December 21, 1998

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

SUBJECT: BEAVER VALLEY POWER STATION, UNIT NO. 1 (TAC NO. M95792)

Dear Mr. Cross:

The Commission has issued the enclosed Amendment No. 219 to Facility Operating License No. DPR-66 for the Beaver Valley Power Station, Unit No. 1 (BVPS-1). This amendment consists of changes to the BVPS-1 Technical Specifications (TSs) in response to your application dated June 18, 1996, as supplemented September 8 and 30, 1998, which submitted Proposed Operating License Change Request Nos. 235 and 109. Proposed Operating License Change Request No. 109, applicable to Beaver Valley Power Station, Unit No. 2, will be processed at a later date.

The amendment (1) makes editorial changes to TS 4.4.5 and associated Bases; (2) revises the Bases for TS 3.4.6.2 to provide consistency with the BVPS-1 Updated Final Safety Analysis Report; and (3) revises Index Page XVII to reflect the revision of page numbers due to shifting of text by License Amendment No. 198.

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

Sincerely,

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

Docket No. 50-334

Enclosures: 1. Amendment No. 219 to DPR-66
2. Safety Evaluation

cc w/encls: See next page

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NAME	DBrinkman:rb	MO'Brien	ESullivan	[illegible]	RCapra Roe
DATE	9/30/98	10/1/98	10/9/98	12/3/98	12/16/98

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DOCUMENT NAME: BV95792.AMD

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

December 21, 1998

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President-Generation Group
Duquesne Light Company
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Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

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cc w/encls: See next page

cc:

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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. 219
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 June 18, 1996, as supplemented September 8 and 30, 1998, 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) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 219, 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 its date of issuance, to be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Capra

Robert A. Capra, 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: December 21, 1998

ATTACHMENT TO LICENSE AMENDMENT NO. 219

FACILITY OPERATING LICENSE NO. DPR-66

DOCKET NO. 50-334

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

Remove	Insert
XVII	XVII
3/4 4-10	3/4 4-10
3/4 4-10a	3/4 4-10a
3/4 4-10b	3/4 4-10b
3/4 4-10c	3/4 4-10c
3/4 4-10d	3/4 4-10d
3/4 4-10g	3/4 4-10g
3/4 4-10h	3/4 4-10h
B 3/4 4-2b	B 3/4 4-2b
B 3/4 4-3f	B 3/4 4-3f
B 3/4 4-3g	B 3/4 4-3g

SURVEILLANCE REQUIREMENTS (Continued)

Category	Inspection Results
C-1	Less than 5 percent of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1 percent of the total tubes inspected are defective, or between 5 percent and 10 percent of the total tubes inspected are degraded tubes.
C-3	More than 10 percent of the total tubes inspected are degraded tubes or more than 1 percent of the inspected tubes are defective.

Note: In all inspections, previously degraded tubes or sleeves must exhibit significant (greater than 10 percent) further wall penetrations to be included in the above percentage calculations.

4.4.5.3 Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections following service under All Volatile Treatment (AVT) conditions, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.

b. If the inservice inspection of a steam generator conducted in accordance with Table 4.4-2 requires a third sample inspection whose results fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until a subsequent inspection demonstrates that a third sample inspection is not required.

SURVEILLANCE REQUIREMENTS (Continued)

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.4-2 during the shutdown subsequent to any of the following conditions:

   1. Primary-to-secondary tube leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.6.2,

   2. A seismic occurrence greater than the Operating Basis Earthquake,

   3. A loss-of-coolant accident requiring actuation of the engineered safeguards, or

   4. A main steamline or feedwater line break.

4.4.5.4 Acceptance Criteria

a. As used in this Specification:

   1. Imperfection means an exception to the dimensions, finish or contour of a tube or sleeve from that required by fabrication drawings or specifications. Eddy-current testing indications below 20 percent of the nominal tube wall thickness, if detectable, may be considered as imperfections.

   2. Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube or sleeve.

   3. Degraded Tube means a tube or sleeve containing imperfections greater than or equal to 20 percent of the nominal wall thickness caused by degradation.

   4. Percent Degradation means the percentage of the tube or sleeve wall thickness affected or removed by degradation.

   5. Defect means an imperfection of such severity that it exceeds the plugging or repair limit. A tube containing a defect is defective. Any tube which does not permit the passage of the eddy-current inspection probe shall be deemed a defective tube.

SURVEILLANCE REQUIREMENTS (Continued)

6. Plugging or Repair Limit means the imperfection depth at or beyond which the tube shall be removed from service by plugging or repaired by sleeving in the affected area because it may become unserviceable prior to the next inspection. The plugging or repair limit imperfection depths are specified in percentage of nominal wall thickness as follows:

   a) Original tube wall 40%

      This definition does not apply to tube support plate intersections for which the voltage-based repair criteria are being applied. Refer to 4.4.5.4.a.10 for the repair limit applicable to these intersections.

   b) ABB Combustion Engineering TIG welded sleeve wall 32%

   c) Westinghouse laser welded sleeve wall 25%

7. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steamline or feedwater line break as specified in 4.4.5.3.c, above.

8. Tube Inspection means an inspection of the steam generator tube from the point of entry (hot-leg side) completely around the U-bend to the top support to the cold-leg.

9. Tube Repair refers to sleeving which is used to maintain a tube in-service or return a tube to service. This includes the removal of plugs that were installed as a corrective or preventive measure. The following sleeve designs have been found acceptable:

   a) ABB Combustion Engineering TIG Welded Sleeves, CEN-629-P, Revision 02 and CEN-629-P Addendum 1.

   b) Westinghouse laser welded sleeves, WCAP-13483, Revision 1.

SURVEILLANCE REQUIREMENTS (Continued)

10. Tube Support Plate Plugging Limit is used for the disposition of an alloy 600 steam generator tube for continued service that is experiencing predominantly axially oriented outside diameter stress corrosion cracking confined within the thickness of the tube support plates. At tube support plate intersections, the plugging (repair) limit is based on maintaining steam generator tube serviceability as described below:

   a) Steam generator tubes, whose degradation is attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with bobbin voltages less than or equal to 2.0 volts will be allowed to remain in service.

   b) Steam generator tubes, whose degradation is attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than 2.0 volts will be repaired or plugged, except as noted in 4.4.5.4.a.10.c below.

   c) Steam generator tubes, with indications of potential degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than 2.0 volts but less than or equal to the upper voltage repair limit(1) may remain in service if a rotating pancake coil or acceptable alternative inspection does not detect degradation. Steam generator tubes, with indications of outside diameter stress corrosion cracking degradation with a bobbin voltage greater than the upper voltage repair limit(1) will be plugged or repaired.

   d) If an unscheduled mid-cycle inspection is performed, the following mid-cycle repair limits apply instead of the limits identified in 4.4.5.4.a.10.a, 4.4.5.4.a.10.b, and 4.4.5.4.a.10.c.

(1) The upper voltage repair limit is calculated according to the methodology in Generic Letter 95-05 as supplemented.

SURVEILLANCE REQUIREMENTS (Continued)

The mid-cycle repair limits are determined from the following equations:

$$V_{MURL} = \frac{V_{SL}}{1.0 + NDE + Gr\left(\frac{CL - \Delta t}{CL}\right)}$$

$$V_{MLRL} = V_{MURL} - (V_{URL} - V_{LRL})\left(\frac{CL - \Delta t}{CL}\right)$$

where:

V_{URL}	=	upper voltage repair limit
V_{LRL}	=	lower voltage repair limit
V_{MURL}	=	mid-cycle upper voltage repair limit based on time into cycle
V_{MLRL}	=	mid-cycle lower voltage repair limit based on V_{MURL} and time into cycle
Δt	=	length of time since last scheduled inspection during which V_{URL} and V_{LRL} were implemented
CL	=	cycle length (the time between two scheduled steam generator inspections)
V_{SL}	=	structural limit voltage
Gr	=	average growth rate per cycle length
NDE	=	95-percent cumulative probability allowance for nondestructive examination uncertainty (i.e., a value of 20-percent has been approved by NRC)[2]

Implementation of these mid-cycle repair limits should follow the same approach as in TS 4.4.5.4.a.10.a, 4.4.5.4.a.10.b, and 4.4.5.4.a.10.c.

(2) The NDE is the value provided by the NRC in GL 95-05 as supplemented.

TABLE 4.4-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE

INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	No	Yes
No. of Steam Generators per Unit	Three	Three
First Inservice Inspection	All	Two
Second & Subsequent Inservice Inspections	One (1)	One (2)

Table Notation:

(1) The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 9 percent of the tubes if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

(2) The other steam generator not inspected during the first inservice inspection shall be inspected. The third and subsequent inspections should follow the instructions described in (1) above.

TABLE 4.4-2

STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S tubes per S.G.	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug or repair defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N/A	N/A
			C-2	Plug or repair defective tubes and inspect additional 4S tubes in this S.G.	C-1	None
					C-2	Plug or repair defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this S.G., plug or repair defective tubes and inspect 2S tubes in each other S.G. Notification to NRC pursuant to Specification 6.6	All other S.G.s are C-1	None	N/A	N/A
			Some S.G.s C-2 but no additional S.G.s are C-3	Perform action for C-2 result of second sample	N/A	N/A
			Additional S.G. is C-3	Inspect all tubes in each S.G. and plug or repair defective tubes. Notification to NRC pursuant to Specification 6.6.	N/A	N/A

$S = \frac{9}{n}\ \%$ Where n is the number of steam generators inspected during an inspection.

3/4.4.5 STEAM GENERATORS (Continued)

no NDE detectable cracks extending outside the thickness of the support plate. Refer to GL 95-05 for additional description of the degradation morphology.

Implementation of these SRs requires a derivation of the voltage structural limit from the burst versus voltage empirical correlation and then the subsequent derivation of the voltage repair limit from the structural limit (which is then implemented by this surveillance).

The voltage structural limit is the voltage from the burst pressure/bobbin voltage correlation, at the 95-percent prediction interval curve reduced to account for the lower 95/95-percent tolerance bound for tubing material properties at 650°F (i.e., the 95-percent LTL curve). The voltage structural limit must be adjusted downward to account for potential degradation growth during an operating interval and to account for NDE uncertainty. The upper voltage repair limit; V_{URL}, is determined from the structural voltage limit by applying the following equation:

$$V_{URL} = V_{SL} - V_{Gr} - V_{NDE}$$

where V_{Gr} represents the allowance for degradation growth between inspections and V_{NDE} represents the allowance for potential sources of error in the measurement of the bobbin coil voltage. Further discussion of the assumptions necessary to determine the voltage repair limit are discussed in GL 95-05.

Safety analyses were performed pursuant to Generic Letter 95-05 to determine the maximum MSLB-induced primary-to-secondary leak rate that could occur without offsite doses exceeding a small fraction of 10 CFR 100 (concurrent iodine spike), 10 CFR 100 (pre-accident iodine spike), and without control room doses exceeding GDC-19. The current value of this allowable leak rate and a summary of the analyses are provided in Section 14.2.5 of the UFSAR.

The mid-cycle equation in SR 4.4.5.4.a.10.d should only be used during unplanned inspections in which eddy current data is acquired for indications at the tube support plates.

SR 4.4.5.5 implements several reporting requirements recommended by GL 95-05 for situations which the NRC wants to be notified prior to returning the SGs to service. For the purposes of this reporting requirement, leakage and conditional burst probability can be calculated based on the as-found voltage distribution rather than the projected end-of-cycle (EOC) voltage distribution (refer to GL 95-05

3/4.4.6.2 OPERATIONAL LEAKAGE (Continued)

APPLICABLE SAFETY ANALYSES (Continued)

affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere conservatively assumes a 10 gpm primary-to-secondary LEAKAGE. With the exception described below for the main steamline break (MSLB) analyzed in support of voltage-based steam generator tube repair criteria.

Primary-to-secondary LEAKAGE is a factor in the dose releases outside containment resulting from a steamline break (SLB) accident. To a lesser extent, other accidents or transients involve secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR). The leakage contaminates the secondary fluid.

The MSLB is more limiting for site radiation releases. The primary-to-secondary LEAKAGE assumed in the safety analysis for the MSLB accident is described in UFSAR Section 14.2.5. The radiological consequences of a MSLB outside of containment was reanalyzed in support of the tube support plate voltage-based repair criteria stated in SR 4.4.5.4.a.10. For this analysis, the thyroid dose was maximized at 10% of the 10 CFR Part 100 guideline of 300 rem for the co-incident iodine spike case. RCS leakage was based on projection rather than on technical specification leakage limits. The analysis indicated that offsite doses would remain within regulatory criteria with the assumed primary-to-secondary leakage (described in UFSAR Section 14.2.5) should steam generator tubes fail due to the depressurization associated with a MSLB.

A similar analysis was performed using a control room thyroid dose of 30 rem as the criterion. The control room was assumed to be manually isolated and pressurized at T=30 minutes for a period of one hour, at which time filtered emergency intake would be automatically started. The control room would be purged with fresh air at T=8 hours following release cessation. The analysis indicated that control room doses would remain within regulatory criteria with the assumed primary-to-secondary leakage (described in UFSAR Section 14.2.5) should steam generator tubes fail due to the depressurization associated with a MSLB.

LCO

RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

No pressure boundary LEAKAGE is allowed, being indicative of material deterioration. LEAKAGE of this type is

3/4.4.6.2 OPERATIONAL LEAKAGE (Continued)

LCO (Continued)

unacceptable as the leak itself could cause further deterioration, resulting in higher LEAKAGE. Violation of this LCO could result in continued degradation of the RCPB. LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. Should pressure boundary LEAKAGE occur through a component which can be isolated from the balance of the Reactor Coolant System, plant operation may continue provided the leaking component is promptly isolated from the Reactor Coolant System since isolation removes the source of potential failure.

b. Unidentified LEAKAGE

One gallon per minute (gpm) of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment sump level monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary.

c. Primary-to-Secondary LEAKAGE through Any One SG

Operating experience at PWR plants has shown that sudden increases in leak rate are often precursors to larger tube failures. Maintaining an operating LEAKAGE limit of 150 gpd per steam generator will minimize the potential for a large LEAKAGE event at power. This operating LEAKAGE limit is more restrictive than the operating LEAKAGE limit in standardized technical specifications. This provides additional margin to accommodate a tube flaw which might grow at a greater than expected rate or unexpectedly extend outside the thickness of the tube support plate. This reduced LEAKAGE limit, in conjunction with a leak rate monitoring program, provides additional assurance that this precursor LEAKAGE will be detected and the plant shut down in a timely manner.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 219 TO FACILITY OPERATING LICENSE NO. DPR-66

DUQUESNE LIGHT COMPANY
OHIO EDISON COMPANY
PENNSYLVANIA POWER COMPANY

BEAVER VALLEY POWER STATION, UNIT NO. 1

DOCKET NO. 50-334

1.0 INTRODUCTION

By letter dated June 16, 1996, as supplemented September 8 and 30, 1998, the Duquesne Light Company (the licensee) submitted a request for changes to the Beaver Valley Power Station, Unit No. 1 (BVPS-1) Technical Specifications (TSs). The requested changes would (1) make editorial changes to TS 4.4.5 and associated Bases; (2) revise the Bases for TS 3.4.6.2 to provide consistency with the BVPS-1 Updated Final Safety Analysis Report (UFSAR); and (3) revise Index Page XVII to reflect the revision of page numbers due to shifting of text by License Amendment No. 198. The September 8 and 30, 1998, letters did not change the initial proposed no significant hazards consideration determination or expand the amendment request beyond the scope of the November 18, 1998, Federal Register notice; these letters only provided updated TS pages to be consistent with the UFSAR.

2.0 EVALUATION

The proposed changes to TS 4.4.5 would (1) define the acronym "AVT" in TS 4.4.5.3.a as "All Volative Treatment," (2) clarify in TS 4.4.5.3.b. that the change in inspection frequency when tube inspection results fall into Category C-3 shall be increased to at least once-per-20-months rather than stating that the inspection frequency shall be reduced to at least once-per-20-months; (3) correct typographical errors such as in TS 4.4.5.4.a., which were introduced by the licensee in License Amendment No. 198, dated April 1, 1996; and (4) TS Tables 4.4-1 and 4.4-2 would be simplified to note that BVPS-1 has three steam generators rather than using the generic tables that were proposed for use with plants having two, three, or four steam generators. These proposed changes do not change any TS requirements, are only editorial or clarifying in nature and are therefore, acceptable.

The proposed revisions to the Bases for TS 3.4.6.2 would make these Bases consistent with the UFSAR. The NRC staff has no objection to these proposed changes.

The proposed revision to Index Page XVII would correct page numbers that were changed by License Amendment No. 198. These changes are purely administrative in nature and are acceptable.

3.0 STATE CONSULTATION

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

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 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 (63 FR 64109). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 CONCLUSION

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

Principal Contributor: D. Brinkman

Date: December 21, 1998