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1. WCAP-16779-NP Revision 0, "Overpressure Protection Report for AP1000 Nuclear Power Plant"

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ENCLOSURE 1

WCAP-16779-NP Revision 0

"Overpressure Protection Report for AP1000 Nuclear Power Plant"

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April 2007

Overpressure Protection Report for AP1000 Nuclear Power Plant

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WCAP-16779-NP
Revision 0

Overpressure Protection Report for AP1000 Nuclear Power Plant

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1 PURPOSE OF REPORT

This report documents the overpressure protection provided for the Reactor Coolant System (RCS) in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Articles NB-7200 and NC-7200 (Reference 1). This report documents the overpressure protection provided in the Westinghouse NSSS scope (Reference 2).

2 DESCRIPTION OF OVERPRESSURE PROTECTION

2.1 Overpressure protection is provided for the RCS and its components to prevent a rise in pressure of more than 10% above the system design pressure of 2,485 psig, in accordance with NB-7311. This protection is afforded for the following events that envelop those credible events that could lead to overpressure of the RCS if adequate overpressure protection was not provided.

a. Loss of External Electrical Load and/or Turbine Trip (Reference 3)
b. Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power (Reference 4)
c. Loss of Forced Reactor Coolant Flow (Reference 5)
d. Loss of Normal Feedwater Flow (Reference 6)
e. Loss of Offsite Power to the Station Auxiliaries (Reference 7)

2.2 The extent of the RCS is as defined in 10 CFR Part 50, and includes:

a. The reactor vessel including control rod drive mechanism housings;

b. The reactor coolant side of the steam generators;

c. Reactor coolant pumps;

d. A pressurizer attached to one of the reactor coolant loops;

e. The pressurizer safety valves;

f. The interconnecting piping, valves and fittings between the principal components listed above;

g. The piping, fittings and valves leading to connecting auxiliary or support systems up to and including the second isolation valve (from the high pressure side) on each line ; and

h. The Passive Residual Heat Removal (PRHR) system and the Core Makeup Tanks (CMT) are not isolated from the RCS and, as such, are considered part of the RCS for the purpose of this report.

2.3 The pressurizer provides volume surge capacity and is designed to mitigate pressure increases (as well as decreases) caused by load transients. The size of the pressurizer and the available steam volume at hot full power (close to 50 percent of the total pressurizer volume) is large enough to accommodate large power mismatch without actuation of the fixed high pressurizer pressure reactor trip. In addition, a pressurizer spray system condenses steam at a rate sufficient to prevent large pressure variations during a step reduction in power level equivalent to ten percent of full rated load.

The spray nozzle is located in the top head of the pressurizer. Spray is initiated when the pressure-controlled spray demand signal is above a given setpoint. Isolated output signals from the pressurizer pressure protection channels are used for pressure control. The spray rate

increases proportionally with an increasing compensated error signal until it reaches a maximum value. The compensated error signal is the output of a proportional plus integral controller, the input to which is an error signal based on the difference between actual pressure and a reference pressure.

In the event of unavailability of the pressurizer spray, steam dump and the rapid power reduction system and a complete loss of steam flow to the turbine, protection of the RCS against overpressurization is afforded by the pressurizer safety valves in conjunction with the steam generator safety valves and a reactor trip initiated by the Protection and Safety Monitoring System.

There are two pressurizer safety valves with a minimum required capacity of 750,000 lb/hr for each valve at 2,575 psia (system design pressure plus 3% allowance for accumulation per Reference 8). The pressurizer safety valves are totally enclosed pop-type, spring-loaded, self-activated valves with backpressure compensation. The set pressure of at least one safety valve will be no greater than system design pressure of 2,485 psig in accordance with Section NB-7410. The pressurizer safety valves discharge lines do not have loop seals. The pressurizer safety valves discharge directly to the containment atmosphere through a rupture disk.

Figure 2-1 shows a schematic arrangement of the primary system pressure-relieving devices.

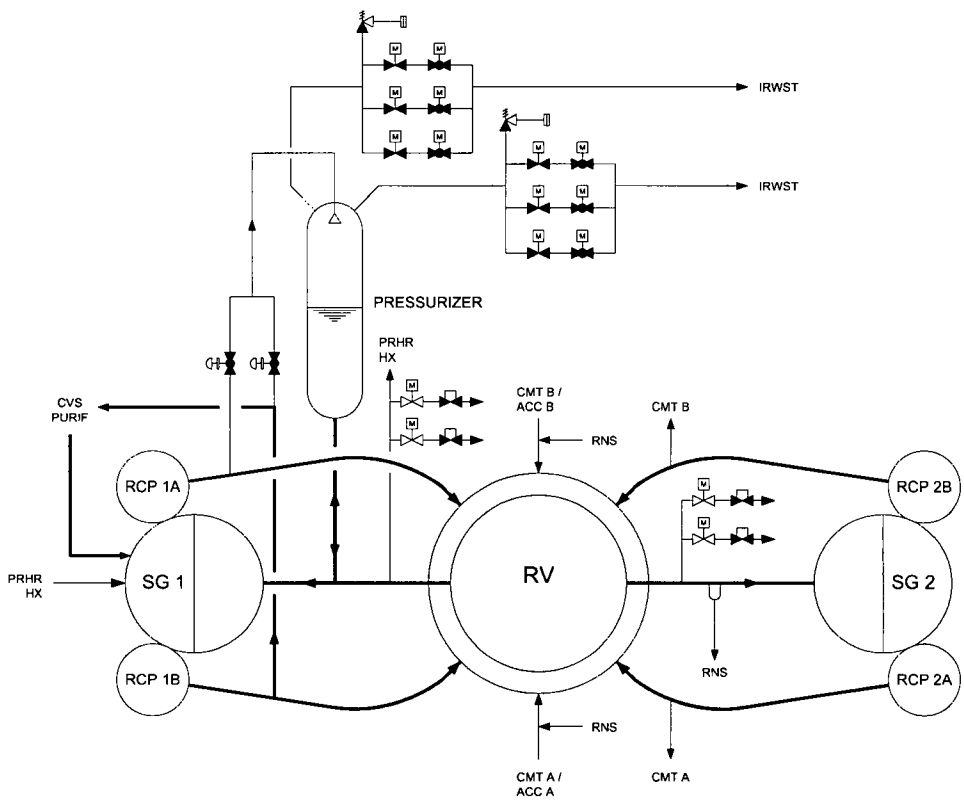

Figure 2-1 – Schematic of Primary System Pressure Relieving Devices

2.4 Overpressure protection for the main steam system is provided by the steam generator safety valves to prevent a rise in pressure of more than 10% above the steam generator shell-side design pressure in accordance with NC-7311. This protection is afforded for the same events as listed in Section 2.1 that envelop those credible events that could lead to system overpressurization if adequate overpressure protection was not provided. The main steam system safety valve capacity is based on providing enough relief to remove 110% of nominal steam flow at a nominal power of 3,415 MWt.

The set pressure of the lowest set main steam safety valve is selected to be 1,185 psig which is equal to the steam generator shell design pressure. The set pressure of the highest set main steam safety valve is selected to be 1,242 psig which ensures that all six main steam safety valves in each steam line are fully open below the maximum pressure limit of 1,303.5 psig which is 10% above the design pressure. The four intermediate main steam safety valves are set to open at different, staggered pressures between the minimum and maximum set pressures on each steam line.

3 SIZING OF PRESSURIZER SAFETY VALVES

3.1 The sizing of the pressurizer safety valves was based on a complete loss of steam flow to the turbine with the reactor operating at greater than 102% of the nominal power of 3,415 MWt. In this analysis, feedwater flow is isolated at the start of the transient and no credit is taken for operation of pressurizer level control system, pressurizer spray system, rod control system, steam dump system, the rapid power reduction system or steam line power-operated relief valves. The reactor is maintained at full power (no credit for reactor trip) and steam relief through the steam generator safety valves is considered. The total pressurizer safety valve capacity is required to be at least as large as the maximum surge rate into the pressurizer during this transient.

This sizing procedure results in a safety valve capacity well in excess of the capacity required to prevent exceeding 110% of system design pressure for the events listed in Section 2.1. An analysis, demonstrating the conservative nature of this sizing procedure at an NSSS power of 3,483.3 MWt or 102% of nominal power, is presented in the following section.

3.2 Each of the overpressure transients listed in Section 2.1 has been analyzed and reported in the AP1000 Design Control Document (DCD). The analysis methods, computer codes, plant initial conditions and relevant assumptions are also discussed in the AP1000 DCD for each transient.

Review of these transients shows that the loss of load/turbine trip event results in the maximum system pressure and the maximum safety valve relief requirements. This transient is presented in detail below.

For a loss of load/turbine trip event, the turbine stop valves close rapidly (in 0.3 seconds) on loss of trip fluid pressure actuated by one of a number of possible turbine trip signals. This will cause a sudden reduction in steam flow, resulting in an increase in pressure and temperature in the steam generator shell. As a result, the heat transfer rate in the steam generators is reduced, causing the reactor coolant temperature to rise, which in turn causes coolant expansion, pressurizer insurge, and RCS pressure rise.

With the control systems in operation, such as the automatic steam dump system, the pressurizer spray and, if above 50 percent power level, the rapid power reduction system, the sudden power mismatch as well as the excess steam generation would normally be accommodated. Reactor coolant temperature and pressure do not significantly increase and the reactor trip is avoided. If the turbine condenser was not available, the excess steam generation would be dumped to the atmosphere and main feedwater flow would be lost. For this situation, the PRHR System would ensure adequate long term residual and decay heat removal capability. Should the steam dump system and the rapid power reduction system fail to operate, the steam generator safety valves will lift to provide pressure control.

In this analysis, the behavior of the nuclear power plant is evaluated for a complete loss of steam load from the nominal full power level plus a two percent calorimetric error without any control system in operation. As additional conservatism, the main feedwater flow is terminated at the time of turbine trip, with no credit taken for startup feedwater or PRHR to mitigate the consequences of the transient.

The loss of load/turbine trip transient is analyzed by employing the detailed digital computer program LOFTRAN (References 9 and 10). The program simulates the neutron kinetics, RCS, pressurizer, pressurizer spray and safety valves, steam generators, and steam generator safety valves. The program computes pertinent plant variables including temperatures, pressures, and power level.

Major assumptions are as follows.

As required by the NRC and in line with the General Design Criteria (GDC) 17 of 10 CFR 50 Appendix A, the loss of load/turbine trip accident is analyzed with and without offsite power available. The loss of offsite power (LOOP) is not considered as a single failure but rather as a potential consequence of the analyzed event. So for the case which assumes the LOOP, the reactor coolant pumps will coastdown 3 seconds after turbine trip actuation.

a. Initial operating conditions

The initial reactor power and RCS temperatures are assumed to be at their maximum values consistent with the steady state full power operation including allowances for calibration and instrument errors. Namely 102% of 3,415 MWt and 573.6 +6.5°F. The initial RCS pressure is assumed at both a maximum and a minimum value consistent with the steady state full power operation including allowances for calibration and instrument errors. Namely 2,250 +50 psia for the case with LOOP and 2,250 -50 psia for the case without LOOP. This results in the maximum possible increase in coolant pressure for both cases of the loss of load/turbine trip event.

b. Moderator and Doppler coefficients of reactivity

The analysis assumes a zero moderator temperature coefficient and a least negative Doppler-only power and Doppler temperature coefficients, as this results in maximum pressure relieving requirements.

c. Reactor control

From the standpoint of the maximum pressures attained, it is conservative to assume that the reactor is in manual control. If the reactor were in automatic control, the control rod banks would move prior to trip and reduce the severity of the transient.

d. Steam release

No credit is taken for the operation of the steam dump system or steam generator power-operated relief valves. The steam generator pressure rises to the safety valve setpoints where steam release through safety valves limits secondary steam pressure to levels below the 110% of shell design pressure.

e. Rapid power reduction

No credit is taken for the operation of the rapid power reduction system. The reactor power is maintained at its initial value maximizing the RCS heat up and resulting in a pressure increase.

f. Pressurizer spray and safety valves

No credit is taken for the effect of pressurizer spray in reducing or limiting the coolant pressure. Safety valves are operable. A 3% accumulation, on the safety valves set pressure, is accounted for to open the valve fully.

g. Feedwater flow

Main feedwater flow to the steam generators is assumed to be lost at the time of turbine trip. No credit is taken for startup feedwater or PRHR flow since a stabilized plant condition will be reached before auxiliary feedwater initiation is normally assumed to occur; however, the PRHR would be expected to start on a trip of the main feedwater pumps. The PRHR would remove core decay heat following plant stabilization.

h. Reactor trip

Reactor trip is actuated by the first Protection and Safety Monitoring System trip setpoint reached. Trip signals are expected to occur due to high pressurizer pressure or overtemperature ΔT for the case without LOOP and due to RCP underspeed for the case with LOOP.

The results of the loss of load/turbine trip transient without LOOP are shown on Figures 3-1 through 3-7. The reactor is tripped on a high pressurizer pressure signal. Figure 3-1 shows the nuclear power. The pressurizer pressure and the reactor coolant pump discharge pressure, which is the point of highest pressure in the RCS, are shown in Figure 3-2. Figure 3-3 shows the pressurizer safety valve relief rate. Figure 3-4 shows the reactor coolant loop hot leg and cold leg temperatures. The steam generator shell side pressure is shown in Figure 6. The main steam safety valve relief rate is shown in Figure 3-6. Figure 3-7 shows the RCS volumetric flow rates as well as the core mass flow rate

The results of the loss of load/turbine trip transient w/LOOP are shown on Figures 3-8 through 3-14. The reactor is tripped on a RCP underspeed signal. Figure 3-8 shows the nuclear power. The pressurizer pressure and the reactor coolant pump discharge pressure, which is the point of highest pressure in the RCS, are shown in Figure 3-9. Figure 3-10 shows the pressurizer safety valve relief rate. Figure 3-11 shows the reactor coolant loop hot leg and cold leg temperatures. The steam generator shell side pressure is shown in Figure 13. The main steam safety valve relief rate is shown in Figure 3-13. Figure 3-14 shows the RCS volumetric flow rates as well as the core mass flow rate.

The results of these analyses show that the overpressure protection provided is sufficient to maintain the peak RCS pressure and the main steam system pressure below the code limit of 110% of the respective system design pressures. The plots of pressurizer safety valve relief rate and main steam safety valve relief rate show that all installed safety valves are required to provide adequate overpressure protection for this pressure limiting event.

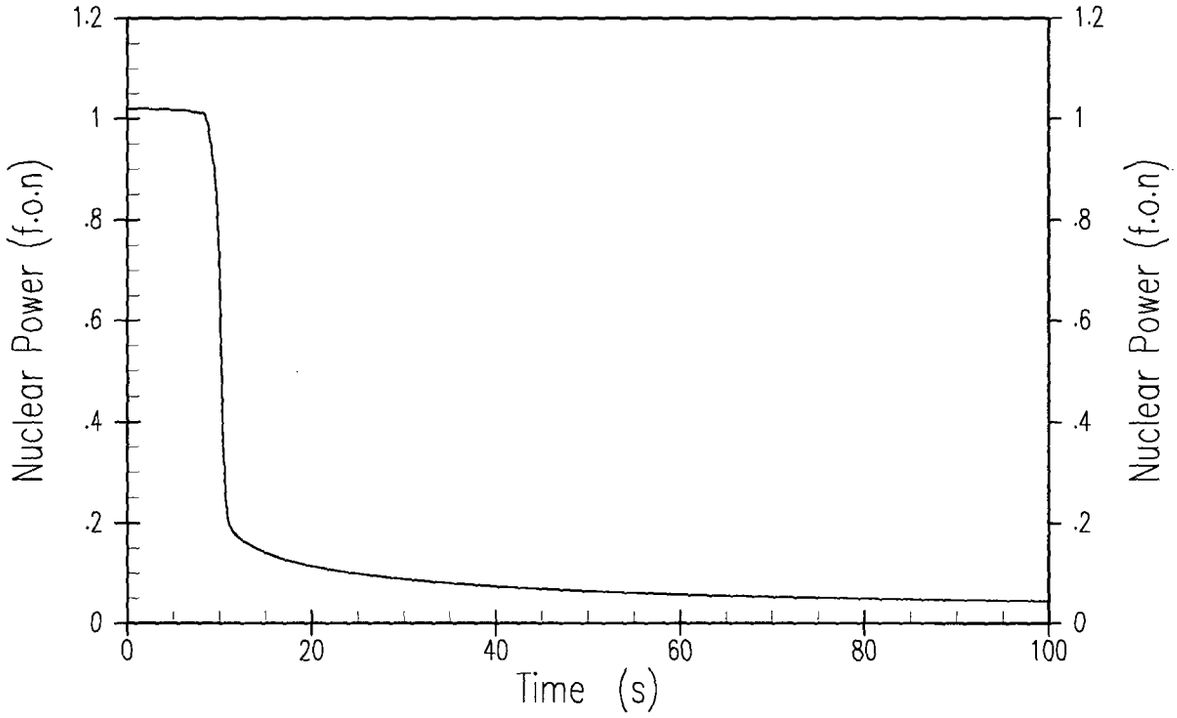

Figure 3-1 Nuclear Power vs. Time – w/o LOOP Case

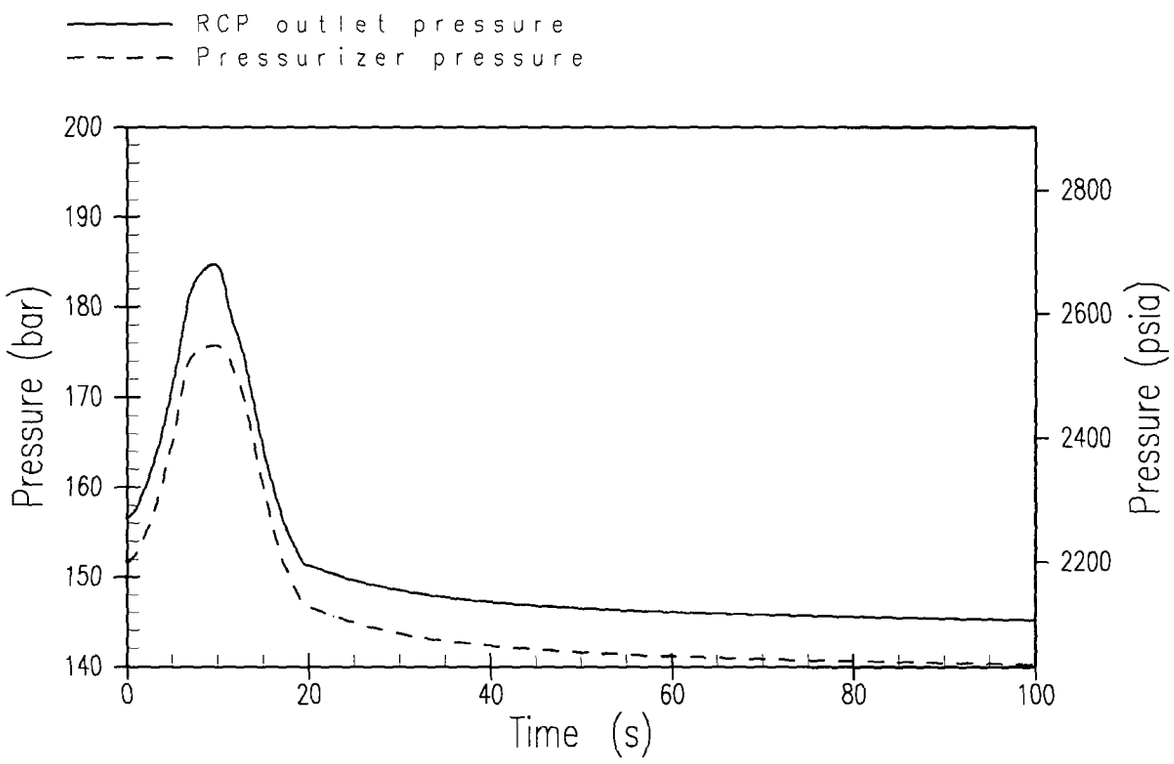

Figure 3-2 RCS Pressure vs. Time – w/o LOOP Case

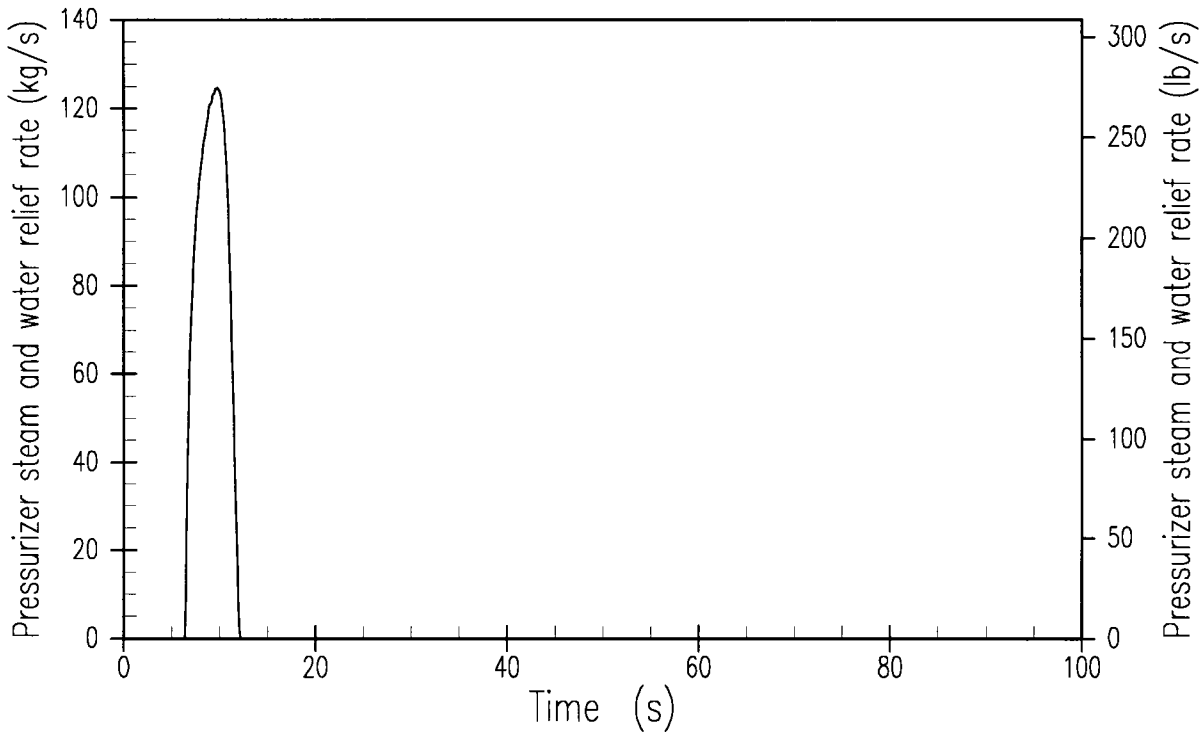

Figure 3-3 PRZ Safety Valve Relief Rate vs. Time – w/o LOOP Case

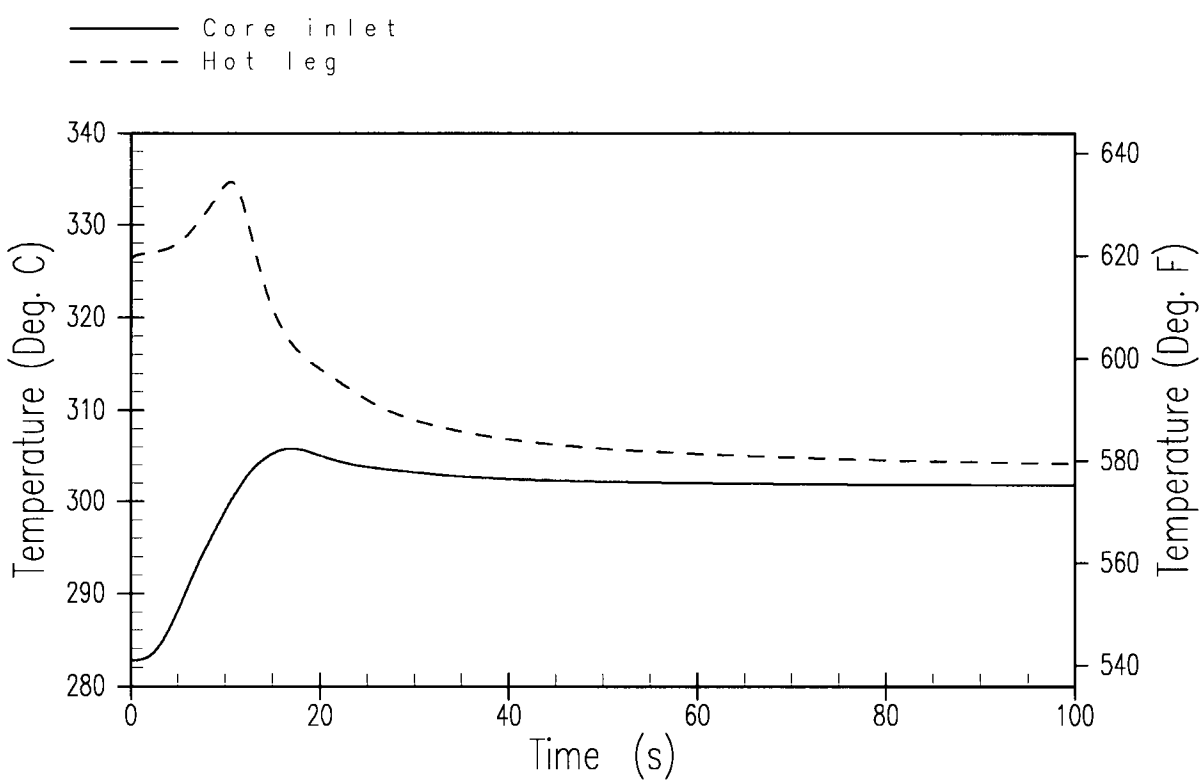

Figure 3-4 Reactor Vessel Temperatures vs. Time – w/o LOOP Case

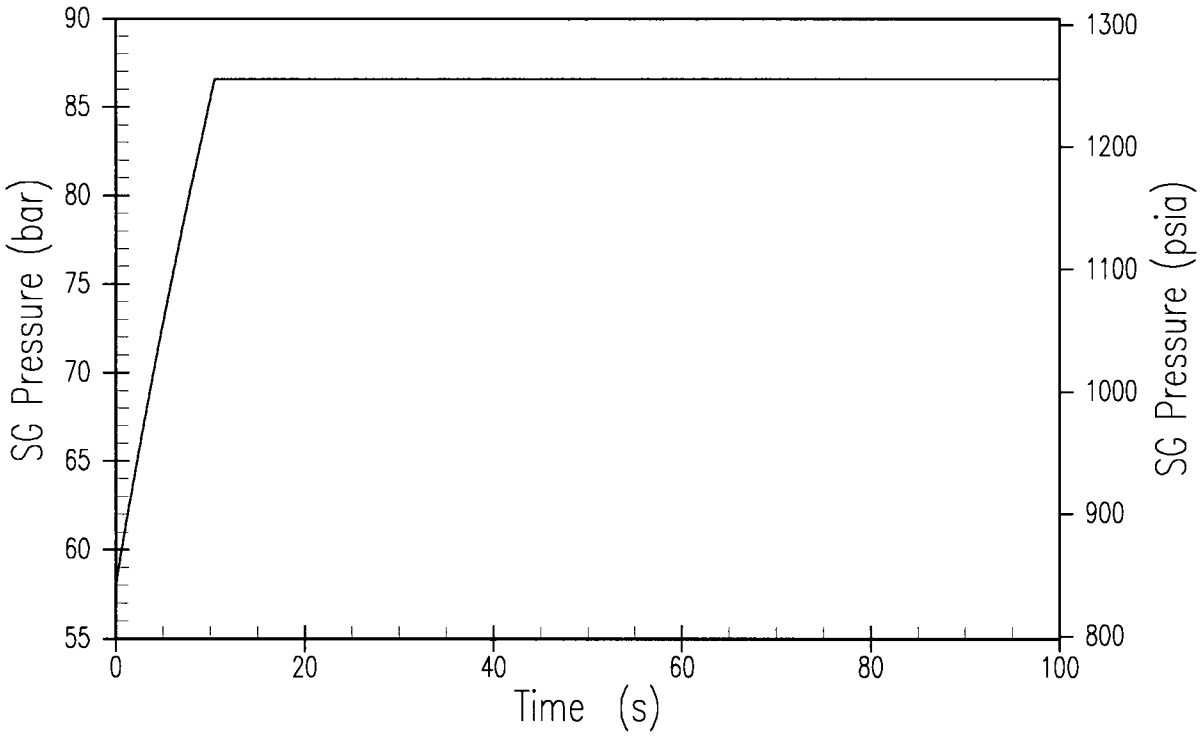

Figure 3-5 SG Pressure vs. Time – w/o LOOP Case

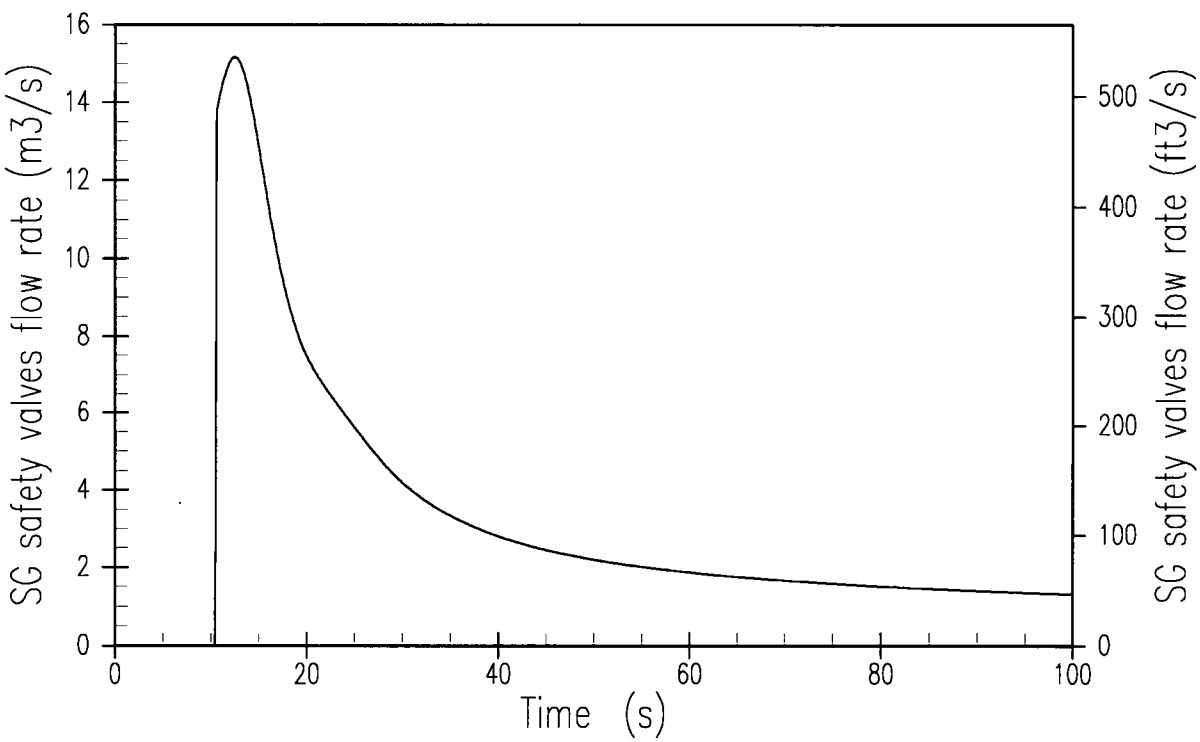

Figure 3-6 SG Safety Valve Flow Rate vs. Time – w/o LOOP Case

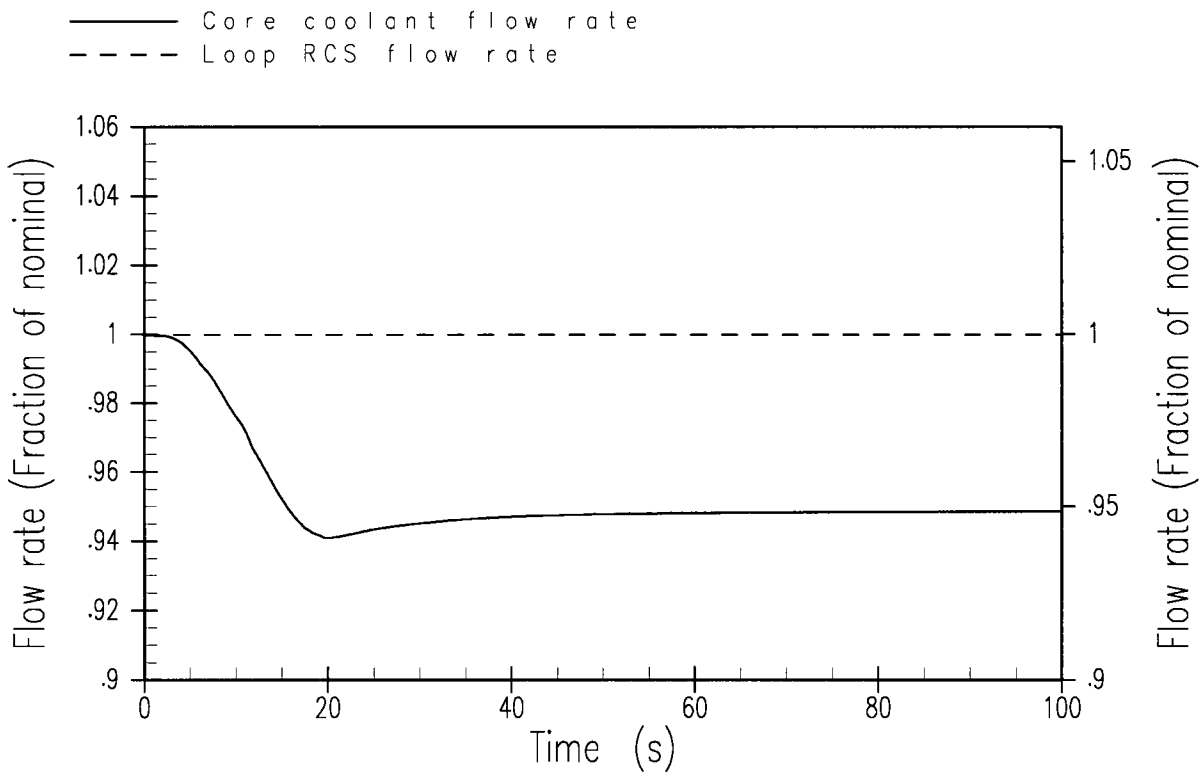

Figure 3-7 RCS Flow Rates vs. Time – w/o LOOP Case

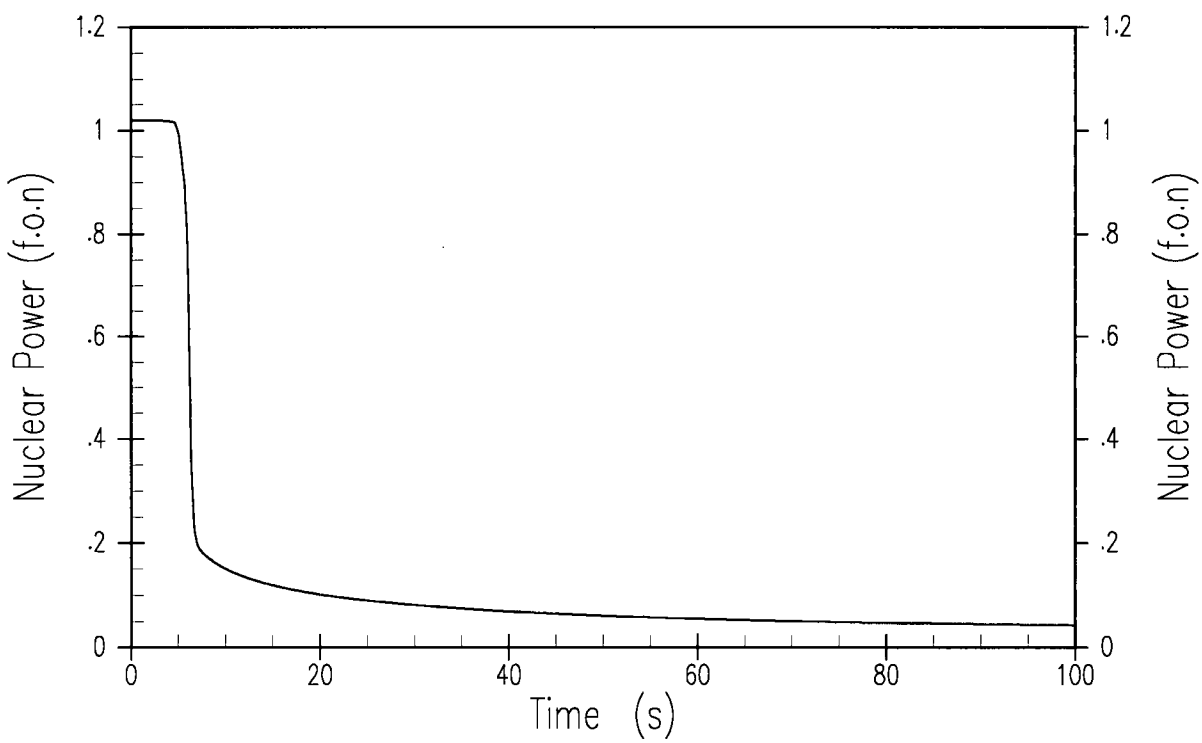

Figure 3-8 Nuclear Power vs. Time – w/LOOP Case

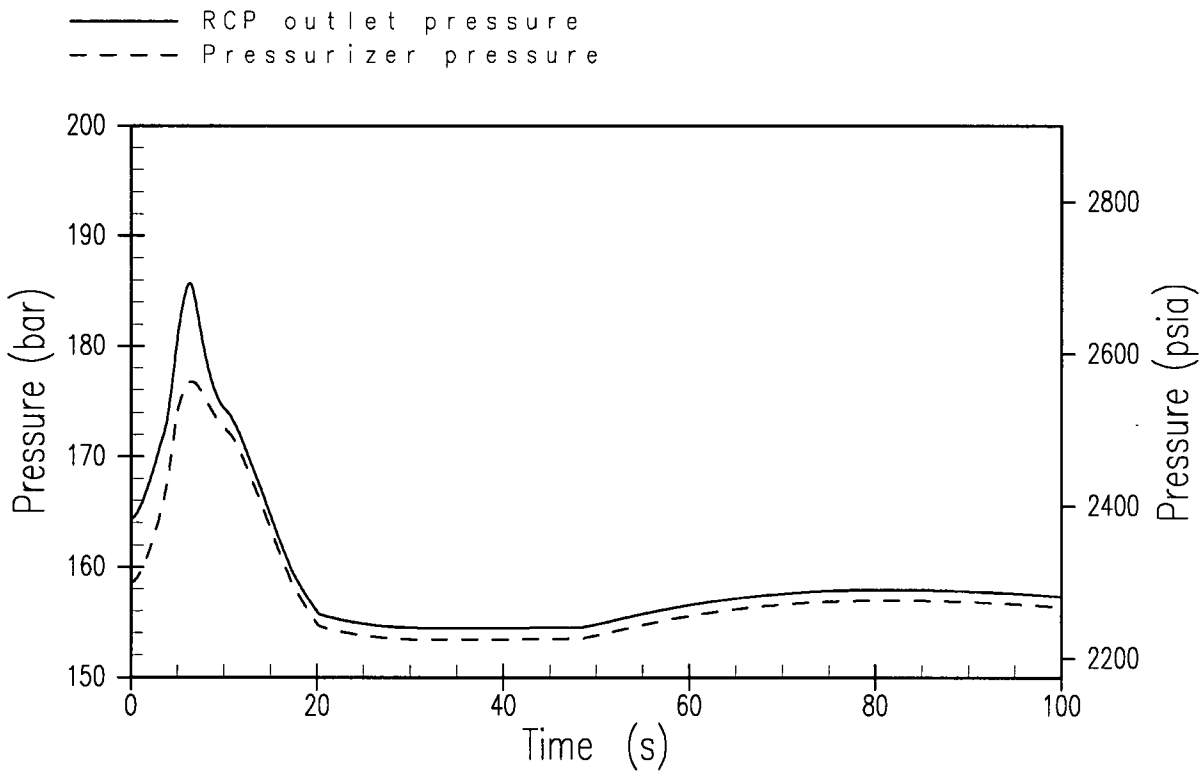

Figure 3-9 RCS Pressure vs. Time – w/LOOP Case

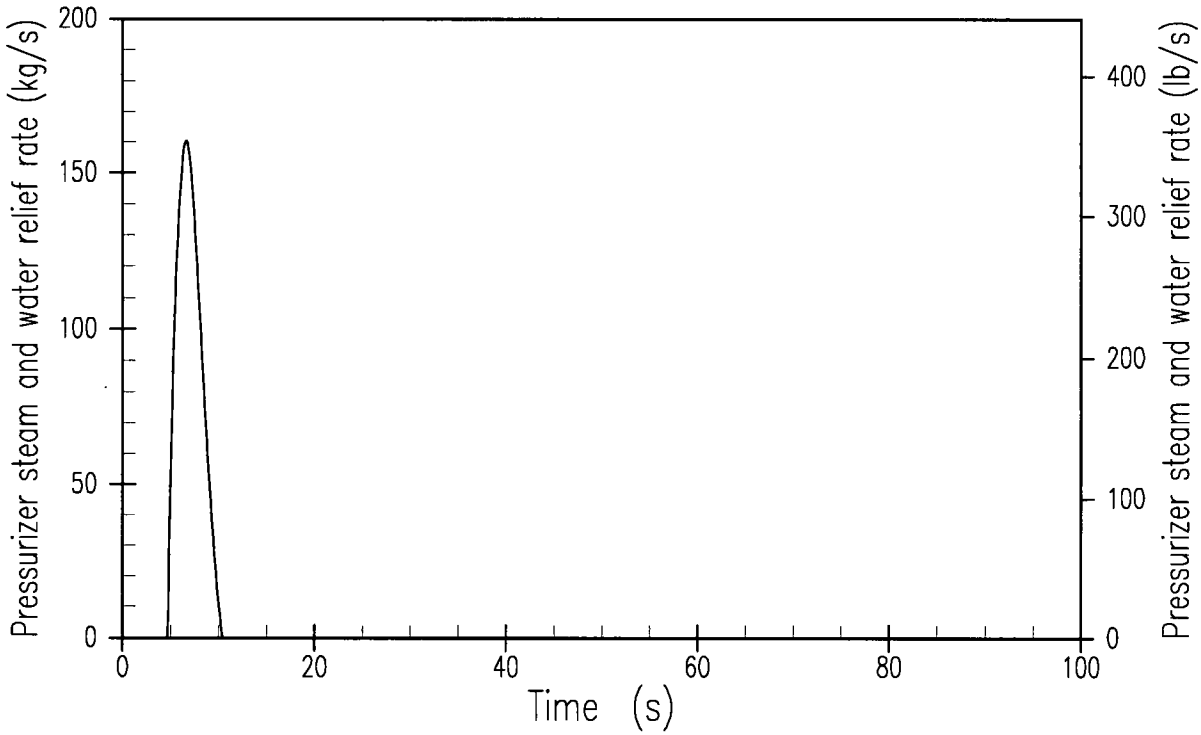

Figure 3-10 PRZ Safety Valve Relief Rate vs. Time – w/LOOP Case

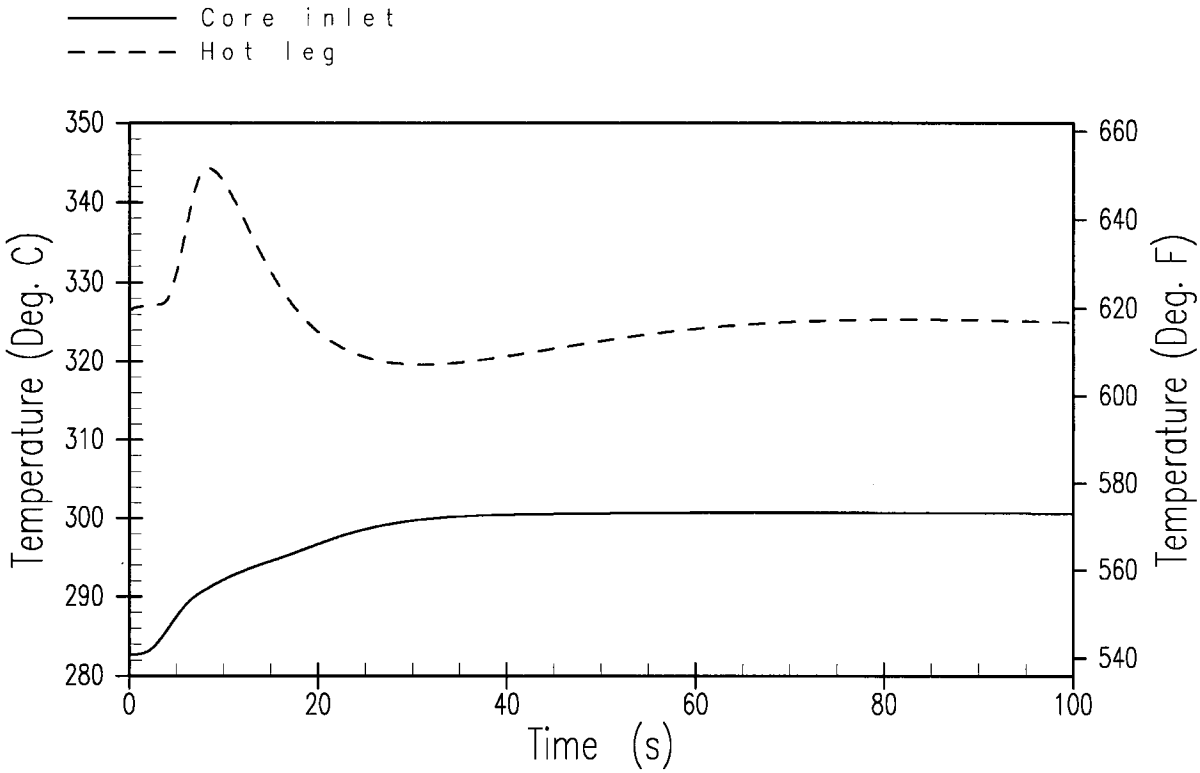

Figure 3-11 Reactor Vessel Temperatures vs. Time – w/LOOP Case

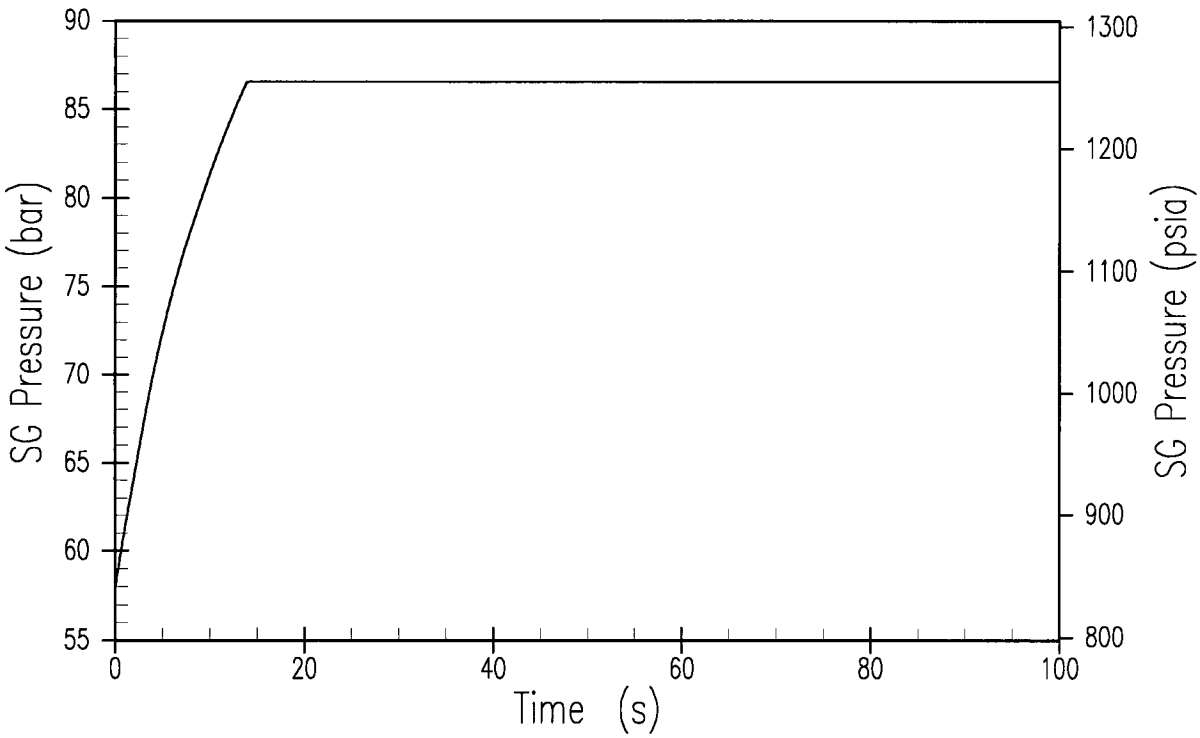

Figure 3-12 SG Pressure vs. Time – w/LOOP Case

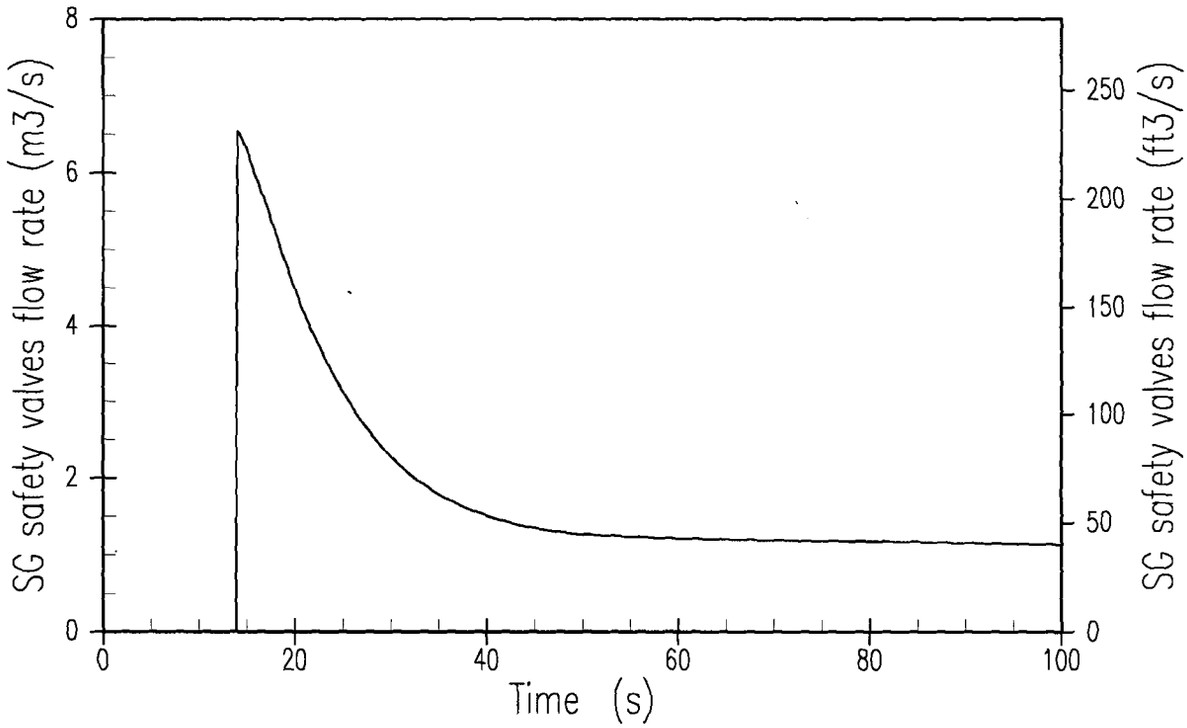

Figure 3-13 SG Safety Valve Flow Rate vs. Time – w/LOOP Case

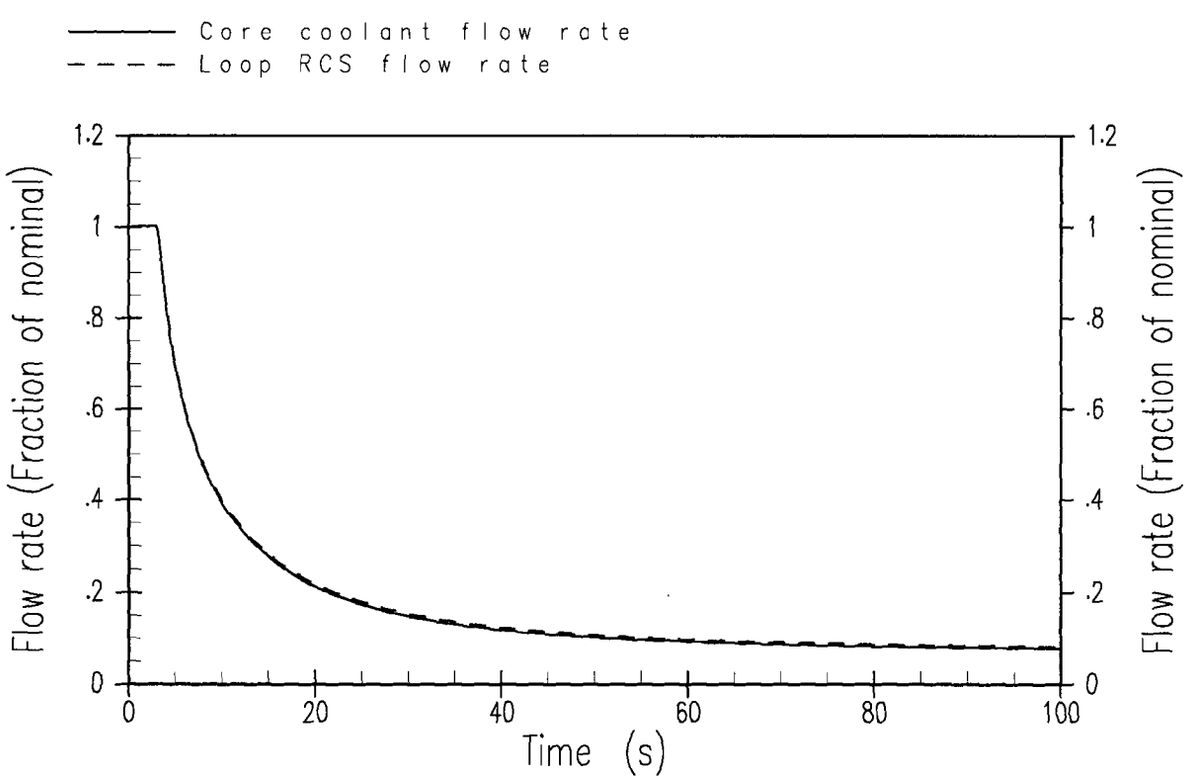

Figure 3-14 RCS Flow Rates vs. Time – w/LOOP Case

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4. "Rod Withdrawal at Power Analysis," APP-GW-GL-700 Rev. 15 AP1000 Design Control Document- Tier 2 Section 15.4.2.

5. "Loss of Forced Reactor Flow Analysis," APP-GW-GL-700 Rev. 15 AP1000 Design Control Document- Tier 2 Section 15.3.2.

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