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Dresden Nuclear Power Station
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August 28, 1990

EDE LTR #90-566

U.S. Nuclear Regulatory Commission
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Licensee Event Report #90-006-0, Docket #050237 is being submitted as required by Technical Specification 6.6, NUREG 1022 and 10 CFR 50.73(a)(2)(iv).

E.D. Eenigenburg
Station Manager
Dresden Nuclear Power Station

EDE/ade

Enclosure

cc: A. Bert Davis, Regional Administrator, Region III
File/NRC
File/Numerical

(ZDVR/22)

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LICENSEE EVENT REPORT (LER)

Form Rev 2.0

Facility Name (1) Dresden Nuclear Power Station, Unit 2 Docket Number (2) 0 15 10 10 10 12 13 17 Page (3) 1 of 0 9

Title (4) Target Rock Safety-Relief Valve Fails Open Due to a Steam Cut Pilot Valve Disc

Event Date (5)			LER Number (6)			Report Date (7)			Other Facilities Involved (8)		
Month	Day	Year	Year	Sequential Number	Revision Number	Month	Day	Year	Facility Names	Docket Number(s)	
0	8	0	2	9	0	9	1	0		0 15 10 10 10 1 1	
				0	0	0	8	2		0 15 10 10 10 1 1	

OPERATING MODE (9) N

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR (Check one or more of the following) (11)

POWER LEVEL (10) 0 8 7	<input type="checkbox"/> 20.402(b)	<input type="checkbox"/> 20.405(c)	<input checked="" type="checkbox"/> 50.73(a)(2)(iv)	<input type="checkbox"/> 73.71(b)
	<input type="checkbox"/> 20.405(a)(1)(i)	<input type="checkbox"/> 50.36(c)(1)	<input type="checkbox"/> 50.73(a)(2)(v)	<input type="checkbox"/> 73.71(c)
	<input type="checkbox"/> 20.405(a)(1)(ii)	<input type="checkbox"/> 50.36(c)(2)	<input type="checkbox"/> 50.73(a)(2)(vii)	<input type="checkbox"/> Other (Specify in Abstract below and in Text)
	<input type="checkbox"/> 20.405(a)(1)(iii)	<input type="checkbox"/> 50.73(a)(2)(i)	<input type="checkbox"/> 50.73(a)(2)(viii)(A)	
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LICENSEE CONTACT FOR THIS LER (12)

Name Daniel G. Daly, Technical Staff System Engineer Ext. 2347

TELEPHONE NUMBER AREA CODE 8 1 5 9 4 2 -12 19 12 10

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS
X	S	B	R	IV	T	10	12	10	Y

SUPPLEMENTAL REPORT EXPECTED (14)

Expected Submission Date (15) X NO

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)

On August 2, 1990, during normal power operation in the Run mode, 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. The reactor was subsequently manually scrammed from 87% power at 0116 hours. All Containment Cooling Service Water and Low Pressure Coolant Injection pumps were manually started for maximum suppression chamber cooling. The maximum average cooldown rate when averaged over a one hour period reached 129.3 degrees F/hr, and maximum bulk suppression chamber temperature was 122 degrees F. The opening of the TRSRV was apparently caused by steam cuts on the first stage pilot valve disc. Analyses were performed to verify that the cooldown rate and the bulk suppression chamber temperature attained during this event were within design limits. A satisfactorily leak tested, rebuilt TRSRV was installed. The Technical Staff will monitor the TRSRV tail pipe temperatures to verify proper pilot valve operation. In addition, any TRSRV pilot valve of greater than eight months service will be replaced during future short unit outages with primary containment drywell accessibility. The TRSRVs are routinely replaced at each refuel outage. A previous TRSRV failure event was reported by LER 50-237/76-34.

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From 0117 hours to 0119 hours, the Turbine Bypass Valves were manually opened to limit the heat input to the suppression chamber and to lower Reactor pressure in an attempt to get the TRSRV to close.

At 0119 hours, 2B and 2D LPCI pumps were started to ensure maximum suppression chamber cooling.

At 0135 hours, an Unusual Event was declared based on failure of a primary system relief valve to close with the reactor water temperature greater than 212 degrees F and suppression chamber bulk temperature cannot be maintained less than 110 degrees F. State and local agency Nuclear Accident Reporting System (NARS) notification was made at 0143 hours, and NRC Emergency Notification System (ENS) notification was made at 0208 hours.

The TRSV was determined to be closed by acoustic monitor indication at 0351 hours.

The Unusual Event was terminated at 0530 hours and the Unit achieved Cold Shutdown at 0600 hours. The suppression chamber reached a maximum bulk temperature of 122 degrees F during this event, and a maximum reactor vessel cooldown rate of 129.3 degrees/hr was experienced.

When the Primary Containment drywell was made accessible, the TRSRV was inspected. Upon inspection of the TRSRV, 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. The TRSRV Bellows Seal pressure switch was found to be separated from its conduit. However, the pressure switch was still functional.

C. APPARENT CAUSE OF EVENT:

This event is being reported per 10CFR50.73(a)(2)(iv), which requires the reporting of any unplanned manual or automatic Engineered Safety Feature [JE] actuation, including the Reactor Protection System.

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 (1115 psig) from a pressure controller.
3. Initiation of Automatic Depressurization System (ADS) [SB] logic.

Self actuation occurs as follows (refer to Figure 1): 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.

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Remote actuation occurs as follows (refer to Figures 1 & 2): 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 2. 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". At greater than 50 psig, a Control Room annunciator also indicates valve actuation. Also, 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 the severely steam cut pilot valve disc (refer to Figure 1). 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 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.

The TRSRV is overhauled during every refuel outage. 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) was put into service on a unit. The pilot and secondary stage valve seats had been inspected, lapped, and satisfactorily leak tested prior to installation.

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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 Qualification (EQ) criteria were evaluated and no discrepancies were noted. In addition, there are 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 and surrounding area are documented at Dresden Station; it is concluded that junction box 2PB-2020 came loose between 01/23/90 and 08/02/90.

The cause of the bellows seal pressure switch being detached from its respective conduit was attributed to an inadequate support design configuration of the conduit and switch.

D. SAFETY ANALYSIS OF EVENT:

General Electric Co. performed a reactor vessel bulk temperature cooldown rate analysis to verify that the cooldown rate was within design limits of a Safety Relief Valve Blowdown (SRVB) event. General Electric used a conservative maximum average cooldown rate when averaged over a one hour period of 153 degrees F/hr for this event based on saturated steam pressures. This cooldown rate is significantly less than the cooldown rate used in a SRVB design basis analysis performed by General Electric which is 254.3 degrees F/hr when averaged over a one hour period. 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 SRVB events are allowed for 40 years of operation for Dresden Unit 2. There have been two 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.

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 TRSRV piping discharge.

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 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. Post transient visual inspection of the relief lines and supports verified that no relief line damage had occurred.

Installation of T-Quenchers on the relief piping where steam is discharged into the suppression chamber was previously performed 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 analysis of suppression chamber heatup events cases (NEDC-22170, July 1982) for Dresden, demonstrating that the maximum temperature achieved in a stuck open relief valve with a single suppression chamber cooling loop would be 131 degrees F. The local to bulk temperature difference determined analytically for this case was 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 events cases for Dresden). This temperature is well within the condensation stability limits for the T-Quencher; therefore, the loads on the suppression chamber during this event were negligible.

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Consequently, the loads experienced in this event were well within the design margins for these systems. 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.

The interior walls on the suppression chamber are coated with an epoxy paint which has a temperature resistance of 350 degrees F. Since the temperature in the suppression chamber never reached this magnitude, there was no concern for degradation of the paint on the interior walls of the suppression chamber.

Consequently, for the above stated reasons, the safety significance of this event is considered minimal.

E. CORRECTIVE ACTIONS:

The Immediate Corrective actions were as follows:

1. A walkdown of the TRSRV discharge piping in the drywell was conducted to inspect the material condition of the piping and its related supports. No discrepancies resulting from this walkdown were identified (237-200-90-07501).
2. A walkdown was conducted to inspect the material condition of electrical junction boxes throughout the Unit 2 drywell. All accessible junction boxes were visually examined for damage and for potential for failures due to vibration. Two junction boxes were found to be supported 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 (237-200-90-07502).
3. A new support for the TRSRV bellows pressure switch assembly was installed per Work Request (WR) 94381 (237-200-07503).
4. A new seismically designed support and mounting for the junction box were installed in the drywell. This junction box replaced the existing 2PB 2020 and 2PB 2021 junction boxes. The pressure switch wiring contained within these junction boxes was installed in the new box. These repairs were performed under WR 94380 (237-200-07504).
5. A satisfactorily leak tested, rebuilt TRSRV was installed per WR 90929 (237-200-07505).

The subsequent Corrective Actions are as follows:

1. The Unit 3 TRSRV junction boxes will be evaluated for structural adequacy during the next available outage of sufficient duration allowing drywell access (237-200-90-07506).
2. The Technical Staff will establish a program to monitor the TRSRV tailpipe temperatures so as to identify potential pilot valve leakage problems (237-200-90-07507).
3. Replacement of the Units 2 and 3 TRSRV 203-3A Pilot Assembly has been placed on the short outage lists. Instructions are to replace any TRSRV pilot assembly that has been in-service for a duration greater than eight months into the operating cycle if the Unit is placed in Cold Shutdown and the drywell is accessible during the short outage (237-200-90-07508).

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F. PREVIOUS OCCURRENCES:

<u>LER/Number</u>	<u>Title</u>
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50-237/76-34 Unit 2 Failure of Target Rock Valve 2-203-3A to Close.

During automatic blowdown surveillance testing with the Reactor in the Run mode at 15% rated core thermal power, TRSRV 2-203-3A opened and remained open. The cause of this event was due to excessive leakage of the first stage pilot valve. The pilot valve assembly, pilot stage and secondary stage of the TRSRV was replaced.

G. COMPONENT DATA:

<u>Manufacturer</u>	<u>Nomenclature</u>	<u>Model Number</u>	<u>Mfg. Part Number</u>
Target Rock Corporation	Safety-Relief Valve	67F	N/A

An industry-wide NPRDS data search was performed for Target Rock safety-relief valves (model number 67F) that had spuriously opened with the Reactor at power. There were seven reported occurrences. Five of the occurrences were due to excessive pilot valve leakage. The other two occurrences were due to an unknown cause.

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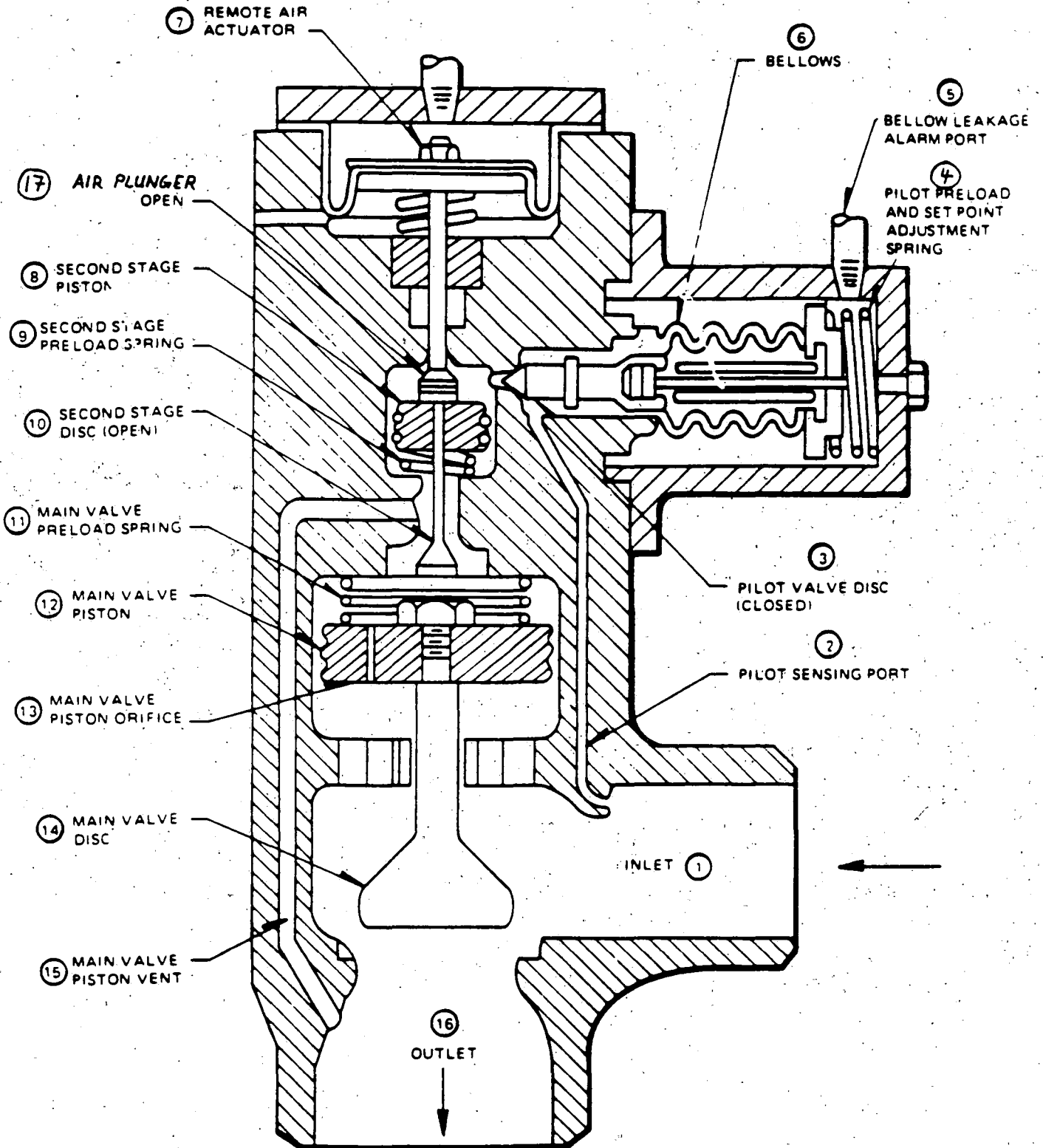
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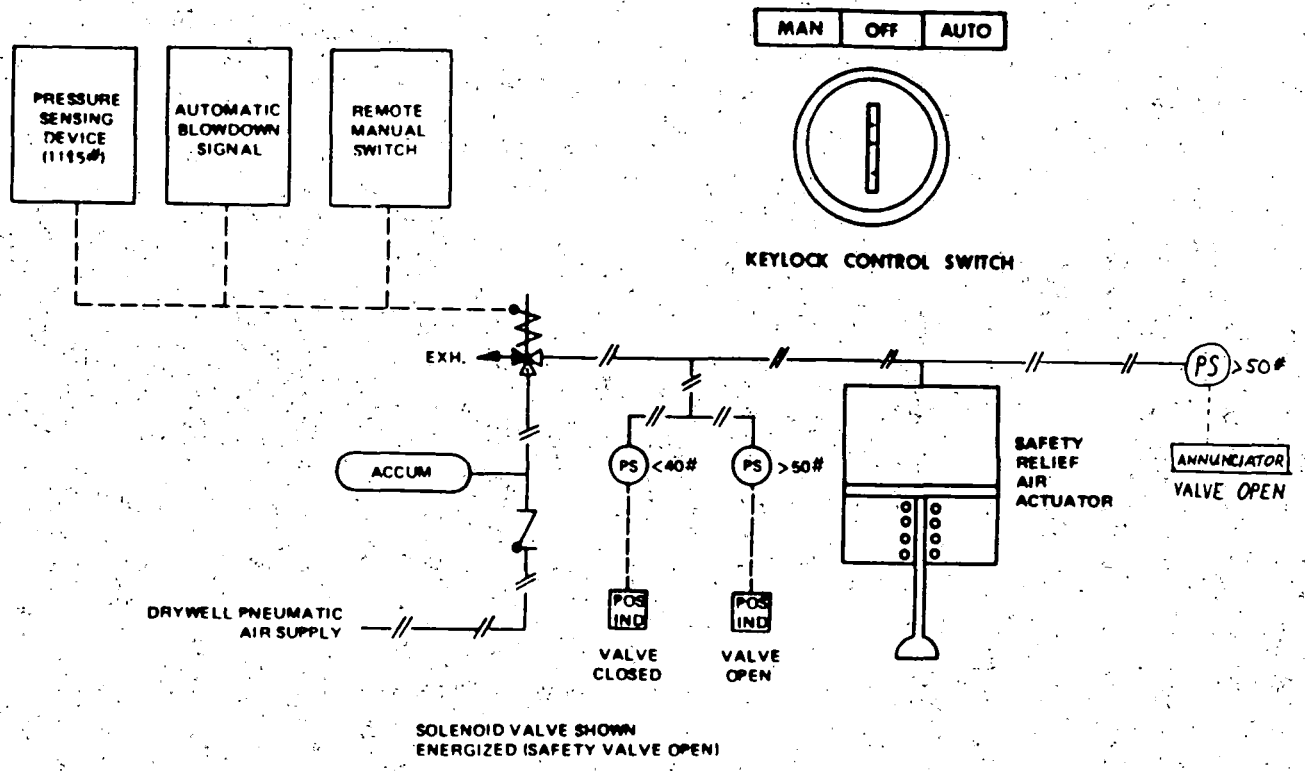


TARGET ROCK SAFETY - RELIEF VALVE (external actuation)

FIGURE 1

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TARGET ROCK SAFETY - RELIEF CONTROLS

FIGURE 2