TRIP REPORT

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Onsite Analysis of the Human Factors of an Event at Dresden Unit 2 on August 2, 1990

(Spurious Safety Relief Valve Opening)

Investigative Team

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INEL/EG&G Idaho, Inc.

ACKNOWLEDGMENTS

We would like to express our appreciation for the cooperation of Dresden Station staff, especially for Gary L. Smith's assistance in arranging for the site visit and interviews with operators, and for the cooperation of the operators who recounted their experiences of this event while on duty. Thanks also to Elaine Beardall for her expeditious word processing.

EXECUTIVE SUMMARY

At 0105, August 2, 1990, Dresden 2 was at 80% power when a safety relief valve (SRV) opened spuriously and remained open. The control room crew successfully executed a manual reactor scram and plant cooldown, but with an excessive plant cooldown rate. This is a report of the visit by a human factors team to analyze the control room crew operations during the event. The team leader was Eugene Trager, NRC/AEOD. Other team members were John Kauffman, NRC/AEOD, and Bruce Kaplan and Brad Stockton from the Idaho National Engineering Laboratory. Data was acquired from the plant logs and recordings and from structured interviews with the control room operators. Data acquisition was enhanced by a detailed time line sequence for the event that had been prepared by the station staff.

The open SRV blowdown to the torus caused an initially rapid rate of temperature rise of the torus (1.3°F/minute). The shift engineer, in command of the Unit 2 control room crew, followed a functional response mode of operation to this symptom and opened both turbine bypass valves (TBVs) for approximately two minutes following a successful scram of the reactor. This reduced the total heat input to the torus, but contributed to a 126°F plant cooldown in one hour, which was in excess of the 100°F/hour normal cooldown limit. (Our analysis of this event demonstrated that plant cooldown without opening the TBVs would not have caused the torus temperature to approach its heat capacity temperature limit.) The TBVs were closed at approximately 600 psig. Plant cooldown and decay heat removal, thereafter, was affected primarily by SRV blowdown to the torus, although all auxilliary steam loads were not secured until later in the event.

Although spurious opening of an SRV is an anticipated event for a boiling water reactor, there was no event-specific guidance for plant cooldown in the plant procedures or training material. Technical guidelines for this event would have permitted the crew to be aware that

the torus can safely absorb the heat load from plant cooldown by an SRV blowdown, provided that the reactor is scrammed. Other human factors noted by the team included a turnover of control room supervision during the event, which can result in the lack of continuity of supervision of response during the event, limited diagnostic support of the Unit 2 shift engineer by the shift technical advisor or senior reactor operator, and the lack of knowledge by the crew of similar events involving a stuck open SRV on other BWRs.

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1.0 INTRODUCTION

1.1 Purpose

The purpose of the site visit on August 8 and 9 and the subsequent analysis was to examine human performance aspects of a transient that occurred at 0105 on August 2, 1990, at Unit 2 of the Dresden Nuclear Power Station. The transient involved the spurious opening of a Target Rock safety relief valve (SRV), 2-203-3A, while Unit 2 was at 80% power and reducing load, a manual reactor scram, and the subsequent cooldown of the plant at an excessive rate. The analysis of this transient is the fourth of a planned series of studies to be conducted for the purpose of analyzing the factors that affect human performance during operational events. Methodologies used by the team will be evaluated for implications concerning the subsequent investigation of human-factors-related events and the improvement of human performance in nuclear power plant operation.

1.2 Scope

The investigation focused on the actions of control room operators following a transient that began at Dresden Unit 2 at approximately 0105 on August 2, 1990. Based on data from discussions, plant logs, and interviews with control room operators and other station staff, analyses were made of technical guidelines for operations, shift organization, plant procedures, training, and operator response. This study addresses the factors affecting human performance in responding to a stuck open SRV, which resulted in torus heating, a manual reactor scram, and actions that resulted in excessive cooldown of the reactor system.

1.3 Team Composition

The investigative team was composed of two members from the Nuclear Regulatory Commission's (NRC's) Office for the Analysis and Evaluation of Operational Data and two members from the Idaho National Engineering Laboratory/EG&G Idaho, Inc. The NRC members were Eugene Trager

(team leader) and John Kauffman. The Idaho National Engineering Laboratory members were Bruce Kaplan, Human Factors Research Unit, and Brad Stockton, Regulatory and Technical Assistance Unit.

2.0 DESCRIPTION OF INVESTIGATION

2.1 Background

The Dresden Nuclear Power Station of Commonwealth Edison is located near Morris, Illinois. Dresden Units 2 and 3 are General Electric boiling water reactors (BWRs) rated at 832 MWe with Mark I containments. Unit 2 began commercial operation in August 1970 and Unit 3 began commercial operation in October 1971.

Unit 2 and Unit 3 are operated from a common control room. The normal control room crew complement is listed in Table 1. The plant operators are licensed for operation of both units, which are similar but not identical. Operators rotate between Unit 2 and Unit 3 watch stations. The shift organization consists of a shift engineer (SE), a shift control room engineer (SCRE), a shift foreman (licensed shift supervisor, or SSL) for each unit, and four nuclear station operators (NSOs) (one NSO for each unit, one NSO who may assist with either unit, referred to as the center desk operator, and one NSO who is a back panel or utility operator). The SE is a qualified shift supervisor, a licensed senior reactor operator (SRO), and has overall responsibility for the safe operation of both units. The SCRE is degreed and SRO-licensed. He normally directs the control room operators and activities for both units. He also serves as the shift technical advisor (STA) during transient and accident conditions after being relieved by the SE. The shift foreman (SSL) is a qualified shift supervisor and a licensed SRO. He reports to the SE and supervises operations in the plant. He also serves as coordinator between the plant operators and the control room. The NSOs are licensed reactor operators (RO) and are tasked with the actual control of the reactor and support

systems within the control room. Equipment operators perform equipment manipulations in the plant.

Once a year, during January, operations personnel are divided into crews that remain on shift for the year. The crews remain intact when rotated between shifts.

2.2 Event Description

On the day of the event, Thursday, August 2, 1990, operations personnel were working the sixth day of the midnight shift. The control room crew had arrived at approximately 10:30 p.m., Wednesday evening, and shift turnover occurred at 11:30 p.m. At approximately 00:25 a.m., August 2, the control room operators began reducing power on Unit 2 from 723 MWe (87%) at a rate of 100 MWe per hour, as requested by the load dispatcher. This is a frequent night shift evolution because of the decreasing network load demand during the late night and early morning hours, especially when cooler-than-normal summer temperatures exist in the area.

The following tabular time line sequence begins with the first event-related control room alarm. The time line is based upon the time line developed by the Dresden staff of the event. Figure 1 is a plant computer plot of torus temperature, reactor pressure and temperature, and steam flow during the event.

Event Details

0105

- Unit 2 is at approximately 80% power. Received 902-4 annunciator H-9 "Acoustic Monitor Actuated." No abnormal indications on the 902-3 front panel, however 902-21 back panel acoustic monitor shows red open light and yellow memory light for the Target Rock safety relief valve (SRV) 2-203-3A. The Target Rock SRV position-indicating lights show the valve as closed, not open.

[The Target Rock SRV and the four Electromatic SRVs form the automatic depressurization system (ADS). They are operated by remote-manual control switches, by the ADS circuitry, or by pilot valves that sense rising reactor pressure vessel (RPV) pressure. In addition, there are eight ASME Code safety valves.]

- Unit 2 operators note an approximate 50 MWe drop in electrical output, a rising torus water temperature, and an increased SRV tailpipe temperature (310°F). The Unit 2 operators are aware that these indicate an open SRV, however, this is not consistent with the SRV position-indicating lights nor with the RPV pressure, which is less than the SRV pressure setpoint.

- The SCRE decides that the Target Rock SRV is open for some reason, possibly because the pilot valve setpoint has drifted sufficiently to cause spurious opening.

- The SE is notified of the event. He reports to the control room promptly and relieves the SCRE, who assumes the STA responsibilities.

  [The two shift foremen leave the control room to supervise local operations later in this event.]

<u>0106</u>

- Unit 2 operators enter abnormal operating procedure DOA 250-1, Relief Valve Failure, and prepare to enter procedure DGP 2-3, Reactor Scram.

<u>0109</u>

- Unit 2 operators start containment cooling service water (CCSW) pumps 2C and 2A for torus cooling. The plant computer does not register the pump running alarm for CCSW pump 2A. This discrepancy had been previously noted during a surveillance.

0109 - 0116

- Torus temperature is observed by the crew to be rising rapidly. (Approximately 1.3°F/minute as determined later.)

- At 95°F torus temperature, the crew enters emergency operating procedure DEOP 200-1, Primary Containment Control.

- To increase torus cooling, the following pumps are started: low pressure coolant injection (LPCI) pump 2A at 0109:58, LPCI pump 2C at 0111:38, CCSW pump 2B at 0114:22, and CCSW pump 2D at 0114:28.

- The SE directs the Unit 2 operators to prepare for a manual reactor scram.

- During this time interval, the instructions in DOA 250-1 for attempting to reclose a stuck open SRV are initiated:

  1. Cycle the control switch from AUTO to MANUAL to AUTO

  2. Cycle the ADS inhibit switch from NORMAL to INHIBIT to NORMAL

  3. Pull fuses to deenergize the SRV control circuit.

- Control adjustments are made to the drywell pneumatic pump back system to maintain the 1 psi differential pressure between the drywell and the torus.

0116

- Manual reactor scram initiated per DOA 250-1 instructions following failure of above efforts to reclose the stuck open SRV, and also, in anticipation of torus temperature reaching 110°F where DEOP 200-1 and the Technical Specifications direct similar initiation of a manual scram. Torus temperature is 104°F and rising.

- DEOP 100 is entered following the scram and immediate actions therein are initiated.

- Group II and III isolations automatically initiated on low reactor water level, +8 inches.

- Generator field breaker opened/generator tripped.

0117

- Turbine EHC trips at high reactor water level (+55 inches).

- Reactor feedwater pumps 2A and 2C trip automatically due to high reactor vessel water level (+55 inches).

0117

- The torus temperature is still increasing at an apparently rapid rate. The SE evaluates the rate of approach to the heat capacity temperature limit (HCTL) for the torus. (See Figure 2 of this report for a reproduction of Figure 200-1-E of DEOP 200-1.) If this limit is exceeded, emergency depressurization is required. The SE decides to depressurize the RPV to 600 psig by dumping steam to the main condenser. He did this to reduce the heat input to the torus and by reducing the RPV pressure, possibly reseat the open SRV. However, he also erroneously believed that the RPV pressure would stabilize when the TBVs were reclosed.

0117:35

- Two turbine bypass valves are opened to near the fully open position.

0119:25

- The TBVs are closed.

- LPCI pumps 2B and 2D are started for full cooling of the torus with the LPCI system.

0120

- Group II and III isolations are automatically initiated by a low reactor water level trip at +8 inches.

0121:50

- NSO starts reactor feedwater pump 2B.

0121:31

- Low reactor water level trip resets.

0135

- Unusual Event declared per Emergency Action Level 2F.

0137 (approximate)

- Auxiliary steam loads secured to reduce the RPV cooldown rate.

- It is evident to the operators that the rate of increase of the torus temperature has been reduced.

0143

- Nuclear Accident Reporting System verification is made to the State of Illinois by the SCRE.

0208

- Emergency Notification System notification is made to the NRC by the SCRE.

0115 - 0215

(The RPV cooldown rate was later determined to be approximately 126°/hour during this time interval. The technical specification limit for normal RPV cooldown is 100°F/hour.)

0230

- Torus temperature has leveled off at 121°F. (The HCTL for the RPV pressure and torus water level conditions is approximately 185°F.)

0344

- Shutdown cooling system (SDC) isolation valves are opened in preparation for SDC startup to continue plant cooldown.

0351

- The Target Rock SRV closes as shown by the acoustic monitor indication and by the increased rate of torus cooldown.

0530

- Terminated Unusual Event.

0600

- Unit 2 is in cold shutdown and the Unusual Event has been terminated.

2.3 Analysis

2.3.1 Technical Guidelines for Operations. Spurious opening or failure to reseat an SRV relief valve is an anticipated event on BWRs, since it requires only a single failure of an active component that is in service during all power operation and hot shutdown modes. The open SRV causes an initially rapid torus temperature increase, which is later moderated by decreasing RPV pressure and the effect of the starting of

torus cooling pumps. As this event at Dresden demonstrated, it is impossible for the control room operators to mentally extrapolate the initial conditions and rates of change of RPV pressure and torus temperature to determine the effects of decay heat, RPV cooldown, and torus cooling upon the ultimate peak temperature of the torus. In addition, the HCTL value increases with decreasing RPV pressure and with increasing water level in the torus. Also, the effectiveness of the torus cooling by the LPCI and CCSW pumps increases with increasing torus temperature due to the increasing differential between torus water temperature and CCSW temperature.

These factors moderate the rate of temperature increase of the torus and, as in this event, eventually the temperature will level off and start to decrease. The initial rate of temperature increase is alarming in view of the consequences of reaching the torus HCTL (i.e., a requirement for emergency depressurization of the RPV). However, the analyses included in the Technical Specification bases indicate that the HCTL is not a concern if the reactor is scrammed before a 110°F torus temperature is reached.

Some relatively simple thermal hydraulic analyses would provide the technical basis for guidelines for operating procedures and training for this event. Securing all steam flow from the RPV, either by closing the MSIV or by securing auxiliary steam loads, is a probable guideline for this event. Curves of the calculated torus temperature and RPV pressure versus time would be adequate training material to provide the operators with an image of the dynamics of this event. Simulator exercises would, of course, be useful for the same purpose.

Lacking these event-specific guidelines for an open SRV, the SE made a functional response to the rising torus temperature, which was a challenge to the containment safety function. However, opening the TBVs to reduce the heat load on the torus caused an unnecessary challenge to the RPV pressure control safety function (excessive cooldown rate). (As discussed in Section 2.3.2, there was no one available to provide diagnostic support to the SE.)

2.3.2 Shift Organization. The SCRE is relieved during plant events by the SE so that the SCRE can fulfill the STA role. This results in a time-consuming turnover of responsibilities during an operational event. The turnover is not specifically required by regulation and can introduce a variety of communication-related problems, such as some loss of the familiarity with the circumstances that the SCRE has already gained. During the Unit 2 event, the SCRE was relieved and his time was then occupied with filling out event notification forms and making the required notifications to state and local officials and to the NRC, and with plant call out. This limited the effectiveness of the SCRE in his performance of the STA function of oversight, advice, and assistance to the SE.

The shift foremen, for both Units 2 and 3, were sent into the plant to perform valve manipulations and other activities. This resulted in the shift foremen not being available to review, assess, and evaluate the response to this event.

The use of the SCRE and the foremen to perform activities that could be performed by shift clerks or equipment operators, respectively, can be an inefficient use of resources that detracts from nuclear safety by preventing three of four senior, licensed personnel from performing the critical functions of monitoring and evaluating plant status and operator actions. Further, ability of the SE to function as the emergency director is impaired when he is also directing plant operators. These weaknesses in the control room organization may have been contributing factors to the excessive reactor pressure vessel cooldown.

2.3.3 Plant Procedures. Abnormal operating procedure DOA 250-1, Relief Valve Failure, does not contain some of the symptoms for this type of event, such as a decrease in MWe, the steam flow/feed flow mismatch, a decrease in steam flow, and difficulties in maintaining the 1 psi differential pressure between the drywell and the torus. When the torus temperature rises above 95°F, DOA 250-1 directs the execution of emergency operating procedure DEOP 200-1, Primary Containment Control, which specifies that DEOP 100, Reactor Control, be executed simultaneously.

However, these emergency operating procedures do not provide guidance for pressure control with one stuck open relief valve [e.g., the need to limit reactor pressure vessel cooldown rate, or ways to limit the reactor pressure vessel (RPV) cooldown rate to the maximum extent possible]. Procedure guidance specific for one open SRV could be added to the abnormal operating procedure DOA 250-1, and then referenced in the emergency operating procedure.

2.3.4 Training. Classroom and simulator training scenarios typically use a stuck open relief valve as the initiating event for an anticipated transient without scram (ATWS). The torus heats rapidly due to failure of the scram, and the torus temperature is a concern of major significance. The operations personnel interviewed stated that they had not been trained for this simpler event (one stuck open safety relief valve followed by a successful scram) to its expected conclusion (i.e., the reactor depressurized and cooled down). The more complicated simulator training prepared the operations personnel for the unlikely worst-case scenario. However, the lack of training for expected simple events failed to highlight the fact that the concerns and response to worst-case scenarios are often different for simple events.

Operations personnel were generally unconcerned with the RPV cooldown rate because they assumed the technical specification cooldown rate limit would have been exceeded anyway. Training could emphasize that the RPV cooldown rate should be limited and should not be ignored, even if it is apparent that it will be slightly exceeded. Further, operators had no estimates of the effect that auxiliary steam loads would have on the RPV cooldown rate and of the expected RPV cooldown rate from the combination of a stuck open SRV, auxiliary steam loads, and from opening the turbine bypass valves. The operators were also surprised by the rate of increase in torus temperature. (The simulator training may not provide realistic torus temperature response.) With the reactor scrammed at 104°F and maximum torus cooling initiated, the heat capacity temperature limit was not in danger of being exceeded. It appears the unexpected heat-up rate of the torus, combined with the procedures discussed previously, may have distracted operator attention from the need to take steps to limit the

cooldown rate. The classroom training lesson plans for a stuck open SRV were in some respects more comprehensive than the plant abnormal and emergency procedures (i.e. the lesson states that the RPV should be cooled down at normal rates if the suppression pool temperature exceeds 120°F). However, the classroom training material does not discuss the need to limit the RPV cooldown rate or ways that this can be achieved (e.g., by securing steam loads).

Operators were generally unaware of generic industry problems involving the Target Rock SRVs, such as spurious opening and their tendency to stick open after actuation, until after the event had occurred.

2.3.5 Operator Response. During the interviews, it became apparent that shift communications and operator response could be improved. Suppression pool cooling was not initially maximized, as required by DEOP 200-1, due to miscommunication or misunderstanding. During the opening of the bypass valves, the operator was not given specific instructions as to the number of valves to be opened, the desired pressure at which the valves should be closed, or the desired rate of depressurization.

3.0 SUMMARY OF FINDINGS

Investigation of the human factors aspects of the crew's attempts to deal with the spurious opening of a SRV produced significant information regarding human-system interactions. Specific findings from this study included:

3.1 Technical Guidelines

- Technical guidelines for procedures and training that are specific for one stuck open relief valve would be very helpful for the reactor operators.

3.2 Shift Organization

- The turnover of responsibilities during this event introduced potential communication problems and had the potential to disrupt the continuity of the event overview.

- Having the SCRE on the phone and the two foreman in the plant left the SE as the only senior, licensed person with full attention to overall plant operations.

- At the same time he was directing plant operations, the SE was also functioning as the emergency director.

3.3 Plant Procedures

- Plant procedures did not provide operator guidance for pressure control with one stuck open relief valve. Procedures did not explain the need for or methods of limiting the reactor pressure vessel cooldown rate.

3.4 Training

- Operator training addressed much more severe events, with little emphasis on simpler events. Training did not highlight how response to simpler events could differ from response to the severe events in which they had been trained.

- The systems that ensure operating experience will be reflected in training did not ensure that previous and similar problems with the Target Rock relief valves at other utilities were reflected in the training and communicated to the operators prior to the event.

3.5 Operator Response

- Operators did open and reclose the bypass valves as directed, but were not given specific instructions as to how many valves to open or the desired rate of depressurization.

TABLE 1. Normal Control Room Crew Complement

A. Control Room Supervision

Shift Engineer (SE)
Shift Control Room Engineer (SCRE)
Center Desk Operator (NSO)
Utility Operator (NSO)

B. Unit 1

Shift Foreman (SSL)
Nuclear Station Operator (NSO)

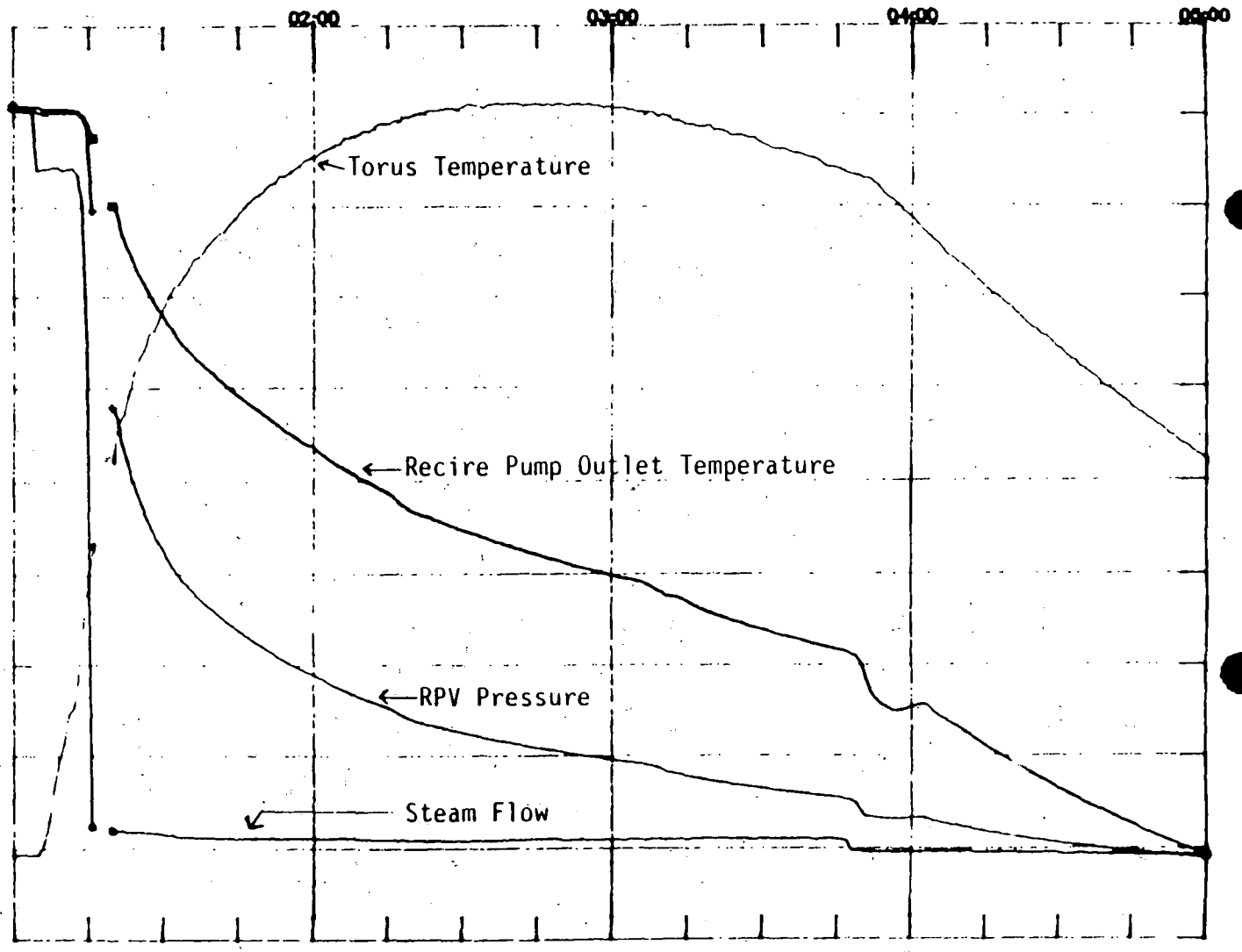
C. Unit 2

Shift Foreman (SSL)
Nuclear Station Operator (NSO)

Point Plot – Dresden Station, Unit 2
Data from 01:00 to 05:00 on 08/02/90
Time interval : 1 Minute(s)

[QN01014 Rev. 0.1]

Torus °F T257	RPV PSI °F C200	Recire Pump outlet °F C208	Steam Flow M#/ hr. C222
124.8538	1102.364	557.3784	8.568648
120.3868	984.7864	528.0602	7.711641
116.1200	867.2091	494.7438	6.824636
111.8531	749.6318	463.4276	5.837631
107.5862	632.0546	432.1112	5.050625
103.3193	514.4774	400.7948	4.163619
99.06241	396.9001	369.4786	3.276614
94.78552	279.3229	338.1622	2.389608
90.51862	161.7457	306.8458	1.502603
86.25171	44.16847	275.5289	0.615597
81.98482	-73.40877	[illegible]	-0.271408



POINT ID	DESCRIPTION	UNITS	SYMBOL
C222	RX OUTPUT STEAM FLOW	M#/HR	•
C208	RECIRC PMP A OUTLET TEMP	DEG. F	■
C200	REACTOR PRESSURE A	PSIG	▲
T257	TORUS BULK WATER TEMP A	DEG. F	!

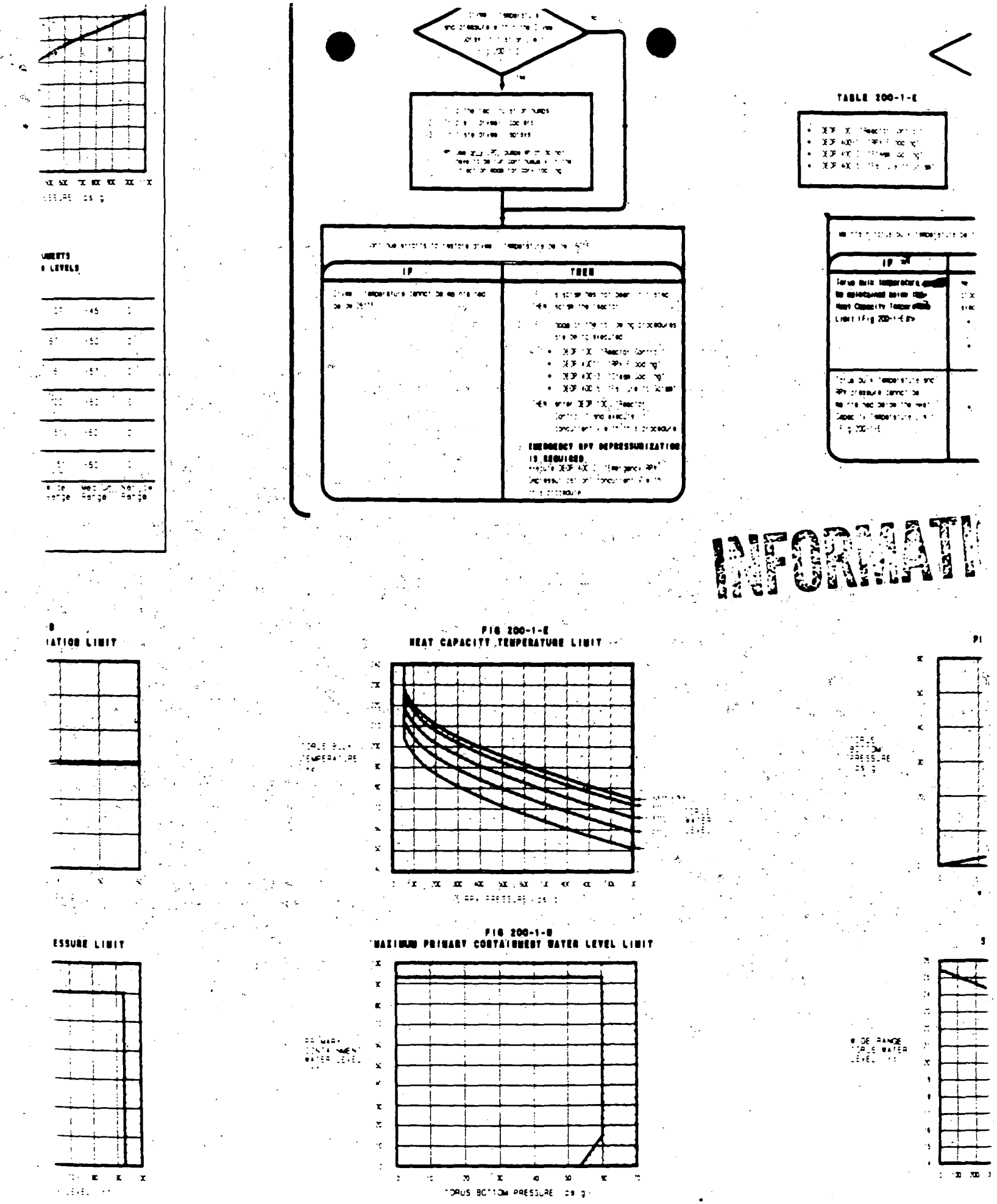


Figure 2. Heat Capacity Temperature Limit
(copied from DEOP 200-1, Primary Containment Control)

LICENSEE EVENT REPORT (L[illegible]

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 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 0	0 0 6	0 0	0 8	2 8	9 0		0 5 0 0 0 1 1 1 0 5 0 0 0 1 1 1

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)
Daniel G. Daly, Technical Staff System Engineer Ext. 2347	8 1 5 9 4 2 1 - 2 9 2 0

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
X	S B	R V	T 0 2 0	Y					

SUPPLEMENTAL REPORT EXPECTED (14)

___ Yes (If yes, complete EXPECTED SUBMISSION DATE) X NO

Expected Submission Date (15): Month | Day | Year

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.

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: 0116 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 Detector 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 chamber cooling. At 0111 hours, 2C LPCI pump was started, and at 0114 hours, 2B and 2D CCSW pumps were started for additional suppression chamber 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 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) [illegible]s forced down moving the second stage disc (10) away from its seat. This permits the pressure from the [illegible]p of the main valve piston (12) to be vented via the second stage disc (10) and out the main valve [illegible]ston vent (15). A differential pressure is created across the main valve piston (12) due to the sma[illegible] 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 [illegible] panel (annunciator D-23 [illegible]RSRV Inoperable[illegible] This indicates that the self actuation mode [illegible]RV 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.

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 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 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 [illegible] the condensation stability limit of the T-Quencher [illegible] the loads on the suppression chamber during the [illegible] 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 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).

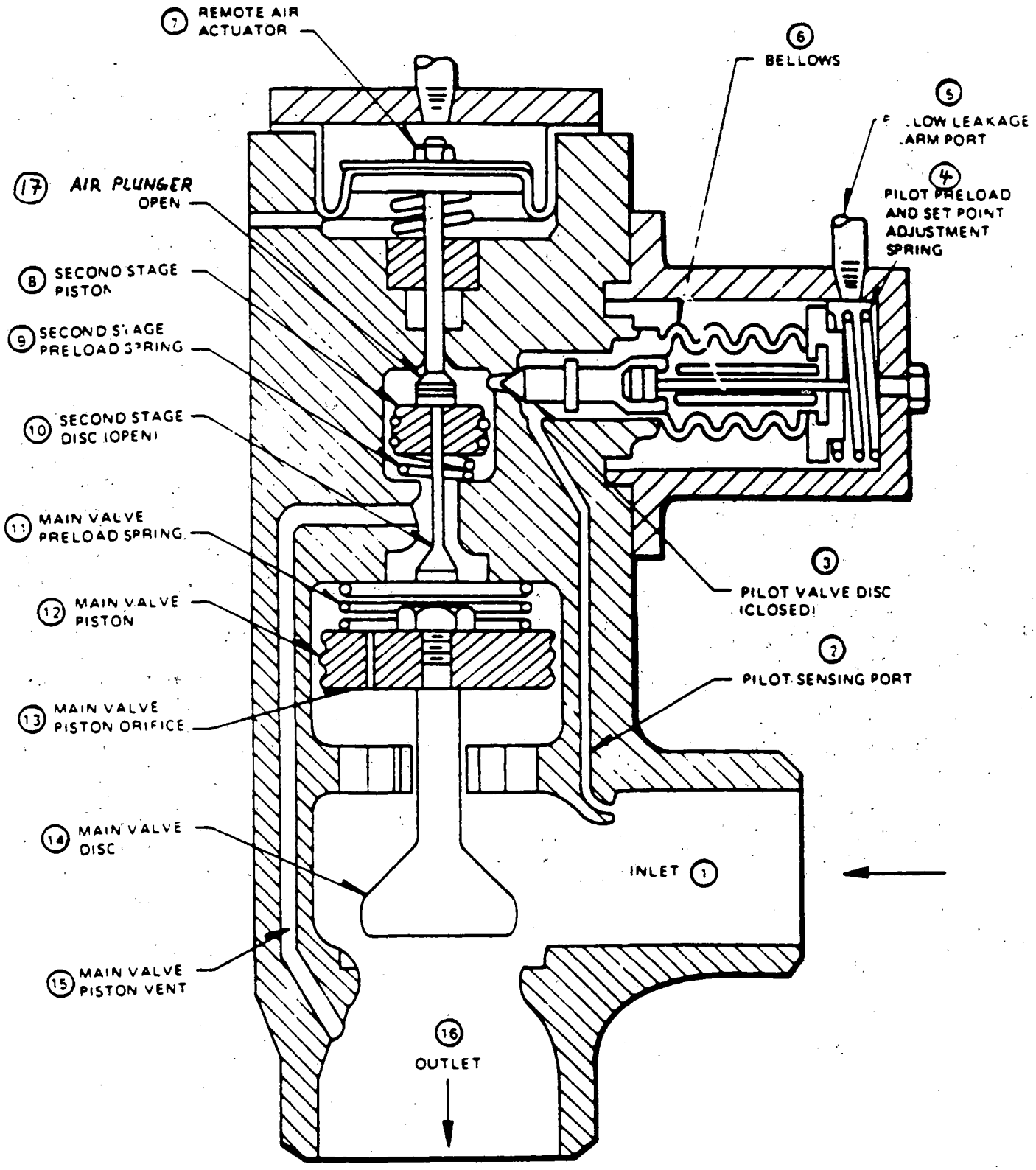
5. A satisfactorily leak tested, rebuilt TRSRV was installed per WR 90929 (237-200-07505).

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).

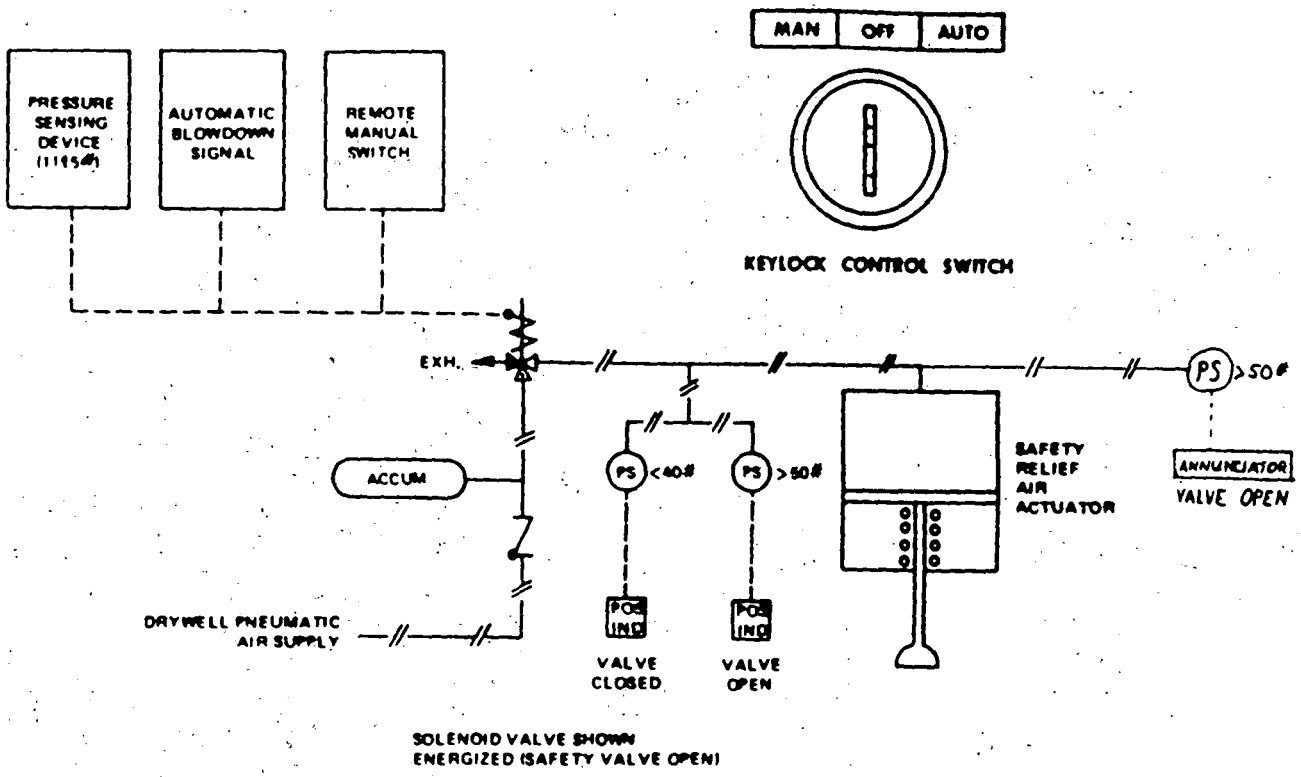
2. The Technical Staff will establish a program to monitor the TRSRV tailpipe temperatures so as to identify potential pilot valve leakage problems (237-200-90-07507).

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).



TARGET ROCK SAFETY - RELIEF VALVE (external actuation)

FIGURE 1



TARGET ROCK SAFETY - RELIEF CONTROLS

FIGURE 2

Commonwealth Edison
Dresden Nuclear Power Station
R.R. #1
Morris, Illinois 60450
Telephone 815 942-2920

August 28, 1990

EDE LTR #90-566

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

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)