## SECOND LEAKAGE EXPERIENCE

On October, 1973, the steam generator was put back in operation and cold trapping operation was proceeded under the coupled situation of sodium and water. On November 16, water leaked again into sodium side after 542 hours of the operation. Gas-Chromatograph and membrane detectors showed rapid increases almost simultaneously at the second time. Operation was stopped with these signs. Plots of the readings are shown in Fig. 7.

According to the facts that two kinds of detectors responded at almost the same time, the leakage was expected to be in sodium region. To make sure the leakage, the same helium leak test as mentioned before was conducted and succeeded to identify the leaked tube between the two. Next step was to find where it happened, downcomer region or herically coiled heat transfer region.

Sodium was filled to such a level that the open end at the bottom of the downcomer was covered so as to separate two region. Helium leak test indicated the downcomer region was damaged. With the previous experience, the damaged part was expected to be the welded portion of spacers on downcomer tube.

When it was lifted up again for the inspection, some reaction products were found arround the open end of downcomer and then the outer tube was cut away to inspect the downcomer, which showed the expectation was right. The damaged part is shown in Fig. 8. It was a similar cracking along the heat affected zone and the causes was considered to be the same as the previous one. The downcomer tube was replaced to new one and spacers were changed to be fixed with inside of the outer tube with screws.

1 MW steam generator was put back into operation on April. 1974 and it is now successfully on the last stage of test schedule. Experiences obtained through these troubles have been referred and utilized to the design and the fabrication following sodium components.

---

**K.3. KNK-Steam Generator Leakage, Evaluations and Improvements** — **K. Dumm, W. Ratzel** — **Fed.Rep.Germany**

**Abstract**:

On September 23rd 1972 the KNK-reactor was shut down due to a sodium-water reaction.
Detection, localisation and possible leak development are described and discussed.
Improvements on the KNK steam generator system due to this leak experience are explained.



## 1. Introduction

The KNK reactor was manually shut down on September 23rd, 1972 due to indications of a sodium-water reaction in one of its two steam generators. The leak was assumed to be a comparably small one as no rupture disc was actuated. At the time the reactor was running at the 30 % output level and the operating conditions of the failed steam generator were as given below:

| | |
|---|---|
| sodium temperature inlet | 420°C |
| sodium temperature outlet | 276°C |
| sodium flow rate | 180 $m^3$/h |
| feed water temperature | 200°C |
| steam temperature | 420°C |
| steam pressure | 79 bar |
| steam flow rate | 13 t/h |

Fig. 1 shows the geometrical arrangement of the secondary sodium loop with its main components. As explained later the event was partly influenced by this geometrical arrangement. The steam generator with the connected pressure relief system is illustrated in Fig. 2. Steam is produced in 28 parallel tube-in-tube units which are connected to headers. Three additional units are available as spare parts; they are not connected to the headers. Each sodium outlet is equipped with an inductive bubble indicator by which hydrogen bubbles originating from a sodium-water reaction can be detected. At the time when the leak occurred each sodium outlet was additionally equipped with a thermocouple for low load stability measurements. In the case of large sodium-water reaction fast pressure relief is achieved by 4 rupture discs. Previously to the steam generator failure described here all rupture discs were designed having an actuating pressure of 20 bar.

## 2. Sodium-water reaction

The presence of a sodium water reaction was indicated by several instruments:

- level increase in the expansion tank.
- pressure increase in the expansion tank which results in an overpressure alarm signal.
- gas bubble signals from nearly all of the bubble indicators.

After an instrumentation check was carried out by the operator, the reactor was manually shut down. The water/ steam side was isolated and depressurized. During the depressurization period the water leakage continued until pressure equilibrium with the sodium side was reached. Therefore pressure and level in the expansion tank increased further; the safety valve starts to blow at 5,5 bar (diff.) and finally the expansion tank was completely filled with sodium up to the safety valve where the sodium solidified. By chance at this point the equilibrium between sodium and steam pressure was attained thereby preventing further pressure build up. As the steam blow down continued hydrogen from the sodium side passed back through the leak and entered the water side. During the incident a part of the air cooler (see Fig. 1) was also filled with hydrogen. By expansion of this hydrogen the steam generator was refilled with sodium and 65 kg of sodium passed through the leak to the water side. This summary of the event is given in Fig. 3.

## 3. Leak detection

The bubble indicators are not designed to produce a "first of its kind" signal. Due to the fact that nearly all indicators signalised the passage of gas bubbles these signals could not be used to identify the failed tube-in-tube unit. As mentioned before all sodium outlet tubes were equipped with thermocouples for low load stability tests. The evaluation of these temperature readings showed temperature fluctuations at every outlet tube during the event. Speaking in terms of KNK-nomenclature the temperature fluctuations at tube no. 39 started about 1 minute earlier than the others. In another approach the water side of the steam generator was pressurized with nitrogen and the water/steam tubes near the feed water and steam headers were controlled by an ultrasonic microphone. The nitrogen pressure was increased up to 5 bar. At this pressure the sound emission from tube no. 39 increased to a level 6-10 times higher than from the other tubes. Having two independent indications tube no. 39 was assumed to be the leaking unit. After dismantling this tube unit the leak was located in the lower superheater region. Fig. 4 shows schematically the leak region. The leak is of a circular shape. At the steam side it has a diameter of approx. 1,3 mm which enlarges conically to the sodium side up to about 4,5 mm. It is positioned in the neighbourhood of a spacer. There was a wastage area of about 18 mm in diameter at the inner surface of the outer tube. The remaining wall thickness of the 2,9 mm thick tube was 0,8 mm in the affected region. The cause of the leak is presented in literature /¯1_7.

## 4. Leak formation evaluations

The pressure gradient in the expansion tank was used to determine the leak rate and the leak size at the time when the pressure increase started.
In the assumption of

$$Na + H_2O \longrightarrow NaOH + 1/2\ H_2$$

the leak rate was calculated to be 7 gr/s with a leak size of approx. 0,7 mm diameter.
The final leak size after dismantling was found to be 1,3 mm diameter (see chapter 3).
Due to results from wastage tests a linear enlargement of the leak diameter in time was assumed. Based on this it was estimated that the leak started at least 30 minutes prior to detection (see Fig. 5). This has to be understood as a minimum time. Much longer times of leak development are possible.

Moreover, a comparison of the hydrogen with the sodium inventory at operating conditions showed that free gaseous hydrogen could not exist before this 30 minute point as the sodium initially had to be saturated with hydrogen partially forming sodium hydride. Free gaseous hydrogen can only remain when the formation rate of the hydrogen is larger than the consumption by solubility and hydride formation.

Finally, bubble meters, pressure and level measurements were not able to detect the leak earlier due to the absence of free gaseous hydrogen. In contrast to this an in sodium detector would be able to detect the leak much earlier.

## 5. Improvements on the systems

The design criterion for the steam generator and secondary system was defined as the sudden and complete rupture of one water tube. The pressure relief system was designed and tested to fulfill this requirement /¯2_7. While the sodium side of the steam generator becomes pressurized up to water pressure the adjacent loop system remains at nearly constant operation pressure.

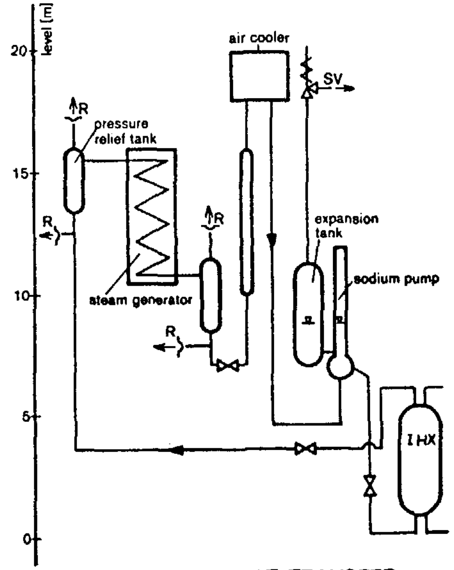
Contrary to the design case during the described incident the whole secondary system was uniformaly pressurized. Therefore in order to avoid overloading the main components such as pump, expansion tank and IHX in the case of a simialr event, the relief pressure of one of the rupture discs was reduced to 9,5 bar as already shown in Fig. 2.

To avoid sodium solidification in the safety valve and the adjacent pipe-work these systems were included in the trace heating system.

By means of an additional fast acting valve the blow down time of the steam generator was reduced in such a manner that 1,5 bar remaining pressure is reached within 100 s. Moreover the steam generator can be pressurized on the water side by nitrogen in order to avoid sodium entrainment.

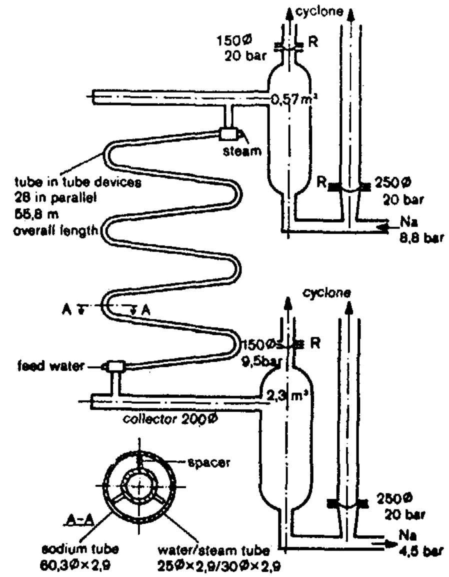
Within the last two years hydrogen detectors for the SNR-300 steam generators have been developed. For an earlier detection of small leaks in the KNK systems it was decided to equip each steam generator with one of these instruments.





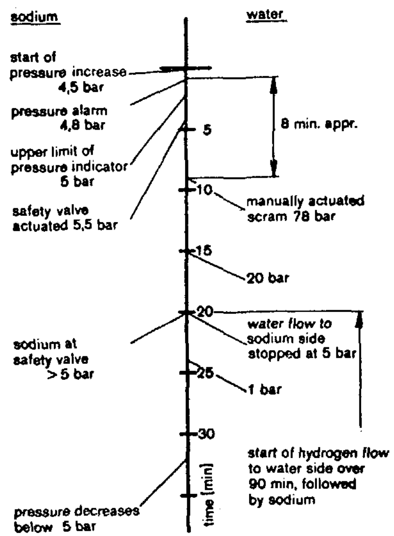

KNK SECONDARY HEAT TRANSFER SYSTEM

Figure 1



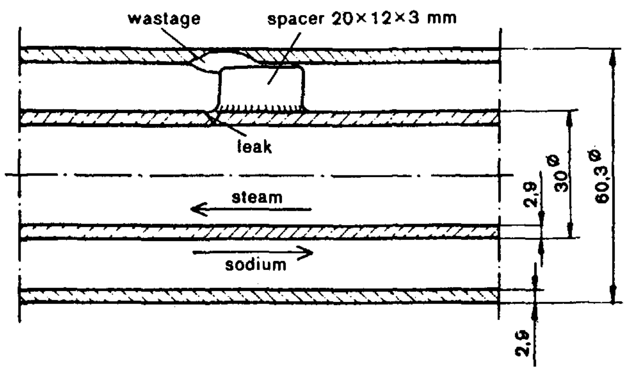

KNK STEAM GENERATOR 28 $MW_{TH}$

Figure 2



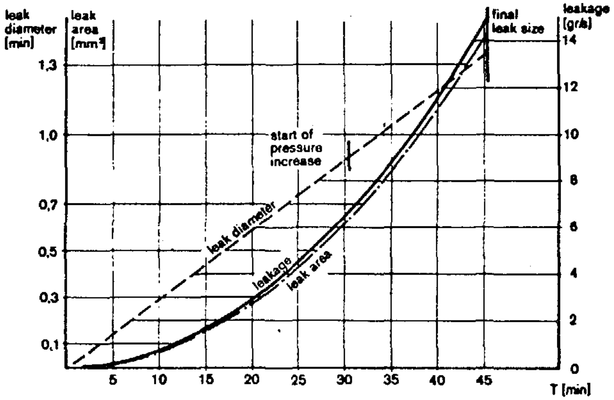

KNK EVALUATION OF STEAM GENERATOR FAILURE

Figure 3




KNK STEAM GENERATOR LEAK ZONE

Figure 4




KNK DEVELOPMENT OF STEAM GENERATOR LEAKAGE

Figure 5


Literature

[1] Lorenz H. et.al.

Metallurgical post-examination of KNK-steam generator failure
Study Group Meeting on Steam Generators for Liquid Metal Fast Breeder Reactors
Bensberg, October 1974

[2] Dumm K. et.al.

The reaction of sodium and water in tubes
Atomkernenergie 14 Jg., (1965) H.5


---

K.4. **Post-failure Metallurgical Investigation of KNK Steam Generator Tube Damage**


*H. Lorenz*
G. Herberg
(presented by E.D.Grosser)

Fed. Rep. Germany



Abstract:

In September 1973 the sodium-cooled reactor KNK was shut down due to a steam generator tube damage.
Failure location and results of the metallurgical examination of the damage are described. The cause of the damage is discussed.


## 1. General

The "Kompakte Natriumgekühlte Kernreaktoranlage" (KNK) is a 58 MWth (~ 20 MWe) sodium-cooled reactor plant, designed and constructed by Interatom by contract of the "Gesellschaft für Kernforschung" (GfK). The construction was started in May 1966. The criticality was achieved in August 1971. In February 1973 the reactor plant was handed over to the customer.
On September 23, 1972 the reactor was being started up for a run at 50 - 60 % of rated power. About 5 hours after having started the runup, 25 % of the full rated power had been achieved.

## 2. Occurence and Course of the Disturbance

In this stage of operation an alarm signal was given: "Excessive pressure in surge tank of System II".
A control of the inert gas supply didn't show any failures.
At the same time the level indicator in the control room recorded an increased level.
At 6.24 o'clock a manual scram was tripped due to the unclear operational conditions, and thus the plant had been shut down. Simultaneously with the level rise and the scram trip, various bubble detectors at the sodium outlet of Several steam generator tubes gave an alarm
The conclusion, drawn from these alarm signals, was that there might be a leak in the steam generator system II.

## 3. Steam Generator Design

The KNK plant has two secondary sodium loops each with a 30 - MW (th) once-through steam generator system. Each of the two steam generators has four headers (sodium inlet and outlet headers, steam and feed-water headers) which are connected by 31 coaxial serpentine tube as heat exchanger. Such a serpentine heat exchanger tube is shown in Fig. 1.
The material for the headers, tubes and sleeves is the ferritic low-alloyed Ni containing Nb-stabilized 2 1/4 Cr - 1 Mo-steel 10 CrMoNiNb 910 (material no. 1.6770).
The inner tube is centered in the outer one by spacers each consisting of 3 tabs - welded on the inner tube and staggered by 120° at 44 locations along the coaxial tube.
In order to avoid friction abrasion, the tabs are stellited at the contact surface to the outer tube (Fig. 2).