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JAN 7 1983

Docket No. 50-334

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

Dear Mr. Carey:

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION ON N-1 LOOP OPERATION

After reviewing your October 8, and November 22, 1982 letters, which furnish additional information on the subject issue, our Reactor Systems Branch has generated additional questions.

The first question, regarding reanalysis of the limiting large-break LOCA, is a reiteration of a question raised in our August 17, 1982 letter. In your reply you declined to submit a reanalysis, arguing that the existing analysis with the 1975 Westinghouse model is still valid. In a subsequent telephone conversation with your staff, we stated the need for a reanalysis using the currently accepted version of the computer code. By this letter, our request made in the phone conversation is formally transmitted.

We request that you furnish responses to the enclosed questions within 45 days of receipt of this letter. If more time is needed, please feel free to discuss it with the Project Manager, Mr. P. Tam.

Sincerely,

original signed by:
S. A. Varga

Steven A. Varga, Chief
Operating Reactors Branch No. 1
Division of Licensing

Enclosure:
Request for Additional Information

cc w/enclosure:
See next page

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REQUEST FOR ADDITIONAL INFORMATION

BEAVER VALLEY POWER STATION

N-1 LOOP OPERATION

1. Large-Break LOCA Analysis

You have demonstrated compliance with 10 CFR50.46 for large break LOCA, using the 1975 Westinghouse evaluation model. However, at the time when these calculations were performed, the 1975 model had been superseded by the 1978 model, which in turn was later superseded by the currently accepted 1981 model. Consequently, the large break LOCA analysis was performed with a code which contained several errors. To confirm the adequacy of the ECCS analysis, please recalculate the limiting-case large-break LOCA with the 1981 Westinghouse model.

2. Major Rupture of a Main Feedwater Pipe

The feedline break analysis for (N-1) loop operation was performed using assumptions which differed significantly from those used in the original FSAR for N-loop operation. Specifically, the (N-1) analysis assumed full safety injection and the availability of offsite power, while the original FSAR assumed no safety injection and a complete loss of offsite power. The Westinghouse generic analysis for feedline breaks, WCAP-9230, identifies safety injection rate as one of the three most significant factors affecting the severity of this transient. Furthermore, the worst single active failure was determined to be "the loss of one safeguard train, which in turn results in the loss of one motor driven auxiliary feedwater pump <u>and</u> the loss of one high head safety injection charging pump."

The feedline break transient is one that is likely to be more severe for (N-1) loop than for N-loop operation. Please reanalyze this transient assuming the worst single failure stated above. Alternatively, you may submit an analysis of how the loss of one train of safety injection would alter the (N-1) calculation.

3. <u>Steam Generator Tube Rupture (SGTR)</u>

You have not submitted an analysis of this transient for N-1 loop operation. The ability to reduce primary system pressure following such an event depends to a great extent on pressurizer sprays and on heat transfer in the intact steam generator. For N-1 loop operation, there is only one intact steam generator, and there is the possibility of degraded pressurizer spray (see below). Demonstrate that the consequences of this event, during N-1 loop operation, are comparable to or less severe than for N-loop operation.

4. <u>Method of Loop Isolation</u>

With the exception of specifying that primary loop isolation valves will be closed, you have provided no details about the status of the isolated loop. In particular, it is not clear whether the secondary side will be isolated, and if so, how this will be done. It is also not clear how much water, if any, will be left in the isolated loop, both on the primary and secondary sides. Please provide this information.

5. Pressurizer Sprays

The original Beaver Valley FSAR states that the pressurizer sprays are fed from primary coolant pump discharge, with backup supply from the chemical and volume control system.

Demonstrate that the loss of one coolant loop does not defeat or degrade the operation of pressurizer sprays.

Also, verify that the effect of any such degradation does not significantly worsen the consequences of any transient or accident.

6. Removing Power to the Loop Isolation Valves

Inadvertent opening of a block valve in the isolated loop could result in the insertion of cold water into the core. To prevent such an insertion and the resulting reactivity addition, motive power should be removed from the isolation valves after they have been closed. Furthermore, the control room should have continuous positive indication of valve position during N-1 operation.

If you do not agree that these should be conditions for N-1 loop operation, please provide a detailed technical justification for an alternative approach.

7. Overpressurization of the Isolated Loop

At least two potential mechanisms exist for overpressurizing the isolated loop; (1) heating due to inadvertent operation of the primary coolant pump (PCP) and (2) inadvertent operation of PCP seal injection from the charging pumps. Such an event could lead to steam generator tube rupture or other modes of breaking the primary coolant boundary. (A loop isolation valve would not qualify as part of the primary coolant boundary).

Propose and justify a method of either preventing or mitigating such overpressurization.

8. Isolation Valve Bypass Lines

In the original FSAR, Figure 4-1 indicates the presence of some bypass lines on the loop isolation valves. The licensee should Describe in detail the status and flow in these lines during N-1 loop operation, and describe how this flow affects transient and accident analyses.

10. Pressurized Thermal Shock (PTS)

There is reason to believe that the problems associated with pressurized thermal shock are more severe for N-1 loop operation than for normal operation. The generic information submitted by the Westinghouse Owners Group for use in the evaluation of PTS did not include an analysis of N-1 loop operation. There are two potentially significant differences. First, the presence of the cold leg isolation valve limits mixing of emergency core cooling water with the primary coolant in the cold leg. Thus the ECC water would enter the downcomer at a lower temperature than would otherwise be expected, causing a more rapid temperature reduction.

Second, stagnation in the isolated loop will occur for all transients and accidents, not just for those in which loss of circulation is normally expected.

Demonstrate that N-1 loop operation with closed isolation valves can be accomplished without significant additional risk from PTS.