



UNITED STATES
NUCLEAR REGULATORY COMMISSION

WASHINGTON, D.C. 20555-0001

August 31, 1995

Mr. J. E. Cross
Senior Vice President and
Chief Nuclear Officer
Nuclear Power Division
Duquesne Light Company
Post Office Box 4
Shippingport, PA 15077

SUBJECT: BEAVER VALLEY POWER STATION, UNIT NOS. 1 AND 2 (TAC NOS. M90466
AND M90467)

Dear Mr. Cross:

The Commission has issued the enclosed Amendment No. 191 to Facility Operating License No. DPR-66 and Amendment No. 74 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 August 31, 1994, as supplemented by a letter dated May 18, 1995. The May 18, 1995, letter requested withdrawal of the proposed changes to TSs 6.5.2.8.h and 6.5.2.8.i which deal with audits of the BVPS-1 and BVPS-2 fire protection program and withdrawal of the proposed 25-percent grace period for all audit frequencies (Item 6 in August 31, 1994 application). We have granted your May 18, 1995, request for withdrawal.

These amendments delete BVPS-2 License Conditions 2.C.(3), 2.C.(5), 2.C.(7), 2.C.(8), 2.C.(9), and 2.C.(10) to reflect completion of activities required by these license conditions and make the following revisions to the BVPS-1 and BVPS-2 TSs:

1. Eliminate references to specific frequencies for each of the TS required audits (TS 6.2.2.8).
2. Eliminate references to reviews and audits of the Emergency Plan and Security Plan (TSs 6.5.2.8 and 6.8.1).
3. Include the Offsite Dose Calculation Manual and Process Control Program and associated implementing procedures in the list of required audits (TS 6.5.2.8).
4. Editorial changes which were necessitated by a reorganization (TS 6.2.1, 6.2.3.1, 6.2.3.4, 6.5.1.7, 6.5.2.2, 6.5.2.8, 6.5.2.9, and 6.5.2.10).
5. Eliminate reference to Appendix A of 10 CFR Part 55 (TS 6.4.1).
6. Separate the Inservice Inspection (ISI) and Inservice Testing (IST) Programs surveillance requirements and simply reference 10 CFR 50.55a(g) for ISI and 10 CFR 50.55a(f) for IST (TS 4.0.5).

9509070134 950831
PDR ADOCK 05000334
P PDR

NRC FILE CENTER COPY

DF01

J. Cross

- 2 -

August 31, 1995

Copies of our related Safety Evaluation and Notice of Partial Withdrawal are enclosed. The Notice of Issuance will be included in the Commission's biweekly Federal Register notice; the Notice of Partial Withdrawal will be published separately in the Federal Register.

Sincerely,

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

Docket Nos. 50-334/412

Enclosures: 1. Amendment No. 191 to
License No. DPR-66
2. Amendment No. 74 to
License No. NPF-73
3. Safety Evaluation
4. Notice of Partial Withdrawal
cc w/encls: See next page

DISTRIBUTION:

Docket File
PUBLIC
OGC
GHill(4)
GBagchi
PMadden

MO'Brien
DBrinkman
ACRS(4)
JLinville, RGN-I
RWessman
RGramm

CGrimes
PDI-2 Reading
SVarga
JStolz
SBlack
CMcCracken

~~FAllenspach~~

OFFICE	PDI-2/LA	PDI-2/PM	BC:ECGB	BC:EMEB	BC:TQMB
NAME	MO'Brien	DBrinkman:bf	GBagchi	RWessman	SBlack
DATE	8/16/95	7/13/95	8/2/95	8/7/95	8/17/95
OFFICE	BC:SPCR #11 FA	OGC	PDI-2/D		
NAME	CMcCracken		JStolz		
DATE	8/18/95	8/16/95	8/11/95		

OFFICIAL RECORD COPY
DOCUMENT NAME: A:\M90466.AMD

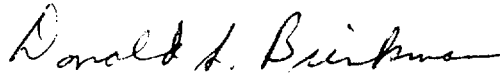
J. Cross

- 2 -

August 31, 1995

Copies of our related Safety Evaluation and Notice of Partial Withdrawal are enclosed. The Notice of Issuance will be included in the Commission's biweekly Federal Register notice; the Notice of Partial Withdrawal will be published separately in the Federal Register.

Sincerely,



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

Docket Nos. 50-334/412

Enclosures: 1. Amendment No. 191 to
License No. DPR-66
2. Amendment No. 74 to
License No. NPF-73
3. Safety Evaluation
4. Notice of Partial Withdrawal

cc w/encs: See next page

J. E. Cross
Duquesne Light Company

Beaver Valley Power Station
Units 1 & 2

cc:

Jay E. Silberg, Esquire
Shaw, Pittman, Potts & Trowbridge
2300 N Street, NW.
Washington, DC 20037

Nelson Tonet, Manager
Nuclear Safety
Duquesne Light Company
Post Office Box 4
Shippingport, PA 15077

Commissioner Roy M. Smith
West Virginia Department of Labor
Building 3, Room 319
Capitol Complex
Charleston, WVA 25305

John D. Borrowes
Director, Utilities Department
Public Utilities Commission
180 East Broad Street
Columbus, OH 43266-0573

Director, Pennsylvania Emergency
Management Agency
Post Office Box 3321
Harrisburg, PA 17105-3321

Ohio EPA-DERR
ATTN: Zack A. Clayton
Post Office Box 1049
Columbus, OH 43266-0149

Dr. Judith Johnsrud
National Energy Committee
Sierra Club
433 Orlando Avenue
State College, PA 16803

Bureau of Radiation Protection
Pennsylvania Department of
Environmental Resources
ATTN: R. Barkanic
Post Office Box 2063
Harrisburg, PA 17120

Mayor of the Borough of
Shippingport
Post Office Box 3
Shippingport, PA 15077

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Resident Inspector
U.S. Nuclear Regulatory Commission
Post Office Box 181
Shippingport, PA 15077

George S. Thomas
Vice President, Nuclear Services
Nuclear Power Division
Duquesne Light Company
P.O. Box 4
Shippingport, PA 15077



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. 191
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 August 31, 1994, as supplemented May 18, 1995, 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.

9509070137 950831
PDR ADOCK 05000334
P PDR

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) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 191, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance, to be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION



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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 31, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 191

FACILITY OPERATING LICENSE NO. DPR-66

DOCKET NO. 50-334

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.

<u>Remove</u>	<u>Insert</u>
3/4 0-2	3/4 0-2
3/4 0-3	3/4 0-3
6-1	6-1
6-3	6-3
6-5	6-5
6-7	6-7
6-8	6-8
6-10	6-10
6-11	6-11
6-12	6-12

SURVEILLANCE REQUIREMENTS

1. At least HOT STANDBY within the next 6 hours,
2. At least HOT SHUTDOWN within the following 6 hours, and
3. At least COLD SHUTDOWN within the subsequent 24 hours.

This specification is not applicable in MODES 5 or 6.

4.0.1 Surveillance Requirements shall be met during the OPERATIONAL MODES or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.

4.0.2 Each Surveillance Requirement shall be performed within the specified time interval with a maximum allowable extension not to exceed 25% of the surveillance interval.

4.0.3 Failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by Specification 4.0.2, shall constitute noncompliance with the OPERABILITY requirements for a Limiting Condition for Operation. The time limits of the ACTION requirements are applicable at the time it is identified that a Surveillance Requirement has not been performed. The ACTION requirements may be delayed for up to 24 hours to permit the completion of the surveillance when the allowable outage time limits of the ACTION requirements are less than 24 hours. Surveillance Requirements do not have to be performed on inoperable equipment.

4.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made unless the Surveillance Requirement(s) associated with a Limiting Condition for Operation has been performed within the stated surveillance interval or as otherwise specified. This provision shall not prevent passage through or to OPERATIONAL MODES as required to comply with ACTION requirements.

4.0.5 Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2 and 3 components shall be applicable as follows:

- a.1 Inservice inspection of ASME Code Class 1, 2 and 3 components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g).

SURVEILLANCE REQUIREMENTS

a.2 Inservice testing of ASME Code Class 1, 2 and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(f).

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

ASME Boiler and Pressure Vessel
Code and applicable Addenda
terminology for inservice
inspection and testing activities

Required frequencies for
performing inservice
inspection and testing
activities

Weekly
Monthly
Quarterly or every 3 months
Semiannually or every 6 months
Every 9 months
Yearly or annually

At least once per 7 days
At least once per 31 days
At least once per 92 days
At least once per 184 days
At least once per 276 days
At least once per 366 days

c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice inspection and testing activities.

d. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.

e. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

6.1 RESPONSIBILITY

6.1.1 The General Manager, Nuclear Operations shall be responsible for overall facility operation and shall delegate in writing the succession to this responsibility during his absence.

6.2 ORGANIZATION

6.2.1 ONSITE AND OFFSITE ORGANIZATIONS

Onsite and Offsite organizations shall be established for facility operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear power plant.

- a. Lines of authority, responsibility and communication shall be established and defined for the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the FSAR.
- b. The General Manager, Nuclear Operations shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.
- c. The Senior Vice President, Nuclear Power Division shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.
- d. The individuals who train the operating staff and those who carry out health physics and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

UNIT STAFF (Continued)

- 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 to 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 REVIEW AND AUDIT

6.5.1 ONSITE SAFETY COMMITTEE (OSC)

FUNCTION

6.5.1.1 The OSC shall function to advise the General Manager, Nuclear Operations on all matters related to nuclear safety and shall provide review capability in the areas of:

- a. nuclear power plant operations
- b. radiological safety
- c. maintenance
- d. nuclear engineering
- e. nuclear power plant testing
- f. technical advisory engineering
- g. chemistry
- h. quality control
- i. instrumentation and control

COMPOSITION

6.5.1.2 The Onsite Safety Committee Supervisor is the OSC Chairman and shall appoint all members of the OSC. The membership shall consist of a minimum of one individual from each of the areas designated in 6.5.1.1.

OSC members and alternates shall meet or exceed the minimum qualifications of ANSI N18.1-1971 Section 4.4 for comparable positions. The nuclear power plant operations individual shall meet the qualifications of Section 4.2.2 and the maintenance individual shall meet the qualifications of Section 4.2.3.

AUTHORITY

6.5.1.7 The OSC shall:

- a. Recommend to the General Manager, Nuclear Operations written approval or disapproval of items considered under 6.5.1.6.a through .d above.
- b. Render determinations in writing with regard to whether or not each item considered under 6.5.1.6.a through .e above constitutes an unreviewed safety question.
- c. Provide written notification within 24 hours to the Senior Vice President, Nuclear Power Division and the Offsite Review Committee of disagreement between the OSC and the General Manager, Nuclear Operations; however, the General Manager, Nuclear Operations shall have responsibility for resolution of such disagreements pursuant to 6.1.1 above.

RECORDS

6.5.1.8 The OSC shall maintain written minutes of each meeting and copies shall be provided to the General Manager, Nuclear Operations and Chairman of the Offsite Review Committee.

6.5.2 OFFSITE REVIEW COMMITTEE (ORC)

FUNCTION

6.5.2.1 The ORC shall function to provide independent review and audit of designated activities in the areas of:

- a. nuclear power plant operations
- b. nuclear engineering
- c. chemistry and radiochemistry
- d. metallurgy
- e. instrumentation and control
- f. radiological safety
- g. mechanical and electrical engineering
- h. quality assurance practices

COMPOSITION

6.5.2.2 The chairman and all members of the ORC shall be appointed by the Senior Vice President, Nuclear Power Division. The membership shall consist of a minimum of five individuals who collectively possess a broad based level of experience and competence enabling the committee to review and audit those activities designated in 6.5.2.1 above and to recognize when it is necessary to obtain technical advice and counsel. An individual may possess expertise in more than one specialty area. The collective competence of the committee will be maintained as changes to the membership are made.

ALTERNATES

6.5.2.3 All alternate members shall be appointed in writing by the ORC Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in ORC activities at any one time.

CONSULTANTS

6.5.2.4 Consultants shall be utilized as determined by the ORC Chairman to provide expert advice to the ORC.

AUDITS

6.5.2.8 Audits of facility activities shall be performed under the cognizance of the ORC. These audits shall encompass:

- a. The conformance of facility operation to provisions contained within the Technical Specifications and applicable license conditions.
- b. The performance, training and qualifications of the entire facility staff.
- c. The results of actions taken to correct deficiencies occurring in facility equipment, structures, systems, or methods of operation that affect nuclear safety.
- d. The performance of activities required by the Quality Assurance Program to meet the criteria of Appendix "B", 10 CFR 50.
- e. Not used.
- f. Not used.
- g. Any other area of facility operation considered appropriate by the ORC or Senior Vice President, Nuclear Power Division.
- h. The Facility Fire Protection Program and implementing procedures at least once per 24 months.
- i. An independent fire protection and loss prevention program inspection and audit shall be performed at least once per 12 months utilizing either qualified off-site licensee personnel or an outside fire protection firm.
- j. An inspection and audit of the fire protection and loss prevention program shall be performed by a qualified outside fire consultant at least once per 36 months.
- k. The OFFSITE DOSE CALCULATION MANUAL and implementing procedures.
- l. The PROCESS CONTROL PROGRAM and implementing procedures for processing and packaging of radioactive waste.

AUTHORITY

6.5.2.9 The ORC shall report to and advise the Senior Vice President, Nuclear Power Division on those areas of responsibility specified in Section 6.5.2.7 and 6.5.2.8.

RECORDS

6.5.2.10 Records of ORC activities shall be prepared, approved and distributed as indicated by the following:

- a. Minutes of each ORC meeting shall be prepared for and approved by the ORC Chairman or Vice-Chairman within 14 days following each meeting.
- b. Reports of reviews encompassed by Section 6.5.2.7 above, shall be documented in the ORC meeting minutes.
- c. Audit reports encompassed by Section 6.5.2.8 above, shall be forwarded to the Senior Vice President, Nuclear Power Division and to the management positions responsible for the areas audited within 30 days after completion of the audit.

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 10CFR 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.

SAFETY LIMIT VIOLATION (Continued)

- 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.
- 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 Each procedure and administrative policy of 6.8.1 above and changes of intent thereto, shall be reviewed by the OSC and approved by 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 considered by the committee, as applicable. Changes to procedures and administrative policies of 6.8.1 above that do not receive OSC review, such as correcting typographical errors, reformatting procedures and other changes not affecting the purpose for which the procedure is performed shall receive an independent review by a qualified individual and approved by a designated manager or director.



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. 74
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 August 31, 1994, as supplemented May 18, 1995, 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. 74, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto are hereby incorporated in the license. DFCO shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. Further, Facility Operating License No. NPF-73, Paragraphs 2.C.(3), 2.C.(5), 2.C.(7), 2.C.(8), 2.C.(9), and 2.C.(10) are hereby amended to read as follows:

(3) Initial Startup Test Program (Section 14 of the SER, and Supplements 3 and 5)

Deleted

(5) Inservice Inspection (Section 6.6 of SER Supplement 5)

Deleted

(7) Plant Safety Monitoring System (PSMS)

Deleted

(8) Detailed Control Room Design Review (DCRDR)

Deleted

(9) Safety Parameter Display System (SPDS)

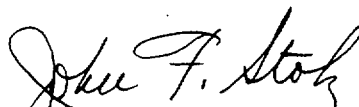
Deleted

(10) Fire Protection Modifications (Section 9.5.1 of SER Supplement 6)

Deleted

4. This license amendment is effective as of the date of its issuance, to be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION



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

- Attachments: 1. Pages 4* and 5* of Facility
Operating License No. NPF-73
2. Changes to the Technical
Specifications

Date of Issuance: August 31, 1995

*Pages 4 and 5 are attached, for convenience, for the composite license to reflect this change.

ATTACHMENT TO LICENSE AMENDMENT NO. 74

FACILITY OPERATING LICENSE NO. NPF-73

DOCKET NO. 50-412

Replace the following pages of the Facility Operating License (FOL) and 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.

	<u>Remove</u>	<u>Insert</u>
FOL	4 5	4 5
Appendix A	3/4 0-2 3/4 0-3 6-1 6-2 6-3 6-6 6-8 6-9 6-10 6-11 6-12 6-13 6-14 6-15	3/4 0-2 3/4 0-3 6-1 6-2 6-3 6-6 6-8 6-9 6-10 6-11 6-12 6-13 6-14 6-15

- (3) Initial Startup Test Program (Section 14 of the SER, and Supplements 3 and 5)

Deleted

- (4) Fresh Fuel Storage

The following criteria apply to the storage and handling of new fuel assemblies in the fuel handling building:

- (a) No more than two fuel assemblies shall be out of approved shipping containers or fuel assembly storage racks at any one time.
- (b) The minimum edge-to-edge distance between the above two new assemblies, the shipping container array, and the storage rack arrays shall be at least 12 inches.
- (c) New fuel assemblies shall be stored in such a manner that water would drain freely from the assemblies in the event of flooding and subsequent draining of the fuel storage area.

- (5) Inservice Inspection (Section 6.6 of SER Supplement 5)

Deleted

- (6) Formal Federal Emergency Management Agency Finding

In the event that the NRC finds that the lack of progress in completion of the procedures in the Federal Emergency Management Agency's final rule, 44 CFR Part 350, is an indication that a major substantive problem exists in achieving or maintaining an adequate state of emergency preparedness, the provisions of 10 CFR Section 50.54(s)(2) will apply.

- (7) Plant Safety Monitoring System (PSMS)

Deleted

(8) Detailed Control Room Design Review (DCRDR)

Deleted

(9) Safety Parameter Display System (SPDS)

Deleted

(10) Fire Protection Modifications (Section 9.5.1 of SER Supplement 6)

Deleted

D. Exemptions

The following exemptions are authorized by law and will not endanger life or property or the common defense and security, and certain special circumstances are present. With the granting of these exemptions, the facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.

- (1) The facility requires an exemption from the requirements of General Design Criterion (GDC) 4, Appendix A to 10 CFR 50. The staff has described in detail in Supplement 4 and Supplement 5 to the Safety Evaluation Report the technical basis and "special circumstances" associated with this exemption. The staff's environmental assessment was published on March 27, 1987 (52 FR 9979). Therefore, pursuant to 10 CFR 50.12(a)(1), 10 CFR 50.12(a)(2)(ii) and (iv), Beaver Valley Power Station, Unit 2 is exempt from the requirements of GDC 4, Appendix A to 10 CFR 50 with respect to the dynamic loading effects associated with the postulated pipe breaks described in detail in Section 3.6.3 of Supplement 4 to the Safety Evaluation Report. These dynamic loading effects include pipe whip, jet impingement, and break-associated dynamic transients. Specifically, this eliminates the need to install jet impingement barriers and pipe whip restraints associated with postulated pipe breaks in the pressurizer surge line, reactor coolant bypass system,

SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be met during the OPERATIONAL MODES or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.

4.0.2 Each Surveillance Requirement shall be performed within the specified time interval with a maximum allowable extension not to exceed 25% of the surveillance interval.

4.0.3 Failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by Specification 4.0.2, shall constitute noncompliance with the OPERABILITY requirements for a Limiting Condition for Operation. The time limits of the ACTION requirements are applicable at the time it is identified that a Surveillance Requirement has not been performed. The ACTION requirements may be delayed for up to 24 hours to permit the completion of the surveillance when the allowable outage time limits of the ACTION requirements are less than 24 hours. Surveillance Requirements do not have to be performed on inoperable equipment.

4.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made unless the Surveillance Requirement(s) associated with a Limiting Condition for Operation has been performed within the stated surveillance interval or as otherwise specified. This provision shall not prevent passage through or to OPERATIONAL MODES as required to comply with ACTION requirements.

4.0.5 Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2 and 3 components shall be applicable as follows:

a.1 Inservice inspection of ASME Code Class 1, 2 and 3 components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g).

a.2 Inservice testing of ASME Code Class 1, 2 and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(f).

SURVEILLANCE REQUIREMENTS

- b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

<u>ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing activities</u>	<u>Required frequencies for performing inservice inspection and testing activities</u>
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days

- c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice inspection and testing activities.
- d. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.
- e. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

6.1 RESPONSIBILITY

6.1.1 The General Manager, Nuclear Operations shall be responsible for overall facility operation and shall delegate in writing the succession to this responsibility during his absence.

6.2 ORGANIZATION

6.2.1 ONSITE AND OFFSITE ORGANIZATIONS

Onsite and offsite organizations shall be established for facility operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear power plant.

- a. Lines of authority, responsibility and communication shall be established and defined for the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the FSAR.
- b. The General Manager, Nuclear Operations shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.
- c. The Senior Vice President, Nuclear Power Division shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.
- d. The individuals who train the operating staff and those who carry out health physics and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

6.2.2 UNIT STAFF

The unit organization shall be subject to the following:

- a. Each duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1.
- b. At least one licensed Operator shall be in the control room when fuel is in the reactor.

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 to the working hour guidelines shall be documented and available for NRC review.

6.2.3 INDEPENDENT SAFETY EVALUATION GROUP (ISEG)

FUNCTION

6.2.3.1 The ISEG shall function to examine unit operating characteristics, NRC issuances, industry advisories, Licensee Event Reports, and other sources of unit design and operating experience information, including units of similar design, which may indicate areas for improving unit safety. The ISEG shall make detailed recommendations for revised procedures, equipment modifications, maintenance activities, operations activities, or other means of improving unit safety to corporate management. If not otherwise implemented, all recommendations shall then be made to the Senior Vice President, Nuclear Power Division.

COMPOSITION

6.2.3.2 The ISEG shall be composed of at least five, dedicated, full-time engineers located on site. Each shall have either:

- (1) A bachelor's degree in engineering or related science and at least 2 years professional level experience in his field, at least 1 year of which experience shall be in the nuclear field, or
- (2) At least 5 years of nuclear experience and hold or have held a Senior Reactor Operator license, or
- (3) At least 10 years of professional level experience in his field, at least 5 years of which experience shall be in the nuclear field.

A minimum of 50% of these personnel shall have the qualifications specified in (1) above.

RESPONSIBILITIES

6.2.3.3 The ISEG shall be responsible for maintaining surveillance of unit activities to provide independent verification⁽¹⁾ that these activities are performed correctly and that human errors are reduced as much as practical.

RECORDS

6.2.3.4 Records of activities performed by the ISEG shall be prepared, maintained, and a summary report shall be forwarded each calendar month to the Senior Vice President, Nuclear Power Division.

(1) Not responsible for sign-off function.

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 REVIEW AND AUDIT

6.5.1 ONSITE SAFETY COMMITTEE (OSC)

FUNCTION

6.5.1.1 The OSC shall function to advise the General Manager, Nuclear Operations on all matters related to nuclear safety and shall provide review capability in the areas of:

- a. nuclear power plant operations
- b. radiological safety
- c. maintenance
- d. nuclear engineering
- e. nuclear power plant testing
- f. technical advisory engineering
- g. chemistry
- h. quality control
- i. instrumentation and control

COMPOSITION

6.5.1.2 The Onsite Safety Committee Supervisor is the OSC Chairman and shall appoint all members of the OSC. The membership shall consist of a minimum of one individual from each of the areas designated in 6.5.1.1.

OSC members and alternates shall meet or exceed the minimum qualifications of ANSI N18.1-1971 Section 4.4 for comparable positions. The nuclear power plant operations individual shall meet the qualifications of Section 4.2.2 and the maintenance individual shall meet the qualifications of Section 4.2.3.

AUTHORITY

6.5.1.7 The OSC shall:

- a. Recommend to the General Manager, Nuclear Operations written approval or disapproval of items considered under 6.5.1.6.a through d above.
- b. Render determinations in writing with regard to whether or not each item considered under 6.5.1.6.a through e above constitutes an unreviewed safety question.
- c. Provide written notification within 24 hours to the Senior Vice President, Nuclear Power Division and the Offsite Review Committee of disagreement between the OSC and the General Manager, Nuclear Operations; however, the General Manager, Nuclear Operations shall have responsibility for resolution of such disagreements pursuant to 6.1.1 above.

RECORDS

6.5.1.8 The OSC shall maintain written minutes of each meeting and copies shall be provided to the General Manager, Nuclear Operations and Chairman of the Offsite Review Committee.

6.5.2 OFFSITE REVIEW COMMITTEE (ORC)

FUNCTION

6.5.2.1 The ORC shall function to provide independent review and audit of designated activities in the areas of:

- a. nuclear power plant operations
- b. nuclear engineering
- c. chemistry and radiochemistry
- d. metallurgy
- e. instrumentation and control
- f. radiological safety
- g. mechanical and electrical engineering
- h. quality assurance practices

COMPOSITION

6.5.2.2 The chairman and all members of the ORC shall be appointed by the Senior Vice President, Nuclear Power Division. The membership shall consist of a minimum of five individuals who collectively possess a broad based level of experience and competence enabling the committee to review and audit those activities designated in 6.5.2.1 above and to recognize when it is necessary to obtain technical advice and counsel. An individual may possess expertise in more than one specialty area. The collective competence of the committee will be maintained as changes to the membership are made.

ALTERNATES

6.5.2.3 All alternate members shall be appointed in writing by the ORC Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in ORC activities at any one time.

CONSULTANTS

6.5.2.4 Consultants shall be utilized as determined by the ORC Chairman to provide expert advice to the ORC.

MEETING FREQUENCY

6.5.2.5 The ORC shall meet at least once per calendar quarter during the initial year of facility operation following fuel loading and at least once per six months thereafter.

QUORUM

6.5.2.6 A quorum of ORC shall consist of the Chairman or his designated alternate and at least four members including alternates. No more than a minority of the quorum shall have line responsibility for operation of the facility.

REVIEW

6.5.2.7 The ORC shall review:

- a. The safety evaluations for 1) changes to procedures, equipment, or systems and 2) tests or experiments completed under the provision of Section 50.59, 10 CFR, to verify that such actions did not constitute an unreviewed safety question.

REVIEW (Continued)

- b. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.
- c. Proposed tests or experiments which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.
- d. Proposed changes in Technical Specifications or licenses.
- e. Violations of applicable statutes, codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance.
- f. Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affect nuclear safety.
- g. All REPORTABLE EVENTS.
- h. All recognized indications of an unanticipated deficiency in some aspect of design or operation of safety-related structures, systems, or components.
- i. Reports and meeting minutes of the OSC.
- j. The results of the Radiological Environmental Monitoring Program prior to submittal of the annual report provided in accordance with Specification 6.9.1.10.

AUDITS

6.5.2.8 Audits of facility activities shall be performed under the cognizance of the ORC. These audits shall encompass:

- a. The conformance of facility operations to provisions contained within the Technical Specifications and applicable license conditions.
- b. The performance, training, and qualifications of the entire facility staff.
- c. The results of actions taken to correct deficiencies occurring in facility equipment, structures, systems, or methods of operation that affect nuclear safety.
- d. The performance of activities required by the Quality Assurance Program to meet the criteria of Appendix "B", 10 CFR 50.

AUDITS (Continued)

- e. Not used.
- f. Not used.
- g. Any other area of facility operation considered appropriate by the ORC or the Senior Vice President, Nuclear Power Division.
- h. The Facility Fire Protection Program and implementing procedures at least once per 24 months.
- i. An independent fire protection and loss prevention program inspection and audit shall be performed at least once per 12 months utilizing either qualified off-site licensee personnel or an outside fire protection firm.
- j. An inspection and audit of the fire protection and loss prevention program shall be performed by a qualified outside fire consultant at least once per 36 months.
- k. The OFFSITE DOSE CALCULATION MANUAL and implementing procedures.
- l. The PROCESS CONTROL PROGRAM and implementing procedures for processing and packaging of radioactive waste.

AUTHORITY

6.5.2.9 The ORC shall report to and advise the Senior Vice President, Nuclear Power Division on those areas of responsibility specified in Section 6.5.2.7 and 6.5.2.8.

RECORDS

6.5.2.10 Records of ORC activities shall be prepared, approved and distributed as indicated by the following:

- a. Minutes of each ORC meeting shall be prepared for and approved by the ORC Chairman or Vice Chairman within 14 days following each meeting.
- b. Reports of reviews encompassed by Section 6.5.2.7 above, shall be documented in the ORC meeting minutes.
- c. Audit reports encompassed by Section 6.5.2.8 above, shall be forwarded to the Senior Vice President, Nuclear Power Division and to the management positions responsible for the areas audited within 30 days after completion of the audit.

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.
- 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.

PROCEDURES (Continued)

- f. Fire Protection Program implementation.
- g. PROCESS CONTROL PROGRAM implementation.
- h. OFFSITE DOSE CALCULATION MANUAL implementation.

6.8.2 Each procedure and administrative policy of 6.8.1 above and changes of intent thereto, shall be reviewed by the OSC and approved by 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 considered by the committee, as applicable. Changes to procedures and administrative policies of 6.8.1 above that do not receive OSC review, such as correcting typographical errors, reformatting procedures and other changes not affecting the purpose for which the procedure is performed shall receive an independent review by a qualified individual and approved by a designated manager or director.

6.8.3 Temporary changes to procedures of 6.8.1 above may be made provided:

- a. The intent of the original procedure is not altered.
- b. The change is approved by two (2) members of the plant management staff, at least one (1) of whom holds a Senior Reactor Operator's License on the unit affected.
- c. The change is documented, reviewed by the OSC and approved by 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, within 14 days of implementation.

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.

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.

6.8.6 The following programs shall be established, implemented, and maintained:

a. Radioactive Effluent Control Program

A program shall be provided conforming with 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to MEMBERS OF THE PUBLIC from radioactive effluents as low as reasonably achievable. The program (1) shall be contained in the ODCM, (2) shall be implemented by operating procedures, and (3) shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

- 1) Limitations on the operability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM,
- 2) Limitations on the concentrations of radioactive material released in liquid effluents to UNRESTRICTED AREAS conforming to 10 CFR Part 20, Appendix B, Table 2, Column 2,

PROCEDURES (Continued)

- 3) Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.106 or 10 CFR 20.1302, as appropriate, and with the methodology and parameters in the ODCM,
- 4) Limitations on the annual and quarterly doses or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released from each unit to UNRESTRICTED AREAS conforming to Appendix I to 10 CFR Part 50,
- 5) Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days,
- 6) Limitations on the operability and use of the liquid and gaseous effluent treatment systems to ensure that the appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a 31-day period would exceed 2 percent of the guidelines for the annual dose or dose commitment conforming to Appendix I to 10 CFR Part 50,
- 7) Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the SITE BOUNDARY conforming to the doses associated with 10 CFR Part 20, Appendix B, Table 2, Column 1,
- 8) Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50,
- 9) Limitations on the annual and quarterly doses to a MEMBER OF THE PUBLIC from Iodine-131, Iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50,
- 10) Limitations on the annual dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR Part 190.



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO AMENDMENT NOS. 191 AND 74 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 August 31, 1994, as supplemented May 18, 1995, 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) and for deletion of certain license conditions. The requested changes would delete BVPS-2 License Conditions 2.C.(3), 2.C.(5), 2.C.(7), 2.C.(8), 2.C.(9), and 2.C.(10) to reflect completion of activities required by these license conditions and make the following revisions to the BVPS-1 and BVPS-2 TSs:

1. Eliminate references to specific frequencies for each of the TS required audits (TS 6.2.2.8).
2. Eliminate references to reviews and audits of the Emergency Plan and Security Plan (TSs 6.5.2.8 and 6.8.1).
3. Include the Offsite Dose Calculation Manual and Process Control Program and associated implementing procedures in the list of required audits (TS 6.5.2.8).
4. Editorial changes which were necessitated by a reorganization (TS 6.2.1, 6.2.3.1, 6.2.3.4, 6.5.1.7, 6.5.2.2, 6.5.2.8, 6.5.2.9, and 6.5.2.10).
5. Eliminate reference to Appendix A of 10 CFR Part 55 (TS 6.4.1).

9509070138 950831
PDR ADOCK 05000334
PDR

6. Separate the Inservice Inspection (ISI) and Inservice Testing (IST) Programs surveillance requirements and simply reference the regulations 10 CFR 50.55a(g) for ISI and 10 CFR 50.55a(f) for IST (TS 4.0.5).

The May 18, 1995, letter requested withdrawal of the proposed changes to TS 6.5.2.8 dealing with audits of the Beaver Valley Power Station, Units 1 and 2, fire protection program and withdrawal of a proposed 25-percent grace period for all audit frequencies (Item 6 in August 31, 1994 application). The May 18, 1995, letter did not change the initial proposed no significant hazards consideration determination. The NRC staff has granted the May 18, 1995, request for withdrawal.

2.0 EVALUATION

2.1 TSs 6.5.2.8 and 6.8.1

The licensee proposed to relocate the audit frequencies from TSs 6.5.2.8.a, 6.5.2.8.b, 6.5.2.8.c, and 6.5.2.8.d. The licensee also proposed to delete TSs 6.5.2.8.e, 6.5.2.8.f, 6.8.1.d, and 6.8.1.e which apply to the Emergency and Security Plans. In addition the licensee proposed new TSs 6.5.2.8.k and 6.5.2.8.l with respect to the Offsite Dose Calculation Manual and the Process Control manual for processing and packaging of radioactive waste.

Section 182a of the Atomic Energy Act (the "Act") requires applicants for nuclear power plant operating licenses to include TSs 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 TSs include items in five specific categories, including (1) safety limits, limiting safety system settings and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; and (5) administrative controls. However, the regulation does not specify the particular requirements to be included in a plant's TSs.

The Commission has provided guidance for the contents of TSs in its "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors" ("Final Policy Statement"), 58 FR 39132 (July 22, 1993), in which the Commission indicated that compliance with the Final Policy Statement satisfies Section 182a of the Act. In particular, the Commission indicated that certain items could be relocated from the TSs to licensee-controlled documents, consistent with the standard enunciated in *Portland General Electric Co.* (Trojan Nuclear Plant), ALAB-531, 9 NRC 263, 273 (1979). In that case, the Atomic Safety and Licensing Appeal Board indicated that "technical specifications are to be reserved for those matters as to which the imposition of rigid conditions or limitations upon reactor operation is deemed necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety."

Consistent with this approach, the Final Policy Statement identified four criteria to be used in determining whether particular limiting conditions for operation are required to be included in the TSs, as follows: (1) installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary; (2) a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier; (3) a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier; (4) a structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.¹

The Commission's policy statement provides that many of the existing TS limiting conditions for operation which do not satisfy these four specified criteria may be relocated to the Updated Final Safety Analysis Report (UFSAR), such that future changes could be made to these provisions pursuant to 10 CFR 50.59. Other requirements may be relocated to more appropriate documents (e.g. Security Plan, Quality Assurance (QA) plan, and Emergency Plan) and controlled by the applicable regulatory requirement. While the content of the TSs administrative controls is specified in 10 CFR 50.36(c)(5), particular details of the administrative controls may be relocated to licensee-controlled documents where 10 CFR 50.59 or comparable regulatory controls exist.

Administrative controls in existing TSs related to the review and audit functions, including specified frequency provisions, should be relocated to a licensee-controlled document that provides adequate control over changes to these provisions and which provides an appropriate change control mechanism. As such, these review and audit provisions should be relocated to the Quality Assurance Program described or referenced in the facility's UFSAR and controlled pursuant to 10 CFR 50.54.

The licensee proposed that the review and audit functions and frequencies (other than fire protection related) specified in existing TS 6.5.2.8 be relocated from the TSs to the UFSAR (Section 1.3.4.1 for BVPS-1 and Table 1.8-1 for BVPS-2) specifying that the TSs identified audits will be accomplished with a frequency commensurate with their safety significance not to exceed a biennial (2 years) frequency as defined by Quality Services Unit Program commitments to NRC

¹The Commission recently adopted amendments to 10 CFR 50.36, pursuant to which the rule was revised to codify and incorporate these criteria. See Final Rule, "Technical Specifications," 60 FR 36953 (July 19, 1995). The Commission indicated that reactor core isolation cooling, isolation condenser, residual heat removal, standby liquid control, and recirculation pump are to be included in the TSs under Criterion 4, although it recognized that other structures, systems and components could also meet this criterion (60 FR 36956).

Regulatory Guides 1.33 and 1.144. The licensee commitment to the QA Regulatory Guides is considered part of the Quality Assurance Program (QAP) described in the UFSAR such that future changes would be made pursuant to 10 CFR 50.54(a).

These particular TS provisions are not necessary to assure safe operation of the facility, given that the requirements in the QA program implement the Commission's regulations pertaining to these review and audit functions as set forth below. The review and audit functions define an administrative framework to confirm that plant activities have been properly conducted in a safe manner. The reviews and audits serve also to provide a cohesive program that provides senior level licensee management with assessments of facility operation and recommends actions to improve nuclear safety and reliability. As such, the review and audit program does not include any elements that are delineated in the Final Policy Statement criteria, as discussed above, for determining which limiting conditions are required to be included in the TSs. As documented in the Final Policy Statement, the review and audit functions constitute requirements that can be relocated to the Quality Assurance plan and controlled by the applicable regulatory requirement. The security and emergency plans' review and audit requirements are proposed to be relocated to their respective plans in accordance with Generic Letter (GL) 93-07. The emergency and security plans implement the Commission's regulations discussed below for these reviews and audits. Such an approach would result in an equivalent level of regulatory authority while providing for a more appropriate change control process. In addition, the following considerations support relocating these items from the TSs:

The current audit frequencies specified in TSs 6.5.2.8.a, 6.5.2.8.b, and 6.5.2.8.c are being extended to a maximum interval of once per 2 years. Audit requirements are specified in the QA program to satisfy 10 CFR Part 50, Appendix B, Criterion XVIII. The licensee has committed to or relies upon the guidance in ANSI N18.7 (endorsed by NRC Regulatory Guide 1.33) and ANSI N45.2 (endorsed by NRC Regulatory Guide 1.28) to meet the requirements of Appendix B to 10 CFR Part 50. Audits are also governed by 10 CFR 50.54(t), 10 CFR 50.54(p), and 10 CFR Part 73. Therefore, duplication of these requirements does not enhance the level of plant safety. Control of changes to the QA program description are governed by the provisions of 10 CFR 50.54(a). In accordance with ANSI N18.7, the licensee will audit safety related activities, including TS identified audits where the frequencies were relocated, at least once every 2 years. The NRC staff finds these proposed audit frequency extensions acceptable since these changes are in accordance with ANSI N18.7 and the licensee will perform the audits on a frequency commensurate with the safety significance of operational phase activities.

The licensee will continue to implement a QA program in accordance with the requirements of 10 CFR Part 50, Appendix B, and commitments to ANSI N18.7, and Regulatory Guide 1.28 which provides appropriate controls for the approval of changes to the audit functions and frequencies. Changes to the QA program, including departures from the referenced ANSI standards, that constitute a reduction in commitment, can be made in the future pursuant to 10 CFR 50.54(a).

The staff concludes that this regulatory requirement provides sufficient control for the audit functions and frequencies, so that removing these requirements from the TS is acceptable.

The licensee similarly proposes to relocate the requirements to establish, implement, and maintain procedures related to the Emergency Plan (existing TS 6.5.2.8.e and 6.8.1.e) and Security Plan (existing TS 6.5.2.8.f and 6.8.1.d), including related requirements for periodic reviews of these programs and implementing procedures, as recommended in Generic Letter 93-07, "Modification of the Technical Specification Administrative Control Requirements for Emergency and Security Plans," dated December 28, 1993. The Security Plan requirements specified in 10 CFR 50.54, 73.40, 73.55, and 73.56, and the Emergency Plan requirements are specified in 10 CFR 50.54(q) and 10 CFR Part 50, Appendix E, Section V, provide adequate regulatory controls for these programs. Duplication of the requirements contained in the regulations would not enhance the level of safety for the facility. On this basis, the NRC staff has concluded that the existing TS requirements can be relocated to the respective plans, and removed from TSs. Future changes in these requirements must be made in accordance with 10 CFR 50.54(p) for the security plan and 10 CFR 50.54(q) for the emergency plan.

On this basis, the NRC staff concludes that these provisions are not required to be in the TSs under 10 CFR 50.36 or Section 182a of the Atomic Energy Act, and are not required to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety. In addition, the NRC staff finds that sufficient regulatory controls exist under 10 CFR 50.54 to adequately control future modifications to these provisions. Accordingly, the NRC staff has concluded that these requirements may be relocated from the TSs to the respective licensee-controlled documents.

The licensee proposes to add TSs 6.5.2.8.k and 6.5.2.8.l to formalize the requirements to audit the Offsite Dose Calculational Manual and implementing procedures and the Process Control Program and implementing procedures. The NRC staff finds the addition of these audit requirements acceptable since these are additional commitments which will enhance the licensee's audit program.

2.2 Title Changes

The licensee proposed to change the title "Vice President, Nuclear Group" to "Senior Vice President, Nuclear Power Division" in numerous places in the Administrative Controls section of the TSs to reflect an organizational change. The duties and responsibilities of the position have not changed. The proposed change is in title only and is merely administrative. Therefore, the NRC staff finds the proposed change acceptable.

2.3 TS 6.4.1

The proposed amendment would delete the phrase "Appendix A of" prior to 10 CFR Part 55 in TS 6.4.1. The proposed deletion is an administrative change which the NRC staff finds acceptable since the requirements of Appendix A have now been included in 10 CFR Part 55.

2.4 TS 4.0.5

The licensee also proposed to delete the phrase "except where specific written relief has been granted" for both the inservice inspection (ISI) and inservice testing (IST) programs. The proposed deletion would permit the licensee to implement proposed relief requests without prior NRC approval during the first 12 months of a 10 year ISI/IST interval provided the relief requests were determined to be clearly impractical. The NRC requires that licensees must establish and implement their ISI/IST programs in accordance with the requirements of 10 CFR 50.55a. NRC staff guidance regarding the implementation of ISI/IST relief requests during the first 12 months of a 10 year ISI/IST interval is provided in Sections 2.5 and 6.3 of NUREG-1482. NUREG-1482 states that licensees may implement proposed relief requests without prior NRC approval, during the first 12 months of a 10 year ISI/IST interval, that have been determined to be clearly impractical provided the NRC is informed of those ASME Code requirements which are impractical to meet and relief is requested from those requirements. We have determined that this proposed change is in accordance with the guidance provided in NUREG-1482 and is therefore, acceptable.

2.5 Deletion of Certain License Conditions

Six license conditions in the BVPS-2 Facility Operating License would be deleted by the proposed amendment. Our evaluations of the proposed deletions are as follows:

2.5.1 License Condition 2.C.(3), Initial Startup Test Program

This license condition required any changes to the BVPS-2 Initial Test Program described in Section 14 of the FSAR made in accordance with the provisions of 10 CFR 50.59 to be reported in accordance with the requirements of 10 CFR 50.59(b) within one month of such change. The licensee completed the BVPS-2 Initial Test Program on June 9, 1989, and submitted the final supplement (Supplement 6) to the BVPS-2 Startup Report on August 4, 1989. Therefore, the action required by this license condition has been completed and License Condition 2.C.(3) may be deleted.

2.5.2 License Condition 2.C.(5), Inservice Inspection

This license condition required the licensee to submit the BVPS-2 ISI program in accordance with 10 CFR 50.55a(g)(4) for NRC staff review by June 1, 1988. By letter dated June 1, 1988, the licensee submitted the first 10-year ISI program for BVPS-2. Therefore, the action required by this license condition has been completed and Licensee Condition 2.C.(5) may be deleted.

2.5.3 License Condition 2.C.(7), Plant Safety Monitoring System (PSMS)

This license condition required the licensee to submit to the NRC on or before November 27, 1987, a verification and validation (V&V) plan to demonstrate the reliability of the Plant Safety Monitoring System. This license condition further required that the V&V plan be implemented before startup after the second refueling outage. By letter dated December 4, 1990, the licensee informed the NRC that the required actions had been completed. Therefore, this requirement has been satisfied and License Condition 2.C.(7) may be deleted.

2.5.4 License Condition 2.C.(8), Detailed Control Room Design Review (DCRDR)

This license condition required resolution of all open DCRDR issues before startup from the second refueling outage. By letter dated December 3, 1990, the licensee informed the NRC that all such actions had been completed. Therefore, this requirement has been satisfied and License Condition 2.C.(8) may be deleted.

2.5.5 License Condition 2.C.(9), Safety Parameter Display System (SPDS)

This license condition required the licensee to perform the necessary activities, provide acceptable responses, and implement all proposed corrective actions related to SPDS items specified in License Amendment No. 16 before startup from the second refueling outage. By letter dated December 4, 1990, the licensee confirmed that these items had been implemented. Therefore, this requirement has been satisfied and License Condition 2.C.(9) may be deleted.

2.5.6 License Condition 2.C.(10), Fire Protection Modifications

This licensee condition required completion of the installation of back draft dampers by September 30, 1987, to mitigate overpressurization caused by discharge of the carbon dioxide system. By letter dated September 30, 1987, the licensee provided a status report regarding the installation of the required dampers, associated controls, and logic testing. The September 30, 1987, letter stated that completion of all carbon dioxide testing for all zones by September 30, 1987, was unattainable. The September 30, 1987, letter went on to state that additional hardware modifications were required and that the expected completion date was December 31, 1987.

By letter dated December 31, 1987, the licensee informed the NRC that the required additional hardware modifications had been completed and that the systems were placed in operation prior to the scheduled due date of December 31, 1987. Therefore, this requirement has been satisfied and License Condition 2.C.(10) may be deleted.

3.0 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.

4.0 ENVIRONMENTAL CONSIDERATION

The amendments 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 (59 FR 65812). The amendments also relate to changes in recordkeeping, reporting, or administrative procedures or requirements. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (c)(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 amendments.

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

Principal Contributors: Donald S. Brinkman
Frederick R. Allenspach

Date: August 31, 1995

UNITED STATES NUCLEAR REGULATORY COMMISSION

DUQUESNE LIGHT COMPANY

OHIO EDISON COMPANY

PENNSYLVANIA POWER COMPANY

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

THE TOLEDO EDISON COMPANY

BEAVER VALLEY POWER STATION, UNITS 1 AND 2

DOCKET NOS. 50-334 AND 50-412

NOTICE OF PARTIAL WITHDRAWAL OF APPLICATION FOR
AMENDMENT TO FACILITY OPERATING LICENSE

The United States Nuclear Regulatory Commission (the Commission) has granted the request by Duquesne Light Company (the licensee) to withdraw a portion of its August 31, 1994, application for a proposed amendment to Facility Operating License Nos. DPR-66 and NPF-73 for Beaver Valley Power Station, Units 1 and 2, located in Beaver County, Pennsylvania.

The proposed amendment involved deletion of certain license conditions and the following changes to the technical specifications (TSs):

1. Elimination of the references to specific frequencies for each of the Technical Specification required audits.
2. Elimination of the references to reviews and audits of the Emergency Plan and Security Plan.
3. Separation of the Inservice Inspection (ISI) and Inservice Testing (IST) Programs surveillance requirements and removal of the requirement that relief requests be granted before they are implemented for both IST and ISI.

4. Editorial changes which were necessitated by a reorganization.
5. Elimination of the reference to Appendix A of 10 CFR Part 55.
6. Elimination of the requirement to perform an independent fire protection and loss prevention program inspection annually.
7. Inclusion of the Offsite Dose Calculation Manual and Process Control Program and associated implementing procedures into the list of required audits.

On May 18, 1995, the licensee submitted a letter to the NRC requesting withdrawal of the proposed changes to the TSs dealing with audits of the Beaver Valley Power Station fire protection program and withdrawal of a proposed 25-percent grace period for all audit frequencies (Item 6 of August 31, 1994 application).

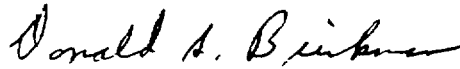
The Commission has previously issued a Notice of Consideration of Issuance of Amendments to Facility Operating Licenses, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing, which was published in the FEDERAL REGISTER on December 21, 1994 (59 FR 65812).

For further details with respect to this action, see the application for amendment dated August 31, 1994, and the licensee's letter of May 18, 1995, which withdrew the portion of the application for license amendment. The above documents are available for public inspection at the Commission's Public

Document Room, 2120 L Street, NW., Washington, DC 20555 and at the B. F. Jones Memorial Library, 663 Franklin Avenue, Aliquippa, Pennsylvania 15001.

Dated at Rockville, Maryland, 31 day of August 1995.

FOR THE NUCLEAR REGULATORY COMMISSION



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