Commonwealth Edison
Dresden Nuclear Power Station
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December 7, 1992

CWS LTR #90-713

U.S. Nuclear Regulatory Commission
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Washington, D.C. 20555

Licensee Event Report #90-006-01, Docket #050237 is being submitted to modify a corrective action that was provided in the original report.

for

C.W. Schroeder
Station Manager
Dresden Nuclear Power Station

CWS/glt

Enclosure

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

150054

(ZDVR/793)

IE22
1/1

LICENSEE EVENT REPORT (LER)

Form Rev 2.0

Facility Name (1)	Docket Number (2)	Page (3)
Dresden Nuclear Power Station, Unit 2	0 5 0 0 0 2 3 7	1 of 1 0

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)
									N/A	0 5 0 0 0
0 8	0 2	9 0	9 0	0 0 6	0 1	0 8	2 8	9 0	N/A	0 5 0 0 0

OPERATING MODE (9): N

POWER LEVEL (10): 0 8 7

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

20.402(b)	20.405(c)	X 50.73(a)(2)(iv)	73.71(b)
20.405(a)(1)(i)	50.36(c)(1)	50.73(a)(2)(v)	73.71(c)
20.405(a)(1)(ii)	50.36(c)(2)	50.73(a)(2)(vii)	Other (Specify in Abstract below and in Text)
20.405(a)(1)(iii)	50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	
20.405(a)(1)(iv)	50.73(a)(2)(ii)	50.73(a)(2)(viii)(B)	
20.405(a)(1)(v)	50.73(a)(2)(iii)	50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

Name	TELEPHONE NUMBER	
	AREA CODE	
Neil Spooner Technical Staff System Engineer Ext. 2789	8 1 5	9 4 2 - 2 9 2 0

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 V	T 0 2 0	Y					

SUPPLEMENTAL REPORT EXPECTED (14)

		Expected Submission Date (15)	Month	Day	Year
Yes (If yes, complete EXPECTED SUBMISSION DATE)	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 pool cooling. The maximum average cooldown rate when averaged over a one hour period reached 129.3 degrees F/hr, and maximum bulk suppression chamber water 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. The Station will routinely replace the entire TRSRV at each refuel outage. A previous TRSRV failure event was reported by LER 50-237/76-34.

PLANT AND SYSTEM IDENTIFICATION:

General Electric - Boiling Water Reactor - 2527 MWt rated core thermal power

Nuclear Tracking System (NTS) tracking code numbers are identified in the text as (XXX-XXX-XX-XXXXX)

EVENT IDENTIFICATION:

Target Rock Safety-Relief Valve Fails Open Due to a Steam Cut Pilot Valve Disc.

A. CONDITIONS PRIOR TO EVENT:

Unit: 2 Event Date: August 2, 1990 Event Time: 0105 Hours

Reactor Mode: N Mode Name: Run Power Level: 87%

Reactor Coolant System (RCS) Pressure: 997 psig

B. DESCRIPTION OF EVENT:

On August 2, 1990 at 0025 hours, with Dresden Unit 2 in the Run mode at 90% rated core thermal power, Operations Department personnel commenced decreasing load at a rate of 100 MWe/hr from 723 MWe per the Load Dispatcher's request. At 0105 hours, annunciator H-19 [IB], (Acoustic Monitor [IV] Actuated), on the 902-4 Control Room [NA] panel was received, indicating that a main steam relief valve had opened. No other alarms were received at this time. The position indicating lights on the 902-3 panel for the four Main Steam Electromatic Relief Valves (2-203-3B through 3D) and the one Target Rock Safety/Relief Valve (TRSRV) 2-203-3A indicated closed.

The Center Desk Nuclear Station Operator (NSO) assisted the Unit NSO by monitoring the Unit while the Unit 2 NSO inspected the Control Room back panels. The Unit 2 NSO discovered the TRSRV Acoustic Monitor had actuated. The TRSRV tailpipe temperature indicated 310 degrees F, as shown on Leak Detection Recorder 2-260-20A. While the Unit 2 NSO was checking the back panels, the Center Desk NSO further verified that the TRSRV was open by observing that the Total Steam Flow Recorder [IG] indicated a drop of approximately 600,000 lb/hr steam flow. The Shift Engineer was notified of these findings, and he immediately reported to the Control Room.

After verifying the TRSRV was open, Operations personnel immediately proceeded to carry out actions in accordance with Dresden Operating Abnormal Procedure (DOA) 250-1, Relief Valve Failure. The valve could not be closed. At 0109 hours, due to the increasing suppression chamber water temperature resulting from the blowdown, 2C Containment Cooling Service Water (CCSW) [CC] pump, 2A CCSW pump, and 2A Low Pressure Coolant Injection (LPCI) [BO] pump were started for suppression pool cooling. At 0111 hours, 2C LPCI pump was started, and at 0114 hours, 2B and 2D CCSW pumps were started for additional suppression pool cooling. After completing preparations for a manual scram per Dresden General Procedure (DGP) 2-3, Reactor Scram, the Reactor was manually scrammed at 0116 hours.

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 pool 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 water 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 water 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.

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 vale 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.

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

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 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 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 the Dresden Final Safety Analysis Report (FSAR) and the General Electric 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 water 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 supression 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 pool 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 have 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 water temperature only reached 122 degrees F, and both loops of suppression pool 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 analysis of suppression pool heatup event 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.

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 were identified.

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.

3. A new support for the TRSRV bellows pressure switch assembly was installed per Work Request (WR) 94381.

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.

5. A satisfactorily leak tested, rebuilt TRSRV was installed per WR 90929.

The subsequent Corrective Actions were as follows:

1. The Unit 3 TRSRV junction boxes were evaluated for structural adequacy during the next available outage of sufficient duration allowing drywell access. The structural adequacy was determined to be satisfactory.

2. The Technical Staff has established a program to monitor the TRSRV tailpipe temperatures so as to identify potential pilot valve leakage problems.

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

3. Replacement of the Units 2 and 3 TRSRV 203-3A Pilot Assembly was placed on the short outage lists. Instructions were to replace any TRSRV pilot assembly that had 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.

Since the initiation of this activity, no removed TRSRV pilot valve assembly has shown evidence of excessive steam cutting on the valve disc to justify its replacement. Investigation revealed that the eight-month in-service duration for pilot valve replacement had no technical basis, but was an arbitrary time initially chosen by the Station. In addition, the tailpipe temperature monitoring program mentioned in item 2 above has been effective in measuring small temperature fluctuations (1-2 degrees F) due to load changes. Since the initiation of the program, a sufficient amount of baseline data has been obtained on both Units to determine when a high tailpipe temperature condition exists. Tailpipe temperatures are also continuously monitored and recorded in the Control Room on Panel 902(3)-21. Annunciation of high tailpipe temperature, triggered from the recorder, is also available on alarm window 902(3)-4, H-17.

In compliance with ASME/ANSI OM-1 1981, Section 3.3, removal of the TRSRV pilot valve assembly requires an as-found lift setpoint determination prior to any dissasembly or adjustment to the valve. In addition, OM-1, Section 4.1 requires that valves designed to operate under steam be tested with saturated steam; however, an alternate test medium may be used provided correlation data exist between the test media and steam. Since as-found testing of the TRSRV using steam cannot be performed with the valve in-place, and correlation data betweeen an alternate medium and steam are not available, the entire TRSRV must be removed and sent to a remote testing facility to perform the as-found test. Station compliance to OM-1 1981 took effect on March 1, 1992. The subject commitment, however, was made prior to this date under a former code which allowed for the removal of the pilot valve assembly without performing the as-found test.

Based upon inspections of the TRSRV pilot valves previously removed, the current tailpipe temperature monitoring program, and the change of code requiring an as-found lift setpoint determination prior to any maintenance or adjustment, TRSRV pilot assemblies in service for greater than eight months into an operating cycle will not be replaced if the unit enters cold shutdown with the drywell accessible. The Station will continue to remove the entire TRSRV (including the pilot valve), and replace it with a rebuilt, leak tested TRSRV every refuel outage. In addition, TRSRV tailpipe temperatures will continue to be monitored in order to identify potential pilot valve leakage.

F. PREVIOUS OCCURRENCES:

LER/Number Title

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 were replaced.

G. COMPONENT DATA:

Manufacturer	Nomenclature	Model Number	Mfg. Part Number
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.

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION Form Rev 2.

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Dresden Nuclear Power Station	0 5 0 0 0 2 3 7	9 0	-	0 0 6	-	0 1	0 9 OF 1

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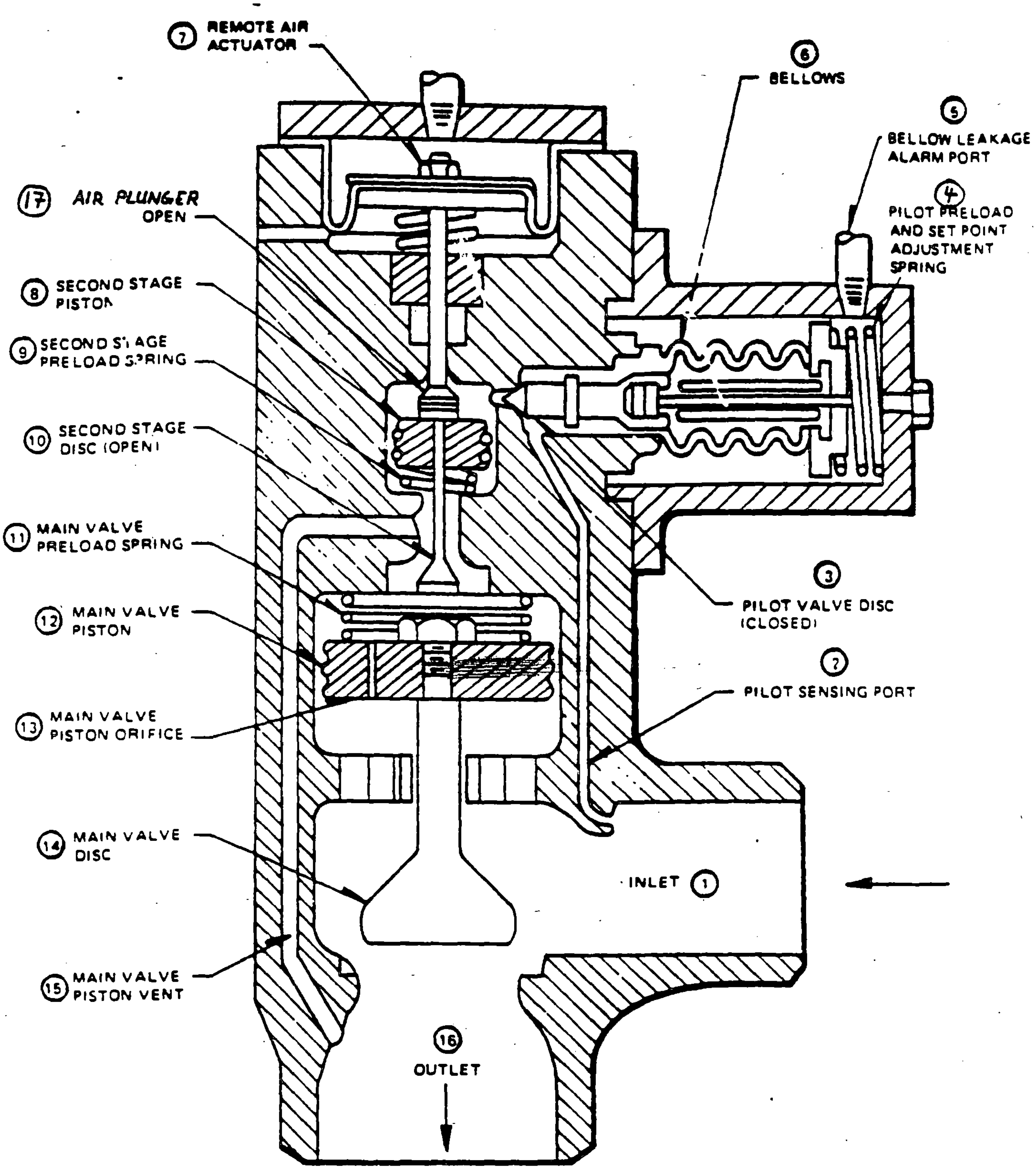

TARGET ROCK SAFETY - RELIEF VALVE (external actuation)

FIGURE 1

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION Form Rev 2.0

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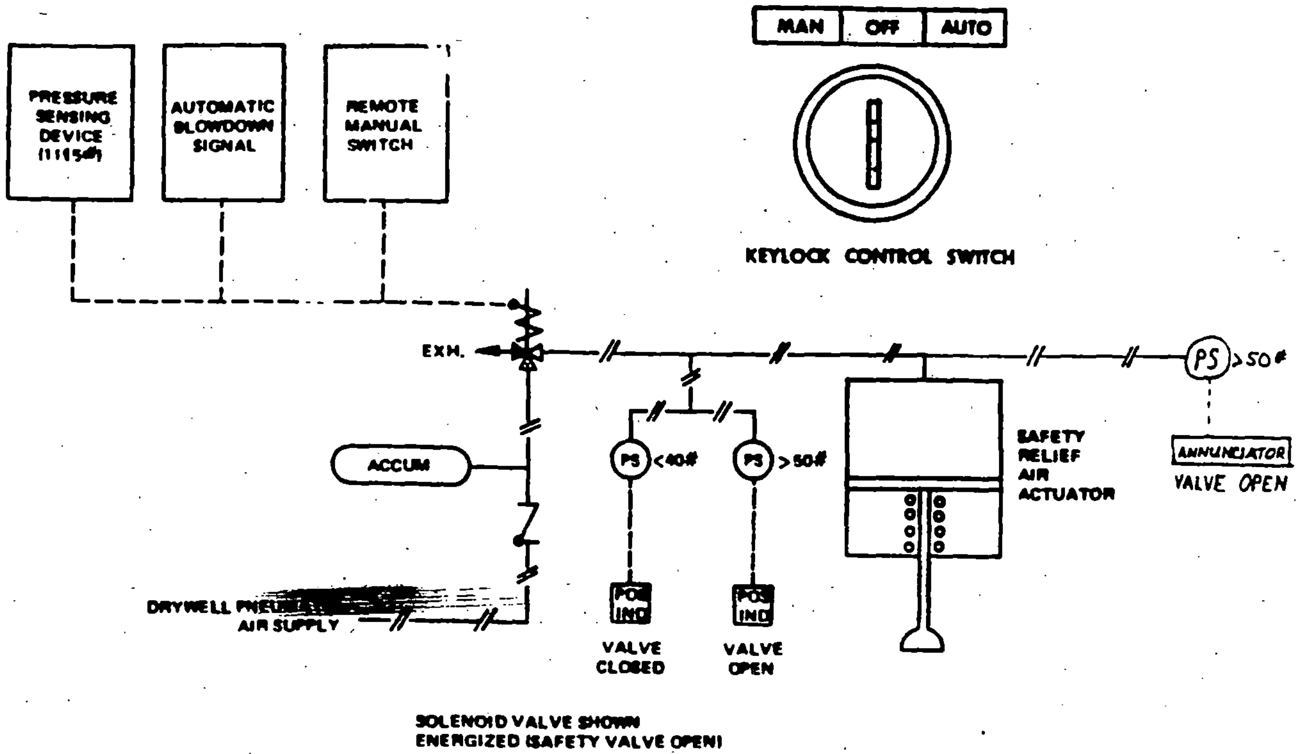

TARGET ROCK SAFETY - RELIEF CONTROLS

FIGURE 2