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NUCLEAR ENERGY DIVISION

DOCUMENT TITLE REACTOR VESSEL OVERPRESSURE PROTECTION

☐ SPECIFICATION ☐ DRAWING ☒ OTHER

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ABSTRACT

This report provides sufficient information and documentation to show compliance with all requirements of Article NB-7000 of the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, Division 1, 1971 Edition with Addenda to and including Winter 1972, in the area of overpressure protection design of the Nine Mile Point 2 nuclear pressure vessel and other Reactor Coolant Pressure Boundary (RCPB) components. The effects on the pressure transients of valve capacity are also shown.

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1. INTRODUCTION

1.1 The Reactor Coolant Pressure Boundary (RCPB) overpressure protection system is designed to satisfy the requirements of Section III, Nuclear Power Plant Components, of the ASME Boiler and Pressure Vessel Code. The general requirements for protection against overpressure, as given in Article NB-7000 of Section III of the Code, recognize that reactor vessel overpressure protection and protection of other RCPB components is one function of the reactor protective systems, and allows the integration of pressure relief devices with the protective systems of the nuclear reactor. Therefore the scram protective system is considered a complementary pressure protection device.

2. DESIGN BASIS

2.1 The safety/relief valve capacity of this plant is sized to limit the primary system pressure, including transients, to the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components. The essential ASME requirements which are all met by this analysis are:

a. It is recognized that the protection of vessels and other RCPB components 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.

b. The safety/relief valve sizing evaluation assumes credit for operation of the scram protective system which may be tripped by any one of two sources; ie, a direct or flux signal. The direct scram signal is derived from position switches mounted on the main steam line 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. Credit is taken for the dual purpose safety/relief valves in their ASME Code qualified modes of safety operation.

c. 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) of the protected vessel.

d. The rated capacity of the pressure relieving devices shall be sufficient to prevent a rise in pressure within the RCPB of more than 110 percent of the design pressure (1.10 x 1250 psig = 1375 psig) for events defined in Paragraph 4.3.1.

2.1 (Continued)

e. 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 RCPB, extensive analytical models representing all essential dynamic characteristics of the associated systems are simulated on a large digital computing facility. These models include the hydrodynamics of the flow loops, 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 nonlinear 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 these models (Paragraphs 8.2.a and 8.2.b) are documented as licensing topical reports. Included within these models 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 nonlinear representation and the models thereby allow full investigation of the various valve response times, valve capacities, and actuation setpoints that are available in applicable hardware systems.

3.1.2 Typical valve characteristics, as modeled, are represented in Figure 1 for the spring-action safety mode of the safety/relief valves. The associated bypass, turbine control valve, and main steam isolation valve characteristics are, of course, also represented fully in the models.

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 <u>Operating Conditions</u>

4.2.1 Operating Power	3463 MWT (104.2 percent of reactor rated power) [REDY]
	3466 MWT (104.3 percent of reactor rated power) [ODYN]
Vessel Dome Pressure	1020 psig
Steam Flow	15.013×10^6 lb/hr (105 percent of reactor rated steam flow)

The specified operating conditions are the most severe with respect to overpressure transient response, 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 closure of the turbine steam bypass system valves) 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 (REDY)* - Figure 2

b. Control rod drive scram motion - Figure 2

4.5 Safety/Relief Valve Transient Analysis Specifications

a. Valve groups - spring-action safety mode - 5

b. Pressure set point (maximum safety limit)

spring-action safety mode - 1177-1217 psig

4.6 Safety/Relief Valve Capacity

4.6.1 Sizing of the safety/relief 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.

* Figure 2 represents input to REDY computer program. For ODYN, scram reactivity for each transient analyzed is calculated within the ODYN computer program.

5. EVALUATION OF RESULTS

5.1 The parametric relationship between peak vessel (bottom) pressure and safety/relief valve capacity for the MSIV closure transient with high neutron flux scram is described in Figure 3. Also shown in Figure 3 is the parametric relationship between peak vessel (bottom) pressure and safety/relief valve capacity for the generator load rejection (with a coincident closure of the turbine bypass valves) 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.

5.2 The time response of the vessel pressure to the MSIV transient with flux scram and the generator load rejection (with a coincident closure of the turbine bypass valves) and direct scram for 18 valves is illustrated in Figure 4. This shows that the pressure at the vessel bottom exceeds 1250 psig for less than five second, 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.

5.3 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 arrangements of the safety/relief valves are shown in Figures 5 and 6.

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.

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 G. Dikkers and Company to meet ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, 1974 Edition with Addenda to and including Summer 1976. They comply with ASME III, Paragraph NB-7640 as safety valves with auxiliary actuating devices. Quantities, set points and associated capacities are as follows:

6.3.1 (Continued)

Quantity	Opening Set Point psig	Minimum Reclosing Pressure psig	ASME Rated Capacity at 103 percent of Set Pressure lb/hr minimum
2	1148	1022	882,000
4	1175	1046	902,000
4	1185	1055	910,000
4	1195	1064	917,000
4	1205	1072	925,000

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 RCPB 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 RCPB pressure protection.

8. REFERENCES

8.1 American Society of Mechanical Engineers, ASME Boiler and Pressure Vessel Code; Section III, Nuclear Power Plant Components

8.2 General Electric Documents:

a. Analytical Methods of Plant Transient Evaluation of GE-BWR, by R. B. Linford - NEDE-10802 (REDY)

b. One-Dimensional Core Transient Model for Boiling Water Reactors - NEDO-24154 (ODYN)

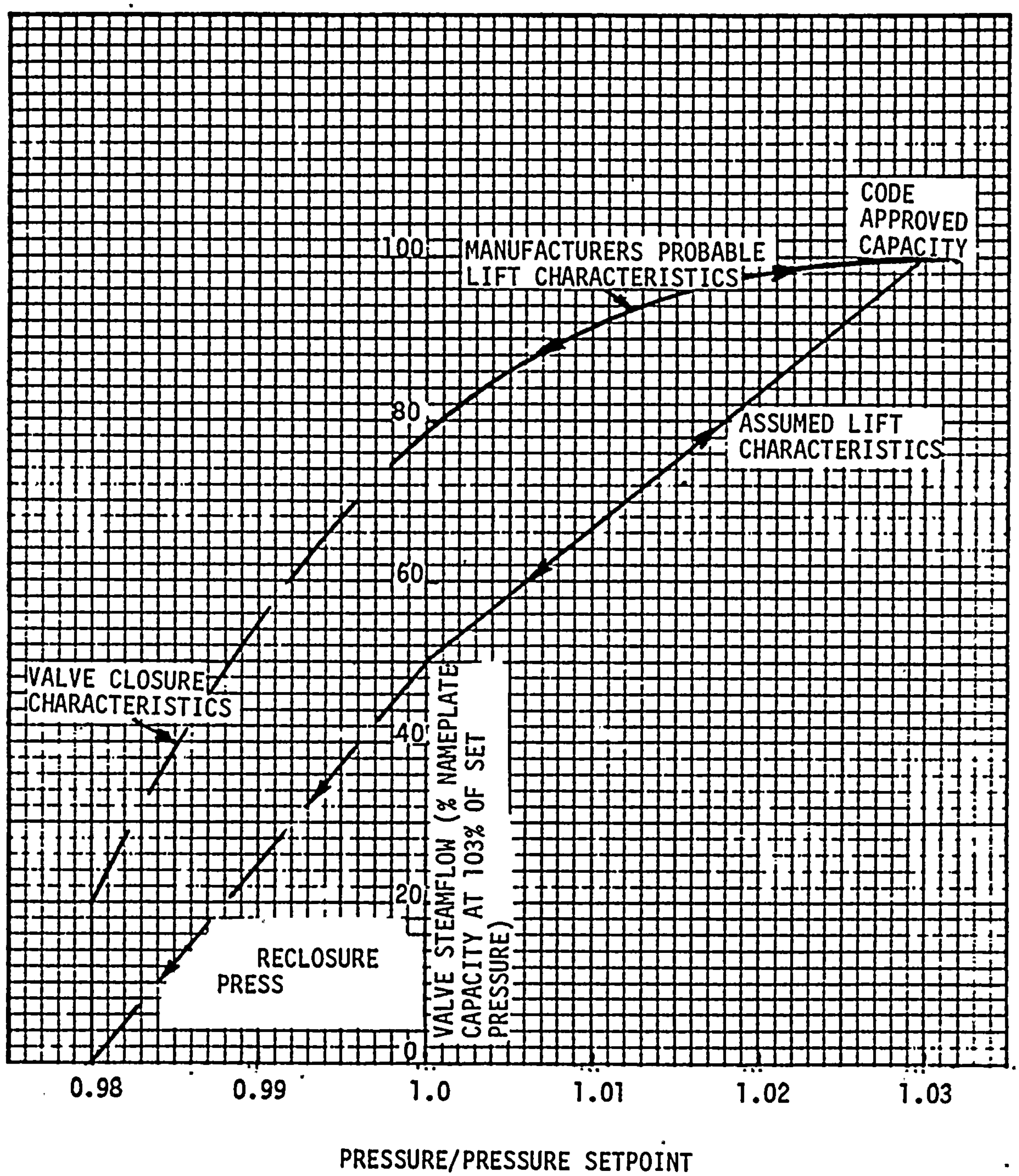

FIGURE 1 TYPICAL DUAL SAFETY/RELIEF VALVE CAPACITY CHARACTERISTICS - SPRING ACTION SAFETY MODE

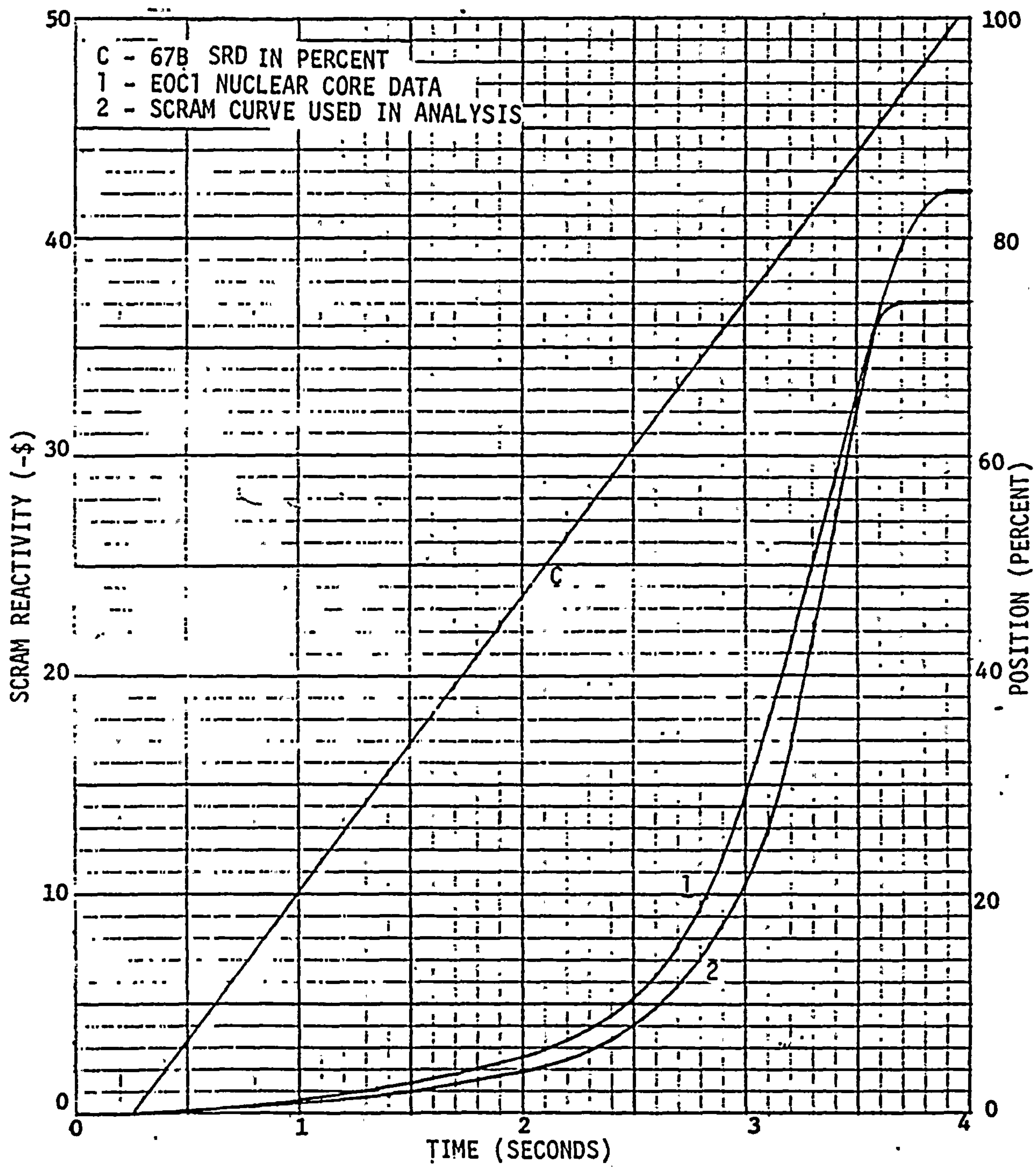

FIGURE 2 SCRAM REACTIVITY CURVE (REDY) AND SCRAM ROD DRIVE VERSUS TIME

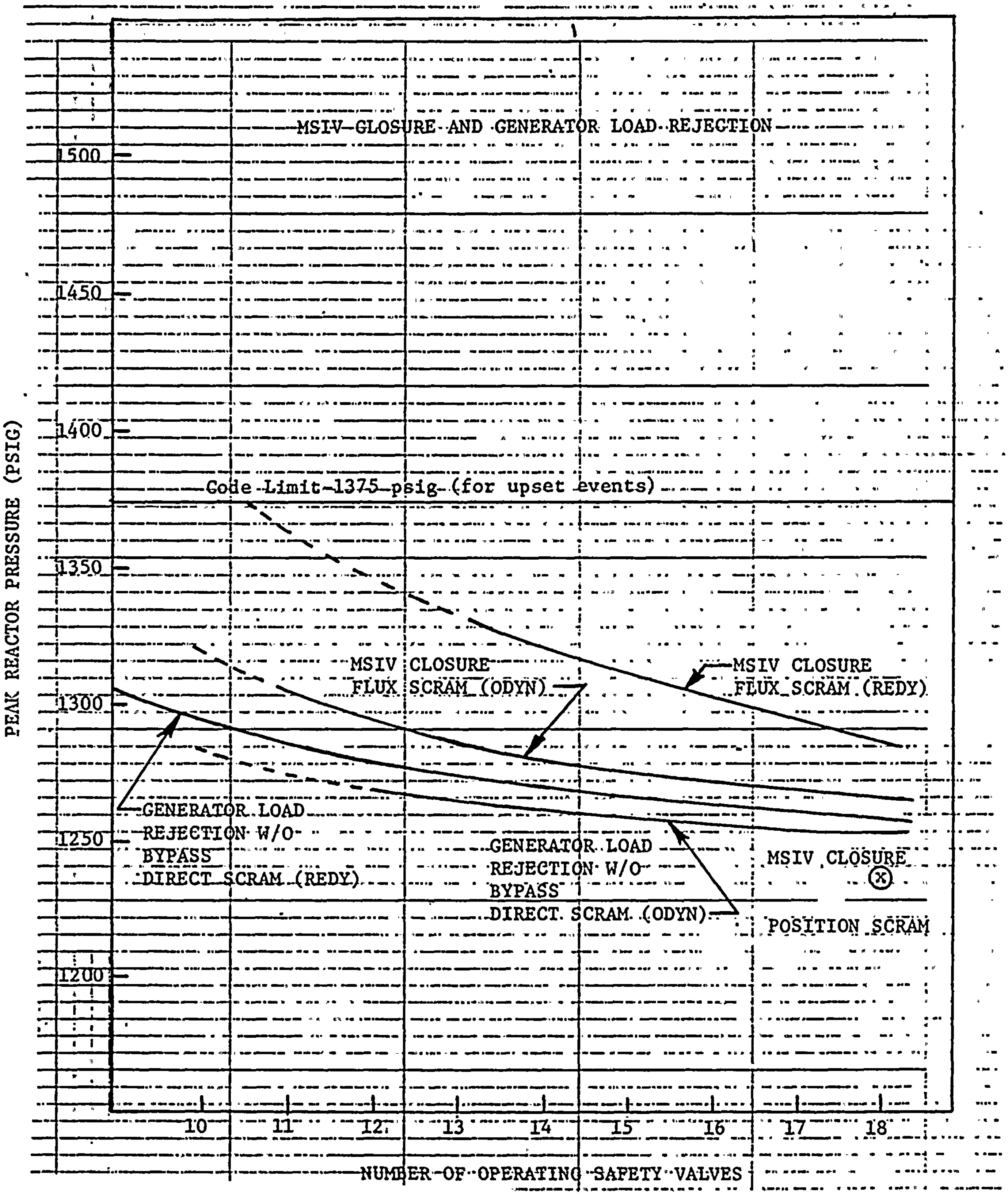

FIGURE 3 PEAK REACTOR PRESSURE VERSUS SAFETY VALVE CAPACITY

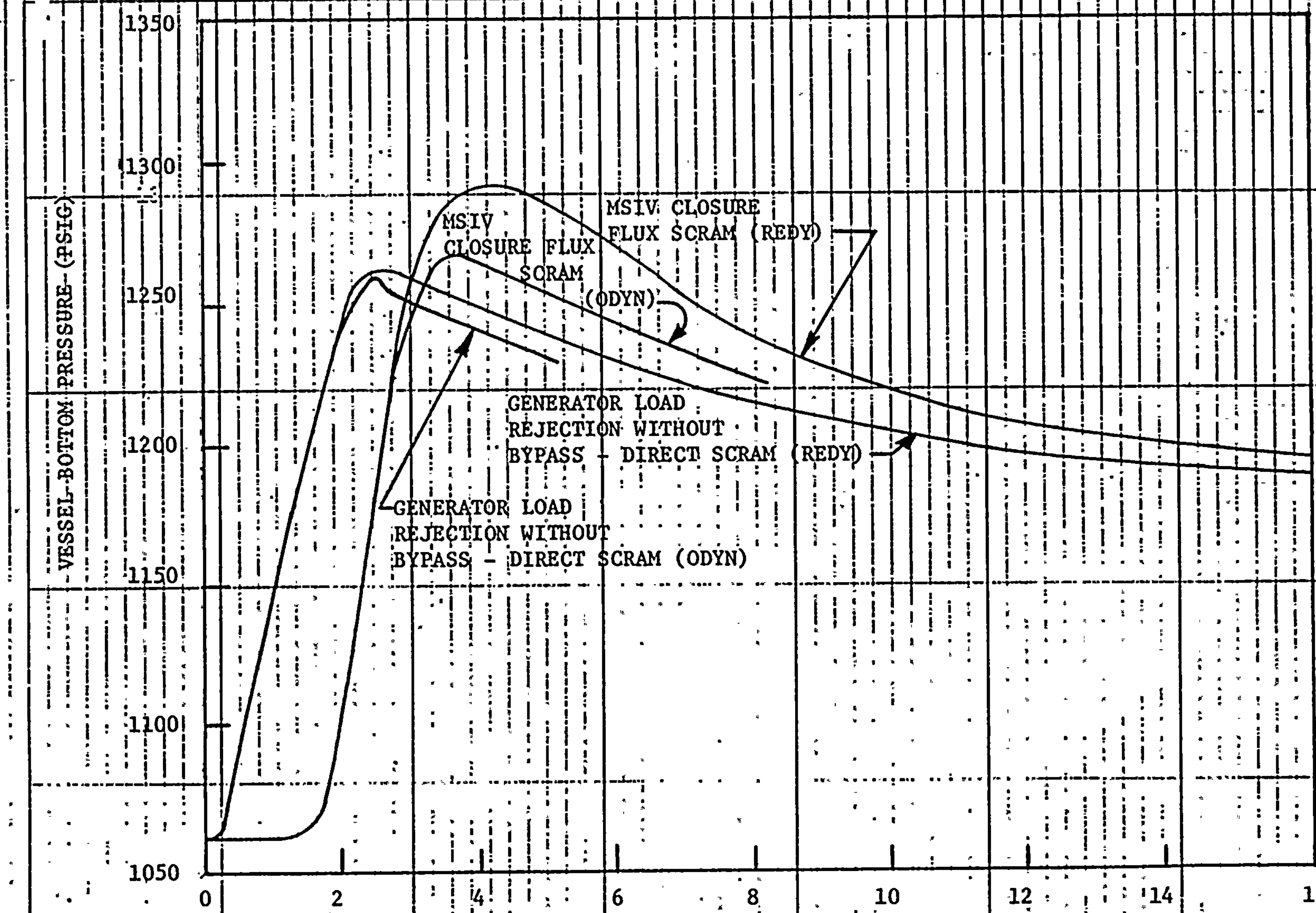

FIGURE 4 TIME RESPONSE FOR PRESSURIZATION TRANSIENTS

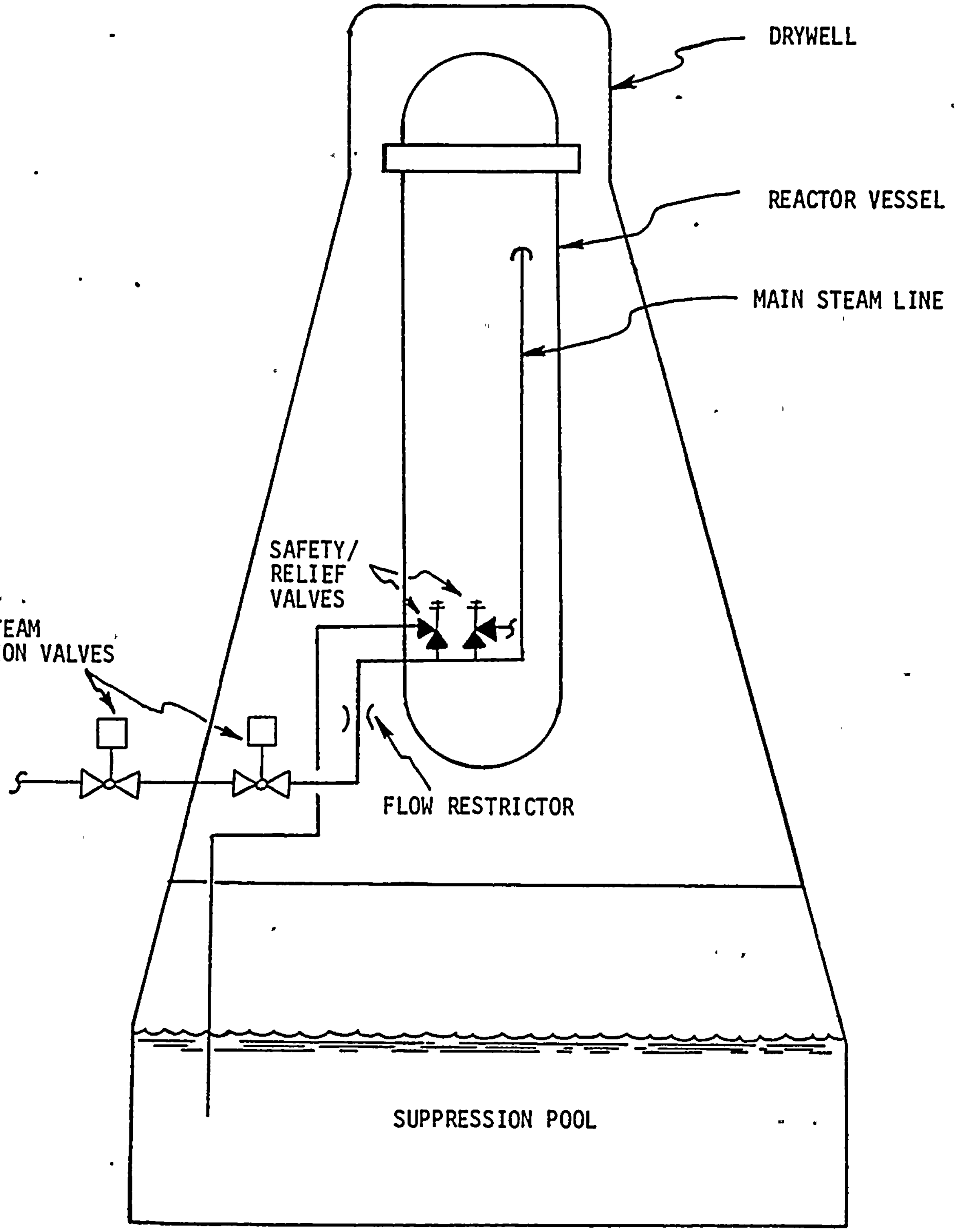

FIGURE 5 SCHEMATIC ELEVATION

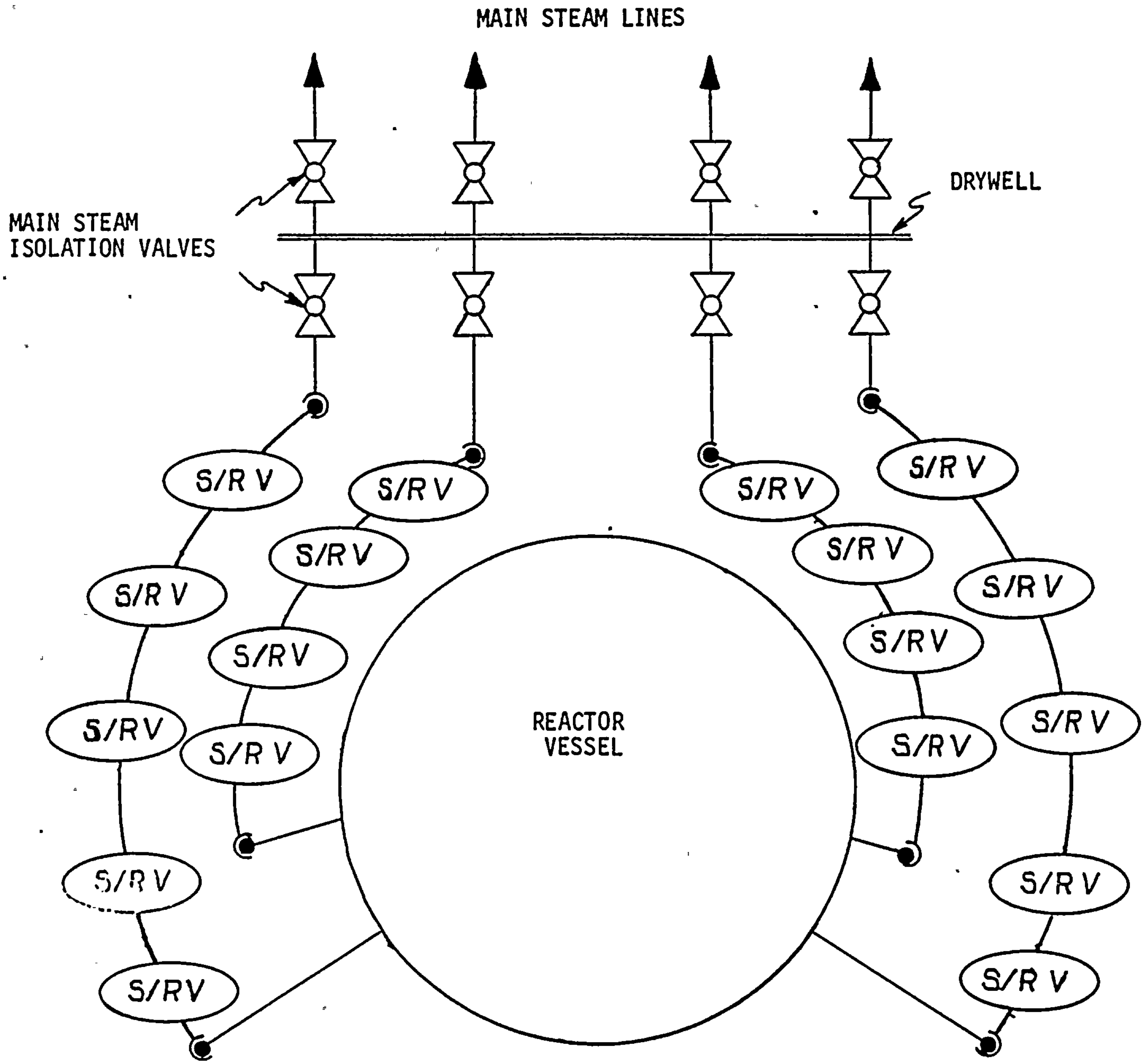

FIGURE 6 SCHEMATIC PLAN