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YANKEE

1671 Worcester Road, Framingham, Massachusetts 01701

2.C.2.1
FYR 82-82

August 1, 1982

United States Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. Dennis M. Crutchfield, Chief
Operating Reactors Branch No. 5
Division of Licensing

References: (a) License No. DPR-3 (Docket No. 50-29)
(b) YAEC Letter to USNRC, dated March 30, 1982 (FYR 82-38)
(c) YAEC Letter to USNRC, dated July 1, 1982 (FYR 82-72)

Subject: TMI Item II.D.1, Safety and Relief Valves

Dear Sir:

The following additional information on the testing of safety and relief valves as installed at the Yankee Plant is submitted in accordance with the requirements of NUREG-0578, Section 2.1.2, as later qualified by NUREG-0737, Item II.D.1 and the USNRC letter, dated September 29, 1981.

Reference (c) committed to supply additional detailed plant specific test condition justification by 8/1/82. The attached report "Valve Inlet Fluid Conditions for Pressurizer Safety and Relief Valves, Yankee Nuclear Power Station" fulfills that commitment.

If you have any questions or desire additional information, please contact us.

Very truly yours,

YANKEE ATOMIC ELECTRIC COMPANY

J. A. Kay

J. A. Kay
Senior Engineer - Licensing

Attachment

A046

VALVE INLET FLUID CONDITIONS
FOR PRESSURIZER SAFETY AND RELIEF VALVES
YANKEE NUCLEAR POWER STATION

YANKEE ATOMIC ELECTRIC COMPANY
Nuclear Engineering Department
July 1982

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1.0 INTRODUCTION

In the aftermath of the Three Mile Island (TMI) accident, the Nuclear Regulatory Commission (NRC) has required that utilities operating and constructing pressurized water reactor (PWR) power plants demonstrate the operability of pressurizer safety and relief valves. These requirements were issued in NUREG-0578 (Reference 1) and later clarified in NUREG-0737 (Reference 2). In response to these requirements, the PWR Industry is conducting a comprehensive program to test various types of safety and power-operated relief valves (PORVs) utilized in domestic PWR units. The objective of the test program is to demonstrate valve operability for fluid conditions which are prescribed in conventional licensing analyses.

As a supplement to the test program, Yankee Atomic Electric Company (YAEC) has initiated supporting studies for the Yankee Nuclear Power Station (YNPS). The particular study which is the subject of this report is intended to provide supporting data to demonstrate that the fluid test conditions being used in the PWR Industry Valve Test Program are applicable to the YNPS.

The objective of this study is to develop information to assist in the justification of the applicability to the YNPS of the inlet fluid conditions selected for the testing of pressurizer safety and relief valves in the PWR Industry Valve Test Program. This report is intended to document the fluid conditions under which the safety and relief valves are shown, in safety analysis reports/reload analyses, to actuate. Cold pressurizations and high pressure injection events are also considered. Cold pressurization events are characterized as low temperature overpressure protection (LTOP) events.

The scope of this study was to review the various sources containing information on pressurization events at the YNPS and to present the inlet fluid conditions for those events for which safety and/or PORV actuation is calculated to occur.

The sources of information on valve inlet fluid conditions include plant safety analysis, Systematic Evaluation Program (SEP) documentation, and the most recent fuel reload analysis. In addition, since the YNPS utilizes the PORV for low temperature overpressure protection (LTOP), the PORV inlet fluid conditions were based on LTOP analyses performed by YAEC. Finally, the actuation of safety valves and/or PORV as a result of the extended operation of the high pressure safety injection (HPSI) pumps was investigated.

2.0 DESCRIPTION OF YNPS DESIGN

2.1 General

The YNPS nuclear steam supply system, a pressurized light-water reactor designed by the Westinghouse Electric Corporation, consists of four closed piping loops connected in parallel to the reactor vessel. Each loop contains a steam generator, a main coolant pump, a check valve to prevent reverse flow, and a hot and a cold leg isolation valve. This system transfers the heat generated in the nuclear core to the steam generators where steam is produced to drive the turbine generator.

The steam and power conversion system is designed to accept a turbine trip without reactor trip at full reactor power without overpressurizing the secondary system. Decay heat can be removed from the reactor core by using emergency feed pumps and steam generator safety valves. No portion of the steam and power conversion system downstream of the non-return valves (NRVs) is needed for plant safety purposes.

A simplified flow diagram of the primary and secondary systems is shown in Figure 2-1.

2.2 Reactor and Main Coolant System (MCS)

The reactor is rated to produce 600 MWt. The active portion of the reactor approximates a right circular cylinder approximately 7.5 feet in height and 6.3 feet in diameter. This nearly cylindrical core contains 76 fuel assemblies which are fueled with slightly enriched uranium dioxide pellets. The pellets are clad in tubes made of Zircaloy.

The reactor is controlled by 24 neutron absorbing control rods and dissolved boron in the moderator. Inherent stability and control is provided by the strong negative temperature coefficient of the moderator. During rapid reactivity insertions, the prompt negative Doppler coefficient also serves to quickly limit the transient. Borated demineralized water is circulated in the reactor coolant system at a flow rate and temperature consistent with achieving the reactor core thermal-hydraulic performance. The borated water also acts as a neutron moderator, a reflector, and as a neutron absorber for chemical shim control.

Water in the reactor and in the main coolant system is normally maintained at a system pressure of 2000 psig. The inlet water temperature to the core is 515°F and the average core outlet temperature is 561°F.

Main coolant system pressure is controlled by use of the pressurizer where water and steam are maintained in equilibrium by electrical heaters or water sprays. Steam can be formed (by the heaters) or condensed (by the pressurizer spray) to minimize pressure variations due to contraction and expansion of the reactor coolant.

A detailed schematic of the MCS is shown in Figure 2-2.

2.3 Overpressure Protection

Overpressure protection of the main coolant system is provided by safety and relief valves located at the top of the pressurizer. The valves are all located such that water cannot condense and form a seal in the inlet lines.

Two spring-loaded code safety valves and a solenoid-operated relief valve are provided to accommodate the pressure surges which exceed the pressure limiting capacity of the pressurizer and spray system. The solenoid-operated relief valve operates at a pressure of 2400 psig to minimize the need for employing the code safety valves. The safety and relief valves discharge through a rupture disc directly to the vapor container.

In addition, each main coolant system loop is equipped with a water relief valve, located on the 5-inch bypass pipe and set to discharge 90 gpm at 2735 psig. The valves are provided to prevent overpressure in isolated loops due to water expansion caused by operating a main coolant pump, or by injecting hot feedwater into the steam generator when the main coolant water is cold. The discharge piping from these valves is directed to the low pressure surge tank.

Under reduced temperature operation, additional overpressure protection is provided by the low pressure setpoint feature of the pressurizer solenoid relief valve. The low pressure setpoint is in the range of 485 psig. This setpoint provides low temperature overpressure protection (LTOP).

2.4 Shutdown Cooling

The shutdown cooling system for YNPS is shown in Figure 2-3. This system is designed for a maximum pressure of 425 psig and a maximum temperature of 370°F. The shutdown cooling system is provided to remove the heat generated by radioactive decay of fission products in the reactor core during extended shutdown periods. Following trip, the reactor is cooled and depressurized by discharging steam from the steam generators through the turbine steam bypass to the main condenser (to the atmosphere if the condenser is unavailable). The shutdown cooling system is placed in service after the main coolant temperature has been reduced to approximately 330°F and the pressure to less than 300 psig. The shutdown cooling system then reduces the main coolant temperature to 140°F or less and operates continuously to maintain this temperature as long as is required by maintenance or refueling operations.

The shutdown cooling system consists of a heat exchanger, circulating pump, piping, valves, and instruments arranged in a low pressure auxiliary loop parallel with the main coolant loops. The shutdown cooling pump takes suction from the hot leg of the main coolant piping - on the reactor side of the loop stop valves - and recirculates the main coolant water through the tube side of the shutdown cooler and back into the cold leg of the main coolant piping. The main coolant is contained in a closed system, and reactor decay heat load is transferred through the shutdown cooler to the component cooling system, which, in turn, is cooled by river water. This arrangement of providing the intermediate cooling medium of the component cooling system was selected in order to assure that any possible leakage of radioactive main coolant would not enter the river water.

Complete backup of the system is provided by the low pressure surge tank pump and heat exchanger, which are identical units connected in parallel.

The shutdown cooling system is designed to remove the maximum expected decay heat from a core which has operated for an equivalent of 10,000 full power hours. Three hours after shutdown, the expected decay heat is approximately 16,000,000 Btu per hour. The cooling system will also reduce the main coolant temperature at such a rate as to cool to 140°F in approximately 55 hours.

2.5 Engineered Safety Features

The engineered safety features for the plant include an emergency core cooling system consisting of a safety injection system, a recirculation system and the vapor containment system. These systems are provided to protect plant personnel and the public from the effects of an unlikely major loss of fluid from the main coolant system. The safety injection and recirculation systems function to supply borated water to the reactor vessel for cooling the core. The vapor containment system serves to minimize the release of radioactivity to the environment.

The safety injection system (SIS) provides makeup to the primary system for events resulting in a reduction in inventory beyond the capacity of the charging system. Inventory reduction can result from either a break in the primary system or excessive cooldown causing a primary system shrink. SI is automatically initiated on a low reactor coolant system pressure, 1700 psig, or a high vapor container pressure, 5 psig. Components of the Yankee SIS are listed in Figure 2-4. The system is shown in Figure 2-5. Basically, this system operates in two modes: injection and recirculation.

In the injection mode, pump suction is taken from the safety injection tank. A performance curve for the injection mode, as a function of system pressure, is given in Figure 2-6. When the safety injection tank level approaches eleven (11) feet, pump suction is realigned to the vapor container sump.

In the recirculation mode, the SI pumps operate by taking suction from the vapor container sump delivering water to all four loops. Water continues to feed the sump by spilling out the break in the primary system.

2.6 Charging and Volume Control System

The charging and volume control system is composed of pumps, vessels, piping and valves, both inside and outside the containment, arranged to accomplish the functions of water charging to the main coolant system, water removal from the main coolant system, boric acid addition and removal for reactivity control, pressurizer vessel cooling and decontamination, noncondensible gas removal, chemical addition, corrosion control of main coolant system, and water charging to auxiliary systems and equipment.

A chemical shutdown system is provided to supply sufficient boron to the main coolant system to ensure that the reactor is 5 percent shut down with the control rods fully inserted in the cold beginning-of-cycle condition (original design basis), supply sufficient boron to borate a cold isolated loop to full shutdown concentration, provide a supply of boron solution for makeup to the safety injection tank, supply sufficient boron to the charging and volume control system to maintain a boron shim when necessary during power operation, and remove low concentrations of boron from the main coolant system as required during operation.

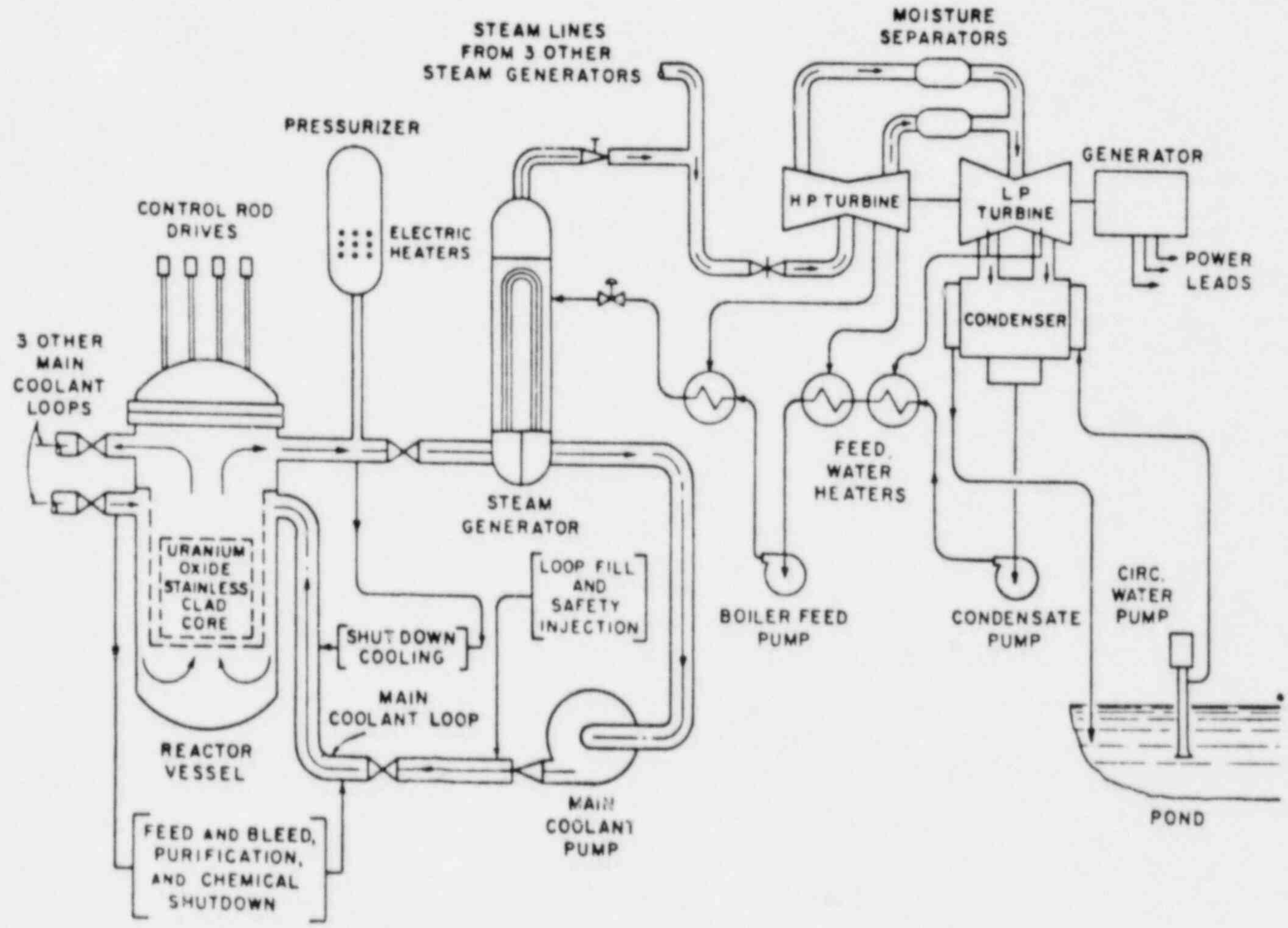

Figure 2-1

Simplified Flow Diagram
Primary and Se.ondary Systems

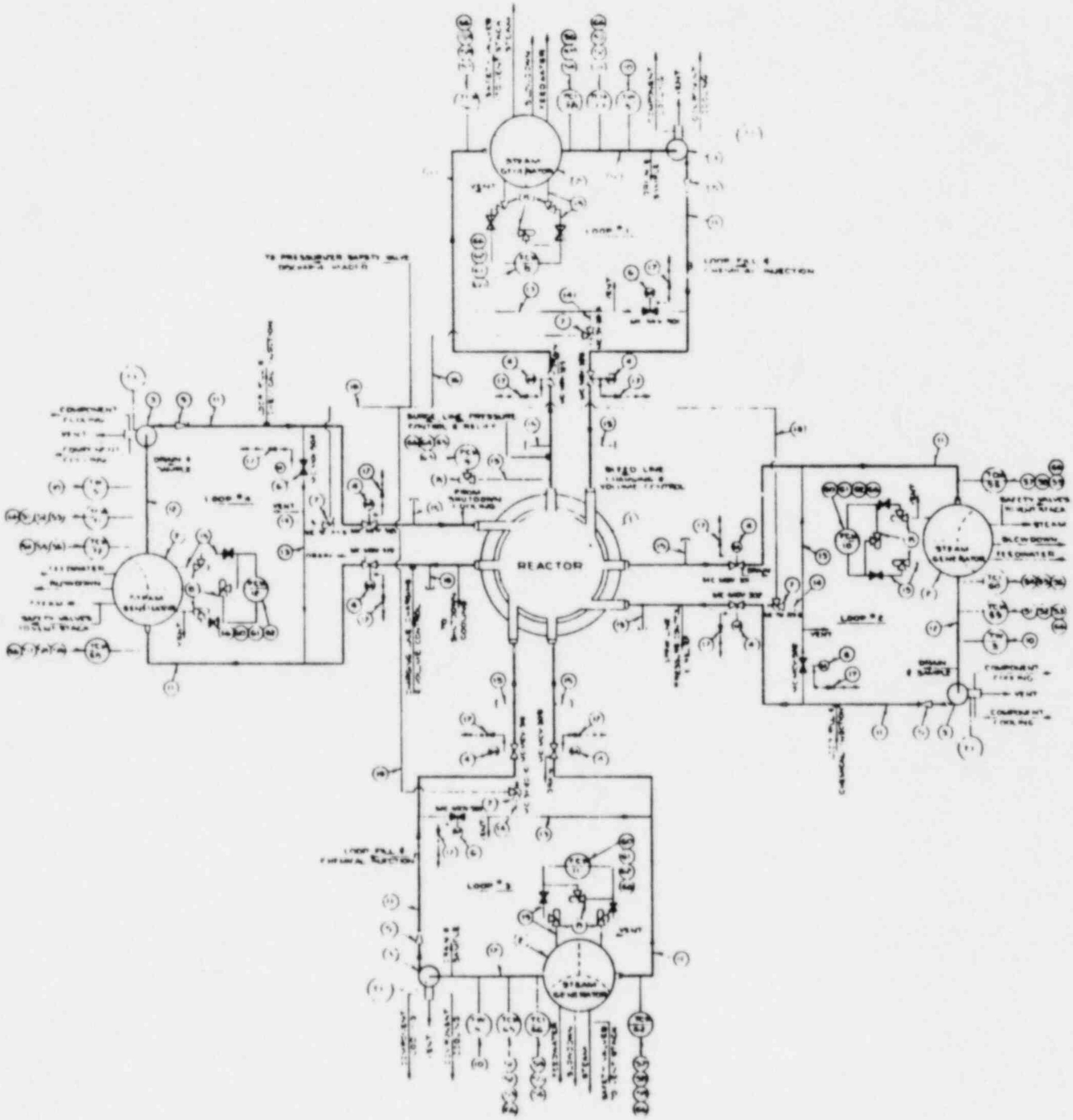

Figure 2-2

Main Coolant System

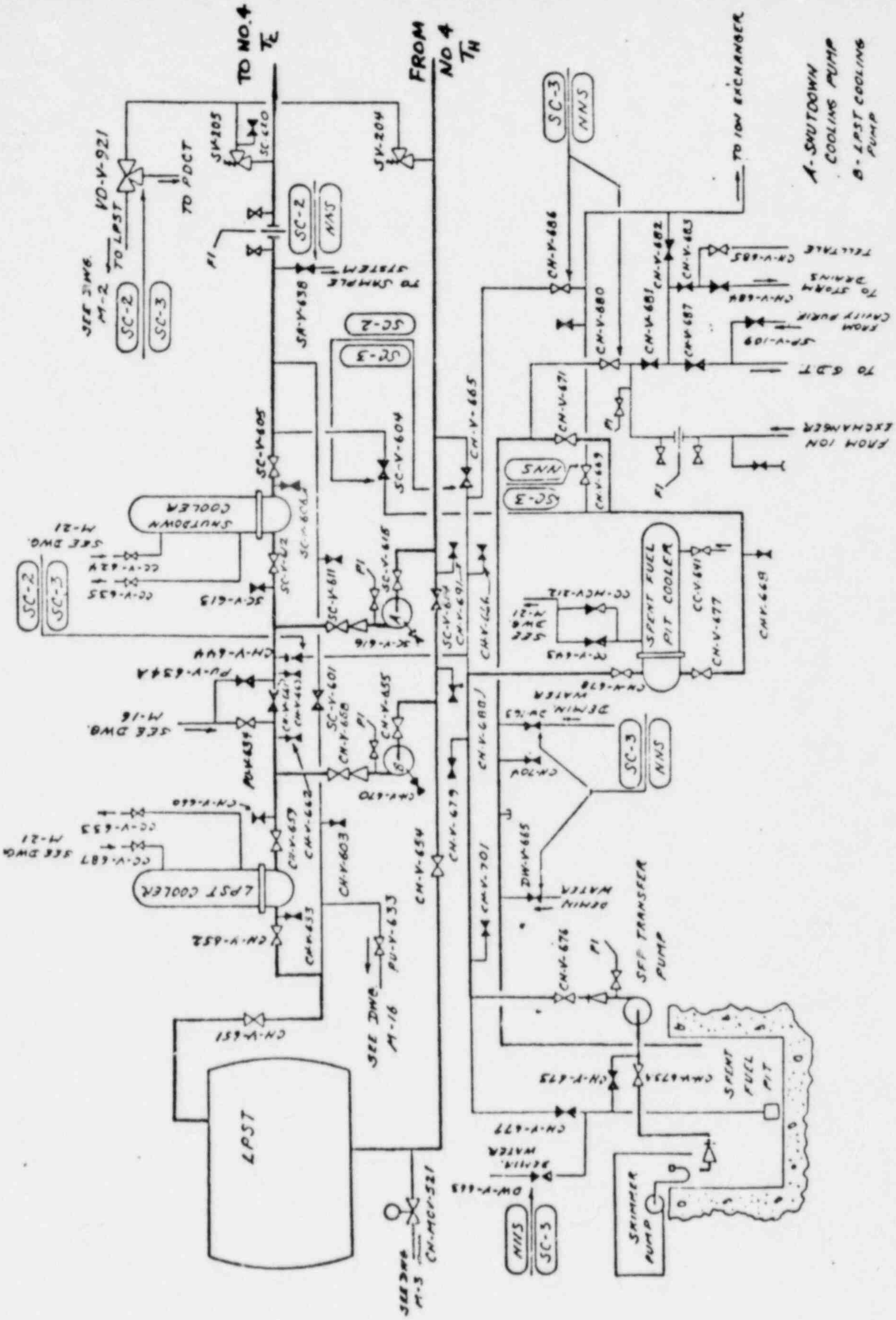

Figure 2-3

Shu:down Cooling

SI Accumulator

Quantity	1
Minimum usable volume	700 ft^3
Nitrogen cover pressure	473 $\pm$ 10 psig
Maximum flow rate (approximate)	5000 gpm
Minimum boron concentration	2200 ppm

Safety Injection Tank

Quantity	1
Nominal capacity	125,000 gallons
Minimum Technical Specification volume	117,000 gallons
Minimum boron concentration	2200 ppm

High Pressure Safety Injection (HPSI) Pumps

Quantity	3
Capacity (per pump)	50%
Shutoff pressure	870 psid
Runout flow	300 gpm

Low Pressure Safety Injection (LPSI) Pumps

Quantity	3
Capacity (per pump)	50%
Shutoff pressure	690 psid
Runout flow	1100 gpm

Figure 2-4

Yankee Emergency Core Cooling System Components

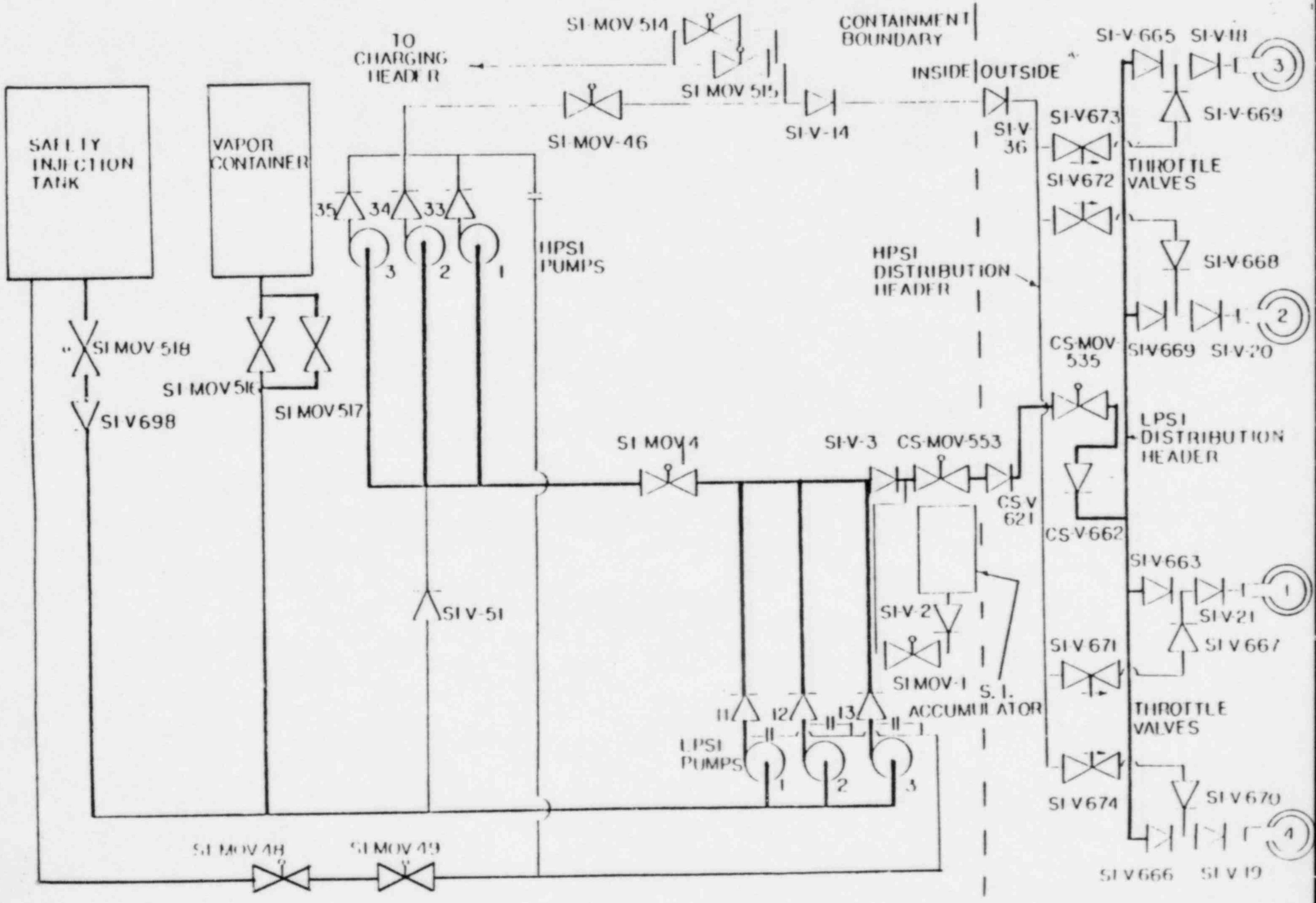

Figure 2-5

Yankee ECCS

3 LPSIs + 3 HPSIs + Accumulator

NO SPILLAGE

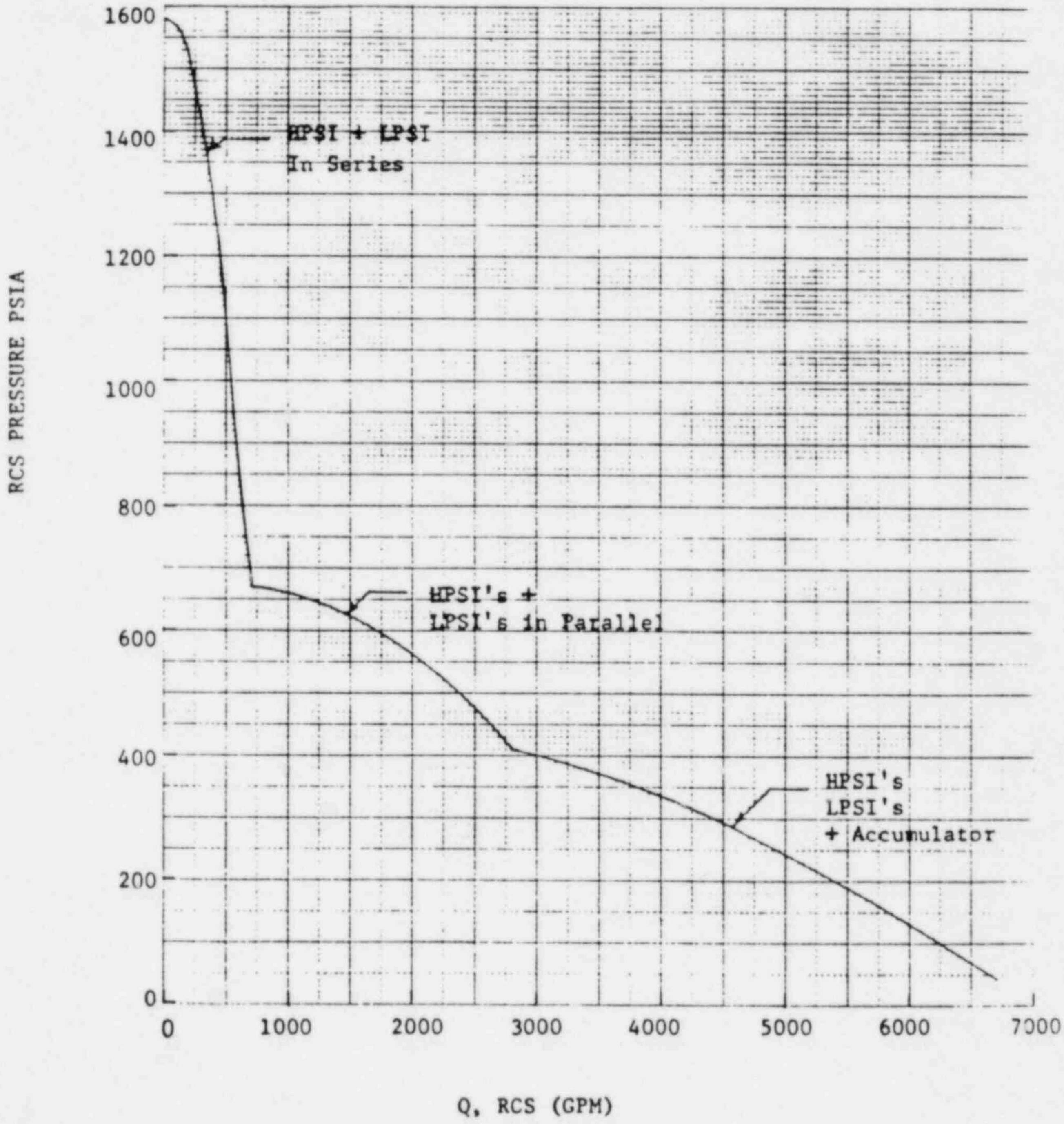

Figure 2-6

Yankee ECCS Performance

3.0 DESCRIPTION OF SAFETY/RELIEF VALVE ACTUATING TRANSIENTS

3.1 General

In order to determine the inlet fluid conditions (pressure, temperature, pressurization rate, etc.) imposed on the primary safety/relief valves for the YNPS, a detailed review of existing safety analyses performed in support of plant operation was conducted. The scope of this review covered:

1. Overpressurization events included in safety analysis reports (SARs) and core reload analysis.

2. Inadvertent actuation of the high pressure safety injection (HPSI) system.

3. Low temperature overpressurization (LTOP) events.

A general description of events in each of these categories is given below.

3.2 SAR/Reload Pressurization Events

The limiting overpressurization event documented in SAR/core reloads for the YNPS is the complete loss of load incident. A bounding analysis of this event was documented in Reference 3.

A loss of load event can be described as a rapid and large reduction in power demand on the reactor while operating at power. The large reduction in power demand (or steam flow) results in a corresponding decrease in the rate of heat removal from the main coolant system. Such an incident could lead to system overpressurization and subsequent core damage if suitable protection were not provided.

The most probable cause of a rapid loss of load is a turbine trip. For a turbine trip, the reactor would be tripped directly (unless below approximately 10 percent power) from a signal derived from the turbine stop valves. The steam bypass system would accommodate the excess steam generation. The steam bypass functions to limit considerably the increase in main coolant temperature and pressure for this transient.

The primary system is protected against overpressurization by:

1. A steam generator low level trip.

2. A pressurizer high level trip.

3. Pressurizer and steam generator safety valves.

4. Main coolant loop safety valves.

In addition, a high main coolant pressure trip set at approximately 2300 psia has been added which provides increased protection for loss of load events.

Loss of load events can also occur from loss of condenser vacuum or from inadvertent closure of the non-return valves in each main steam line. Under these conditions, the steam dump system would not be available.

Both the pressurizer power-operated relief valve and the steam bypass valve are provided to prevent the spring-loaded safety valves from opening, and are not intended to be a part of the system overpressurization protection. In the event the steam dump valves fail to open following a large loss of load, the steam generator safety valves may lift and the reactor may be tripped by the high pressurizer pressure signal. The steam generator shell side pressure and main coolant temperatures will increase rapidly. The pressurizer safety valves and steam generator safety valves are, however, sized to protect the main coolant system and steam generators against overpressure for all load losses without assuming the operation of the steam dump system, pressurizer spray, pressurizer power-operated relief valves, automatic rod control, nor direct reactor trip on turbine trip.

The steam generator safety valve capacity is sized to remove the maximum calculated steam flow (105 percent of maximum guaranteed steam flow) from the steam generator without exceeding 110 percent of the steam system design pressure. The pressurizer safety valve capacity is sized based on a complete loss of heat sink with the plant initially operating at the maximum calculated turbine load along with operation of the steam generator safety valves. The pressurizer safety valves are then able to maintain the main coolant system pressure within 110 percent of design pressure without direct or immediate reactor trip action.

In order to demonstrate that the main coolant system is adequately protected during a complete loss of load transient, the analysis reported does not take credit for the steam dump system, or the pressurizer power-operated relief valve, or the direct reactor trip on turbine trip. In such a case, when credit is not taken for the immediate trip initiated by the turbine trip or subsequent steam generator low level signal, the reactor is tripped by the high pressurizer level or pressure trip.

3.3 Inadvertent Actuation of HPSI

The extended high pressure injection transient is characterized in licensing terms as an "Increase in Reactor Coolant System Inventory Event" in which the high pressure safety injection pumps are inadvertently actuated to discharge into the MCS during normal power operation. The rate of increase in MCS inventory is dependent upon the head-flow curve for the high pressure safety injection pumps. For the YNPS, as seen in Figure 2-6, the HPSI pump's shutoff head (1600 psia) is below normal operating pressure. Therefore, this event is of no concern for the YNPS since mass additions to the MCS cannot occur above the HPSI pump shutoff head.

3.4 Low Temperature Pressurization Transients

During low temperature modes of plant operation, system pressure must be maintained below specific limits to preclude brittle fracture in the reactor coolant pressure boundary. Inadvertent inputs of mass and/or energy into the MCS can result in undesirable pressure increases. Particularly rapid and severe pressure transients can occur when the pressurizer is operated in a water-solid condition (without a volume of steam or gas).

Overpressurization under low temperature conditions can be avoided by:

1. provision of sufficient relieving capacity,

2. preclusions of the initiating events by administrative control and/or operating procedures,

3. a combination of 1 and 2.

Low temperature overpressure protection is provided by the low setpoint on the PORV. Specific events having the potential of causing reactor vessel overpressurization at low temperature include:

1. Inadvertent ECCS operation,

2. Charging without bleed flow,

3. Pressurizer heater operation without bleed flow,

4. Loss of shutdown cooling heat removal capacity,

5. Reactor coolant flow initiating transients.

These events involve either a mass or energy addition to the main coolant system with the potential for threatening the capacity of the PORV. The most limiting LTOP transient for YNPS, as demonstrated in References 4 and 5, results from a single coolant pump start during filled pressurizer conditions with a steam generator to MCS temperature difference of 100^{o}F.

4.0 SAFETY/RELIEF VALVE INLET CONDITIONS

This section summarizes safety/relief valve inlet conditions for each category of overpressurization event discussed in Section 3.0. Details of the analysis results and assumptions used can be obtained from References 3 - 5. Reference 6 contains an evaluation of the latest core reload. It is noted that the analysis of transients reported for SAR/reload cores does not take credit for the mitigation of the event by pressurizer spray or the operation of the PORV, but only for the code safety valves. Thus, the calculated peak pressures are conservatively high. The pressure ramp rate presented for the safety valves is estimated at the time that the pressure is approaching the safety valve setpoint, with the PORV assumed inoperable.

The valve inlet conditions for each event category are presented in Table 4.1. The highest peak pressure and greatest ramp are 2490 psig and 34.2 psi/sec, respectively, for the loss of load event. The lowest setting for opening pressure for the two code safety valves is 2485 psig. The safety/relief valve inlet fluid was steam for all events except LTOP transients.

Table 4.1

Calculated Pressurizer Safety/Relief Valve
Inlet Fluid Conditions During Pressurization Transients

Pressurization Transient	Peak Pressurizer Pressure (psig)	Pressure Ramp Rate (psi/sec)	Fluid Condition
Loss of load	2,490.0	34.2	Steam
MCP start, LTOP	523.3	13.3	Liquid
Extended HPSI	N/A	N/A	N/A

5.0 SUMMARY

Overpressurization transients for the YNPS have been reviewed for the purpose of determining limiting safety/relief inlet fluid conditions for verifying the applicability of the EPRI test program. Based on this review, it was determined that the peak pressure that would be experienced by the safety/relief valves at the YNPS is 2490 psig at a corresponding pressure ramp of 34.2 psi/sec. Steam conditions would be present for all postulated events except for low temperature overpressure cases.

6.0 REFERENCES

1. NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations", Nuclear Regulatory Commission, July 1979.

2. NUREG-0737, "Clarification of TMI Action Plan Requirements", Nuclear Regulatory Commission, November 1980.

3. YAEC Letter to USAEC, Proposed Change No. 115, dated March 29, 1974.

4. YAEC Letter to USNRC, WYR 76-125, dated December 1, 1976.

5. YAEC-1124, "An Analytical Model Used in PWR Overpressurization Analysis", dated February 1977.

6. YAEC-1240, "Yankee Nuclear Power Station Core XV Performance Analysis", March 1981.