July 27, 1977

Docket No. 50-334

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Duquesne Light Company
ATTN: Mr. C. N. Dunn, Vice President
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Pittsburgh, Pennsylvania 15219

Gentlemen:

The Commission has issued the enclosed Amendment No. 10 to Facility Operating License No. DPR-66 for the Beaver Valley Power Station Unit No. 1. This amendment consists of changes to the Technical Specifications in response to your application dated March 25, 1977.

This amendment relates to a revised enthalpy rise hot channel factor ($F^N_{\Delta H}$) Technical Specification for Beaver Valley Unit No. 1 to account for new fuel rod bow information.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Enclosures:
1. Amendment No. 10
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

Const. 1

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Form AEC-318 (Rev. 9-53) AECM 0240

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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. 10
License No. DPR-66

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

   A. The application for amendment by Duquesne Light Company, filed on behalf of itself, Ohio Edison Company, and Pennsylvania Power Company (the licensees), dated March 25, 1977, 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, Facility Operating License No. DPR-66 is hereby amended as indicated below and by changes to the Technical Specifications as indicated in the attachment to this license amendment:

   A. Revise paragraph 2.C.(2) to read as follows:

      (2) Technical Specifications

      The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 10 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

   B. Delete paragraph 2.C.(7) and renumber the remaining paragraph 2.C.(7).

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Attachment:
Changes to the Technical Specifications

Date of Issuance: July 27, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 10

FACILITY OPERATING LICENSE NO. DPR-66

DOCKET NO. 50-334

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Pages

3/4 2-8
B 3/4 2-4
B 3/4 2-5

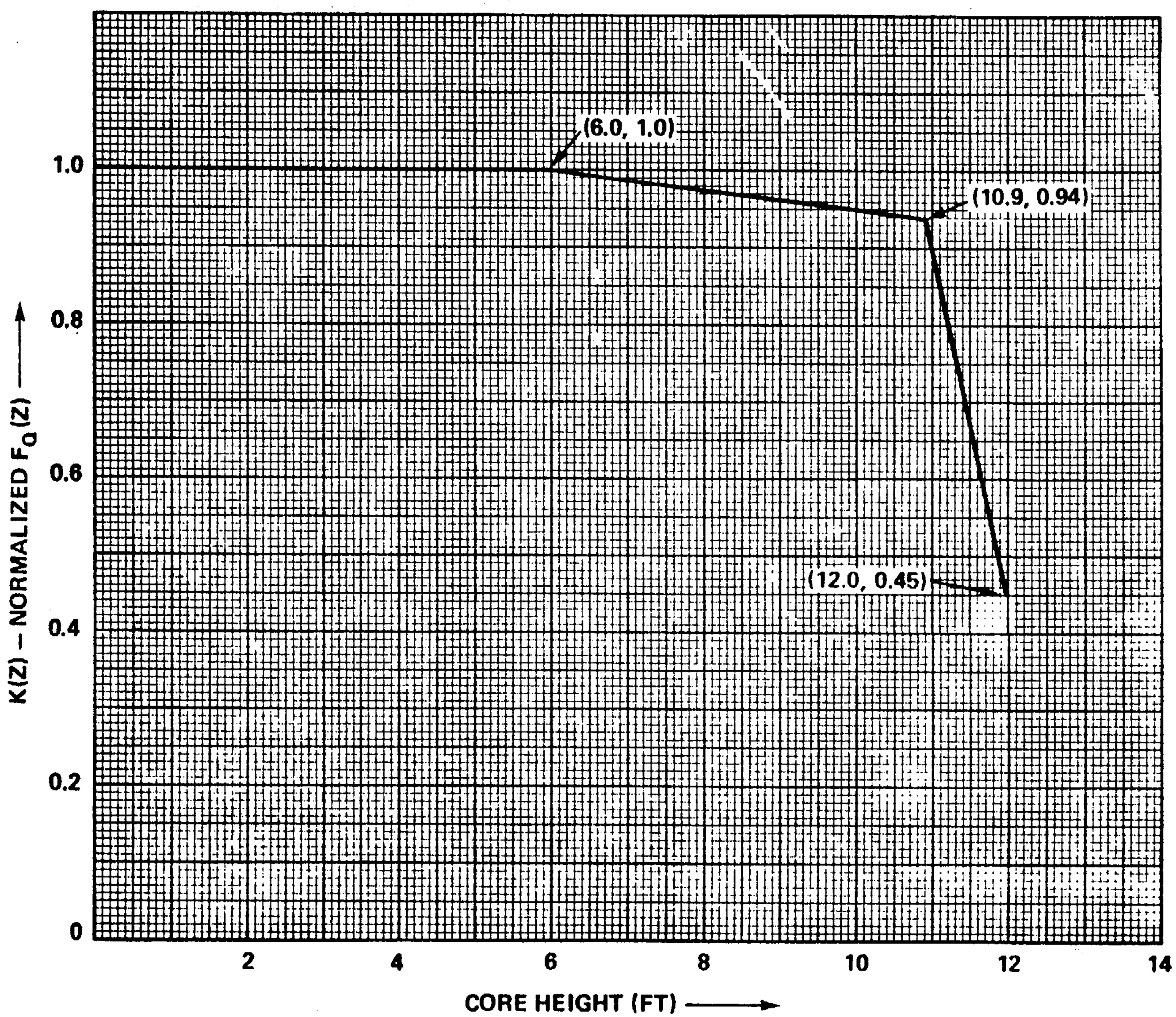

FIGURE 3.2-2

K(Z) – NORMALIZED $F_Q(Z)$
AS A FUNCTION OF CORE HEIGHT

POWER DISTRIBUTION LIMITS

NUCLEAR ENTHALPY HOT CHANNEL FACTOR - $F^N_{\Delta H}$

LIMITING CONDITION FOR OPERATION

3.2.3 $F^N_{\Delta H}$ shall be limited by the following relationship:

$$F^N_{\Delta H} \leq 1.55 \left[1.0 - \frac{0.095\ Bu}{15{,}000\ MWD/MTU}\right] [1.0 + 0.2\ (1-P)] \text{ for } \leq 15{,}000 \text{ MWD/MTU burnup fuel}$$

$$F^N_{\Delta H} \leq 1.36\ [1.0 + 0.2\ (1-P)] \text{ for } > 15{,}000 \text{ MWD/MTU burnup fuel}$$

where $P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$

Bu = FUEL REGION AVERAGE BURNUP IN MWD/MTU

APPLICABILITY: MODE 1

ACTION:

With $F^N_{\Delta H}$ exceeding its limit:

a. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to $\leq$ 55% of RATED THERMAL POWER within the next 4 hours,

b. Demonstrate thru in-core mapping that $F^N_{\Delta H}$ is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours, and

c. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION may proceed provided that $F^N_{\Delta H}$ is demonstrated through in-core mapping to be within its limit at a nominal 50% of RATED THERMAL POWER prior to exceeding this THERMAL POWER, at a nominal 75% of RATED THERMAL POWER prior to exceeding this THERMAL power and within 24 hours after attaining 95% or greater RATED THERMAL POWER.

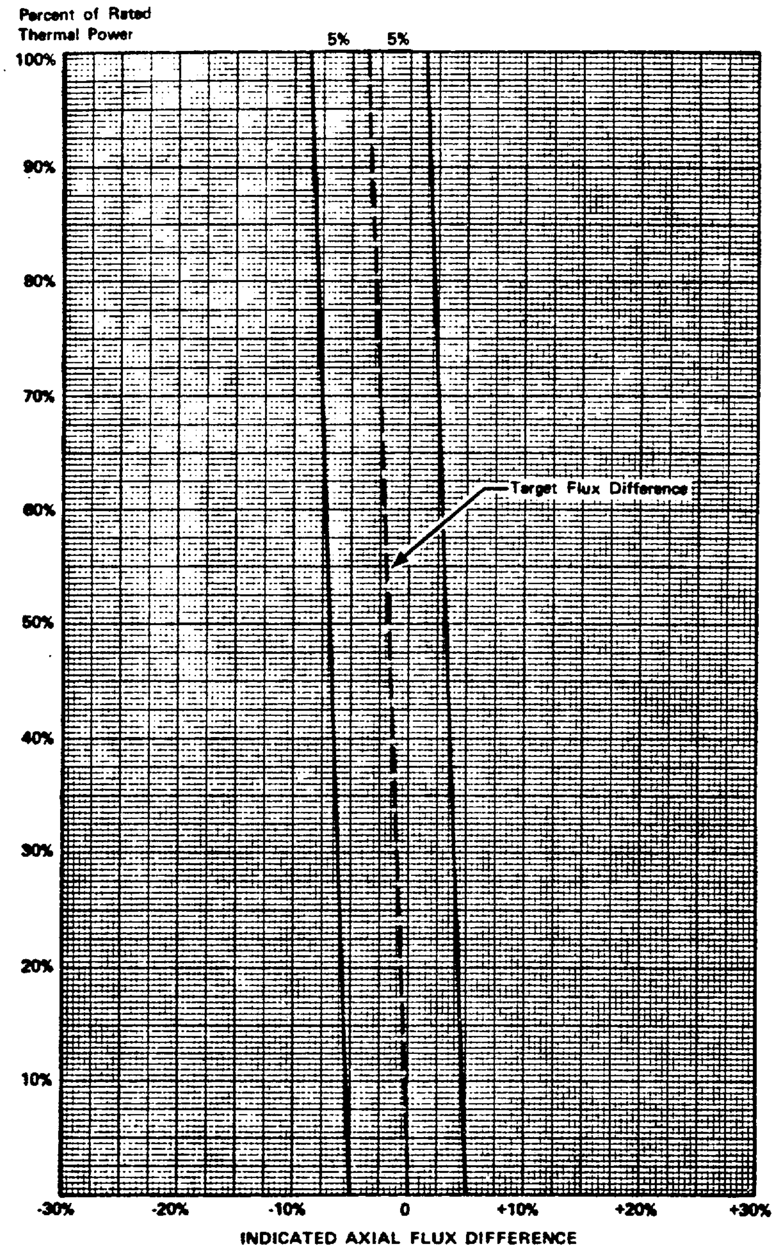

Figure B 3/4 2-1 TYPICAL INDICATED AXIAL FLUX DIFFERENCE VERSUS THERMAL POWER AT BOL

FIGURE B 3/4. 2-1

3/4.2.2 and 3/4.2.3 HEAT FLUX AND NUCLEAR ENTHALPY HOT CHANNEL FACTORS- $F_Q(Z)$ and $F^N_{\Delta H}$

The limits on heat flux and nuclear enthalpy hot channel factors ensure that 1) the design limits on peak local power density and minimum DNBR are not exceeded and 2) in the event of a LOCA the peak fuel clad temperature will not exceed the ECCS acceptance criteria limit of 2200°F.

Each of these hot channel factors are measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to insure that the hot channel factor limits are maintained provided:

a. Control rod in a single group move together with no individual rod insertion differing by more than $\pm$ 12 steps from the group demand position.

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.5.

c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained.

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE is maintained within the limits.

e. The part length rods are fully withdrawn from the core.

The relaxation in $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits. $F^N_{\Delta H}$ will be maintained within its limits provided conditions a thru d above, are maintained.

When an F_Q measurement is taken, both experimental error and manufacturing tolerance must be allowed for. 5% is the appropriate allowance for a full core map taken with the incore detector flux mapping system and 3% is the appropriate allowance for manufacturing tolerance.

BASES

a. abnormal perturbations in the radial power shape, such as from rod misalignment, effect $F^N_{\Delta H}$ more directly than F_Q,

b. although rod movement has a direct influence upon limiting F_Q to within its limit, such control is not readily available to limit $F^N_{\Delta H}$, and

c. errors in prediction for control power shape detected during startup physics tests can be compensated for in F_Q by restriting axial flux distributions. This compensation for $F^N_{\Delta H}$ is less readily available.

A recent evaluation of DNB and test data from experiments of fuel rod bowing in subchannels containing thimble cells has identified that it is appropriate to impose a penalty factor to the accident analyses DNBR results. This evaluation has not been completed, but in order to assure that this effect is accommodated in a conservative manner, an interim thimble cell rod bow penalty as a function of fuel burnup, is applied to the measured values of the enthalpy rise hot channel factor, $F^N_{\Delta H}$.

3/4.2.4 QUADRANT POWER TILT RATIO

The quadrant power tilt ratio limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during startup testing and periodically during power operation.

The limit of 1.02 at which corrective action is required provides DNB and linear heat generation rate protection with x-y plane power tilts. A limiting tilt of 1.025 can be tolerated before the margin for uncertainty in F_Q is depleted. The limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The two hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the power by 3 percent for each percent of tilt in excess of 1.0.

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient & accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of 1.30 throughout each analyzed transient.

The 12 hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The 18 month periodic measurement of the RCS total flow rate is adequate to detect flow degradation and ensure correlation of the flow indication channels with measured flow such that the indicated percent flow will provide sufficient verification of flow rate on a 12 hour basis.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 10 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

Introduction

By letter dated March 25, 1977, Duquesne Light Company (the licensee) requested an amendment to Facility Operating License No. DPR-66. The purpose of the request was to revise the enthalpy rise hot channel factor ($F^N_{\Delta H}$) Technical Specifications for Beaver Valley Power Station Unit No. 1 to account for new fuel rod bow information.

Discussion

On August 9, 1976, Westinghouse Electric Corporation presented data to the NRC staff which showed that previously developed methods for accounting for the effect of fuel rod bowing on departure from nucleate boiling may not contain adequate thermal margin when unheated rods (such as thimble tubes) are present. We have evaluated the impact of the Westinghouse data on all operating pressurized water reactors (PWR's). Models for treating the effects of fuel rod bowing on thermal-hydraulic performance have been derived for all PWR's. The models are based on the propensity of the individual fuel designs to bow and on the thermal analysis methods used to predict the coolant conditions for both normal operation and anticipated transients. As a result of these evaluations, we have concluded that for some facilities the current technical specification operating limits do not provide sufficient thermal margin. In these cases, additional thermal margin is required to assure, with high confidence that departure from nucleate boiling (DNB) does not occur during anticipated transients.

Background

In 1973 Westinghouse Electric presented to the NRC staff the results of experiments in which a 4 x 4 bundle of electrically heated fuel rods was tested to determine the effect of fuel rod bowing to contact on the thermal margin. The departure from nucleate boiling ratio (DNBR) is a measure of the thermal margin available prior to the point at which DNB occurs. The tests were performed at conditions representative of PWR coolant conditions. The results of these experiments showed that, for the highest power density at the highest coolant pressure expected in a Westinghouse reactor, the DNBR reduction due to a heated rod bowed to the point of contact with adjacent heated rods was approximately 8%.

Fuel bundle coolant mixing and heat transfer computer programs such as COBRA IIIC and THINC-IV were able to predict the results of these experiments. Because the end point could be predicted, i.e., the DNBR reduction at contact, there was confidence that the DNBR reduction due to partial rod bow, that is, rod bow to a point less than contact with the adjacent rod, could also be correctly predicted.

On August 9, 1976, Westinghouse met with the NRC staff to discuss further experiments with the same configuration of fuel bundle (4 x 4) using electrically heated rods. However, for this set of experiments one of the center 4 fuel rods was replaced by an unheated tube of the same size as a Westinghouse thimble tube. This new test configuration was tested over the same range of power, flow and pressure as the earlier tests. However, with the unheated, larger diameter rod the reduction in DNBR was much larger than in the earlier (1973) tests.

The data consisted of points corresponding to no intentional bowing (that is, a certain amount of bowing due to tolerances cannot be prevented) and to contact. No data were taken at partial clearance reductions between rods.

We attempted to calculate the Westinghouse results with the COBRA IIIC computer code but could not obtain agreement with the new data. Westinghouse was also unable to obtain agreement between their experimental results and the THINC-IV computer code.

On August 19, 1976, Combustion Engineering (CE) presented results of similar experiments to the NRC staff. These tests were performed using a 21-rod bundle of electrically heated rods and an unheated guide tube. Results were presented for not only the case of full contact, but also the case of partial bowing.

Both sets of data (Westinghouse and CE) showed similar effects due to variations in coolant conditions. For both cases, the DNBR reduction became greater as the coolant pressure and the rod power increased.

Because both sets of data showed that plant thermal margins might be less than those intended, we derived an interim model to conservatively predict the DNBR reduction. Since the data with unheated rods could not be predicted by existing analytical methods, empirical models were derived. These models are contained in Reference 1 and were transmitted to the licensee in our letter dated March 9, 1977. Using these empirical models, we calculated DNBR reductions to be applied to all operating pressurized water reactors. We have permitted the calculated reduction by DNBR to be offset by certain available thermal margins on a case-by-case basis. These "credits" may be either generic to a given fuel design or plant specific. The derivation of the Beaver Valley Power Station Unit No. 1 DNBR reduction due to row bow is described in Section 4.1 of Reference 1.

Evaluation

The licensee has proposed Technical Specification changes which would provide for additional DNBR margin to offset the reduction in DNBR due to rod bow. The credits which the licensee has taken to offset the DNBR penalty are:

$F^N_{\Delta H}$ limits as listed in Table 4.2 in Reference 1

We have evaluated the proposed Technical Specification changes using the procedure given in Reference 1 and concluded that the reduction in $F^N_{\Delta H}$ limits and credits for excess flow are adequate to offset the loss of thermal margin indicated by the recent Westinghouse rod bow data; and, therefore, the proposed changes are acceptable.

Environmental Conclusions

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR §51.5(d)(4) that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

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1 Revision 1 to Interim Safety Evaluation Report on Effects of Fuel Rod Bowing on Thermal Margin Calculations, dated February 16, 1977

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) 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: July 27, 1977

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-334

DUQUESNE LIGHT COMPANY

OHIO EDISON COMPANY

PENNSYLVANIA POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 10 to Facility Operating License No. DPR-66, issued to Duquesne Light Company, Ohio Edison Company and Pennsylvania Power Company (the licensees), which revised Technical Specifications for operation of the Beaver Valley Power Station Unit No. 1 (the facility) located in Beaver County, Pennsylvania. The amendment is effective as of its date of issuance.

This amendment relates to a revised enthalpy rise hot channel factor ($F^N_{\Delta H}$) Technical Specification for Beaver Valley Power Station Unit No. 1 to account for new fuel rod bow information.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement, negative declaration or environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated March 25, 1977, (2) Amendment No. 10 to License No. DPR-66, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Beaver Area Memorial Library, 100 College Avenue, Beaver, Pennsylvania. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 27th day of July 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors