December 19, 1984

Docket No. 50-334

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Mr. J. J. Carey, Vice President
Nuclear Division
Duquesne Light Company
Post Office Box 4
Shippingport, PA 15077

Dear Mr. Carey:

SUBJECT: Issuance of Amendment (TAC No. 55104)

The Commission has issued the enclosed Amendment No.86 to Facility Operating License No. DPR-66 for the Beaver Valley Power Station, Unit No. 1. The amendment consists of changes to the Technical Specifications in response to your application dated May 21, 1984.

The amendment changes the Technical Specifications for Beaver Valley Unit No. 1 to reflect the revised capsule removal schedule recommended by Westinghouse Topical Report WCAP-9860. The Bases have also been revised to reference 10 CFR 50 Appendix H for capsule removal and evaluation. The changes would bring the surveillance schedule into conformance with Appendix H, "Reactor Vessel Material Surveillance Program Requirements" of 10 CFR 50.

In addition, reference to the augmented inservice inspection program prescribed for the first three refueling outages are deleted since the program has been completed.

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

Sincerely,

Peter S. Tam, Project Manager
Operating Reactors Branch No. 1
Division of Licensing

Enclosures:
1. Amendments No.86 to DPR-66
2. Safety Evaluation

cc w/enclosures:
See next page

ORB#1:DL	ORB#1:DL	C-ORB#1:DL	OELD	AD:OR:DL
CParrish	PTam	SVarga		GLainas
12/11/84	12/12/84	12/17/84	12/17/84	12/19/84

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P PDR

Mr. J. J. Carey
Duquesne Light Company

Beaver Valley Power Station
Unit 1

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

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.86
License No. DPR-66

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

   A. The application for amendment by Duquesne Light Company, Ohio Edison Company, and Pennsylvania Power Company (the licensees) dated May 21, 1984, 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 Appendices A and B, as revised through Amendment No. 86 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This amendment is effective on issuance, to be implemented no later than 30 days after issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Attachment:
Changes to the Technical Specifications

Date of Issuance:
December 19, 1984

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 86 TO FACILITY OPERATING LICENSE NO. DPR-66

DOCKET NO. 50-334

Revise Appendix A as follows:

Remove Pages	Insert Pages
3/4 4-26	3/4 4-26
3/4 4-28	3/4 4-28
3/4 4-29	3/4 4-29
B 3/4 4-9	B 3/4 4-9

TABLE 4.4-3

REACTOR VESSEL MATERIAL IRRADIATION SURVEILLANCE SCHEDULE

Capsule	Vessel Location	Lead Factor	Withdrawal Time (EFPY)
V	165°	1.37	1 EFPY (Removed)
U	65°	.89	3 EFPY
W	245°	.89	6 EFPY
Y	295°	.89	15 EFPY
X	285°	1.37	EOL
T	55°	.58	Standby
Z	305°	.58	Standby
S	45°	.43	Standby

3/4.4.10 STRUCTURAL INTEGRITY

ASME CODE CLASS 1, 2 and 3 COMPONENTS

LIMITING CONDITION FOR OPERATION

3.4.10 The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification 4.4.10.

APPLICABILITY: ALL MODES

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) from service.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.10 Each ASME Code Class 1, 2, and 3 component shall be demonstrated OPERABLE in accordance with Specification 4.0.5.

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The second portion of the heatup analysis concerns the calculation of pressure-temperature limitations for the case in which a 1/4T deep outside surface flaw is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and thus tend to reinforce any pressure stresses present. These thermal stresses are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Since the thermal stresses at the outside are tensile and increase with increasing heatup rates, each heatup rate must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced as follows: A composite curve is constructed based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration. The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist wherein, over the course of the heatup ramp, the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion. Then, composite curves for the heatup rate data and the cooldown rate data are adjusted for possible errors in the pressure and temperature sensing instruments by the values indicated on the respective curves.

The actual shift in NDTT of the vessel material will be established periodically during operation by removing and evaluating, in accordance with 10 CFR 50 Appendix H,reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 86 TO FACILITY OPERATING LICENSING NO. DPR-66

DUQUESNE LIGHT COMPANY

OHIO EDISON COMPANY

PENNSYLVANIA POWER COMPANY

BEAVER VALLEY POWER STATION, UNIT NO. 1

DOCKET NO. 50-334

INTRODUCTION

By letter dated May 21, 1984, Duquesne Light Company (the licensee) proposed administrative changes to the Technical Specifications (TS) set forth in Appendix A to the license for the purpose of 1) revising the Reactor Pressure Vessel Material Irradiation Surveillance Schedule shown in Table 4.4-3; 2) referencing 10 CFR 50 Appendix H rather than ASTM E185-70 for surveillance specimens in bases 3/4.4.9 on page B 3/4 4-9 (Appendix H now endorses ASTM E185-73, -79 and -82); and 3) eliminating surveillance requirements for the first three refueling outages that are no longer applicable for inspection of the reactor vessel nozzles (4.4.10 a,b, and c on page 3/4 4-29).

No physical changes to the facility or equipment will be made as a result of this revision. The revision represents administrative changes required to up date the Technical Specification to 10 CFR 50 Appendix H and updated ASTM E185-82 requirements.

EVALUATION AND DISCUSSION

The function of the Reactor Pressure Vessel Material Irradiation Program is to evaluate the toughness changes of the vessel belt line materials and weld metal when exposed to fast neutron fluence. The surveillance program is designed to show that the reactor pressure vessel coolant boundary is designed with sufficient margin to ensure that, when stressed under operating, maintenance, testing, and postulated accident conditions: (1) the boundary behaves in a nonbrittle manner, and (2) the probability of rapidly propagating fracture is minimized as required by general design criterion 31 of Appendix A. This is accomplished throughout the service life of the vessel by testing in-vessel samples and calculation of changes in fracture toughness of the reactor vessel materials caused by neutron radiation and the thermal environment.

The licensee's request for a change in the surveillance schedule is based on 1) analysis of the first capsule (V) removed from the reactor vessel and reported in "Analysis of Capsule V from the Duquesne Light Company, Beaver Valley Unit No. 1 Reactor Vessel Radiation Surveillance Program", WCAP-9860, and 2) changes in the standard practice for conducting surveillance tests as indicated in ASTM E185-82.

WCAP-9860 indicates that the capsule received a fast-neutron fluence of 2.55 x 10^{18} n/cm^2 compared to the calculated value of 2.58 x 10^{18} n/cm^2 and that based upon these fluence measurements the vessel ¼ thickness fluence after 1.02 effective-full-power years (EFPY) is 1.14 x 10^{18} n/cm^2 compared to a calculated fluence of 1.15 x 10^{18} n/cm^2. The agreement between fluence calculations and fluence measurements is excellent which provides reliability for analytical projections of future vessel toughness.

Revision 2 of Regulatory Guide 1.99 indicates that the fluence factor trend curve, considering the effects of nickel, should be steeper at lower fluences and flatter at the higher fluences.

The staff's review of the acceptability of the licensee's proposed changes to Table 4.4-3 utilized applicable portions of WCAP-9860, but does not constitute a technical review of WCAP-9860 in its entirety.

The change requested on page B 3/4 4-9 is in referencing 10 CFR 50, Appendix H, rather than ASTM E185-70 as the applicable document for removing and evaluating irradiation surveillance specimens. This change constitutes a clarification of the requirements.

The changes to TS Surveillance requirement 4.4.10 constitute an administrative change in the elimination of inspection requirements for the first three refueling outages which have been completed. The indications detected during the inspection of the subject reactor vessel nozzle attachments, following the third refueling outage and reported in Westinghouse Electric Corporation letter to DLC IS-GCE-022, were evaluated as being acceptable with no corrective action mandated.

Based on review of applicable documents, the staff finds that the proposed amendment to the DLC Technical Specification represents administrative changes required to update the Technical Specifications consistent with 10 CFR 50 code of Federal Regulations and involves no physical changes in the plant safety related systems, structures or components. The staff therefore finds the proposed amendment to be acceptable.

ENVIRONMENTAL CONSIDERATION

This amendment involves a change in administrative procedure and requirements. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Section 51.22 (c)(10) pursuant to 10 CFR 51.22(b) and no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

CONCLUSION

We have 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; and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: December 19, 1984

Principal Contributor:

Samuel D. Reynolds Jr.