UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

Docket 50-3
50-247

SEP 2 5 1979

GL-79045

TO ALL POWER REACTOR LICENSEES

SUBJECT: TRANSMITTAL OF REPORTS REGARDING FOREIGN REACTOR OPERATING EXPERIENCES

The enclosed reports are provided to you for information and use in your reactor evaluations in light of the Three Mile Island Unit 2 accident. Enclosure 1 is an internal Westinghouse report which describes an incident involving a stuck-open power-operated relief valve that occurred at the Beznau Unit 1 reactor in Switzerland on Augsut 20, 1974. This report is now a part of the official records of the President's Special Commission investigating the TMI-2 accident. Enclosure 2 is an internal NRC staff memo on this incident. Enclosure 3 is a report on a steam generator tube "rupture" incident at the Doel 2 nuclear power plant in Belgium.

If you have any questions about the enclosed information, please let us know.

Sincerely,

D. F. Ross, Jr., Director
Bulletins and Orders Task Force

Enclosures:
1. Technical Report on Beznau Unit 1 Incident of August 20, 1974: TG-1 Trip/Reactor Trip/Safety Injection Actuation
2. Memorandum dated May 15, 1979; Ashok Thadani to D. F. Ross, Jr.
3. Memorandum dated September 13, 1979; Darrell G. Eisenhut to Multiple Addressees.

MA

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cp

Westinghouse
Nuclear Europe

Paule VJ Esposito info



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SUBJECT : TECHNICAL REPORT ON NOK 1 INCIDENT OF AUGUST 20, 1974

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References (1) Telex SE-G-74-195 (8/26/74) to NOK by H. Cordle
(2) Letter (8/27/74) NKA-3940 from L. Barshaw.

You will find attached the technical report on NOK 1 Incident of August 20, 1974 prepared by WNE inspection team who went to Beznau on August 23.

This report, which should be sent to Beznau, summarizes our observations on the course of the transient, the damage as we viewed it, our calculations and conclusions.

Despite what is indicated in the referenced (2) letter, in order to have a more complete report, we added some recommendations for future changes.

T. CECCHI
SYSTEMS ANALYSIS

SA/251

<u>TECHNICAL REPORT ON BEZNAU UNIT ONE</u>

<u>INCIDENT OF AUGUST 20, 1974 : TG-1 TRIP/</u>

<u>REACTOR TRIP/SAFETY INJECTION ACTUATION.</u>


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September 2, 1974

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## TECHNICAL REPORT ON BEZNAU UNIT ONE INCIDENT OF AUGUST 20, 1974 : TG-1 TRIP/REACTOR TRIP/SAFETY INJECTION ACTUATION

## I - INTRODUCTION

This report is produced as a result of a site visit following the incident on Beznau I which took place on August 20, 1974. The object of the visit was to make a rapid evaluation of whether the consequences of the incident would jeopardize safety. This report confirms the telex of Aug. 28, 74 on this subject.

The scope of this report, therefore, is limited to a description of the sequence of events and of the damage observed together with a possible explanation and assessment of safety issues. It is not meant to be a comprehensive analysis of the effects of the incident.

## II - SEQUENCE OF EVENTS DURING THE INCIDENT

On August 20, 1974, a trip of one of the two turbines on the Beznau I reactor followed by failure of the steam dump system to operate resulted in a reactor trip and the opening of the pressurizer relief valves. One of these valves subsequently failed to close and the extended blowdown of the pressurizer resulted in the rupture of the pressurizer relief tank bursting disk. Examination following the incident revealed that the pressurizer relief valve which had failed to close had been damaged, as had some of the supports to the pressurizer relief line itself.

The sequence of events, with times where known, is reconstructed below :

Initial conditions :

Date : August 20, 1974 Time : 11.20 a.m.

Pressurizer pressure : 154 bar Pressurizer level : 50%

Pressurizer relief tank level : 80%

Power output of turbogenerator 1 : 187 MW (e)

" " " " 2 : 177 MW (e)

| Time | Event |
|---|---|
| | Disturbance occurs on the external grid network. |
| | TG1 trips out on high casing vibration. |
| 11 hrs 20 min 07.8 sec | Vibration causes low Δ p signal from hydrogen seal oil system. |
| | Steam dump valves fail to open. |
| | SG steam pressures rise. |
| | Pressurizer pressure rises. |
| | Pressurizer level rises. |
| 20 11.9 | Both pressurizer relief valves open. |
| 20 17.8 | Turbotrol of TG2 drops into the emergency mode. |
| 20 23.0 | One pressurizer relief valve closes in accordance with automatic signal, pressure continues to fall and level continues to rise. |
| | Pressurizer relief tank pressure rises. |
| | Pressurizer relief tank level rises. |
| | TG2 power level falls then rises to an overpower of 214 MW (e). |
| 21 00.4 | Reactor trips on pressurizer low pressure. |
| 21 01.2 | TG2 trips. |
| | SG steam pressures rise. |
| | SG water levels fall. |
| | Pressurizer level falls. |
| 23 03.5 | Secondary side safety valves lift. |
| 23 13.9 | Steam is formed in the RCS hot legs and pressurizer level rises past 100% and remains off-scale for 3 to 5 minutes. A reasonable assumption is that water discharge occurs through the open relief valve. |
| | Operator shuts pressurizer relief line isolation valve. (Reported verbally as 2 to 3 minutes after the trip). |

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| | |
|---|---|
| | Pressurizer level falls rapidly as steam bubbles in RCS collapse. |
| | Pressurizer relief tank bursting disk ruptures. |
| | Pressurizer relief tank pressure falls. |
| | Pressurizer relief tank level falls. |
| 11 hrs 23 min 43.5 sec | High containment pressure recorded (peak 1.1 bar abs.). |
| 24 51.2 | High containment temperature recorded (53.4°C). |
| 25 17.8 | High containment activity recorded (17.3 mr/hr). |
| 32 14.3 | SIS initiated as pressurizer level falls to 5%. |
| | Pressurizer level rises as SI water is added to the RCS. |
| | SIS stopped manually. |
| Subsequently | Procedure begun to bring reactor to cold shutdown condition, using the atmospheric steam relief valves. |

Fig. 18 shows the record of pressurizer pressure and level transients following incident initiation.

## III - TRANSIENT BEHAVIOR OF MAIN PLANT VARIABLES DURING THE INCIDENT

A turbine trip in a two turbine plant is equivalent to a 50% load rejection and no reactor trip should be initiated if control systems work correctly. Since in Beznau I the steam dump system did not work at all, initially the main variables behaved as follows :

1. Steam Generator steam pressure rose (to about 66 bars) but not enough in order to actuate safety valves.

2. Feedwater flow, steam flow and steam generator level decreased normally as expected.

3. The reactor being in automatic control, the nuclear power decreased. When reactor was tripped after about 49 seconds, it was at 76%.

4. Pressurizer pressure rose rapidly from 154 bars to a maximum of 160 bars (pressurizer relief valves actuation) in about 11 seconds.

5. Reactor coolant system average temperature rose rapidly from 298.5°C to a maximum of 305.5°C in about 50 seconds.

6. Cold leg temperature rose rapidly from 275°C to 290°C, then decreased to 240°C in 10 minutes, to 220°C in next 100 minute and to 140°C in next 170 minutes.

7. Pressurizer level rose from 50% to 67% in about 50 seconds.

Due to the fast pressurizer pressure increase, both pressurizer relief valves were rapidly actuated. Their actuation took place almost simultaneously. However, it is very probable that the valve actuated by the compensated pressure error signal (signal elaborated by a PID controller) opened some seconds before the other one due to the derivative term of the PID controller.

When pressure decreased below relief valves actuation setpoint the valve directly controlled from an uncompensated pressure signal did not shut. This resulted in a depressurization at rate of about 0.75 bar/sec, resulting in a reactor trip by low pressu in approximately 49 seconds.

The reactor trip signal tripped the turbine which was still in operation, resulting in a further steam pressure increase (above 70 bars) which produced steam generator safety valves actuation, lowering the pressure to about 65 bars.

Reactor coolant system average temperature decreased to about 285°C and pressurizer level to 23% in about 1 minute after reactor trip. At this point pressurizer pressure had fallen to hot leg saturation (70 bars). Subsequently, hot leg flashing resulted in an increase of pressurizer level until the pressurizer filled about 3 minutes after reactor trip, resulting in probable liquid water discharge from the relief valve and bulk boiling in the core. x Then the operator isolated the failed relief valve, and pressurizer level decreased reaching the setpoint (5%) for safety injection actuation (safety injection is actuated by coincident low pressurizer pressure and level S.I. signals) about 11 minutes after reactor trip. The system then started refilling. When pressurizer level reached about 70%, safety injection pumps were shut off manually.

The reactor was then brought normally to cold shutdown conditions.

## IV - DAMAGE TO THE RELIEF PIPE RESTRAINTS AND SUPPORTS

For pipe layout, see isometric, fig. 1 attached.

The relief line to the power relief valves comes out of the pressurizer top and runs directly down (vertical run of 6.8 m). It passes through a grating floor. No impact evidence between the floor and the pipe insulation exists. (Gap about 25 mm). At the bottom of the vertical run there is a console type restraint. (Location 1 in fig. 1). The main dimensions are given in fig. 2. There is contact evidence, as shown on the figure, but no damage.

The pipe then runs horizontally to the restraint 2 (fig. 1). This restraint limits motion of the pipe in a horizontal direction, perpendicular to the pipe axis (See fig. 3). Scratches on the shoes indicate that the pipe moved about 26 mm axially. The top part of the insulation is slightly smashed (See fig. 3).

../...

x Nuclear Power was then at [illegible]%.

The line then runs vertically down (2.77 m) and separates into two branches each having a stop valve and a relief valve. Fig. 7, 8 and 9 show the damage to the valve.

Examination of the pressurizer relief valve which failed to close revealed that the yoke had broken off completely. One arm of the cast iron yoke had broken at the top and the other arm at the bottom taking part of the yoke ring with it. The top break showed the presence of a very large flaw (inclusion). All broken faces showed classic brittle failure together with evidence that the faces had rubbed together following failure. In addition it was reported that the valve spindle had been slightly bent. This was not observed since repairs had already been started.

Fig. 6 and 7 show the pedestal of the support between the two valves. Fig. 4 is a sketch of the support and details the damage.

The damage corresponds to a rotation of the pipe around a horizontal axis perpendicular to the pipe axis. No evidence of translation has been found. Considering fig. 7, the back bolts were strained much more than the front ones.

The bolts of the undamaged valve support have been inspected. It was found that the paint was cracked at the bolt joints, but no other damage could be found.

After the valves the two branches of the pipe drop to the lower floor. Fig. 10 shows the penetration corresponding to the damaged branch.

At the lower floor, the restraint R4 (See fig. 1) has been pulled off the floor (see detail in fig. 14). The motion has been imposed on the frame by the bar of the hanger passing through a 50 mm slot in the frame (See fig. 11).

Restraint R5, which is only a column supporting a sliding shoe, shows a motion of 70 mm as shown in fig. 5.

The pipe then joins a header and passes through the floor (R6 on fig. 1). There is evidence of 25 mm upward displacement.

At the lower floor the header has an elbow. Motion is restrained by a snubber. The bolts fixing the snubber to the concrete were found to be loose.

## V - EVALUATION OF THE INCIDENT

This evaluation covers the incident transient effects and a preliminary estimate of magnitude and probable causes of damage to the pressurizer relief piping and supports.

### 1. Comparison with design transients

This Beznau I incident is similar to the two following incidents which are normally considered among reactor coolant system design transients :

- Loss of load (up to pressurizer relief valves actuation).
- RCS depressurization (from pressurizer relief valves actuation).

From the standpoints of core power, heat transfers and systems pressures and temperatures, the reported incident is less severe than the design transients considered above.

The magnitude and variation rate of the temperature and pressure transients resulting from the incident are indeed fully covered by the values used for equipment design.

Plant variable behavior during the transient did not result in an uncontrolled or damaging situation, and the released activity

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remained well below dangerous limits. All existing protection systems (steam generator safety valves, reactor trip, safety injection) worked properly and were adequate to handle the incident avoiding core and equipment damage.

2. <u>Evaluation of damage to the pressurizer relief line, the relief valves and supports.</u>

The relief line between the pressurizer and the power relief valves is part of the reactor coolant pressure boundary and therefore is important to the safety of the plant.

The one power relief valve which failed to close was isolated in accord with design intent by the operator closing the appropriate relief isolation valve and hence no uncontrolled loss of coolant occurred.

The review of the relief line equipment showed damage to the relief line supports and the pressurizer relief valve PCV-456.

The damage evaluation and probable causes are treated below.

a) <u>Discussion of the incident related to cause of damage.</u>

Examination of the relief line and supports along with the records of primary reactor coolant system parameters leads to the following observations.

(1) It is probable that the observed damage to the supports is the result of hydraulic shocks from a sequence of water and steam discharge through the relief line.

(a) The pressurizer relief line from the relief valve to the pressurizer can fill with condensate. This distance is approximately 19 meters, and can contain a volume of 0.06 $m^3$. Opening of the relief valves will cause a rapid discharge of the water. The resulting dynamics are one possible cause of the piping displacements observed.

(b) Based upon the recorder chart of pressurizer water level, it appears probable that some water discharge occurred later in the transient when the pressurizer was completely filled. The records indicate that this event could only have occurred after automatic closure of the undamaged valve (PCV-455C).

Dynamics related to this event are another possible cause of the observed piping displacements and support damage.

(2) It is not possible from available evidence to provide one sequence of events which uniquely explains the observed results of the transient.

It is not certain that the valve damage was the consequence of the same hydraulic shock that resulted in the support damage.

The observed sequence of events indicates that one likely scenario is as follows :

(a) The undamaged relief valve, PCV-455C, opens first on the derivative compensated pressure controller a few seconds before the second valve opens.

(b) The water slug formed by condensed pressurizer steam in the relief line is largely discharged through the undamaged valve. We note that this portion of the line showed little or no support damage.

(c) The second valve, PCV-456, opens on continued pressure increase and the transient, combined with the large flaw in the valve yoke results in valve failure.

With this hypothesis, there is no reason to expect a hydraulic shock higher than in opening of the first valve hence piping displacement sufficient to damage supports might not yet have occurred.

(d) The first valve closes automatically upon a reducing pressure signal before pressurizer water level reaches 100%.

(e) Water discharge occurs upon filling the pressurizer creating a substantial hydraulic shock in the relief line. Since the undamaged valve has already closed, the resultant pipe displacement was most pronounced in the portion of line where the damaged valve is located.

Other scenarios can also be postulated, but none has sufficient support of evidence to permit identification of a single sequence of events as the cause of observed damage.

(3) The events which lead to complete filling of the pressurizer and the second water discharge through the relief line required more than a single failure :

(a) The failure of all the secondary steam dump valves to operate.

(b) The failure of the pressurizer relief valve to close. It is likely that such a failure would not have occurred even with an initial hydraulic shock without existence of a large flaw in the relief valve yoke.

(4) Considering the valve PCV-456 itself, when in the open position, there is a spring force producing a tension of about 60,000 to 80,000 Newtons in the yoke. When the disk lifts, this force can be amplified due to dynamic effects. The presence of the flaw in one of the arms overstressed that arm (area reduction and stress concentration), which caused it to break.

This caused a moment to be applied to the other arm, resulting in bending of the spindle and rupture of the base.. The broken metal surface appearance was typical of brittle failure with some polishing due to rubbing contacts following yoke separation. The yoke the rose about 2,5 cm, the normal stroke of the valve. With the broken yoke, the valve failed to close. Dynamic forces due to the free motion of the operator body may have contributed to damage to the support.

(5) Appendix A calculates the forces and stresses on the relief line piping in two locations, suspected to be among the most stressed. It is seen there that, within the calculation assumption the piping could have been marginally overstressed. However, since a dye penetrant check of the PVC-456 valve to pipe weld was reported to show no defect, we cannot see any reason to think that the plant would operate in unsafe condition with the line in the present state. This statement assumes of course that all the support system of the piping will have been returned to its design condition before the reactor goes back to power.

To gain further assurance on the safety of the line we would recommend that a dye penetrant check of all welds near the fixed points be made at the earliest convenience. The locations include the pressurizer nozzle, the relief tank nozzle and the intermediate supported or restrained points.

b) Operational Considerations

(1) Plant operation with one pressurizer power relief valve closed off does not present a safety problem. The high pressure reactor trip and the pressurizer safety valves provide the necessary protection against overpressure of the reactor coolant pressure boundary.

The existence of the power relief valves is to prevent unnecessary opening of the main code safety valves during certain plant design transients.

(2) The safety injection system functioned normally with a reported total injection rate of 40 l/sec. The injected water raised the pressurizer level from 5% to 75%. Assuming the injection water to be initially at 16°C and atmospheric pressure in the RWST and to end up in the pressurizer at 285°C and 110 bars then the quantity of water leaving the RWST must have been about 10 $m^3$. This would cause a decrease in RWST level of about 0.7%. The injection time would be about 4.1/2 minutes assuming a constant injection rate.

(3) The reason why the turbotrol gear of turbine 2 dropped into the emergency mode is not known. It was reported that the effect of this would be to lock the turbine inlet control valves in their last position. Thus they would no longer respond to changes in steam pressure. This may account for the overpower excursion experience on turbogenerator 2 just prior to its tripping.

(4) The failure of the steam dump valves to open was reported to be the result of a wrong wiring connection which was not discovered during testing. The control circuitry of the steam dump valves had been out for maintenance at some previous date. Before being put back on line, the circuitry had been tested in two halves. Each half was checked independently of the other half and each half checked out satisfactorily. A fault at the interface of the two halves thus remained unrevealed.

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VI - OTHER RECOMMENDATIONS

1. The piping displacements and support damage which occurred have indicated the possibility that the pressurizer relief line was marginally overstressed. The likelihood is that the displacements resulted from either discharge of a water slug initially in the line or from relief of water when the pressurizer was completely filled.

   The initial evaluation of stress was deduced from observed support displacement and support bolt strains. As such, no definitive indication of possible stress levels with this transient exists as basis for an evaluation of fatigue damage for the entire piping length.

   We would recommend a dynamic analysis be performed, considering at a minimum the effects of the steam condensate initially in the line. The force time history function can then be used for evaluation of fatigue damage as well as the adequacy of restraints.

2. The failure of the power relief valve yoke is more probable due to the use of cast-iron materials of construction where impact properties are poor and flaws of the type involved in this failure can remain undiscovered.

   We therefore recommend such non-destructive tests as are feasible be made to ascertain that no flaws of this type exist in the valve currently installed.

   Further consideration might be given to replacing these yokes with a less brittle material.

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3. The test procedures following maintenance of the control system to the steam dump valves should be rewritten to eliminate the possibility of unrevealed faults.

4. It would be useful to provide means (i.e. 2 separate alarms : one actuated by the uncompensated pressure signal and the other by the compensated pressure error signal) in order to know if certainly each pressurizer relief valve opens during a pressure excursion.

APPENDIX A

## Stress and Force Evaluation in the pipe between valves

### 1. Damage to the support

The two bolts on the right side on figure 3 were strained about 3 mm. The two bolts on the left side were also strained but only to the point of getting loose.

### 2. Evaluation of the moment applied to the support

Bolt size : M10 → Shaft size (diameter)

$$8.888 < d < 9.128 \text{ mm}$$

(Catalogue MARC-GERARD - 1970)

Section (average) $\frac{\pi}{4}\left(\frac{8.888 + 9.128}{2}\right)^2 = 63.73 \text{ mm}^2$

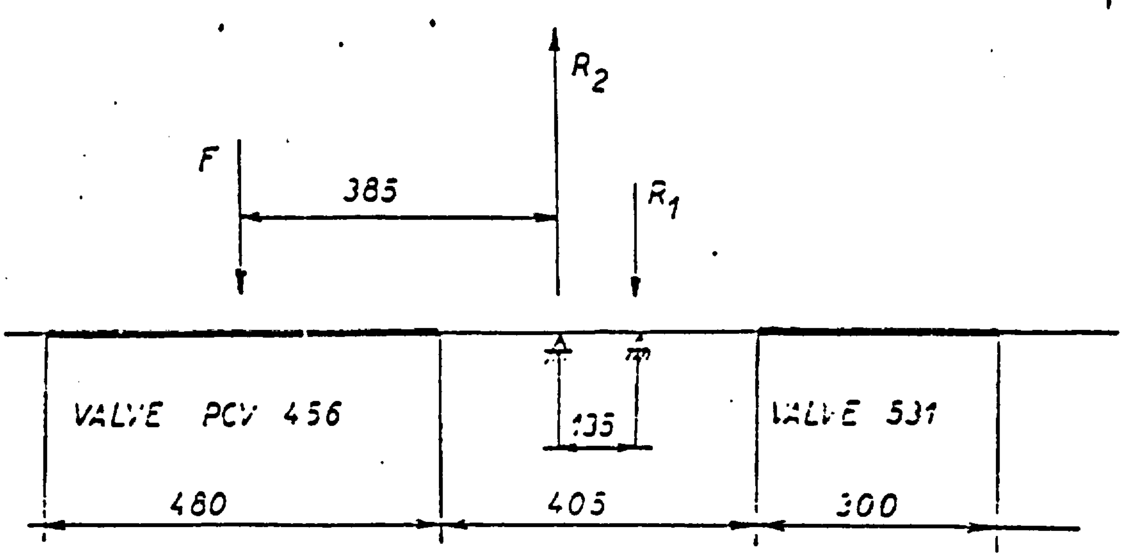
Assume for the bolt material a yield stress of

$$\sigma_Y = 32 \text{ kg/mm}^2$$

Hence the moment to strain the two bolts is

$$M = 63.73 \times 32 \times 2 \times .135 = 550.6 \text{ kg.m}$$

### 3. Force required to create that moment



If one neglects the effect of the supports located downstream of valve 456, one can write the equation

$$385*F = 135*R_1$$

Knowing that $R_1 * .135 = 550.6$ kgm

Hence F = 1430 kg

It is felt that such a force is in the possible range.

4. Stresses in the pipe (Primary stresses only)

Pipe : 3" sch 160

Hence : OD = 3.5 in = 88.9 mm t = 11.13 mm

$$\text{Bending modulus} = \frac{I}{v} = 47.17\ 10^3\ \text{mm}^3$$

Bending stress :

$$\sigma_B = \frac{M}{I/v} = \frac{550.6\ 10^3}{47.17\ 10^3} = 11.67\ \text{kg/mm}^2$$

Pressure stress (ASME III, Article NB 36 52)

$$\sigma_p = \frac{p*OD}{2t} = \frac{164.5*10^{-2}*88.9}{2*\ 11.13} = 6.57\ \text{kg/mm}$$

Combination (Article NB 36 52)

$$B_1 \frac{PD_o}{2t} + B_2 \frac{D_o}{2I} M_i$$

$B_1$ and $B_2$ are taken from table 3683.2-1

$B_1 = B_2 = 1$

Hence $\sigma_{tot} = 6.57 + 11.67 = 18.24\ \text{kg/mm}^2$

## 5. Allowable stresses

SA 376 Grade 316

$S_m$ at room temp. = 20 ksi = 14 kg/mm$^2$

$S_m$ at 650°F (=343°C) = 16.6 ksi = 11.6 kg/mm

Allowable stress = 1.5 $S_m$ (ASME III, article NB 36 52)

1.5 $S_m$ = 21 kg/mm$^2$ (room temperature)
= 17.4 kg/mm$^2$ (343°C)

## 6. Conclusion for primary stresses in the pipe

Since it appears that hot fluid has been carried by the pipe for a time of about 3 min, the hot allowable stress needs to be taken. Then it appears that the actual stress is slightly higher than the allowable :

$$18.24 > 17.4 \text{ kg/mm}^2$$

It should be noted that the figure of 18.24 kg/mm$^2$ is a minimum, since it corresponds to the plastification of the support (M = 550.6 kgm).

## 7. Primary and Secondary stresses in the pipe

The evaluation of secondary stresses (article NB 3653.1) requires the knowledge of the temperature gradients in the pipe. It was thus not possible to evaluate these stresses.

## 8. Primary stresses at the reducer

Bending moment

$$M = 1430 \times \left(385 - \frac{1}{2}(405 - 135)\right) \text{ kg mm}$$
$$= 357 \text{ kgm}$$

reducer $2\frac{1}{2}$ " sch 160

OD = 2.875 in = 73.02 mm     t = .375 in = 9.52 mm

$\frac{I}{v}$ = 1.64 in$^3$ = 26.9 cm$^3$

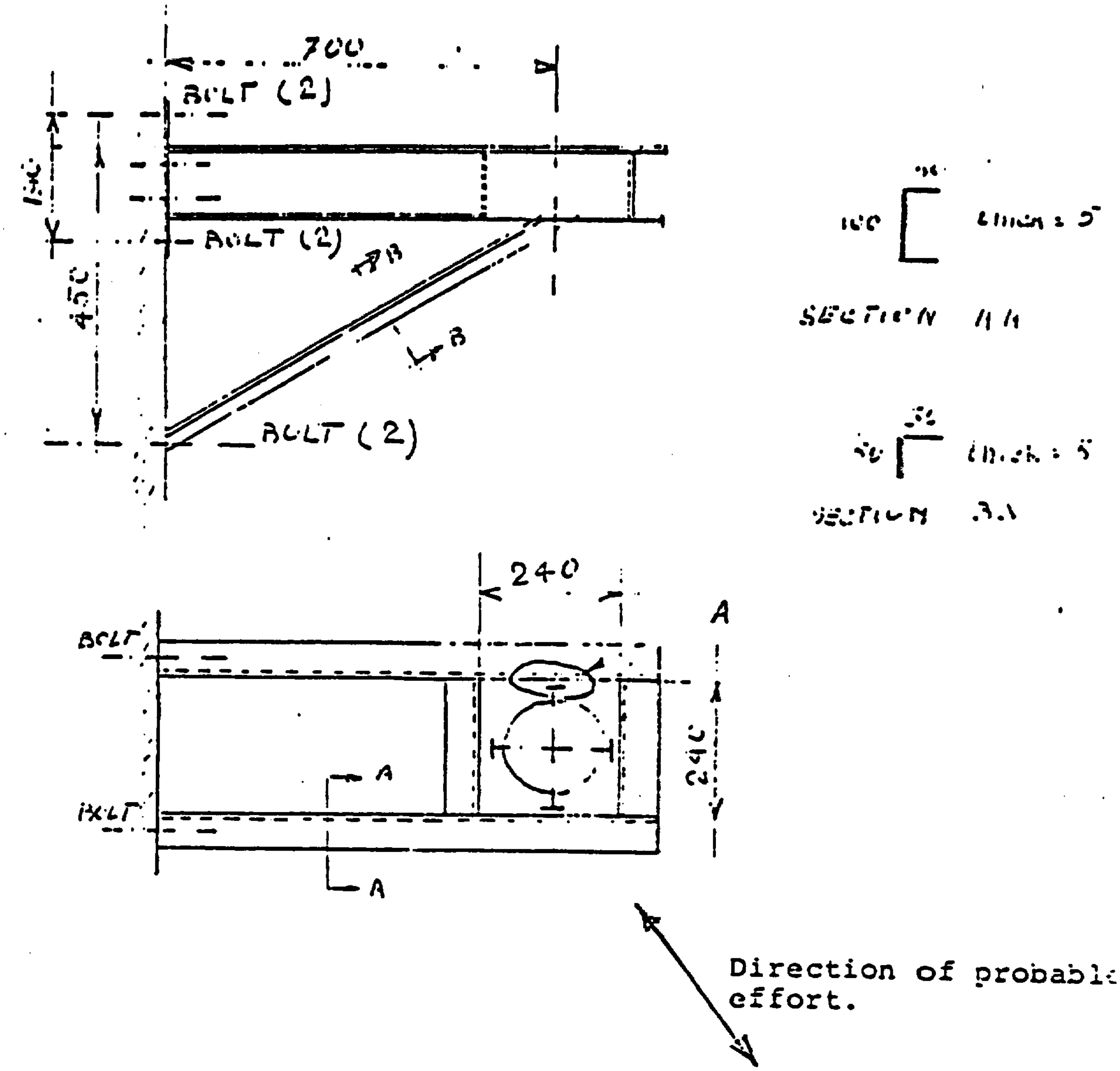
Pressure stress = $\frac{p*OD}{2t}$ = 6.28 kg/mm$^2$

Bending stress = $\frac{M}{I/v}$ = 13.28 kg/mm$^2$

Total stress = 19.56 kg/mm$^2$

This stress should be considered more as indicative since it depends so much on the assumption of the force location.

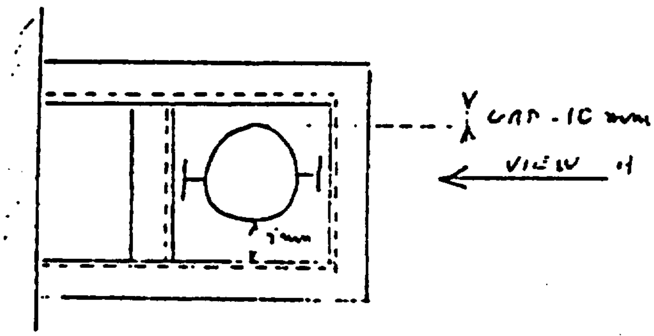
The same conclusion holds as for the pipe stress.

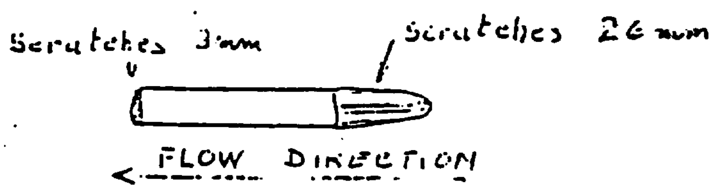



Bolts (6 total) : Hexagonal head = 25 mm

<u>Damage</u> : - no general distortion
- no rubbing evidence
- contact evidence in A

Figure 2 - Restraint R-1

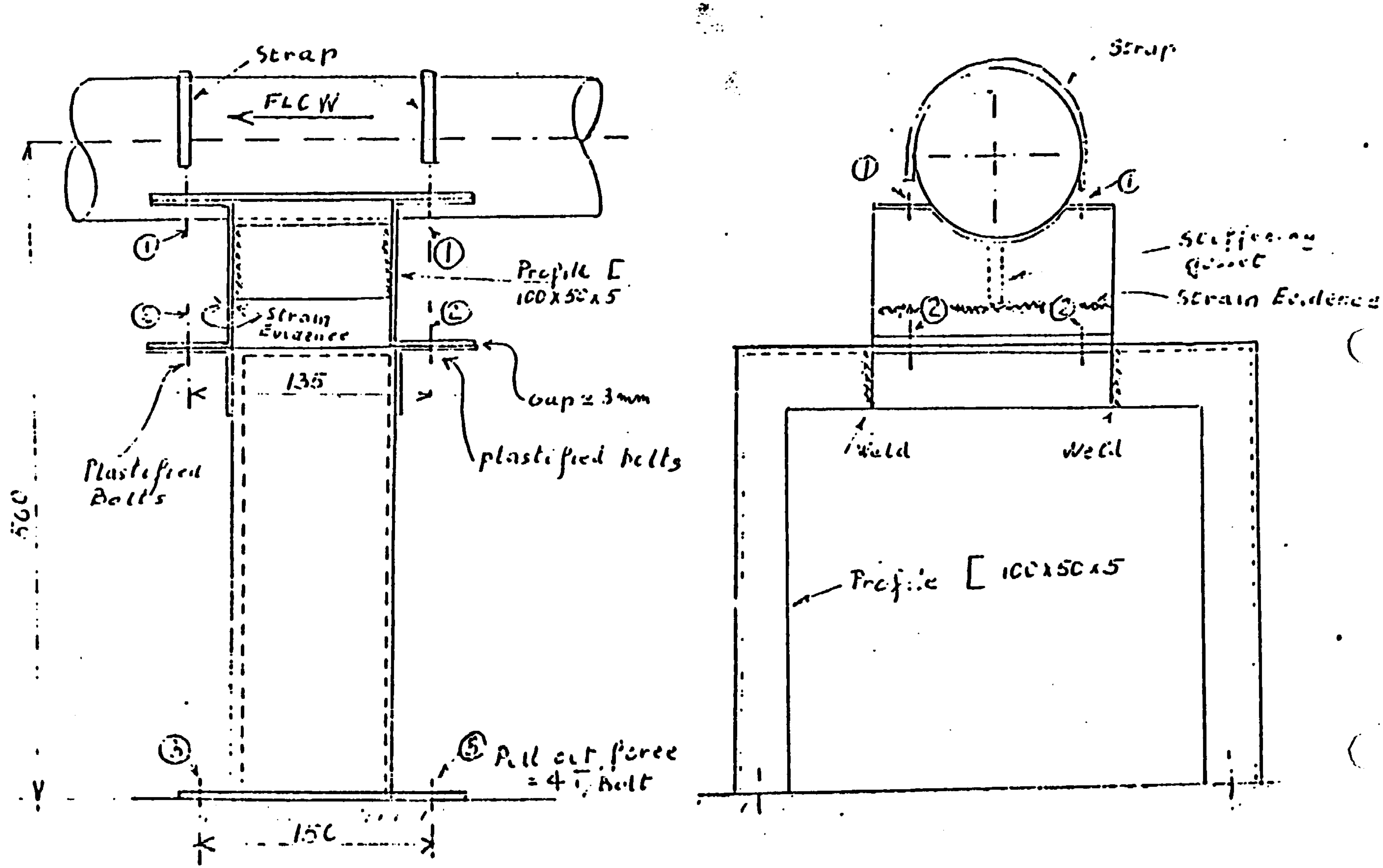



View A (marks on the shoes)

Damage : - top of insulation slightly smashed
- scratches on shoes as shown on view A

Figure 3 - Restraint R-2




Bolts 1 (4 total) M-10
2 (4 total) M-10
3 (4 total) pull out

force = 4T/bolt

Damage : - no evidence at straps, pipe and bolts (1) and (3)
- all 4 bolts (2) have been strained
- gap measured as shown
- strain evidence in the [ profile as shown

Figure 4 - Restraint R-3

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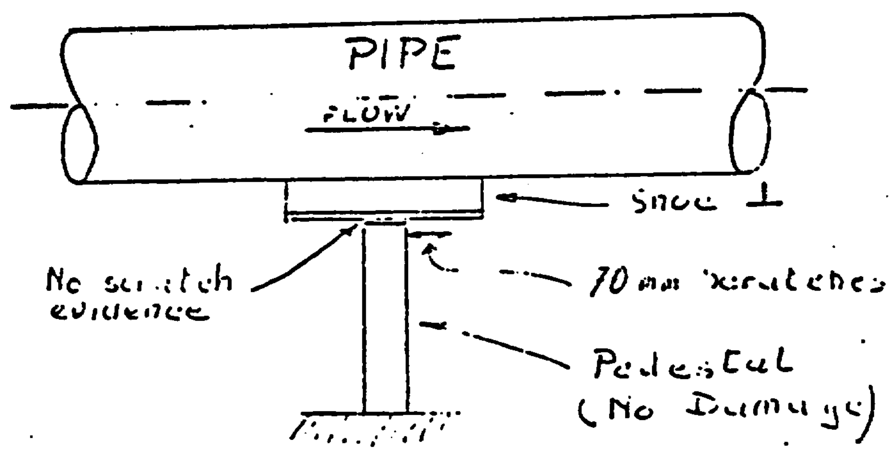



Figure 5 - Restraint R-5 Motion Evidence

BEZNAU - UNIT N° 1 (NOK)

STEAM DUMP FAILURE INCIDENT

Aug. 21, 74

PRESSURIZER RELIEF LINE

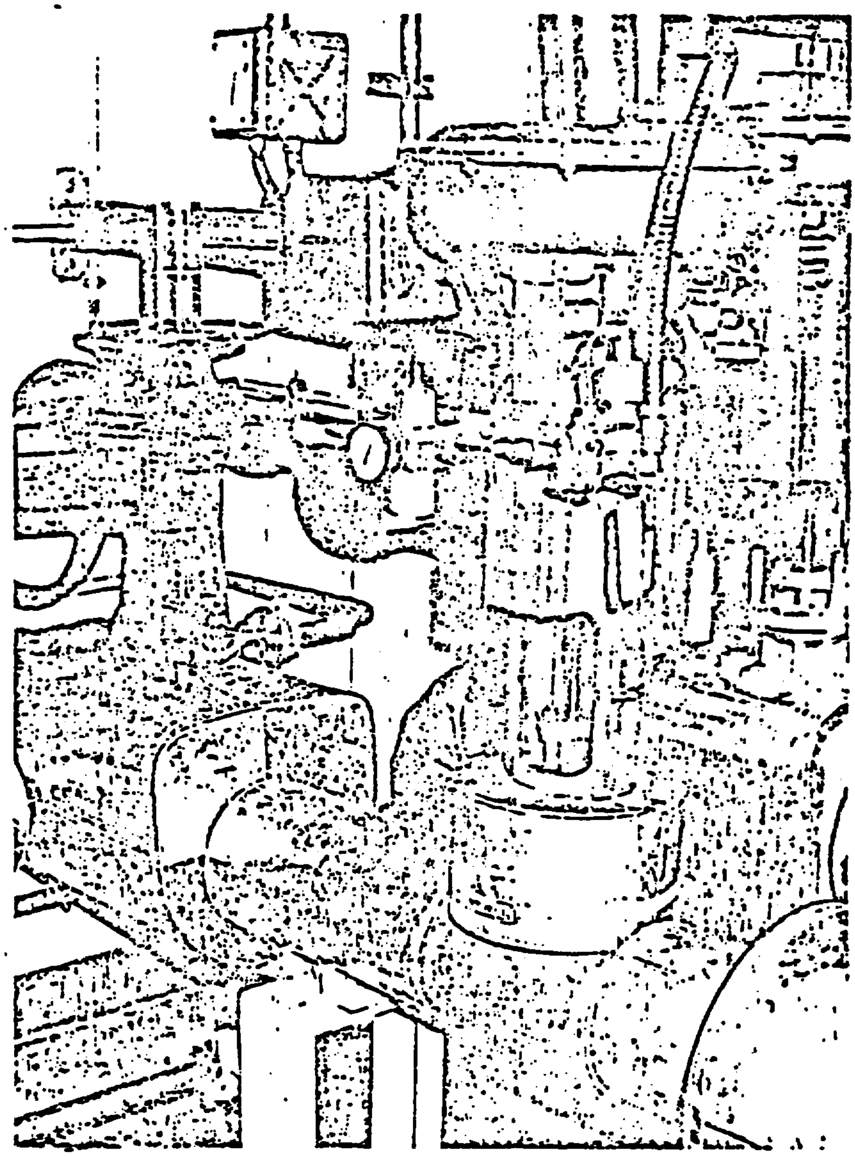


Figure 6 - Undamaged Relief Valve.

B_ [illegible] (NOK)

STEAM DUMP FAILURE INCIDENT

Aug. 21, 74

PRESSURIZER RELIEF LINE

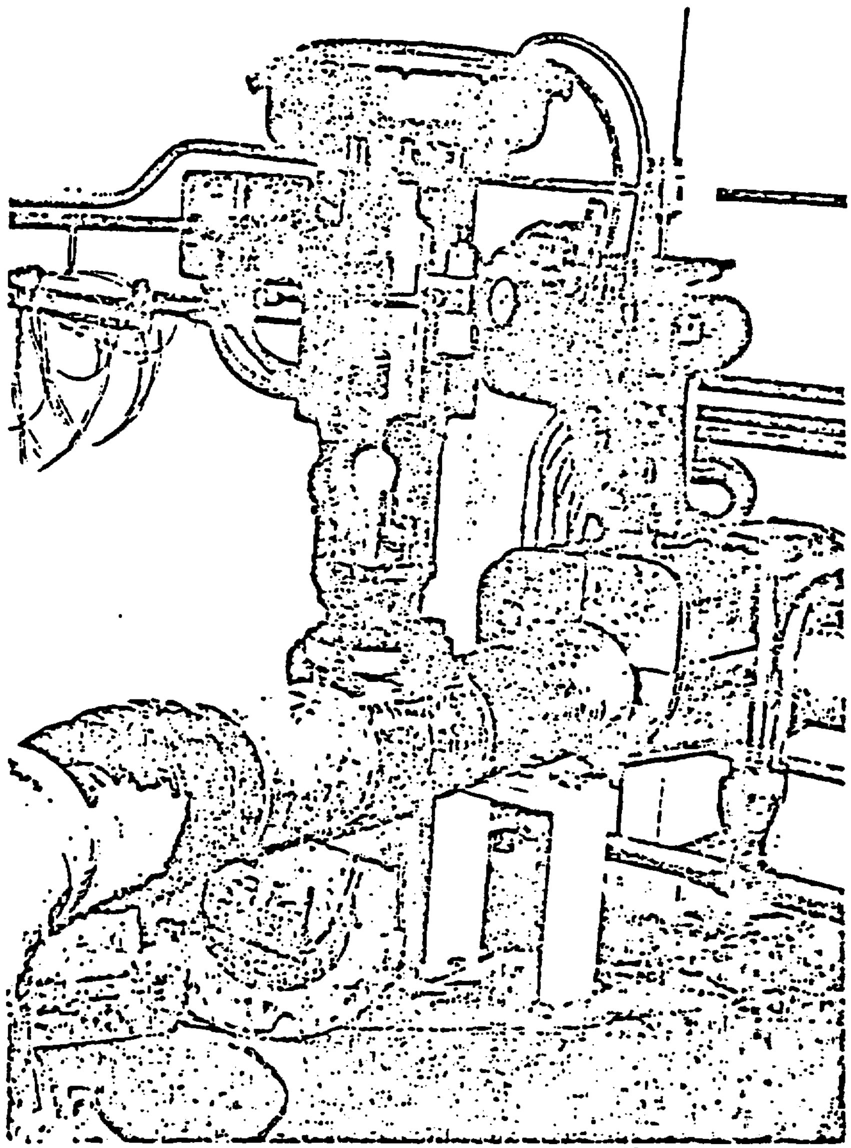


Figure 7 Damaged relief valve
General view showing the two fractured arms
and the liefted operator.

BEZNAU - UNIT N° 1 (NOK)

STEAM DUMP FAILURE INCIDENT

Aug. 21, 74

PRESSURIZER RELIEF LINE

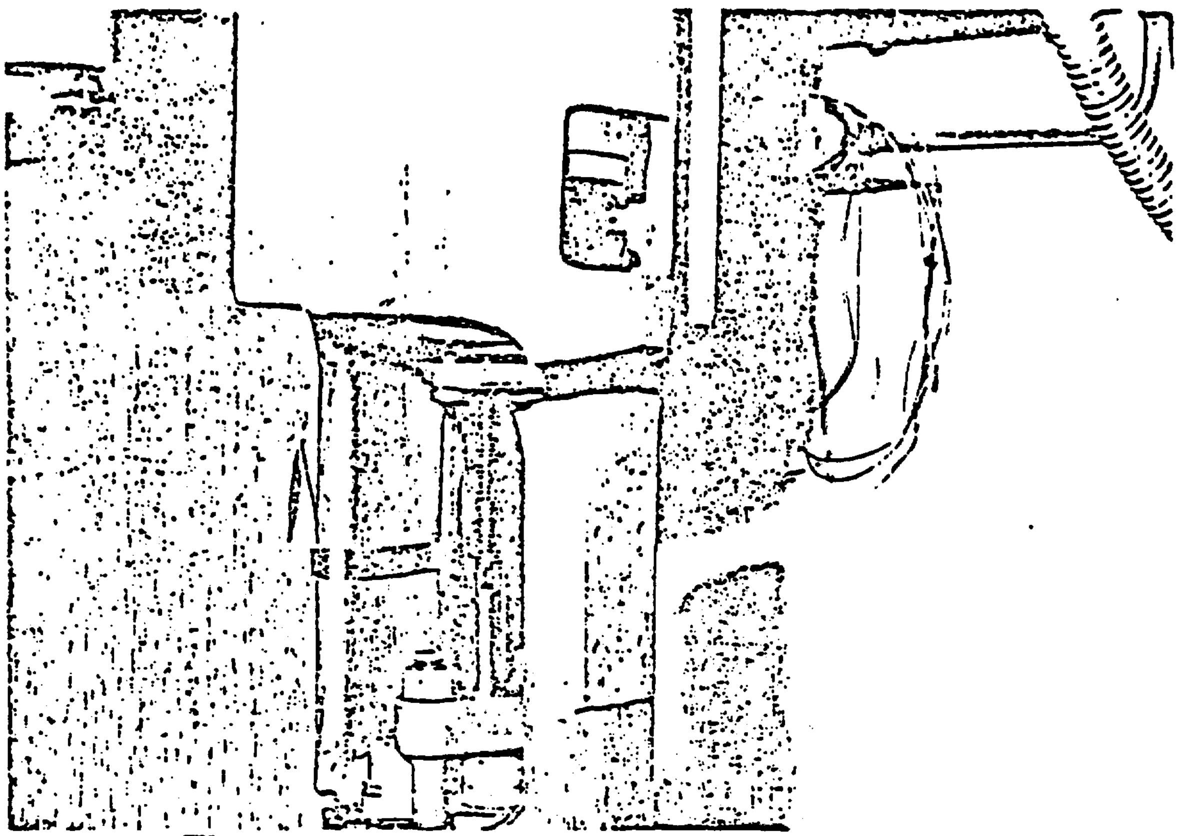


Figure 8 - Damaged Valve.

Detail of fractured yoke

BEZNAU - UNIT N° 1 . (NOK)

STEAM DUMP FAILURE INCIDENT

Aug. 21, 74

PRESSURIZER RELIEF LINE

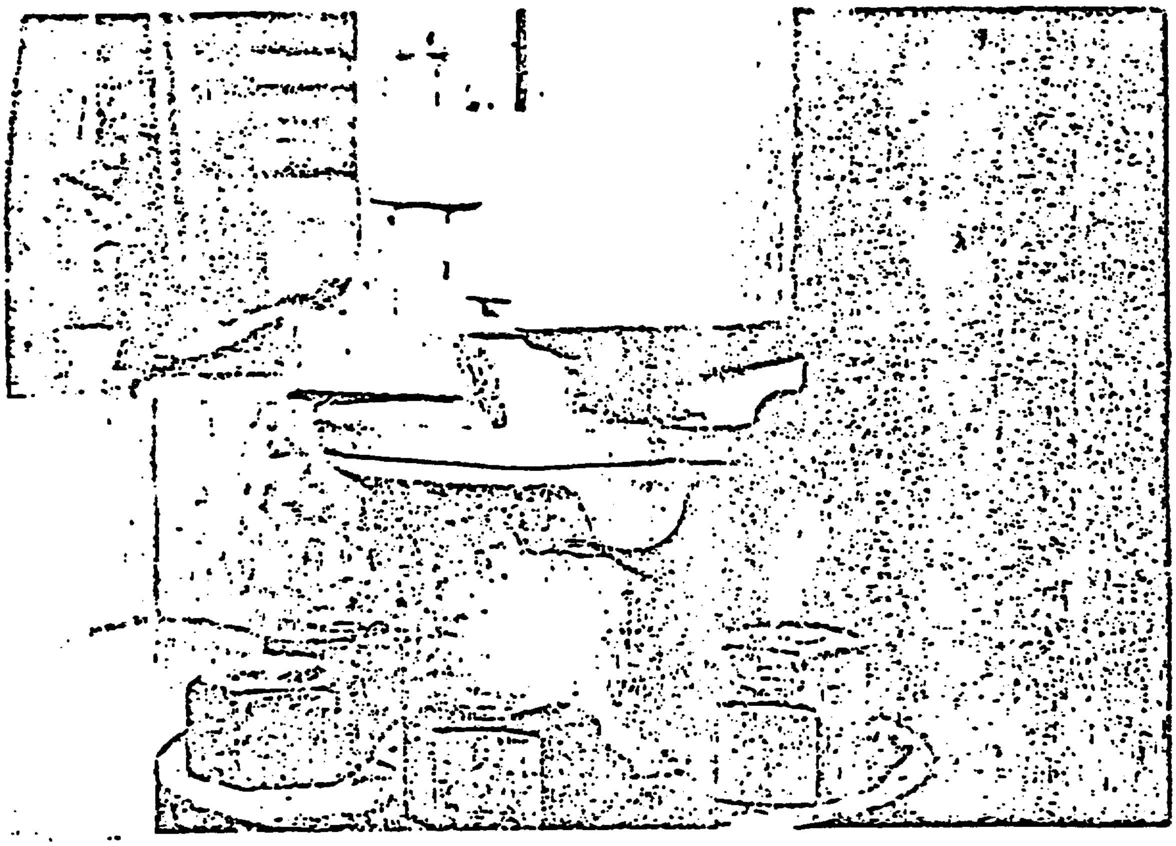


Figure 9 - Damaged Valve.

Detail of fractured bonnet.

BEZNAU - UNIT N° 1 (NOK)

STEAM DUMP FAILURE INCIDENT

Aug. 21, 74

PRESSURIZER RELIEF LINE.

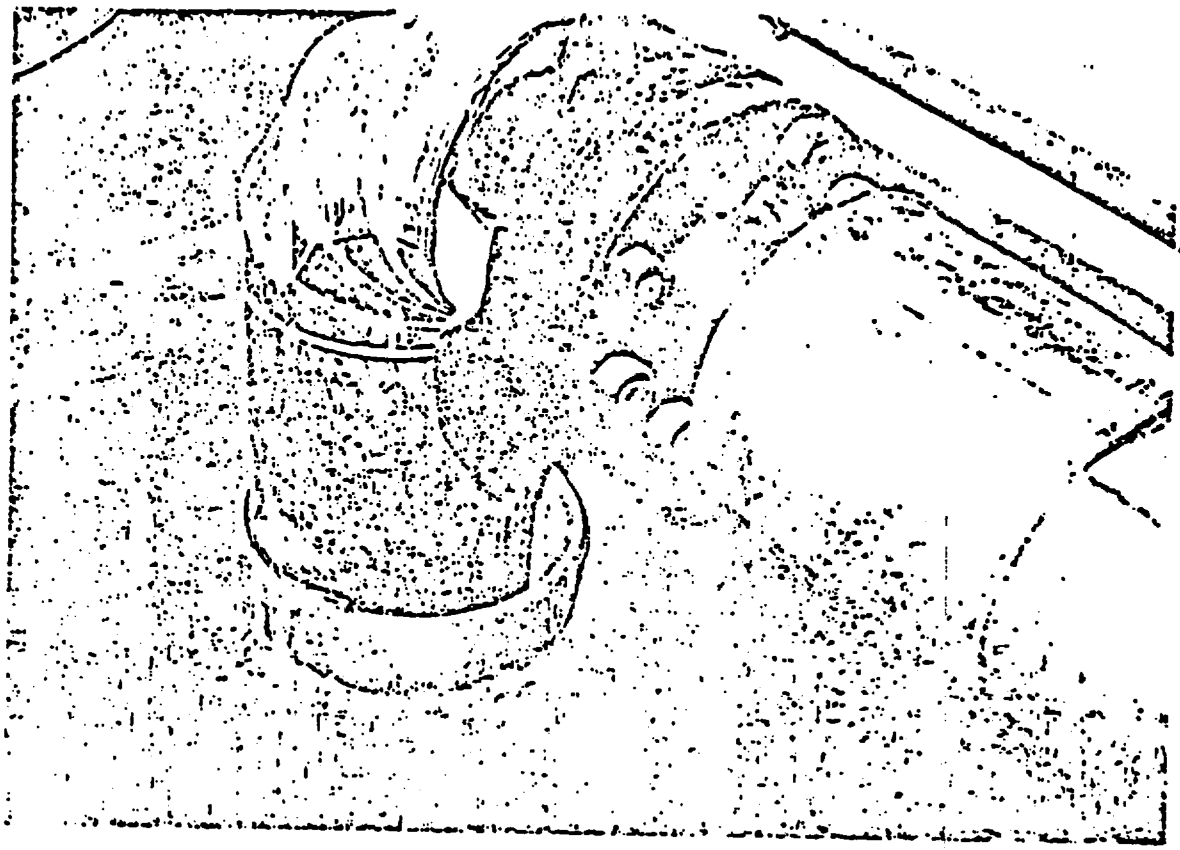


Figure 10 - Elbow after damaged valve.

BEZNAU - UNIT N° 1 (I

STEAM DUMP FAILURE INCIDENT

Aug. 21, 74

PRESSURIZER RELIEF LINE

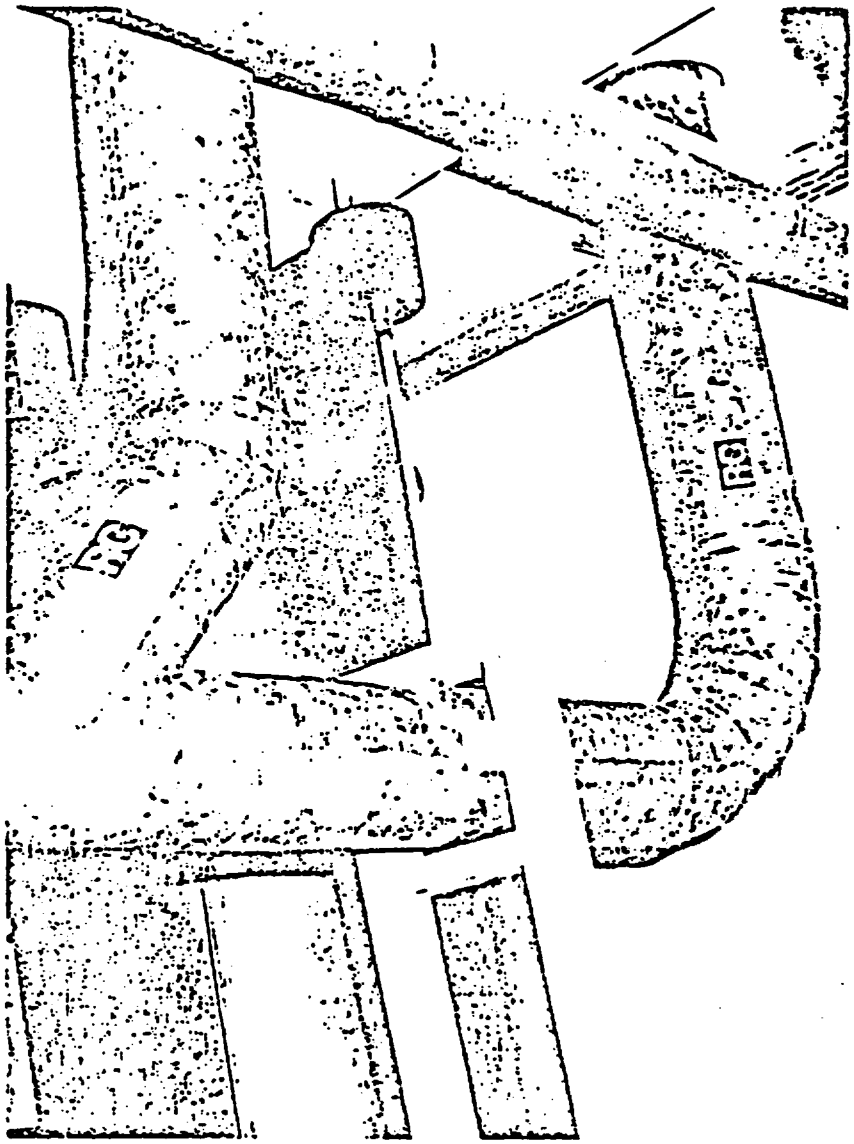


Figure 11 - Support R4 (1)
General arrangement

100 x 50 x 5 profiles
50 mm slot

top view

BEZNAU - UNIT N° 1 (NOK)

STEAM DUMP FAILURE INCIDENT

Aug. 21, 74

PRESSURIZER RELIEF LINE

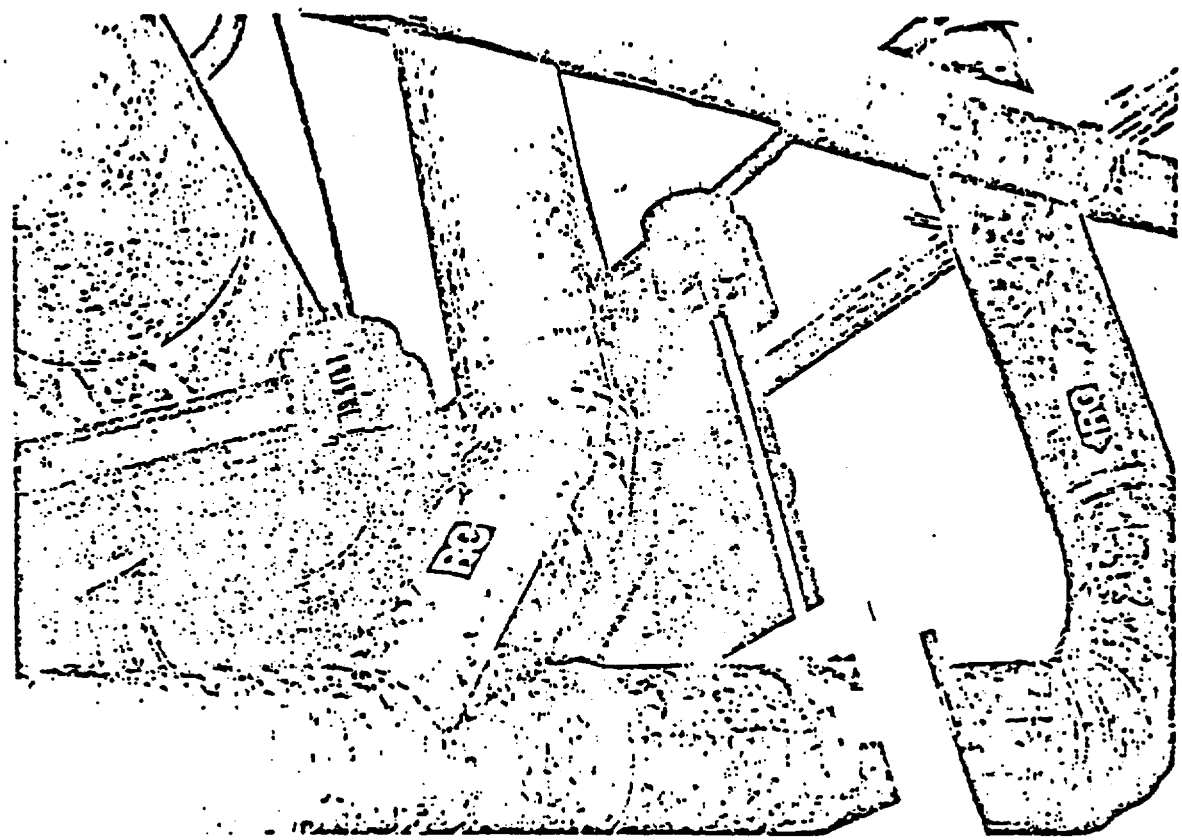


Figure 12 - Support R4 (2)

BEZNAU - UNIT N° 1 (NC

STEAM DUMP FAILURE INCIDENT

Aug. 21, 74

PRESSURIZER RELIEF LINE

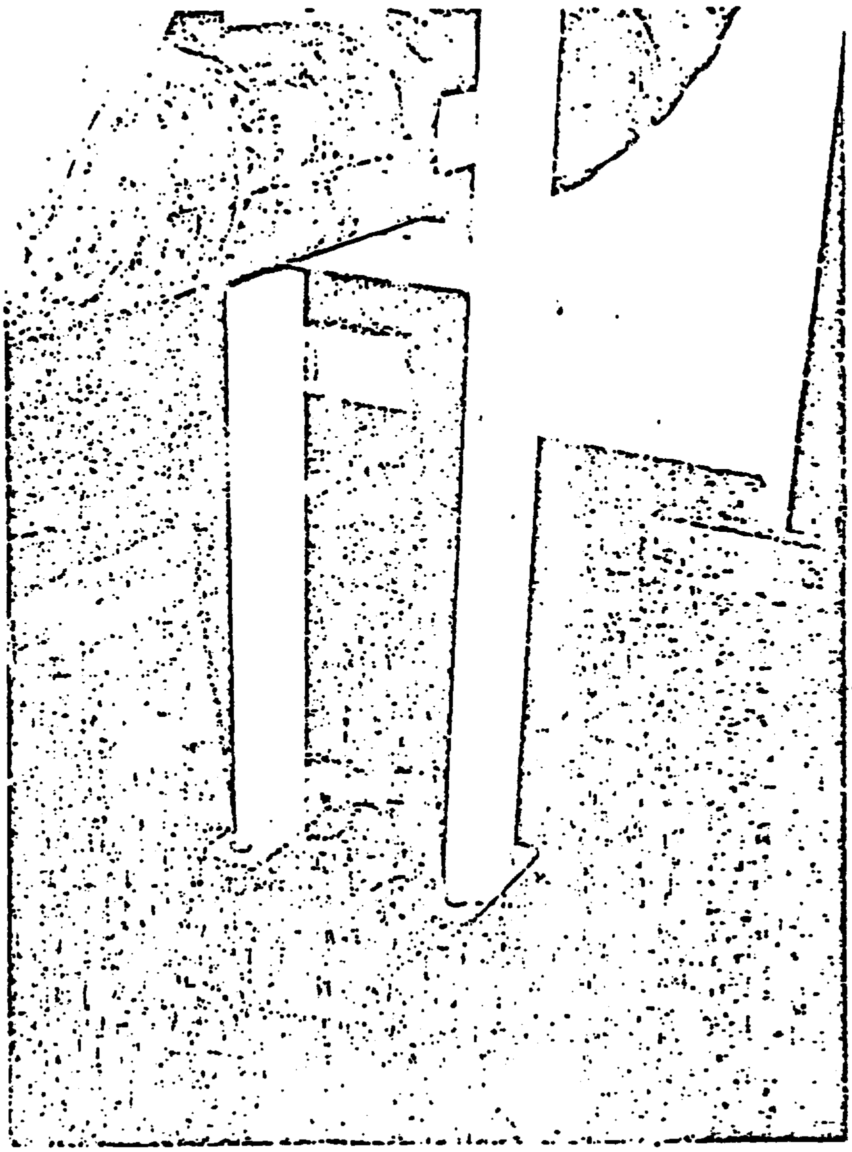


Figure 13 - Support R4 (3)
Attachment to floor
Concrete damage (back of the restraint)

F ?NAU - UNIT N° 1 (NOK)

STEAM DUMP FAILURE INCIDENT

Aug. 21, 74

PRESSURIZER RELIEF LINE.

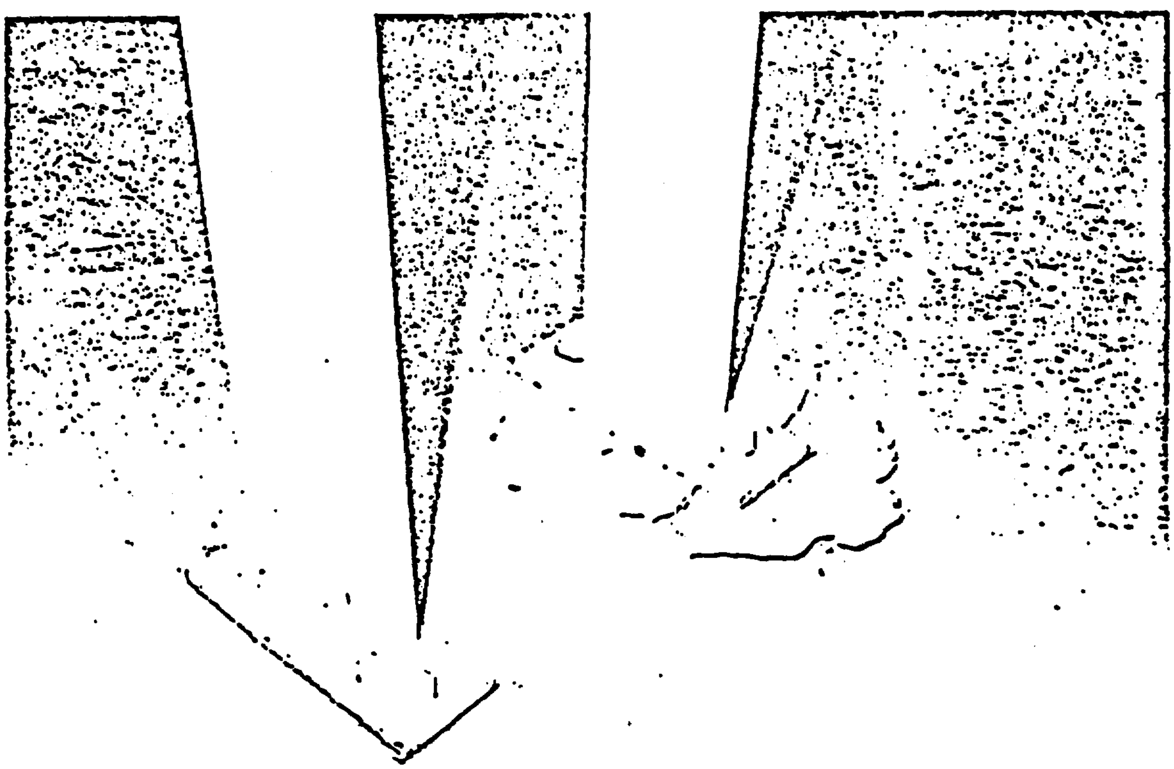


Figure 14 - Support R4 (4)

Detail of concrete damage.

BEZNAU - UNIT N° 1 (NOK)

STEAM DUMP FAILURE.INCIDENT

Aug. 21, 74

PRESSURIZER RELIEF LINE.

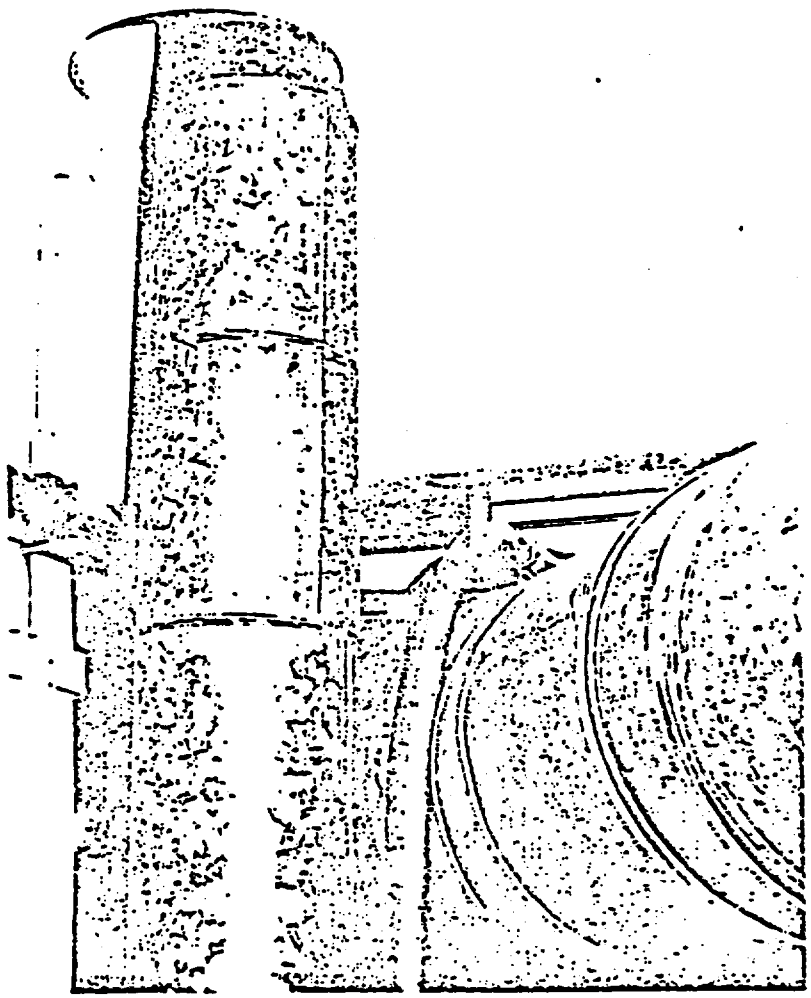


Figure 15 - Ceiling Penetration (1)

BEZNAU - UNIT N° 1 (NOK)

STEAM DUMP FAILURE INCIDENT

Aug. 21, 74

PRESSURIZER RELIEF LINE.

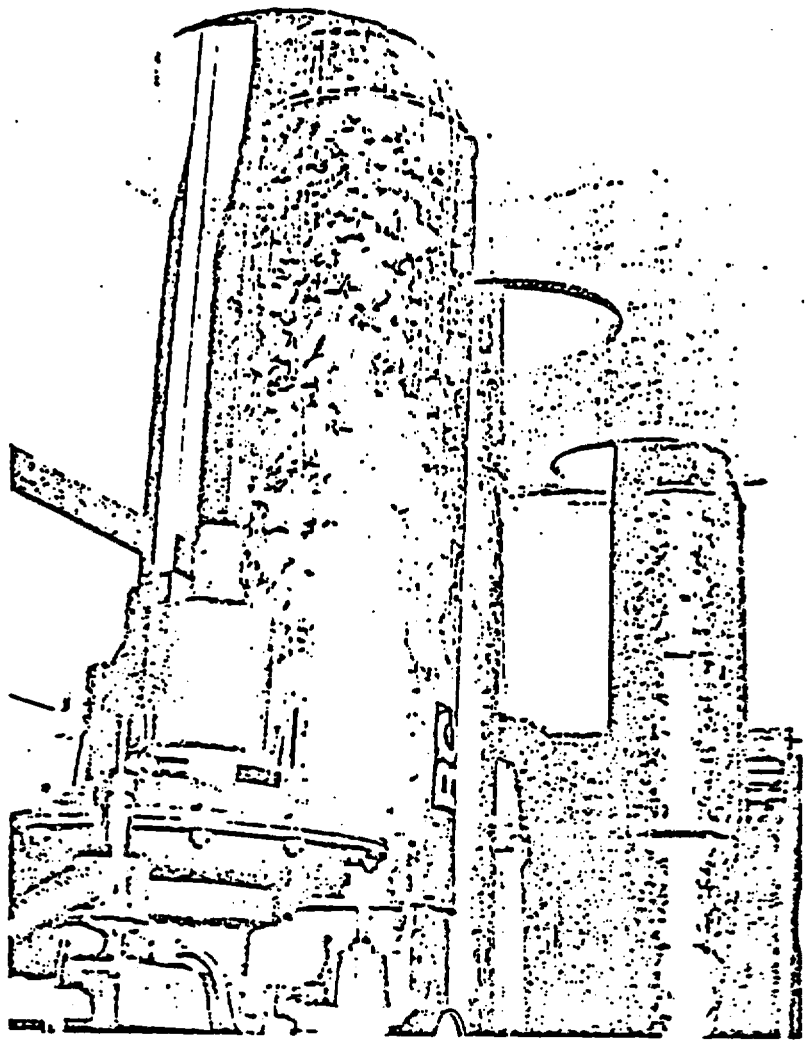
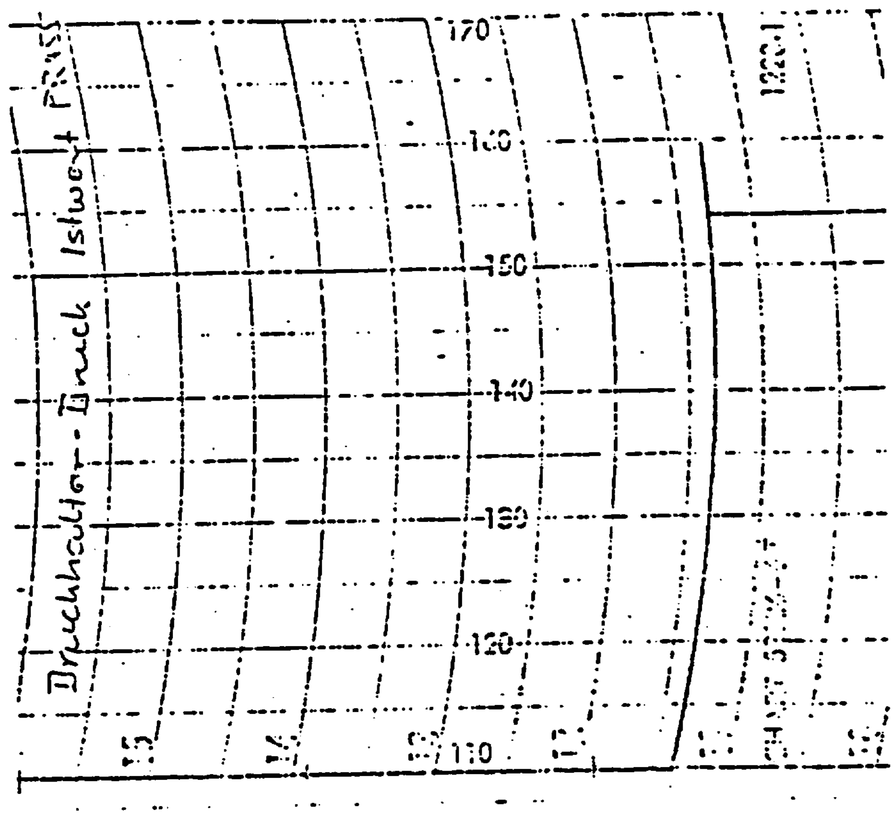


Figure 17- Ceiling Penetration (3)



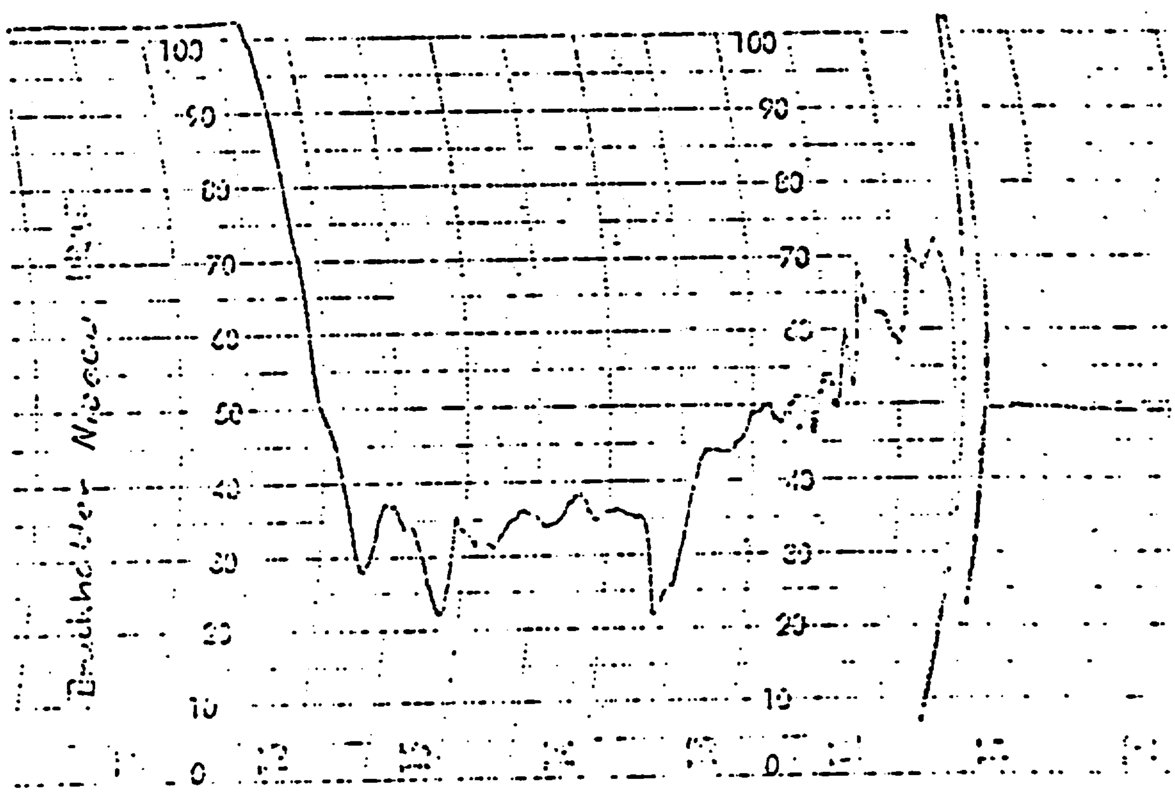




Figure 18 : Record of Pressurizer Pressure and Level Transients following incident initiation.

45 46 blank

NOK REPORT ON BEZNAU ACCIDENT OF AUGUST 20, 1974

1. TRIP TG-1/REACTOR TRIP/SI/

On August 20, 1974 at 11:20 a.m. a trip on turbine TG-1 occurred resulting to high bearing and casing vibrations (Bearing 6:60 )

At trip time, generator 2 was delivering about 140 MVar. Resulting from a failure of the steam dump system to operate, with the consequence that the relief valve did not open. That resulted in a rapid rise of coolant temperature, steam pressure and pressurizer level and pressure.

At 160 bar of pressure in the primary, the pressurizer pressure relief valves opened, lowering rapidly the pressure in the primary. About 10 seconds after valve opening, the pressure had reached such a low level that the pressurizer pressure relief valves were reactuated to close. Due to a disturbance, valve PCV-456, failed to close, resulting in a lowering of RCS pressure up to 100 bar after about 1 minute. Reactor tripped resulting from a low pressure signal (126.5 bar).

Due to the opening of the pressurizer relief valve, the pressure in RCS dropped to about 70 bar, corresponding to a saturation temperature of 284°C. Consequently, steam appeared in the primary hot leg, filling the pressurizer.

Two or 3 minutes after trip, the operator recognised the failure of the relief valve and isolated it with the power operated valve 531. The water level began to drop, and 11 minutes after trip, automatic SI was initiated by low pressure and level in the pressurizer.

SI systems worked normally and about 40 litres per second of water was spilled through the four SI pump nozzles into the primary, causing a rise of pressure to 110 bars and a further rise of level to 70 %. The SI pumps were then turned off and the power operated valves of the spray pipings were closed.

From that moment on, the pressurizer level could be controlle through charging pumps and release of steam, assuming the primary to cool down.

About 3 minutes after trip, the containment pressure alarm signal was actuated because of too high pressure, and 1 minute later the high activity alarm. Maximum pressure in containment reached 100 mbar over normal. The operators activated the containment fan coolers. Since several safety alarms of the pressurizer relief tank were on, it was quickly assumed that the rupture disc was broken and that the discharge channel was defectuous. After TG-1 trip, due to steam dump failure, steam pressure rose to 66 bar.

The turbatrol of TG-2 was actuated as an emergency after TG-1 trip. TG-2 was unregular in behaviour, and the position of the control valve remained constant during the pressure transient. The performances of TG-2 rose to about 214 MWe due to higher steam pressure (rise from 52 bar to 66 bar).

After TG-2 trip, following reactor trip, steam pressure rose to over 70 bar, actuating the safety valves and thus lowering pressure to about 65 bar.

## 2. CHRONOLOGICAL SEQUENCE OF EVENTS

August 20, 1974

## 2.1. Reactor Trip

| | |
|---|---|
| Beginning of incident | 11 h 20' 12" |
| TG-1 main breaker off | |
| Pressurizer pressure low-trip | 39,7" later |
| Reactor trip breaker open | 39,8" later |
| TG-2 main breaker off | 40,3" later |
| SI actuation (pressurizer pressure and level low) | 11'55,9" later |

## 2.2. Events as Registered on Alarm Typewriter

| TIME | | |
|---|---|---|
| 11:15 | TG-1 power high | 135,5 MVar |
| 11:20 | Allowable oil pressure of TG-1 too low | |
| 11:20 | Pressurizer pressure high. | 158.2 bar |
| 11:20 | Pressurizer pressure high. | 159.9 bar |
| | Reactor Trip. | |
| 11:21 | Tavg RCS-A high | 302.2°C |
| 11:21 | Steam pr. upstream of TG-1 stop valve high. | 66.3 bar |
| 11:21 | Tavg RCS-A high | 305.2°C |
| 11:21 | SG-A steam pressure high. | 67.3 bar |
| 11:21 | SG-B steam pressure high. | 67.2 bar |
| 11:21 | Steam pr. upstream of TG-1 stop valve. | 77.6 bar |
| 11:21 | SG-A steam pressure high. | 73.3 bar |
| 11:21 | SG-A steam pressure high. | 65.4 bar |
| 11:22 | Safety oil pressure of TG-2 too low. | |
| 11:22 | Tavg RCS-A | 285.2°C |

| TIME | | |
|---|---|---|
| 11:23 | Steam pressure upstream of TG-2 stop valve. | 68.1 bar |
| 11:23 | Pressurizer relief tank temperature high. | 62.8°C |
| 11:24 | Pressurizer level | 79 % |
| 11:24 | Pressurizer level | 88 % |
| 11:24 | Containment pressure high | 1.1 bar abs |
| 11:24 | Pressurizer relief tank level low. | 20.2 % |
| 11:24 | Pressurizer relief tank pressure high. | 0.59 bar |
| 11:25 | Pressurizer relief tank pressure | 0.15 bar |
| 11:25 | SG-A+B steam pressures normal. | 63.7 bar |
| 11:25 | Containment activity high | 17.3 mr/h |
| 11:26 | Loop B RCS flow low. | 88 % |
| 11:27 | Containment air temperature high | 53.4 °C |
| 11:32 | Pressurizer level low. | 6.8 % |
| 11:32 | Pressurizer level normal. | 18 % |
| 11:33 | Surge line temperature too low. | 271.1°C |
| 11:34 | Pressurizer level, high. | 58 % |

## 2.3. Sequence of Events for Pressurizer and Pressurizer Relief Tank

| TIME | | |
|---|---|---|
| 11 h 20' | 11.1" | Pressurizer pressure above control range. |
| | 11.9" | Pressurizer relief valve. |
| | 22.8" | Pressurizer relief tank pressure high |
| | 23.0" | Pressurizer relief valve locked |
| | 23.0" | Pressurizer pressure normal |
| | 23.1" | Pressurizer relief tank level high |
| | 24.2" | Pressurizer level high. |
| | 33.0" | Pressurizer relief tank pressure too high. |
| | 35.0" | Pressurizer pressure under normal. |

| TIME | |
|---|---|
| 11 h 21' 00.4" | Pressurizer pressure low - Trip. |
| 01.2" | Pressurizer pressure low - SIS unlocked. |
| 05.1" | Pressurizer relief tank level high. |
| 13.5" | Pressurizer pressure low - SIS unlocked. |
| 11 h 23' 27.6" | Pressurizer level high - 1 channel tr |
| 43.3" | Pressurizer relief tank level too hig |
| 43.5" | Containment pressure too high. |
| 47.1" | Pressurizer relief tank level low. |
| 11 h 24' 29.4" | Pressurizer relief tank pressure norma |
| 51.2 | Containment temperature high. |
| 11 h 25' 17.8" | Containment activity high. |

## 3. ANALYSIS OF THE CAUSES OF THE INCIDENT

TG-1 tripped due to high casing vibrations, especially in casing 6. It had already been noticed that TG-1 was sensitive to shocks. At the moment of incident, TG-1 was set to function under maximum effort, so that it could support a maximum of vibrations.

The trip is not unfamiliar and would not have affected the primary if steam dump had normally been actuated.

An inspection of containment after primary had cooled down, showed that the yoke between the PCV-456 valve housing and air engine was broken, and probably due to a dynamic effort on the piping at opening of the valve.

Consequently, the valve failed to close and initiated a rapid fall of pressure in primary. The pressurizer relief tank rupture disc broke, due to a prolonged surge of primary coolant in the tank. Items 2 and 3 show the disc broke when the relief valve had already closed.

WATER COLLECTED IN CONTAINMENT SUMP

| | | |
|---|---|---|
| Regen. hold up water Tank A | 38 % - 100 % | = 9.8 $m^3$ |
| Regen. hold up water Tank B | 16 % - 36 % | = 3.2 $m^3$ |
| Total quanlity of water collected | | =13.0 $m^3$ |
| Pressurizer relief tank | 80 % - 19 % | =11.2 $m^3$ |
| Water out of system. | | = 1.8 $m^3$ |

Since no further damage was noticed in containment, it could be assumed these 1.8 $m^3$ of water were blown out.

## 4.1. Thermal Stresses in RCS

Beside a rapid water temperature rise of about 6°C after TG-1 tripped, a rapid primary pressure rise from 154 bar to 160 bar, there was also an important temperature transient in area of SI nozzles. However, since the reactor's main pumps operated all the time, thus mixing cold spray water with hot coolant, it can be assumed that other components didn't undergo high temperature gradients. Furthermore, nozzle temperature and stress remained within design limits.

## 4.2. Damages to Relief Systems

During inspection in containment after cooling of primary, the following damages in the pressurizer relief systems were observed :

- relief valve PLV 456 : Mechanism broken on both sides and bent spindle.
- One anchor point of the relief system piping after valve was loose.
- Relief tank pressure disc broken.

Further damages in containment were not noticed.

It must be said that the relief tank is not designed to accept steam from the pressurizer for a prolonged time. The damages to the relief valve is therefore a direct cause to the breaking of the rupture disc.

4.3. Turbines

TG-1
The cause of vibrations to the casing are most probably the stresses and shocks. The P signal from hydrogen seal oil system is due to casing vibrations.
Damages to the seal or casing are most improbable.

TG-2
The oscillation from 172 MWe to 110 MWe, and then to 215 MWe suggested that the bolts of the high pressure cylinder were loosened and had lost some of their tension.
A too small stress was noticed, due to leakage of the seals of the high pressure cylinder. Due to too high rotational momentum at 215 MWe, the coupling between turbine and generator was closely controlled.

5. When reviewing the sequence of events, the failure of two systems, namely the steam dump and the pressurizer relief system, we came to the conclusion that it did not bring to an uncontrolable nor a damaging situation. During the incident, no activity (in gas or liquid form) in the surrounding area reached an uncontrollable level.

The generator safety valves maintained the steam pressure within allowable limits. The SIS brought back the primary to a safer pressure, allowing normal cooldown conditions.

6. PROPOSAL FOR MODIFICATIONS

6.1 Control of generator 1

Generator 1 reaching rapidly to casing vibrations, it will be tried to see if the regulator can be modified in order to have a quick action.

6.2. Pressure Regulator

Tests will be made to see if the first row of impellers in the pressure regulator of the turbine must not be reviewed in order to limit power to 190 MWe.

6.3. Steam Dump System

a) Revisions and calibrations should be made in steam dump system (before opening of steam dump valve.)

b) Studies will be made, to make periodic controls of steam dump while in operation. It should help to insure better safety limits (for example : unwanted opening of steam dump valve).

c) A control type writer linked to the steam dump will be installed in order to control the opening of steam dump valves and to check the good working of oil pumps.

6.4. Pressurizer Relief System

The first measure to be taken, is to repair the damaged valve, the piping supports and review boltings.
The pressurizer relief tank rupture disc must be replaced.
With these repairs start-up should be possible.

To see how the relief system piping can be better secured and how shock at opening of relief valve can be avoided are further measures to be taken.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

MAY 1 5 1979

MEMORANDUM FOR: D. F. Ross, Jr., Deputy Director, DPM

FROM: Ashok Thadani, Task Manager

SUBJECT: STUCK OPEN POWER OPERATED RELIEF VALVE AT FOREIGN PWR

In the process of gathering data on power operated relief valves (PORVs) for our report on Westinghouse plants, we were informed by Westinghouse that they were aware of only one instance of a PORV failing to reclose after opening. No failure of this nature had been observed on any U.S. reactor plant designed by W. The failure, according to W, occurred at one of the NOK reactors in Switzerland. Our survey of all operating U.S. W reactors also indicates that the failure of a PORV to reclose has not been observed on any U.S. Westinghouse reactor.

To follow up on the apparent foreign reactor PORV failure, we contacted Howard Faulkner of NRC International Programs and informed him of our need for additional information. Our basic need was to determine whether this failure did indeed occur and, if so, if it could occur on a U.S. PWR (due to similar system and component design).

A phone conversation between NRC (H. Faulkner, Ashok Thadani and Scott Newberry) and the Swiss Federal Office of Energy of Switzerland was arranged for the morning of May 15 to obtain this information. Howard Faulkner informed the Swiss that we would treat this information as confidential and would telecopy them a copy of what we intended to include in our W evaluation report prior to its issuance.

A sequence of events for the turbine trip and associated PORV failure to close described by Mr. F. Weehuizen, Head of Energy Section, is attached.

We requested additional information to supplement that in the phone conversation:

1. Event reports pertaining to the event

2. PORV description, manufacturer and failure mode



(55)

Based upon this phone conversation, we note that:

1. As demonstrated by this event, pressurizer level will remain above the trip set point for ECCS actuation for a stuck open PORV. ECCS did not actuate automatically until the operator shut the PORV isolation valve.

   In this case we do not know how soon the coincident signal (Lo Level/ Lo Press.) would have automatically initiated HPI and the subsequent operator actions since the PORV was isolated at 7 minutes.

2. The indications in the control room of actual PORV position and relief tank parameters appear to have provided the operator with sufficient information to make a reasonably rapid assessment of the problem and take appropriate action.

   Since this event occurred about five years ago and because of its relevance on our current deliberations on W designed plants, we recommend that complete information package including plant data be obtained and reviewed, as well as the role of the operator.

We therefore recommend that all operating Westinghouse reactors modify the pressurizer level/pressure coincidence ECCS actuation as already directed by I&E bulletins 79-06 and 79-06A and that we continue to pursue the PORV design, manufacturer and transient sequence to make a determination as to the likelihood of this event on a U.S. PWR and to obtain more information on turbine bypass system failure modes as a lower priority consideration.

A C Thadani

A. Thadani
Task Manager

cc: E.G. Case
R. Mattson
R. L. Tedesco
T. Novak
H. Faulkner
S. Newberry

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# Enclosure

1. Trip of 1 turbine due to generator disturbance (plant has a twin turbine arrangement - only 1 turbine tripped - no direct reactor trip unless both turbines trip)

2. Secondary system pressure increased - turbine bypass (5 relief valves to condenser) did not open due to a controller malfunction caused by operator error during previous maintenance period.

3. Primary system temperature, pressure and pressurizer level increase. PORV opens.

4. Primary pressure decreases. After 10 seconds PORV should have shut but remained open.

5. Reactor trip on low pressure (pressurizer level still above low level trip, therefore ECCS has not yet actuated on coincident low pressure - low level)

6. Reactor Coolant System pressure decreases to saturation. Voiding in hot legs. Operator observes flow oscillations and reactor coolant pump vibrations. He did not trip the reactor coolant pumps.

7. 2-3 minutes after the reactor trip, the PORV isolation valve was shut by the operator. He had received increasing pressure and temperature indication in pressure relief tank. He also had open indication of PORV (direct from limit switch on valve stem) in the control room.

8. High containment pressure alarm (~1.4 psig).
   High containment activity (pressure relief tank rupture disc ruptured).

9. Pressurizer level decreased. 11 minutes after the reactor trip, ECCS actuated on coincident low pressure/low level ECCS performed as designed

10. Pressure increased to 110 bars (~1600 psi).
    Pressurizer level increased to 70% of indicated range.
    Operator tripped HPI and maintained pressurizer level using charging pump (CVCS).

11. No core uncovery.
    No fuel damage.
    No hydrogen generation.

Additional Notes:

1. Main feedwater was maintained throughout the event.

2. Secondary system reactor trips are:

   - low steam generator level
   - both turbines trip.

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3. Total reactor coolant lost to containment sump = 1.8 cubic meters.

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SEP 13 1979

MEMORANDUM FOR: H. R. Denton, Director, NRR
E. G. Case, Deputy Director, NRR
D. Ross, Deputy Director, DPM
R. Mattson, Director, DSS

FROM: Darrell G. Eisenhut, Acting Director
Division of Operating Reactors

SUBJECT: INCIDENT AT BELGIUM DOEL 2 REACTOR

In response to our following up on a rather large, sudden steam generator tube rupture at the Doel 2 nuclear power plant in Belgium, we have received the attached report. You may find this incident particularly interesting since the unit underwent a transient where pressurizer level apparently went offscale high. Strip chart recordings of the event are enclosed.

We hope to be obtaining more information on this event in the near future.

Darrell G. Eisenhut, Acting Director
Division of Operating Reactors

Enclosures:
As Stated

cc: S. Hanauer
F. Schroeder
B. Grimes
P. Check
G. Lainas
S. Levine
V. Stello
W. Russell

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Nied 8/27

CENTRE D'ÉTUDE DE L'ÉNERGIE NUCLÉAIRE
C.E.N./S.C.K.
ÉTABLISSEMENT D'UTILITÉ PUBLIQUE

Veuillez adresser votre réponse
en deux exemplaires aux
LABORATOIRES DU C.E.N./S.C.K.
Boeretang 200 B-2400 MOL

Tel: (014) 31 18 01
Telex SCKCEN-Mol 31922
Adr. télégr.: Centratom Mol

SEP 3 1979

Mr. Joseph D. LAFLEUR, Jr.
Deputy Director
Office of International Programs
UNITED STATES NUCLEAR REGULATORY COMMISSION
WASHINGTON D.C. 20555
U.S.A.

MOL, le 21.08.79.

V/lettre

V/réf.

N/réf.
Centrale BR3
FM./mb
5.5126/71

Dear Dr. LAFLEUR,

As a first answer to the telex of Mr. H.J. FAULKNER NRC-BHDA, dated 8.8.79, I send you here enclosed a report describing the steam generator leak incident at the Unit 2 of the Doel nuclear power plant.

This report has been transmitted to me by "Tractionel Engineering", a division of the compagny "Société de Traction et d'Electricité" in Brussels ; as you most probably know, this division is playing the role of engineering office for the benefit of the Doel plant operator compagny (EBES).

I hope you will find in this report satisfactory answers to all your questions ; do not hesistate to ask for eventual additional informations.

Yours sincerely,

F. Motte

F. MOTTE
BR3 Plant Superintendent.

Enclosure : "Report on the incident at Doel 2 nuclear power plant Severe leakage in steam generator B on June 25, 1979".

(60)

PD/VEF
20.07.79

SEP 3 1979

# REPORT ON THE INCIDENT AT DOEL 2 NUCLEAR POWER PLANT

# SEVERE LEAKAGE IN STEAM GENERATOR B ON JUNE 25, 1979.

## 1. STATUS OF THE POWER PLANT AT THE MOMENT OF THE INCIDENT

The primary system was being heated up after repair works at the actuation system of the main steam valve.
At the moment of the incident, temperature in the primary system was ± 255°C (refer to point A on Fig. 1 & 2) and pressure had reached its rated value of 157 $kg/cm^2$ (refer to point A on Fig. 3 & 4).
The reactor was subcritical with all rods in.
Secondary pressure in the steam generators was ± 45 $kg/cm^2$, the saturation pressure corresponding to 255°C (refer to point A on Fig. 6 & 7).
For some time, A-loop steam generator had shown a low activity value along the secondary side (below admissible limits) that indicated a small leakage.

## 2. SEQUENCE OF THE EVENTS (refer also to various computer data given in attachment)

### 2.1. Initiating phase

About 7:20 PM, a quick pressure decrease is recorded in the primary system (about 2 $kg/cm^2$ per minute : see Fig. 4), which results in accelerating the operating charging pump. A second charging pump is started manually. The letdown

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station of the CV system closes automatically. It is confirmed that the relief valves are closed and their isolation valves are preventively closed. The level in the pressurizer quickly decreases (see Fig. 5) and the electrical heaters are automatically disconnected.
At the same time, a quick level increase is recorded in B-loop steam generator (see Fig. 7 point B). The activity measurement channels of the blowdown system record a maximum value.
The combination of all those signals indicates a severe leakage in B-loop steam generator. The faulted steam generator is then immediately completely isolated along the steam side and the discharge valve to the atmosphere is set at maximum pressure.

Meanwhile the third charging pump is started (was set apart to be maintained), but the three charging pumps are not sufficient to compensate the loss of fluid in the steam generator. Indeed, the CV tank is readily empty and the charging pumps are automatically supplied from the 2R11 refuelling water storage tank. To increase the subcooling primary pump B is stopped and letdown starts through A-loop steam generator (see Fig. 3, point B).

## 2.2. Actuation of safety injection

About 20' after the incident started, the threshold pressure (118.5 $kg/cm^2$) to actuate the safety injection is reached. The emergency diesels start within the required time lapse but are not necessary. Phase A isolation and ventilation isolation of the reactor building are achieved. The vital components not yet in operation are started.

When reaching the 108 kg/$cm^2$ value, all HP SI-pumps discharge into the primary system, and the pressure decrease is stopped (see Fig. 3, point C).
To prevent the secondary pressure in the faulted steam generator from reaching the opening pressure of the safety valves, the primary pressure is successfully decreased (see Fig. 3, point D) through maximum spray in the pressurizer (re-start of primary pump B and use of both spray lines).
During this phase, the level in the pressurizer quickly increases and it fills up completely (see Fig. 5). Spray is temporary stopped and pressure stabilizes at zero flow pressure of HP SI-pumps.
The automatically started auxiliary feedwater supply results in a pressure decrease in B-loop steam generator (see Fig. 7, point C). The auxiliary feedwater supply pump of the faulted disconnected steam generator is locally stopped and isolated (Fig. 7, point D). This cannot be performed from the control room since the SI-signal still prevails. The auxiliary feedwater supply tank is filled up from Doel 1.

## 2.3. Cancelling of SI-signal

Pressure decrease was now mandatory :

a) to avoid the opening of safety valves of the faulted steam generator.
b) to start, as soon as possible, the shutdown cooling system (low pressure circuit 1) to stop the letdown of slightly contaminated steam through the A-loop steam generator.

First,the safety injection signal had to be cancelled. This had to be performed more than once (each time requiring 5 minutes interval) because of a relay fault.

After definitively cancelling the SI-signal, two HP SI-pumps are stopped and soon thereafter a third one (Fig. 3, point F). While considering the subcooling margin, the last HP SI-pump is stopped. Pressure successively decreases to reach $\pm$ 65 $kg/cm^2$ (Fig. 3, point H) (saturation pressure is $<$ 15 $kg/cm^2$ at that moment).

It is then tried to initiate the CV-discharge line, but valves do not open. Some time goes by before the reason therefore is determined. Due to phase A isolation there is no longer a compressed-air supply in the reactor building. After re-opening the compressed-air supply line the discharge line is opened (Fig. 3, point I). Pressure decreases, first quickly, then slower.

The loss of compressed-air supply has also resulted in the closure of CC-valves to the primary pumps. The pumps have run for a long time without cooling of the thermal shield, however without alarm temperatures were reached.

2.4. Initiation of the residual heat removal system

As the CV-system permittted only a slow pressure decrease, the interlock, which maintains the isolation of the RHRS up to a pressure of 28 kg/cm$^2$, has been bypassed at 31 kg/cm$^2$ There was indeed a sufficient margin compared to the design pressure of the system (42 kg/cm$^2$). Thanks to this operation the letdown through A-loop steam generator could be stopped earlier and the discharge of slightly contaminated steam could be reduced (Fig. 3, Point J).

## 2.5. Further sequences

The abovementioned operation allowed a primary pressure decrease below the value of secondary pressure in the faulted B-loop steam generator. The secondary level decreases, which creates a dilution risk. The boric acid concentration is controlled every half hour (stabilized howerver at $\pm$ 1500 ppm).

Thanks to the cooling down, pressure decreases slowly in B-loop steam generator and reaches a value lower than the primary pressure. From this moment on, attention is paid to always maintain the primary pressure higher than that in the steam generator.

Despite the cold water so discharged in the steam generator, pressure goes on decreasing slowly (due to the presence of a warm water film at the water surface).

As the level of water in the steam generator approaches the upper limit of the broad level measurement pressure is sufficiently low ($\pm$ 12 kg/cm$^2$) to inject nitrogen.

The secondary drain line is coupled with system B for liquid waste, and the steam generator discharges into it through the nitrogen pressure.

The nitrogen is only slightly contaminated after this and can be discharged via the annulus between primary and secondary containments.

## 2.6. Comments and conclusion

The incident has been handled as prescribed and no damages have occured to the environment or the installation.
The procedures have to be reviewed considering the following :

- cancelling of phase A isolation to restore compressed air supply in the reactor building.

Attachment 1 - Computer data

1. Initiating phase

19 21'06" pressurizer pressure below reference pressure

19 22'51" demand for charging pump higher speed

19 23'31" disconnecting pressurizer heaters by low level

19 23'32" CV letdown station valves closed

19 25'42" closing of isolation valves of relief valves and spray valves

19 26'14" low pressure in primary system

19 30'30" very low pressure in pressurizer

10 30'30" high level in B steam generator

19 38'32" B primary pump disconnected

2. Safety injection phase

19 40'18" low pressure in pressurizer

19 40'19" safety injection through low pressure in pressurizer

19 40'19" diesels started

19 40'19" reactor building ventilation isolation

19 40'20" phase A reactor building isolation

19 40'24" actuation signal HP SI-pumps

19 40'33" HP SI-valves opened

19 43'28" very large auxiliary feedwater flow to A SG

19 44'39" very large auxiliary feedwater flow to B SG

19 53'12 auxiliary feedwater supply pump B disconnected

19 56'37" very low level in auxiliary feedwater supply tank

19 57'11" pressurizer level normal

19 57'29" pressurizer heaters re-started

19 58'48" high level in pressurizer

3. SI-signal cancelling phase

20 00'15" automatic starting signal of diesels cancelled and SI-pumps starting signal cancelled

20 00'21" back to SI

20 03'24" LP compressed air in reactor building
20 05'59" safety injection ordered
20 06'05" safety injection
20 10'59" reactor building ventilation isolation ordered
20 21'15" HP SI-pump B disconnected
20 25'22" HP SI-pump A disconnected
20 38'33" valve CC 096 closed
20 40'25" valve CC 099 closed
20 48'54" compressed air supply to reactor building restored
20 49'00" primary pumps CC-valves re-opened

4. Actuation of RHRS

22 35'54" valve RC 003 opened

*Temperature hot leg*

TEMPERATUUR WARM BEEN

FIGUUR 1

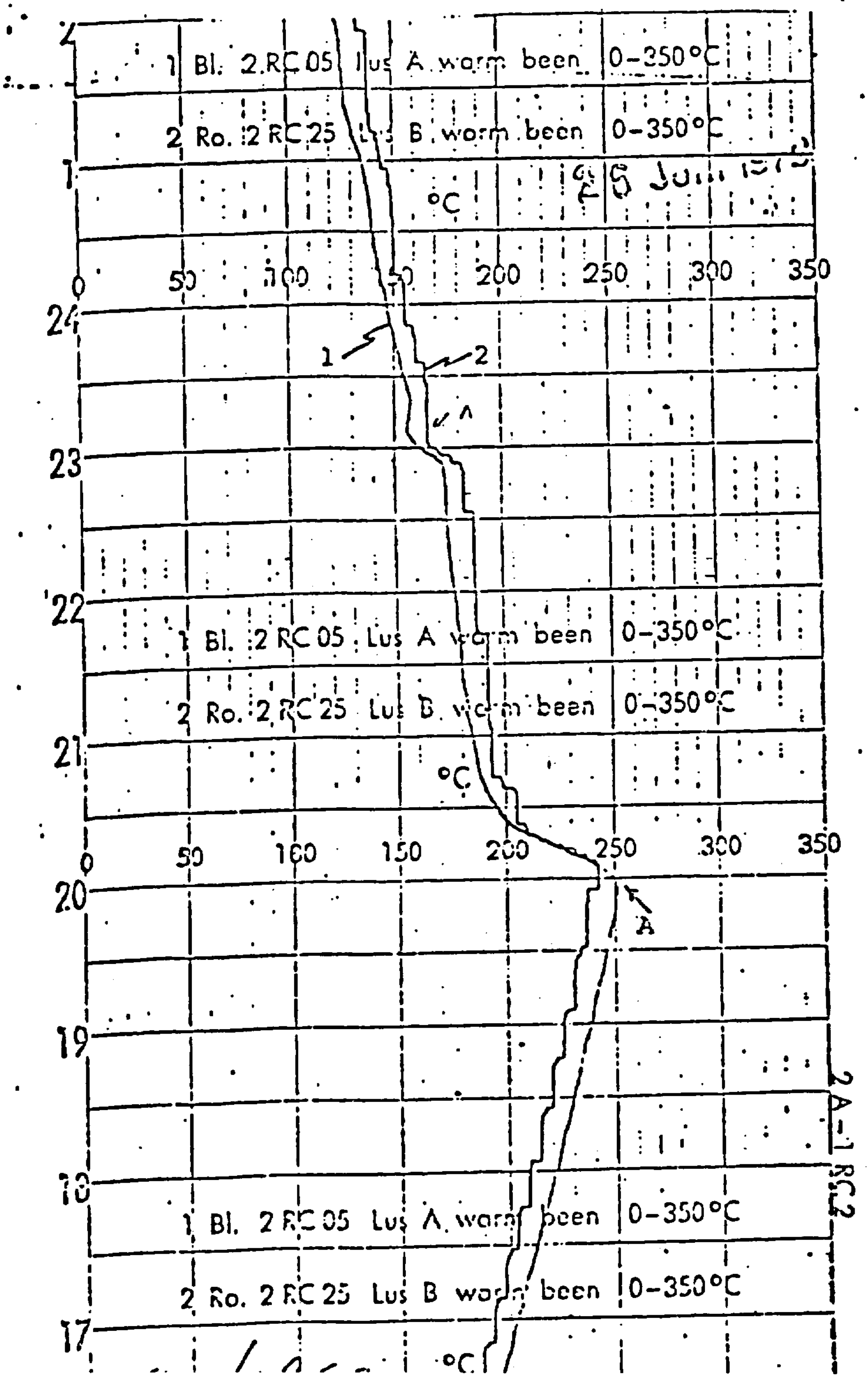
Schrijver : 2 A - 1 RC 2

*loop A. hot leg*

1. Bl. RC 05 lus A warm been 0 - 350°C

*loop B hot leg*

2. Ro. RC 25 lus B warm been 0 - 350°C



*Temperature cold leg*

**TEMPERATUUR KOUD BEEN** **FIGUUR 2**

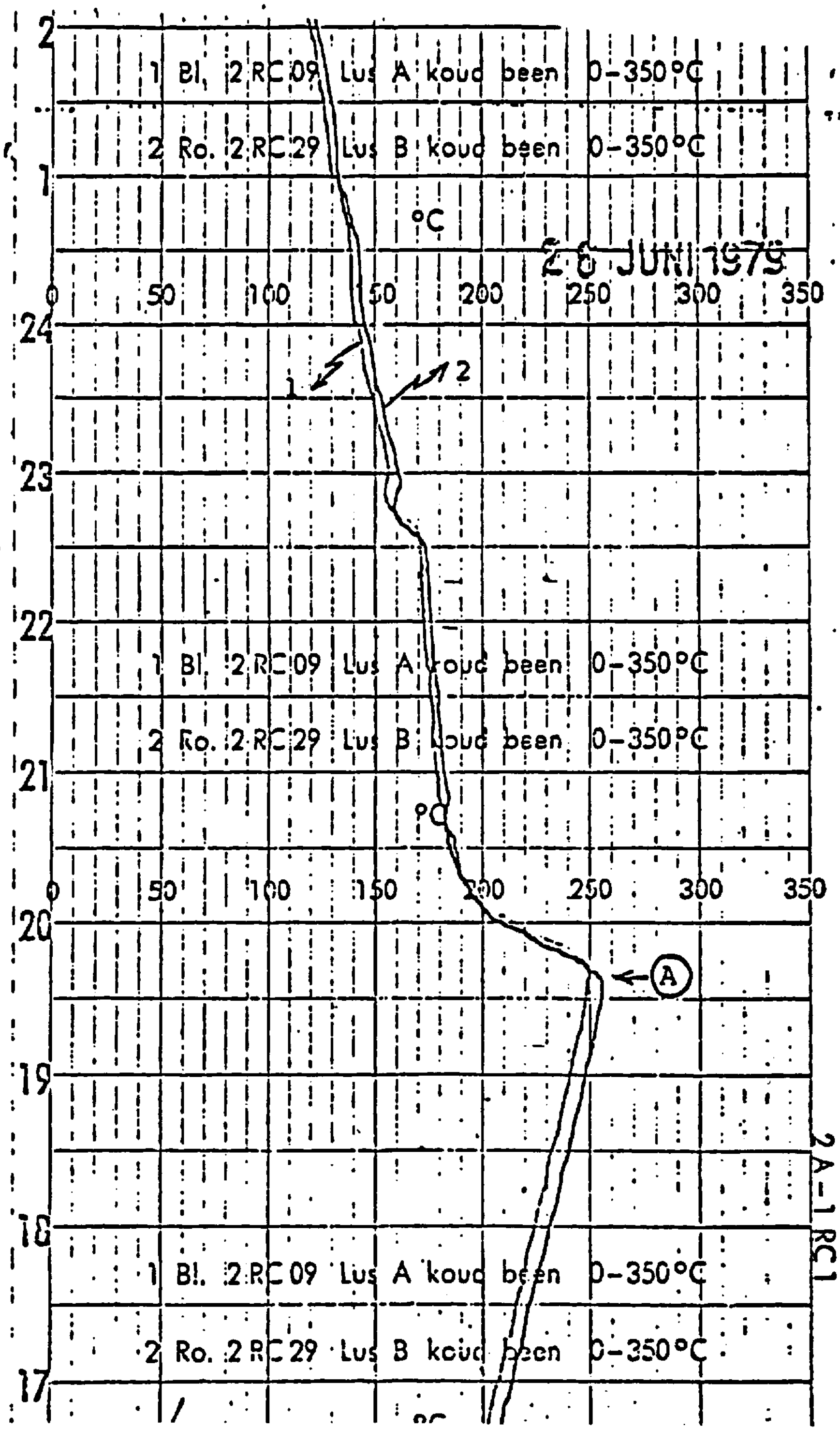
Schrijver : 2. A - 1 RC 1

*loop A cold leg*

1. Bl. RC 09 lus A koud been 0 - 350°C

*loop B cold leg*

2. Ro. RC 29 lus B koud been 0 - 350°C

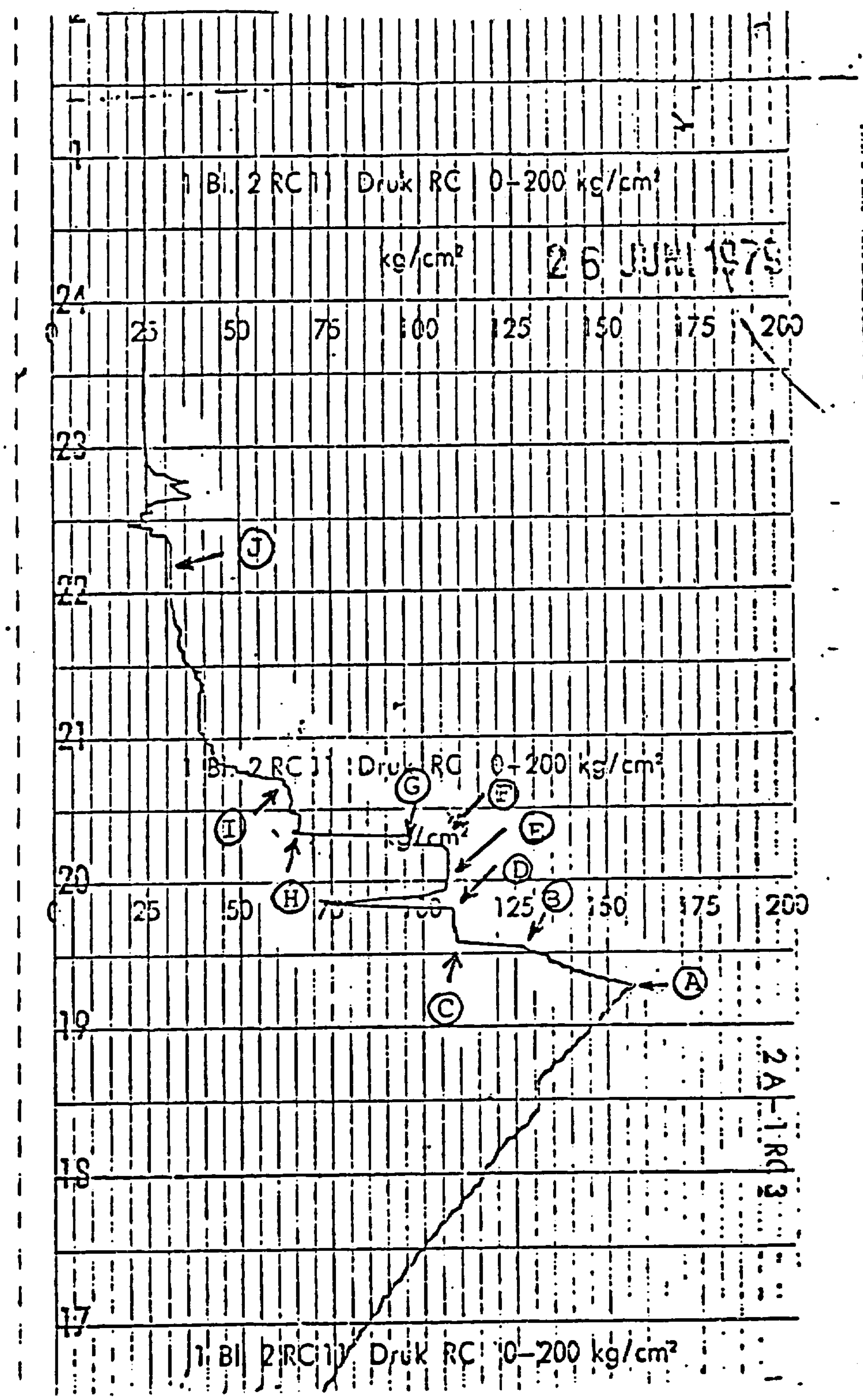


Pressure RC

DRUK RC FIGUUR 3

Schrijver : 2 A - 1 RC 3

1. Bl. RC 11 druk RC pressure RC 0 - 200 kg/cm2

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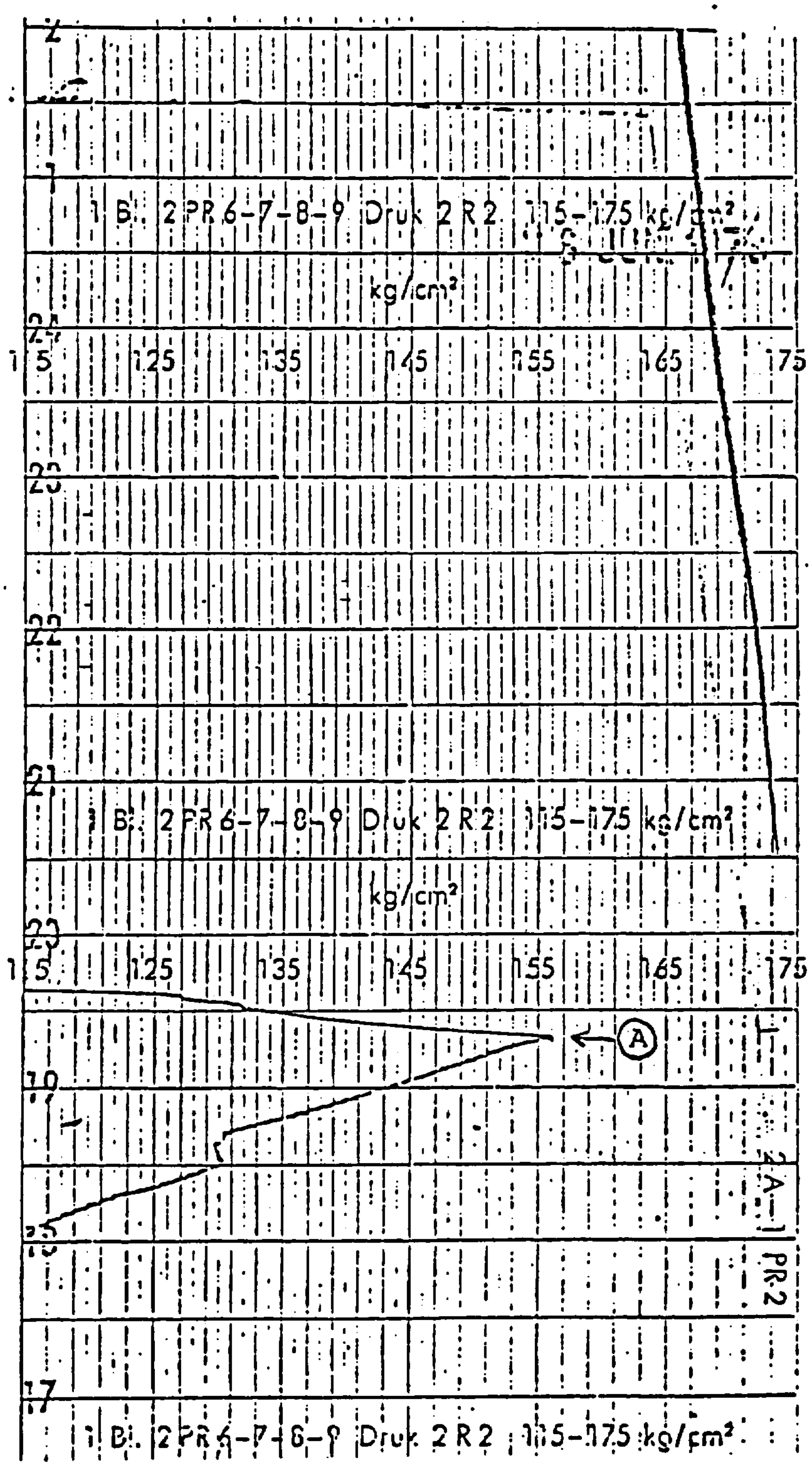
Pressuriser Pressure

DRUK R 2 FIGUUR 4

Schrijver : 2. A - 1 PR 2

1. Bl. PR 6 - 7 - 8 - 9 Druk R2 115 - 175 kg/cm2
Pressuriser pressure

Meting in dienst spoor 1 :

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Pressurizer level

PEIL R 2 FIGUUR 5

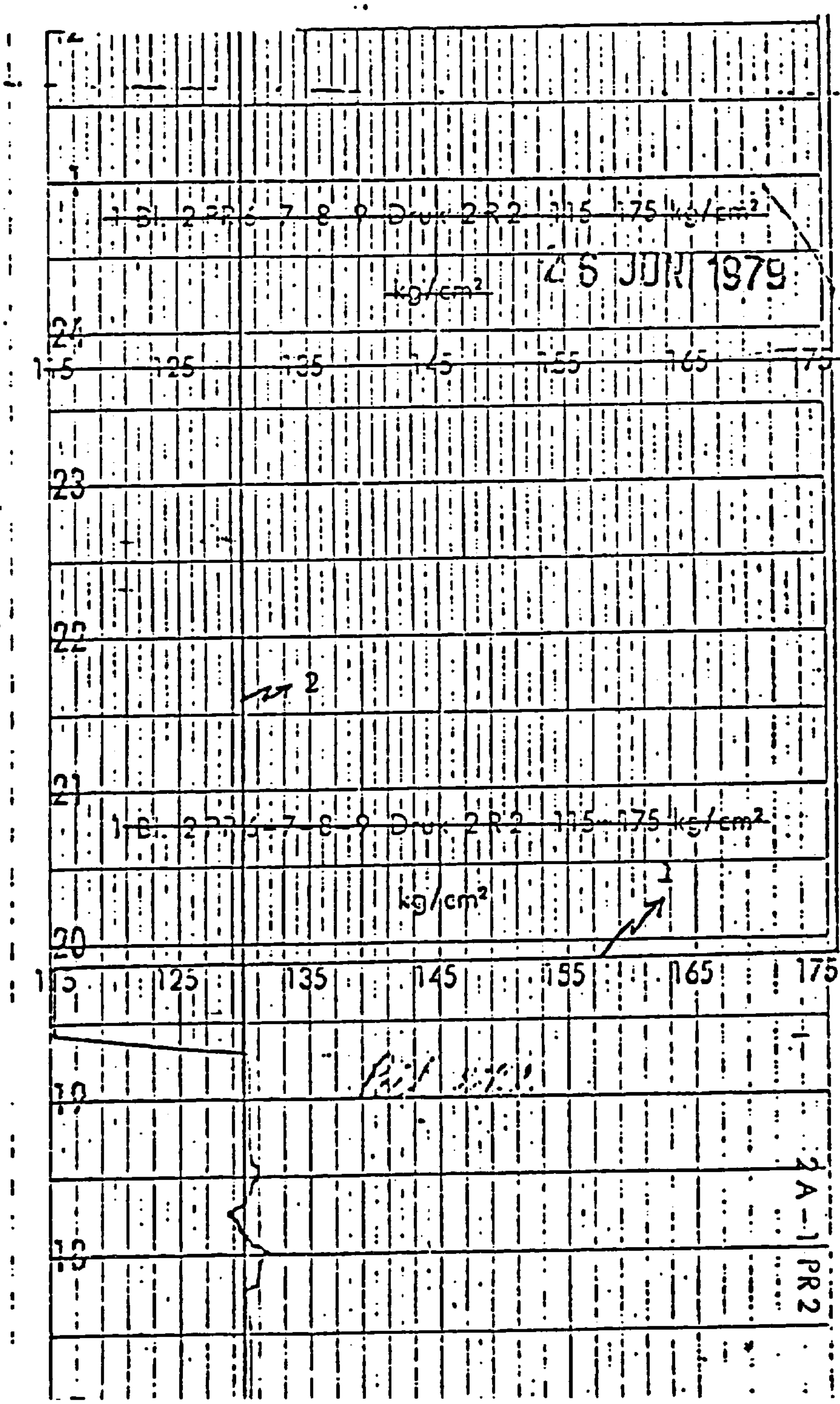
Schrijver : 2 A - 1 PR 1 Pressurizer level

1. Bl. L. PR 11 - 12 - 13 Peil R 2 0 - 100 %

Reference level

2. Ro. L. Ref. Ref. Peil R 2 0 - 100 %

Meting in dienst spoor 1 :




verkeerde schaal vervangen door 0 → 100 %

wrong scale Replaced by 0 - 100 %.

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SG A - level and pressure

**SG. A PEIL - DRUK** **FIGUUR 6**

Schrijver : 2 A - 4 FW MS. 1

SG A level

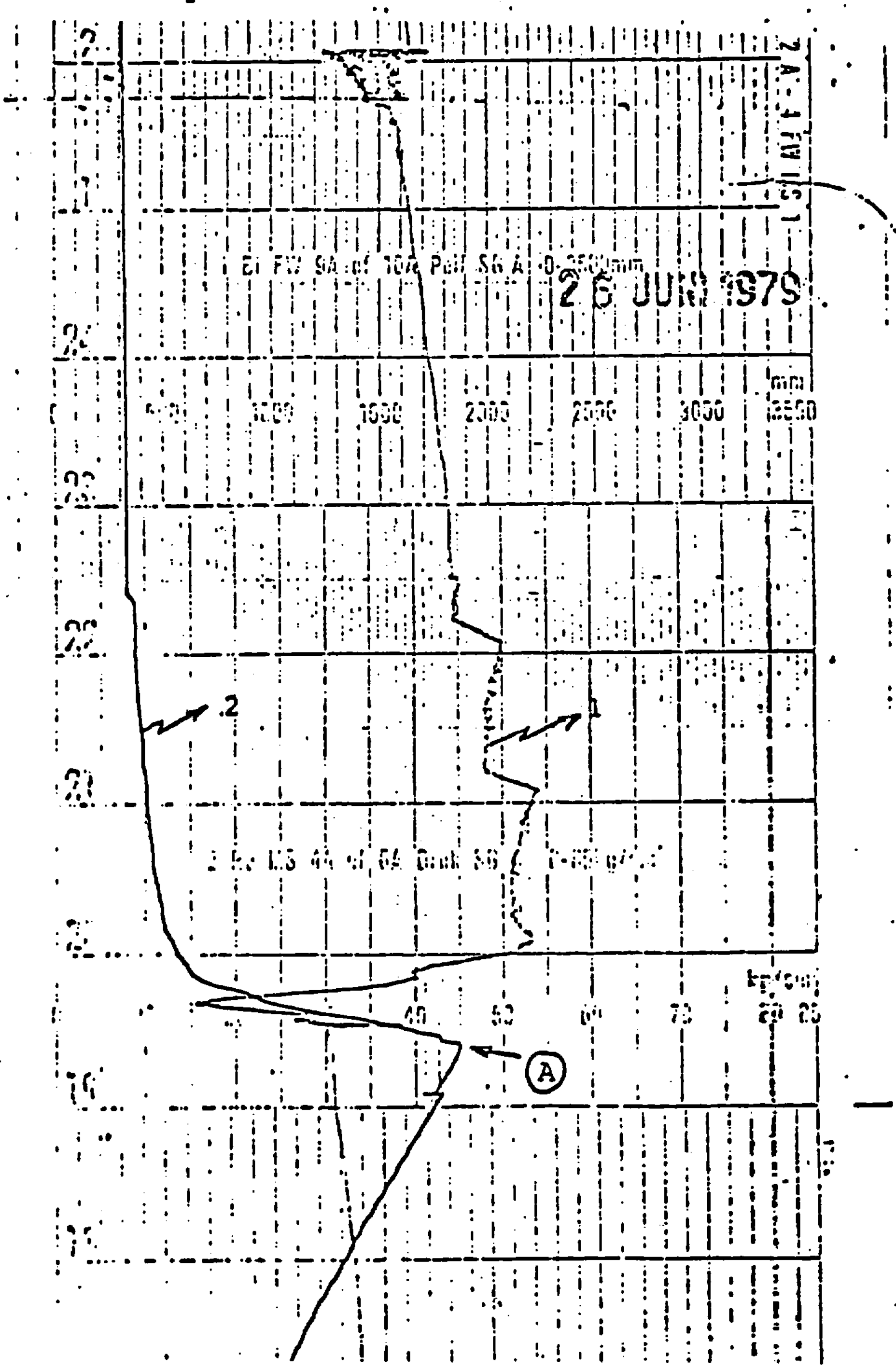
1. Bl. L. FW 9 A - 10 A Peil SGA 0 - 3500 mm

SG A pressure

2. Ro. P. MS 4 A - 6 A Druk SGA 0 - 85 kg/cm2.

Meting in dienst spoor 1 :

spoor 2 :

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SG B - level and pressure

SG. B PEIL - DRUK    FIGUUR 7

Schrijver : 2. A - 4 FW MS 5

SG B level

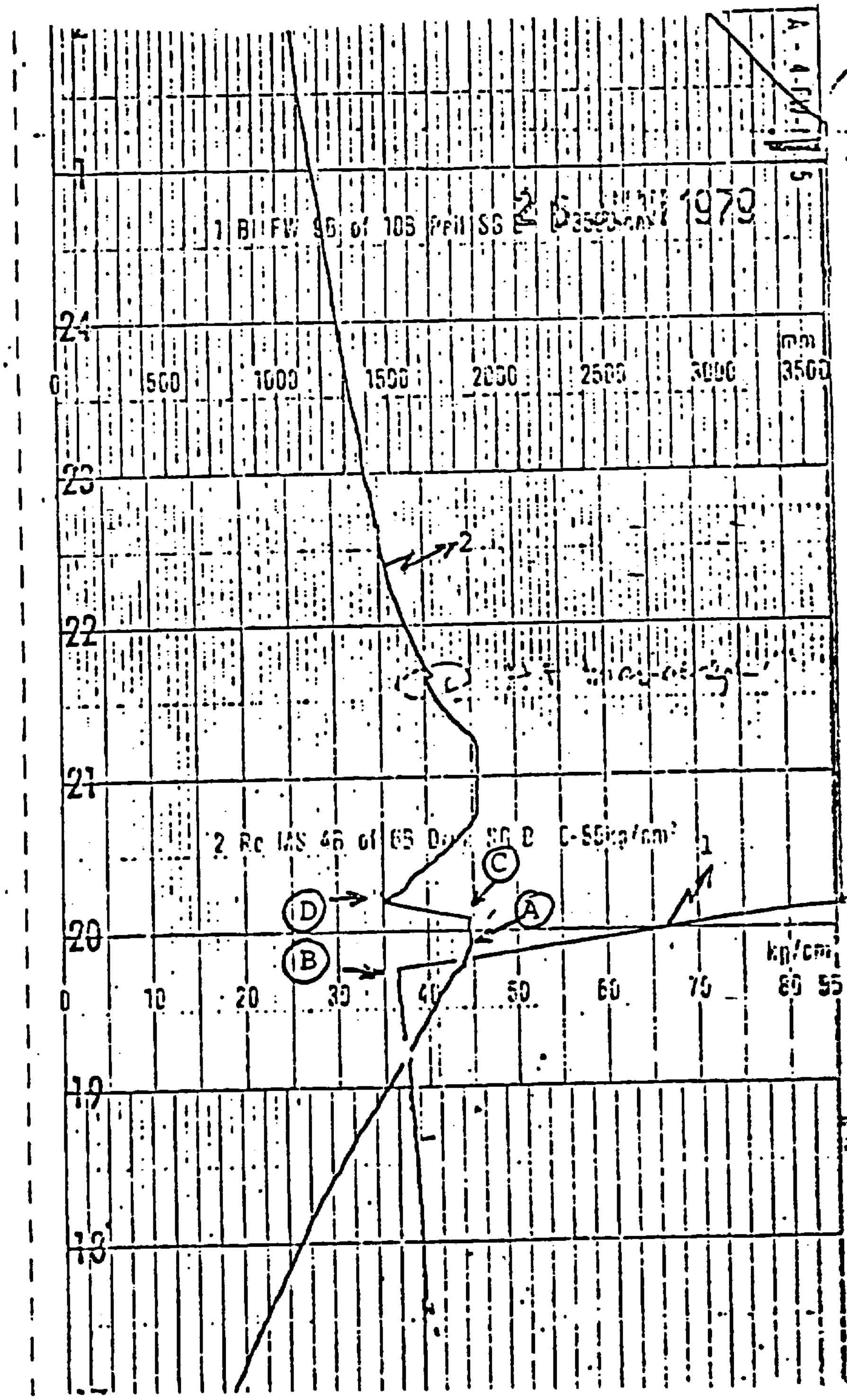
1. Bl. L. FW 9 B - 10 B Peil SG B    0 - 3500 mm

SG B pressure

2. Ro. P. MS 4 B - 6 B Druk SG B    0 - 85 kg/cm2

Meting in dienst spoor 1 :

spoor 2 :

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NRC BHDA
017 WAE071(0640)(1-065698G242)PO 08/30/79 0640
ICS IPMIIHA IISS
IISS FM WUI 30 0640
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UWE9303 BE32221TG3/229
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DR JOSEPH D LAFLEUR JR
DEPUTY DIRECTOR
OFFICE OF INTERNATIONAL PROGRAMS
U STATES NUCLEAR REGULATORY COMMISSION
WASHINGTONDC(20555)
COMPLEMENTARY TO MY LETTER REF 5.5126/71 OF AUGUST 21 PLEASE
FIND HEREAFTER SPECIFIC ANSWERS TO THE FOUR QUESTIONS RAISE BY
YOUR DR FAILKNER ON THE DOEL 2 STEAM GENERATOR INCIDENT
1. THE MAGNITUDE

COL (20555) 5.5126/71 21 2 1.
3/229 DR JOSEPH D LAFLEUR JR P2/50
OF THE LEAK WAS ESTIMATED AS ABOUT 30 TONS/HOUR AND
DEVELOPPED RADPIDLY
2. THE LEAK IS LOCATED ON THE TOP OF PIPE NR 1/24 OF STEAM GENERATOR
B DOEL 2 IN THE EXTRA-DOS OF THE U-BEND
3. SUSPECTED CAUSE STRESS-CORROSION DUE TO OVALIZATION
4. DENTING MAXIMUM 450 MICROMETER
NO FLOW
COL 30 2. 1/24 2 3. 4. 450

3/229 DR JOSEPH D LAFLEUR JR P3/25
SLOT DEFORMATION AT ALL NO TUBE WALL THINNING FOUND
THESE ANSWERS WERE FORMULATED BY THE DOEL 1 AND 2

PLANT SUPERINTENDENT
YOURS SINCERELY
F MOTTE
COL 1 2
RETR MSG
NNN
NN
WU TWX WSH

NRC BHDA

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