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SUBJECT: Forwards ASME Boiler & Pressure Vessel Code,Section 3 overpressure rept per FSAR Question 211.204.

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September 12, 1980

Mr. B. J. Youngblood, Chief
Licensing Branch No. 1
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Docket Nos. 50-387
50-388

SUSQUEHANNA STEAM ELECTRIC STATION
FSAR QUESTION 211.204
ER 100450 FILE 841-2
PLA-544

Dear Mr. Youngblood:

As requested in FSAR Question 211.204, attached is a copy of the ASME Boiler and Pressure Vessel Code, Section III overpressure report for Susquehanna.

Very truly yours,

J. H. Henderson for
N. W. Curtis
Vice President-Engineering and Construction-Nuclear

CTC:mks

Attachment

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Abstract

This report provides sufficient information and documentation to show compliance with all requirements of Article 9 of ASME Pressure Vessel Code - Section III, 1968, Nuclear Vessels (up to and including Summer 1970 Addenda) in the area of the vessel overpressure protection design of the Susquehanna 1 and Susquehanna 2 nuclear pressure vessels. The effects on the vessel pressure transients of valve capacity are also shown.

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List of Illustrations

1. INTRODUCTION

1.1 The vessel overpressure protection system is designed to satisfy the requirements of Section III, Nuclear Vessels, of the ASME Boiler and Pressure Vessel Code. The general requirements for protection against overpressure as given in Article 9 of Section III of the Code recognize that reactor vessel overpressure protection is one function of the reactor protective systems and allows the integration of pressure relief devices with the protective systems of the nuclear reactor. Hence, credit is taken for the SCRAM protective system as a complementary pressure protection device.

2. DESIGN BASIS

2.1 Safety Valve Capacity. The safety valve capacity of this plant is adequate to limit the primary system pressure, including transients, to the requirements of the ASME Boiler and Pressure Vessel Code, Section III, 1968, Nuclear Vessels (up to and including Summer 1970 Addenda). The essential ASME requirements which are all met by this analysis are:

2.1.1 It is recognized that the protection of vessels in a nuclear power plant is dependent upon many protective systems to relieve or terminate pressure transients. Installation of pressure relieving devices may not independently provide complete protection.

2.1.2 The safety valve sizing evaluation assumes credit for operation of the SCRAM protective system which may be tripped by any one of two sources; i.e., a direct, or flux signal. The direct SCRAM signal is derived from position switches mounted on the main steamline isolation valves or the turbine stop valves or from pressure switches mounted on the dump valve of the turbine control valve hydraulic actuation system. The position switches are actuated when the respective valves are closing and following 10 percent travel of full stroke. The pressure switches are actuated when a fast closure of the control valves is initiated. Further, no credit is taken for power operation of the pressure relieving devices. Credit is taken for the dual purpose safety/relief valves in their ASME Code qualified mode of safety operation.

2.1.3 The nominal pressure setting of at least one safety/relief valve connected to any vessel or system shall not be greater than a pressure at the safety/relief valves corresponding to the design pressure (1250 psig) anywhere in the protected vessel.

2.1.4 The rated capacity of the pressure relieving devices shall be sufficient to prevent a rise in pressure within the protected vessel of more than 110 percent of the design pressure (1.10X1250psig = 1375 psig) for events defined in Paragraph 4.3.1.

2.1.5 Full account is taken of the pressure drop on both the inlet and discharge sides of the valves. All combination safety/relief valves discharge into the suppression pool through a discharge pipe from each valve which is designed to achieve sonic flow conditions through the valve; thus providing flow independence to discharge piping losses.

3. METHOD OF ANALYSIS

3.1 To design the pressure protection for the nuclear boiler system, extensive analytical models representing all essential dynamic characteristics of the system are simulated on a large computing facility. These models include the hydrodynamics of the flow loop, the reactor kinetics, the thermal characteristics of the fuel and its transfer of heat to the coolant, and all the principal controller features, such as feedwater flow, recirculation flow, reactor water level, pressure, and load demand. These are represented with all their principal non-linear features in models that have evolved through extensive experience and favorable comparison of analysis with actual BWR test data.

3.1.1 A detailed description of this model is documented in licensing topical report NEDO-10802, Analytical Methods of Plant Transient Evaluations for the GE-BWR, R.B. Linford. Included within this model, then, are components of the reactor vessel pressure protection system, which system is the subject of this report. Dual safety/relief valves are simulated in the non-linear representation, and the model thereby allows full investigation of the various valve response times, valve capacities, and actuation setpoints that are available in applicable hardware systems.

3.1.2 Typical capacity characteristics as modeled are represented in Figure 1 for the safety/relief valves. The associated bypass, turbine control valve, and mainsteam isolation valve characteristics are, of course, also represented fully in the model.

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4. SYSTEM DESIGN

4.1 A parametric study was conducted to determine the required steam flow capacity of the safety/relief valves based on the following assumptions.

4.2 Operating Conditions

4.2.1	Operating Power	-	3439MWt (104.4 percent of reactor rated power)
	Vessel Dome Pressure	-	1020 psig
	Steamflow	-	14.153×10^6 lbs/hr (105 percent of rated steam flow)

These conditions are the most severe because the maximum stored energy exists at these conditions. At lower power conditions the transients would be less severe.

4.3 Transients

4.3.1 The overpressure protection system must accommodate the most severe pressurization transient. There are two major transients, the closure of all main steam line isolation valves and a turbine/generator trip with a coincident failure of the turbine steam bypass system valves to open, that represent the most severe abnormal operational transients resulting in a nuclear system pressure rise. The evaluation of transient behavior with final plant configuration has shown that the isolation valve closure is slightly more severe when credit is taken only for indirect derived scrams, therefore, it is used as the overpressure protection basis event.

4.4 Scram

a. SCRAM reactivity curve - Figure 2

b. Control rod drive SCRAM motion - Figure 3

4.5 Safety/Relief Valve Transient Analysis Specifications

a. Valve groups - 5

b. Pressure setpoint - 1165 - 1205 psig (+1 percent assumed error)

4.6 Safety Valve Capacity

4.6.1 Sizing of the safety valve capacity is based on establishing an adequate margin from the peak vessel pressure to the vessel code limit (1375 psig) in response to the reference transients.

5. EVALUATION OF RESULTS

5.1. Safety Valve Capacity

5.1.1 The parametric relationship between peak vessel (bottom) pressure and safety valve capacity for the MSIV transient with high flux and position trip scram is described in Figure 4. Also shown in Figure 4 is the parametric relationship between peak vessel (bottom) pressure and safety valve capacity for the turbine trip with a coincident failure of the turbine bypass valves to open and direct scram, which is the most severe transient when direct scram is considered. Pressures shown for flux scram will result only with multiple failure in the redundant direct scram system.

5.1.2 The time response of the vessel pressure to the MSIV transient with flux scram and the turbine trip with a coincident failure of the turbine bypass valves to open and direct scram for 16 valves is illustrated in Figure 5. This shows that the pressure at the vessel bottom exceeds 1250 psig for less than 6 seconds which is not long enough to transfer any appreciable amount of heat into the vessel metal which was at a temperature well below 550°F at the start of the transient.

5.1.3 From the analytical models described in Paragraph 3 together with engineering studies, it has been determined that the safety/relief valve reclosing pressures, as specified in Paragraph 6.3.1, are acceptable.

6. SAFETY/RELIEF VALVE CHARACTERISTICS

6.1 Schematic Arrangement. The schematic arrangement of the safety/relief valves are shown in Figures 6 and 7.

6.2 Pressure Drop in Inlet and Discharge

6.2.1 Pressure drop on the piping from the reactor vessel to the valves is taken into account in calculating the maximum vessel pressures reported above.

6.2.2 Pressure drop in the discharge piping to the suppression pool is limited by proper discharge line sizing to prevent back pressure on each safety/relief valve from exceeding 40 percent of the valve inlet pressure, thus assuring choked flow in the valve orifice and no reduction of valve capacity due to the discharge piping. Each safety/relief valve has its own separate discharge line.

6.3 Safety/Relief Valve Description

6.3.1 These valves were manufactured by Crosby Valve and Gage Company to ASME, Section III Code, 1971 Edition. They comply with ASME III, Paragraph NB-7640 as safety valves with auxiliary actuating devices. Quantities and set points are as follows:

Quantity	Opening Set Point psig	Reclosing Pressure psig	ASME Rated Capacity at 103 percent of Set Pressure lb/hr minimum
2	1146	1020	862,400
4	1175	1046	883,950
4	1185	1055	891,380
3	1195	1064	898,800
3	1205	1072	906,250

7. CONCLUSION

7.1 Safety requirements have long demanded very high reliability in the reactor SCRAM functions. Recognition of this reliability as being completely adequate justification for these functions to contribute to vessel pressure protection is reflected in the Section III Code provisions. Actual General Electric design practice very conservatively applies the code provisions which results in margins even beyond those necessary to satisfy code limits which further enhances the reliability of vessel pressure protection.

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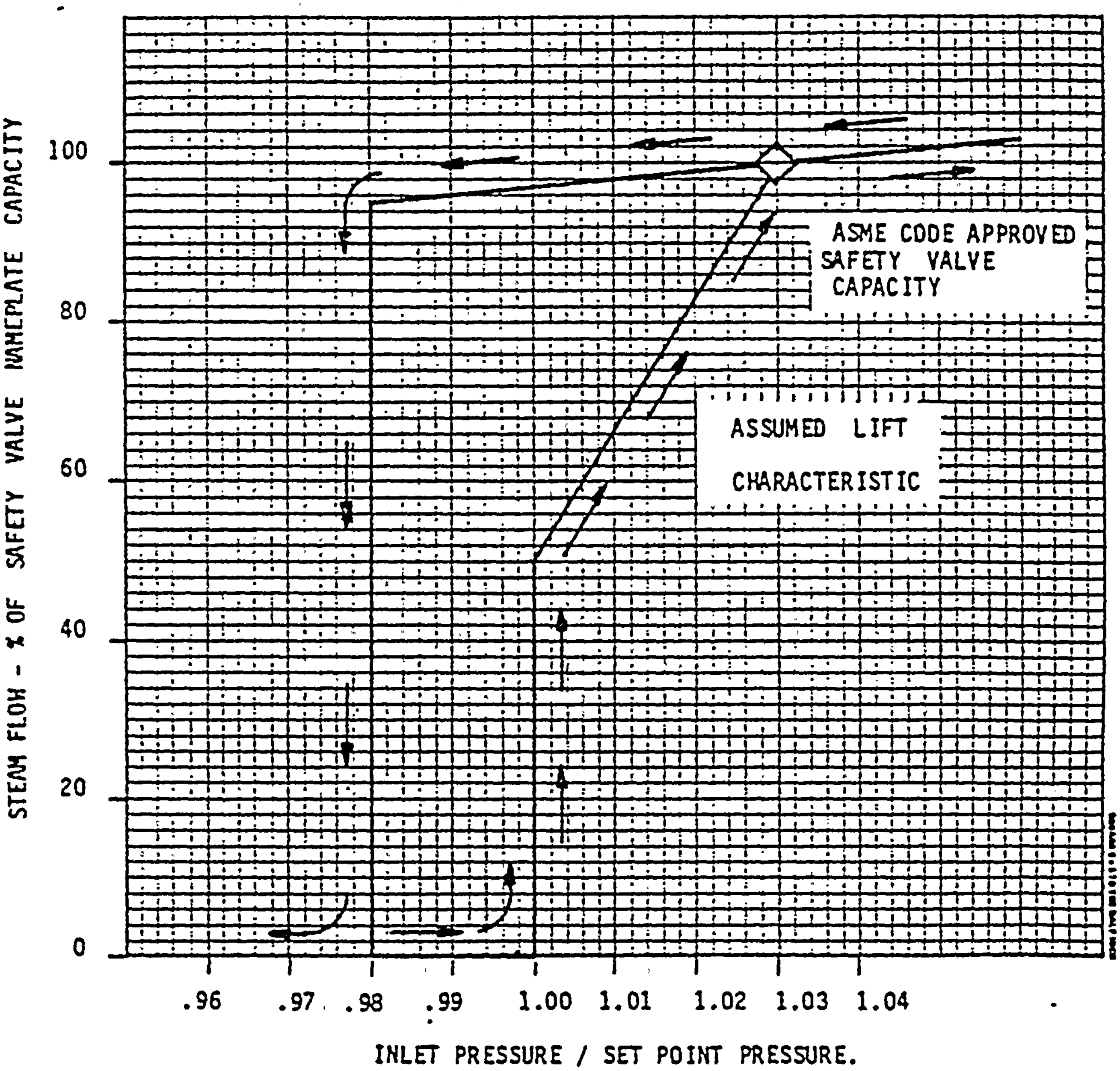

FIGURE 1 TYPICAL S/R VALVE CAPACITY CHARACTERISTIC

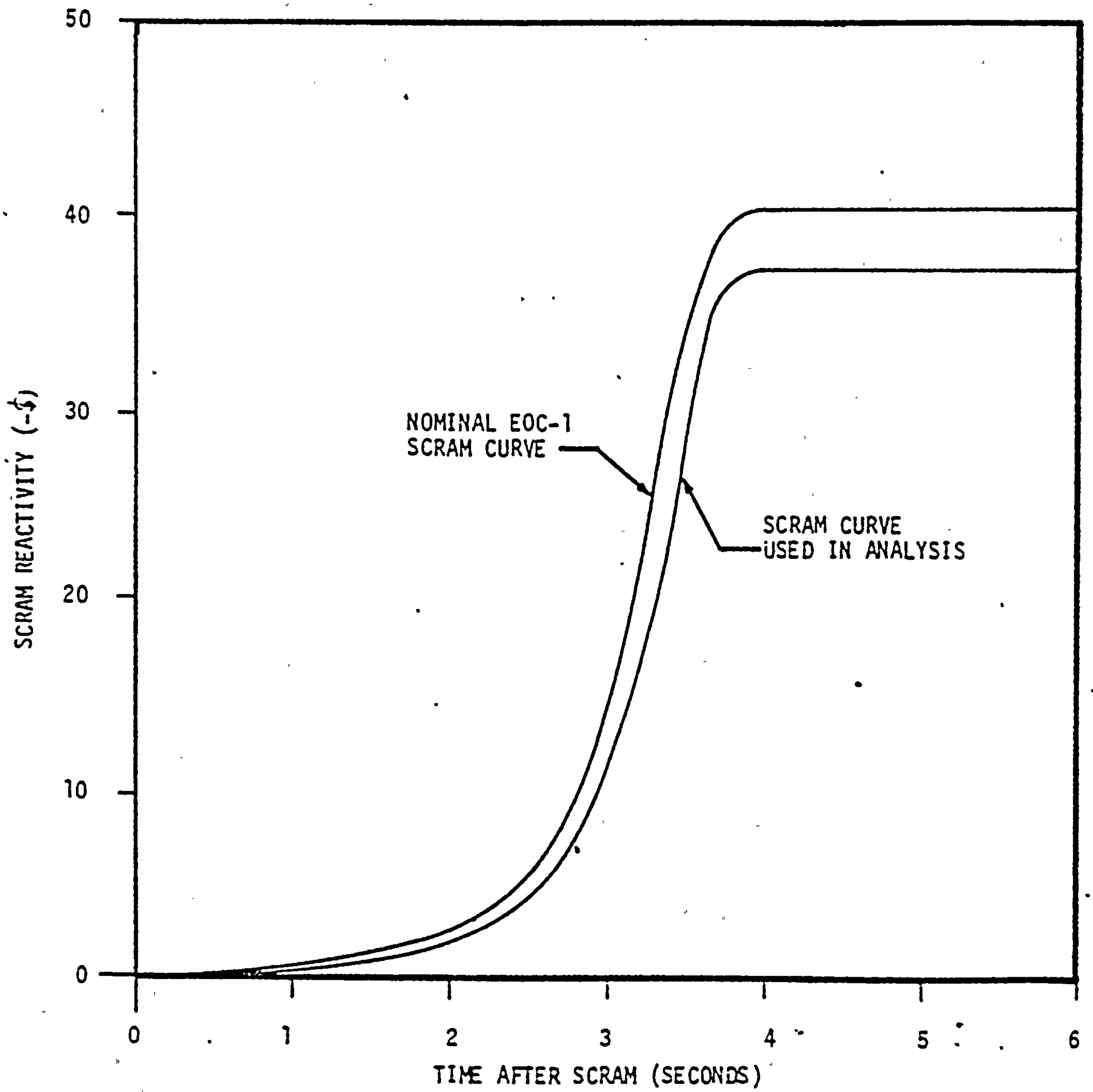

Figure 2 SCRAM Reactivity Versus Time

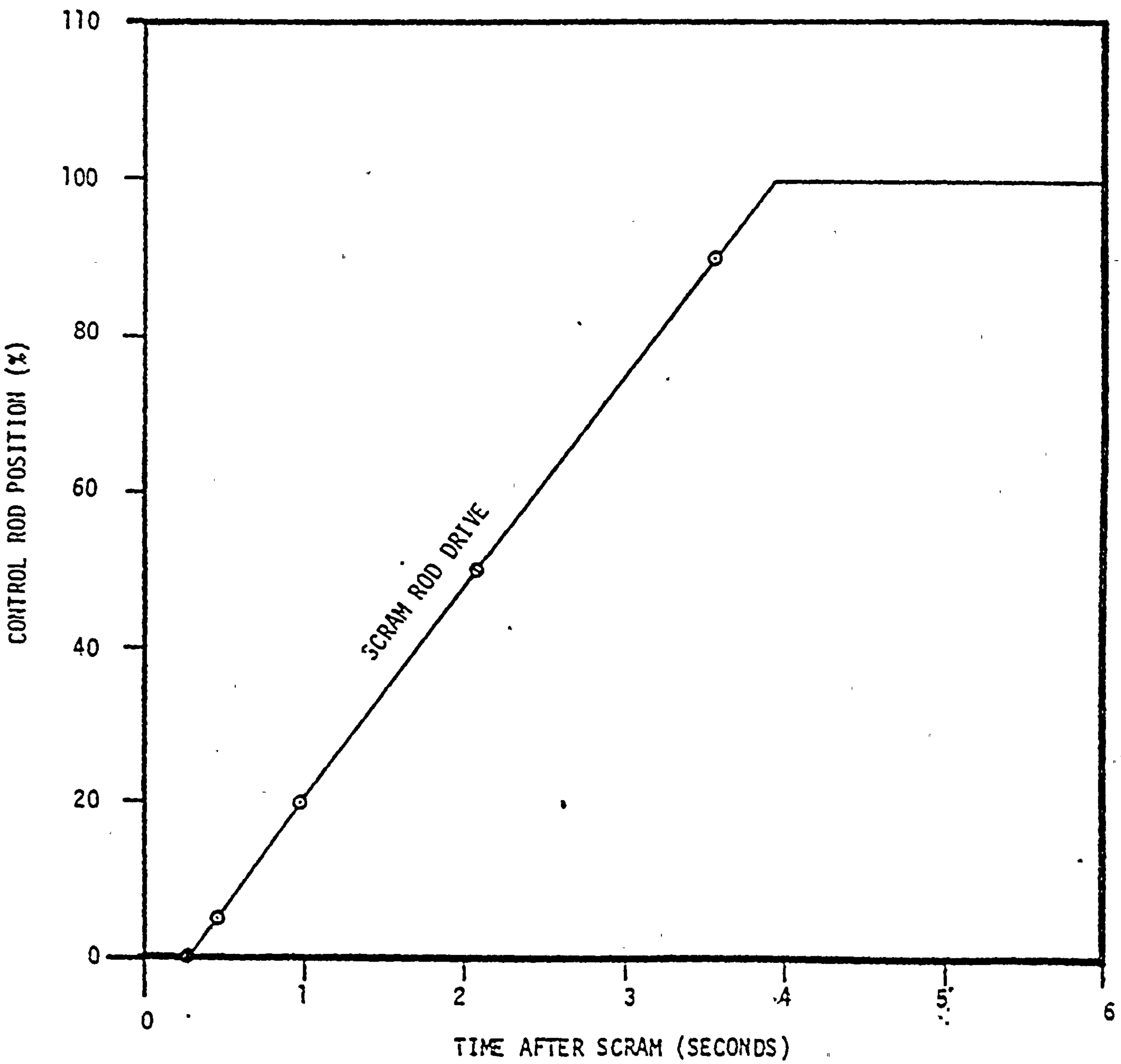

Figure 3 SCRAM Rod Drive Versus Time

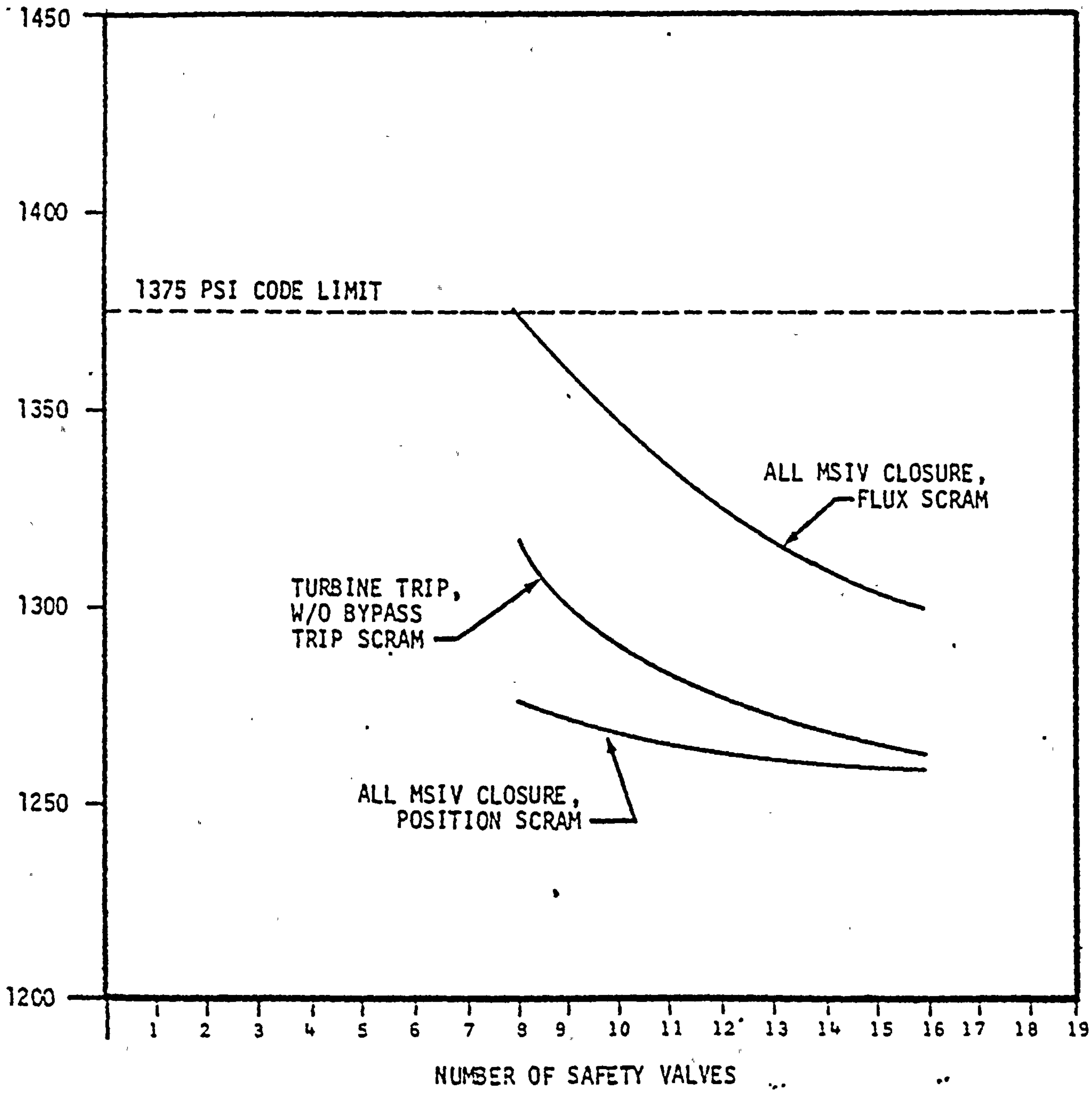

Figure 4 Peak Vessel Pressure Versus Safety/Relief Valve Capacity

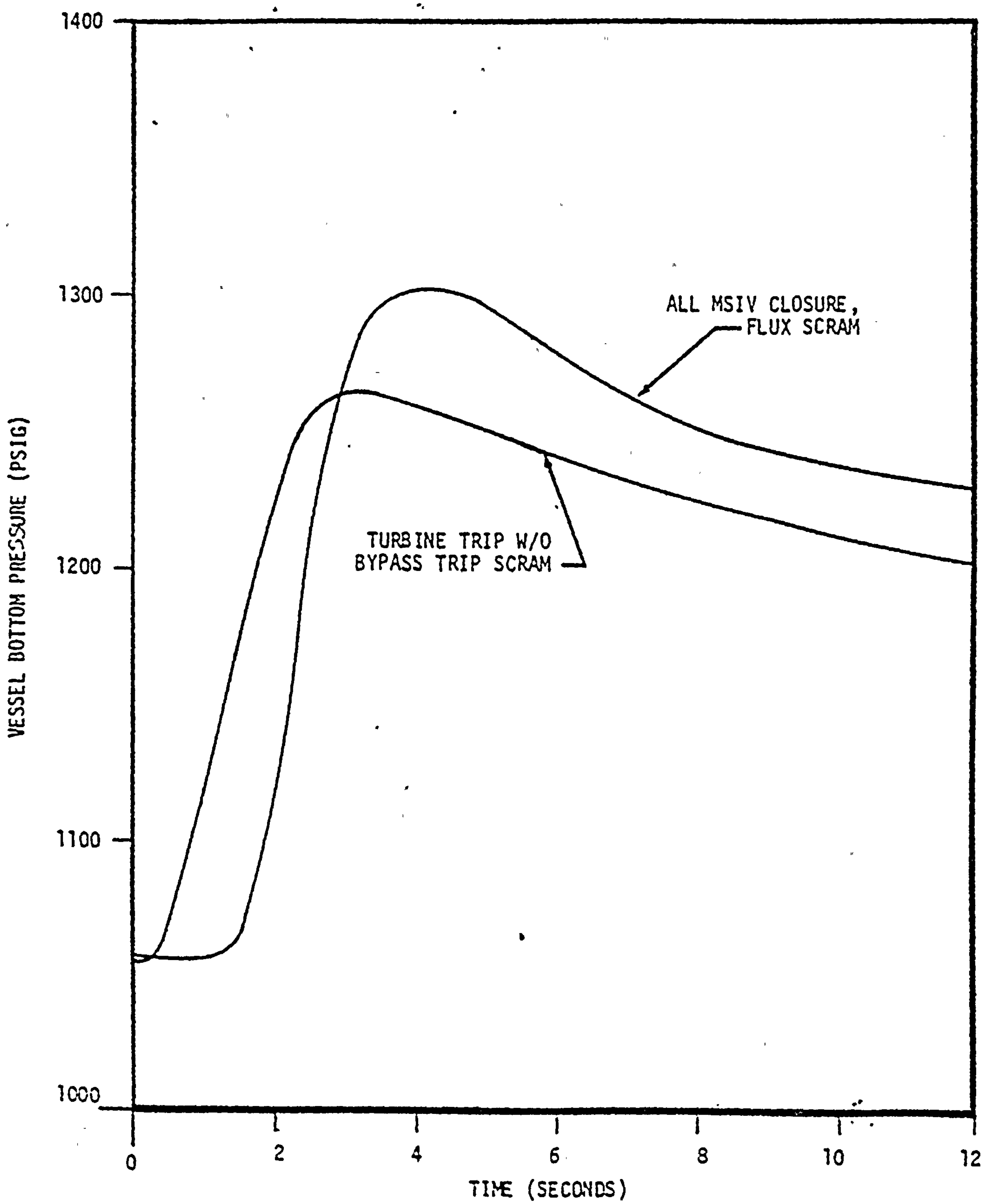

Figure 5 Time Response of Pressurization Transients

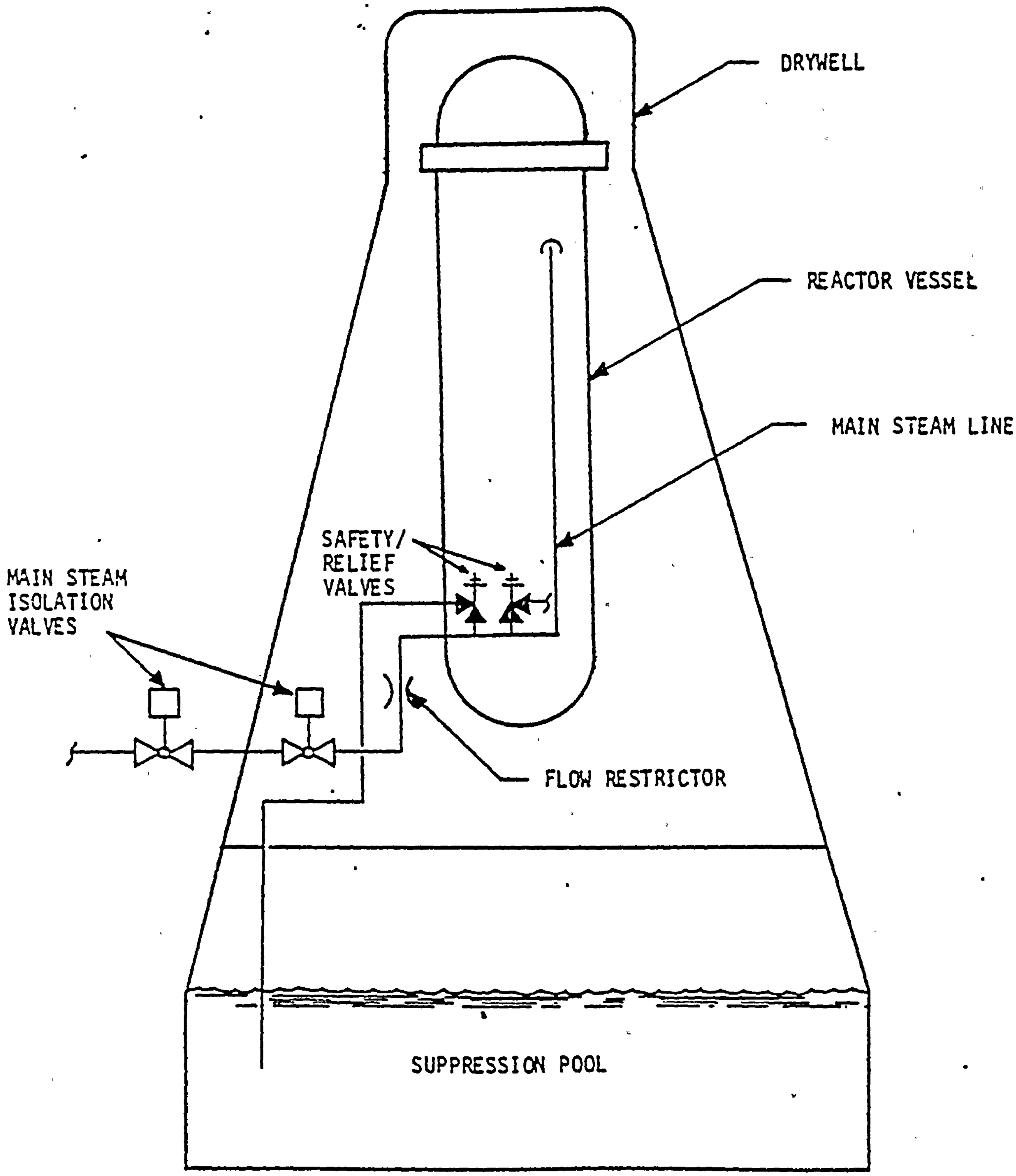

Figure 6 Safety/Relief Valve Schematic Elevation

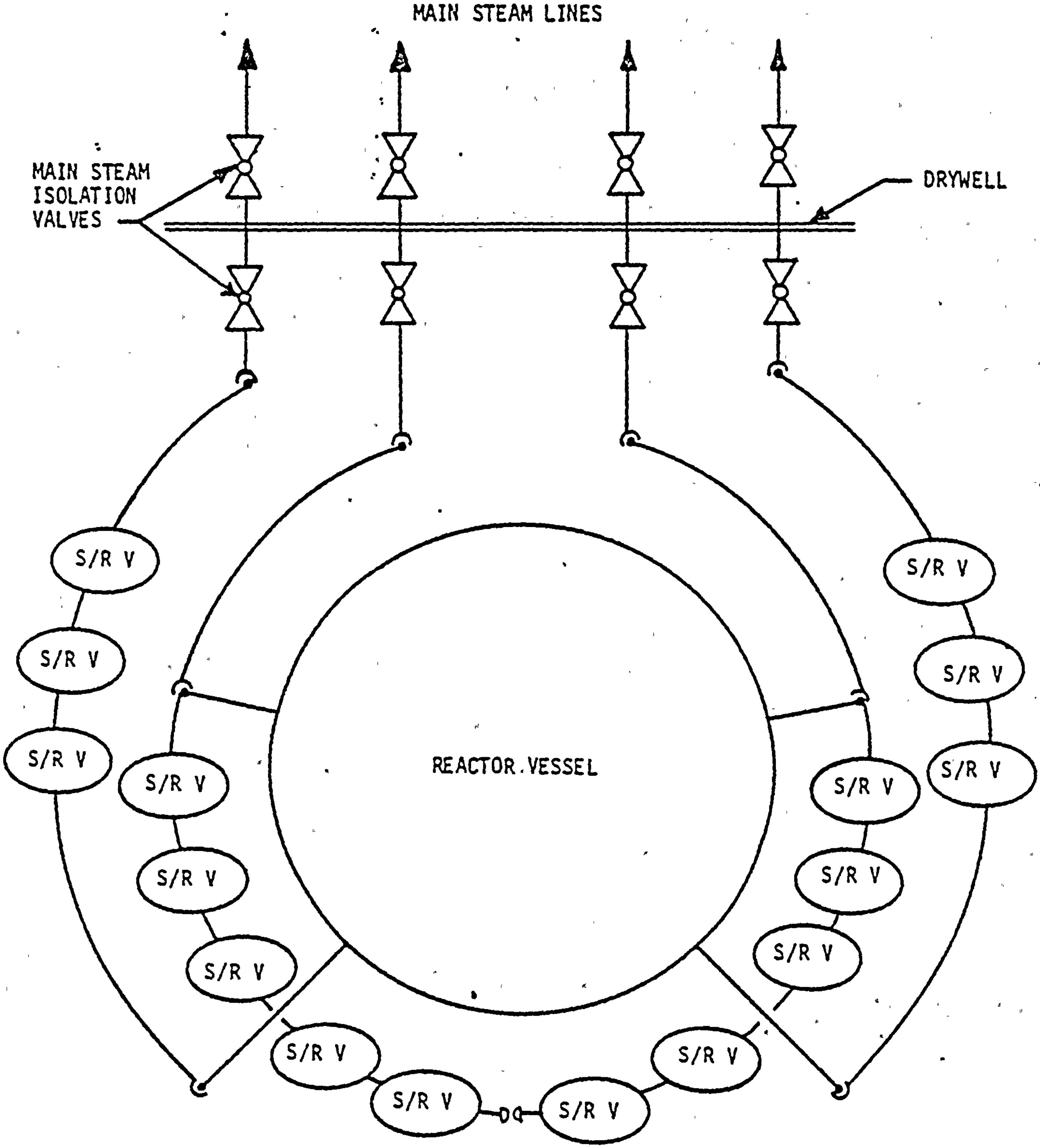

Figure 7 Safety/Relief Valve Schematic Plan