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PWR Loss of Feedwater ATWS: Analysis and Sensitivity Study

by

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The incident at the Salem Nuclear plant has presented a renewed interest in the analysis of the consequences of anticipated transients without scram (ATWS). This paper presents the results of an analysis of a complete loss of feedwater ATWS for a typical 4-loop PWR. The loss of feedwater transient was selected since previous analyses[1] have shown that this transient produces one of the more limiting overpressure conditions in the primary system. These results provide a detailed analysis of this transient using current analytical techniques and show the sensitivity to several important parameters and plant modeling techniques.

The RELAP5/MOD1 computer code[2] has been used for this analysis. The code version is designated as Cycle 13 with additional modifications provided by both INEL and BNL. A significant code modification implemented for this analysis provided the capability to represent the Doppler feedback as a function of reactor power. This required an iterative solution of the point kinetics equations so that the Doppler feedback and reactor power are determined simultaneously.

The basic input model was adapted from a model developed by INEL for use in small break loss of coolant accident analysis.[3] The model consists of 145 control volumes and 146 flow paths; two primary coolant loops are

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represented, one of which simulated 3 reactor loops. The pressurizer is modeled with 6 volumes and is connected to the three loop representation. The heated region of each steam generator secondary (i.e., adjacent to the steam generator tubes) is represented by 4 volumes. For this analysis, a number of additional system representations important in overpressurization transient, have been added to the basic input model, including the pressurizer safety/relief valves and spray and the steam generator dump, safety valves and auxiliary feedwater systems.

The transient was initiated by the termination of feedwater to all 4 steam generators. The liquid inventory in the steam generator secondary side provides continued heat removal capability for approximately 30 seconds into the transient. At 30 seconds the turbine is assumed to trip and the secondary side heat removal capability is reduced. The primary system pressure (Fig. 1) quickly rises to the pressurizer relief valve setpoint; the secondary system pressure also increases following the turbine trip and actuates the steam dump and the secondary safety valves. Auxiliary feedwater is initiated at 60 secs but the capacity (5% rated) is not sufficient to replenish the inventory lost through the bypass and safety valves. As the secondary side inventory continues to deplete, the primary to secondary heat transfer is degraded, increasing the heat source/sink mismatch. This combined with the filling of the pressurizer causes a rapid increase in primary system pressure. However, moderator temperature feedback eventually decreases reactor power and the primary pressure reaches a peak of 3062 psia. The details of the pressure transient are shown in Fig. 1.

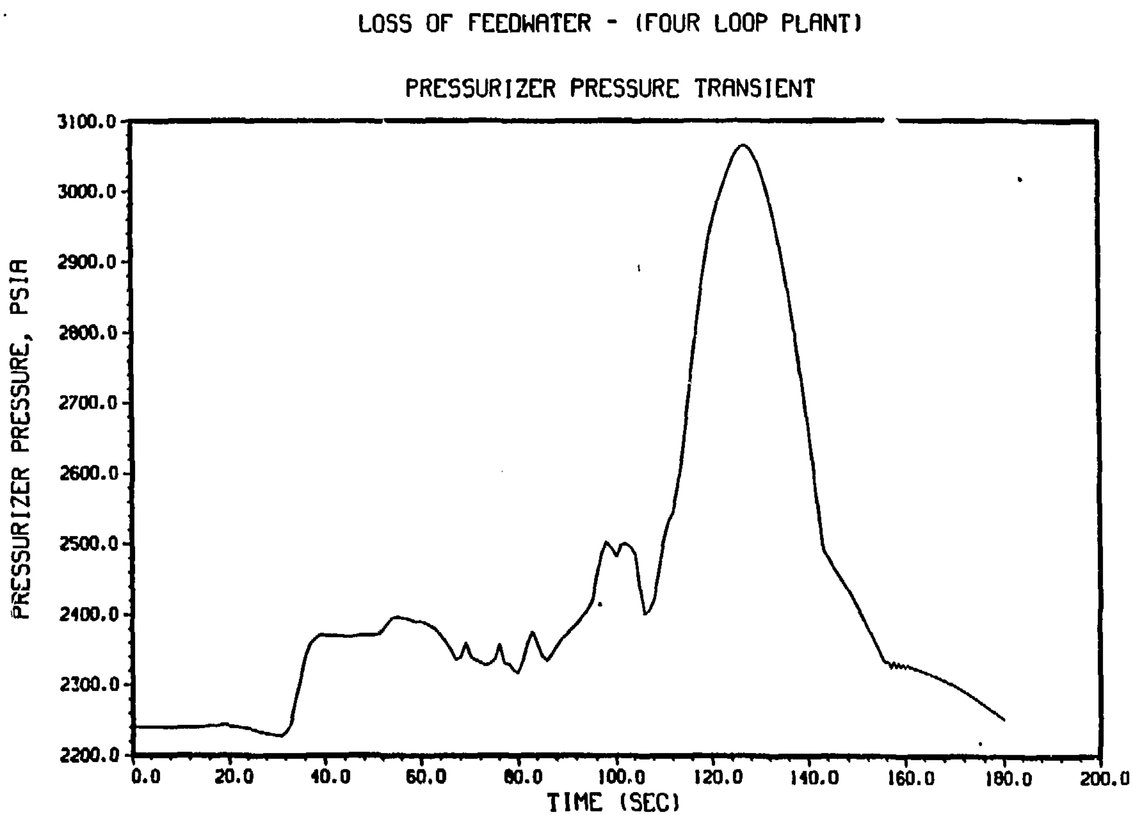
A significant characteristic of this ATWS is the steam generator dryout and subsequent loss of heat sink. The calculations described above utilized a 4 node model to represent the heated section of the steam generator secondary (each node approximately 8 feet long). The use of an 8 node model, to refine the dryout calculation, showed that the peak pressurizer pressure was reduced by 390 psi. This result is significant since it indicates additional margin between the peak system pressure and the overpressurization limit of 3200 psi. In addition the result indicates the importance of steam generator modeling in ATWS analyses. Additional sensitivities that have been evaluated in this analysis include the following: blocked primary relief valve (158 psi); reduced primary safety valve liquid flow rate (115 psi); and moderator temperature coefficient changed by 1 pcm/°F (83 psi).

References

1. "Westinghouse Anticipated Transients Without Trip Analysis," WCAP-8330, Westinghouse Electric Company, August 1974.

2. V. Ransom et al., "RELAP5/MOD1 Code Manual," Vols. 1 and 2, NUREG/CRF-1826, Idaho National Engineering Laboratory, 1982.

3. C.D. Davis, "Documentation of the RELAP5 Zion Model," CBD-6-80, Idaho National Engineering Laboratory, October 1981.


LOSS OF FEEDWATER - (FOUR LOOP PLANT)
PRESSURIZER PRESSURE TRANSIENT
PRESSURIZER PRESSURE, PSIA
3100.0
3000.0
2900.0
2800.0
2700.0
2600.0
2500.0
2400.0
2300.0
2200.0
0.0
20.0
40.0
60.0
80.0
100.0
120.0
140.0
160.0
180.0
200.0
TIME (SEC)