluation attached to this amendment.	The amendment shall be implemented within 60 days from November 25, 1997
209	The licensee is authorized to relocate certain Technical Specification requirements to licensee-controlled documents. Implementation of this amendment shall include the relocation of these technical specification requirements to the appropriate documents, as described in the licensee's application dated March 14, 1997, as supplemented July 29 and August 13, 1997, and evaluated in the staff's safety evaluation attached to this amendment.	The amendment shall be implemented within 60 days from December 10, 1997

INDEX

BASES

DESIGN FEATURES

ADMINISTRATIVE CONTROLS

INDEX

ADMINISTRATIVE CONTROLS

TABLE INDEX

Figure Index

g. Snubber Service Life Monitoring*

The service life of hydraulic and mechanical snubbers shall be monitored to ensure that the service life is not exceeded between surveillance inspections. The maximum expected service life for various seals, springs, and other critical parts shall be determined and established based on engineering information and may be extended or shortened based on monitored test results and failure history. Critical parts shall be replaced so that the maximum service life will not be exceeded during a period when the snubber is required to be OPERABLE. The parts replacements shall be documented and the documentation shall be retained in accordance with the applicable record retention provision of the quality assurance program description referenced in the Updated Final Safety Analysis Report.

* For purposes of establishing a baseline for the determination of service life monitoring, this program will be implemented over 3 successive refueling periods.

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility and Radiation Protection staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for the Health Physics Manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975, and the technical advisory engineering representative who shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design and response analysis of the plant for transients and accidents.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the facility staff shall be maintained under the direction of the Nuclear Training Manager and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and 10 CFR Part 55.

6.5 DELETED

6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified in accordance with 10 CFR 50.72 and/or a report be submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and

b. Each REPORTABLE EVENT shall be reviewed by the OSC, and results of this review shall be submitted to the ORC.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The facility shall be placed in at least HOT STANDBY within one (1) hour.

b. The Safety Limit violation shall be reported to the Commission within one hour and to the General Manager, Nuclear Operations and to the ORC within 24 hours.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the On-Site Safety Committee (OSC). This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

SAFETY LIMIT VIOLATION (Continued)

d. The Safety Limit Violation Report shall be submitted to the Commission, the ORC and the General Manager, Nuclear Operations within 30 days of the violation.

6.8 PROCEDURES

6.8.1 Written procedures shall be established, implemented, and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, February 1978.

b. Refueling operations.

c. Surveillance and test activities of safety related equipment.

d. Not used.

e. Not used.

f. Fire Protection Program implementation.

g. PROCESS CONTROL PROGRAM implementation.

h. OFFSITE DOSE CALCULATION MANUAL implementation.

6.8.2 Deleted

6.8.3 Deleted

6.8.4 A Post-Accident monitoring program shall be established, implemented, and maintained:

A program which will provide the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples following an accident. The program shall include the following:

(i) Training of personnel,
(ii) Procedures for sampling and analysis, and
(iii) Provisions for maintenance of sampling and analysis equipment.

PROCEDURES (Continued)

6.8.5 A program for monitoring of secondary water chemistry to inhibit steam generator tube degradation shall be implemented. This program shall be described in the station chemistry manual and shall include:

a. Identification of a sampling schedule for the critical parameters and control points for these parameters;

b. Identification of the procedures used to measure the values of the critical parameters;

c. Identification for process sampling points;

d. Procedures for the recording and management of data;

e. Procedures defining corrective actions for off control point chemistry conditions; and

f. A procedure identifying:

   1) the authority responsible for the interpretation of the data, and
   2) the sequence and timing of administrative events required to initiate corrective action.

(Next Page is 6-14)

SPECIAL REPORTS (Continued)

f. Miscellaneous reporting requirements specified in the Action Statements for Appendix C of the ODCM.

g. DELETED

h. Steam Generator Tube Inservice Inspection Results Report, Specification 4.4.5.5.

i. Liquid Hold Up Tanks, Specification 3.11.1.4.

j. Gas Storage Tanks, Specification 3.11.2.5.

k. Explosive Gas Monitoring Instrumentation, Specification 3.3.3.11.

6.10 DELETED

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

(Next Page is 6-23)

6.13 PROCESS CONTROL PROGRAM (PCP)

Changes to the PCP:

a. Shall be documented and records of reviews performed shall be retained in accordance with the applicable record retention provision of the quality assurance program description referenced in the Updated Final Safety Analysis Report. This documentation shall contain:

   1) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

   2) A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.

b. Shall become effective after review and acceptance by the OSC and the approval of the General Manager Nuclear Operations, predesignated alternate or a predesignated Manager to whom the General Manager Nuclear Operations has assigned in writing the responsibility for review and approval of specific subjects.

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

Changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained in accordance with the applicable record retention provision of the quality assurance program description referenced in the Updated Final Safety Analysis Report. This documentation shall contain:

   1) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

   2) A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.1302, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

b. Shall become effective after review and acceptance by the OSC and the approval of the General Manager Nuclear Operations, predesignated alternate or a predesignated Manager to whom the General Manager Nuclear Operations has

OFFSITE DOSE CALCULATION MANUAL (ODCM) (Continued)

assigned in writing the responsibility for review and approval of specific subjects.

c. Shall be submitted to the Commission in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Annual Radioactive Effluent Release Report for the period of the report in which any change to the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (e.g., month/year) the change was implemented.

6.16 Moved to the PROCESS CONTROL PROGRAM.

6.17 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions(1). This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 40.0 psig.

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.10% of containment air weight per day.

Leakage Rate acceptance criteria are:

a. Containment leakage rate acceptance criterion is $\leq$ 1.0 L_a for the overall Type A leakage test and < 0.60 L_a for the Type B and Type C tests on a minimum pathway leakage rate (MNPLR) basis. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are < 0.60 L_a on a maximum pathway leakage rate (MXPLR)(2) basis for Type B and Type C tests and < 0.75 L_a for Type A tests.

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(1) Exemptions to Appendix J of 10 CFR 50 dated November 19, 1984, December 5, 1984, and July 26, 1995.

(2) For penetrations which are isolated by use of a closed valve(s), blind flange(s), or de-activated automatic valve(s), the MXPLR of the isolated penetration is assumed to be the measured leakage through the isolation device(s).

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

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. 87
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 March 14, 1997, as supplemented July 29, 1997, and August 13, 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-73 is hereby amended to read as follows:

   (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 in the license. DLCO shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

   In addition, the license is amended by changes to paragraph 2.C.(11) to the Facility Operating License No. NPF-73 as follows:

   (11) Additional Conditions

   The Additional Conditions contained in Appendix D, as revised through Amendment No. 87 , are hereby incorporated into this license. Duquesne Light Company shall operate the facility in accordance with the Additional Conditions.

3. This license amendment is effective as of its date of issuance, to be implemented within 60 days. Implementation of this amendment shall include the relocation of these technical specification requirements to the appropriate documents, as described in the licensee's application dated March 14, 1997, as supplemented July 29 and August 13, 1997, and evaluated in the staff's safety evaluation attached to this amendment.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Director
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachments: 1. Page 1 to Appendix D of License* No. NPF-73
2. Changes to the Technical Specifications

Date of Issuance: December 10, 1997

* Page 1 of Appendix D is attached, for convenience, for the composite license to reflect this change.

ATTACHMENT TO LICENSE AMENDMENT NO. 87

FACILITY OPERATING LICENSE NO. NPF-73

DOCKET NO. 50-412

1. Revise Appendix D of the License as follows:

Remove Page	Insert Page
1	1

2. Replace the following pages of 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
XIV	XIV
XV	XV
XVI	---
3/4 7-27	3/4 7-27
6-2	6-2
6-3	---
6-6	6-6
6-7	6-7
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6-21	6-21
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6-23	---
6-24	6-24
6-25	6-25

APPENDIX D

ADDITIONAL CONDITIONS
OPERATING LICENSE NO. NPF-73

Duquesne Light Company, Ohio Edison Company, The Cleveland Electric Illuminating Company, and The Toledo Edison Company shall comply with the following conditions on the schedules noted below:

Amendment Number	Additional Condition	Implementation Date
83	The licensee is authorized to1291Xrelocatecertain Technical Specification requirements to licensee-controlled documents. Implementation of this amendment shall include the relocation of these technical specification requirements to the appropriate documents, as described in the licensee's application dated September 9, 1996, and evaluated in the staff's safety evaluation attached to this amendment.	The amendment shall be implemented within 60 days from April 14, 1997
87	The licensee is authorized to relocate certain Technical Specification requirements to licensee-controlled documents. Implementation of this amendment shall include the relocation of these technical specification requirements to the appropriate documents, as described in the licensee's application dated March 14, 1997, as supplemented July 29 and August 13, 1997, and evaluated in the staff's safety evaluation attached to this amendment.	The amendment shall be implemented within 60 days from December 10, 1997

INDEX

ADMINISTRATIVE CONTROLS

INDEX

ADMINISTRATIVE CONTROLS

e. Hydraulic Snubbers Functional Test Acceptance Criteria

The hydraulic snubber functional test shall verify that:

1. Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression.

2. Snubber bleed, or release rate, where required, is within the specified range in compression or tension. For snubbers specifically required to not displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified.

f. Mechanical Snubbers Functional Test Acceptance Criteria

The mechanical snubber functional test shall verify that:

1. The force that initiates free movement of the snubber rod in either tension or compression is less than the specified maximum drag force.

2. Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression.

3. Snubber release rate, where required, is within the specified range in compression or tension. For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified.

g. Service Life Monitoring

The service life of hydraulic and mechanical snubbers shall be monitored to ensure that the service life is not exceeded between surveillance inspections. The maximum expected service life for various seals, springs, and other critical parts shall be determined and established based on engineering information and may be extended or shortened based on monitored test results and failure history. Critical parts shall be replaced so that the maximum service life will not be exceeded during a period when the snubber is required to be OPERABLE. The parts replacements shall be documented and the documentation shall be retained in accordance with the applicable record retention provision of the quality assurance program description included in the Updated Final Safety Analysis Report. Service life will be defined to commence at plant startup subsequent to initial fuel load.

UNIT STAFF (Continued)

c. At least two licensed Operators shall be in the control room during reactor start-up, scheduled reactor shutdown and during recovery from reactor trips.

d. An individual qualified in radiation protection procedures shall be onsite when fuel is in the reactor.

e. All CORE ALTERATIONS after the initial fuel loading shall be directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.

f. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety-related functions; senior reactor operators, reactor operators, radiation control technicians, auxiliary operators, meter and control repairman, and all personnel actually performing work on safety related equipment.

The objective shall be to have operating personnel work a normal 8-hour day, 40-hour week while the plant is operating. However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance or major plant modifications, on a temporary basis, the following guidelines shall be followed:

a. An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time.

b. An individual should not be permitted to work more than 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, nor more than 72 hours in any seven day period, all excluding shift turnover time.

c. A break of at least eight hours should be allowed between work periods, including shift turnover time.

d. Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

Any deviation from the above guidelines shall be authorized by the General Manager, Nuclear Operations or predesignated alternate, or higher levels of management. Authorized deviations at the working hour guidelines shall be documented and available for NRC review.

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility and Radiation Protection staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for the Health Physics Manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975, and the technical advisory engineering representative who shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design and response analysis of the plant for transients and accidents.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the facility staff shall be maintained under the direction of the Nuclear Training Manager and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and 10 CFR Part 55.

6.5 DELETED

6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified in accordance with 10 CFR 50.72 and/or a report be submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and

b. Each REPORTABLE EVENT shall be reviewed by the OSC, and the results of this review shall be submitted to the ORC.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The facility shall be placed in at least HOT STANDBY within one (1) hour.

b. The Safety Limit violation shall be reported to the Commission within one hour. The Safety Limit violation shall be reported to the General Manager, Nuclear Operations and to the ORC within 24 hours.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the On-Site Safety Committee (OSC). This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

SAFETY LIMIT VIOLATION (Continued)

d. The Safety Limit Violation Report shall be submitted to the Commission, the ORC and the General Manager, Nuclear Operations within 30 days of the violation.

6.8 PROCEDURES

6.8.1 Written procedures shall be established, implemented, and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, February 1978.

b. Refueling operations.

c. Surveillance and test activities of safety related equipment.

d. Not used.

e. Not used.

f. Fire Protection Program implementation.

g. PROCESS CONTROL PROGRAM implementation.

h. OFFSITE DOSE CALCULATION MANUAL implementation.

6.8.2 Deleted

6.8.3 Deleted

6.8.4 A Post-Accident monitoring program shall be established, implemented, and maintained. The program will provide the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples following an accident. The program shall include the following:

(i) Training of personnel,
(ii) Procedures for sampling and analysis, and
(iii) Provisions for maintenance of sampling and analysis equipment.

(Next page is 6-14)

SPECIAL REPORTS (Continued)

b. Inoperable Seismic Monitoring Instrumentation, Specification 3.3.3.3.

c. Inoperable Meteorological Monitoring Instrumentation, Specification 3.3.3.4.

d. Seismic event analysis, Specification 4.3.3.3.2.

e. Sealed source leakage in excess of limits, Specification 4.7.9.1.3.

f. Miscellaneous reporting requirements specified in the ACTION Statements for Appendix C of the ODCM.

g. DELETED

h. Steam generator tube inservice inspection, Specification 4.4.5.5.

i. Inoperable accident monitoring, Specification 3.3.3.8.

j. Liquid Hold-Up Tanks, Specification 3.11.1.4.

k. Gas Storage Tanks, Specification 3.11.2.5.

l. Explosive Gas Monitoring Instrumentation, Specification 3.3.3.11.

6.10 DELETED

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.1601 of 10 CFR 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring

HIGH RADIATION AREA (Continued)

issuance of a Radiological Work Permit(1) or Radiological Access Control Permit. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them.

c. An individual qualified in radiation protection procedures who is equipped with a radiation dose rate monitoring device. This individual shall be responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by a facility health physics supervisor in the Radiological Work Permit or Radiological Access Control Permit.

6.12.2 The requirements of 6.12.1, above, also apply to each high radiation area in which the intensity of radiation is greater than 1000 mrem/hr. In addition, locked doors shall be provided to prevent unauthorized entry into such areas and the keys shall be maintained under the administrative control of the Shift Supervisor on duty and/or a facility health physics supervisor.

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(1) Health physics personnel, or personnel escorted by health physics personnel in accordance with approved emergency procedures, shall be exempt from the RWP issuance requirement during the performance of their radiation protection duties, provided they comply with approved radiation protection procedures for entry into high radiation areas.

(Next page is 6-24)

6.13 PROCESS CONTROL PROGRAM (PCP)

Changes to the PCP:

a. Shall be documented and records of reviews performed shall be retained in accordance with the applicable record retention provision of the quality assurance program description included in the Updated Final Safety Analysis Report. This documentation shall contain:

   1) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

   2) A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.

b. Shall become effective after review and acceptance by the OSC and the approval of the General Manager Nuclear Operations, predesignated alternate or a predesignated Manager to whom the General Manager Nuclear Operations has assigned in writing the responsibility for review and approval of specific subjects.

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

Changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained in accordance with the applicable record retention provision of the quality assurance program description included in the Updated Final Safety Analysis Report. This documentation shall contain:

   1) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

   2) A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.1302, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

b. Shall become effective after review and acceptance by the OSC and the approval of the General Manager Nuclear Operations, predesignated alternate or a predesignated Manager to whom the General Manager Nuclear Operations has assigned in writing the responsibility for review and approval of specific subjects.

c. Shall be submitted to the Commission in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Annual Radioactive Effluent Release Report for the period of the report in which any change to the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (e.g., month/year) the change was implemented.

6.16 Moved to the PROCESS CONTROL PROGRAM.

6.17 CONTAINMENT LEAKAGE RATE TESTING PROGRAM

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54 (o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions(1). This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995.

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(1) Exemptions to Appendix J of 10 CFR 50, as stated in the operating license.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 209 AND 87 TO FACILITY OPERATING

LICENSE NOS. DPR-66 AND NPF-73

DUQUESNE LIGHT COMPANY

OHIO EDISON COMPANY

PENNSYLVANIA POWER COMPANY

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

THE TOLEDO EDISON COMPANY

BEAVER VALLEY POWER STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-334 AND 50-412

1.0 INTRODUCTION

By letter dated March 14, 1997, as supplemented July 29, 1997, and August 13, 1997, the Duquesne Light Company (the licensee) submitted a request for changes to the Beaver Valley Power Station, Unit Nos. 1 and 2 (BVPS-1 and BVPS-2), Technical Specifications (TSs). The requested changes would relocate certain administrative control TSs from the BVPS-1 and BVPS-2 TSs to the licensee's operational quality assurance program description, which is presented in Section 17.2 of the BVPS-2 Updated Final Safety Analysis Report (UFSAR). Section 17.2 of the BVPS-2 UFSAR contains the quality assurance program description for both BVPS-1 and BVPS-2. The following TSs are being relocated to the quality assurance program description.

BVPS-2 TS 6.2.3 (Independent Safety Evaluation Group)

BVPS-1 and BVPS-2 TS 6.5.1 (Onsite Safety Committee)

BVPS-1 and BVPS-2 TS 6.5.2 (Offsite Review Committee)

BVPS-1 and BVPS-2 TS 6.8.2 (Procedures, Review and Approval)

BVPS-1 and BVPS-2 TS 6.8.3 (Temporary Procedure Changes, Review and Approval)

BVPS-1 and BVPS-2 TS 6.10.1 (Records Retention, At least 5 Years)

BVPS-1 and BVPS-2 TS 6.10.2 (Records Retention, Duration of Operating License)

The July 29, 1997, and August 13, 1997, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the amendment request beyond the scope of the May 7, 1997, Federal Register notice.

2.0 BACKGROUND

Section 182a of the Atomic Energy Act (the "Act") requires applicants for nuclear power plant operating licenses to include TS as part of the license. The Commission's regulatory requirements related to the content of TSs are set forth in 10 CFR 50.36. That regulation requires that the TS include items in five specific categories: (1) safety limits, limiting safety system settings, and limiting safety settings; (2) limiting conditions for operations; (3) surveillance requirements; (4) design features; and (5) administrative controls. The regulation, however, does not specify the particular requirements to be included in the plant TSs.

Section 50.36 provides, with respect to limiting conditions for operations (LCO), four criteria to be used in determining whether particular safety functions are required to be included in the TSs. While the four criteria specifically apply to LCO, in adopting the revision to the rule, the Commission indicated that the intent of these criteria can be utilized to identify the optimum set of administrative controls in the TSs (60 FR 36957). Administrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure safe operation of the facility in a safe manner. The specific content of the administrative controls section of the TSs is, therefore, that information that the Commission deems essential for the safe operation of the facility that is not already covered by regulations. Accordingly, the staff has determined that requirements that are not specifically required under 10 CFR 50.36(c)(5), and which are not otherwise necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety, can be removed from administrative controls. Existing TS requirements, therefore, may be relocated to more appropriate documents (e.g., Security Plan, Quality Assurance Plan, and Emergency Plan) and controlled by the applicable regulatory requirement. Similarly, while the required content of TS administrative controls is specified in 10 CFR 50.36(c)(5), particular details of administrative controls may be relocated to licensee-controlled documents where the provisions of 10 CFR 50.54, 10 CFR 50.59, or other regulations provide adequate regulatory control.

NRC Administrative Letter 95-06 provides guidance to licensees requesting amendments that relocate the administrative controls within its NRC-approved quality assurance program description (QAPD), where subsequent changes are controlled by 10 CFR 50.54(a). Administrative Letter 95-06 provides specific guidance in the areas of (1) Independent Safety Engineering Group (ISEG), (2) Reviews and Audits, (3) Procedure Review Process, and (4) Records and Record Retention.

3.0 EVALUATION

The licensee proposes to relocate, from the TSs to the QAPD, certain administrative controls related to the four areas addressed below. The QAPD is located in Section 17.2 of the BVPS Unit 2 UFSAR. (Section A.2.2 of the BVPS-1 UFSAR references the Unit 2 UFSAR, Section 17.2 for the QAPD.)

Unless otherwise noted in the following evaluation, the licensee proposes to relocate administrative controls intact, with no changes in requirements. All editorial changes are nonsubstantive and are made either to clarify or to accommodate UFSAR text numbering and formatting conventions and are, therefore, acceptable.

3.1 Independent Safety Engineering Group

The licensee proposes that the ISEG function (TS 6.2.3) be relocated to the QAPD with no changes in requirements, except for composition (TS 6.2.3.2). The licensee proposes to reduce the ISEG permanent staffing level from five to three personnel; ISEG personnel qualification requirements remain unchanged.

The principal function of the ISEG, as stated in NUREG-0737, "is to examine plant operating characteristics, NRC issuances, Licensing Information Service advisories, and other appropriate sources of plant design and operating experience information that may indicate areas for improving plant safety." The BVPS ISEG has performed this function since 1987. Since its inception, the BVPS ISEG has reviewed all NUREG-0737 issues for both units and has provided detailed recommendations to station management for revised procedures, equipment modifications, maintenance activities, operations activities, or other means of improving operational safety. Although the ISEG contribution is recognized, the level of ISEG involvement depends on unit operating conditions and emergence of regulatory and industry issues.

In addition, many of the issues addressed by NUREG-0737 are now performed by other BVPS organizations. Because many ISEG reviews are redundant, the requirement to maintain an ISEG staffing level of five dedicated engineers is burdensome in that it restricts station management's capability to utilize resources to their maximum advantage and does not result in an increase in the protection afforded to the health and safety of the public.

The BVPS ISEG will continue to perform its review function with the proposed reduction in staffing. The experience gained over the last 10 years through organizational changes and process improvements has provided confidence that the ISEG function will continue to be fulfilled. At times when increased ISEG staffing may prove beneficial for evaluating issues and recommending plant safety improvements, the ISEG staffing will be increased, consistent with the efficient use of site resources to address both continuing and emergent plant issues. The proposed change would provide flexibility to accomplish the ISEG function without compromising the overall ISEG purpose, scope, or thoroughness of reviews.

3.2 Reviews and Audits

The licensee proposes that review and audit functions specified in TS 6.5 be relocated to the QAPD intact, with no change in requirements. This includes both the onsite review function (TS 6.5.1) and the offsite review function (TS 6.5.2).

3.3 Procedure Review Process

Plant operating procedures and administrative policies are specified by TS 6.8.1. The licensee proposes that the process for review and approval of these procedures and administrative policies (TS 6.8.2) be relocated to the QAPD intact, with no change in requirements; the licensee further proposes that the process for controlling temporary changes to these procedures (TS 6.8.3) be relocated to the QAPD.

3.4 Records and Record Retention

The licensee proposes that the record retention process specified by TS 6.10 be relocated to the QAPD intact, with no change in requirements.

4.0 SUMMARY

The NRC staff finds the reduction in ISEG staffing level to be acceptable. With this exception, the proposed relocation of the TS administrative controls to the QAPD description has been accomplished with no change in requirements or substantive editorial changes. Relocation is consistent with the guidelines provided by NRC Administrative Letter 95-06 and is, therefore, acceptable. As part of the quality assurance program, subsequent changes to these administrative controls will be controlled pursuant to 10 CFR 50.54(a).

In conclusion, requirements related to the administrative controls described above, which the licensee proposes be relocated, are not required to be in the TS under 10 CFR 50.36 or Section 182a of the Atomic Energy Act, are not necessary to assure operation of the facility in a safe manner, and are not required to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to public health and safety and are, therefore, acceptable.

5.0 STATE CONSULTATION

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

6.0 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 and changes surveillance requirements. 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 (62 FR 24986). The amendment also relates to changes in recordkeeping, reporting, or administrative procedures or requirements. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (10). 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.

7.0 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.

Principal Contributor: K. Heck

Date: December 10, 1997