



UNITED STATES  
NUCLEAR REGULATORY COMMISSION

WASHINGTON, D.C. 20555-0001

June 1, 1993

Docket Nos. 50-334  
and 50-412

Mr. J. D. Sieber, Senior Vice President  
and Chief Nuclear Officer  
Nuclear Power Division  
Duquesne Light Company  
Post Office Box 4  
Shippingport, Pennsylvania 15077-0004

Dear Mr. Sieber:

SUBJECT: ISSUANCE OF AMENDMENT NOS.173 AND 52 TO FACILITY OPERATING LICENSES  
DRP-66 AND NPF-73, IN RESPONSE TO CHANGE REQUEST NOS. 205/71; STEAM  
GENERATOR TUBE SLEEVING (TAC NOS. M85081 AND M85082)

The Commission has issued the enclosed Amendment Nos.173 and 52 to  
Facility Operating License Nos. DPR-66 and NPF-73 for the Beaver Valley Power  
Station, Unit Nos. 1 and 2, in response to your application dated December 30,  
1992, as supplemented March 12, 1993.

The amendments revise the Appendix A Technical Specifications (TSs) relating  
to steam generator (SG) tubing. The amendments permit sleeving of SG tubes at  
the tube support plate and tube sheet regions in accordance with the process  
performed by the vendor Babcock and Wilcox (B&W). TS 4.4.5.2 through 4.4.5.5  
and Bases section 4.4.5 have been revised to reflect the sleeving option and  
to make minor editorial corrections.

The amendments enclosed address only that portion of your December 30, 1992,  
application which relates to the use of kinetic welded (the B&W process)  
straight sleeves with postweld heat treatment. The staff is still reviewing  
the use of laser welded sleeves (Westinghouse process).

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Mr. J. D. Sieber

- 2 -

June 1, 1993

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/

Gordon E. Edison, Senior Project Manager  
Project Directorate I-3  
Division of Reactor Projects - I/II  
Office of Nuclear Reactor Regulation

Enclosures:

- 1. Amendment No.173 to DPR-66
- 2. Amendment No.52 to NPF-73
- 3. Safety Evaluation

cc w/enclosures:  
See next page

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Mr. J. D. Sieber

- 2 -

June 1, 1993

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,



Gordon E. Edison, Senior Project Manager  
Project Directorate I-3  
Division of Reactor Projects - I/II  
Office of Nuclear Reactor Regulation

Enclosures:

1. Amendment No. 173 to DPR-66
2. Amendment No. 52 to NPF-73
3. Safety Evaluation

cc w/enclosures:  
See next page

Mr. J. D. Sieber  
Duquesne Light Company

Beaver Valley Power Station  
Units 1 & 2

cc:

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

DUQUESNE LIGHT COMPANY

OHIO EDISON COMPANY

PENNSYLVANIA POWER COMPANY

DOCKET NO. 50-334

BEAVER VALLEY POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.173  
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 December 30, 1992, as supplemented March 12, 1993, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;
  - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
  - C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;
  - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
  - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

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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. 173, 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 of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION



Walter R. Butler, Director  
Project Directorate I-3  
Division of Reactor Projects - I/II  
Office of Nuclear Reactor Regulation

Attachment:  
Changes to the Technical  
Specifications

Date of Issuance: June 1, 1993

ATTACHMENT TO LICENSE AMENDMENT NO.173

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 4-8	3/4 4-8
3/4 4-9	3/4 4-9
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 4-10e
B 3/4 4-1	B 3/4 4-1
B 3/4 4-1a	B 3/4 4-1a
B 3/4 4-2	B 3/4 4-2
B 3/4 4-2a	B 3/4 4-2a

DPR-66  
REACTOR COOLANT SYSTEM

3/4.4.5 STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

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3.4.5 Each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more steam generators inoperable, restore the inoperable generator(s) to OPERABLE status prior to increasing  $T_{avg}$  above 200°F.

SURVEILLANCE REQUIREMENTS

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4.4.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.

4.4.5.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4. Steam generator tubes shall be examined in accordance with Article 8 of Section V ("Eddy current Examination of Tubular Products") and Appendix IV to Section XI ("Eddy Current Examination of Nonferromagnetic Steam Generator Heat Exchanger Tubing") of the applicable year and addenda of the ASME Boiler and Pressure Vessel Code required by 10CFR50, Section 50.55a(g). When applying the exceptions of 4.4.5.2.a through 4.4.5.2.c, previous defects or imperfections in the area repaired by sleeving are not considered an area requiring reinspection. The tubes selected for each inservice inspection shall include at least 3 percent of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

- a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50 percent of the tubes inspected shall be from these critical areas.
- b. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:
  1. All nonplugged tubes that previously had detectable wall penetrations greater than 20 percent, and

SURVEILLANCE REQUIREMENTS (Continued)

2. Tubes in those areas where experience has indicated potential problems, and
  3. At least 3 percent of the total number of sleeved tubes in all three steam generators. A sample size less than 3 percent is acceptable provided all the sleeved tubes in the steam generator(s) examined during the refueling outage are inspected. These inspections will include both the tube and the sleeve, and
  4. A tube inspection pursuant to Specification 4.4.5.4.a.8. If any selected tube does not permit the passage of the eddy current probe for a tube or sleeve inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.
- c. The tubes selected as the second and third samples (if required by Table 4.4-2) during each inservice inspection may be subjected to a partial tube inspection provided:
1. The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found, and
  2. The inspections include those portions of the tubes where imperfections were previously found.

The results of each sample inspection shall be classified into one of the following three categories:

<u>Category</u>	<u>Inspection Results</u>
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.



SURVEILLANCE REQUIREMENTS (Continued)

3. A loss-of-coolant accident requiring actuation of the engineered safeguards, or
4. A main steam line 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.
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%
  - b. Babcock & Wilcox kinetic welded sleeve wall 40%

SURVEILLANCE REQUIREMENTS (Continued)

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 steam line 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) Babcock & Wilcox kinetic welded sleeves, BAW-2094P, Revision 1 including kinetic sleeve "tooling" and installation process parameter changes.
  - b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug or repair all tubes exceeding the plugging or repair limit) required by Table 4.4-2.

4.4.5.5 Reports

- a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged or repaired in each steam generator shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2.
- b. The complete results of the steam generator tube and sleeve inservice inspection shall be submitted to the Commission in a Special Report pursuant to Specification 6.9.2 within 12 months following the completion of the inspection. This Special Report shall include:
  1. Number and extent of tubes and sleeves inspected.
  2. Location and percent of wall-thickness penetration for each indication of an imperfection.
  3. Identification of tubes plugged or repaired.

DPR-66  
REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

- c. Results of steam generator tube inspections which fall into Category C-3 shall be reported to the Commission pursuant to Specification 6.6 prior to resumption of plant operation. The written report shall provide a description of investigations conducted to determine the cause of the tube degradation and corrective measures taken to prevent recurrence.

TABLE 4.4-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE  
INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	No			Yes		
	Two	Three	Four	Two	Three	Four
No. of Steam Generators per Unit						
First Inservice Inspection	All			One	Two	Two
Second & Subsequent Inservice Inspections	One (1)			One (1)	One (2)	One (3)

Table Notation:

- (1) The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 3 N percent of the tubes (where N is the number of steam generators in the plant) 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.
- (3) Each of the other two steam generators not inspected during the first inservice inspections shall be inspected during the second and third inspections. The fourth and subsequent inspections shall 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 Specifi- cation 6.6.	N/A	N/A

$s = \frac{3N}{n}$  Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection.

3/4.4 REACTOR COOLANT SYSTEMBASES3/4.4.1 REACTOR COOLANT LOOPS

The plant is designed to operate with all reactor coolant loops in operation and maintain DNBR above the design DNBR limit during all normal operations and anticipated transients. In Modes 1 and 2, with one reactor coolant loop not in operation, THERMAL POWER is restricted to less than or equal to 31 percent of RATED THERMAL POWER until the Overtemperature  $\Delta T$  trip is reset. Either action ensures that the DNBR will be maintained above the design DNBR limit. A loss of flow in two loops will cause a reactor trip if operating above P-7 (11 percent of RATED THERMAL POWER) while a loss of flow in one loop will cause a reactor trip if operating above P-8 (31 percent of RATED THERMAL POWER).

In MODE 3, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, due to the initial conditions assumed in the analysis for the control rod bank withdrawal from a subcritical condition, two operating coolant loops are required to meet the DNB design basis for this Condition II event.

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

The operation of one Reactor Coolant Pump or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a Reactor Coolant Pump with one or more RCS cold legs less than or equal to 275°F are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by either (1) restricting the water level in the pressurizer and thereby providing a volume for the primary coolant to expand into or (2) by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 25°F above each of the RCS cold leg temperatures.

Power is removed from the isolated loop stop valves (hot leg and cold leg) to ensure that no reactivity addition to the core can

3/4.4 REACTOR COOLANT SYSTEMBASES3/4.4.1 REACTOR COOLANT LOOPS, (continued)

occur while the loop is isolated due to inadvertent opening of the isolated loop stop valves. Isolated loop startup is limited to Modes 5 and 6 in accordance with the NRC SER on N-1 loop operation. Verification of the isolated loop boron concentration prior to opening the isolated loop stop valves provides a reassurance of the adequacy of the shutdown margin in the remainder of the system. Restoration of power to the hot leg stop valve allows opening this valve to complete the recirculation flowpath in conjunction with the relief line bypassing the cold leg stop valve and ensures adequate mixing in the isolated loop. This enables the temperature and boron concentration of the isolated loop to be brought to equilibrium with the remainder of the system. Limiting the temperature differential between the isolated loop and the remainder of the system prior to opening the cold leg stop valve prevents any significant reactivity effects due to cool water addition to the core.

Startup of an idle loop will inject cool water from the loop into the core. The reactivity transient resulting from this cool water injection is minimized by delaying isolated loop startup until its temperature is within 20°F of the operating loops. Making the reactor subcritical prior to loop startup prevents any power spike which could result from this cool water induced reactivity transient.

3/4.4.2 and 3/4.4.3 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 345,000 lbs. per hour of saturated steam at the valve set point. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss of load assuming no reactor trip until the first Reactor Protective System trip set point is reached (i.e., no credit is taken for a direct reactor trip on the loss of load) and also assuming no operation of the power operated relief valves or steam dump valves.

DPR-66  
REACTOR COOLANT SYSTEM

BASES

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3/4.4.2 and 3/4.4.3 SAFETY VALVES (Continued)

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code.

3/4.4.4 PRESSURIZER

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

3/4.4.5 STEAM GENERATORS

One OPERABLE steam generator in a non-isolated reactor coolant loop provides sufficient heat removal capability to remove decay heat after a reactor shutdown. The requirement for two OPERABLE steam generators, combined with other requirements of the Limiting Conditions for Operation ensures adequate decay heat removal capabilities for RCS temperatures greater than 350°F if one steam generator becomes inoperable due to single failure considerations. Below 350°F, decay heat is removed by the RHR system.

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those parameter limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these parameter limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant

BASES

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3/4.4.5 STEAM GENERATORS (Continued)

operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 500 gallons per day per steam generator). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 500 gallons per day per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

Wastage-type defects are unlikely with the all volatile treatment (AVT) of secondary coolant. However, even if a defect of similar type should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging or repair will be required of all tubes with imperfections exceeding the plugging or repair limit. Degraded steam generator tubes may be repaired by the installation of sleeves which span the degraded tube section. A steam generator tube with a sleeve installed meets the structural requirements of tubes which are not degraded, therefore, the sleeve is considered a part of the tube. The surveillance requirements identify those sleeving methodologies approved for use. If an installed sleeve is found to have through wall penetration greater than or equal to the plugging limit, the tube must be plugged. The plugging limit for the sleeve is derived from R.G. 1.121 analysis which utilizes a 20 percent allowance for eddy current uncertainty in determining the depth of tube wall penetration and additional degradation growth. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20 percent of the original tube wall thickness.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be reported to the Commission pursuant to Specification 6.6 prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.



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. 52  
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 December 30, 1992, as supplemented March 12, 1993, 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. 52, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto are hereby incorporated in the license. DLCO shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION



Walter R. Butler, Director  
Project Directorate I-3  
Division of Reactor Projects - I/II  
Office of Nuclear Reactor Regulation

Attachment:  
Changes to the Technical  
Specifications

Date of Issuance: June 1, 1993

ATTACHMENT TO LICENSE AMENDMENT NO.52

FACILITY OPERATING LICENSE NO. NPF-73

DOCKET NO. 50-412

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 4-11	3/4 4-11
3/4 4-12	3/4 4-12
3/4 4-13	3/4 4-13
3/4 4-14	3/4 4-14
---	3/4 4-14a
---	3/4 4-14b
3/4 4-16	3/4 4-16
B 3/4 4-3	B 3/4 4-3
---	B 3/4 4-3a

NPF-73  
REACTOR COOLANT SYSTEM

3/4.4.5 STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

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3.4.5 Each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more steam generators inoperable, restore the inoperable generator(s) to OPERABLE status prior to increasing  $T_{avg}$  above 200°F.

SURVEILLANCE REQUIREMENTS

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4.4.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.

4.4.5.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4. Steam generator tubes shall be examined in accordance with Article 8 of Section V ("Eddy Current Examination of Tubular Products") and Appendix IV to Section XI ("Eddy Current Examination of Nonferromagnetic Steam Generator Heat Exchanger Tubing") of the applicable year and addenda of the ASME Boiler and Pressure Vessel Code required by 10CFR50, Section 50.55a(g). When applying the exceptions of 4.4.5.2.a through 4.4.5.2.c, previous defects or imperfections in the area repaired by sleeving are not considered an area requiring reinspection. The tubes selected for each inservice inspection shall include at least 3 percent of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

- a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50 percent of the tubes inspected shall be from these critical areas.
- b. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

SURVEILLANCE REQUIREMENTS (Continued)

1. All nonplugged tubes that previously had detectable wall penetrations greater than 20 percent, and
  2. Tubes in those areas where experience has indicated potential problems, and
  3. At least 3 percent of the total number of sleeved tubes in all three steam generators. A sample size less than 3 percent is acceptable provided all the sleeved tubes in the steam generator(s) examined during the refueling outage are inspected. These inspections will include both the tube and the sleeve, and
  4. A tube inspection pursuant to Specification 4.4.5.4.a.8. If any selected tube does not permit the passage of the eddy current probe for a tube or sleeve inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.
- c. The tubes selected as the second and third samples (if required by Table 4.4-2) during each inservice inspection may be subjected to a partial tube inspection provided:
1. The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found, and
  2. The inspections include those portions of the tubes where imperfections were previously found.

The results of each sample inspection shall be classified into one of the following three categories:

<u>Category</u>	<u>Inspection Results</u>
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.

SURVEILLANCE REQUIREMENTS (Continued)

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.
- 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,

SURVEILLANCE REQUIREMENTS (Continued)

3. A loss-of-coolant accident requiring actuation of the engineered safeguards, or
4. A main steam line 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.
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%
  - b. Babcock & Wilcox kinetic welded sleeve wall 40%

SURVEILLANCE REQUIREMENTS (Continued)

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 steam line 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) Babcock & Wilcox kinetic welded sleeves, BAW-2094P, Revision 1 including kinetic sleeve "tooling" and installation process parameter changes.
  - b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug or repair all tubes exceeding the plugging or repair limit) required by Table 4.4-2.

4.4.5.5 Reports

- a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged or repaired in each steam generator shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2.
- b. The complete results of the steam generator tube and sleeve inservice inspection shall be submitted to the Commission in a Special Report pursuant to Specification 6.9.2 within 12 months following the completion of the inspection. This Special Report shall include:
  1. Number and extent of tubes and sleeves inspected.
  2. Location and percent of wall-thickness penetration for each indication of an imperfection.
  3. Identification of tubes plugged or repaired.

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SURVEILLANCE REQUIREMENTS (Continued)

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- c. Results of steam generator tube inspections which fall into Category C-3 shall be reported to the Commission pursuant to Specification 6.6 prior to resumption of plant operation. The written report shall provide a description of investigations conducted to determine the cause of the tube degradation and corrective measures taken to prevent recurrence.

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 50.72 (b)(2) of 10 CFR Part 50	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 50.72 (b) (2) of 10 CFR Part 50	N/A	N/A

$s = \frac{9}{n}^2$  Where n is the number of steam generators inspected during an inspection.

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3/4.4.5 STEAM GENERATORS (Continued)

decay heat removal capabilities for RCS temperatures greater than 350°F if one steam generator becomes inoperable due to single failure considerations. Below 350°F, decay heat is removed by the RHR system.

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those parameter limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these parameter limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 500 gallons per day per steam generator). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 500 gallons per day per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

Wastage-type defects are unlikely with the all volatile treatment (AVT) of secondary coolant. However, even if a defect of similar type should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging or repair will be required of all tubes with imperfections exceeding the plugging or repair limit. Degraded steam generator tubes may be

BASES

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3/4.4.5 STEAM GENERATORS (Continued)

repaired by the installation of sleeves which span the degraded tube section. A steam generator tube with a sleeve installed meets the structural requirements of tubes which are not degraded, therefore, the sleeve is considered a part of the tube. The surveillance requirements identify those sleeving methodologies approved for use. If an installed sleeve is found to have through wall penetration greater than or equal to the plugging limit, the tube must be plugged. The plugging limit for the sleeve is derived from R.G. 1.121 analysis which utilizes a 20 percent allowance for eddy current uncertainty in determining the depth of tube wall penetration and additional degradation growth. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20 percent of the original tube wall thickness.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be reported to the Commission pursuant to Specification 6.6 prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION  
RELATED TO AMENDMENT NO. 173 TO FACILITY OPERATING LICENSE NO. DPR-66  
AMENDMENT NO. 52 TO FACILITY OPERATING LICENSE NO. 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

In a letter of December 30, 1992, as supplemented March 12, 1993, Duquesne Light Company (the licensee) submitted an amendment request to the licenses for Beaver Valley Power Station, Units 1 and 2, proposing steam generator (SG) tube sleeving in accordance with the Westinghouse and the Babcock and Wilcox processes. The March 12, 1993, supplement provided additional and clarifying information in support of the original request and did not change the NRC's proposed finding of no significant hazards consideration determination.

The Westinghouse process consists of a laser welding technique to secure a sleeve inside a steam generator tube to serve as the primary barrier within a degraded tube. The technical justification supporting the Westinghouse process is given in WCAP-13483, Rev. 1, "Beaver Valley Units 1 and 2 Westinghouse Series 51 Steam Generator Sleeving Report," which is still under review. Therefore, this safety evaluation applies only to the Babcock and Wilcox process described below.

The Babcock and Wilcox (B&W) process uses a kinetic welding technique to secure the sleeves inside SG tubes. The technical justification supporting the B&W process is given in BAW-2094PA, Rev. 1, "Recirculating Steam Generators Kinetic Sleeve Qualification for 7/8 Inch OD Tubes," a topical report that was previously found acceptable for referencing by the NRC.

2.0 BACKGROUND

Tubes in an operating pressurized water reactor (PWR) SG can be degraded by such corrosion phenomena as wastage, pitting, intergranular attack, stress corrosion cracking, crevice corrosion, as well as by such other phenomena as denting and vibration wear. Tubes that become excessively degraded reduce the integrity of the primary-to-secondary pressure boundary and must be removed from service or repaired. Historically, SG tubes that have degraded below a

calculated minimum wall thickness value, termed the "plugging limit," were plugged at both the inlet and outlet ends of the tube. Installing plugs in SG tubes decreases the heat transfer surface area available for reactor core cooling.

Recently, SG tubes experiencing localized degradation have been fitted with sleeves over the degraded area to reestablish the integrity of the primary-to-secondary pressure boundary. The sleeves are expanded and sealed inside the tubes to provide an acceptable leak resistance, load-carrying path. Installing sleeves inside the SG tubes slightly reduces the heat transfer capability and slightly restricts the primary flow through the sleeved tubes.

The licensee has performed eddy current testing (ECT) to measure the degradation of the SG tube and developed similar testing for sleeves. When ECT measurements exceed the plugging limit in a tube, the tube can be either sleeved or plugged; and when the measurements exceed the plugging limit in a sleeve, the tube is plugged. The plugging limit conservatively accounts for the uncertainties in ECT measurements and contains an additional margin for expected or postulated degradation during service. The amendment identifies the plugging limit for the B&W process as 40 percent of the sleeve wall.

Since 1983, B&W has installed kinetically welded sleeves in recirculating SG tubes located in Europe and the United States. The sleeves are of three basic designs: (1) kinetic welds in both tube-to-sleeve joints (joints) completely within the tube sheet, (2) kinetic or rolled joints within the tube sheet and a freespan kinetic weld above the tube sheet, and (3) freespan kinetic welds on each side of a tube support plate. The kinetic welds are given a postweld heat treatment (PWHT) for stress relief.

### 3. DESCRIPTION AND EVALUATION OF BAW-2094PA, Rev. 1

The proposed amendment would modify a portion of the Technical Specifications that specifies that degraded tubes must be removed from service by plugging the tubes. The modification would allow the option of sleeving to repair degraded tubes. The sleeving process consists of cleaning the tube and sleeve surfaces at the contact areas, expanding the sleeves against the tubes, attaching the sleeves to the tubes by welding or hard rolling, inspecting the assembly to confirm weld acceptability and to establish an inspection baseline. The design, installation, and inspection of the sleeves will be done in accordance with American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III (Code).

By adherence to industry standards, the pressure boundary integrity is preserved. The structural analysis has been performed for SG pressure, temperature, and to determine minimum primary pressure-boundary wall thickness. The structural integrity has been demonstrated by means of fatigue testing, tensile testing, thermal cycling, and leak testing. Corrosion resistance of the sleeves has been demonstrated by means of corrosion testing.

The technical justification supporting the B&W process is given in BAW-2094PA, Rev. 1, a topical report that was previously found acceptable for referencing in licensing applications by the NRC; therefore, the staff did not conduct a detailed review. The staff concluded that the licensee's use of the B&W sleeves will be bounded by the parameters stated in BAW-2094PA, Rev. 1, which does not address bowed sleeves. The licensee has committed to PWHT of the welded joints.

Based on the preceding analysis, the NRC staff concludes that the operation of Beaver Valley Power Station, Units 1 and 2, in accordance with the proposed amendments to include the repair of SG tubes using kinetic welded sleeves that are installed with the B&W process is acceptable.

The staff also finds that the use of the eddy current equipment and techniques as described in BAW-2094PA, Rev. 1, or their equivalent, to be currently acceptable. The licensee has committed to utilize advanced state-of-the-art eddy current inspection techniques as they are developed and verified, after a 10 CFR 50.59 review.

#### 4.0 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.

#### 5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the 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 (58 FR 17912). Accordingly, the amendments meet 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 amendments.

#### 6.0 CONCLUSION

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

Principal Contributor: D. Naujock  
Date: June 1, 1993