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October 8, 1982

Director of Nuclear Reactor Regulation
United States Nuclear Regulatory Commission
Attn: Mr. Steven A. Varga, Chief
Operating Reactors Branch No. 1
Division of Licensing
Washington, DC 20555

Reference: Beaver Valley Power Station, Unit No. 1
Docket No. 50-334, License No. DPR-66
Request for Additional Information on Two-Loop Operation

Gentlemen:

In accordance with your letter of August 17, 1982, we are providing the information requested on two-loop operation. Included as Attachment A is the additional information which responds to each question with the exception of Question 7. The Westinghouse Electric Corporation is presently reviewing this question against their analysis. We will provide a response by November 22, 1982 to this question.

This information is being submitted later than the requested response date in concurrence with our NRC Project Manager. If you have any questions on this subject, please contact my office.

Very truly yours,

J J Carey

J. J. Carey
Vice President, Nuclear

Attachment

cc: Mr. W. M. Troskoski, Resident Inspector
U. S. Nuclear Regulatory Commission
Beaver Valley Power Station
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U. S. Nuclear Regulatory Commission
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DUQUESNE LIGHT COMPANY
Beaver Valley Power Station, Unit No. 1

Request for Additional Information on Two-Loop Operation
Response to NRC letter dated August 17, 1982

Attachment A

Question 1

The licensee has demonstrated compliance with 10 CFR 50.46 for large break LOCA. However, since the time of the N-1 loop operation submittal, the Westinghouse evaluation model has undergone several changes and corrections. Has the large break analysis for N-1 loop operation been performed with the latest evaluation model? If so, provide the results. If not, please confirm the adequacy of the submitted analysis by recalculating the limiting-case large LOCA with currently approved model.

Response

The large break N-1 Analysis was performed with the October 1975 version of the Westinghouse ECCS Evaluation Model. The currently approved model is the 1981 Westinghouse ECCS Evaluation Model. There are currently no plans to redo the N-1 analysis using the current model because the October 1975 version is still considered valid by the NRC.

Question 2

The submittal does not contain a small break analysis. Either justify its omission or analyze the small break LOCA with the currently approved evaluation model.

Response

WCAP-9280 "Westinghouse Emergency Core Cooling System Evaluation Model for Analyzing Small LOCA's During Operation with a Loop Out of Service for Plants Without Loop Isolation Valves", contains results showing dramatic reduction in small break Peak Clad Temperature (PCT) for N-1 loop analyses. Since small break is not, nor has it ever been the limiting case, by demonstrating that N-1 loop analyses always results in greatly reduced PCT's on a generic basis should be sufficient.

Question 3

For the uncontrolled boron dilution transient, demonstrate that the acceptance criteria of SRP 15.4.6 can be met for isolated loop operation. These criteria require that during power operation, hot standby, cold shutdown and startup, a minimum of 15 minutes be available from the time an alarm announces an unplanned moderator dilution to the time of loss of shutdown margin. For refueling, the minimum time is 30 minutes.

Response

This question is not applicable because of the fact that for the uncontrolled boron dilution event, cold shutdown, hot standby, and time from alarm annunciation were not part of the original licensing basis for Beaver Valley Unit 1. However, because the shutdown margin for N-1 loop operation is 0.63% k greater than that for N-loop operation, the time from boron dilution initiation to the time criticality is attained can only be longer for N-1 loop as compared to N-loop. The N-loop times are given in Section 14.1.4 of the Beaver Valley Unit 1 FSAR.

Our submittal of October 27, 1978, contained proposed Technical Specification changes addressing N-1 loop operation. Technical Specification 3.1.1.1 identifies the shutdown margin requirements for this operating condition. During refueling, the high flux at shut-down alarm is set at one-half decade above background and an audible count rate is provided in the control room. Should a dilution event occur, it would be identified by both an increasing audible count rate and the alarming of the high flux at shut-down alarm. By procedure, the operator is required to initiate immediate boration. The time for this activity would be less than 15 minutes.

The criterion of SRP 15.4.6 which requires the identification of a boron dilution transient before the shutdown margin is lost does not appear to be achievable. If the plant is maintaining the minimum shutdown margin as defined in Appendix A of the Technical Specifications, any dilution event would immediately result in an inadequate shutdown margin. However, the time required for the dilution event to continue to the point where criticality is achieved has been analyzed in our FSAR, Section 14.1.4.

During refueling, the time required to reach criticality was determined to be 24 minutes following the initiation of a dilution transient. For a dilution transient during start-up, this time is 53.7 minutes. During power operation, if dilution continues after reaching the low-low insertion limit alarm, it takes approximately 14 minutes before the total shutdown margin is lost due to dilution.

In all the above cases, there is ample time for the operator to determine the cause of the event, isolate the primary grade water sources and initiate boration. This information is tabulated in the FSAR on Table 14.1-2.

Question 4

Transients involving accidental depressurization of the reactor coolant system were analyzed in the original FSAR and found not to be limiting. Justify that this is also the case for N-1 loop operation.

Response

For the N-1 loop accident analysis amendment, the RCS depressurization accident was not considered. This is due to the fact that the N-loop scenario for this event is to have the initial power level at 102% of nominal and core average temperature 4°F above nominal. For N-1 loop, this would correspond to a power level of 67% and a core average temperature 10°F lower than that for N-loop. For these reasons, the margin to DNB would increase and, therefore, the N-loop analysis would be bounding for N-1 loop operation.

Question 5

Propose startup tests for the purpose of demonstrating operational stability with N-1 loop operation. These should include isolation of the loop containing the pressurizer.

Response

At this time, no additional N-1 loop startup tests are proposed other than those already outlined for N-loop operation. If, however, during a cycle, a switchover is made from N to N-1 loop operation, a flow calorimetric test will be performed and flux maps will be taken at zero power, an intermediate power level, and full power level. Because the pressurizer surge line would not be isolated if its respective loop was isolated, operating in this configuration should not affect operational stability. Stable two loop operation has been demonstrated at the Virginia Electric Power and Light Company, Surry Power Station which is very similar in design to Beaver Valley, Unit No. 1.

Question 6

Are there any variation to operator emergency procedures for N-1 loop operation? Do the present W operator emergency guidelines address N-1 loop operation? If not, justify the technical adequacy of your procedures since it is our understanding the W guidelines provide the technical basis to your procedures. If so, describe the modifications to the guidelines in detail.

Response

The Westinghouse Generic Emergency Response Guidelines do not specifically address N-1 loop operation. Variations to these guidelines for N-1 loop operation have not been addressed at this time. We have reviewed the immediate actions addressed in our existing LOCA procedure for technical adequacy to determine the degree to which they support N-1 loop operation.

To support two loop operation, the following steps would be taken:

1. The instrumentation, alarms, bistables and valve positions for the out of service loop would be made identifiable to the operator and administratively controlled to avoid confusion during an event. These items would be a part of the procedure for removing the loop from service.

2. The auxiliary feedwater flow to the out of service loop would be isolated as part of the procedure for removing a loop from service. Flow verification could not be made and therefore would require identification of this instrument as being out of service for the affected loop.

3. The surveillence tests would provide for monitoring of the following where necessary:

   - verification of the closed position of the out of service main steam isolation valve
   - instrument channel checks for protection and control instrumentation
   - auxiliary feedwater system alignments

The safety injection flow verifications performed in the emergency procedures are not affected when a loop is removed from service since the injection points are not within the isolable portion of the loops.

The protection system inputs from the isolated loop would be defeated or by-passed to accommodate surveillence testing of isolated loops and maintain annunciators operable for shared (3 loop) inputs.

Emergency procedures currently prohibit isolating a steam generator during an accident in the event of a steam generator tube leak and would not be revised for two loop conditions.

Emergency procedure E-0, Immediate Actions and Diagnostics is adhered to in the identification of the following:

1. Spurious actuation of safety injection
2. Loss of reactor coolant
3. Loss of secondary coolant
4. Steam generator tube rupture

In consideration of the above administrative controls and statements, two loop operation would not affect the immediate actions of these procedures and as such, our emergency procedures would not require revisions. As new emergency guidelines are currently in development, their adequacy for supporting two loop operation would have to be determined.

Question 7

For steam line breaks with an isolated loop, the time to attain criticality and the time to empty the pressurizer are longer than for normal operation (see Table 2.5-2 of the License Amendment Request). By contrast, the time to reach 2,000 ppm boron is much shorter for N-1 operation. Please provide a detailed explanation of this behavior.

Response

We are unable to provide a response to this question at this time. The Westinghouse Electric Corporation is presently studying their analysis with respect to the time it takes the 20,000 ppm boron to reach the loops. We expect to be able to provide a response to this question by November 22, 1982.