LER 366/82-095

Event Description: RHRSW Loops A and B Unavailable

Date of Event: August 17, 1982

Plant: Hatch 2

Summary

On August 1 7, 1982 while residual heat removal service water (RHRSW) loop B was out of service for maintenance, the personnel responsible for closing the B loop strainer inlet valve inadvertently closed the A loop strainer inlet valve instead. This blocked the only open flow path in the A loop and resulted in both trains of RHRSW being simultaneously unavailable. The A loop strainer valve was opened and the A loop of RHRSW returned to operable status.

The RHRSW system provides cooling water from the ultimate heat sink (the Altamaha River) to remove decay heat via the residual heat removal (RHR) heat exchangers. By means of a crosstie with the RHR system, the RHRSW system can also supply makeup to the reactor coolant system (RCS) when all emergency core cooling systems have failed. The RHRSW system consists of two independent trains consisting of two pumps each. Each train supplies cooling water from the intake structure to one RHR heat exchanger. During power operation, the RHRSW system is not operating. When required, the system is placed in operation by remote manual means. Pressure control is achieved by a flow control valve on the RHR heat exchanger outlet. Each RHRSW train has a rated decay heat removal capacity of 100%. This implies that two RHRSW pumps supplying a single RHR heat exchanger can provide adequate decay heat removal capacity.

The duration of the combined unavailability of both trains of RHRSW is not clearly indicated in the licensee event report (LER), but it is reported that one train was recovered before an 8-hour Technical Specification time limit was exceeded. A one-hour duration was assumed and this event was modeled as a one-hour unavailability of both trains of RHR. The non-recovery probability for RHR was revised to 0.054 to reflect the RHRSW failures (based on data included in "Faulted Systems Recovery Experience," Nuclear Safety Analysis Center (NSAC)-161, May 1992). For sequences involving potential RHR or power conversion system (PCS) recovery, the non-recovery estimate was revised to 0.054 x 0.52 (PCS non-recovery), or 0.028.

The increase in core damage probability (CDP), or importance, over the duration of the event is 7.7×10^{-6}. The base-case CDP over the duration of the event is 7.7×10^{-9} resulting in an estimated conditional core damage probability of 7.7×10^{-6}. The dominant sequence involves a postulated transient with a successful reactor shutdown, failure of the power conversion system, success of main feedwater, and failure of all three modes of RHR.