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

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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, EPRI 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, Combustion Engineering Incorporated (CE) under contract to the Electric Power Research Institute (EPRI), conducted generic studies to provide supporting data for the fluid test conditions being used in the EPRI Valve Test Program for CE designed plants. The CE generic study was documented in Reference 3. In addition to the CE generic study, the Yankee Atomic Electric Company (YAEC) has initiated a plant specific study for the Maine Yankee Atomic Power Station (MYAPS). The particular study, which is the subject of this report, is intended to provide supporting information (beyond Reference 3) to demonstrate that the fluid conditions being used in the EPRI Valve Test Program are applicable to the MYAPS.

The objective of this study is to develop information to assist in the justification of the applicability to the MYAPS of the inlet fluid conditions selected for the testing of pressurizer safety and relief valves in the EPRI 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 MYAPS 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 reports (FSAR), and the most recent fuel reload analysis. In addition, since the MYAPS utilizes PORVs for low temperature overpressure protection (LTOP), 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 MYAPS DESIGN

2.1 General

The Nuclear Steam Supply System (NSSS) of the MYAPS consists of a pressurized water reactor designed by Combustion Engineering Inc., with three parallel heat transfer loops, each containing one steam generator and one reactor coolant pump. A pressurizer is connected to one of the reactor vessel outlet pipes to maintain and control reactor coolant system pressure. In addition, each loop contains two remotely-operated stop valves and bypass piping to permit isolation of one loop from the reactor. A quench tank is provided to receive, condense, and cool discharges from the pressurizer relief valves and safety valves. All components are located inside the containment building and are arranged as shown in Figure 2-1.

2.2 Reactor Coolant System (RCS)

The reactor is rated to produce 2630 MWt. The fuel assemblies are arranged to approximate a right circular cylinder with an equivalent diameter of 136 inches and an active length of 136.7 inches. This nearly cylindrical core contains 217 fuel assemblies fueled with slightly enriched uranium dioxide pellets. The pellets are clad in tubes made of Zircaloy.

The reactor is controlled by 85 neutron absorbing control element assemblies (CEAs) and dissolved boron in the moderator. Inherent stability and control is provided by the negative temperature coefficient of the moderator at full power. 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 required 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 reactor coolant system is normally maintained at a system pressure of 2250 psia. The inlet water temperature to the core is 550°F and the average core outlet temperature is 600°F.

Reactor 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.

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.

Three spring-loaded code safety valves and two power-operated relief valves (PORVs) are provided to accommodate the pressure surges which exceed the pressure limiting capacity of the pressurizer and spray system. The PORVs operate at a pressure of 2400 psia to minimize the need for employing the code safety valves. The steam discharged by the safety and relief valves is piped to the quench tank inside containment.

Under reduced temperature operation, additional overpressure protection is provided by the low pressure setpoint feature of the PORVs. 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 or Residual Heat Removal (RHR) system for MYAPS is designed for a maximum pressure of 600 psig and a maximum temperature of 450°F. The RHR system is provided to remove the heat generated by radioactive decay of fission products in the reactor core during extended shutdown periods and to reduce plant temperature during cooldown operations below 400°F. The RHR system is capable of reducing plant temperature from 400°F to a refueling temperature of 125°F in four hours. Typically, 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 RHR system is placed in service after the reactor coolant temperature has been reduced to approximately 400°F and the pressure to less than 350 psig. The RHR system then reduces the reactor coolant temperature to 125°F or less and operates continuously to maintain this temperature as long as is required by maintenance or refueling operations.

The RHR system uses components of the low pressure safety injection (LPSI) system and the containment spray system for reactor coolant circulation and temperature control. These components include:

1. LPSI pumps.

2. RHR heat exchangers.

3. Associated piping and valves.

The LPSI pumps take suction from the Reactor Coolant System (RCS) Loop 2 hot leg through the RHR isolation valves. The high capacity, low head pumps discharge to the RHR heat exchangers. Component cooling water on the shell side of the heat exchangers removes heat from the reactor coolant RHR flow. The temperature of the RHR return flow is controlled by varying the proportion of RHR heat exchanger tube side flow and bypass flow. Heat exchanger bypass flow is adjusted to maintain constant system flow. The bypass flow and the heat exchanger effluent combine and enter the RCS through the cold leg safety injection nozzles from the LPSI header. The RHR system is designed to remove 118.4×10^6 Btu/hr.

2.5 Engineered Safety Features

The engineered safeguards system includes the safety injection systems and the containment spray system. These systems are provided to protect plant personnel and the public from the effects of a loss-of-coolant incident. The engineered safeguards function to localize, control, mitigate and terminate such incidents and to hold environmental exposure levels within the acceptable levels.

The systems are principally designed to cool the core, to reduce the pressure within the containment, and to remove fission products from the containment atmosphere.

Both a high pressure and a low pressure safety injection system are provided to inject borated water into the reactor vessel immediately after a loss-of-coolant incident. They are designed to prevent fuel and cladding damage that could interfere with adequate emergency core cooling, and to minimize the extent of the cladding-water reaction. The systems will perform satisfactorily even in the unlikely event of loss of all off-site electrical power and with only one of the two on-site diesel generators to supply emergency power.

In the event of a loss-of-coolant incident, the high pressure safety injection system (HPSI) uses two of the three charging pumps of the charging and volume control system as high pressure safety injection pumps to inject borated water into the reactor coolant system.

During normal plant operation, two of three charging pumps provide all the reactor coolant makeup requirements. Normally, one charging pump is adequate for makeup and the second pump is used as a standby. The third pump is a spare that is used to replace either of the other pumps when maintenance is required.

Upon a safety injection actuation signal (SIAS), the charging pumps will function as HPSI pumps. Their suction will be automatically transferred from the volume control tank to the refueling water storage tank (RWST) and their output will be diverted from the charging flow path to the HPSI header and loop injection lines. Under this mode of operation, the HPSI pump's total capacity is adequate to maintain a reactor coolant system overpressure for a line break equivalent to 1-1/2 inch pipe. The HPSI system is also designed for post-accident core cooling and for the injection of large quantities of borated water for adding shutdown capability during the rapid cooldown of the reacto coolant system which might result from the rupture of a steam line.

The low pressure safety injection (LPSI) system is designed to inject borated water into the reactor vessel to flood and cool the core upon the depressurization of the reactor coolant system following a major loss-of-coolant accident. This system includes three safety injection tanks and two LPSI pumps.

The safety injection tanks are approximately 40 percent filled with borated water and are provided with a 205 psig nitrogen cover gas. The gas pressure provides the driving head required to transfer the tank liquid into the reactor vessel when the reactor coolant system pressure falls below that of the cover gas. This borated water flows through check valves located in the line connecting each tank to the reactor coolant system.

The two LPSI pumps are also used to inject borated water from the refueling water storage tank into the reactor vessel upon depressurization of the reactor coolant system. A spare pump is provided which can be used either as an LPSI pump or alternately as a containment spray pump. This pump can be lined up from either one of the two suction headers from the refueling water storage tank. The LPSI pumps can also be used during the post-accident core cooling operation.

All safety injection equipment starts on a safety injection actuation signal. Upon a containment pressure of 5 psig or upon a low pressurizer pressure of 1585 psig, the instrumentation circuitry will produce the SIAS required to automatically start the safety injection pumps and position the valves for borated water injection to the reactor coolant system.

Both high pressure and low pressure headers of the safety injection systems feed into the three reactor coolant loops. The feed lines of the high pressure header join the transport lines from the low pressure safety injection header penetrating the containment, to connect into the safety injection tank outlet to the cold leg of each loop.

As previously stated, the safety injection systems also function during post-accident core cooling. This function is either initiated by operator action or by a recirculation actuation signal derived from RWST low level switches. Upon such a signal, the containment spray pump suctions are transferred from the RWST to the containment sump. This signal also stops the LPSI pumps and transfers the HPSI pump suctions from the RWST to the outlets of the residual heat exchangers. The spray pumps must be operating to satisfy the NPSH requirements of the HPSI pumps during the recirculation mode of operation. At the discretion of the operator, the HPSI pumps may be stopped and the LPSI system used for post-accident core cooling.

The containment spray system is designed to spray chemically treated water into the containment following a loss-of-coolant incident. This will depressurize the reactor containment by continuously reducing the pressure to about 10 psig in approximately 24 hours after a double-ended rupture of the largest pipe in the reactor coolant system. In addition, sodium hydroxide added to the containment spray solution removes radioactive iodine from the containment atmosphere.

2.6 Charging and Volume Control System

The Chemical and Volume Control System (CVCS) is designed to perform the following functions:

1. Control the reactor coolant system volume by compensating for coolant contraction or expansion resulting from changes in reactor coolant temperature;

2. Maintain the reactor coolant activity level within prescribed limits by removing corrosion and fission products;

3. Inject chemicals into the reactor coolant system to control coolant chemistry and minimize corrosion;

4. Control the reactor coolant boric acid concentration, thereby providing the reactivity control required for startup, operation and shutdown;

5. Provide a means for pressure testing the reactor coolant system;

6. Supply high pressure safety injection flow into the reactor coolant system upon a safety injection signal.

Letdown flow from a cold leg of the reactor coolant system passes through the tube side of the regenerative heat exchanger for an initial temperature reduction. The cooled fluid is then reduced in pressure by a letdown control valve to the operating pressure of the letdown heat exchanger. The final reduction to operating temperature and pressure of the purification subsystem is made by the letdown heat exchanger and a letdown backpressure valve. The flow then passes through a prefilter, demineralizer and a postfilter, and is then sprayed into the cover-gas of the volume control tank. A small fraction of the letdown flow bypasses the demineralizer, part of which is directed through the boronometer, which measures the boron concentration of the reactor coolant, and part through the letdown gross activity monitor which measures the coolant radioactivity level. The charging pumps take suction from the volume control tank and pump the coolant into the reactor coolant system at the desired rate. One letdown control valve and charging pump are normally in operation to maintain a balance between letdown and charging flow. The charging flow passes through the shell side of the regenerative heat exchanger for recovery of heat from the letdown flow before being returned to the reactor coolant system.

A makeup system provides for changes in reactor coolant boric acid concentration and for reactor coolant chemistry control. Concentrated boric acid solution is prepared in a mixing tank and stored in both the mix tank and the boric acid storage tank. Any of three boric acid pumps may be used to transfer the concentrated boric acid to a blending tee where it is mixed with demineralized water at a predetermined ratio. The blended boric acid solution is then introduced into the volume control tank. A chemical addition tank is used to flush chemical additives into the volume control tank.

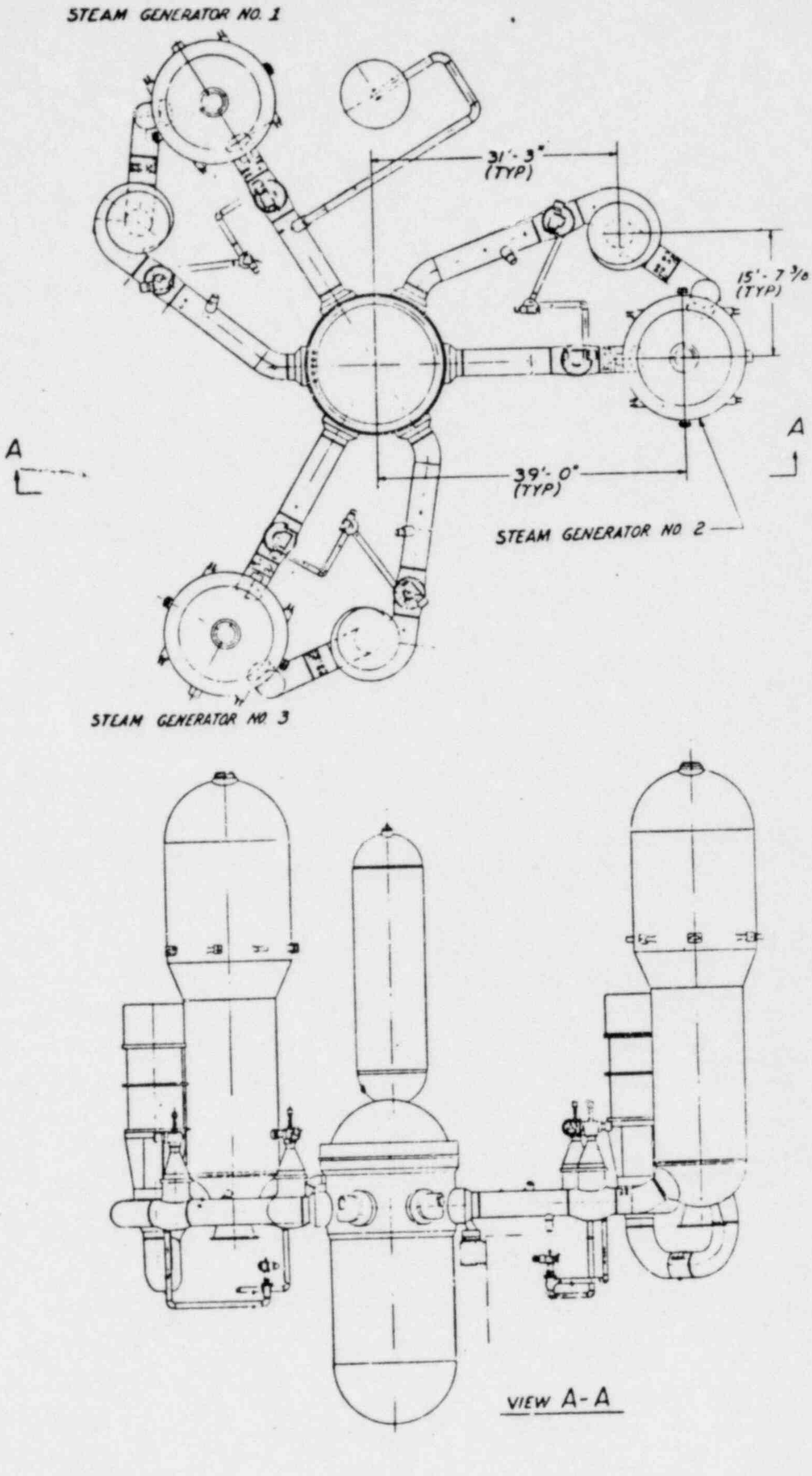


Figure 2-1

Reactor Coolant System
Arrangement

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 MYAPS, 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 (FSARs) and core reload analysis.

2. Inadvertent actuation of the 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 FSAR/Reload Pressurization Events

The limiting overpressurization event documented in FSAR/core reloads for the MYAPS is the complete loss of load incident. A bounding analysis of this event was documented in Reference 4.

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 reactor 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 15 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 primary 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 pressure trip.

3. Pressurizer and steam generator safety valves.

Loss of load events can also occur from loss of condenser vacuum or from inadvertent closure of the excess flow check valves (EFCVs) in each main steam line. Under these conditions, the steam dump system would not be available.

Both the pressurizer power-operated relief valves and the steam dump and bypass system valves are provided to prevent the spring-loaded safety valves from opening. In the event the steam dump valves fail to open following a large loss of load, the steam generator shell side pressure and main coolant temperatures will increase rapidly. The steam generator safety valves lift and the reactor may be tripped by the high pressurizer pressure signal. The pressurizer safety valves and steam generator safety valves are, however, sized to protect the reactor 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, nor direct reactor trip on turbine trip.

In order to demonstrate that the reactor coolant system is adequately protected during a complete loss of load transient, the analysis reported does not take credit for the steam dump system, pressurizer spray, the pressurizer power-operated relief valves, 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 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 coolant system during normal power operation. The rate of increase in reactor coolant system (RCS) inventory is dependent upon the head-flow curve for the high pressure safety injection pumps. For the MYAPS, the HPSI pump's shutoff head, as shown in Figure 3-1, is above normal operating pressure. Therefore, the potential exists for mass additions to the RCS from inadvertent actuation of the HPSI system. The results of an analysis of this event for the MYAPS is presented in Section 4.0 of this report.

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 RCS 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 PORVs. 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,

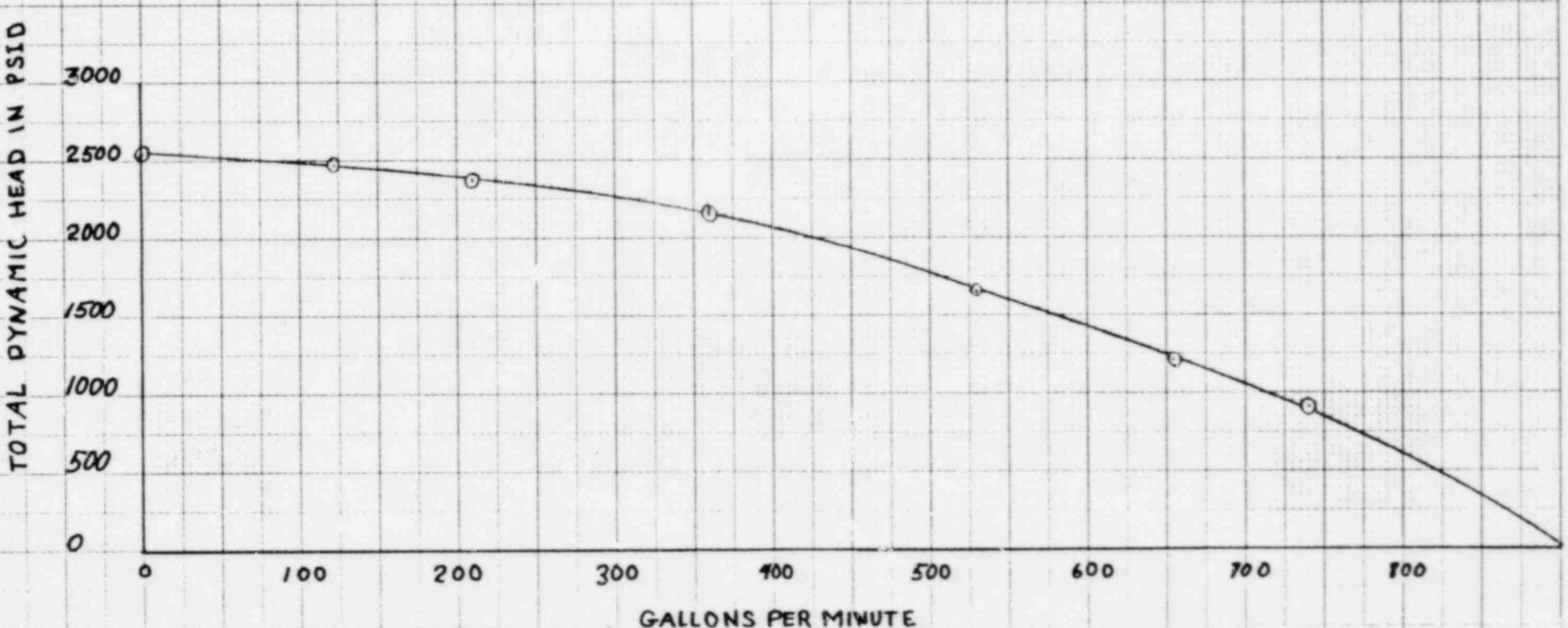
4. Loss of shutdown cooling heat removal capacity,

5. Reactor coolant flow initiating transients.

These events involve either a mass or energy addition to the reactor coolant system with the potential for threatening the capacity of the PORVs. The most limiting LTOP transient for the MYAPS, as demonstrated in References 5 and 6, results from a single coolant pump start during filled pressurizer conditions with a steam generator to RCS temperature difference of $100^{\circ}F$.

Figure 3-1

MAINE YANKEE ATOMIC POWER
CENTRIFUGAL CHARGING PUMP
CHARACTERISTIC CURVE SHEET



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 - 6. Reference 7 contains an evaluation of the latest core reload. Analysis details for the extended HPSI event are included below. It is noted that the analysis of transients reported for FSAR/reload cores does not take credit for the mitigation of the event by pressurizer spray or the operation of the PORVs, 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 2574 psig and 63.1 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.

To evaluate valve inlet fluid conditions for the extended HPSI event an analysis was performed using the RETRAN-02 computer code (Reference 8). In this analysis, both HPSI pumps were inadvertently initiated while operating at full power conditions. HPSI flow slowly increases the mass and inventory of the reactor coolant system, raising levels in the pressurizer. As the steam space in the pressurizer is compressed, the RCS pressure increases, eventually reaching the high pressure trip and PORV setpoint. To maximize the peak pressure and challenges to the code safety valves, the analysis assumed that the PORVs do not open. Following the reactor trip, HPSI continues to increase the total mass in the primary system. The pressurizer level continues to increase at an average rate of approximately 1 ft/min (Figure 4-1). In approximately 5 minutes following the event, the primary code safety valves are challenged and cycle as the pressurizer continues to fill (Figure 4-2). If no operator action is assumed and the event continues, the pressurizer would be filled in approximately 27 minutes. At this point, $568^{0}F$ liquid would be discharged through the pressurizer safety valves. The peak pressure within the pressurizer is 2496 psig prior to the filled condition. The peak pressure ramp rate is 6.5 psi/sec for the steam discharge conditions.

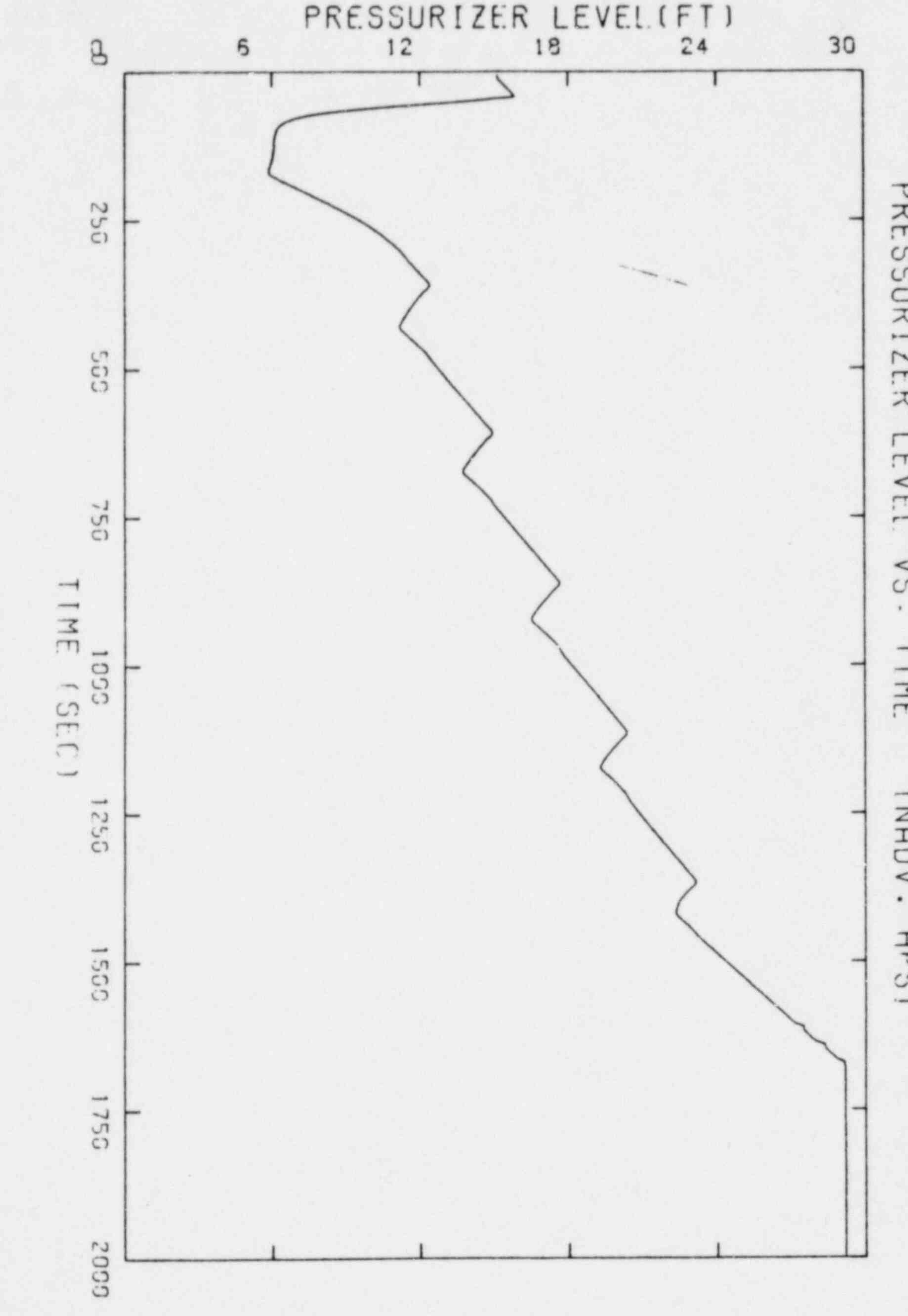
Table 4.1

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	2,574	63.1	Steam
RCP start, LTOP	549	12.5	Liquid
Extended HPSI	2,496	6.5	Steam*

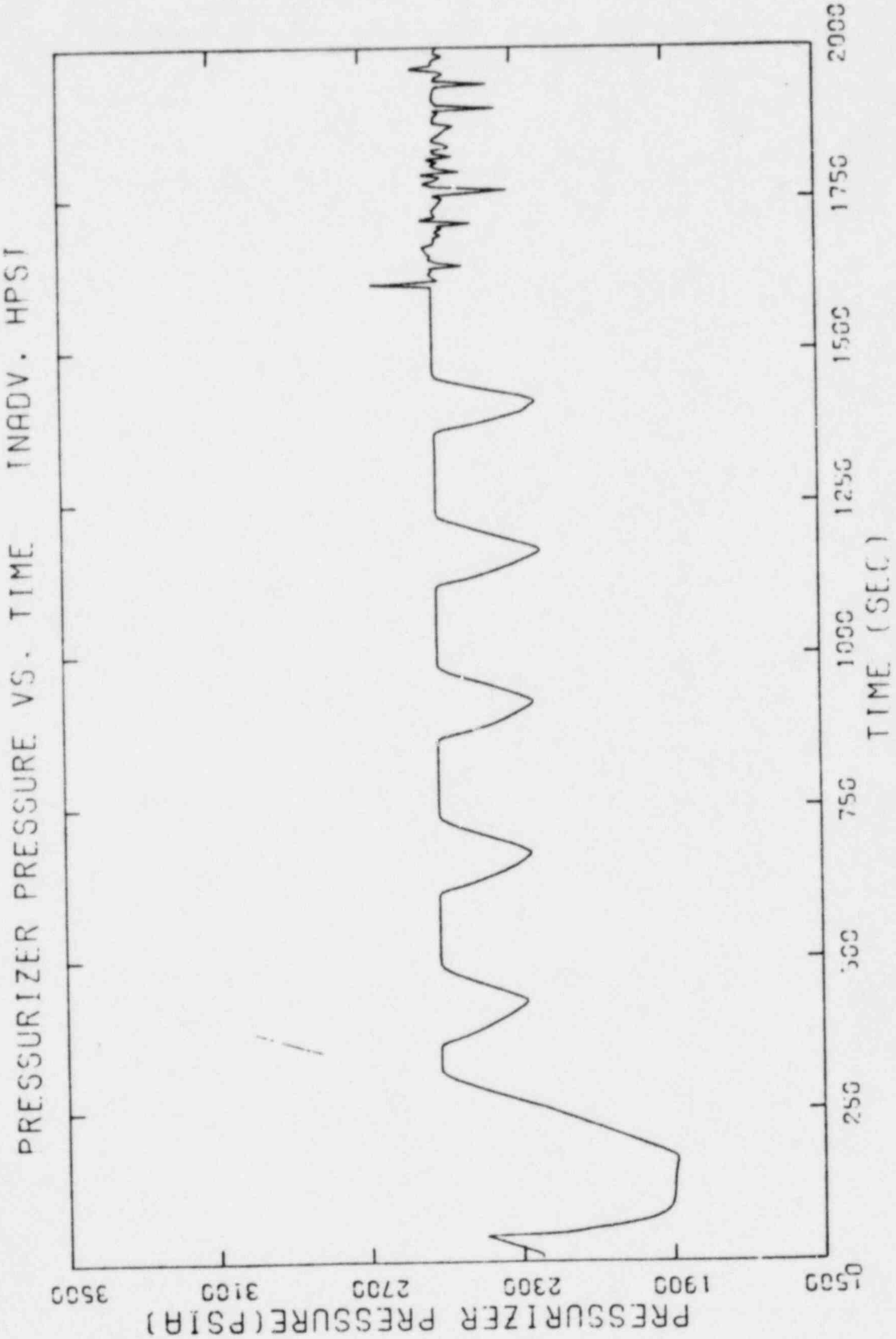
*Steam conditions persist for at least 27 minutes. Thus, there is ample time for the operator to take corrective action to prevent liquid discharge.

Figure 4-1



PRESSURIZER LEVEL VS. TIME INADV. HPSI
PRESSURIZER LEVEL (FT)
0
6
12
18
24
30
TIME (SEC)
250
500
750
1000
1250
1500
1750
2000

Figure 4-2



PRESSURIZER PRESSURE VS. TIME INADV. HPSI
PRESSURIZER PRESSURE(PSIA)
3500
3100
2700
2300
1900
1500
0
250
500
750
1000
1250
1500
1750
2000
TIME (SEC)

5.0 SUMMARY

Overpressurization transients for the MYAPS 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 MYAPS is 2574 psig at a corresponding pressure ramp of 63.1 psi/sec. Steam conditions would be present for all postulated events except for low temperature overpressure cases. For extended HPSI events, based on conservative application of HPSI pump head/flow performance and setpoint tolerances, the two HPSI pumps are marginally capable of lifting the safety valves. The flow for the extended HPSI condition is less than 200 gpm per pump at the PORV setpoint and less than 100 gpm per pump at the safety valve setpoint.

A plant specific analysis of the extended event is performed by utilizing a RETRAN model. The results of that analysis show the PORV opening on steam. Transition to water will not occur since it is assumed that operator action will be taken within 20 minutes. A worst case conservatively analyzed event disregarding the PORVs would result in a transition to liquid discharge out through the safety valves after a period of 27 minutes. This is more than ample time for the operator to take action to trip the HPSI pumps and normalize pressurizer conditions.

Additionally, even without operator action, the PORVs would successfully pass the water which falls within the as-tested condition in the EPRI Test Program. One PORV is more than sufficient to avoid a challenge to the safety valves on any extended HPSI event.

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. EPRI NP-2318-LD, "Valve Inlet Fluid Conditions for Pressurizer Safety and Relief Valves in Combustion Engineering-Designed Plants", dated April 1982.

4. YAEC-1132, "Justification for 2630 MWt Operation of the Maine Yankee Atomic Power Station", dated July 1977.

   YAEC-1148, "Justification for Operation of the Maine Yankee Atomic Power Station with a Positive Moderator Temperature Coefficient", dated April 1978.

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

6. MYAPCo letter to USNRC, WMY 76-135, dated December 2, 1976.

7. YAEC-1259, "Maine Yankee Cycle 6 Core Performance Analysis".

8. EPRI NP-1850 CCM, dated May 1981.