~~DR 5579~~

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May 23, 1973

Mr. F.E. Kruesi
Director of Regulatory Operations
United States Atomic Energy Commission
Washington, D.C. 20545

Dear Sir:

Thanks for your letter of May 9, 1973 and copy of ROE-72-15.

The drawings will be very helpful. Some additional facts are needed. They are the detailed dimensions of the --

A. "26" main steam piping O.D. and I.D.

B. The Reducers from 26" to 12" and 12" Cap. The original reduction was reported to be 20" to 16" saddle and 16" to 12".

C. Piping Specs

D. Safety Valve -- Original was 4"______#Std.____#Bolted Flange.

E. The Safety Valves described seem to be unsatisfactory. Mountings created objectionable stresses.

Great weight about 300# for Original and 400# for each of replacements. Drawings of the Original and Replacement Valves could be helpful.

F. Welding design proceedures and heat treatment could have been factors. Better practice would have reduced the Stresses to absolute Safety.

G. None of the "repairs" take care of the thermal stresses which causedthe violent explosion.

The replacement was said to be 5"______#Std.

Lack of time has not allowed me to go to the Homested Library. If you could lend me a copy of Hearing I would not need to have copies made for careful study.

My impressions from incomplete data are:

1. Original Safety Valve Headers were dangerous.
2. Replacements are also hazards.
3. Headers were not necessary. No explosion could have occurred in non existant headers.

Yours truly,

J.C. Hobbs

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The report ROE 72-17

Copy of ROE 71-12 please

ROE 72-15 concludes Explosion was caused by dynamic loading!

J.C.H.-- The piping included undesirable features which multiplied the destructive forces. Those features were unnecessary The shear diagrams confirm.

The report indicates that all fractures started near welds. No explosion could have occurred in the safety valve piping if none existed!

The destructive force was the result of piping design which resulted in shear and tensil and Tortion Stresses multiplied by reaction of Safety Valve discharge forces and vibration.

ROE 71 12

RUPTURE OF MAIN STEAM SAFETY VALVE PIPE NOZZLE

NOTE

The incident described in this Reactor Operating Experience was also reported in the newly established AEC series of "Reactor Construction Experience" reports (RCE 71-1). The RCE reports are directed primarily to individuals directly responsible for plant construction and quality assurance and control. Thus, the emphasis of the RCE reports is often on different aspects of an event than those of special interest to operating personnel. Since this particular incident is also of special interest to reactor operating personnel, the operational aspects of the incident are discussed in this report.

Summary

During hot functional testing of a new pressurized water reactor plant, a SIX-INCH DIAMETER PIPE NOZZLE between a pressurized main steam line and a safety valve FAILED COMPLETELY. Seven men, involved in testing the valve, were injured by the escaping steam. At the time of the incident, the nuclear core had not been placed in the reactor vessel. The incident provided an unusual opportunity to observe the non-nuclear plant transient aspects associated with a rupture in the main steam system.

DESCRIPTION OF MAIN STEAM SYSTEM, SAFETY VALVE ASSEMBLY AND PIPE NOZZLE BREAK

The secondary system is comprised of three steam generators each with one separate main steam line passing from each generator through the containment wall to a common main steam header. Four safety valves and two relief valves are installed on each of the three separate main steam lines (see Figure 1).

All safety valves were welded to a 6-inch schedule 80 pipe nozzle, with the nozzle weld preparation formed by counterboring and tapering the nozzle to schedule 40. The schedule 80/40 pipe nozzle was in turn welded to a 26-inch OD main steam pipe spool piece (see Figure 2).

These welds and the pipe nozzle were subjected to a "cold hydro" test at 1356 psig and had subsequently been exposed to elevated temperatures and pressures for approximately nine days prior to the incident. During this 9 day period, system temperatures varied from 520° — 540°F and pressures varied from 800—1000 psig.

The failure occurred in the reduced section of the six-inch pipe nozzle which connected the safety valve to a 26-inch steam line. The internal fracture path coincided exactly with the end of the machined internal diameter taper.

ORNL-DWG 71-14080

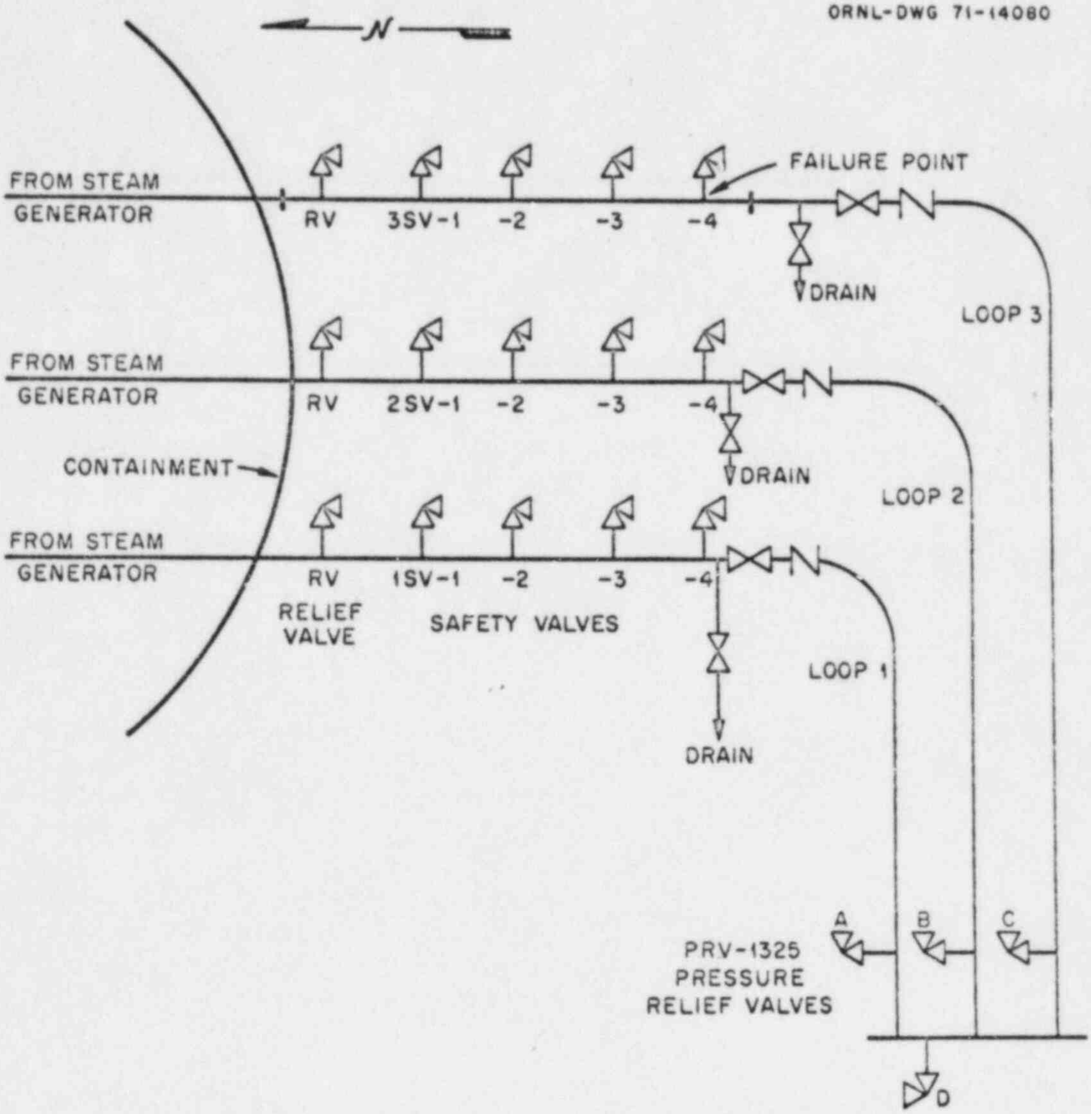

Figure 1. Main Steam Piping System

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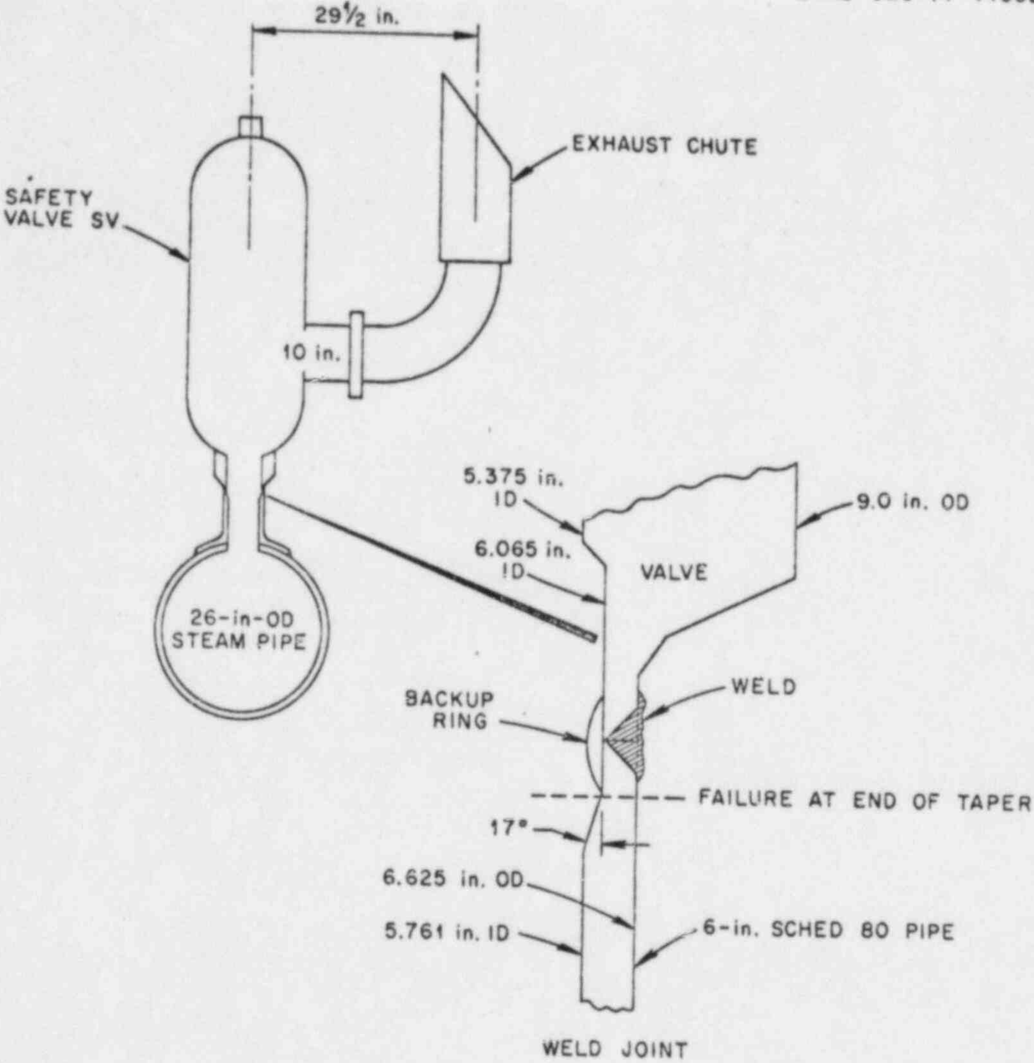

Figure 2. Safety Valve Connection

ROE 71 12

Circumstances

A. Description of Incident. For several days prior to the nozzle failure, the final stages of the hot functional testing program had been in progress using the reactor coolant pumps to heat the primary system. At the time of the incident, the lift set pressure was being checked for the steam generator secondary system safety valve. Verification of safety valve set points and adjustments as necessary were being performed using a pneumatic test device attached to the safety valve which allows the set points to be verified without raising the system pressure up to lift pressure. Eight of a total of twelve safety valves had been tested on two of three steam lines and all settings were found to be very close to the certified settings. The primary system was at 533°F and 2225 psi and the secondary system was at 900 psig. Constant level was being maintained in the steam generator associated with the safety relief valve nozzle which failed. Steam lines had been blown down briefly for one to three minutes the day of the incident by drain valves prior to testing.

It was determined that the failure occurred about the time that a member of the test group was opening the valved air pressure regulator of the pneumatic test device installed on the safety valve. The opening of this valved air pressure regulator balances the safety valve spring force to make a determination of the safety valve set pressure. No warning of the impending failure was evident to the men in the vicinity of the safety valve. A loud noise was heard, followed by a shower of steam, insulation, scaffolding, metal parts and construction debris. The men in the vicinity of the valve were either knocked to the flooring by the force of the steam release or were forced to lie down due to lack of air to breathe. The rapid release of steam displaced the air from the area above the severed pipe nozzle requiring the men to stay in a position near the floor. The men immediately made their way out of the area and down a stairway, away from the immediate scene of the accident, without assistance. The men were transported to a local hospital by ambulance and treated for burns and injuries. One man was released after immediate treatment at the hospital, but the other six were admitted for treatment of their injuries.

Crafts workmen, construction supervisory personnel and operations and test personnel witnessed the incident from several vantage points. The following description of the incident is based on interviews of the eye-witnesses:

The initial noise was immediately followed by a second louder noise. The initial steam accumulation in the area of the break spread in an almost horizontal plane, followed by the formation of a vertical column of steam which rose an estimated 150 feet into the air.

Upon inspection following the incident, it was observed that there was an area of localized cutting of insulation on a nearby pipe line

located in about the same horizontal plane as the break. This damage suggested an initial crack in one quadrant of the pipe. Steam was apparently directed in a fan shape, hori-ontally toward the nearby pipe line for a brief period of time prior to a complete severance of the pipe nozzle and prior to the expulsion of the total valve assembly from the area by the force of the steam jet. Such a sequence of events is also suggested by the reported two stages of sound, the post-accident appearance of the fracture surface, and the direction of travel of the separated valve. The valve was propelled in a direction approximately 180° opposite to the quadrant in which the initial crack is thought to have occurred. In its rebounding flight, it struck supporting structures, carried away an angle brace and dented and moved the stack from the auxiliary boilers causing its supports to bend and break away. The valve came to rest on the turbine building mezzanine floor eighteen feet below its original position.

B. Initial Plant Conditions and Action of Control Operators During Incident. Two personnel were on duty in the control room prior to the incident, a control operator and a shift foreman. The plant was being operated in accordance with a hot functional test program with three reactor coolant pumps in service to provide system heat (533°F) with system pressure being automatically comtrolled by the pressurizer at 2225 psig. One charging pump was operating in automatic control with letdown through a 45 GPM letdown orifice and feeding one train of the mixed bed demineralizer. Makeup to the system was being controlled automatically. Secondary system pressure was approximately 900 psig with the main steam isolation and main steam bypass valves closed. All three steam generator blowdown lines were valved to the blowdown tank and throttled to achieve system temperature and pressure control. Steam generator levels were being maintained at approximately 70 percent level. The motor-driven auxiliary feed pumps were lined up to take suction from the condensate tank and feed the steam generators as required to maintain proper level. The main feedwater and feedwater bypass valves were closed and the main feedwater pumps were secured. The steam-driven auxiliary feed pump was also secured and its discharge valves closed. No feedwater was being added to any steam generator at the time of the incident.

At the instant of the pipe rupture, a loud noise was heard by the shift foreman and the control operator. The noise was followed by a rapid decrease in indicated pressurizer level and pressure. In addition, level decreased rapidly in the steam generator associated with the failed safety valve pipe nozzle. Reactor coolant temperature decreased rapidly. The control room personnel secured the reactor coolant pumps. Two additional charging pumps were placed in service and letdown was stopped to minimize the effects of pressurizer level and pressure decrease. Pressurizer heaters were manually de-energized prior to reaching the automatic heater cutoff set point on pressurizer low level. Even though the pressurizer level decreased off scale, it seemed reasonably certain (and was later verified by analysis) that the pressurizer steam bubble did not

expand out of the surge line. Pressurizer pressure did not decrease below the safety injection set point of 1715 psi owing to the timely actions of the operators, which increased the system mass and minimized system shrinkage. Two boric acid transfer pumps were started to provide makeup to the charging pumps in addition to the automatic makeup which was in service at the time of the accident. The extremely loud noise produced by the escaping steam rendered communications by the plant intercom system ineffective and plant operators reported in person to the control room for instuctions. The bottom blowdown valve on the affected steam generator was opened fully to aid in depressurization of the steam generator and it was blown dry. The overall transient caused the reactor coolant system temperature to decrease approximately 213°F within a one-hour period.

A detailed survey of the reactor coolant and secondary systems was conducted to assure that there was no damaged equipment, pipe or other abnormalities. No other abnormalities were found. Pressurizer level was restored to normal, no-load, operating level which resulted in regaining a coolant pressure of approximately 2050 psig. Normal charging and letdown was established.

In preparation for starting the reactor coolant pump associated with the affected steam generator, the temperature and pressure of the two unaffected steam generators was reduced to minimize the thermal and pressure transients. Steam generator temperature and pressure were reduced by intermittent use of bottom blowdown and feedwater addition and by steaming through the main steam isolation bypass valves. Reactor coolant pressure was reduced to approximately 1250 psig by use of the auxiliary spray system.

A review of all available plant temperature instrument readings was made. Selected surface temperatures were measured on the steam generator and coolant loops to assure that there were no large temperature differences between components and the primary/secondary interface. The reactor coolant pump associated with the affected steam generator was started with no abnormalities noted. After the plant temperatures were stabilized, the remaining reactor coolant pumps were started and the plant was cooled down utilizing normal operating procedures.

C. Transient Aspects of the Pipe Nozzle Failure. Since the safety valve was blown clear, it can be assumed that from the moment of the accident. unimpeded steam discharge took place through a break area of 0.176 ft^2. This was slightly smaller than the design basis steam line break evaluated in the Final Safety Analysis Report (FSAR) for the plant. If this type of failure were to occur during reactor operation, protection against overheating of the reactor core would normally be provided by the safety injection system initiated by coincident signals of pressurizer low pressure and low level. In this incident, the automatic initiation of the safety injection system was precluded by the corrective

actions of the operators. The available instrumentation in the plant recorded a sufficient number of plant parameters during the accident to enable a comparison to be made between the calculations of plant behavior for a steam line rupture and the actual experience. Although the timely actions of the plant operators resulted in a transient behavior of the reactor coolant system which differed considerably from the accident calculations presented in the FSAR, an analytical investigation revealed that the calculational technique used to evaluate steam line rupture accidents could yield realistic predictions of the transient behavior during the incident, given input data concerning operator action. A delay of a few minutes in implementing the actions which were taken to increase primary system mass and minimize shrinkage would have resulted in emptying the pressurizer and surge line, in which case the pressurizer pressure would have decreased rapidly through the safety injection pressure set point of 1715 psig.

Although there was no fuel in the reactor at the time of the accident, the operators did follow the emergency boration procedure by starting both boric acid transfer pumps and delivering fluid from boric acid tanks to the suction of the charging pumps. Under normal reactor operating conditions, these actions would have resulted in the additional boration of the reactor coolant to prevent the reactor from returning to criticality.

The effects of the transient on plant equipment were reviewed by the applicable equipment designers. The affected steam generator and reactor internals were subjected to analysis and examination to determine if possible detrimental effects occurred and were found to be in satisfactory condition.

Causes and Results

At the time of this writing the cause of the failure has not been determined by the post-incident investigation. The stress analysis performed for the failed valve nozzle indicated that the sum of known stresses was significantly less than the yield strength and ultimate strengths of the failed nozzle material. It was postulated during the course of the investigation that a full capacity steam discharge (at existing steam conditions) at the time of the incident may have caused the failure. To achieve the full discharge, one of the following conditions must have been satisfied: (a) the set point of the safety valve lift device was exerting an equivalent of an additional 100 psi steam line pressure via 50 psig set into its air pressure regulator. If this were the case, the 900 psig reported in the steam line would have caused the valve to open. However, this would require that either the safety valve set pressure as certified by the manufacturer was incorrect, or the set pressure decreased during shipment, handling or after installation. (b) The air pressure regulator of the valve device was set above 120 psi. At this setting, the equivalent steam line pressure would have been greater than 1140 psig enabling the valve to discharge at full capacity. However, the air pressure

regulator is a sensitive device requiring manual adjustmant. Therefore, a deliberate action would be required to change the 50 psig as reported to allow 120 psi to enter the valve lift device.

Because the valve manufacturer's representative performing the valve check had paused during the test to check steam line pressure just prior to the incident, the equivalent steam line pressure could have been no greater than that necessary to bring about a balance valve condition. This condition would have caused a "simmering" of the valve which would have been noticeable, but was not detected.

Although full capacity steam discharge is improbable, it cannot be ruled out entirely. The stresses computed for a full capacity steam discharge condition were nearly equal to the ultimate strength determined for the nozzle material. Further, should the total stress, under these conditions, be coincident with a stress intensification factor due to pipe wall imperfection, stresses may have resulted which exceeded the ultimate strength of the nozzle material. Thus, it appears that the inlet nozzle may have been marginally designed for a full discharge load condition.

The results of the investigation (available at this writing) indicate that the fracture initiated at one location on the pipe nozzle and proceeded circumferentially around it in both the clockwise and counterclockwise directions terminating about 180° opposite the fracture starting point. All areas of the fracture showed evidence of extensive plastic deformation and the internal diameter fracture path coincided exactly with the end of the machined inner diameter taper (see Figure 2). The fracture path was not associated with the heat-affected zone of the weld although it did penetrate the heat affected zone in several locations. No significant material deficiencies were found with respect to chemical analysis, microstructure, pre-existing defects, or mechanical properties. No evidence of fatigue damage was observed in any area of the fracture.

Corrective Action

The twelve main steam line safety valve inlet nozzles were redesigned to provide for a conservative design and to negate the effects of possible stress intensification factors. The extent of the revision included increasing inlet pipe nozzle size to 8-inch schedule 160 thereby enlarging the nozzle diameter and decreasing the thickness differences between the nozzle wall and the valve body wall. The seven relief valve pipe nozzles have also been modified using the same approach taken for the modification of the safety valve pipe nozzles.

Conclusion

The lack of any material deficiency and the extreme plastic strain viewed on the fracture surface, indicate that failure was caused by overloading. The peak stress must reach the ultimate stress to initiate a

failure and although the preliminary stress analysis of the static condition did not account for the overload, there are mechanisms which may be postulated for generating the required increase in stress. These mechanisms become significant especially since the stresses computed for a full capacity steam discharge were nearly equal to the ultimate strength determined for the pipe nozzle material.

Although this report is based upon interim findings, sufficient information is available to identify the probable mode of failure and to indicate that the inlet pipe nozzle apparently was marginally designed for a full discharge load condition or other increased stress condition.

P. S. Colby, Carolina Power and Light Co., to Dr. P. A. Morris, AEC Division of Reactor Licensing, Docket 50-261, July 3, 1970, available at AEC Public Document Room.