Commonwealth Edison
1400 Opus Place
Downers Grove, Illinois 60515

September 13, 1990

Mr. A. Bert Davis
Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, IL 60137

Subject: Dresden Station Unit 2
Response to Confirmatory Action Letter
CAL-RIII-90-014
NRC Docket No. 50-237

References: (a) Confirmatory Action Letter (CAL-RIII-90-014) from A.B. Davis (NRC) to C. Reed (CECo), dated August 3, 1990.

(b) E. Eenigenburg (CECo) letter to U.S. Nuclear Regulatory Commission, dated August 28, 1990, transmitting Licensee Event Report 90-006-0.

Mr. Davis:

On August 2, 1990, Dresden Unit 2 was manually scrammed following the spurious opening of the Main Steam Target Rock Safety Relief Valve (TRSRV). Following the reactor scram, the suppression chamber bulk water temperature increased to 122 degrees F, and the reactor coolant cooldown rate exceeded 100 degrees F per hour. On August 3, 1990, a drywell entry discovered an electrical junction box related to the TRSRV circuitry which had fallen from its original mounting, and conduit which had separated from the TRSRV bellows seal pressure switch. As a result of this event, Reference (a) requested Commonwealth Edison Company (CECo) to provide the results of evaluations being performed to determine: (1) the impact and consequences of the cooldown rate that the reactor coolant system experienced; (2) the impact and consequences of the temperature reached in the torus (suppression chamber); (3) the root cause for the condition of the electrical equipment (junction box and TRSRV bellows seal pressure switch conduit); and (4) the root cause for the opening of the TRSRV. Attachment 'A' provides CECo's response to that request.

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Please direct any questions or comments on this response to this office.

Respectfully,

Milton H. Richter

M.H. Richter
Nuclear Licensing Administrator

Attachment: A - Response to Confirmatory Action Letter (CAL-RIII-90-014)

Figures: 1 - Dresden Unit 2 Reactor Pressure - TRSRV Blowdown Event
2 - Target Rock Safety Relief Valve
3 - Target Rock Safety Relief Valve Controls

cc: NRR Document Control Desk
B. Siegel - NRR Project Manager
S. DuPont - Senior Resident Inspector, Dresden

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ATTACHMENT A

RESPONSE TO CONFIRMATORY ACTION LETTER (CAL-RIII-90-014)

BACKGROUND

On August 2, 1990, with Unit 2 operating at approximately 87% power, the 2-203-3A Main Steam Target Rock Safety Relief Valve (TRSRV) acoustic monitor alarmed, indicating that the valve had spuriously opened and was relieving reactor pressure to the suppression chamber. After confirming that the TRSRV had spuriously opened, attempts were made in accordance with Station procedures to close the valve; however, the TRSRV would not close. As suppression chamber temperature approached 110 degrees F, the reactor was manually scrammed as required by the Technical Specifications. All Containment Cooling Service Water (CCSW) and Low Pressure Coolant Injection (LPCI) pumps were manually started for maximum suppression chamber cooling. The maximum average cooldown rate when averaged over a one hour period was 129.3 degrees F/hr (using recirculation loop temperature data), and the maximum bulk suppression chamber temperature was 122 degrees F.

Subsequent to the reactor scram, a primary containment drywell entry was made to inspect the TRSRV on August 3, 1990. Upon inspection, it was discovered that an electrical junction box (2PB-2020) related to the valve circuitry, which had been attached to junction box 2PB-2021, had fallen off and was resting on piping below its original mounting. All wires/cables were still connected, and the electrical control of the valve was unaffected. Additionally, the TRSRV Bellows Seal pressure switch was found to be separated from its conduit; however, the pressure switch was still functional.

As a result of this event, Confirmatory Action Letter CAL-RIII-90-014 was issued to Commonwealth Edison Company which contained several specific requests. The response to those requests are provided below.

Request:

Provide the results of the analysis of the effect of the cooldown rate that the reactor primary system experienced as a result of the thermal transient conditions experienced during this event. The analysis should discuss any long term consequences of the temperature transient and its duration.

Response:

Utilizing recirculation loop temperature data, the maximum cooldown rate of the reactor vessel bulk coolant, when averaged over a one hour period, was 129.3 degrees F/hr. General Electric Company performed a reactor vessel bulk coolant temperature cooldown rate evaluation to verify that the cooldown rate was within design limits of a Safety Relief Valve Blowdown (SRVB) event. The SRVB design basis event was developed to bound actual single valve blowdowns that would occur at operating plants. In the design basis blowdown event, the cooldown proceeds from 546 degrees F to 375 degrees F in the first ten minutes, which corresponds to a rate of 1026 degrees F/hr. The cooldown rate then proceeds at 100 degrees F/hr to shutdown

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conditions. Taking these two cooldown rates into consideration, this averages to be a cooldown rate of 254.3 degrees F/hr when averaged over a one hour period. For this TRSRV event, General Electric used a conservative maximum average cooldown rate of 149 degrees F/hr (when averaged over a one hour period) based on saturated reactor steam pressures. This cooldown rate is significantly less than the 254.3 degrees F/hr cooldown rate (when averaged over a one hour period) used in the SRVB design basis analysis previously performed by General Electric (refer to Figure 1).

A review was performed to ensure the SRVB design allowances discussed in Dresden's Final Safety Analysis Report (FSAR) and General Electric's SRVB design analysis were not exceeded. The review revealed that twelve (12) SRVB events are allowed for forty (40) years of operation for Dresden Unit 2. There have been two (2) SRVB events for Dresden Unit 2 prior to this event; a TRSRV opening event occurred in 1976, and a Main Steam Safety Valve opening event occurred in 1970. Based on this information, Dresden Unit 2 is well within the SRVB allowables. Consequently, it has been concluded that this blowdown event is bounded by the SRVB design basis analysis and that no detrimental long term effects to the plant exist.

Request:

Provide the results of the analysis of the high temperature reached in the suppression chamber (torus). The analysis should discuss any long term consequences of the temperature transient and its duration.

Response:

The maximum bulk suppression chamber temperature attained during this event was 122 degrees F. The key structures affected by the magnitude of the temperature experienced are the TRSRV discharge line, associated supports and the suppression chamber in the area of the outlet of the TRSRV discharge line.

The transient experienced with the TRSRV spuriously opening and remaining open at power is bounded by the Mark I design basis analysis. Since the TRSRV remained open and did not cycle, the only load experienced on the discharge line resulted from the initial actuation. This loading combination has been reviewed with Sargent & Lundy (the architect engineer for Dresden Station) and is well within allowable stresses for the piping and supports. Due to the design margins of the actuation transient and the negligible magnitude of the steady-state discharge loads, fatigue is not a concern. A post transient visual inspection of the TRSRV discharge line and supports within the drywell verified that no damage had occurred.

All of the safety relief valve discharge lines are equipped with T-Quenchers at their outlet (within the suppression chamber) to mitigate concerns regarding air bubble loads and condensation stability of the original rams-head discharge devices. Condensation stability of the T-Quencher configuration has been demonstrated, provided that local temperatures remain below 200-204 degrees F (based on mass flux out of the T-Quencher). General Electric performed an analysis of suppression chamber heatup event cases

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(NEDC-22170, July 1982) for Dresden Station demonstrating that the maximum temperature achieved with a stuck open relief valve and a single suppression chamber cooling loop would be 131 degrees F, with a local to bulk temperature difference of 30 degrees F. Other test data has shown that the local to bulk temperature difference with two loops of suppression chamber cooling would not exceed approximately 38 degrees F. Since the maximum bulk suppression chamber temperature only reached 122 degrees F, and both loops of suppression chamber cooling were utilized during the event, the local temperature in the vicinity of the T-Quencher can be estimated to be approximately 160 degrees F (based on General Electric's analysis of suppression chamber heatup event cases for Dresden). This temperature is well within the condensation stability limits for the T-Quencher. Also, the bases for Technical Specification 3.7.A., Primary Containment, discuss that excessive steam condensing loads can be avoided if the peak bulk temperature of the suppression chamber is maintained below 160 degrees F during any period of relief valve operation with sonic conditions at the discharge exit. Due to the mixing of the water inside the suppression chamber by operation of the suppression pool cooling loops, the maximum suppression chamber bulk temperature of 122 degrees F attained during this event was well below this peak bulk temperature of 160 degrees F. Therefore, the loads on the suppression chamber during this event were negligible.

Consequently, the loads experienced in this event were well within the design margins for these systems/structures. Additionally, the temperatures realized in the event are bounded by prior analysis and condensation stability of the T-Quencher can be demonstrated. The low magnitude of the steady-state loads as well as the design margins for the transient valve actuation loads support the conclusion that no significant impact on the fatigue life of the plant has occurred, and long term effects from this event are minimal.

Request:

Provide a root cause analysis of the electrical equipment found in a degraded condition as observed on the August 3, 1990, drywell entry.

Response:

The cause of junction box 2PB-2020 detaching and falling from junction box 2PB-2021 is believed to be inadequate junction box support/attachment. All Environmental Qualified (EQ) criteria were evaluated and no discrepancies were noted. In addition, there were scratch marks on junction box 2PB-2021 from a snubber lightly rubbing against it. This snubber is one of several utilized on the TRSRV. It was inspected and showed no signs of damage. Numerous inspections of the junction box (2PB-2020) and surrounding area are documented at Dresden Station between 1985 and 1990. It is concluded that junction box 2PB-2020 came loose between January 23, 1990 and August 2, 1990. The exact time of separation is not known. It was determined that a single junction box would replace the existing 2PB-2020 and 2PB-2021 junction boxes. The replacement junction box was installed on a new

seismically designed support and mounting, which provided clearance from the snubber. The wiring contained within junction boxes 2PB-2020 and 2PB-2021 was installed in the replacement box and verified to be correct.

The cause of the TRSRV Bellows Seal pressure switch being detached from its respective conduit was attributed to inadequate support of the conduit and switch. A new support for the TRSRV Bellows Seal pressure switch assembly was installed.

A walkdown was conducted to inspect the mounting of electrical junction boxes throughout the Unit 2 drywell. All such accessible junction boxes were visually examined for damage and for potential concerns due to vibration. Two junction boxes were found to be supported only by their respective conduits. However, the conduits were solidly connected to the junction boxes and were adequately supported. These boxes only contain cable and cable splices, neither of which are shock-sensitive. Consequently, these two junction boxes were judged to be adequately supported although their long term acceptability is currently being confirmed. Any required resupporting will be performed during the upcoming Unit 2 refueling outage (currently scheduled to begin on September 23, 1990). The Unit 3 TRSRV junction boxes will be evaluated for structural adequacy during the next available outage of sufficient duration allowing drywell access.

Request:

Conduct a full investigation of the failed TRSRV for a root cause analysis including a review of the maintenance history on this valve.

Response:

The TRSRV operates through self-actuation (safety mode) at 1135 psig reactor pressure, or through remote actuation of a solenoid valve which admits a pneumatic supply to an air operator. This remote actuation may occur from the following sources:

1. Remote manual switch in the Control Room.

2. High reactor pressure (1124 psig) from a pressure controller.

3. Initiation of Automatic Depressurization System (ADS) logic.

Self-actuation occurs as follows (refer to Figure 2): Pressure is sensed at the pilot sensing port (2). The bellows (6) expands at a pressure setpoint of 1135 psig. This moves the pilot valve disc (3) allowing pressure to be transferred to the second stage piston (8). The second stage (8) is forced down moving the second stage disc (10) away from its seat. This permits the pressure from the top of the main valve piston (12) to be vented via the second stage disc (10) and out the main valve piston vent (15). A differential pressure is created across the main valve piston (12) due to the small size of the main valve piston orifice (13) as compared to the main valve

piston vent (15). Reactor pressure then lifts the main valve piston (12) and the main valve disc (14). The final result is that reactor steam is directed through the discharge line to the suppression chamber. When steam pressure is approximately 50 psig below the setpoint pressure of 1135 psig, the pilot preload and setpoint adjustment spring (4) forces the pilot valve (3) closed. The second stage disc (10) then closes equalizing the pressure across the main valve piston (12). Spring force from the main valve preload spring (11) closes the main valve disc (14). If the bellows (6) ruptures, a pressure switch (5) results in an alarm in the Control Room on the 902-4 panel (annunciator D-23, TRSRV Inoperable) at 25 psig. This indicates that the self-actuation mode of the TRSRV is inoperable.

Remote actuation occurs as follows (refer to Figures 2 and 3): A DC solenoid admits nitrogen pressure of approximately 85 psig to the remote air actuator (7) resulting in the stroking of the air plunger (17) which in turn pushes down the second stage piston (8). This permits the pressure from the top of the main valve piston (12) to be vented via the second stage disc (10) and out the main valve piston vent (15). A differential pressure is created across the main valve piston (12) due to the small size of the main valve piston orifice (13) as compared to the main valve piston vent (15). Reactor pressure then lifts the main valve piston (12) and the main valve disc (14). The final result is that reactor steam flow is directed through the discharge pipe to the suppression chamber. When the solenoid is de-energized, pressure is vented from the remote air actuator (7). The second stage disc (10) then closes equalizing the pressure across the main valve piston (12). Spring force from the main valve preload spring (11) closes the main valve disc (14).

The TRSRV valve controls are shown in Figure 3. The control switch is a three-position, key operated switch. The three positions are "MANUAL", "OFF" and "AUTO. In the "MANUAL" position, the solenoid is always energized. In the "OFF" position, the solenoid only energizes from an automatic depressurization signal; it does not energize from a relief signal sent from the controller. In the "AUTO" position, the solenoid energizes from either an automatic depressurization signal or a relief signal sent from the controller. The accumulator-check valve arrangement stores sufficient nitrogen to operate the TRSRV in the event of a loss of drywell pneumatic air (nitrogen). The pressure switches act as position indicators in the following manner: at less than 40 psig, the TRSRV indicates "Closed"; at greater than 50 psig, the TRSRV indicates "Open"; and at greater than 50 psig, a Control Room annunciator also indicates valve actuation. Additionally, an acoustic monitor in the drywell monitors the TRSRV discharge piping and alarms in the Control Room when there is pipe vibration due to steam flow.

The apparent cause of the failure of the TRSRV was steam cuts in the pilot valve disc (refer to Figure 2). Excessive steam leakage through the pilot sensing port (2) and past the pilot valve (3) via the severe steam cuts allowed pressure to be transferred to the second stage piston (8). The second stage piston (8) was forced down, moving the second stage disc (10) away from its seat. This permitted pressure from the top of the main valve piston (12) to be vented via the second stage disc (10) and out the main valve piston vent (15). This created a differential pressure across the main valve piston (12). Reactor pressure then lifted the main valve piston (12) and main valve disc (14), thus, opening the valve. When reactor pressure reached approximately 100 psig, the main valve preload spring (11) force overcame the reactor pressure force and the main valve disc (14) subsequently closed. The root cause for the steam cuts on the pilot valve disc is an inherent design deficiency.

The TRSRV indicator on the front panel showed a closed position due to the position indicators receiving their signals from the pressure switches on the drywell pneumatic air line between the solenoid and the TRSRV. The indicator/pressure switches performed satisfactorily since the TRSRV air operator was not actuated when the TRSRV opened. The air operator was not responsible for the valve failure.

A previous occurrence of a TRSRV spuriously opening due to excessive leakage of the first stage pilot valve was experienced at Dresden Station in 1976. The valve in that event had been in-service for more than one cycle. To prevent further occurrences, Dresden started rebuilding the TRSRV on each Unit every refuel outage. Although Dresden did not experience another TRSRV spuriously opening until this recent event, steam cuts and degradation of the first stage pilot valve discs and secondary stage valve discs have been noted. A maintenance history review for this valve was performed. This valve was installed on Unit 2 on February 11, 1989. This was the first time this particular pilot stage assembly (includes pilot and second stage valves) had been put into service on a unit. The pilot and secondary stage valve seats had been inspected, lapped, and satisfactorily leak tested prior to installation.

A satisfactorily leak tested, rebuilt TRSRV was installed in Unit 2 prior to start-up. Replacement of the TRSRV 203-3A Pilot Assembly has been placed on the short outage list for each unit (Units 2 and 3). Instructions are to replace any TRSRV pilot assembly that has been in-service for a duration greater than eight (8) months into the operating cycle if the Unit is placed in Cold Shutdown and the drywell is accessible during the short outage.

FIGURE 1

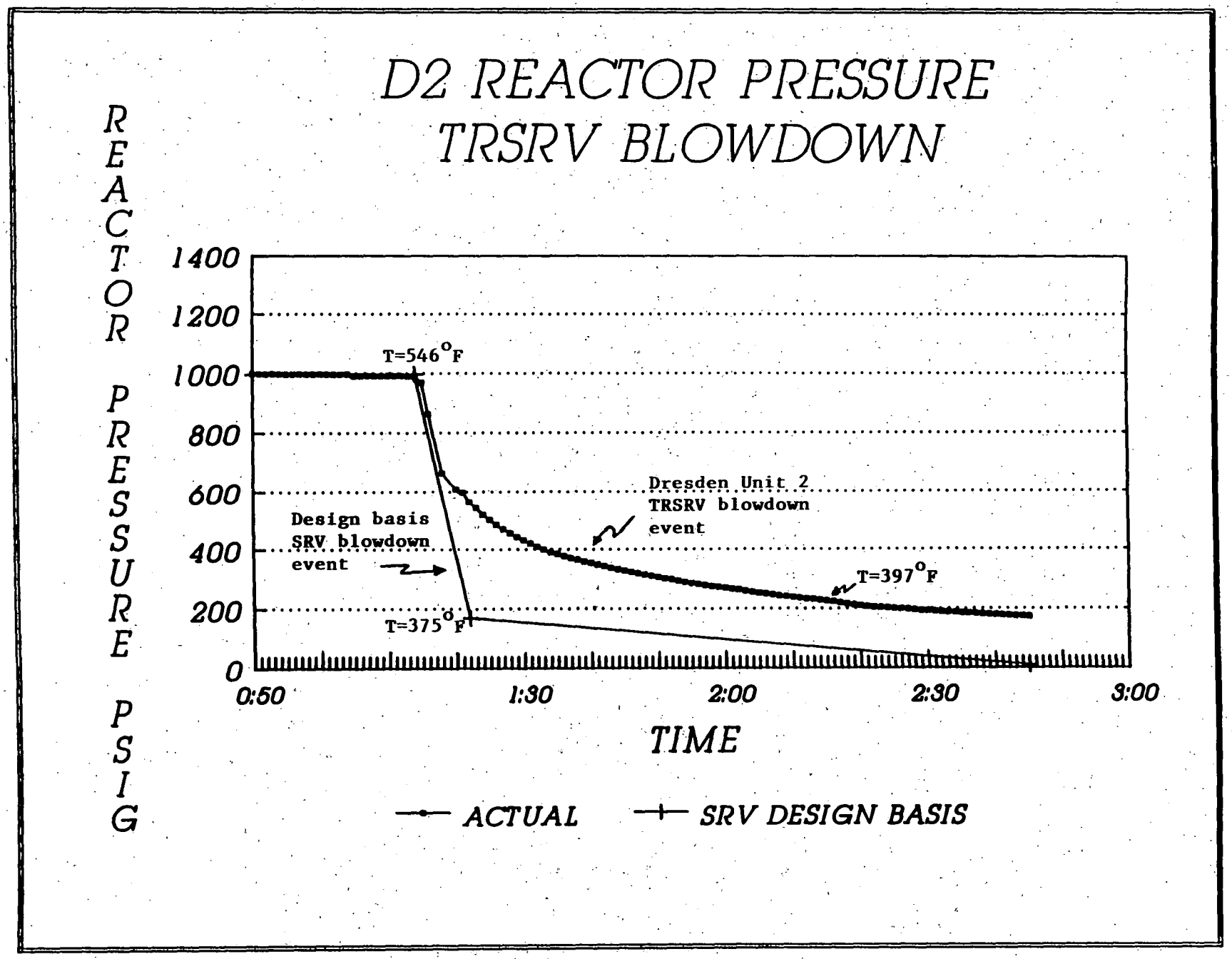
D2 REACTOR PRESSURE
TRSRV BLOWDOWN
REACTOR PRESSURE PSIG
1400
1200
1000
800
600
400
200
0
T=546°F
Design basis
SRV blowdown
event
Dresden Unit 2
TRSRV blowdown
event
T=397°F
T=375°F
0:50
1:30
2:00
2:30
3:00
TIME
ACTUAL
SRV DESIGN BASIS

FIGURE 2

TARGET ROCK SAFETY RELIEF VALVE

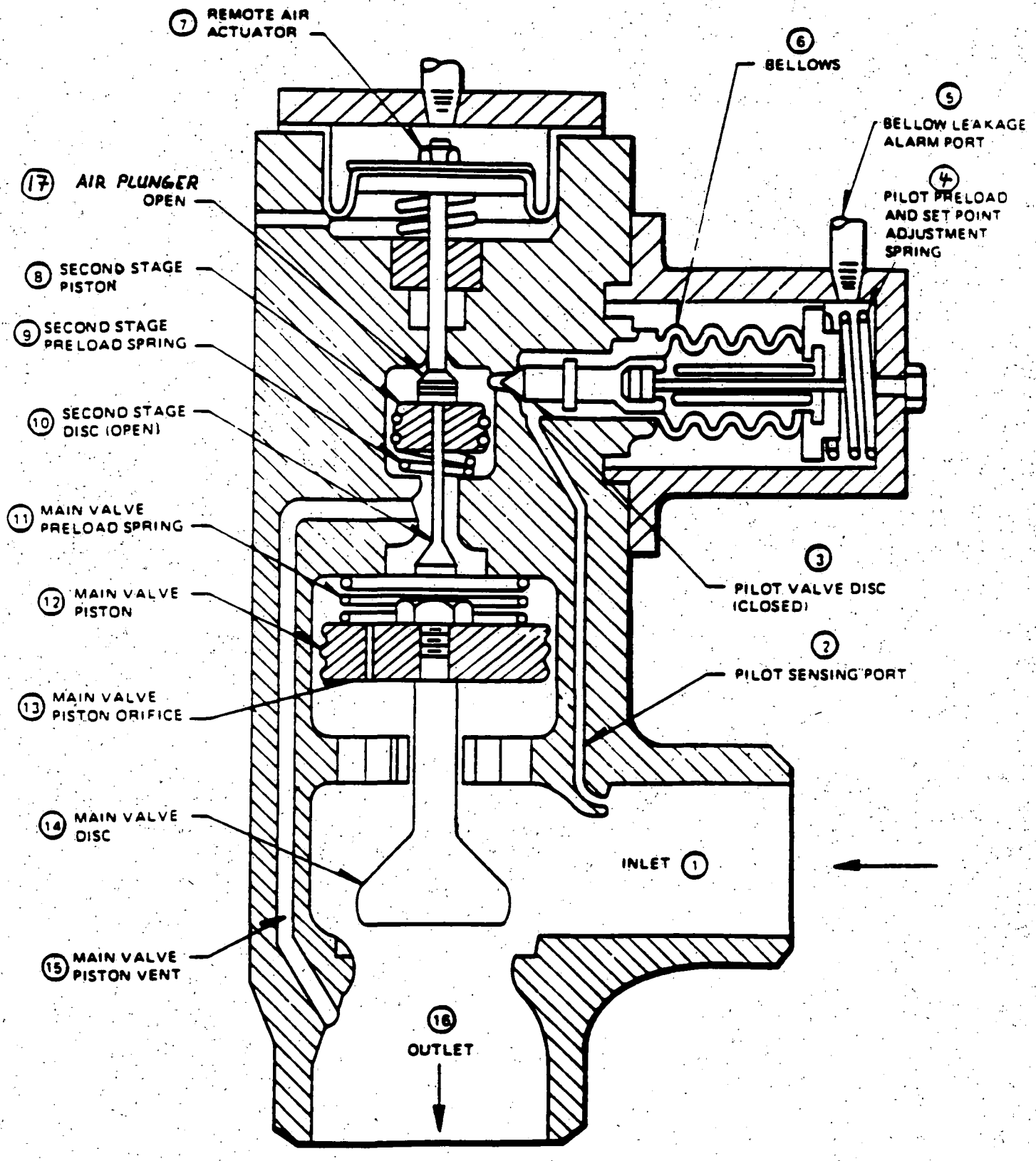

FIGURE 3

TARGET ROCK SAFETY RELIEF VALVE CONTROLS