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TECHNICAL REPORT ON OPERATING EXPERIENCE WITH BWR PRESSURE RELIEF SYSTEMS

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Division of Operating Reactors
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington D. C. 20555

TECHNICAL REPORT ON OPERATING EXPERIENCE WITH BOILING WATER REACTOR PRESSURE RELIEF SYSTEMS

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ABSTRACT

The staff has conducted a review of operating experience with Boiling Water Reactor (BWR) pressure relief systems because of the number of unanticipated events with the power actuated relief valves and the safety/relief valves utilized in these systems. This experience includes (1) valves that failed to open properly, (2) valves that opened properly and failed to reseat, and (3) valves that inadvertently opened and failed to reseat.

This report describes the pressure relief systems utilized in operating BWR facilities, the operating experience involving failures of pressure relief system valves, the safety considerations associated with such failures, and possible corrective measures to reduce the likelihood of future failures.

1.0 DISCUSSION OF BWR PRESSURE RELIEF SYSTEMS

A typical Boiling Water Reactor (BWR) manufactured by the General Electric Company is schematically represented in Figure 1. The heat generated by the nuclear fuel is transferred to the coolant water which flows adjacent to the fuel elements. The coolant is heated to boiling and the steam which results is routed via steam lines to the turbine generator where it is utilized to generate electricity.

BWRs may experience pressure transients during operation as a result of a mismatch between the reactor power level and the electrical load demand. When the electrical load demand is less than the power generated by the reactor (e.g., due to a sudden loss of electrical load), an increase in the pressure of the reactor coolant generally results. An increase in the reactor coolant pressure will reduce the voids (vapor bubbles resulting from boiling) in the core cooling water, thereby causing an increase in reactor power.* Therefore, pressure transients must be limited both to prevent damage or rupture of the reactor coolant pressure boundary (RCPB) due to excessive pressure and to prevent possible core damage due to excessive power.

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*This occurs because BWR power is strongly affected by the presence of voids in the core cooling water; a reduction in voids results in more effective neutron moderation in the cooling water and an increased fission rate in the core.

1.1 System Design Considerations

The BWR pressure relief system is designed to limit reactor pressure during normal operational transients and to prevent overpressurization of the RCPB under the most severe abnormal operational transient (e.g., closure of the main steamline isolation valves or fast closure of the turbine stop valves at full power). These design functions are accomplished through the use of a plant-unique combination of safety valves (SVs), power actuated relief valves (PARVs), and dual function safety/relief valves (SRVs). As shown in Figure 2, these valves are installed in the horizontal section of the main steamlines inside the containment drywell upstream of the first set of main steam isolation valves.* The design features of each of these three types of valves are described in Section 1.2 of this report. The combination of valves utilized in the pressure relief system of each operating BWR-2, BWR-3, and BWR-4** facility is presented in Table 1.

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*The SVs utilized at Nine Mile Point Unit No. 1 and at Oyster Creek are installed on the reactor vessel head.

**BWR-1 through BWR-6 are the product line designations for the Boiling Water Reactor Nuclear Steam Supply Systems designed and supplied by the General Electric Company (GE). This report addresses operating experience with the pressure relief system valves utilized in BWR-2, BWR-3, and BWR-4 facilities. There are a few operating BWR facilities which pre-date the BWR-2 product line; however, these facilities are not discussed herein because of their unique designs. Facilities of the BWR-5 and BWR-6 design have not yet been licensed for operation.

As shown in Table 1, the typical pressure relief system utilized in older operating BWR facilities consists of (1) a large number of spring-loaded SVs which protect the RCPB from overpressurization during the most severe abnormal operational transient, and (2) a smaller number of PARVs which supplement the SVs by relieving pressure surges associated with normal operational transients that might otherwise cause the higher setpoint SVs to actuate.*

The relief capacity of the SVs must be sufficient to limit the reactor coolant pressure during the most severe abnormal operational transient to less 1375 psig, which is 110% of the nominal 1250 psig RCPB design pressure, with sufficient margin to account for uncertainties in the design and operation of the facility and with certain conservative assumptions, including allowance of no credit for PARV operation.**

Most SVs discharge directly to the containment drywell whereas the PARVs have been designed to discharge beneath the water level in the containment pressure suppression chamber (torus).

*The PARVs have a lower pressure setpoint (nominally 1130 psig) than the spring-loaded SVs (nominally 1225 psig).

**For RCPB overpressure protection, credit is only allowed for self-actuated pressure relief system valves.

Consequently, the relief capacity of the PARVs must be sufficient to limit the reactor coolant pressure during normal operational transients so as to prevent SV discharges.

As shown in Table 1, the typical pressure relief system utilized in the newer operating BWR facilities consists of dual function SRVs, either exclusively or in conjunction with a small number of SVs.

The SRVs self-actuate on increased system pressure, have a nominal setpoint of 1100 psig, and are designed to discharge beneath the water level in the torus. As a result, for those facilities which utilize SRVs exclusively in their pressure relief systems, a single type of valve provides system overpressure protection both during normal operational transients and during the most severe abnormal transient. The required relief capacity of the SRVs is based on limiting the reactor coolant pressure during the most severe abnormal operational transient to less than 110% of the RCPB design pressure with sufficient margin to account for uncertainties in the design and operation of the facility and with certain conservative assumptions.

For those facilities which utilize a small number of SVs to supplement the SRVs, the total relief capacity of the SRVs and SVs must be sufficient to prevent overpressurization of the RCPB during the most severe abnormal transient. In addition, the relief capacity of the SRVs must be sufficient to preclude SV discharge to the drywell during normal operational transients.

In addition to the RCPB overpressure protection design functions of the BWR pressure relief system, a specified number of the PARVs or SRVs utilized in the pressure relief system of each BWR facility are also used in the Automatic Depressurization System (ADS), which is one of the Emergency Core Cooling Systems. In the event of certain postulated small break loss of coolant accidents (LOCAs),* the ADS is designed to reduce reactor coolant

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*For certain sizes of postulated BWR pipe ruptures, the rate of fluid loss would exceed the normal makeup capability of the system but would not be rapid enough to cause the reduction of the system pressure necessary to permit the low pressure emergency core cooling systems to function when required. In such a situation, operation of either the High Pressure Coolant Injection System (HPCI) or the ADS would be required to assure proper operation of the low pressure cooling systems. It should be noted that certain of the older operating BWR facilities do not have a qualified HPCI system and, therefore, rely on two redundant trains of ADS.

system pressure to permit the low pressure emergency core spray and/or low pressure coolant injection systems to function. The ADS performs this design function by automatically actuating certain pre-selected PARVs or SRVs following receipt of specific signals from the protection system.

1.2 Valve Design Descriptions

A typical spring-loaded SV is shown in Figure 3. In the event that the reactor coolant system pressure at the valve inlet reaches the valve setpoint, the pressure exerted on the valve disc is sufficient to overcome the spring pressure and the valve will actuate.

A typical PARV is shown in Figure 4. During normal plant operation, reactor coolant system steam pressure enters the main valve through the Chamber A and passes upward around the disc guide in Chamber B. Steam also enters Chamber C through a clearance space between the main valve disc and the disc guide. The main valve disc is held in the closed position by the steam pressure in Chamber C. PARV actuation is accomplished by energizing the solenoid in the pilot valve assembly. When the solenoid is energized, the pilot valve opens and allows the steam in Chamber C to be vented to atmosphere through port F. The resulting differential pressure on the main valve disc causes it to open, thereby permitting steam to escape from Chamber B to the valve outlet.

The electrical control system for a PARV is designed to provide power to energize the pilot valve solenoid in any one of the following ways:

(1) When the reactor system pressure reaches the valve setpoint, a pressure sensing switch senses the increasing pressure, actuates, and signals the pilot valve solenoid to energize;

(2) The reactor operator can manually open the valve, either for testing purposes or to relieve system pressure during a transient, by providing power to the pilot valve solenoid through a switch on one of the control room panels;

(3) If a PARV is utilized in the ADS and a postulated small break LOCA occurs, a protection system signal will energize the pilot valve solenoid provided that certain plant conditions exist.

The valve is designed to reseat when the solenoid is de-energized, the pilot valve closes, and the steam pressure in the chamber beneath the main valve disc is restored.

A typical SRV is shown in Figures 5, 6, and 7. Figures 5 and 6 depict the manner in which an SRV self-actuates; Figure 7 depicts the manner in which an SRV may be externally actuated.

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Self-actuation of an SRV is accomplished through a two-stage pilot valve. When reactor pressure reaches the valve setpoint pressure, the first stage pilot valve opens against the preset setpoint spring and allows steam pressure to enter into the chamber above the second stage piston. The second stage piston operates to open the second stage pilot valve which, in turn, opens a path for steam in the chamber above the main valve piston to be exhausted. The resulting differential pressure across the main valve piston causes the SRV to open.

External actuation of an SRV is accomplished through a diaphragm type air operator. When instrument air or nitrogen is admitted to the air operator chamber, the resultant differential pressure across the diaphragm causes the air operator stem to stroke the second stage pilot valve disc which results in the opening of the main valve.

The external control system for an SRV is designed to provide instrument air or nitrogen to the air operator by energizing a solenoid valve in the air or nitrogen supply line. Electrical power to energize the solenoid valve is provided either through a remote, manual actuation switch on one of the control room panels or through the ADS (for those SRVs which are part of that system.)

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2.0 OPERATING EXPERIENCE WITH BWR PRESSURE RELIEF SYSTEMS

Over 100 reactor years of BWR operating experience have been accumulated since the first commercial operation of a BWR. This experience includes a number of malfunctions of the valves utilized in the BWR pressure relief systems. These malfunctions can be characterized by the following subsets: (1) failures of a valve to open properly on demand, (2) spurious opening of a valve with subsequent failure of the valve to properly reseat, and (3) proper opening of a valve with subsequent failure of the valve to properly reseat.

The failure of a pressure relief system valve to open on demand results in a decrease in the total available pressure relieving capacity of the system. In addition, if the failed valve also serves as part of the ADS, a degradation of the capability of the ADS to perform its design function could result.

Spurious openings of pressure relief system valves or failures of valves to properly reseat after opening can result in unnecessary thermal transients on the reactor vessel and the vessel internals, unnecessary hydrodynamic loading of the containment system's pressure suppression chamber (torus) and its internal components, and potential increases in the release of radioactivity to the environs.

Operating experience related to (1) failure of pressure relief system valves to open and (2) inadvertent reactor coolant system blowdown events due to spurious valve openings or failures of valves to properly reseat after opening is discussed below. These discussions are based on reported failures of either the PARVs or the SRVs utilized in BWR-2, BWR-3, and BWR-4 facilities. Operating experience with the spring-loaded SVs has been essentially failure free.

The operating experience discussed in this report is based upon Licensee Event Reports submitted to the NRC during the period from 1969 through April 1978 and upon information obtained from General Electric Company (GE), valve manufacturers, and licensees of operating BWR facilities. While it is recognized that the operating experience information presented in this report may not be all inclusive, we believe that it is sufficiently complete and representative of operating experience to date for the purpose of this report.

2.1 Inadvertent Blowdown Events Due to Pressure Relief System Valve Malfunctions

As shown in Table 2, there have been a total of 53 inadvertent blowdown events due to pressure relief system valve malfunctions which have been reported from 1969 through April 1978. These events have varied in severity from a very short duration pressure transient to a rapid depressurization and cooldown of the primary coolant system from approximately 1100 psig (normal operating pressure is approximately 1000 psig) to a few hundred psig. During more than one-third of these events, the maximum allowable primary system cooldown rate (100°F per hour) permitted by the facility Technical Specifications was exceeded.

2.1.1 SRV Malfunctions

Forty-nine of the inadvertent blowdown events have resulted from a malfunction of an SRV. Of these events, (a) twenty-five events involved the spurious opening of a valve with subsequent failure to properly reseat, (b) twelve events involved the proper opening of a valve during a pressure transient with subsequent failure to properly reseat, (c) eleven events involved the proper opening of a valve (manually) during in-plant testing with subsequent failure to properly reseat, and (d) one event involved the inadvertent actuation of the SRVs used in the ADS during required surveillance testing.

The majority of the inadvertent blowdown events due to SRV malfunctions have been attributed to failures in the two-stage pilot valve section. Erosion of the first-stage pilot valve seat, which results in the leakage of steam into the second-stage actuating chamber and subsequent actuation of the main valve, has been the primary cause of the valve failures. (This same leakage path acts to prevent the proper reseating of the valve once it has opened). Accumulation of foreign material (dirt, rust) in the pilot valve section and mechanical failure of pilot valve section internal parts have been the causes of a lesser number of the valve failures.

2.1.2 PARV Malfunctions

Four of the inadvertent blowdown events have resulted from a malfunction of a PARV. Of these events, (a) three events involved the proper opening of a valve during a pressure transient with subsequent failure to properly reseat, and (b) one event involved the proper opening of a valve (manually) during in-plant testing with a subsequent failure to properly reseat. These failures have been attributed to mechanical damage to the main valve internals (e.g., scored valve rings), buildup of foreign material on the pilot valve seat, and steam erosion of the pilot valve seat.

2.2 Failures of Pressure Relief System Valves to Open Properly

As shown in Table 3, there have been a total of twenty-seven events involving the failure of pressure relief system valves to open properly on demand which have been reported from 1969 through April 1978.

Of these twenty-seven operational events, twenty events involved failures of pressure relief system valves to open during in-plant

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testing (usually during reactor startup) and the remaining seven events involved the failure of a valve to open properly during a pressure transient. None of these events resulted in RCPB overpressurization. The failure of more than one valve during a single pressure transient event occurred only on one occasion when four of eleven SRVs failed to open during a pressure transient following a reactor scram; however, in that instance, the pressure rise peaked at 1120 psig and no system damage resulted.

Only one of these events involved a major degradation of the ADS. In that event, three of the four SRVs utilized in the ADS were determined to be inoperable due to the common mode failure of the air actuator diaphragm due to overheating. The overheating had been the result of improper installation of insulation on the valve.* Although this event resulted in a major degradation of the ADS for the affected facility, the diverse HPCI system was operable and capable of providing emergency core cooling protection in the event of a postulated small break LOCA.

*As a result of this event, the NRC took positive action via an OI&E Bulletin (Bulletin 76-06, "Diaphragm Failures in Air Operated Auxiliary Actuators for Safety/Relief Valves," July 21, 1976) to ensure that other licensees inspected their SRVs to determine that a similar condition did not exist. As a result of this NRC action, a similar problem with SRV diaphragms was detected at another facility. This second occurrence involved the deterioration (delamination) of the diaphragms for 9 of the 11 SRVs. Although it is believed that the deterioration had not yet progressed to the point where the valves would have failed to function on demand, for the purposes of this report we have included this event as a "potential failure to open properly."

2.2.1 SRV Malfunction

Eleven of the twenty-seven events involved the failure of an SRV to open properly. Six of these failures occurred during a pressure transient event (five in the self-actuating mode of valve operation); the remaining five failures occurred during in-plant testing. The five failures which have occurred in the self-actuated mode of valve operation have primarily been attributed to leakage in the bellows setpoint assembly of the first stage pilot valve. The six failures which have occurred in the externally actuated mode of valve operation have been attributed to degradation of the air/nitrogen actuator diaphragm or failures in the air/nitrogen supply system (e.g., valves misaligned, solenoid valve failures).

2.2.2 PARV Malfunctions

Sixteen of the twenty-seven events involved the failure of a PARV to open properly. Fifteen of these failures occurred during in-plant valve testing; one failure occurred during a pressure transient event. The majority of these failures have been attributed to improper alignment or adjustment of the pilot valve mechanism which prevented the opening of the pilot valve. Loose parts or leaking seal rings in the main valve section have been the causes of a lesser number of these valve failures.

2.3 Potential Failures of Pressure Relief System Valves to Open Properly

As shown in Table 4, there have been seventeen instances where pressure relief system valves were determined to be in a condition in which they would have failed to open properly had they been required to do so.

2.3.1 SRV Malfunctions

Sixteen of the seventeen potential failure-to-open events involved SRVs. Twelve of these events involved leakage or suspected leakage* in the bellows setpoint assembly of the first stage pilot valve.

All but one of the bellows leakage failures ocurred during plant operation and were detected by the bellows leakage detection system. Three of these eleven operational events involved the occurrence of bellows leakage alarms on two SRVs simultaneously. In these instances, the affected facilities were shut down immediately to investigate and resolve the problem. The single remaining bellows leakage event occurred during bench testing of the SRVs while the affected facility was shut down. In this instance, three SRVs were found to have leaking bellows assemblies. The defective valves were repaired prior to the resumption of power operation.

*In several instances, bellows leakage alarms which were received on the leak detection system were later determined to have been caused by an electrical failure of the bellows pressure switch.

Two of the remaining four SRV-related potential failure-to-open events involved electrical problems (e.g., air supply solenoid electrically grounded) which affected only one SRV. The third event, which occurred during bench testing of the SRVs while the affected facility was shut down, involved the failure of all six of the SRVs utilized in the pressure relief system to open at the required setpoint pressure. Two valves opened at an elevated pressure and four valves failed to open at all. These valves were of a unique design and were subsequently replaced with the type of SRV utilized in other operating BWR facilities. The fourth event, which also occurred while the affected facility was shut down, involved the deterioration (delamination) of the air actuator diaphragms for 9 of the 11 SRVs. The degraded diaphragms were replaced prior to the resumption of power operation.*

2.3.2 PARVs

The only PARV-related potential failure-to-open event involved the failure of an electrical actuating switch

*Of the sixteen SRV-related potential failure-to-open events, this event is the only one which involved a potential major degradation of the ADS.

in the ADS. This failure rendered three of the six ADS valves inoperable; however, the remaining three valves would have adequately performed the design function of the ADS if a postulated small break LOCA had occurred.

2.4 Experience Summary

Table 5 summarizes the operating experience with SRVs and PARVs.

3.0 EVALUATION OF INADVERTENT BLOWDOWN EVENTS DUE TO PRESSURE RELIEF SYSTEM VALVE MALFUNCTIONS

Safety considerations related to inadvertent pressure relief system blowdown events include: (1) the imposition of unnecessary or excessive thermal stresses on the reactor vessel and on the vessel internals, (2) the imposition of unnecessary hydrodynamic loads on the torus and its internal components, and (3) potential increases in the release of radioactive material to the environs.

Each of these considerations is discussed in the following sections.

3.1 Thermal Stress Considerations

During an inadvertent pressure relief system blowdown event, the reactor coolant temperature will drop due to one or more of the following: (1) the release of energy through the open valve(s), (2) a rapid decrease in reactor heat generating capacity (reactor scram)*, and (3) the addition of cold feedwater or makeup water to maintain reactor coolant inventory. This drop in reactor coolant temperature will, in turn, induce a thermal transient on the reactor vessel and the vessel internals.

The reactor vessel wall will cool down at a rate determined mainly by its mass and the magnitude of the reactor coolant thermal

*If the reactor is initially operating at high power, the inadvertent opening of a pressure relief system valve may cause a scram on high power, or the reactor may scram on low pressure. A turbine trip would also occur which would remove the plant load.

transient. Since only the vessel inside wall is directly exposed to the reactor coolant, the temperature of the outer wall will lag behind the temperature of the inner wall. Thermal stresses due to the differential temperature across the vessel wall will result. The significance of these thermal stresses is discussed in Section 3.1.1.

Unlike the reactor vessel, the vessel internals are immersed in the reactor coolant and will be uniformly cooled throughout the temperature transient. The vessel internals are also thinner and less massive than the vessel wall, so their temperature more closely follows the coolant temperature. Since the vessel internals do not develop significant temperature gradients during pressure relief system blowdown events, the thermal stresses developed in the vessel internals during an inadvertent pressure relief system blowdown event are not of significant concern.

3.1.1 Reactor Vessel Thermal Stress Considerations

In addition to assuring that the reactor vessel can accommodate, with sufficient safety margin, the maximum stress loadings anticipated to occur, vessels are also designed for fatigue considerations.

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As part of the design procedures for a reactor vessel, a specific number of stress cycles of various magnitudes are postulated to occur. These correspond to the many operational events which are likely to occur during the anticipated 40 year life of the vessel. A listing of the various types of stress cycles, utilized in the design of the reactor vessel for an older operating BWR facility, including the number of times each is postulated to occur during the life of the vessel, is shown in Table 6. In the case described, two pressure relief system valve blowdown events were postulated. For newer BWR facilities, about eight such events have been postulated to occur during the life of the vessel.

As discussed in Section 2.1, inadvertent pressure relief system blowdown events have occurred more frequently than originally anticipated during the design of most operating BWR facilities. The staff has evaluated the response of these older vessels to more than the design number of stress cycles associated with pressure relief system blowdown events.

3.1.2 Effects of Stress Cycles

When the reactor coolant system is heated and pressurized during a normal startup operation, the vessel is subjected to stresses due to the change in vessel internal pressure and due to the transient effects of unequal temperatures in the various regions of the vessel. The magnitudes of these stresses are kept within allowable bounds by proper design and operation. When the system is depressurized and cooled down, these stresses are removed and a stress cycle has been completed.

When a number of design stress cycles of a specifc type are postulated to occur during the life of a reactor vessel, it does not imply that the vessel will fail if that number is exceeded, rather it 's a way of incorporating an estimate of the number of such cycles the vessel will experience into the vessel design considerations. After estimating the number of all the various anticipated stress cycles, the designer quantifies the combined fatigue effect on the vessel. This is accomplished in accordance with procedures specified by the ASME design codes.

The fractional amount that each stress cycle fatigues the vessel is known as a usage factor. The usage factors for each type of stress cycle are multiplied by the number of cycles of each type which are postulated to occur and are summed for each of the various regions of the reactor vessel. The resultant total usage factor must be less than unity for each region of the vessel, where unity is the design limit specified by the code. The total usage factor, not the usage factor associated with a particlar stress cycle, is used as the main indicator of vessel fatigue strength. Usually, vessels are designed with the total of the individual usage factors equal to less than unity because it is recognized that the number of cycles of each type that will occur during the life of the vessel cannot be accurately predicted. (A typical design total usage factor is 0.5). In addition, stresses per cycle are calculated in a conservative manner. There is additional margin to failure because the ASME code limit is itself conservative (it contains a margin of a factor of 2 on stress or 20 on the number of cycles, whichever is controlling). Should the ultimate fatigue limit (without all the safety factors) be reached during the life of a vessel, it is generally believed that the result would not be immediate vessel failure but probably the onset of fatigue cracking. Such shallow cracking would probably not be serious unless the cracks were allowed to grow longer and deeper over a period of years.

Cyclic stresses of concern during an inadvertent pressure relief system blowdown event are caused by the cooling of the inner surface of the vessel while the bulk of the vessel wall remains at a higher temperature. This differential temperature results in thermal stresses in the wall because the inner surface tries to contract or shrink relative to the outer surface. The stress magnitude is proportional to the temperature difference and is normally kept low by limiting the total water temperature change or by limiting the rate of temperature change. At a typical Technical Specification cooldown limit of 100°F/hour, the thermal stresses are well within allowable limits. For many of the reported inadvertent pressure relief system blowdown events, either the cooldown rate was less than 100°F/hour or the total temperature change was not significant. These particular events are not of safety significance because the resulting thermal stresses were quite low.

Because some of the inadvertent pressure relief system blowdown events resulted in faster than normal cooldown rates over a significant temperature range, the staff performed an analysis of a hypothetical worst case event.

The resulting usage factor for such a postulated worst case event was determined to be less than 0.001. By comparison, the usage factor per pressure relief system blowdown event based on more realistic assumptions is about 0.0001, which is of the same order as the usage factor due to a typical startup-shutdown cycle. Assuming that the average blowdown event has a factor of 0.0001 and that 100 such events occur during the life of the vessel, the total usage factor would be increased by only 0.01 as a result of inadvertent blowdown events.

Thus, it is concluded that these BWR pressure relief system blowdown events are not likely to significantly affect the reactor vessel fatigue life even if they were to continue to occur at a frequency even greater than that indicated by operating experience.

3.2 Pressure Suppression Pool Dynamic Loading Considerations

The steam discharge from an SRV or a PARV is routed through piping from the drywell to the suppression pool (Figure 8). There, the steam is condensed and the energy is absorbed by the water in the suppression pool. Prior to relief valve actuation, the discharge piping between the relief valve and

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the suppression pool is filled with air and the discharge piping below the surface of the suppression pool is filled with water.

3.2.1 Dynamic Loading Description

During relief valve actuation, high pressure steam compresses the air column and accelerates the water leg in the submerged section of the discharge line. When the water leg has been discharged, the compressed air is released into the pool. The air bubble expands in the pool causing a short duration, high pressure load on the suppression chamber (torus) structures and components. In addition, the momentum of the displaced pool water causes the air bubble to over expand and contract and subsequently collapse, causing vibratory loads on the torus structures and components. The steam subsequently discharged into the suppression pool causes low amplitude pressure oscillations on the torus structures and components which continue for the remainder of the blowdown event. In addition to the above-mentioned loads, the high mass flow rates through the discharge lines create reaction forces on the discharge line piping supports, and the pool motion induced by the discharge flow causes drag loads on the structures and components located within the pool.

As the steam discharge continues, the temperature of the suppression pool will rise as the energy of the steam is absorbed by the pool. At a point referred to as the "threshold temperature", the steam condensation process would become unstable and the pressure oscillation loads could increase by a factor of ten or more. This effect is referred to as the steam quenching vibration phenomenon. The threshold temperature for this phenomenon is a function of the discharge mass flow rate and is considered to occur when the bulk pool temperature is approximately 150°F to 170°F.

3.2.2 Summary of Operating Experience

A large number of pressure relief system actuations have occurred in both domestic and foreign BWR facilities. In a number of cases, typically early in the life of a given facility, localized damage to the discharge line restraints in the suppression pool and to the suppression pool baffles has occurred. The cause of this localized damage has been attributed to the reaction loads and to the pressure forces generated during the discharge of the air bubble. In these cases, the affected structures were repaired or modified such that additional structural capacity was provided. In no case did this localized damage result in a loss of containment function or a release of radioactivity.

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In an event in April 1972 at a foreign BWR facility, significant steam quenching vibrations were encountered when the pool temperature was in excess of 160°F. These vibrations caused the suppression pool liner to separate from reinforcing beams in the bottom of the structure. However, this facility utilized a straight pipe relief valve discharge device, while domestic facilities typically utilize a split (e.g., ramshead) discharge device. The ramshead device is considered to have a relatively higher threshold temperature. At least one relief valve discharge event has occurred in a domestic facility where the suppression pool temperature exceeded 160°F, with no visible evidence of damage to the suppression chamber.*

3.2.3 Ongoing Review Efforts

In response to the concerns relating to relief valve loads, letters were sent in 1975 to all licensees of operating BWR plants requesting that they report on the potential magnitude of relief valve loads and on the structural capability of the suppression chamber and internal structures to

*Current practice for domestic BWR plants is to restrict the maximum allowable bulk suppression pool temperature to limits specified in the facility Technical Specifications for specific plant operating modes, such that the threshold temperature would not be reached. Further, plant operating procedures include specific provisions to minimize the potential for exceeding the threshold temperature.

tolerate such loads. In addition, the consideration of relief valve loads has become an integral part of the staff's review of construction permit and operating license applications for all BWR pressure suppression containment designs (i.e., Mark I, II, and III).

As a result of generic suppression pool hydrodynamic concerns, Owners Groups were formed by utilities with plants utilizing the Mark I and Mark II containment designs. Through these Groups, generic, analytical and experimental programs have been developed to address relief valve loads. For the operating facilities, the relief valve related tasks of the Mark I containment Long Term Program (LTP) are intended to improve the quantification of relief valve loads, to confirm the suppression chamber structural margins, and to confirm the adequacy of the suppression pool temperature limits.

3.2.4 Conclusions

The staff believes that there is no immediate (i.e., short term) potential hazard for the major containment structures from the vibratory loads associated with relief valve operation due to the slowly progressive nature of the material fatigue mode of failure associated with cyclic loadings. Based upon the test results and analyses reported by the

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General Electric Company in "Steam Vent Clearing Phenomena and Structural Response of the BWR Torus," NEDO 10859, April 1973, substantial fatigue life margin is available in the torus structure to accommodate the potential relief valve actuations that may occur during the conduct of the LTP. The Mark I Owners Group has recently performed additional in-plant tests at the Monticello facility to identify and quantify the stresses in the torus structures associated with relief valve operation. The need for structural modifications to provide conservative margins to assure structural integrity throughout the life of these facilities will be determined during the LTP. This part of the LTP, currently scheduled for completion in early 1979, will permit necessary modifications to be instituted before any more than a small fraction of the fatigue life predicted by the GE analysis has been utilized. We are continuing our review of this matter and will take appropriate action should this program fail to resolve our concerns on an acceptable schedule.

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3.3 Radiological Considerations

As discussed previously, depressurization of the reactor system via SRVs or PARVs is accomplished by transferring steam from the reactor vessel to the torus. Since the steam contains radioactive gases and some non-gaseous (soluble and particulate) radioactive isotopes which are entrapped in small amounts of water which may be swept along with the steam (called carryover), a small fraction of the primary coolant radioactivity inventory can be transferred to the torus along with the steam. The staff has been evaluating the increased liquid and airborne radioactivity concentrations in the torus from this process and has generally concluded that they are not significant for the following reasons:

1. Most (generally more than 99%) of the soluble and particulate radioactivity remains in the reactor coolant water and thus stays in the reactor vessel. That which is carried over into the torus is diluted by at least a factor of 20 in the suppression pool water.

2. The increased radioactivity in the suppression pool water can be removed by conventional reactor water treatment systems, for example, the condensate polishing system.

3. Gaseous radioactivity in the steam will be released to the air space in the torus. This gaseous activity is basically

no different than that which would have been transferred to the condenser air ejector and offgas system during normal operation and normal shutdown. Even though there may be increases in radioactivity concentrations within the torus, it is not expected that there will be significant increases in the radioactive effluents to the environment since the torus is part of the primary containment and is basically sealed from the environment. Generally, the torus air space is not vented to the environment, but it can be vented if personnel access is required. If this is necessary, the gaseous radioactivity will have been decayed and will be vented to the environment through charcoal and particulate filters such that radioactive effluents should be less than that which is specified in the facility Technical Specifications.

Consequently, we have concluded that a blowdown via the SRVs or PARVs does not have a significant impact on the environment appreciably different than that encountered during normal reactor shutdown.

3.4 Conclusion

The inadvertent blowdown events which have occurred to date as a result of pressure relief system valve malfunctions have neither significantly affected the structural integrity or capability of the reactor vessel, the reactor vessel internals, or the pressure-suppression containment system nor resulted in any significant radiation releases to the environment.

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The staff has concluded that such events, even if they were to occur at a more frequent rate than that indicated by operating experience, would not likely have any significant effects on the reactor vessel or the vessel internals. The staff has also concluded that pressure relief valve blowdown events will not result in offsite radiological consequences appreciably different from those encountered during a normal reactor shutdown.

With respect to the pressure-suppression containment system, due to the slowly progressive nature of the material fatigue mode of failure associated with the dynamic loading conditions resulting from pressure relief valve blowdown events and due to the substantial fatigue life margin currently available in the affected structures, the staff has concluded that additional short term actions are not required to assure that the integrity and functional capability of the system will be maintained. The staff has also determined that the currently ongoing programs to provide additional containment system structural safety margins for the long term (i.e., the anticipated 40 year lifetime of BWR facilities) are acceptable.

4.0 EVALUATION OF EVENTS INVOLVING FAILURES OF PRESSURE RELIEF SYSTEM VALVES TO OPEN PROPERLY

The safety considerations related to the failure of a pressure relief valve to open properly include the potential degradation of the overpressurization protection provided for the reactor vessel and the primary coolant system, and the continued capability of the ADS to function as designed to mitigate the consequences of a postulated LOCA. Each of these considerations is addressed below.

4.1 Primary System Overpressure Considerations

4.1.1 SRVs

Since the self-actuation mode of a SRV provides the automatic overpressure protection for facilities in which such valves are installed, only failures to open in the self-actuation mode of operation need be examined.

The majority of the SRV failure or potential failure to open events (self-actuation mode) have been attributed to bellows assembly leakage. Since the bellows assembly of each SRV is continuously monitored for leakage, the likelihood of such failures occurring without being detected is low. Upon detection of a leaking SRV bellows assembly, appropriate operator action is required.*

*Since an additional SRV (beyond that which is taken credit for in the plant overpressure protection analysis) is installed in most operating facilities, plant shutdown usually would not be required unless bellows leakage was detected on two valves.

During each refueling outage a specified percentage of the installed SRVs are required to be bench-tested to verify their operability in the self-actuated mode. The percentage of valves that are required to be tested ranges from 20% to 50% based on plant unique considerations; however, during the past few years most licensees have elected to test more than the minimum required number of valves.

As discussed in Section 2.2, operating experience includes five SRV failure-to-open events during pressure transients where the failure occurred in the self-actuation mode. Of these, only one event involved the failure of more than one valve. The peak primary coolant pressure resulting from any of these events was 1140 psig which is well within the reactor coolant pressure boundary design pressure (1250 psig). As discussed in Section 2.3, operating experience includes thirteen potential failure-to-open events involving the self-actuation mode of operation of an SRV. Of these, only one event, which was detected during bench testing while the affected facility was shutdown, involved the potential for system overpressure. The valves involved in this event were of a unique design and were replaced prior to resumption of power operation.

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In summary, based on the demonstrated performance and established surveillance programs for SRVs, the staff has determined that they provide reliable overpressure protection for the RCPB.

4.1.2 PARVs

With respect to the PARV failure-to-open events, it should be noted that credit is not given for the relieving capacity of these valves in the overpressure analysis for the facility. As discussed in Section 1.1, the design of the PARVs is based on establishing an adequate pressure relieving capacity to avoid opening of the SVs during normal operational events.

Operating experience indicates that in only one case have more than two PARVs been in a condition where they would have failed to open during a pressure transient at a given time. Even in this case, it is not likely that SV actuation would have been required in the event of a pressure transient. However, in the event that SV actuation would be required to avoid an overpressurization event, the radiological consequences of such an actuation, as discussed below, are not significant.

When SV blowdown occurs, the steam is transferred first to the drywell and thence to the torus. The amount of radioactivity transferred to the drywell is similar to that transferred to the torus during a relief valve blowdown. This results in increased airborne radioactivity concentrations in the drywell. As the steam condenses into water, it will wash the soluble and particulate radioactivity down into the torus. The torus water could then be processed through a treatment system to remove soluble radioactivity.

There would also be increased airborne concentrations of noble gases and some radioiodines within the drywell. Even though the drywell aiborne concentrations would be increased by this event, this is not expected to lead to significant increases in radioactive effluents. The drywell is part of the primary containment system and is isolated from the environment. Most of the radioactivity should be of short half life such that, if necessary, the drywell need not be purged until most of the radioactivity has decayed to acceptable levels. In addition, charcoal and particulate filters (the Standby Gas Treatment System) can be used to filter any leakage or ventilation exhaust from the drywell. The use of these filters, coupled with sampling of the drywell atmosphere and the technical specification requirements related to radioactive effluent releases, provide adequate assurance that SV blowdown would not have any impact on the environment appreciably different than that encountered during normal reactor shutdown.

4.2 Automatic Depressurization System Operability Considerations

4.2.1 SRVs

As discussed previously a specified number of SRVs are utilized in the externally actuated mode as part of the facility's ADS. Consequently, only failures to open in the externally-actuated mode of SRV operation need be considered.

As discussed in Section 2.2, operating experience includes six events where SRVs have failed to open in the externally-actuated mode of operation. Of these, only one event involved more than one valve in the ADS.* In that event, three of the four ADS valves were determined to be inoperable due to the common mode failure of the air actuator diaphragm due to overheating. The overheating had been the result of improper installation of insulation on the valve. Although this event resulted in a major degradation on the ADS, the diverse HPCI system was available to provide protection in the event of a postulated small break LOCA.

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*In general, the ADS will include a larger number of SRVs than that which is necessary to accomplish its design function. At least one extra SRV is usually included.

As discussed in Section 2.3, operating experience includes one potential failure-to open event involving the externally-actuated mode of SRV operation. This event, which was discovered during maintenance while the affected facility was shut down, involved the deterioration of the air actuator diaphragms for 9 of the 11 SRVs. It is believed that this deterioration had not yet progressed to the point where the valves would have failed to function on demand. In any event, the diverse HPCI system was operable.

Based on the demonstrated performance and established surveillance programs for SRVs used in the ADS, we conclude that such valves provide reliable protection against a postulated small break LOCA. The existence of a diverse HPCI system at most operating BWR facilities further enhances the reliability of the protection provided against such an accident.

4.2.2 PARVs

As discussed in Sections 2.2 and 2.3, operating experience includes a number of events where PARVs utilized in the ADS failed to open. Of these, only one constituted a significant degradation of the ADS. In that event, three of the six ADS valves were rendered inoperable by the failure of the electrical actuating switch. However, the remaining three valves would have adequately performed the design function of the ADS in the event of a postulated small break LOCA.

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Other failures of PARVs have been limited to one or two valves, and, in all cases, either a sufficient number of valves were operable to perform the ADS function and/or the diverse HPCI system was operable.

4.3 Conclusion

The events involving failures of pressure relief system valves to open properly which have occurred to date have not resulted in a significant degradation of either the overpressurization protection provided for the reactor vessel and the primary coolant system or the protection provided for a postulated small break LOCA.

Only a few of these events have involved the simultaneous failure of more than one pressure relief valve to open properly. Although even in those instances the consequences were minimal, the staff believes that such occurrences are undesirable and that appropriate actions should continue to be taken to assure that the high level of reliability of the overpressurization protection provided by the pressure relief system is maintained.

Only two of these events have involved a significant degradation of the capability of the ADS to perform its design function in the event of a postulated small break LOCA. Although in each of these instances adequate protection capability was available either through the remaining operable ADS valves or the diverse HPCI system, the staff believes that such events are undesirable and that appropriate actions should continue to be taken to assure that the high level of reliability of the ADS is maintained.

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5.0 CORRECTIVE ACTIONS

For some time licensees, manufacturers of pressure relief valves and GE have been working with the NRC to improve pressure relief valve maintenance programs. In general, an increased number of pressure relief valves are periodically inspected, tested, and refurbished at operating BWRs. The reduction in the reported failure rate for pressure relief valves from approximately one failure per 40,000 hours of operation in 1974 to one failure per 80,000 hours of operation in 1976 (a factor of 2 improvement) can, for the most part, be attributed to these improved maintenance programs.

With respect to the reliability of SRVs, GE, working with the valve manufacturers, has developed a two-phase program to further improve reliability of the SRVs. A short term program has been initiated which involves minor modifications to the existing 3-stage SRV design and increases monitoring to detect valve leakage prior to inadvertent actuation. Since almost all the inadvertent blowdown events caused by SRV malfunctions have occurred on valves which have had less than 100 psi of "simmer margin,"* this short term effort is designed to increase the simmer margin of such valves. In order to increase the simmer margins at their facilities, several licensees have increased

*Simmer margin is the differential pressure between normal system operating pressure and the SRV setpoint.

the SRV setpoint pressure for their valves. In some cases this required a grinding out of the valve discharge "throat" diameter in order to assure adequate relieving capacity.

The long term program involves the development and installation of an improved self-actuation mechanism for the SRVs. The most significant changes incorporated in this new design are the elimination of one of the pilot stages and the elimination of the bellows assembly. The new design is expected to improve the performance of the SRVs by eliminating failures of valves to open due to bellows leakage and by eliminating spurious valve openings due to leakage past the first stage pilot seat. This second improvement is accomplished by preventing such leakage from building up pressure on the actuating mechanism of the main valve.

Valves of this new design have undergone a testing program which is currently under review by the NRC staff. The NRC also has under review applications to utilize valves of this new design or two facilities. It is anticipated that licensees of other operating facilities will shift to this new design, as appropriate, in the future.

With respect to NRC-initiated action, on several occasions the NRC staff has sent IE Bulletins or Circulars to licensees of operating facilities informing them of particular problems with pressure relief valves.

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In addition, the NRC staff has been considering revising facility Technical Specification requirements related to pressure relief valve inspection and testing. Some of the changes that the staff presently has under consideration are:

(1) A variable frequency operational testing schedule based on the number of PARVs and SRVs that failed to open on demand since the last testing period.

(2) Increased refurbishing and bench testing requirements for PARVs and SRVs that would vary as a function of the number of valves that are spuriously opened or opened and failed to reseat properly during a specified period.

(3) Increased surveillance of the structural integrity of relief valve line restraints in the pressure suppression pool based on the frequency of relief valve actuation.

One of the considerations as to whether or not to implement such Technical Specification requirements will be an assessment of continuing operating experience with respect to pressure relief valve failures.

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6.0 CONCLUSION

As discussed in this report, the staff has concluded that the events involving BWR pressure relief system valve malfunctions which have occurred to date have neither resulted in significant impact on the nuclear facility nor involved any significant radiation release to the environment.

Although the events which have occurred to date have had only minimal consequences, the staff believes that appropriate actions should continue to be taken to improve the reliability of the pressure relief system valves and, thereby, to reduce the frequency of occurrence of such events. This belief is primarily based on the fact that such events involve a reduction in the original design safety margins provided for overpressurization events and for a postulated small break LOCA.

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TABLE 1

BWR TYPE	FACILITY	VALVE COMPLEMENT		
		SAFETY/RELIEF VALVES	SAFETY VALVES	POWER-ACTUATED RELIEF VALVES
2	Nine Mile Point	-	16	6
2	Oyster Creek	-	16	5
3	Dresden 2	1	8	4
3	Dresden 3	1	8	4
3	Millstone 1	6	-	-
3	Monticello	7	-	-
3	Pilgrim 1	4	2	-
3	Quad Cities 1	1	8	4
3	Quad Cities 2	1	8	4
4	Browns Ferry 1	11	2	-
4	Browns Ferry 2	11	2	-
4	Browns Ferry 3	11	2	-
4	Brunswick 1	11	-	-
4	Brunswick 2	11	-	-
4	Cooper	8	3	-
4	Duane Arnold	6	2	-
4	Fitzpatrick	11	-	-
4	Hatch 1	11	-	-
4	Peach Bottom 2	11	2	-
4	Peach Bottom 3	11	2	-
4	Vermont Yankee	4	2	-

TABLE 2

Inadvertent Blowdowns Events

	1969	1970	1971	1972	1973	1974	1975	1976	1977	1978
Browns Ferry 1						2	1		2	
Browns Ferry 2						2	1			2
Browns Ferry 3										1
Brunswick 1										
Bruwnswick 2							2	1	1	
Cooper						4				
Dresden 2								1		
Dresden 3										
Duane Arnold										
Fitzpatrick								3		
Hatch 1							1	1	3	
Millstone 1			1				1		1	1
Monticello			1	1						2
Nine Mile Pt. 1					1*					
Oyster Creek				1*						
Peach Bottom 2						3	1	2	1	
Peach Bottom 3								2		
Pilgrim 1				2	1		1			
Quad Cities 1										
Quad Cities 2									2*	
Vermont Yankee										
TOTAL	0	0	2	4	2	11	8	10	10	6

*Power Actuated Relief Valve

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TABLE 3

Failures to Open Properly on Demand

	1969	1970	1971	1972	1973	1974	1975	1976	1977	1978
Browns Ferry 1										
Browns Ferry 2										
Browns Ferry 3										
Brunswick 1										
Brunswick 2										
Cooper										
Dresden 2		3*					2*		1*	
Dresden 3					1*	1*				
Duane Arnold						1				
Fitzpatrick										
Hatch 1									2	
Millstone 1										
Monticello				2				1		
Nine Mile Point	*									
Oyster Creek					1*				1*	
Peach Bottom 2						1				
Peach Bottom 3										
Pilgrim 1									2	
Quad Cities 1								1*	1*	
Quad Cities 2								1*	2*	
Vermont Yankee						1		1		
TOTAL	1	3	0	2	2	4	2	4	9	0

*Power Actuated Relief Valve

TABLE 4

Potential Failures to Open Properly

	1969	1970	1971	1972	1973	1974	1975	1976	1977	1978
Browns Ferry 1					1					
Browns Ferry 2										
Browns Ferry 3										
Brunswick 1										
Brunswick 2							1			
Cooper							1	1		
Dresden 2										
Dresden 3										
Duane Arnold									1	1
Fitzpatrick								1		
Hatch 1								1	1	
Millstone 1							1			
Monticello										
Nine Mile Point									1*	
Oyster Creek										
Peach Bottom 2					1	2			1	
Peach Bottom 3										
Pilgrim 1										
Quad Cities 1										
Quad Cities 2										
Vermont Yankee				2						
TOTAL	0	0	0	2	2	2	3	3	4	1

*Power Actuated Relief Valve

TABLE 5

Event Type	Safety Relief Vavles	Power Actuated Pressure Relief Valves	Total
Inadvertent Blowdowns	49	4	53
Failures to Open	11	16	27
Potential Failures to Open	16	1	17

TABLE 6

Reactor Design Cycles (40 year life)

Type of Cycle	Number of Cycles
Bolt up	123
Design hydrostatic test at 1250 psig	130
Startup (100 F/hr heatup rate)	120
Daily reduction to 75 percent power	10,000
Weekly reduction to 50 percent power	2,000
Control rod worth test	400
Loss of Feedwater heaters (80 cycles total)	
Turbine trip at 25 percent power	10
Loss of heating to feedwater heater	70
Scram (200 cycles total)	
Loss of feedwater pumps, isolation valves close	10
Turbine trip, feedwater on, isolation valves stay open	40
Reactor overpressure with delayed scram, feedwater stays on, isolated valves stay open	1
Single relief valve or safety valve blowdown	2
All other scrams	147
Improper start of cold recirculation loop	5
Sudden start of pump in cold recirculation loop	5
Shutdown (100 F/hr cooldown rate)	118
Hydrostatic test at 1563 psig	3
Unbolt	123

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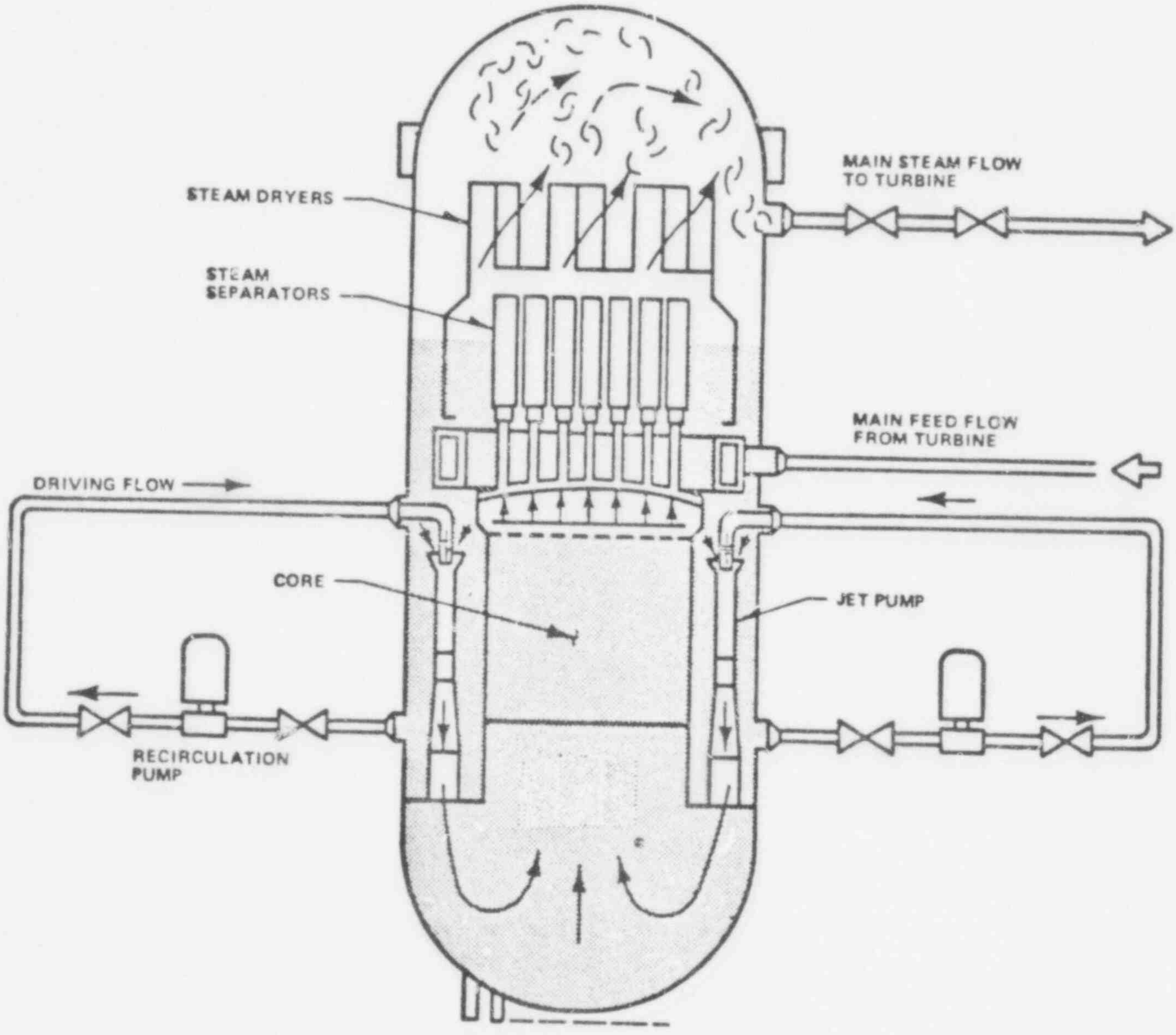

FIGURE 1 SCHEMATIC OF A TYPICAL
BOILING WATER REACTOR

501 297

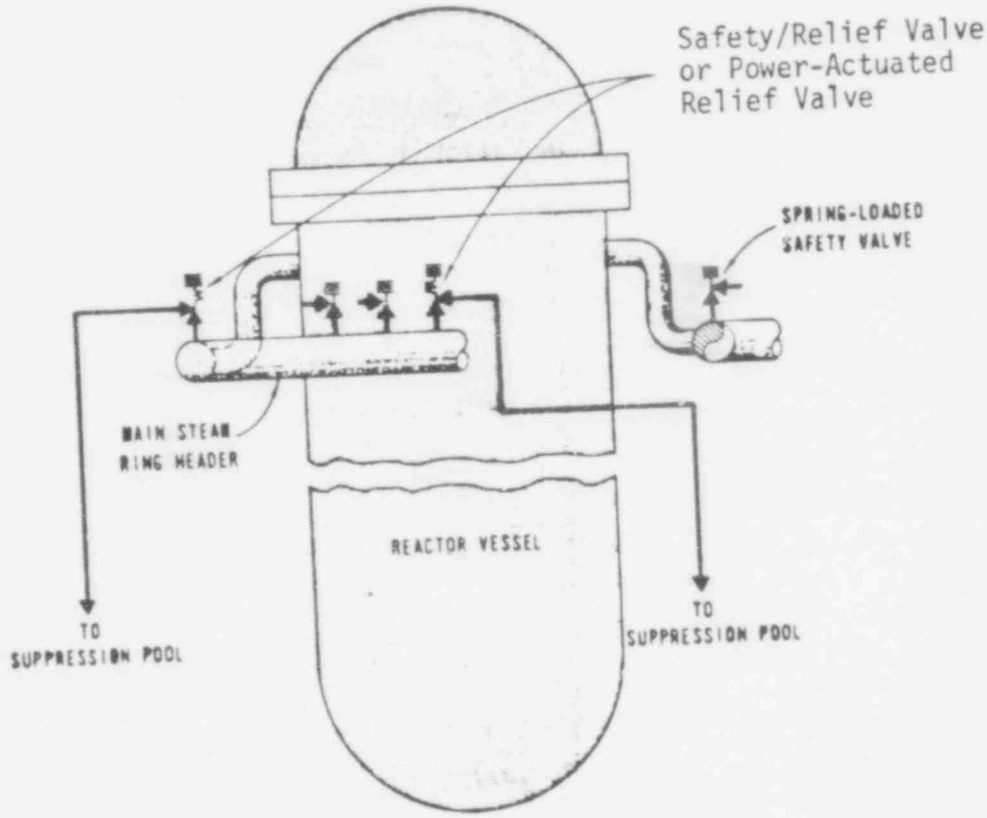

FIGURE 2

PHYSICAL LOCATION OF BWR SAFETY VALVES
AND RELIEF VALVES

501 298

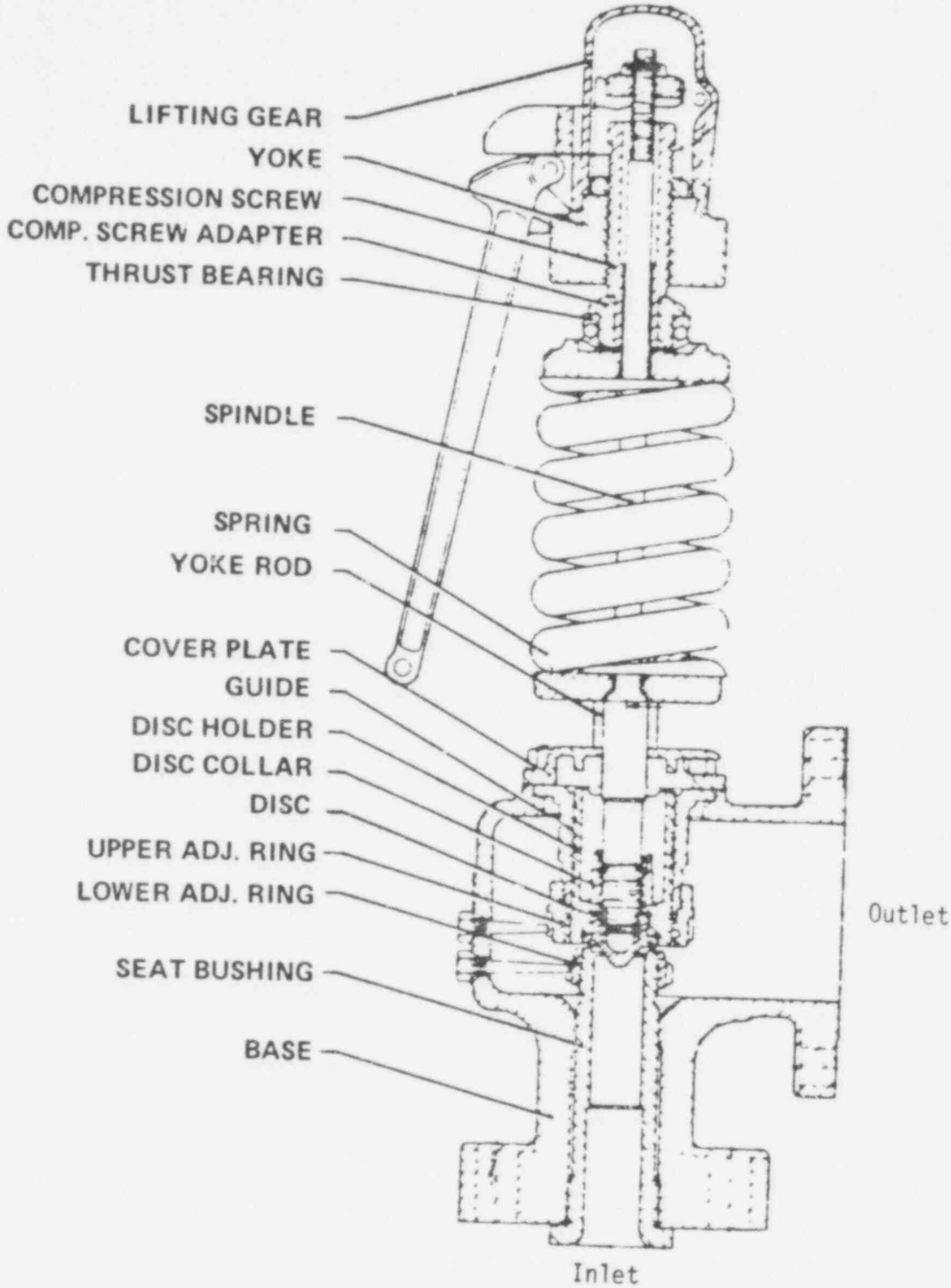

FIGURE 3
SAFETY VALVE

501 299

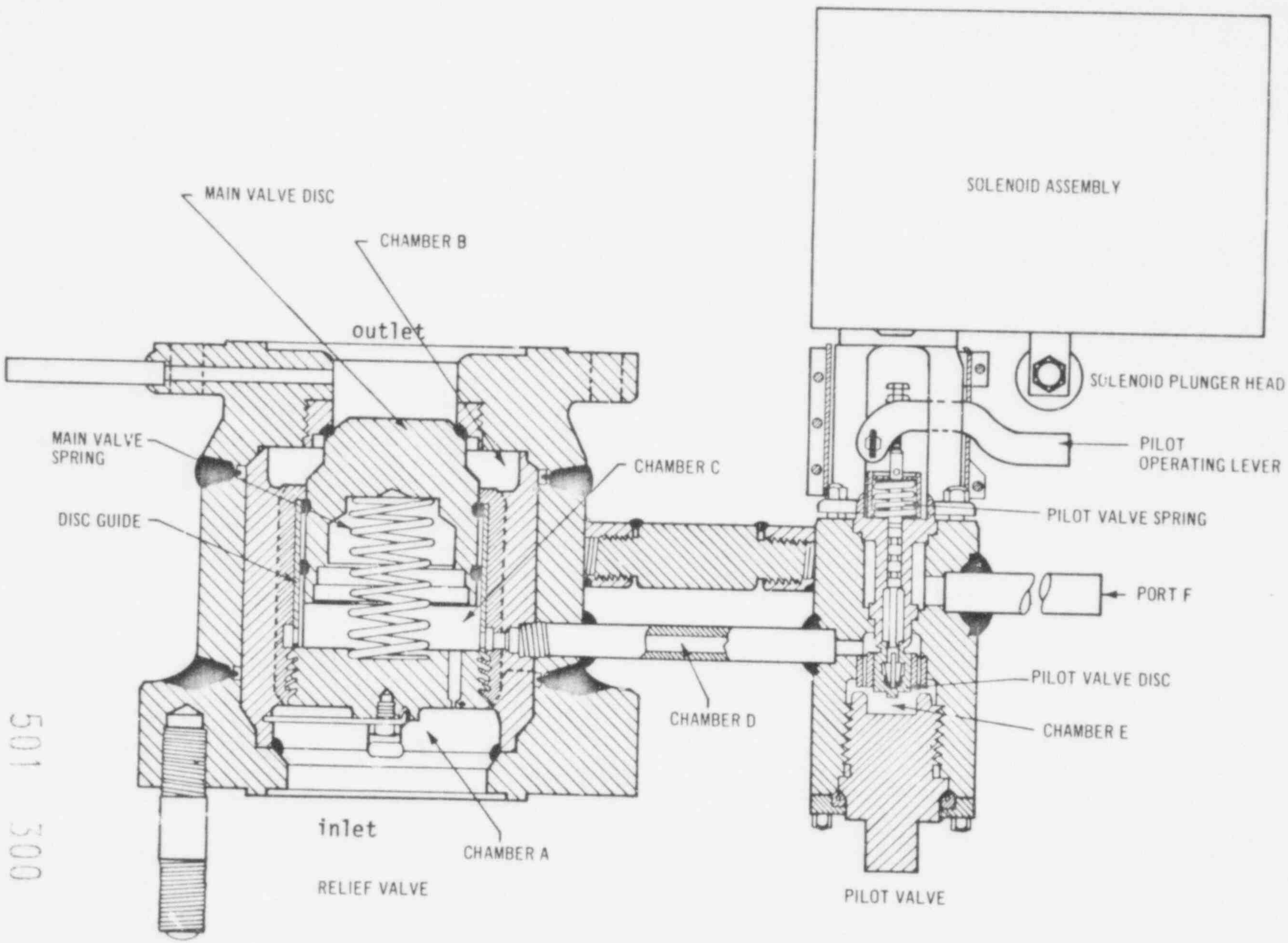

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FIGURE 4
POWER ACTUATED RELIEF VALVE

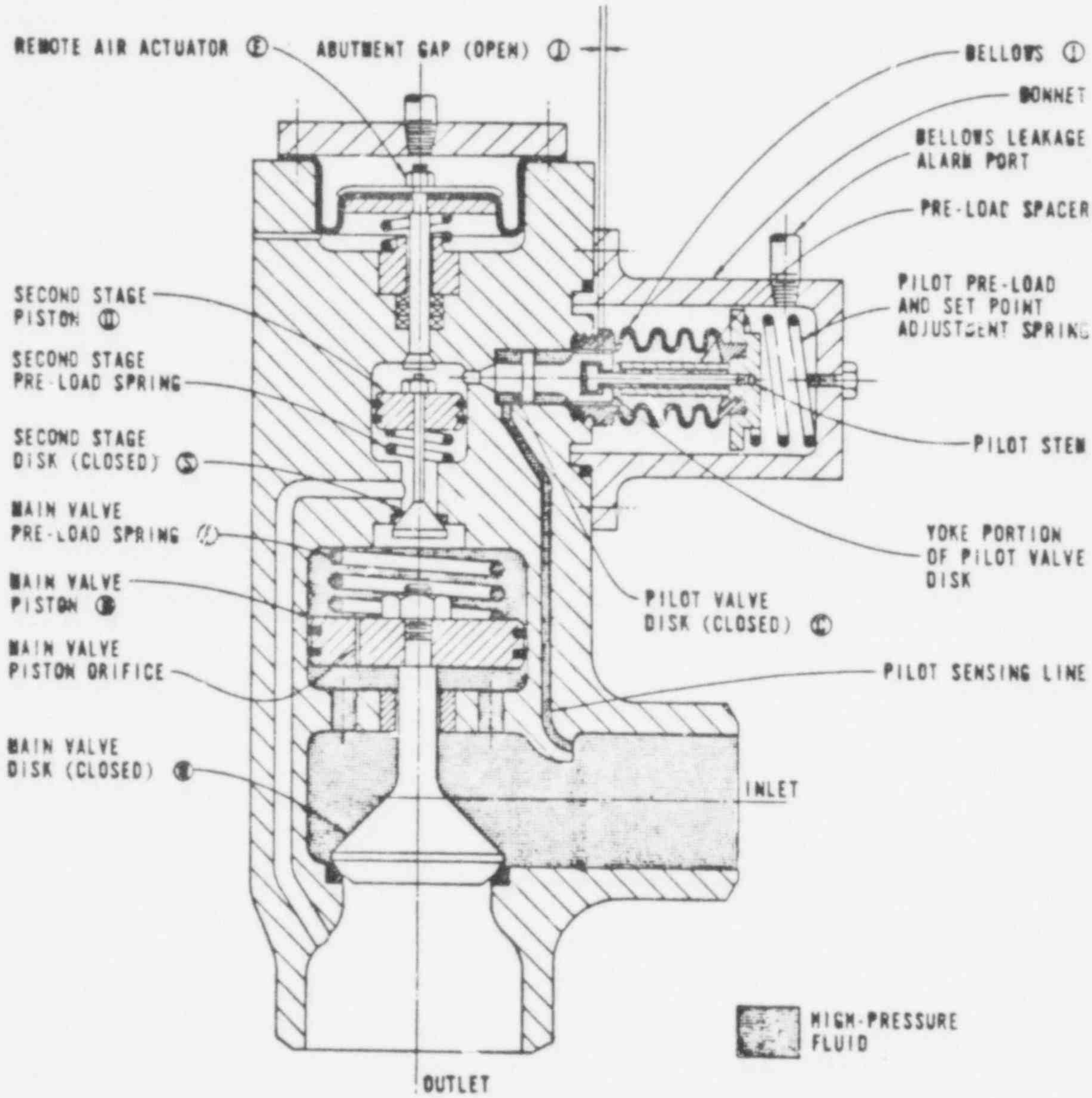

FIGURE 5 - Operational Diagram of Three-Stage Pilot-Operated Pressure Relief Valve in Closed Position.

501 301

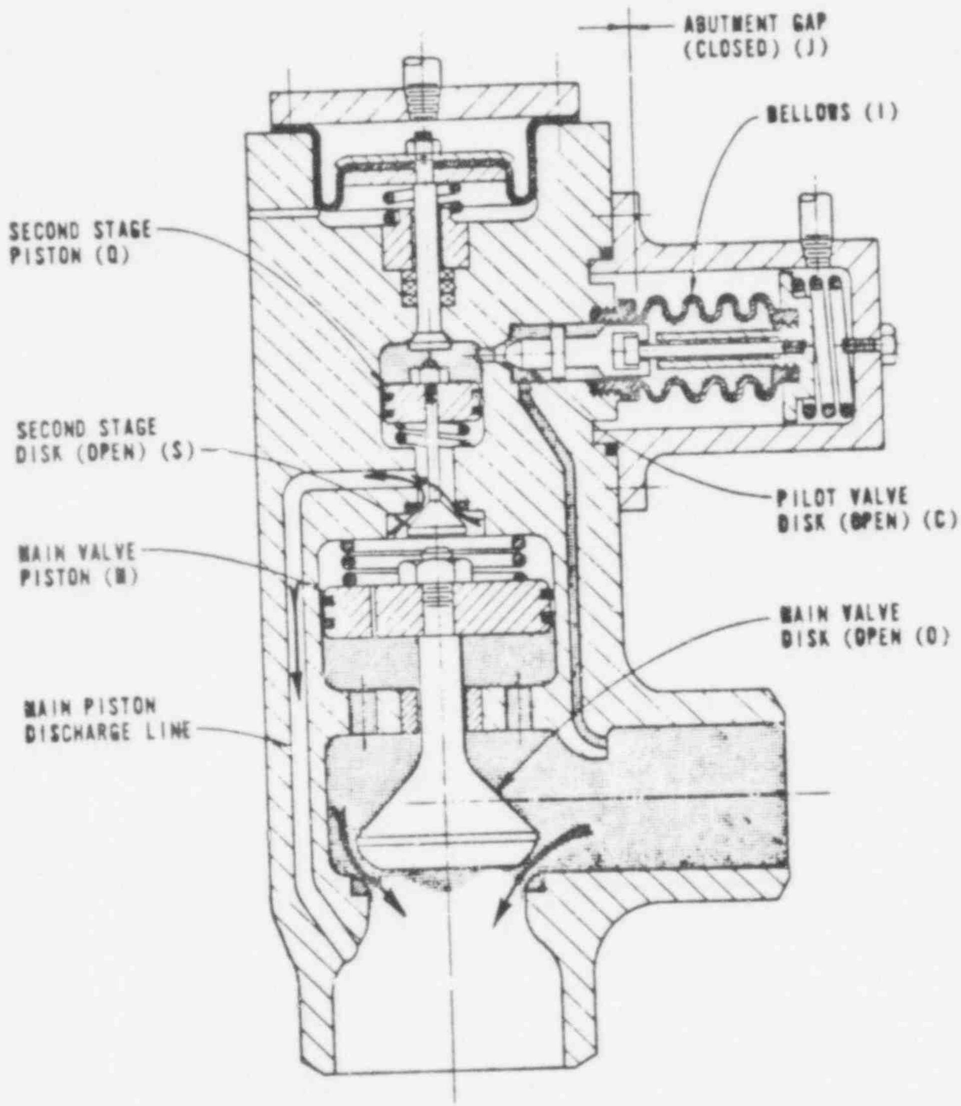

FIGURE 6 - Operational Diagram of Three-Stage Pilot-Operated Pressure Relief Valve in Open Position.

501 302

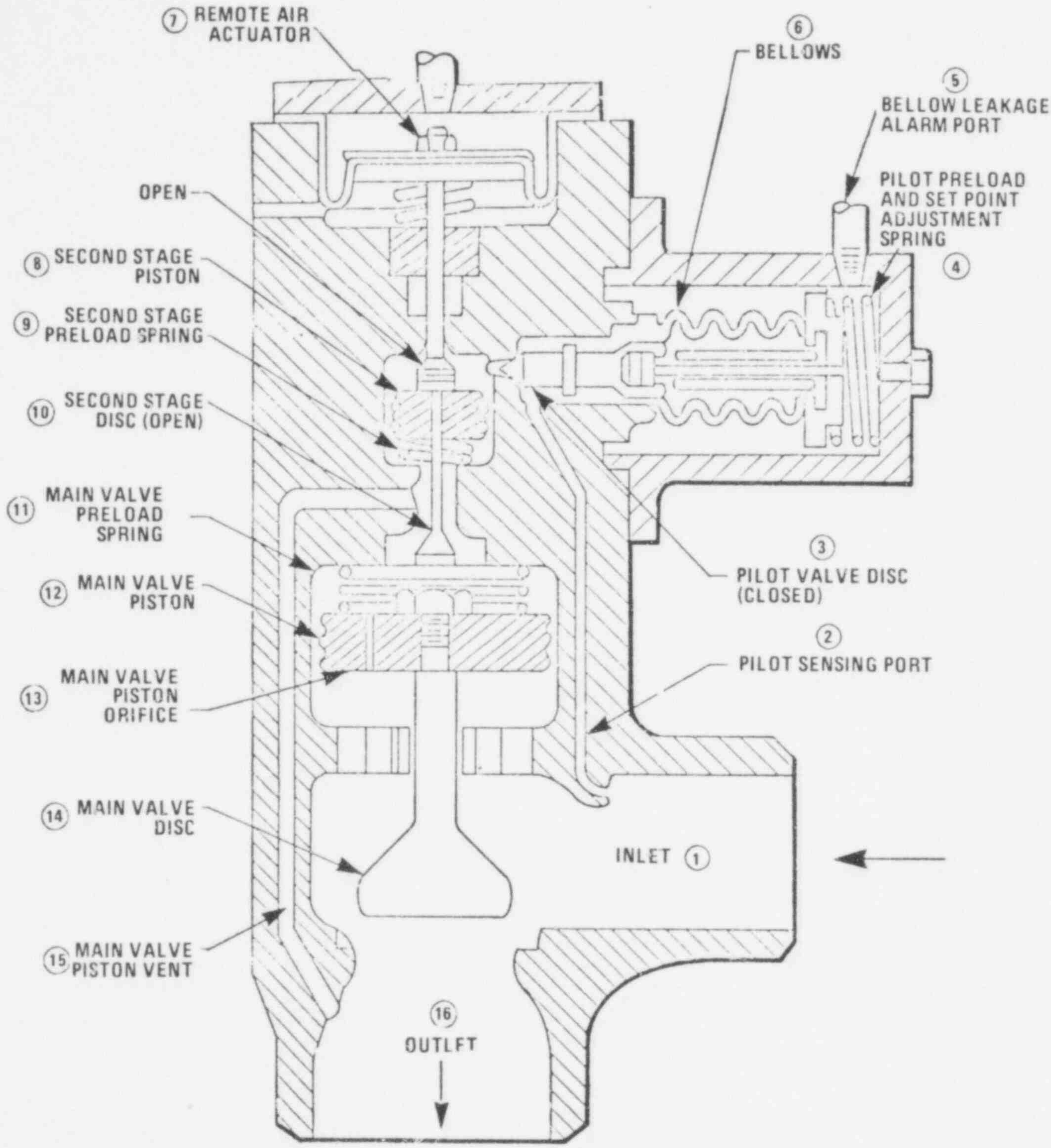

FIGURE 7 *Safety-Relief Valve (External Actuation)*

501 300

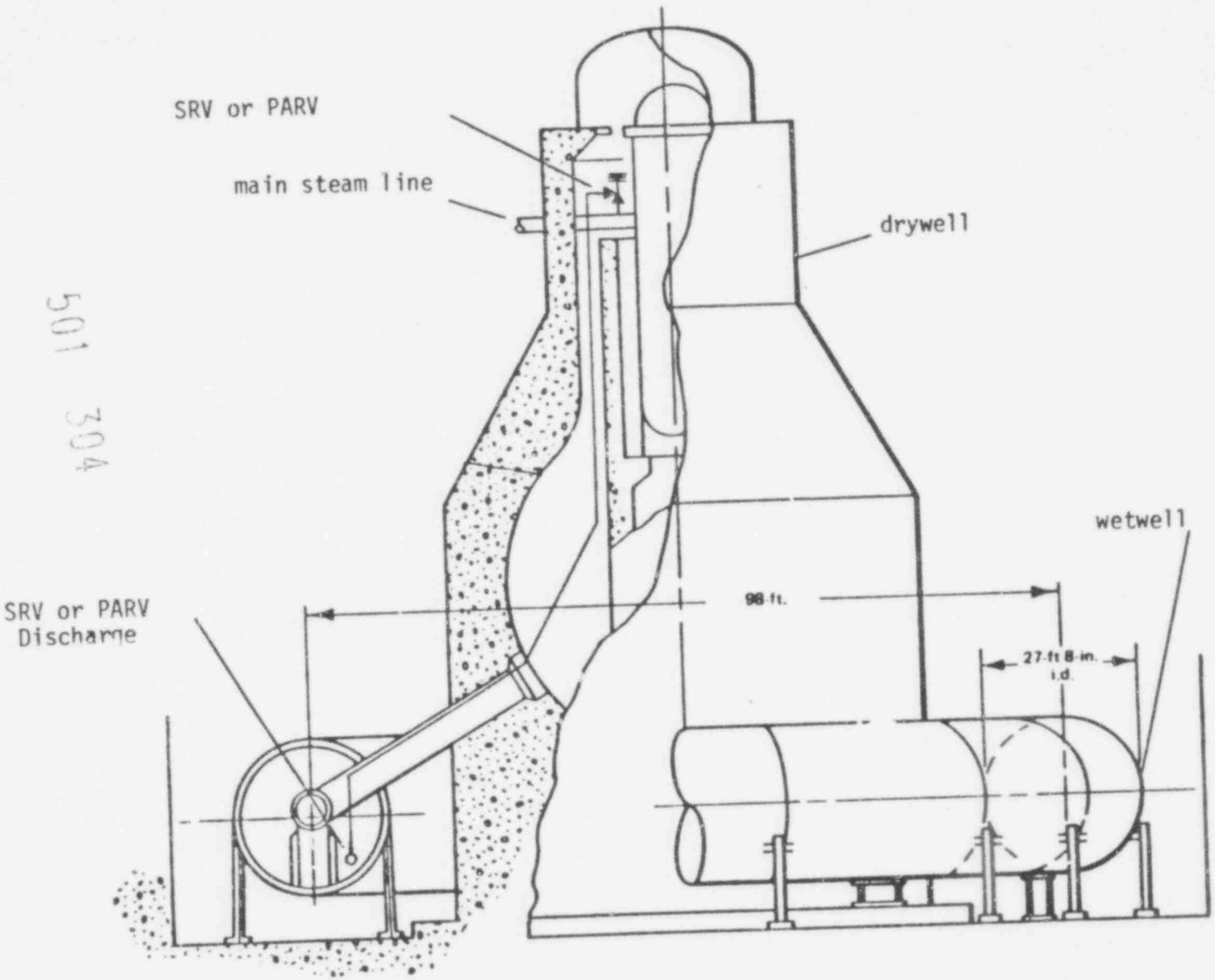

FIGURE 8
RELIEF VALVE DISCHARGE FLOW PATH

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* U.S. GOVERNMENT PRINTING OFFICE: 1978—720-387/268