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File
BWR-SRV
Report on BWR Blowdown ~~Experience~~

FROM: D. Eisenhut

SUBJECT: REPORT ON BWR BLOWDOWN EXPERIENCE

At the request of the Commission we have put together the attached draft report on BWR S/R and relief valve problems. The Commission was particularly interested in experience to date, the safety considerations associated with such events, and possible fixes.

We would appreciate your review of the attached draft report. Your comments should be provided to J. Guibert (x28256) by COB April 24, 1978.

Eisenhut

D. Eisenhut, Assistant Director
for Systems & Projects
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Enclosure:
As stated

cc: V. Stello
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TECHNICAL REPORT ON OPERATING EXPERIENCE WITH BOILING WATER REACTOR PRESSURE RELIEF SYSTEMS

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 relief and safety-relief valves utilized in these systems. This experience includes (1) valves that inadvertently opened and failed to reseat and (2) valves that failed to open properly.

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 described 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 plant electrical loads and the reactor power. When power demand is less than the power generated by the reactor, such an imbalance generally results in an increase in the pressure of the coolant. An increase in coolant pressure due to a sudden loss of load will reduce the voids (vaporous bubbles resulting from boiling) in the core cooling water, thereby causing an increase in reactor power.* Therefore, pressure transients must be limited in order to prevant possible core damage due to excessive power and to prevent damage or rupture of the reactor coolant pressure boundary due to excessive pressure.

The BWR pressure relief system is designed to prevent overpressurization of the reactor coolant pressure boundary under the most severe pressure transients (e.g., main steamline isolation valve closure at 100% reactor power) and to limit reactor pressure during normal operational transients.

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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.

Overpressure protection is provided by the use of various combinations of safety valves, relief valves, and dual function safety/relief valves.

Although there are a few operating* BWRs that predate the BWR-2s, they are not discussed herein because of their unique designs. Table 1 describes the combination of valves utilized in the pressure relief system of each operating BWR-2, BWR-3, and BWR-4 facility. Generally, pressure relief systems used in older operating BWR facilities consist of (1) a large number of spring-loaded safety valves which provide the total pressure boundary from overpressurization during the most limiting transient and (2) a smaller number of power actuated pressure relief valves which supplement the spring-loaded safety valves. The relief valves supplement the safety valves by relieving pressure surges associated with normal operational transients that might otherwise cause the higher setpoint safety valves to actuate. As shown in Figure 2, the safety valves and the relief valves are installed in the horizontal section of the main steam lines inside the containment (drywell) upstream of the first set of main steam isolation valves. Since the relief valves have a lower pressure setpoint (approximately 1130 psig) than the safety valves (approximately 1210 psig), their actuation occurs sooner and is more likely, therefore, they have been designed to discharge beneath the water level of the suppression pool (torus). Due to their higher relieving capacity and the small probability of their operation, the spring-loaded safety valves have been designed to discharge directly to the drywell.

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*Facilities licensed to operate include Big Rock Point, LaCrosse, and Humboldt Bay.

Figure 3 is an engineering drawing of a typical spring-loaded safety valve. When reactor pressure reaches the valve setpoint, the pressure exerted on the valve disc is sufficient to overcome the spring pressure and the valve will actuate. Figure 4 is an engineering drawing of a typical power actuated relief valve. When reactor pressure reaches the valve setpoint, a separate pressure switch is actuated which provides power to energize the solenoid in the pilot valve. When the solenoid is energized, steam is vented from beneath the valve disc and the resulting differential pressure on the valve disc causes it to open.

The relief valves are also used as part of the reactor's Automatic Depressurization System* (ADS) and are similarly actuated by means of protection system signals which energize the solenoid in the pilot valve.

More recent operation BWR facilities utilize dual function safety/relief valves, either exclusively or in conjunction with safety and/or relief valves. As shown in Figure 5, the safety/relief valves are also installed in the horizontal section of the main steam lines within the drywell and are designed to discharge beneath the water level of the pressure suppression pool.

The safety valve function of a safety/relief valve is provided through a two-stage pilot valve section and a main valve section (see Figures 6

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*For certain sizes of pipe ruptures in some BWRs, the rate of fluid loss exceeds the normal makeup capability of the system but would not be rapid enough to cause the reduction of system pressure necessary to permit low-pressure emergency core spray systems to function when required. In such a situation, the coolant system is automatically vented by means of the automatic depressurization system (ADS). This system senses certain plant parameters and if a small pipe break is indicated, automatically actuates certain preselected relief valves until the system is depressurized.

and 7). These two sections are directly coupled to provide a unitized, self-actuated safety valve function. The pilot valve section is the pressure sensing and control element and the main valve is a hydraulically actuated follower valve which provides the relief function. When reactor pressure reaches the valve setpoint pressure, the first stage pilot valve opens against a preset spring and allows steam pressure to open the second stage pilot which, in turn, opens a path for steam behind the main valve piston to be exhausted. Differential pressure across the main valve piston causes the valve to open.

The relief valve function of a safety/relief valve is provided through a remotely controlled (automatic or remote manual*) air operator which is fitted to provide selective operation of the valve at other than its set pressure (see Figure 8). This is a diaphragm type operator which must be energized to open the valve. It is actuated by a solenoid pilot valve which admits instrument air or nitrogen to the diaphragm and strokes the second stage disc which results in the opening of the main valve.

*The relief valve function of a safety/relief valve is utilized only for manual pressure relief or for ADS.

2.0 OPERATING EXPERIENCE WITH BWR PRESSURE RELIEF SYSTEMS

Over 100 reactor years of BWR operating experience has 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 reclose, and (3) proper opening of a valve with subsequent failure of the valve to properly reclose.

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 system, a degradation of the ADS system's capability 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 thermal transients on the reactor vessel and the vessel internals, 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) failures 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 power actuated pressure relief valves or the dual function safety/relief valves utilized in BWR-2, BWR-3, and BWR-4 facilities. Operating experience with the spring-loaded safety valves has been essentially failure free.

The operating experience discussed in this section is based upon Licensee Event Reports submitted from 1969 to the present and upon information obtained from General Electric Company, 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 purposes 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 51 inadvertent blowdown events due to pressure relief system valve malfunctions which have been reported from 1969 until 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 normal operating pressure (approximately 1000 psig) to a few hundred psig. During more than one-third of these events, the maximum allowable primary system cooldown rate of 100°F per hour was exceeded.

2.1.1 Safety/Relief Valve Malfunctions

Fourty-seven of the inadvertent blowdown events have resulted from a malfunction of a safety/relief valve. Of these events, (a) twenty-four events involved a 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) ten events involved the proper opening of a valve (manually) during in-plant testing with subsequent failure to reseat properly, and (d) one event involved the inadvertent actuation of the safety/relief valves used in the ADS during required surveillance testing.

The majority of the inadvertent blowdown events due to safety/relief valve malfunctions have been attributed to failures in a 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 failures of internal parts of the pilot valve section have been the causes of a lesser number of the valve failures.

2.1.2 Pressure Relief Valve Malfunctions

Four of the inadvertent blowdown events have resulted from a malfunction of a power-actuated pressure relief valve. 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 reseat properly. These failures have been attributed to mechanical damage to the relief valve internals (e.g., scored valve rings), buildup of foreign material on the 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 until April 1978.

None of these events resulted in a reactor coolant system overpressurization, and only one event resulted in a major degradation of the ADS (during this event the redundant High Pressure Coolant Injection System was ava'lable to protect against a small break LOCA.)

Sixteen of the twenty-seven operational events involved the failure of a power-actuated pressure relief valve to open properly. Fifteen of these failures occurred during in-plant valve testing. 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.

The remaining eleven of the twenty-seven operational events involved the failure of a dual function safety/relief valve to open properly. Six of these failures occurred during a pressure transient event (five in the self-actuating mode of operation); the remaining five failures occurred during in-plant testing. Failures which have occurred in the self-actuation mode of the valves (five events) have primarily been attributed to leakage in the bellows setpoint assembly of the first stage pilot valve. Failures which have occurred in the power-actuated mode of the valves (six events) 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). Of the twenty-seven operational events, twenty events involved failures of pressure relief system valves to open during in-plant testing (usually during reactor startup) and the remaining seven events involved the failure of a valve to open properly during a pressure transient. The failure of more than one valve during a single pressure transient event occurred only on one occasion when four of eleven safety/relief valves failed to open during a pressure transient following a reactor scram; however, the pressure rise peaked at 1120 psig and no system damage resulted.

2.3 Potential Failures to Open Properly

As shown in Table 4, there have been sixteen additional 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.

Twelve of the sixteen potential failure to open events involved leakage in the bellows setpoint assembly of the first stage pilot of a safety/relief valve. All but two of the bellows leakage failures occurred during plant operation and were detected by the bellows failure detection system. The remaining two bellows leakage failures were detected during bench testing of valves during a plant shutdown. Three of the four remaining potential failure to open events involved electrical problems (e.g., air supply solenoid electrically grounded). The fourth event involved a degradation in the air/nitrogen actuator diaphragm.

2.4 Experience Summary

Table 5 summarizes the experience with safety/relief valves and power-actuated pressure relief valves.

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

Safety considerations related to inadvertent pressure relief valve blowdown events include: (1) the imposition of 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 SRV blowdown event, the reactor coolant temperature will drop due to one or more of the following: (1) the release of energy through the open SRV, (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 cooldown at a rate determined mainly by its mass and the magnitude of the reactor coolant thermal transient. Since only the vessel inside wall is directly exposed to the reactor coolant, the temperature of the outer wall will lag

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

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 are 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 significantly thinner and less massive than the vessel wall, so their temperature more closely follows the coolant temperature. Since the vessel internals do not develop a significant differential temperature during thermal transients and since they are designed to withstand more severe thermal shocks than that which results from the temperature transient caused by an SRV blowdown event,* the thermal stresses developed in the vessel internals during an inadvertent SRV 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.

*The reactor vessel internals are designed to withstand the thermal shock resulting from the blowdown during a LOCA and subsequent reflood with cold (70°F) ECCS water (main condensate tank or suppression pool).

**Fatigue can be visualized as being analogous to the bending of a nail back and forth. If the bending is repeated often enough, the nail will eventually break due to fatigue. If the nail were only bent slightly each time; i.e., if the stresses in the nail were kept low, it would tolerate many more such cycles before it breaks than if it were bent 90° each time. The same principle applies to the reactor vessel because of the various stress cycles it will experience in its lifetime.

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 typical various types of stress cycles, utilized in the design of the reactor vessel for a typical 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 relief valve or safety valve blowdown events were postulated. For newer BWR vessels, about eight such events have been postulated to occur over the life of the vessel.

As discussed in Section 2.1, inadvertent SRV 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 SRV 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 specific 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 is 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 their 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 summated 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 particular 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 SRV 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 or heatup limit of 100°F/hour, the thermal stresses are well within allowable limits. For many of the reported inadvertent SRV blowdown events, either the cooldown or heatup rate was within the technical specification limit 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 SRV 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 event was determined to be less than 0.001. By comparison, the usage factor per SRV 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 usage 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 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 is routed through piping from the drywell to the suppression pool (Figure 5). There, the steam is condensed and the energy is absorbed by the heat capacity of the suppression pool. Prior to SRV actuation, the SRV discharge piping between the SRV and the suppression pool is filled with air and the SRV discharge piping below the surface of the suppression pool is filled with water.

During an SRV 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 discharge, 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) walls. The momentum of the displaced pool water causes the air bubble to over expand and subsequently collapse, causing a negative pressure load on the torus walls. The steam subsequently discharged into the suppression pool causes low amplitude pressure oscillations on the torus walls, which continue for the remainder of the blowdown event. Pressure loads on the torus walls are transmitted through the structure to the torus supports and

to piping attached to the torus. In addition to the pressure loads on the torus boundary, flow through the discharge lines create reaction forces on the 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 "threshhold temperature", if the discharge continues the steam condensation process would become unstable and the pressure oscillations could increase by a factor of ten or more. This effect is referred to as the steam quenching vibration phenomenon. The threshhold temperature for this phenomenon is primarily a function of the discharge flow rate and is considered to occur where the bulk pool temperature is on the order of 150°F to 170°F.

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 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.

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 SRV discharge device, while domestic facilities typically utilize a split (i.e., ramshead) discharge device. The ramshead device is considered to have a relatively higher threshhold temperature. At least one SRV 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 boundary.

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

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

has become an integral part of the 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 SRV loads. For the operating facilities, the SRV related tasks of the Mark I containment Long Term Program are intended to improve the quantification of SRV loads, to confirm the suppression chamber structural margins, and to confirm the adequacy of the suppression pool temperature limits.

The staff believes that there is no immediate (i.e., short term) potential hazard from the vibratory loads associated with SRV 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 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 SRV operations 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 SRV 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 any necessary modifications to be instituted before 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.

3.3 Radiological Considerations

Depressurization of the reactor system via the steam relief valves transfers steam from the reactor vessel to the torus. Since the steam potentially 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 polisher 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 within the reactor technical specifications.

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

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

The safety consideration 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 Safety/Relief Valves

Since the self-actuation mode of a safety/relief valve provides the automatic overpressure protection for facilities in which such valves are installed, only failures to open in the self-actuation mode of operation have been examined.

As discussed in Section 2.2, operating experience includes five safety/relief valve failure to open events during pressure transients where the failure occurred in the self-actuating 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).* Therefore, operating

*The ASME Boiler and Pressure Vessel Code requires that the reactor coolant pressure boundary design pressure should not be exceeded by 110%.

experience to date indicates that these valves are very reliable with respect to overpressurization protection.

The majority of the safety/relief valve failure or potential failure to open events (self-actuating mode) have been attributed to bellows assembly leakage. Since the bellows assembly of each safety/relief valve is continuously monitored for leakage, the likelihood of such failures occurring without being detected is low. Upon detection of a leaking safety/relief valve bellows assembly, appropriate operator action is required; however, since an additional safety/relief valve (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 safety/relief valves are required to be bench-tested to verify their operability. The percentage of valves that are required to be tested ranges from 33% 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.

In summary, based on the demonstrated performance and established surveillance programs for safety/relief valves, the staff has determined that they provide reliable overpressure protection for the reactor coolant pressure boundary.

4.1.2 Power-Actuated Relief Valves

With respect to the power-actuated relief valve failure to open events, it should be noted that the relieving capacity of these valves is not considered in the course of the overpressure analysis for the facility. The design of the pressure relief valves is based on establishing an adequate pressure relieving capacity to avoid opening of safety valves during normal operational events.

Operating experience indicates that in only one case have more than two power-actuated relief valves been in a condition where they would have failed to open on demand at a given time. Even in this case, it is not likely that safety valve actuation would have been required in the event of a pressure transient.

In the event that safety valve actuation would be required to avoid an overpressurization event, the radiological consequences of such actuation are not significant.

When blowdown occurs using the safety valves, the steam is transferred to the drywell and not the torus. The amount

of radioactivity transferred to the drywell is similar to that transferred to the torus with the 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 similar to the relief valve blowdown transient.

There would also be increased airborne concentrations of noble gases and some radioiodines within the drywell. Even though the drywell airborne 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 lide 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 procedural guidance given in technical specifications on radioactive effluent releases provides adequate assurance that blowdown via the steam safety valves would not have any impact on the environment appreciable different than that encountered during normal reactor shutdown.

4.2 Automatic Depressurization System Operability Considerations

4.2.1 Safety/Relief Valves

A specified number of safety/relief valves may be utilized in the power-actuated mode of operation as part of the facility's ADS (e.g., seven of eleven safety/relief valves installed at a particular facility are utilized for ADS). Consequently, only failures to open in the power-actuated mode of safety/relief valve operation need be considered.

As discussed in Section 2.2, operating experience includes six events where safety/relief valves have failed to open in the power-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 of the ADS, the redundant High Pressure Coolant Injection (HPCI) System was available to provide protection in the event of a postulated small break LOCA.

As discussed in Section 5, upon learning of this event the NRC took positive action to inform other licensees via an I&E

circular to ensure that they inspected their facilities to design to assure that a similar condition did not exist. As a result of this NRC action, similar problems with the diaphragms of safety/relief valves were detected at another facility. This second occurrence involved 9 of the 11 safety/relief valves and indicated a deterioration (delamination) of the diaphragm; however, the deterioration had not yet progressed to the point where the valves would have failed to function. In this case the degraded diaphragms were all replaced. The redundant HPCI system was also available for operation at this facility.

As a result of this experience, it was determined that the facility Technical Specification for operating BWRs required revisions to ensure that positive indication of ADS valve operation (e.g., turbine control valve movement) is obtained during the periodic surveillance testing of the ADS valves.

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

4.2.2 Power-Actuated Relief Valves

As discussed in Section 2.2, operating experience includes a number of events where power-actuated relief valves 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.

Other failures of power-actuated relief valves 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 redundant HPCI system was operable.

5.0 CORRECTIVE ACTIONS

For some time licensees, manufacturers, and vendors of boiling water reactors have been working with the NRC to improve 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 safety/relief valves, the General Electric Company, working with valve suppliers, has proposed a two-staged program to further improve reliability of the safety relief valves. A short term program has been initiated which involves minor modifications to the existing 3-stage safety/relief valve design, and increased monitoring to detect valve leakage prior to inadvertent actuation has been initiated. Since almost all the inadvertent blowdown events caused by safety/relief valve 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 the

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*Simmer margin is the differential pressure between normal system operating pressure and the safety relief valve setpoint.

SRV setpoint pressure for their valves. In some cases this required a grinding out of the valve "throat" diameters of the discharge of the valves in order to accommodate the decrease flow rate at increased pressure.

The long term program is the development and installation of an improved design valve actuator for Target Rock for safety relief valves. The most significant changes incorporated in this new design is the elimination of one of the pilot stages and the elimination of the bellows assembly. Therefore, the new design is expected to improve the performance of the safety/relief valves by eliminating valve failures to open due to bellows leakage, and eliminating spurious valve openings due to leakage past the first stage pilot set. This second improvement is accommodated by precluding the leakage from building upper pressure on the actuating mechanism of the main valve.

Valves of this new design have undergone a testing program and is currently under review by the NRC staff. The NRC also has under review licensing applications to utilize valves with this new type on two facilities. It is anticipated that licensees of other operating facilities will shift to this new design as approppriate in the future.

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

In addition, the NRC staff has been considering revising operating 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 relief and safety/relief valves that failed to open on demand since the last testing period.

(2) Increased surveillance of the structural integrity of relief and/or safety/relief valve line restraints in the pressure suppression pool based on the frequency of relief pressure relief valve actuation.

(3) Increased refurbishing and bench testing requirements for relief and safety/relief valves that would vary as a function of the number of valves that are spuriously opened or opened and failed to reset properly during a specified period.

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

With respect to power-actuated relief valves, more attention has recently been given to adverse valve experience. In several cases licensees have replaced power-actuated relief valves with safety/relief valves and in general, newer plants utilize the dual function safety/

relief valves as opposed to the relief valves. Consequently, they have shown generally good performance and reliability.

6.0 CONCLUSION

As discussed in this report, the staff has concluded that the events that have occurred to date have not resulted in significant impact on the nuclear facility and have had no offsite consequences. Further, the staff has generally concluded that even if such events were to continue they would probably have no noticable effect on the reactor pressure vessel.

In spite of the minor nature of such events, the staff generally believes that measures should be undertaken to reduce the frequency of such events because they represent a deviation from the original design envelope for these plants and present generally undesirable situation.

TABLE _

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

*Power Actuated Relief Valve

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	1*									
Oyster Creek					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	
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	0

*Power Actuated Relief Valve

TABLE 5

Event	Safety Relief Valves	Power Actuated Pressure Relief Valves
Inadvertent Blowdowns	47	4
Failures to Open	11	16
Potential Failures to Open		

TABLE 6

Typical 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 tc 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

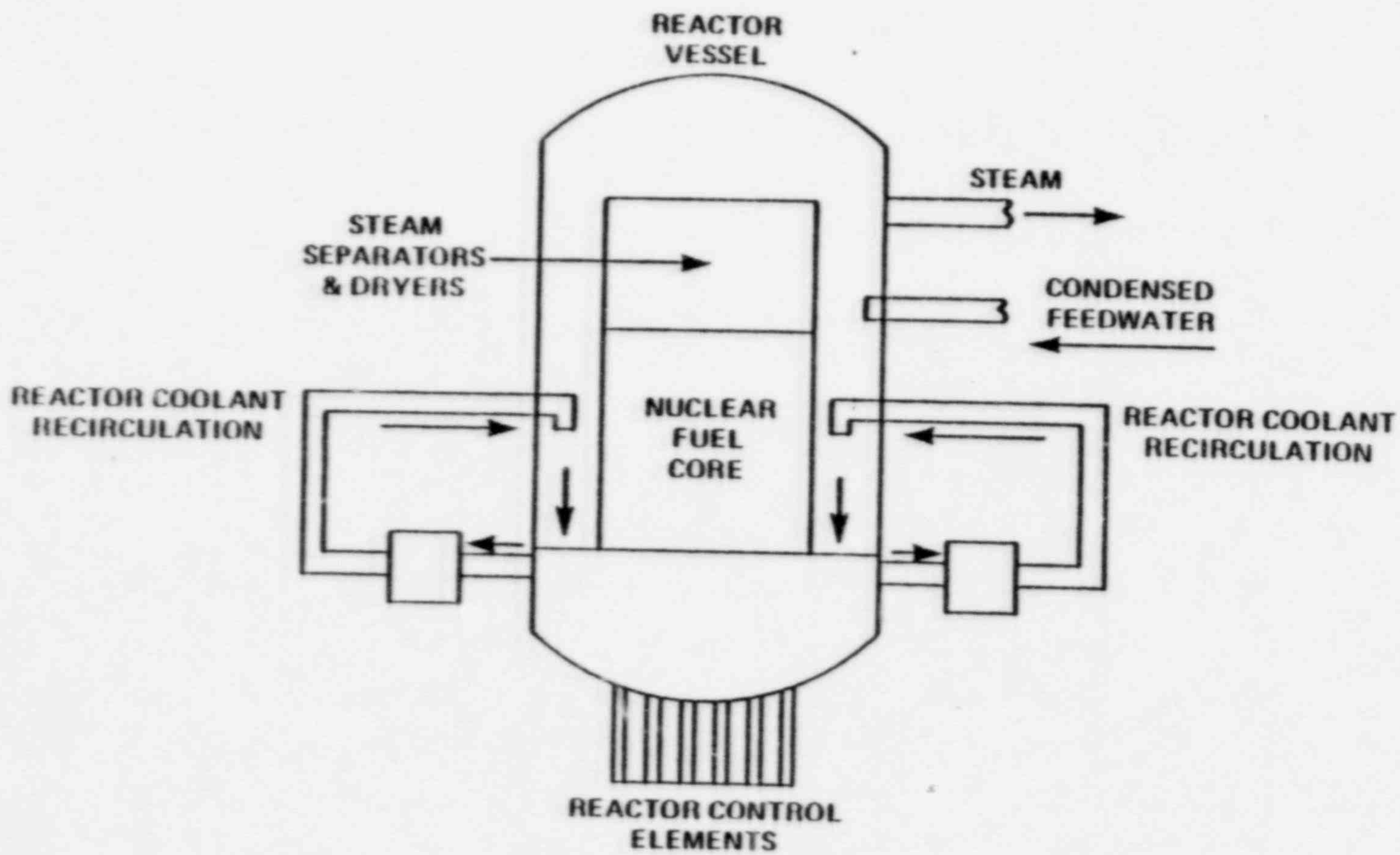

Figure 1. Schematic of a Boiling Water Reactor

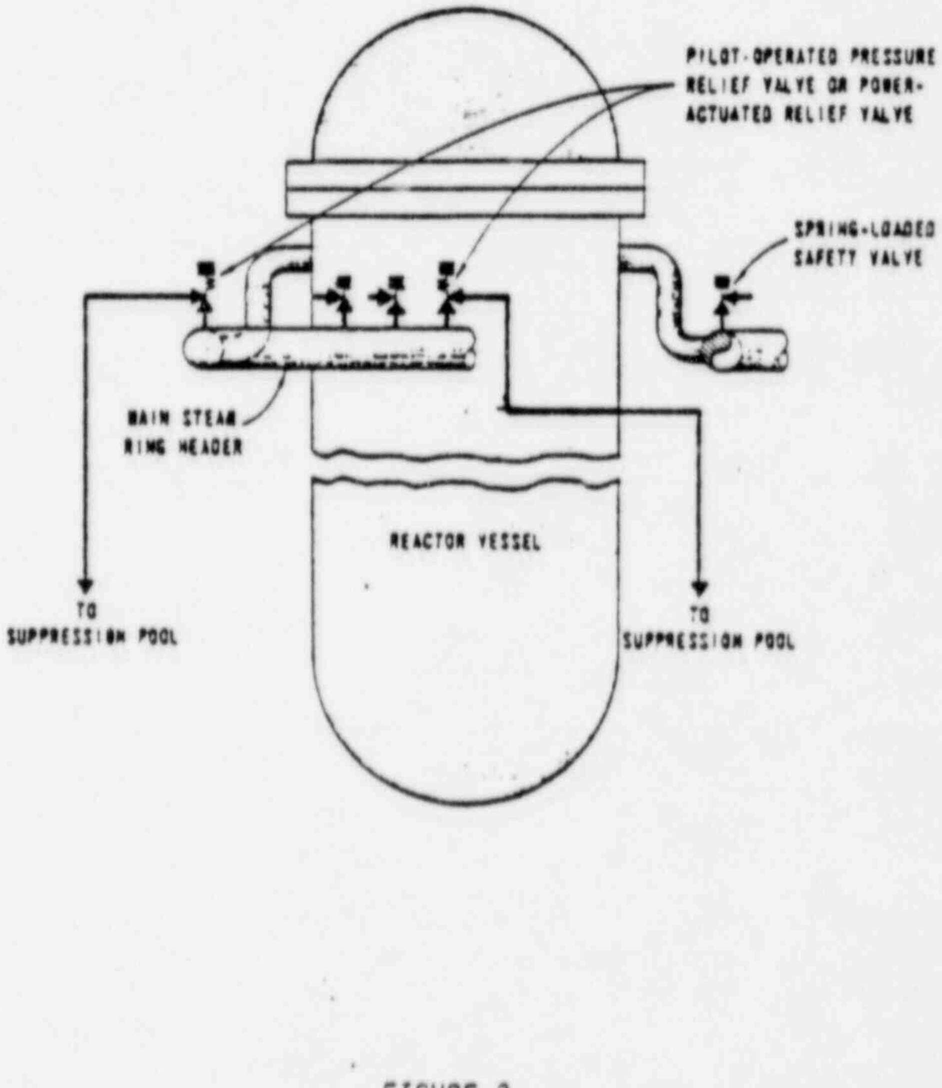

FIGURE 2

PHYSICAL LOCATION OF BWR SAFETY VALVES
AND RELIEF VALVES

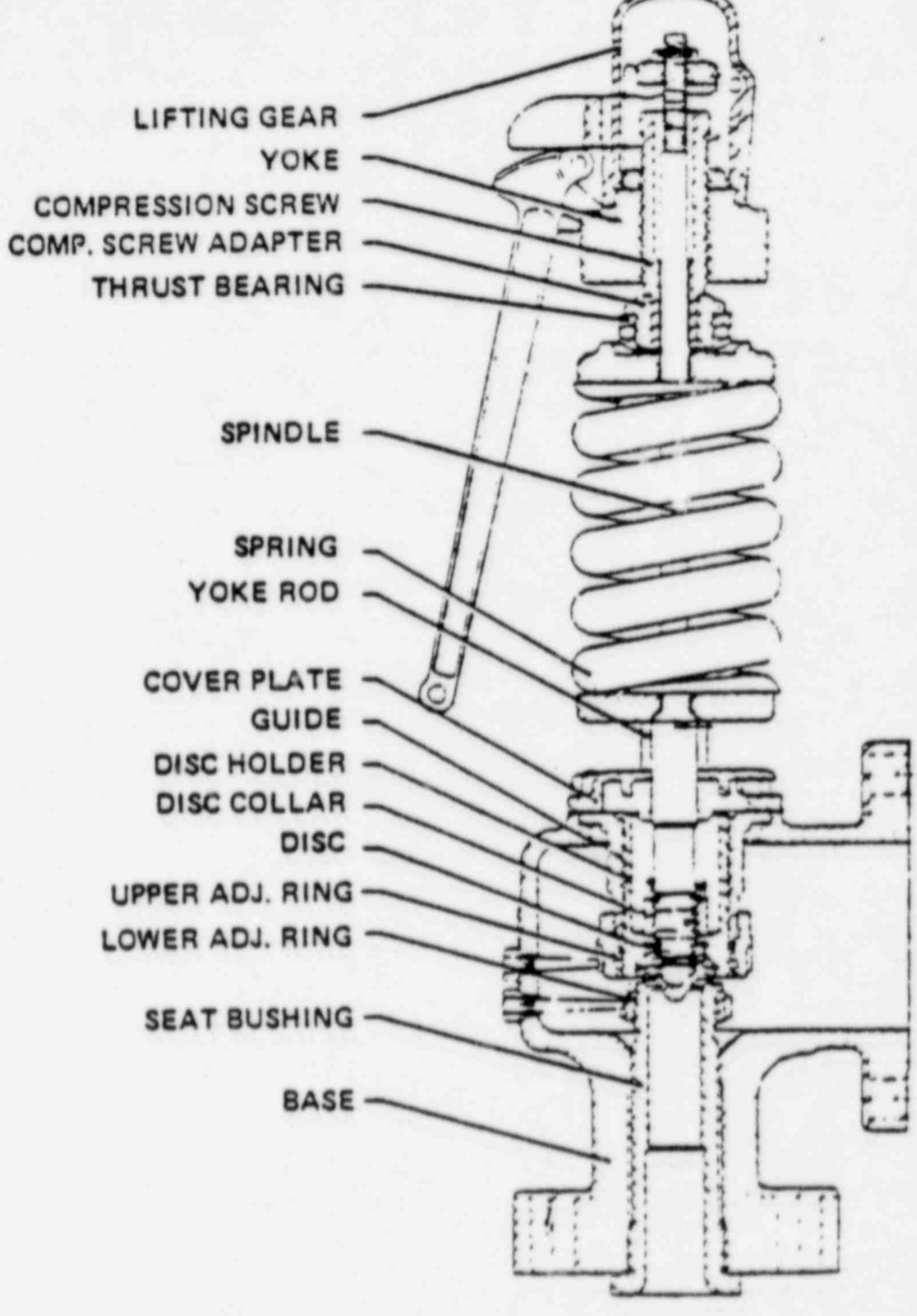

FIGURE 3
SAFETY VALVE

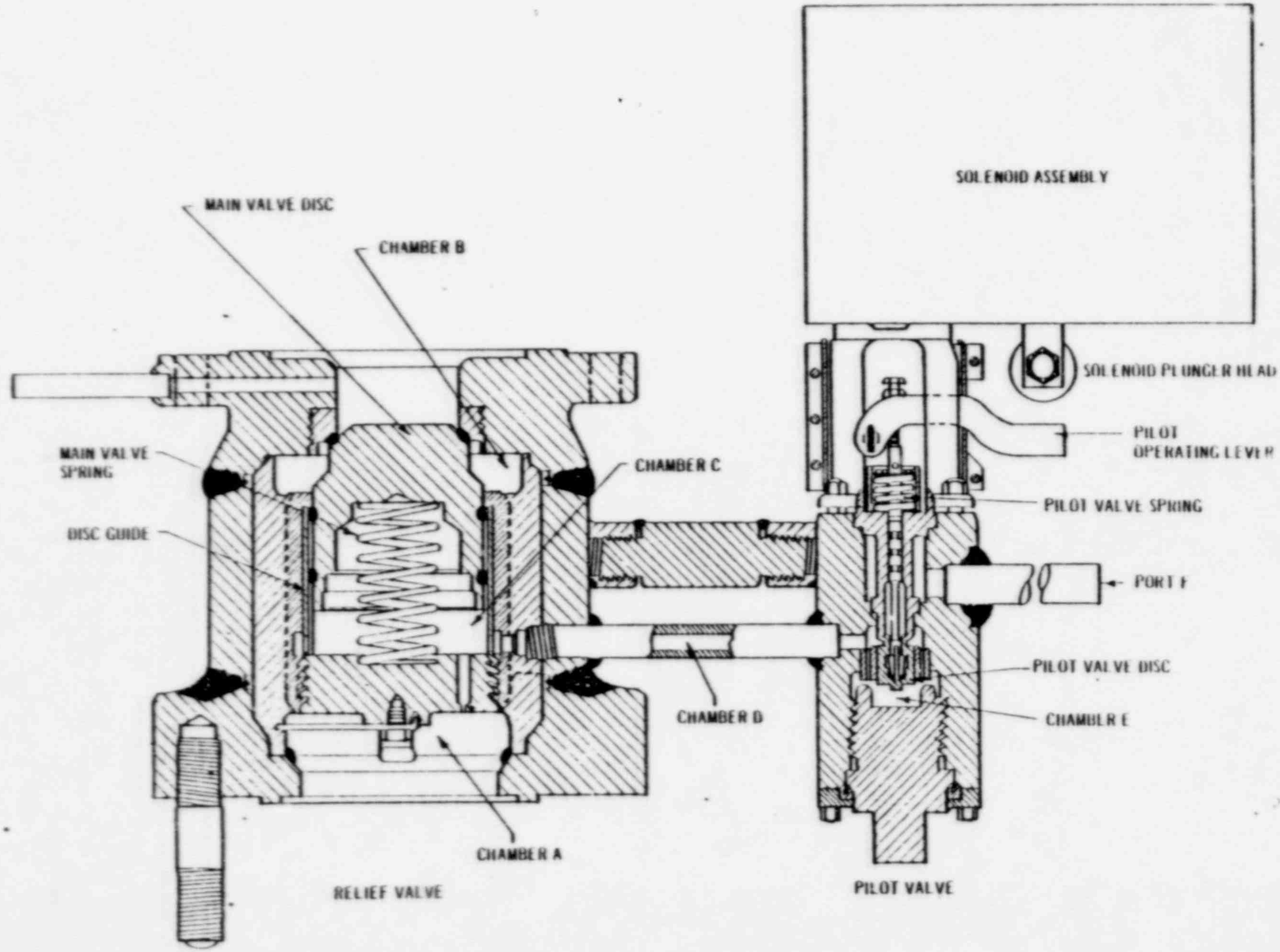

FIGURE 4

POWER ACTUATED RELIEF VALVE

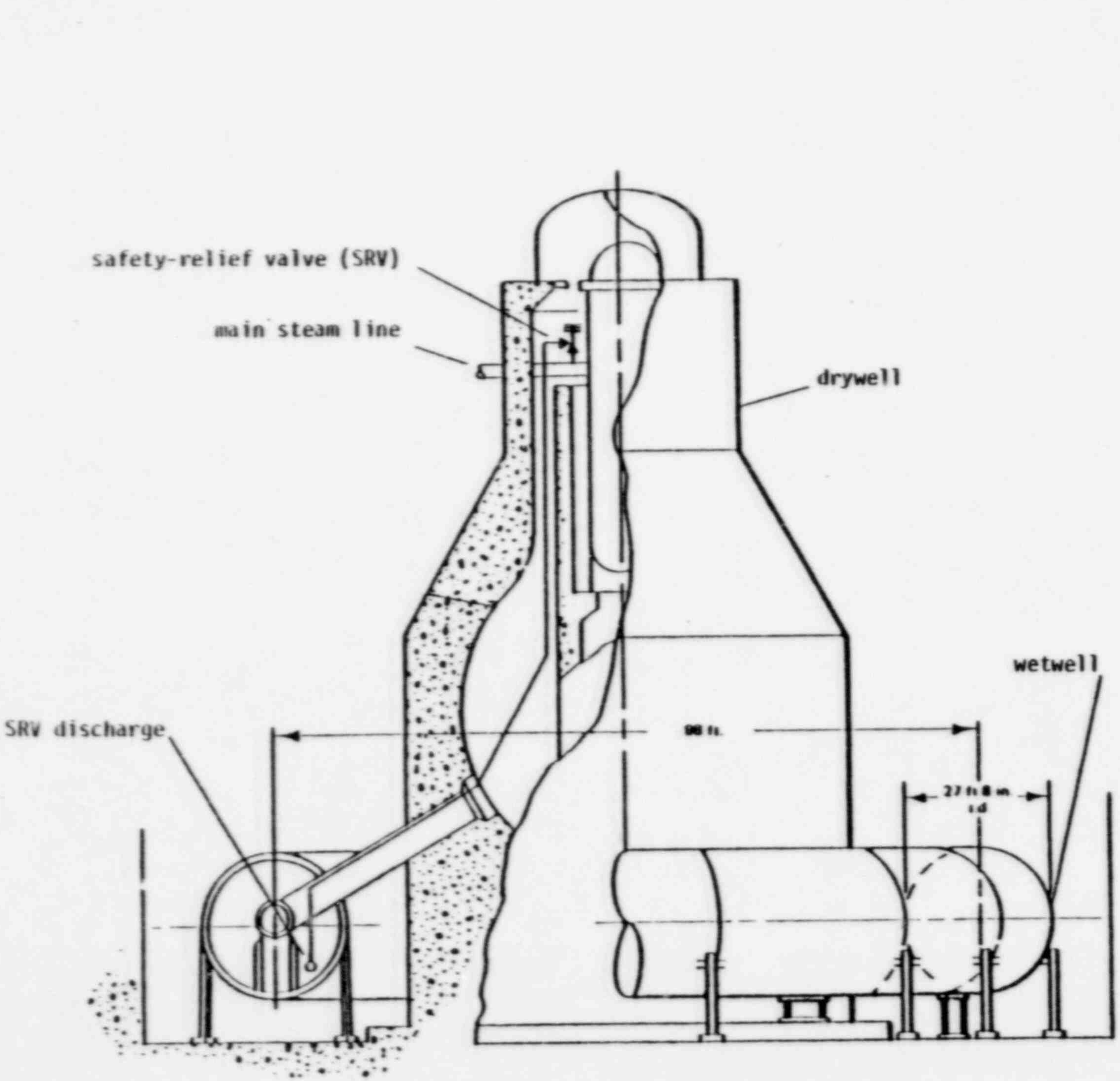

FIGURE 5

SAFETY/RELIEF VALVE DISCHARGE FLOW PATH

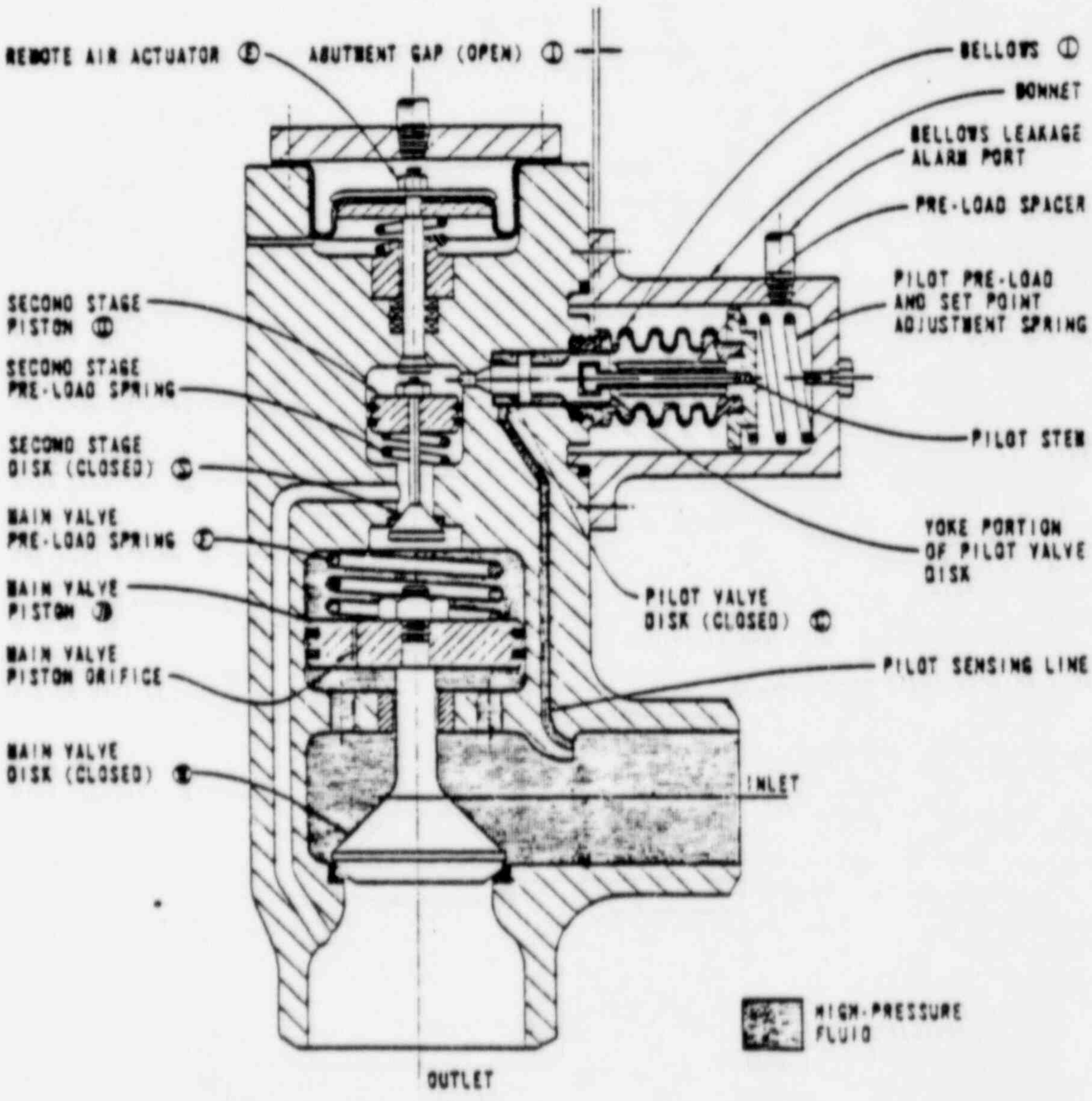

FIGURE 6 - Operational Diagram of Three-Stage Pilot-Operated Pressure Relief Valve in Closed Position.

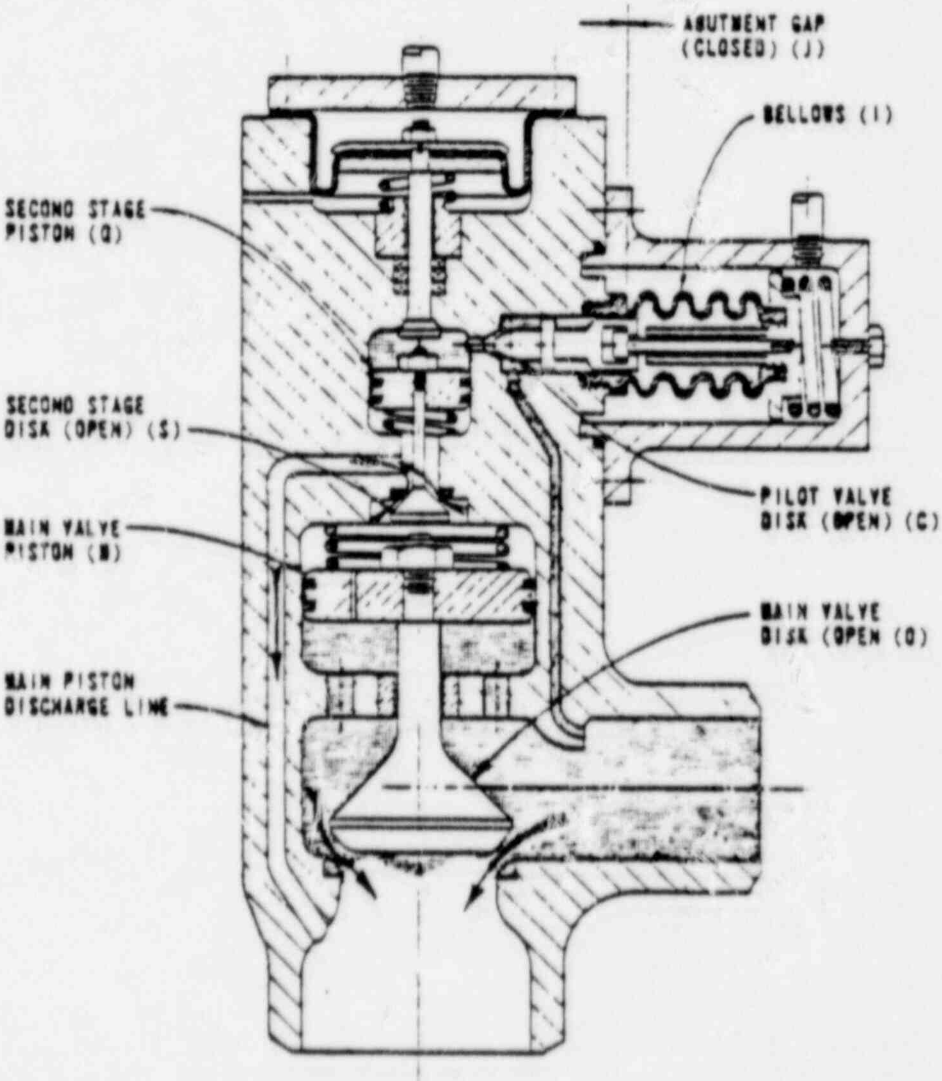

FIGURE 7 - Operational Diagram of Three-Stage Pilot-Operated Pressure Relief Valve in Open Position.

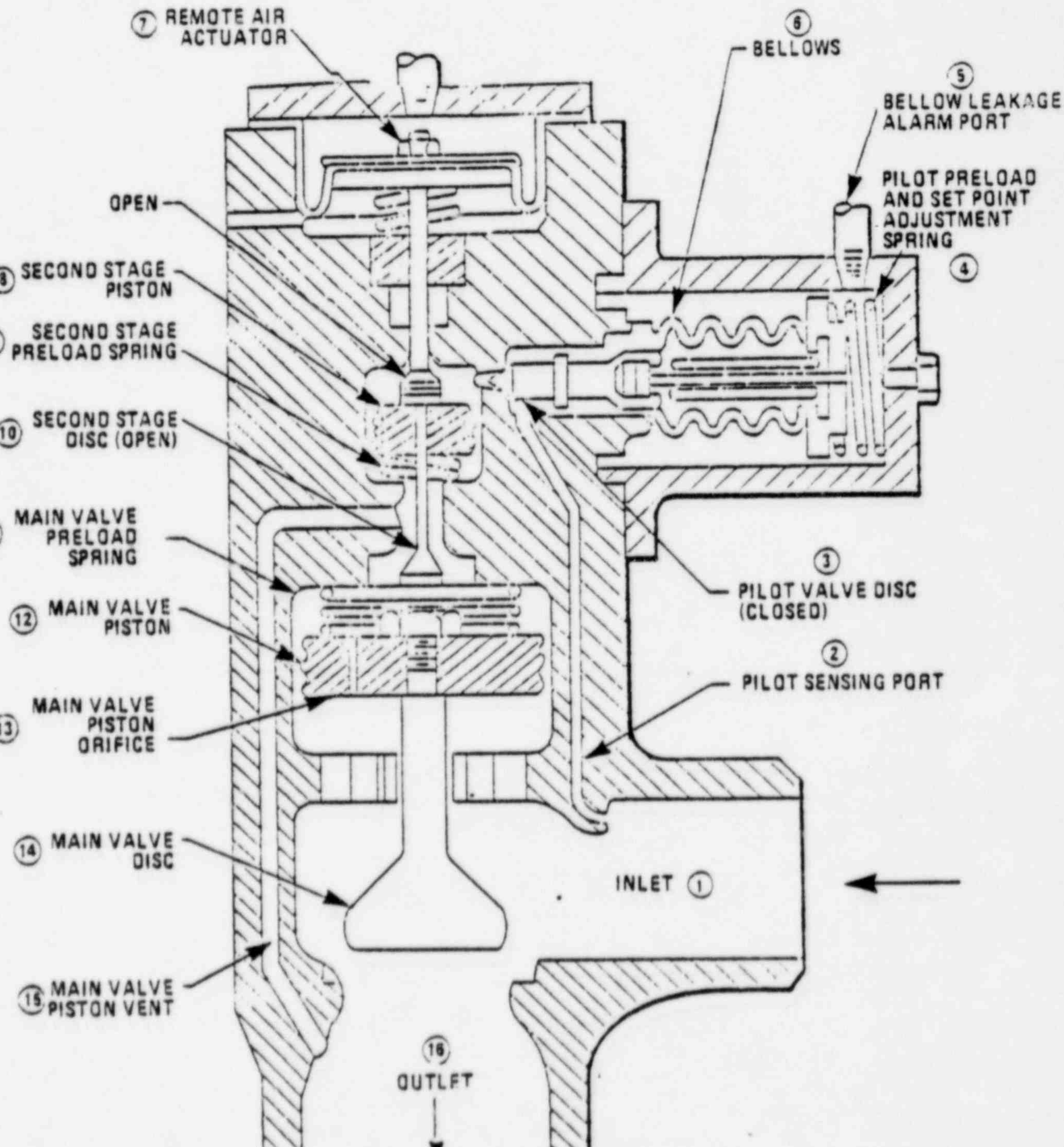

FIGURE 8 Safety-Relief Valve (External Actuation)

F-C-r
19

NRC–17
(01-75)

U. S. NUCLEAR REGULATORY COMMISSION
ROUTING SLIP

Organization

TO	NAME – MAIL STOP	
R. Tedesco		
cc: G. Lainas		
~~J. Kudrick~~		
C. Anderson		
SUBJECT: SRV ACTUATION FOLLOWING DBA		

REMARKS:

Pursuant to your question regarding SRV actuation following DBA, I have had discussions with Bert Sobon. The following is my understanding about this subject.

SRV actuation following DBA is not a credible event because the time required for MSIV closure will allow the reactor to be depressurized below the SRV's setpoints. However, if MSIV closure at time zero is postulated, which is an unrealistic assumption, GE calculations indicate the opening of the two groups of low-set valves (10 for GESSAR 238 plant) 4 to 12 seconds after the DBA. Therefore, it can be concluded that SRV blowdown will not occur simultaneously with the DBA.

Description of the event is provided in Section A.8.3.2 of GESSAR Appendix 3B.

- ☐ As Requested
- ☐ Correction
- ☐ File
- ☐ Information
- ☐ See Me
- ☐ Approval/Signature
- ☐ Comment/Concurrence
- ☐ Necessary Action
- ☐ Note and Return
- ☐ Per Conversation
- ☐ Answer/Acknowledge for

Signature of ____________

FROM (Name)	OFFICE	DATE 6/15/78
T. M. Su	CSB/DSS	PHONE 27711

D-115