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U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

Subject: **Beaver Valley Power Station, Unit No. 1 and No. 2**
BV-1 Docket No. 50-334, License No. DPR-66
BV-2 Docket No. 50-412, License No. NPF-73
Response to Request for Peer Review of Analyses in the NRC's Accident Sequence Precursor (ASP) Analysis Program

Reference: NRC Letter, "Transmittal of NRC Contractor Analysis of Beaver Valley Operational Events," dated May 26, 1994

This letter is in response to the NRC request for review and comment to the analyses of two operational events which occurred at the Beaver Valley Power Station (BVPS) Units 1 and 2 in 1993 (see reference above). We have completed the review of the analyses and our comments are provided in Attachments 1, 2 and 3.

Should you have any questions regarding this submittal, please contact Ed Coholich at (412) 393-5224.

Sincerely,

J. D. Sieber

J. D. Sieber

Attachments

cc: Mr. L. W. Rossbach, Sr. Resident Inspector
Mr. T. T. Martin, NRC Region I Administrator
Mr. G. E. Edison, Project Manager

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ATTACHMENT 1

Comments on NRC Contractor Analysis
of Beaver Valley Operational Event LER Number 334/93-013

This attachment addresses the analysis of LER Number 334/93-013, "Unit 1 Reactor Trip and Required Shutdown, Dual Unit Loss of Offsite Power."

On October 12, 1993, the Beaver Valley site experienced a dual unit loss of offsite power (LOOP). Unit 1 had been operating at 100% power and Unit 2 was in refueling at the time of the event. This event was modeled as a plant-centered LOOP and the short and long-term nonrecovery values and seal LOCA probabilities were modified to reflect this in the event tree. The RCS leak rate associated with the loop 1A cold leg vent valve (RC-27) was small and well within the capabilities of the charging system, and as such, had no impact on the sequence of other events or on the viability of operator recovery of other systems. In addition, the Unit 2 transient was not modeled since the Unit 2 LOOP was of short duration, and all fuel had been moved to the spent fuel pool.

Based on these assumptions, the overall conditional core damage probability for this event is 6.2×10^{-5}, as determined by ORNL. The dominant core damage sequence involves a LOOP, followed by successful trip, failure of emergency power, successful Auxiliary Feedwater (AFW) actuation, a reactor coolant pump seal LOCA, and failure to recover offsite power in the long term. This sequence contributes 3.7×10^{-5} to the core damage frequency, or about 60% of the total.

Duquesne Light Company has reviewed the ORNL event tree and generally agrees with the modeling assumptions. The dominant sequence and the core damage frequency based on the ORNL system probabilities appear reasonable. However, the event tree structure does not take credit for the Unit 1 Appendix R Dedicated Auxiliary Feedwater Pump which is included in top event DF of the BVPS Individual Plant Examination (IPE) Summary Report. This pump is powered by the Emergency Response Facility Diesel Generator and can provide water to the steam generators during a station blackout coincident with failure of the steam driven AFW pump. This would only affect sequences where AFW has failed (i.e., sequence (55) with a core damage frequency of 1.2×10^{-5}). Therefore, with credit taken for top event DF, the core damage frequency for sequence (55) would be reduced by about 54% to 5.5×10^{-6}, using the IPE data, and the overall conditional core damage probability for this event would be reduced by approximately 10% to 5.6×10^{-5}.

ATTACHMENT 2

Comments on NRC Contractor Analysis
of Beaver Valley Operational Event LER Number 412/93-012, Case 1

This attachment addresses the analysis of LER Number 412/93-012, "Emergency Diesel Generator Sequencer Circuit Deficiencies," Case 1 - LOCA with transient-induced LOOP (Attachment 3 addresses Case 2 - LOOP with SI initiated for feed and bleed).

On November 4, 1993, the automatic loading capability of the 2-1 emergency diesel generator (EDG) failed in response to a safety injection (SI) signal during testing. Two days later, on November 6, 1993, the automatic loading capability of the 2-2 EDG to an SI signal also failed during testing.

This failure will only occur when an SI signal is present coincident with a loss of the normal power supply to the ESF bus. Repeated testing demonstrated that the failure mechanism occurred in one out of three initiations. When failure occurred, operator actions would have been necessary to allow manual loading of equipment on the ESF busses.

For Case 1, a postulated LOCA is the initiating event. If offsite power is available, loads are fast transferred to the alternate offsite power source and the SI sequencer would operate properly. If offsite power is not available, the EDGs will start. The normal feeder breaker to the safeguards busses trips open and load shedding occurs. The sequencer would start and "lock-up." The event tree is based on the ASP LOCA tree for class A plants and assumes an operator failure rate of 0.34 to manually load the ESF equipment onto the EDGs within the one-half hour that is available to establish make-up to the reactor coolant system before core damage will occur. Based on these assumptions, the estimated conditional core damage probability for this event is 2.1×10^{-6}, as determined by ORNL. The dominant core damage sequence for Case 1 involves a postulated LOCA, transient-induced LOOP that is recovered in the first half hour, successful reactor trip, and failure to load the ESF busses (reset the MCCs). This sequence contributes 1.1×10^{-6} to the core damage frequency, or about 52% of the Case 1 total.

The other dominant Case 1 sequence involves a postulated LOCA, transient-induced LOOP that is not recovered in the first half hour, successful reactor trip, emergency power restoration, and failure to load the ESF busses. This sequence contributes 1.0×10^{-6} to the core damage frequency, or approximately 48% of the Case 1 total.

Duquesne Light Company has reviewed the ORNL event tree and generally agrees with the modeling assumptions. The dominant sequences and the core damage frequency based on the ORNL system probabilities appear reasonable. However, the assumed operator failure rate of 0.34 to manually load the ESF equipment and reset the Motor-Control-Centers (MCCs) appears to be high. The ESF pumps can be loaded directly from

the control room and all of the Unit 2 emergency MCCs can be reset at the 480 VAC substations located in the emergency switchgear rooms. Based on this, we have a high confidence that power to the MCCs can be restored at one location and it would not be necessary to go to each individual MCC. Therefore the ASP Recovery Class R3 operator failure rate of 0.12 appears to be more reasonable to use. By using the 0.12 operator failure rate, a reduction of approximately 65% in core damage probability would result, and the total Case 1 conditional core damage probability for this event would then be 7.4×10^{-7}.

ATTACHMENT 3

Comments on NRC Contractor Analysis
of Beaver Valley Operational Event LER Number 412/93-012, Case 2

This attachment addresses the analysis of LER Number 412/93-012, "Emergency Diesel Generator Sequencer Circuit Deficiencies," Case 2 - LOOP with SI initiated for feed and bleed.

The plant transient, as discussed in Attachment 2, remains the same; however, for Case 2, a postulated LOOP is the initiating event. If offsite power is not recovered and AFW and HFW fail, feed and bleed is utilized for core cooling. It is assumed the operator will actuate high-pressure injection by manually actuating an SI signal. This will cause the sequencers to "lock-up" since offsite power is not available. Loads already connected to the bus will not be shed. Therefore, the equipment started for the loss of voltage signal will remain operable; however, the additional equipment started by the SI signal will not start.

The event tree is based on the ASP LOOP tree for class A plants and assumes an operator failure rate of 0.34 to manually load the ESF equipment onto the EDGs when AFW is successful, and a guaranteed failure to do so if AFW fails. Based on these assumptions, the estimated conditional core damage probability for this event is 3.8×10^{-6}, as determined by ORNL. The dominant core damage sequence for Case 2 involves a postulated LOOP that is not recovered in the first half hour, successful reactor trip, emergency power restoration, failure of AFW, and failure to load the ESF busses. This sequence contributes 3.6×10^{-6} to the core damage frequency, or about 95% of the Case 2 total.

Duquesne Light Company has reviewed the ORNL event tree and disagrees with the event tree structure and the timing of the modeling assumptions. For the Unit 2 LOOP initiators resulting in a PORV/SV LOCA, FSAR analyses show that the time required for the reactor coolant system pressure to increase to the PORV setpoint and then decrease to the SI setpoint following the opening of a PORV is approximately 50 seconds. This is based on a review of the LOOP and inadvertent relief valve opening analyses which conservatively take no credit for steam dump valves opening. This would provide adequate time for the EDGs to start and sequence on the Charging/HHSI pumps, SI valve MCCs, EDG cooling valves, and AFW pumps and valves before the SI reset sequencer failures occur. Therefore, all essential equipment would have electric power available and would either be running (pumps), or would actuate (valves) when the SI signal is either automatically generated or manually initiated for feed and bleed. ESF manual loading would only have to be performed on some less vital equipment that are not required until some time later (e.g., LHSI pumps), and the ASP Recovery Class R3 operator failure rate of 0.12 seems more reasonable to use. This should reduce the conditional core damage probability for these Case 2 sequences to below 1.0E-07.