3/4.4.2 SAFETY/RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.2 a) The safety valve function of at least 12 of the following reactor coolant system safety/relief valves shall be OPERABLE with the specified code safety valve function lift settings:*

2 safety/relief valves @ 1150 psig +1%/-3%
4 safety/relief valves @ 1175 psig +1%/-3%
4 safety/relief valves @ 1185 psig +1%/-3%
4 safety/relief valves @ 1195 psig +1%/-3%
4 safety/relief valves @ 1205 psig +1%/-3%

APPLICABILITY: OPERATIONAL CONDITIONS 1, and 2, ~~and 3~~. when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER

Insert A

ACTION:

a. With the safety valve function of one or more of the above required safety/relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. With one or more safety/relief valves stuck open, provided that suppression pool average water temperature is less than 90°F, close the stuck open safety/relief valve(s); if unable to close the open valve(s) within 2 minutes or if suppression pool average water temperature is 110°F or greater, place the reactor mode switch in the Shutdown position.

c. With one or more safety/relief valve acoustic monitors inoperable, restore the inoperable monitor(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.2 The acoustic monitor for each safety/relief valve shall be demonstrated OPERABLE by performance of a:

a. CHANNEL CHECK at least once per 31 days, and a

b. CHANNEL CALIBRATION at least once per 18 months.**

*The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures.

**The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

INSERT A

b)

~~3.4.2~~ The safety valve function of at least ~~12~~ 4 of the following reactor coolant system safety/relief valves shall be OPERABLE with the specified code safety valve function lift settings:*

2 safety/relief valves @ 1150 psig +1%/-3%
4 safety/relief valves @ 1175 psig +1%/-3%
4 safety/relief valves @ 1185 psig +1%/-3%
4 safety/relief valves @ 1195 psig +1%/-3%
4 safety/relief valves @ 1205 psig +1%/-3%

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3, when THERMAL POWER is less than 25% of RATED THERMAL POWER.

3/4.4.2 SAFETY/RELIEF VALVES (Continued)

the dual purpose safety/relief valves in their ASME Code qualified mode (spring lift) of safety operation.

The overpressure protection system must accommodate the most severe pressurization transient. There are two major transients that represent the most severe abnormal operational transient resulting in a nuclear system pressure rise. The evaluation of these events with the final plant configuration has shown that the MSIV closure is slightly more severe when credit is taken only for indirect derived scrams; i.e., a flux scram. Utilizing this worse case transient as the design basis event, a minimum of 12 safety/relief valves are required to assure peak reactor pressure remains within the Code limit of 110% of design pressure.

[Insert B]

~~Demonstration of the safety/relief valve lift settings will occur only during shutdown and will be performed in accordance with the provisions of Specification 4.0.5.~~

3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.3.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

INSERT B

Testing of safety/relief valves is normally performed at low power. It is desirable to allow an increased number of valves to be out of service during testing. Therefore, an evaluation of the MSIV closure without direct scram was performed at 25% of RATED THERMAL POWER assuming only 4 safety/relief valves were operable. The results of this evaluation demonstrate that any 4 safety/relief valves have sufficient flow capacity to assure that the peak reactor pressure remains well below the code limit of 110% of design pressure.

Demonstration of the safety/relief valve lift settings will be performed in accordance with the provisions of specification 4.0.5.

ATTACHMENT 2

LOW POWER/FLOW ASME OVERPRESSURIZATION CALCULATION RESULTS

This analysis provides support for a proposed WNP-2 Technical Specification modification to increase the number of safety/relief valves that can be out of service when the plant is at a low power/flow condition. The Technical Specification change is desirable for checkout of safety/relief valves after valve maintenance has been performed.

ANF was requested to perform an ASME overpressurization transient calculation for WNP-2 for the conditions in Table 1. For this event the maximum system pressure is calculated for a containment isolation which is the rapid closure of all main steam isolation valves (MSIVs). The analysis results show that for WNP-2, at Table 1 conditions, four safety/relief valves in service have sufficient capacity to prevent the reactor vessel pressure from reaching the established transient pressure safety limit of 1375 psig (110% of the design pressure) specified by the ASME Pressure Vessel Code.

This overpressurization calculation was performed with the ANF advanced plant simulator code COTRANSA, which includes an axial one-dimensional neutronics model. Neutronics data which represent WNP-2 for Cycle 5 with a 136 assembly reload batch size with Table 1 conditions were used in this calculation. The most critical active component (scram on MSIV closure) was assumed to fail during the transient.

The power rise is terminated by the increased heat transfer and voiding of the core. At about 4.8 seconds, the reactor scram is initiated by reaching the high vessel pressure trip setpoint (1071 psig). Pressures reach the recirculation pump trip setpoint (1170 psig) before the pressurization has been reversed. Loss of core flow leads to enhanced steam production as less subcooled water is available to absorb core thermal power. The calculated maximum pressure in the steam lines, which was 1221 psig, occurred near the vessel at about 11.4 seconds. The maximum vessel pressure was 1226 psig, and it occurred in the lower plenum at about 10.7 seconds. These results and other significant parameters are presented in Table 2, and Figures 1 through 3 show key calculated parameters.

ANF results for a number of boiling water reactors have shown that ASME overpressurization results vary little due to cycle specific neutronic effects, and this variation is well within the margin to the ASME criteria calculated for this low-power case. The conclusion regarding the adequacy of four safety relief valves is applicable to current and future WNP-2 operating cycles with the ANF 8x8 fuel design.

TABLE 1

Initial Parameter	Condition
Core Thermal Power	25% of rated
Core Flow	33% of rated
Steam/Feedwater Flow Rate	3.04×10^6 lb/hr
Core Exposure	EOC 5 (all rods out)
Steam Dome Pressure	1020 psia
Safety/Relief Valve Pressure Set Points	Valves With Highest Pressure Set Points Assumed Operable
Number of Safety/Relief Valves in Service	Four
Jet Pump M-Ratio	4.7
Feedwater Temperature	276 °F

TABLE 2

Calculated Parameter	Result
Peak Neutronic Power	67.3% of Rated @ 4 sec
Peak Steamline Pressure	1236 psia (1221 psig) @ 11.4 sec
Peak Dome Pressure	1231 psia (1216 psig) @ 10.7 sec
Peak Vessel Pressure	1241 psia (1226 psig) @ 10.7 sec
Peak Heat Flux	32.5% of Rated @ 5.9 sec

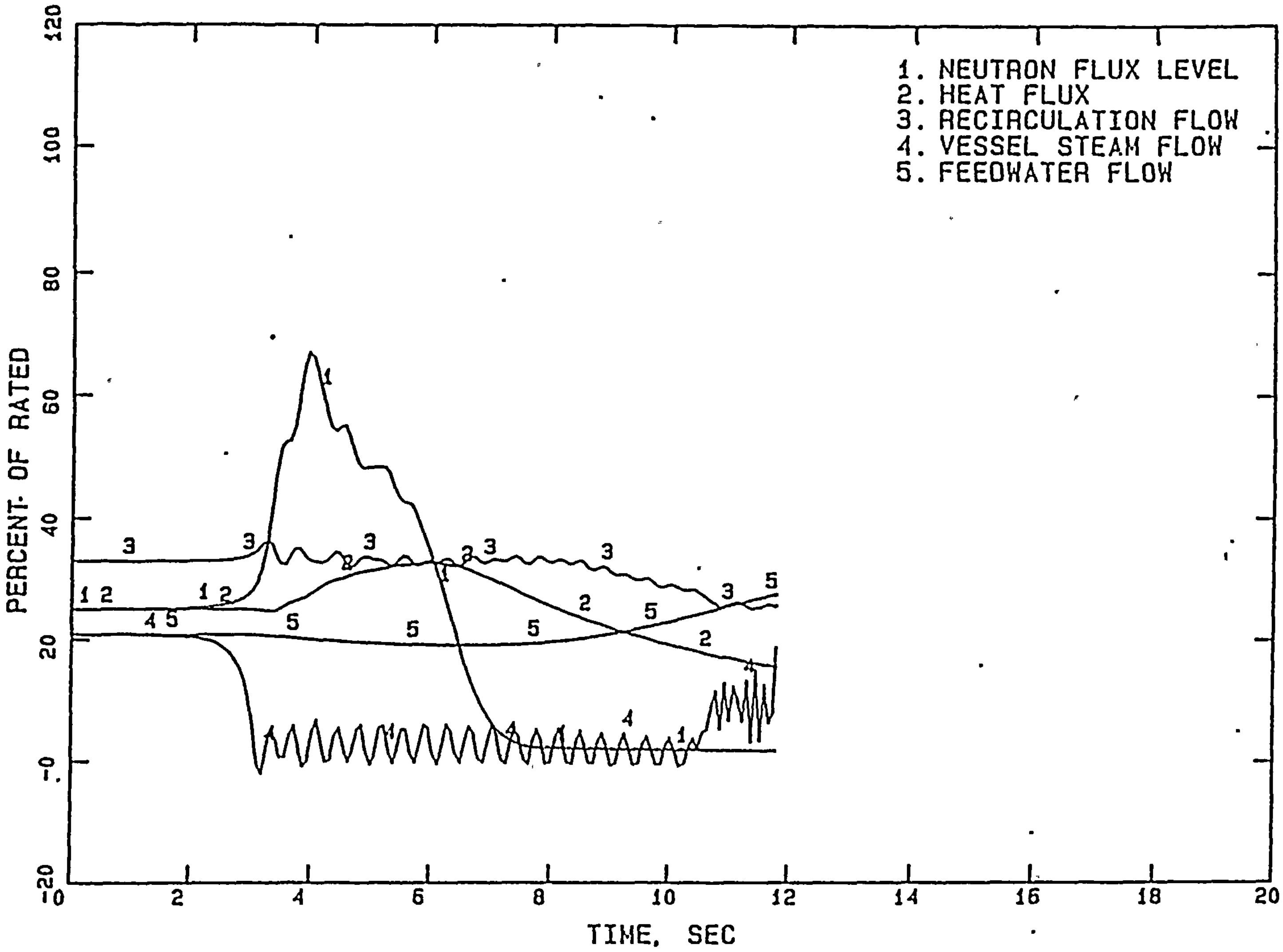

FIGURE 1 LOW POWER/FLOW ASME OVERPRESSIZURATION RESULTS, NORMAL SCRAM SPEED

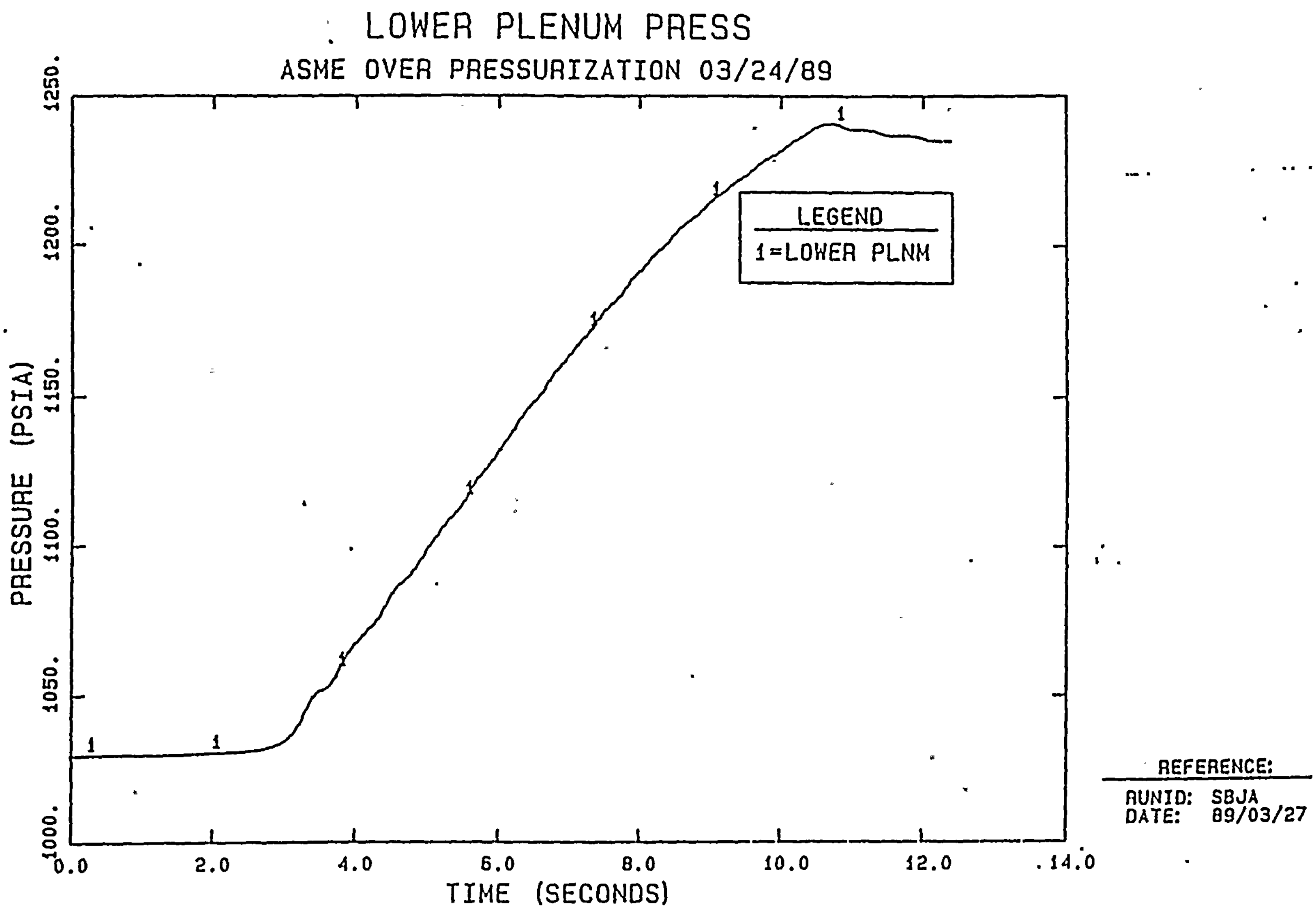

FIGURE 2 LOW POWER/FLOW ASME OVERPRESSURIZATION RESULTS, NORMAL SCRAM SPEED

SBKO 89/03/27.

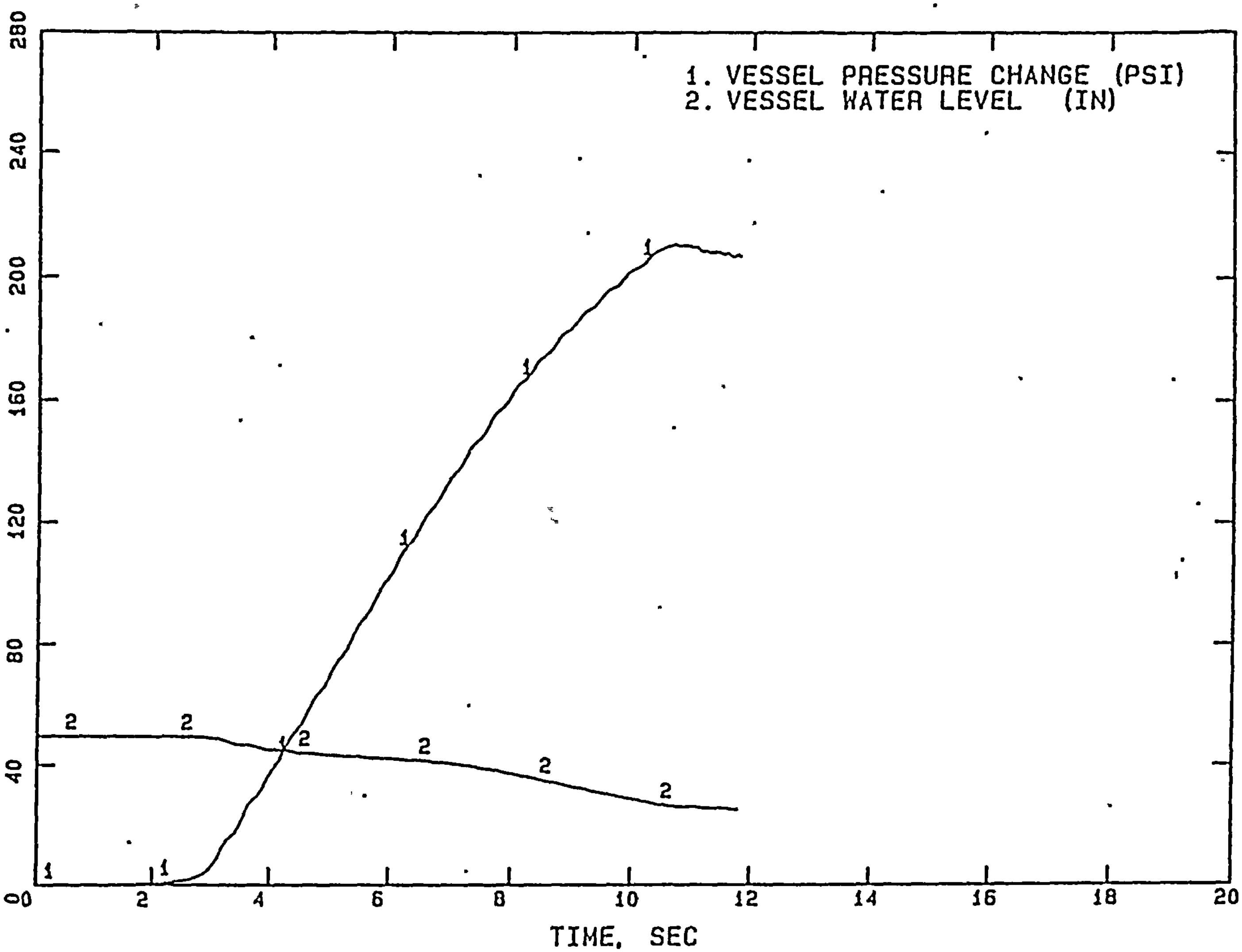

FIGURE 3 LOW POWER/FLOW ASME OVERPRESSURIZATION RESULTS, NORMAL SCRAM SPEED

SEQ. SBQK 89/03/28. 09.37.07.