Duquesne Light

Nuclear Construction Division
Robinson Plaza, Building 2, Suite 210
Pittsburgh, PA 15205

2NRC-4-102
(412) 787-5141
(412) 923-1960
Telecopy (412) 787-2629
July 12, 1984

United States Nuclear Regulatory Commission
Washington, DC 20555

ATTENTION: Mr. George W. Knighton, Chief
Licensing Branch 3
Office of Nuclear Reactor Regulation

SUBJECT: Beaver Valley Power Station - Unit No. 2
Docket No. 50-412
Response to Outstanding Issues

Gentlemen:

This letter forwards responses to the issues listed below. The following items are attached:

Attachment 1: Response to Outstanding Issue 44 of the Beaver Valley Power Station Unit No. 2 Draft Safety Evaluation Report.

Attachment 2: Response to Outstanding Issue 45 of the Beaver Valley Power Station Unit No. 2 Draft Safety Evaluation Report.

Attachment 3: Response to Outstanding Issue 47 of the Beaver Valley Power Station Unit No. 2 Draft Safety Evaluation Report.

Attachment 4: Response to Outstanding Issue 73 of the Beaver Valley Power Station Unit No. 2 Draft Safety Evaluation Report.

DUQUESNE LIGHT COMPANY

SUBSCRIBED AND SWORN TO BEFORE ME THIS
11th DAY OF July, 1984.

Anita Elaine Reiter
Notary Public
My Commission expires October 29, 1986

By E. J. Woolever
E. J. Woolever
Vice President

KAT/wjs
Attachments

cc: Mr. H. R. Denton, Director NRR (w/attachments)
Mr. D. Eisenhut, Director Division of Licensing (w/attachments)
Ms. M. Ley, Project Manager (w/attachments)
Mr. M. Licitra, Project Manager (w/attachments)
Mr. G. Walton, NRC Resident Inspector (w/attachments)

B001
1/1

COMMONWEALTH OF PENNSYLVANIA)
) SS:
COUNTY OF ALLEGHENY)

On this 11th day of July, 1984, before me, a Notary Public in and for said Commonwealth and County, personally appeared E. J. Woolever, who being duly sworn, deposed and said that (1) he is Vice President of Duquesne Light, (2) he is duly authorized to execute and file the foregoing Submittal on behalf of said Company, and (3) the statements set forth in the Submittal are true and correct to the best of his knowledge.

Anita Elaine Reiter
Notary Public

ANITA ELAINE REITER, NOTARY PUBLIC
ROBINSON TOWNSHIP, ALLEGHENY COUNTY
MY COMMISSION EXPIRES OCTOBER 20, 1986

ATTACHMENT 1

Response to Outstanding Issue 44 of the
Beaver Valley Power Station Unit No. 2
Draft Safety Evaluation Report

Draft SER Section 4.4.8: Conclusion (excerpt)

Address the concerns regarding the effect of rod bow on DNBR as described in Section 4.4.4.1 of the SER.

Response:

The phenomenon of fuel rod bowing, as described in WCAP-8691, "Fuel Rod Bow Evaluation," must be accounted for in the DNBR safety analysis of Condition I and Condition II events for each plant application. Applicable generic credits for margin resulting from retained conservatism in the evaluation of DNBR and/or margin obtained from measured plant operating parameters (such as $F^N_{\Delta H}$ or core flow) -- which are less limiting than those required by the plant safety analysis -- can be used to offset the effect of rod bow.

The safety analysis for Beaver Valley Unit 2 maintained sufficient margin (9.1 percent)* to accommodate full and low flow DNBR penalties identified in References A and B (< 3 percent for the worst case which occurs at a burnup of 33,000 MWD/MTU).

The fuel rod diameter, pitch, and bowing variation (including inpile effects) was considered in the preparation of the THINC input values such as axial flow area, equivalent hydraulic diameter, and lateral cross-flow area for the hot channel. This effect (pitch reduction) was used as part of the margin to offset rod bow penalties.

The maximum rod bow penalties accounted for in the design safety analysis are based on an assembly average burnup of 33,000 MWD/MTU. At burnups greater than 33,000 MWD/MTU, credit is taken for the effect of $F^N_{\Delta H}$ burndown, due to the decrease in fissionable isotopes and the buildup of fission product inventory, and no additional rod bow penalty is required.

---

* Design Limit DNBR of 1.30 vs. 1.28
Grid Spacing (K_s) of 0.046 vs. 0.059
Thermal Diffusion Coefficient of 0.038 vs. 0.051
DNB Multiplier of 0.865 vs. 0.88
Pitch Reduction

Reference A: "Partial Response to Request No. 1 for Additional Information on WCAP-8691, Revision 1," letter, E. P. Rahe, Jr. (Westinghouse) to J. R. Miller (NRC), NS-EPR-2515, dated October 9, 1981

Reference B: "Remaining Response to Request No. 1 for Additional Information on WCAP-8691, Revision 1," letter, E. P. Rahe, Jr. (Westinghouse) to R. J. Miller (NRC), NS-EPR-2572, dated March 16, 1982

ATTACHMENT 2

Response to Outstanding Issue 45 of the
Beaver Valley Power Station Unit No. 2
Draft Safety Evaluation Report

Draft SER Section 4.4.3.2: Crud Deposition

Operating experience on two pressurized water reactors (not of Westinghouse design) indicate that a significant reduction in the core flow rate can occur over a relatively short period of time as a result of crud deposition on the fuel rods. In establishing the Technical Specifications for Beaver Valley Unit 2, we will require provisions to assure that the minimum design flow rates are achieved. We also require that the applicant provide a description of the flow measurement capability for Beaver Valley Unit 2 as well as a description of the procedures to measure flow.

Response:

Operating experience to date has indicated that a flow resistance-allowance for possible crud deposition is not required. There has been no detectable long-term flow reduction reported at any Westinghouse plant. Inspection of the inside surfaces of steam generator tubes removed from operating plants has confirmed that there is no significant surface deposition that would affect system flow. The small piping friction contribution to the total system resistance and the lack of significant deposition on piping near steam generator nozzles support the conclusion that an allowance for piping deposition is not necessary. The effect of crud enters into the calculation of core pressure drop through the fuel rod frictional component by use of a surface roughness factor. Present analyses utilize a surface roughness value which is a factor of three greater than the best estimate obtained from crud sampling from several operating Westinghouse reactors.

The operator has at his disposal several methods of detecting significant RCS flow reduction; these are:

a. Flow meter on each RCS loop,

b. If operating in an automatic control rod mode (T_c held constant) a reduction in reactor power would be present for significant reductions in RCS flow,

c. If operating in a manual control rod mode (power held constant) an increase in ΔT across the core would be present for significant reductions in flow,

d. Local changes in flow could be indicated by incore flux maps (assuming significant changes in local power), and

e. Core exit thermocouple readings.

Technical Specifications are being prepared for Beaver Valley Unit 2. These are being drafted to require the operator to verify flow, perform calorimetric power checks, and generate incore flux maps as specified by the Standard Technical Specifications (NUREG-0452, Rev. 4).

ATTACHMENT 3

Response to Outstanding Issue 47 of the
Beaver Valley Power Station Unit No. 2
Draft Safety Evaluation Report

Draft SER Section 4.4.7: ICC Instrumentation (excerpt)

We have reviewed the applicant's submittal of the instrumentation for indication of inadequate core cooling (Section 4.4.6.4) and found it insufficient; therefore, the staff will require the applicant to provide the itemized documentation of a complete ICCI system on a schedule which will permit completion of our review prior to fuel load.

Response:

A description of the ICC instrumentation, a core cooling monitor ($T_{saturation}$ Meter) and a Reactor Vessel Level Instrumentation System (RVLIS) has been provided in FSAR Section 7.7.2. These meet the requirement of NUREG-0737 Item II.F.2 to provide instrumentation for the direction of inadequate core cooling. For more detailed information on the system, see the summary report titled "Westinghouse Reactor Vessel Level Instrumentation System for Monitoring Inadequate Core Cooling" (Westinghouse 1980).

ATTACHMENT 4

Response to Outstanding Issue 73 of the
Beaver Valley Power Station Unit No. 2
Draft Safety Evaluation Report

Draft SER Section 7.6.2.2: RCS Loop Isolation Valve Interlocks

FSAR Section 7.6.6 describes the RCS loop isolation valve interlocks. The description is incomplete and additional information is required to clarify that the design is in conformance with IEEE-279. Additionally, the staff is concerned that, during operation with N-1 loops, the criteria for testing and single failure may not be met due to reduced protection logic. This is an open item.

Response:

Section 7.2.2.2 of the FSAR provides a description of how the Reactor Trip System provides automatic core protection during non-standard operation with a loop isolated by the Reactor Coolant System Loop Isolation Valve interlocks.

Isolation of a loop is under strict administrative control. One of the actions required to continue to meet the single failure criterion during this operation is to place the Δ TOP and the Δ TOT channels (associated with the loop not in service) in their tripped condition. This instrumentation would continue to be in partial trip except during surveillance. The surveillance testing requirements will be described in Technical Specifications similar to those being developed for Beaver Valley Unit One.