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INTERNATIONAL ATOMIC ENERGY AGENCY, INTERNATIONAL WORKING
GROUP ON FAST REACTORS MEETING OF SPECIALISTS ON SODIUM-
WATER REACTIONS, ARGONNE, USA, 5-6 NOVEMBER 1968

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INTERNATIONAL ATOMIC ENERGY AGENCY

INTERNATIONAL WORKING GROUP ON FAST REACTORS

Meeting of Specialists on Sodium-Water Reactions
Argonne, USA, 5-6 November 1968Summary Report

1. INTRODUCTION

The meeting of specialists on sodium-water reactions was the first of its kind within the framework of activity of the International Working Group on Fast Reactors. The meeting was held at Argonne in accordance with the recommendation of the IWGFR and in agreement with the USA authorities. Participants from five countries took part in the meeting chaired by Mr. R.H. Jones of the United States Atomic Energy Commission. The list of participants is found in Attachment 1.

The adopted Agenda is enclosed in Attachment 2.

The first report on the meeting (Attachment 3) included statements made by some participants of the specialists meeting at the panel discussion on the subject held at the International Conference on Sodium Technology and Large Fast Reactor Design on 7 November 1968. Though these review statements covered many aspects discussed at the meeting of specialists, it was felt that a more comprehensive technical report could be of value. The idea of issuing an IAEA/official technical report has been encouraged by the IWGFR which, at its meeting in March 1969, considered it important to hold subsequent meetings on sodium-water reactions. So a technical report on the first meeting at Argonne can be some sort of a reference document for further meetings.

This technical report has been prepared with the mutual co-operation of all the participants and includes summaries of their contributions to the discussion.

2. FUNDAMENTAL STUDIES OF REACTION KINETICS

MR. GOLDMANN: The results of a literature search and some tests on the kinetics of sodium-water reactions were reported.^(1,2)

The tests consisted of contacting drops of sodium with water and steam and measuring the reaction rates by measuring the rates of hydrogen evolution. Principal variables were temperatures of the sodium and water and water purity. Motion pictures were taken to identify various stages of reaction and to observe the effects of the presence of oxygen.

The literature survey uncovered considerable controversy in regard to reaction rate limiting steps. On a molecular scale, the rates are on the order of microseconds, which is considerably faster than turbulent diffusion or other mixing processes. The latter processes take milliseconds or longer and may therefore be assumed to be rate limiting for steam generator leaks.

Reaction rates increase as sodium temperatures approach and exceed the melting and boiling points. For reactions between "cool" (212°F) sodium and (35-140°F) water, the rates are proportional to the surface area of a sodium drop. Reactions with higher viscosity water are much more rapid, presumably due to the lower cooling capability. Inhibitors, such as oil floating on the water surface, slowed reaction rates greatly.

With the use of coloured motion pictures, several reaction stages could be identified. A preheating stage preceded ignition. A heating stage was characterized by a red glow. A deflagration stage was attributed to sodium boiling. In the presence of oxygen, the deflagration stage is reached at much lower temperatures by virtue of a sodium dispersion resulting from an explosive reaction between the hydrogen which is generated in a sodium-water reaction and the surrounding oxygen.

MR. FUZUKAWA: The reaction phenomena between sodium and water will show wide variations depending on their phase states, temperatures, masses and mutual contact mode. To get a wider understanding of this reaction for hazard analysis, about 40 combinations in the reaction conditions were examined experimentally in air. Among the tests, 23 cases were chosen as more interesting ones for presentation in Table I. These were presented, together with an 8 mm film at the meeting.

The most remarkable conclusions are as follows;

- (a) The most violent reactions were observed in the conditions of both liquid state and higher temperature. 50 g sodium was sputtered burning to the distance of 7 m.
- (b) In the case of contacting solid sodium with water, no explosive reaction takes place until complete melting of sodium.
- (c) In the case of pouring a lot of hot sodium on a small amount of water, a steam explosion (expansion) occurs which is very dangerous due to the sputtering of liquid sodium.

3. INSTRUMENTATION FOR RIG USE

MR. LIONS referred to instrumentation already discussed at the conference in Aix-en-Provence, France, in September 1967.

MR. MAUSBECK said the main measuring devices during their sodium-water reaction test work were pressure pick-ups of the piezoelectric-quartz type. The layout data and the calibration procedure are included in (3) and (4).

MR. BRAY mentioned instrumentation used at Dounreay. This item was not given too much attention at the meeting and is not intended to be covered in more detail since the subject has been thoroughly considered at the IAEA Meeting of Specialists on Core Instrumentation held in Karlsruhe in October 1969 [5].

4. SODIUM-WATER REACTION TESTS AND EVALUATION OF RESULTS

Large leaks

MR. GOODMAN reported on a programme of Atomic International and showed a film on tests of mock-ups of steam-generator modules.

MR. LIONS: The results submitted relate to tests on the reaction of sodium and water in a linear steam generator with a nest of 7 tubes used in conjunction with a sodium circuit of approximately 100 kW power.

In these experiments, a tube of water was cut right through at the centre of the nest. The experimental work was concerned essentially with the following points:

- Study of the behaviour of the tubes of the nest in the presence of sodium-water reaction products;
- Measurement of the main physical parameters in the generator and the associated sodium circuit: pressure, flow, temperature, strains;
- Comparison of the experimental results with the theoretical figures calculated on the basis of a programme processed on an analogue simulator;
- Technological tests: shut-off valves of steam generator, discharge circuit for sodium-water reaction products, sodium-hydrogen separator.

Description of the test equipment is given in [6] (Fig. 1).

Mathematical model and calculation hypotheses

The calculation hypotheses as well as the equations employed were presented at the Aix-en-Provence conference on the safety of fast neutron reactors in September 1967 [6]; it will be recalled that the main hypotheses are as follows:

- The reaction between the sodium and the water is instantaneous and adiabatic;
- All the water introduced into the generator reacts; account is not taken of the diffusion of steam through the hydrogen;
- The hydrogen produced occupies the entire cross section of the steam generator;
- The expansion of the hydrogen is isothermic; it takes place at an average temperature of 1000°K ;
- The sodium and the metallic structures are assumed to be, respectively, incompressible and non-deformable;
- The compression of the blanketing gas is adiabatic.

Experimental results

The experimental conditions and the principal results obtained are shown in Table 2.

(a) Comparison of experiments with calculation

The calculated pressure of hydrogen in the steam generator is verified experimentally with a mean error of 15% between experiment and calculation (Figs 2, 7 and 9).

The flow rates of sodium in the associated circuit are verified with a mean error of 30% for the upward flow Q_3 and of 27% for the downward flow Q_4 (Figs 3, 4, 8 and 10).

The calculation of pressures in the sodium circuit and the expansion tanks and the calculation of the rate of displacement of the reaction fronts during a time interval of 0-50 ms were also checked.

(b) Reproducibility of the results

Considerable differences were observed between the experimental results and the calculations, both in the case of flow measurements and that of the pressure measurements during the test GLF - 2A (Figs 5 and 6); these may be due to poor reproducibility of the water tube rupture conditions.

(c) Influence of the geometry of the sodium circuit used in conjunction with the steam generator

These tests demonstrated the important role which the geometry of the sodium circuit associated with the steam generator plays in the protection of the fragile components of the sodium circuit.

(d) behaviour of the nest of tubes

No second ^{tube}/rupture was observed during these tests either in the case of an austenitic steel nest or in that of a ferritic steel nest.

A film on sodium-water experiments in Cadarache was shown.

Conclusions

This series of tests carried out on an industrial-type installation shows that it is possible to keep a check on the results of a violent sodium-water reactor involving high rates of water flow (~ 12 kg/s) without greatly damaging the sodium circuit. It was shown that this was possible under experimental conditions by giving the circuit a special form, the principle of which is shown in the sketch in Fig. 1. The calculation code employed enabled ^{reasonable} approximations to be made of the maximum excess hydrogen pressure measured in the apparatus with an average accuracy of approximately 15% for a small hydrogen bubble and of 50% for a large bubble.

The pressures and flow rates in the sodium circuit were checked with an accuracy acceptable for safety calculations, but time lags were observed between the theoretical and experimental curves; these were due, in particular, to the fact that the calculation was made for an incompressible fluid.

MR. MAUSBECK: In this connection not only the experimental results of the different laboratories are of prime interest, but also the definition of a large leak. B&W Barberton showed that the pressure build-up in the steam generator is proportional to the square root of the number of destroyed tubes. Therefore, the pressure layout of the outer shell of the steam generator is strictly connected with the number of destroyed tubes.

The dependence of maximum pressure on the number of destroyed tubes is based on all ^{of} the tubes breaking at the same time. The same time means in this connection: within some ten milliseconds. We think no credible process can be assumed which gives a complete rupture of more than one tube at the same time as the starting event.

It is possible that other tubes break at the same time as a consequence of initial tube failure. Since the pressure peak is only of short duration, namely for the first moment of the acceleration of the sodium piston after the break, subsequent tube ruptures do not produce pressure peaks which coincide with the initial pressure peak and they do not reach the same pressure height because of the changed situation in the reaction zone. The conclusion is: the design pressure for the steam generator is probably given by the breaking of one tube.

"Large leak" and "small leak" are terms which define the reaction behaviour that occurs. A "small leak" results in a stationary reaction zone. A "large leak" is characterized by a movement of the whole sodium column between the reaction zone and the rupture discs in the pressure relief system - this is equivalent to a movement of the reaction zone itself.

Therefore, small leak problems arise from heating and wastage by reaction products in a small area. This may lead to additional leaks. Large leaks, on the other hand, give the criterion for the pressure design of the system and the sizing of the pressure relief system.

With respect to large leaks, we have performed tests for two different types of steam generators:

- Tube-in-tube steam generator;
- Tube-in-shell steam generator.

The results of both are included in Lit (3) and (7).

MR. BRAY: Sodium/water work in support of the design of the PFR and future sodium heated steam generators has been carried out at Downreay over the last ten years. The large test rig known as NOAH (an anagram of the letters in the chemical symbols $\text{Na}/\text{H}_2\text{O}$) was commissioned in December 1963 and 15 test reactions were carried out in this unit. The rig was then modified to accommodate larger experiments and a second series of tests on the new assembly known as Super-NOAH are now in progress. Other small-scale experiments were made in the laboratory using simpler techniques to study wastage effects.

The specific objectives of the experiments may be summarized as follows:

- (a) To establish the changes in pressures and temperatures during a sodium/water reaction;

- (b) To investigate the effects of these pressures and temperatures on adjacent water tubes in the test bundle and on the main pressure vessel;
- (c) To investigate the corrosion/erosion effects of a sodium/water reaction on the materials used;
- (d) To investigate the problems of clean-up and maintenance after a sodium/water reaction has occurred;
- (e) To acquire information on the problems of dealing with the effluent from a sodium/water reaction.

The original NOAH rig was designed on the basis of testing a $\frac{1}{6}$ scale model of a PFR steam generator. At that time the effects anticipated were mainly mechanical, arising from the rapid changes in temperature and pressure brought about by the sodium/water reaction. The corrosion/erosion damage which occurred in the first model test was much more severe than the mechanical effects and this, together with other information becoming available at that time, led to a change of approach. The emphasis was placed on corrosion/erosion instead of mechanical damage. The size of the models in the test vessel was increased to full scale to study these effects and to see if wastage would be enhanced by a larger jet and an increased amount of water.

The size of experiment that it was possible to carry out in a reasonable time scale, was dictated by the requirements of the original model concept. This limited the sodium capacity to 700 lbs and the water capacity to 70 lbs. The maximum available sodium head above the reaction zone was 5 ft. In order to provide further data on the PFR units a larger system was built enabling the sodium capacity to be increased to 3000 lbs, the water capacity to 700 lbs and the sodium head available to 20 ft. Fifteen tests were carried out in the NOAH rig and one has been mounted so far in the Super-NOAH rig. These are summarized in Table 3.

In general the damage resulting from sodium/water reactions within the models has been below expectation. Some distortion to the internal equipment of variable severity and some bowing of water tubes has occurred local to the reaction zone. Damage to the test vessel wall has been negligible and as many as ten major reactions have been carried out in one vessel without detectable damage or distortion.

The main result of the observations made on reaction damage has been the confirmation of the proposition that secondary tube failures can occur. In the four tests where secondary tube failure occurred with full scale PFR water tubes the periods of failure were respectively 4.5 seconds (test 5), 2.5 seconds (test 8), 3.5 seconds (test 10) and 32.5 seconds (test 16) after initiating water injection.

Study of the secondary tube failures indicates two possible mechanisms:

- (a) Corrosion/erosion or tube wastage;
- (b) High local temperatures and water system pressures.

The degree of damage and the speed with which it occurs, by corrosion/erosion appears to depend on the conditions in the reaction zone permitting the formation of a well-developed jet of water from the initial leak. There is also evidence that small orifice water leaks may have more rapid damaging effects than larger leaks because conditions for true jet formation in the former are more likely.

On the NOAH tests the solid and molten reaction products tended to settle out in the bottom of the test vessel. A typical analysis of these products is NOH 50%, Na_2O 40% and NaH 10%, the mixture having a melting point and specific gravity of approximately 300°C and 2 respectively. Considerable difficulty has been experienced in dealing with these dense reaction products when dismantling the rig.

The results may be summarized as follows:

- (a) The tests produced high temperatures (over 1300°C having been detected by thermocouples within the reaction zone in the models). Temperatures were very erratic and localized inducing thermal stresses in the tubes;
- (b) Pressure wave shocks caused only slight damage to the test-tube bundles and none to the vessel wall. In one test vessel (300 lbs/in² design pressure) as many as ten reactions were performed without detectable distortion or other damage;
- (c) Failure of one pressurized water/steam tube in a bundle can lead to failure of adjacent tubes. These secondary failures may be rapid, occurring within two to five seconds of the initial burst;
- (d) Exact duplication of reaction conditions proved difficult and good reproduction of results was not obtained.

- (e) The tests indicate that the causes of tube failure appear to be due to:
 - (1) Severe corrosion/erosion effects on the tubes near the reaction zone;
 - (11) High local reaction temperatures coupled with the high water system pressures.
- (f) All the full bore tube injection tests resulting in secondary tube failure were on ferritic steel bundles. Small-scale tests indicate that austenitic stainless steels have an improved resistance to sodium/water reaction attack and that high nickel alloys such as Incoloy-800 have even better resistance;
- (g) Dismantling and maintenance work following a reaction highlighted difficult problems due to the formation of heavy deposits of reaction products which settled to form a pool at the base of the reaction vessel;
- (h) The amount of gaseous effluent (mainly hydrogen) produced represents 55-60% of the theoretical maximum available. Considerable quantities of sodium were entrained in the gaseous effluent (up to 1/3 of the test vessel sodium charge was displaced into the surge vessel in some tests).

A film was shown on experiments in Dounreay.

MR. FORD: In the period 1955 through 1957, APDA conducted a total of 37 NaK water reaction tests to study the effects of the following variables on the reaction: size of surge volume, presence and size of an inertia line between the reaction area and surge volume, NaK loop initial temperature, water loop initial temperature and water loop initial pressure. These tests were conducted by opening a plug valve situated between a thermal circulation NaK loop and a thermal circulation water loop. The NaK inventory was seven pounds and the valve opening was 1/2 inch in diameter. Surge volumes in the NaK loop from 0 to 29% of the NaK volume were tested and NaK loop initial temperatures of 200°, 400°, 600° and 300°F were explored. The initial pressure of the water loop

varied between 600 and 1200 psi and the initial temperature between 200° and 400°F. Reaction pressures and temperatures were recorded on a Sanborn recorder which had a 75% response to a square-wave input in 10 milliseconds.

In general, poor reproducibility was experienced during these tests. The reaction lasted for about 8 seconds and maximum temperatures exceeding 1500°F were experienced. The effect of a surge volume on the NaK loop was to lower the peak pressures observed during the reaction. Most pressure peaks were 0 to 100 psi above the ^{initial} water pressure; however, some peaks as high as 800 psi above the initial water pressure were measured. The insertion of an inertia line between the reaction area and the surge volume had the same effect as reducing the size of the surge volume, i.e. greater pressures were experienced. Reaction temperatures fell mainly in the 1200° to 2000°F range, and higher temperatures were experienced for the tests for which an inertia line was included, as opposed to tests where no surge volume existed.

In the same period, APDA conducted a total of 21 NaK water reaction tests in a 7-tube heat exchanger to determine the effects of a complete tube rupture on a steam generator. The steam generator consisted of 7 stainless steel tubes in a carbon steel shell. A 1/2-inch diameter plug valve at the end of the centre bayonet tube was used to initiate the reaction. A NaK volume of 5780 cm³ was required to fill the steam generator and loop. Ten tests were conducted with a 29 to 39% volume connected to the steam generator through 10 feet of 1/2 inch diameter pipe. Eleven tests were conducted with the same equipment; however, a rupture disc rated to burst at 37 psig was installed on the steam generator. Water pressures of 1200 psig were used and the NaK pressure ranged from 2 to 50 psig. The NaK and water temperatures ranged from 100° to 475°F.

The temperatures obtained from these tests were 400° to 1000°F lower than those measured during the earlier series. The pressures measured during the tests where no rupture disc was installed were approximately the same as those measured in the earlier series. The tests in which a rupture disc was installed, however, yielded peak pressures 700 to 1000 psi lower than the pressure measured for the earlier series of tests. The reaction pressure for these tests was less than the initial steam pressure. These tests are summarized in References 8 and 9.

APDA conducted a series of 4 sodium/water reaction tests in 1958 to obtain knowledge of the mechanics and effects of a possible sodium/water reaction on the shell of a steam generator. These tests were conducted on a 7-tube, bayonet tube prototype of a proposed steam generator design for the Enrico Fermi Atomic Power Plant. The maximum pressure from these tests was 175 psi and the reaction zone temperatures were in the area of 1300° to 1950°F. Some distortion of the tubes occurred after each test; however, no secondary tube failures occurred. These tests are summarized in References 10 and 11.

A film on sodium/water experiments conducted by APDA was shown.

Small leaks

EP. MAUSBECK: Contrary to the large leak behaviour the small leaks are of interest not only with respect to reactor safety, but also with respect to economy of the different steam generator designs under discussion now. All experiments with respect to the effect of small leaks on adjacent tube walls are connected with the question of economy if one agrees that a small leak will not cause a major steam generator failure. Besides this aspect the probability of pinhole failures at the outer steam generator wall has to be considered. This effect could cause sodium fires and even hydrogen-air explosions outside the steam generator. In spite of the extreme low probability of this event we decided to have an inert atmosphere around the steam generator in ^{the} case of KfK (compact sodium reactor experiment). This is a safe but inconvenient solution. With small leak experiments, we try to prove that a penetration of the outer wall can be avoided by an adequate geometry. This would lead to a much better solution of this special problem.

Now to the economical point of view of the small leak behaviour. We think it can be said that, independent of the type of steam generator, the capital costs are more or less decreasing with steam generator size. This pays off only if the leak probability is small. The leak probability cannot be taken from the operating history of conventional boilers since the inspection rate is much higher and the temperature profiles are much smoother for the sodium-heated steam generators. I think ^{the} large leak probability can be assumed small enough not to become a lay-out criterion

for a sodium-heated steam generator. So the interest concentrates on small leak probability and small leak behaviour. We feel a small leak cannot be excluded even with the best inspection of tubes and welds during assembly. Besides operation experience with sodium heated steam generators is not yet adequate to extract probability data of it.

Mr. FORD: Small leak sodium/water reaction tests were conducted by the Detroit Edison Company after a sodium/water reaction was experienced in the Enrico Fermi Atomic Power Plant steam generators, which resulted in wastage of tubes adjacent to the leak. These tests were conducted at leak rates from .033 to 1.25 lbs/sec in a static pot containing sodium. Wastage rates as high as 4 mils/sec were measured; however, the wastage data was not reproducible. These tests are summarized in References 12, 13 and 14. As part of a contract with the Babcock & Wilcox Company, APDA conducted a series of 9 small leak sodium/water reaction tests with leak rates on the order of .05 to 1 lb/sec. The results of these tests are shown in Table 4. As in the Detroit Edison tests, the wastage results were not reproducible and difficulties were encountered with plugging of the injection orifice. Wastage rates as high as approximately 1 mil/sec were experienced. These tests are summarized in Reference 15.

As part of the AEC programme of steam generator development, a project was initiated at APDA for the purpose of investigating the effects of small leaks. An initial series of tests is being run to establish the means of estimating the extent of tube wastage during a small leak of water into sodium in a sodium-heated steam generator, and to determine the dominant parameters of tube wastage during a leak so that these parameters may be taken into account in the design of a unit.

Water jets were directed through sodium against target tubes in a typical tube bundle configuration. The target tube spacing was varied from 1/4 to 1 inch from the orifice. The conditions of the test were 2650 psi water pressure and 10 psig and 600°F sodium. Leak rates in the range of 10^{-3} to 10^{-1} lbs/sec were produced. These tests were conducted in a large test facility (Rig 10) containing 3500 lbs of sodium and a reaction vessel 35 inches in diameter and 16-1/2 feet high. The quantity of water injected varied from 0.11 to approximately 1 lb.

Figure 11 shows the results of these tests. As can be seen from the figure, the wastage rate appears to reach a peak and then decrease as the leak rate increases from 10^{-3} to 10^{-1} lbs./sec. The highest wastage rate measured was 3 mils./sec for the 2-1/4 Cr-1 Mo material at a spacing of 1/4 inch. Wastage rates for stainless steel were an order of magnitude lower than those for 2-1/4 Cr-1 Mo and wastage rates for Inccloy-800 were even less than that for the stainless steel. During some tests, target tubes with a wall thickness of 0.120 inches were fully penetrated. During the tests at these leak rates, wastage was observed only on the target tubes. At low flow rates, well defined pits were produced on the target tubes, while a donut shaped hole was produced at higher leak rates. The maximum temperature during these tests was 1376°F.

Several tests were conducted in a small test pot 12 inches in diameter and 42 inches high containing 60 lbs of sodium. The same test conditions were used for this facility as was used in the previously described tests. A comparison of the wastage information from this facility (Fig 43) with the wastage data from the larger facility (Fig 10) is given in Figure 12. The data from all the above tests is summarized in References 16 through 20.

In addition to the wastage tests, further small leak sodium/water reaction tests were conducted to explore the temperatures resulting from pin-hole type defects at leak rates on the order of 10^{-3} lbs/sec and at spacings of 1/4 to 1-1/2 inches in a static system. These tests were conducted to shed light on the mechanism causing tube wastage. A reactor vessel 12 inches in diameter and 42 inches high was utilized for this test series and an array of 16 Chromel Alumel thermocouples was fabricated as the target. The thermocouples were 1/16 inch stainless steel sheathed, swaged to a 40 mil diameter at the hot junction. These were arranged as spokes of a wheel with their tips near the axis and were read out on an optical oscillograph.

Eighteen tests were conducted at an initial sodium and water temperature of 600°F and a water pressure of 2650 psi. The results of these tests are shown in Figure 13. The average of the maximum temperature at a 3/4 inch spacing was 1420°F with a standard deviation of 129°F. During most of the tests, the maximum temperature was stable, varying only

approximately $\pm 10\%$. Also, the maximum temperature location was stable and stayed within an area less than 0.1 inches in diameter at a spacing of $3/4$ inch, and $1/4$ inches in diameter at a spacing of $1-1/4$ inches. The boundary of the reaction zone appeared to be well defined since temperature drops of 240°F within a 60 mil distance were experienced which represents a gradient of $4000^{\circ}\text{F}/\text{inch}$. During two tests, the thermocouple in the centre of the reaction zone read 300°F below the bulk sodium temperature while a thermocouple 30 mils away measured 1300°F representing a gradient of $30\ 000^{\circ}\text{F}/\text{inch}$. The theoretical adiabatic temperature for these conditions is computed to be approximately 3000° to 3100°F .

The conclusions from these tests were that (1) the reaction takes place on the surface of a jet containing a steam-water mixture; (2) the inside of the jet is cooled by expanding water; (3) the reaction appears to occur in a 30 mil layer around the jet and the jet disappears in approximately $1-1/2$ inches for the leak rate used during these tests; (4) the reaction appears to have a flame-like structure which is very stable. These tests are summarized in References 16 and 21.

5. ORIGIN OF WATER LEAK FAILURE AND ITS PROTECTION

Mr. PROBERT: It seems likely we are attempting to eliminate defects in the initial tubing, defects in welds, damage to tubes due to vibration or fatigue, as well as to eliminate corrosion from the external sources. The first two lend themselves to non-destructive techniques. The elimination of vibration is a design problem and that of corrosion is a matter of operation. We have attempted to estimate what the size of some of these leaks would be but without any particular success. We consider it very important to find a way of estimating the rate of leakage from postulated defects.

Mr. WATTS: In a single-wall tube heat exchanger the major protection against a water leak is high quality control of components. The P.F.R. uses a single U tube which is subject to close inspection, and the tube to tube plate welds have been carefully developed. Further experimental work which will help to avoid tube failures has been carried out in the fields of tube/support fretting, vibration and flow modelling.

MR. FORD: Unfortunately for the Fermi project we have had considerable experience with defects in steam generators. We had stress corrosion difficulties in a steam generator which resulted in stress corrosion cracking. This defect in the steam generator did not result in a reaction since it was discovered before sodium and water were in the unit. The condition was caused by improper cleaning of the tubes. The cleaning solution contained compounds that could cause stress corrosion cracking and, after cleaning, the tubes were not properly flushed. After the stress corrosion cracking occurred, two of the tubes were taken out to determine what would have happened if we actually had had water and sodium in the units. Leakage from these two tubes, when they were put into a sodium pot and water put into the tubes, was about 520 grammes in 1-1/2 hours. After 22 hours of exposure in this pot of sodium, where the reaction was slowly taking place, the leakage increased ten-fold.

Besides stress corrosion cracking, a sodium/water reaction occurred in one steam generator due to vibration to the point of failure of two other tubes.

In addition, at the time when the bundle was removed for examination of stress corrosion cracking, one tube was found with a gross defect. This tube apparently failed during previous hydrostatic tests and examinations.

Considerable leakage was found recently at tube to tube sheet joints. These are leaks into the cover gas. Typical size would be about 250-400 ppm per minute.

6. METHODS FOR DETECTING STEAM GENERATOR LEAKS

MR. GOLDMAN: When a leak develops in a sodium-heated steam generator, water or steam will react with sodium to form reaction products such as oxides, hydroxides, hydrogen and hydrides, depending on sodium temperatures and saturation conditions. A leak can therefore be detected in principle by monitoring the sodium outlet of the steam generator for these reaction products. Because of the large sodium flows and the desirability to detect small leaks, the detection method must be capable of sensing impurity concentration changes that are less than 10 ppm. Also, because of the high wastage rates which may be associated with a small leak, such a leak should be identified within less than a minute to initiate safety measures before a small leak develops into a massive failure.

Most instruments such as plugging meters or diffusion type hydrogen meters are too slow in their response and/or do not have sufficient sensitivity. On the other hand, instruments which operate on electrochemical principles should be well suited for this purpose. UNC's Liquid Metal Oxygen Meter and Hydrogen Meter are examples.

The Liquid Metal Oxygen Meter was developed under the auspices of the US Atomic Energy Commission(22) and is used routinely at many installations to monitor oxygen activity in sodium. As described in Reference 22, a 10% change in oxide concentration, e.g., a 1 ppm change at the 10 ppm level or 0.1 ppm change at the 1 ppm level, is sensed by the meter as a 2.6 mv change in electrode potential. This change is easily measured and the meter's response is instantaneous for all practical purposes.

Figure 14, taken from Reference 23, shows the response of two oxygen meter electrode assemblies to an injection of sodium hydroxide into 600° F sodium. The injection produced a sharp voltage drop in both electrode assemblies. The drop would be expected to be even larger if the sodium temperature had been higher, since more of the hydroxide would have reacted with the sodium to form oxide, which is associated with a higher oxygen activity.

A hydrogen meter, which like the oxygen meter, operates on an electrochemical principle, is also being developed at UNC under the auspices of the US Atomic Energy Commission.(24) The sensor generates a voltage which is directly related to the hydrogen activity in the sodium. The results of some initial tests, which were performed with early laboratory models in a gas environment, are shown in Figure 15. It can be seen that Sensor No. 8 (solid line) follows the changes in hydrogen concentration (dashed line) extremely well. Assuming a successful development of this meter, it should prove extremely useful in detecting leaks by sensing increases in hydrogen activity in the sodium. In particular, one may envisage its use in conjunction with the oxygen meter, one being designated as the primary instrument for detecting a leak and the other being used for a confirming signal to initiate steam generator isolation, water dump and/or other safety measures.

MR. FURUKAWA: As an improvement to ^{the}nickel diaphragm type water-leak detector, stainless steel bellows 0.015 mm thick were used dipping in the high temperature sodium ^{the inside}with/evacuated. (25, 26)
The merits of this type are the application of greater surface area, mechanically stable configuration, and stable and easily weldable materials.

Further developments of fast response detectors will be possible. However, the more important problems will be ⁽¹⁾the more effective location chosen for ^{the}detector probe or ⁽²⁾the quicker sampling method of sodium, considering the delay time due to the slow sodium flow in the shell of steam generators.

MR. FORD: The leak detection system for the steam generators in the Enrico Fermi Atomic Power Plant consists of two thermal conductivity cells and a gas chromatograph in the cover gas of each steam generator to detect hydrogen produced by small leaks. Also, a pressure detector is installed on the cover gas which alarms at 1 psi above the cover gas pressure and initiates an automatic isolation and dump of the steam generator when the pressure reaches 33 psig. A detector is provided at the rupture disc in the steam generator relief system, which initiates an automatic isolation and dump of the steam generator if the rupture disc is burst.

A large tube surface area is exposed in the cover gas space of the steam generators. Sodium condensation occurs on the cold tube surfaces and reactions between the condensed sodium and hydrogen tend to cause large changes in the hydrogen concentration of the cover gas due to changes in operating temperatures of the steam generator. For example, at no flow a hydrogen concentration of 1700 ppm was measured; however, this value decreased to a few hundred ppm after initiation of water flow through the tubes. After shutdown the feedwater inlet tubes approached the sodium temperatures and the hydrogen concentration again increased to the 1700 ppm range. This effect complicates the detection of leaks in the steam generators by measuring the hydrogen in the cover gas; however, the use of cover gas hydrogen monitors is still feasible for leak detection. This experience is summarized in References 13, 27, 28 and 29.

MR. MAUSBECK: The methods applicable to leak detection in an actual steam generator depend partly on the steam generator design. In ^{the} case of the tube-in-tube steam generator for KNK (compact sodium reactor experiment) the following lay-out was used: bubble meters at the exit of each tube-in-tube unit serve for an identification of the leaking tube. The response time of this measuring device depends largely on the leak size. With extremely small leaks the sodium will be saturated first with hydrogen before hydrogen bubbles will be formed. A bulk detection of hydrogen is ensured by a gas chromatograph in the cover gas. Large leaks will be indicated by pressure measurements at the sodium side of the steam generator and discontinuous level gauges behind the rupture discs, which both lead to a separation of the damaged steam generator.

In case of tube-in-shell steam generators (used in our fast breeder design) the identification of the leaking tube itself is not possible. The fastest detection methods of the leak are:

- oxygen measurement on the sodium side (by plugging meters)
- hydrogen measurement behind a nickel membrane.

7. TEST RESULTS

MR. GOODMAN, MR. BRAY, MR. LIONS and MR. FORD took an active part in the discussion of this item. General conclusions of their presentations are included in item 4.

MR. WATTS concluded from tests carried out in the NOAH and SUPERNOAH rigs that:

- (i) Secondary tube failures can arise from initial small leaks;
- (ii) High pressures, generated during a reaction, rapidly decay;
- (iii) Effluent system components can be successfully used with full plant integrity and safety of personnel.

8. COMPUTING CODES

MR. FORD: The Babcock & Wilcox Company undertook the design of a Full-Size Steam Generator for a 1000 Mwe LMFBR plant for the USAEC-LMFBR component development program. Also included in the program was the design of a model of the steam generator to be fabricated for testing in the Sodium Components Test Installation. APDA was given a sub-contract to determine the effects of a sodium-water reaction on both steam generator designs. This work included the development

of mathematical models to describe the pressure and temperature effects of a reaction. Three mathematical models were developed for this analysis: A piston-type, one-dimensional mathematical model; a finite difference, three-dimensional mathematical model; and a combination of these. The assumptions used in these models are:

- (a) The leak rate is instantaneous and not a function of the back-pressure;
- (b) The reaction is instantaneous according to the equation
$$2\text{Na} + \text{H}_2\text{O} \longrightarrow \text{Na}_2\text{O} + \text{H}_2;$$
- (c) The hydrogen is generated at a temperature of 1800 F and expands isothermally;
- (d) No mixing of the gas and sodium phases;
- (e) The static pressure of the gas acts on the gas liquid interface;
- (f) Gravitational forces are neglected.

Results of the analyses of the Babcock & Wilcox steam generators using these mathematical models is given in references 15 and 30.

MR. FORD: In the analysis of sodium-water reactions for a sodium-heated steam generator, it is difficult to estimate the hydrogen generation rate at every moment even when the water leak rate is known. In usual cases, we have to assume that the rate of hydrogen generation always corresponds stoichiometrically to the water leak rate.

However, the actual chemical reaction takes place in a very complicated manner due to the many factors which influence the reaction. The sodium-water reaction in the large leak case is said to be spasmodic or intermittent. Therefore, the hydrogen generation rate does not always correspond to the water leak rate at that moment.

It is the hydrogen generation rate rather than water leak rate that controls hydrodynamics of the transient two-phase flow of the Na-gas system.

Reported are the results of calculations to estimate the effect of ^{the}hydrogen generation pattern on the pressure effect ^{from a}/ sodium-water reaction.

Several types of hydrogen generation pattern were considered and the change of the gas bubble pressure was calculated as a function of time for each generation pattern. For all ^{of} the calculations the total amount of hydrogen generated in a fixed time period was held constant.

For comparing the effectiveness of the pressure, the time-integral of pressure, or impulse, was used. The peak value of a transient pressure is also important in some cases, but the impulse, I , which is defined by

$$I(t) = \int_0^t P(t) dt$$

is considered to be more important in cases where the time of deformation and damage of a structure is much longer than the duration of the pressure wave in the incident.

The equation for the single-path piston model is:

$$\frac{m}{gCA^2} \cdot \frac{d^2 V}{dt^2} + \Delta P_f + \Delta P_h + P_{cg} = \frac{P_a}{V} \int_0^t Q(t) dt \quad (1)$$

where m = mass of sodium column

A = cross-sectional area of sodium

V = volume of sodium displaced at time t

P_a = atmospheric pressure

Q = rate of generation of hydrogen at the reaction temperature and atmospheric pressure resulting from Na-H₂O reaction.

ΔP_f = frictional pressure drop

ΔP_h = sodium head

P_{cg} = cover gas pressure

t = time

Usually the frictional pressure drop term ΔP_f , the head term ΔP_h , and the cover gas pressure P_{cg} can be neglected compared to the acceleration term for the initial transient period with high rate of hydrogen generation. The simplified equation is

$$\frac{m}{gCA^2} \cdot \frac{d^2 V}{dt^2} = \frac{P_a}{V} \int_0^t Q(t) dt \quad (2)$$

The impulse for several patterns of hydrogen generation rates can be calculated analytically or numerically using Eq (2).

These patterns were selected arbitrarily to check the effect of the hydrogen generation pattern on the value of impulse, so some patterns may be unrealistic.

An example of the calculation results is shown in Fig. 16. In this figure are shown the hydrogen generation rate (input), and the bubble pressure and impulse (outputs) as a function of time. Table 5 summarizes the results.

From the above results we could say that if the average hydrogen generation rate or the total amount of hydrogen generated in a certain time period is constant, the pressure effect on the structure component is not very much affected by the detailed change of hydrogen generation rate, provided the impulse is the controlling factor for evaluating the effectiveness of the pressure.

MR. WATTS: Codes have been used to predict the pressures generated in the hydrogen bubble of a sodium-water reaction, and to indicate sodium velocities in the various parts of a system so as to determine dynamic forces. These pressures and velocities are being checked in a series of reactions on the NOAH and Super-NOAH rigs at Dounreay.

MR. MAUSBECK referred to [7] and [3].

9. DESIGN OF STEAM GENERATORS

MR. MAUSBECK first referred to [7] as concerning methods for calculating effects of reaction on steam generator structure and secondary system (code DRUWA).

In the case of the tube-in-tube steam generator the following safety equipment was developed [7]:

The safety equipment proposed for the German fast breeder prototype power plant is equivalent in layout to the KNK, except the reaction products separation tanks. The last test results showed an improvement by the use of two tanks in series, the first of which was operated as a cyclone. The second one was a simple vessel, open to the air at the top. The connecting tube and the second tank contain a water spray system to wash out the solid reaction products. This system had a retention efficiency of more than 99% as measured by titration. This was also obvious from the visible effect of the small amount of fume loading of the escaping hydrogen. Earlier experiments with much smaller amounts of sodium were always accompanied by a large flame at the outlet of the reaction products tank, even though the titration of the sodium residue in the cyclone always gave retention efficiencies of more than 95%.

With respect to experience on sodium-water reactions in steam generators he referred to [31].

MR. PROBERT:

B&W steam generator design

Under contract to USAEC the Babcock and Wilcox Company has been carrying on work on the design of a full-size steam generator and the design and fabrication of a prototype steam generator that will model the significant problem areas of the full-size steam generator. The full-size steam generator is of a size that three of them would supply steam to a 1000 MW(e) turbine [32]. A typical arrangement of three steam generators and three reheaters is as shown on Fig. 17. The prototype steam generator has a capacity of approximately 30 MW(t) [34]. The relative size of the full-size and prototype steam generator is shown on Fig. 18.

The B&W steam generator is a once-through design having a single wall tube separating the water from the sodium. The general arrangement of the steam generator is shown on Fig. 19. It has the heating surface arranged in helical coil bundles. The design shown on Fig. 19 has a Croloy 2-1/4 boiler bundle and a separate stainless steel superheater bundle. Each of these tube bundles is separately removable from the shell for inspection or repair. Recent studies [35] have shown that with the low fuel costs predicted for LMFBR plants, sodium temperatures and steam temperatures will be somewhat lower than was used for the design shown here, and it has been possible to eliminate the separate stainless steel superheater bundle. Analysis of the single bundle design for its ability to withstand sodium-water reactions has not been completed so for this discussion the two-bundle design will be used.

The outer shell of the steam generator is approximately 1-1/2 inches thick with two sodium inlets in the upper head and one sodium outlet in the lower head. The normal sodium level in the steam generator is a few inches above the top of the superheater coil bundle. There is a 3/8-inch thick liner spaced approximately three inches away from the inside of the vessel shell. This liner extends from above the sodium level to the bottom of the steam generator and is sealed to the sodium outlet nozzle. The space between the liner and shell contains inert gas and this inert gas space communicates with the inert cover gas above the sodium level in the upper portion of the steam generator. The inert gas in the upper head region and between the liner and shell are intended to cushion the shell against some of the rapid pressure excursions due to sodium-water reaction. Also, the liner with the gas space

between it and the shell will protect the shell from local high temperatures that could develop in case a sodium-water reaction occurred in a row of tubes near the outside of a tube bundle.

Sodium flows in through the sodium inlet nozzles at the top through a distributor system and down over the stainless steel tubes of the superheater bundle, then flows over the Croloy 2-1/4 tubes of the boiler bundle and leaves at the sodium outlet nozzle in the lower head. Feedwater flows in through the inlet headers, down through annular rows of downcomer tubes, up through the coiled tubes of the boiler bundle, upward through annular rows of boiler riser tubes, and leaves through the boiler outlet headers. At this location the steam will be slightly superheated. The steam then flows through interconnecting pipes not shown on the figure and enters the superheater inlet header, flows downward through annular rows of superheater downcomer tubes, upward through the tubes of the superheater bundle, upward through superheater riser tubes, and outward through the superheater outlet header to the steam turbine. The tube supports for the coiled tubes in both the superheater and boiler bundles are notched bars arranged in radial rows so that the individual tube coils can expand upward and downward, yet so the weight of the tube bundle can be supported in both the normal vertical position and the horizontal position for shipment. Because of these considerations of the design these tube supports divide the boiler and superheater bundle into eight sectors that are more or less sealed from each other when considering the relief of sodium-water reaction products. This means that the escape path from an assumed leak site is limited to the sector containing the leak until the products have travelled to the top or bottom of the bundle where there are cavities communicating with all eight sectors.

To minimize heat transfer to the boiler downcomer tubes they are arranged with a series of anti-convection baffles. Products from a sodium-water reaction in the downcomer area must go up around the anti-convection baffles to the gas space above the sodium level or downward to the bottom of the steam generator and up through the tube bundles to the gas space. Two twenty-inch rupture discs are located in the upper head communicating with the gas space above the sodium. There are no rupture discs below the sodium level.

Analysis of steam generator shell side structure for accommodating sodium-water reactions

APDA made analyses of sodium-water reactions in various assumed locations throughout the steam generator assuming various numbers of steam generator tubes should fail simultaneously. APDA calculated [36] the pressure and volume in the hydrogen bubble versus time for each assumed leak size and leak location, evaluating the inertia and flow resistance as the products of the reaction were relieved through two parallel escape paths up through the steam generator to the inert gas space above the sodium. It was assumed that when the product reached the inert gas space the rupture discs would burst and the gas pressure above the sodium would be relieved. A typical APDA pressure-time plot is shown on Fig. 20.

Using the APDA data, B&W made analyses of the steam generator shell and internals to confirm that the shell would not fail and become a hazard to nearby personnel even under the very unlikely occurrence of a simultaneous failure of a large number of tubes. Shell analyses were made on an impulse-momentum basis and on a work-energy basis, plus an analysis to see if there could be a local bulge failure if a sodium water reaction occurred at the tube row next to the shell. The significant differences as between the several analyses were in the assumption of how the pressures from the sodium-water reaction were distributed along the shell itself. During^a sodium-water reaction the pressure disturbance is transmitted outward from the leak site at sonic velocity (except for shock waves which were ignored in this analysis) and initially load a local spot on the shell. At later times the pressure disturbance becomes more or less uniform around the circumference of the shell but is a maximum at the leak elevation and decreases to nearly zero at the sodium surface. In all cases the motion of the vessel was calculated using strain hardening data were obtained for Croloy 2-1/4 at room temperature and at elevated temperatures up to 1000 F.

Influence of reaction analysis on design criteria and selection of materials

The design pressure for the sodium side of the steam generator was set at 200 psi, although the operating pressure would be 20 psi or less. This was done in an effort to have a shell thick enough to withstand transient pressures of several thousand psi as shown on Fig. 20. The steam generator shell thickness must be set quite early in the design, certainly before the complete analysis of sodium-water reaction is completed. At the present time this is done by choosing an arbitrary design pressure. As more work is

done on analyzing for sodium water reactions a procedure should be developed for setting the pressure vessel thickness early in the design phase on as realistic a basis as possible. In addition to withstanding the pressure excursions the steam generator must withstand local high temperatures and very corrosive reaction products. Analyses have been made of the liner, shell and tubes for overheating during a sodium-water reaction. It appears that the overheating of the shell will not have penetrated to a sufficient depth to be a serious problem during the time of a sodium-water reaction; especially if there is a liner. Overheating of tubes adjacent to a sodium-water reaction and subsequent failure of these tubes is a distinct possibility, and in fact there were overheating-type failures in the reaction zone during a sodium-water reaction in a Fermi steam generator.

A small experimental programme performed for USAEC by B&W [37] showed that corrosion of Croloy 2-1/4 in high temperature reaction products can be very rapid even in the absence of the reaction itself. Data from APDA tests have shown that tube wastage in the zone of an actual reaction can be extremely rapid for Croloy 2-1/4 material. It may be desirable to investigate alternate materials that are more resistant to this corrosion or to investigate baffling arrangements that would limit the zone of corrosion.

MR. WATTS:

Steam generator design for sodium-water reaction

The PFR steam generators are depicted in a flow diagram (Fig. 21) and in sketches of the superheater (Fig. 22) and evaporator (Fig. 23).

An effluent system relieves pressure from the units in the event of a sodium-water reaction. Major damage to the units can be prevented by using hydrogen detectors or pressure gauges to give warning of a water leak, leading to isolation of the steam generator units and dumping of water and steam in the units to atmosphere through quick acting valves.

Design philosophy for reaction analysis

The PFR effluent system and steam generators have been designed by making various pessimistic assumptions about the sequence of a sodium-water reaction, so that components are generally conservatively sized. A computing code has been used to calculate transient hydrogen pressures and sodium velocities, and experimental results confirm that the calculated values are high.

Conclusion

Experimental results and calculations have given confidence in the design of the PFR steam generator and effluent system to handle a major sodium-water reaction with no damage to other plant or danger to personnel. High standards of control during manufacture are important to avoid the risk of water leaks and sodium-water reactions.

10. DESIGN AND EFFECTIVENESS OF RELIEF SYSTEMS

MR. PROBERT:

Influence of analysis on design and relief system

As the above mentioned analyses were made, ^{the} preliminary results, including engineering judgements by the designers and APDA, were factored back into the design. Several general design rules were observed, based on judgement. The passages for the relief of products of a sodium-water reaction up through the tube bundles were kept as free and open as possible. The anti-convection baffles and tube supports in the downcomer regions were spaced to have minimum obstruction to the flow of reaction products around them. A four-inch safety valve was provided in the upper head communicating with the gas space above the sodium. This safety valve was protected from condensation of sodium or sodium hydroxide in it by a small rupture disc between the steam generator and safety valve. This small safety valve would take care of minor upsets in cover gas pressure and would reclose after the excess pressure had been relieved. On a steady-state basis this four inch safety valve would relieve all of the reaction products from a complete severing of one tube without exceeding the vessel design pressure. In addition to the safety valve two large rupture discs would be installed in the upper head to assure the ability to relieve reaction products from a very large sodium-water reaction. These rupture discs would be set to burst at a higher pressure than the safety valve and should not operate except in the case of a very large accident.

The safety valve and rupture disc outlets must be connected to a relief system that will carry away the hydrogen gas, sodium hydroxide fume and probably some molten sodium. The reaction products must be vented. It is not possible to design a vessel to contain all of the reaction products, because the overpressure of hydrogen that will be required to stop and reverse a sodium-water reaction is extremely high. Although the design of a relief system was not within the scope of the steam generator study carried on so far, some of the requirements of the relief system have been identified.

The relief system must be large enough to handle large amounts of reaction products without significant pressure build-up. The piping must be firmly supported so that it can withstand the reaction forces of high-velocity products passing through it. There must be some type of trap to separate out the molten sodium from the gas stream to prevent a shower of burning sodium over the countryside surrounding the plant. Although in the past the sodium hydroxide fume has been vented directly to the atmosphere it is the writer's feeling that in the future it will be a requirement that some type of filtering system be employed to separate out the sodium hydroxide and vent off only relatively clean hydrogen gas. After the water has been dumped from the steam generator and the reaction has subsided it is necessary to prevent intrusion of air into the steam generator with a resulting fire inside the steam generator. To accomplish this there should be some type of automatic closing valve at the outlet end of the relief system with a purge system to ensure that the gas within the steam generator and vent system is inert.

MR. WATTS: The relief of pressure in a sodium-water reaction is achieved by use of an effluent system. Bursting devices are incorporated in a system of pipework which leads to a surge tank, which is vented to atmosphere. This system has been successfully used in experimental sodium-water reactions.

MR. FORD: The relief system for the Fermi steam generator is described in Reference 13. This system has a 24-inch rupture disc set to burst at 50 to 60 psig. Thirty-inch lines connect each steam generator to internally vaned centrifugal separators. The vent stacks from the separators are closed on the top with rubber membranes and a nitrogen atmosphere is maintained in the relief system. The dump system has provisions for testing its operability without actually performing an isolation and dump, so that the units will not be shocked each time the operability of the system is being checked. Also, there are parallel back-up systems to provide complete redundancy for isolation and dump. A sodium-water reaction took place near the sodium surface in the Fermi No. 1 steam generator and caused very little material transport into the relief system. During this reaction the relief system and isolation and dump system operated satisfactorily to contain the reaction.

11. METHODS FOR SYSTEM RECOVERY

MR. FORD: After the sodium-water reaction in the Fermi steam generator, the primary and secondary pumps of the affected loop were stopped. In attempting to re-start the secondary sodium pump, it was found to be frozen and required several hundred foot-pounds of torque to free it. The sodium in the loop was saturated with impurities and 22 days of cold trapping were required to remove the impurities from the sodium. Based on this experience, installation of cold traps in each secondary sodium loop with the capability of removing reaction products from the sodium in the loop or its dump tank is recommended. When the tube bundle was removed from the shell it was noted that the tube surfaces below the sodium level were completely cleaned by the sodium. However, massive deposits of reaction products were still retained on the tube surfaces in the cover gas space. These deposits were removed by alcohol cleaning. This experience is described in Reference 13.

MR. PROBERT:

Analysis of isolation, cleanup, system recovery and maintenance procedures

It is of vital importance to detect a sodium-water reaction with certainty as early as possible and stop the reaction by eliminating one of the reactants. This means that instrumentation for early leak detection is of greatest importance and that a system for isolating and dumping the steam generator is an absolute necessity. Because the water can be dumped from the water side of the steam generator more rapidly than the sodium can be dumped on the sodium side it appears to be most desirable to stop the reaction by dumping the water from the water side. Also, dumping a reacting mixture of water and sodium to a sodium dump tank might just transfer the problem of containing and venting reaction products to the dump tank. The isolation and dump system that has been used on the Fermi plant contains the essential features that would be required on any once-through steam generator. There must be fast acting valves to close off the connection to the feedwater supply system and the steam header to the turbine, and a fast acting valve to dump the water from the steam generator through the water inlet header. These isolation and dump valves should operate simultaneously on command from a control system triggered by the leak detection system, excess cover gas pressure or failure of a rupture disc. To prevent sodium finding its way into the water system through a tube leak after the steam pressure has been

vented down to atmospheric, there should be a nitrogen purge system that automatically pressurize the steam side of the steam generator to a pressure slightly higher than the sodium side.

After the steam generator has withstood some assumed sodium-water reaction the utility plant operator is faced with a problem of ⁽¹⁾assessing the amount of damage that may have occurred inside the steam generator, ⁽²⁾making an inspection to determine which tube or tubes are leaking, ⁽³⁾repairing or plugging the leaking tubes and adjacent tubes that may be damaged but not leaking, ⁽⁴⁾performing some type of final ⁽⁵⁾inspection to insure that all significant damage has been repaired, and returning the steam generator to service in the minimum possible time. Outages of equipment in large central station power generation plants are very expensive. The cost of an outage of a 1000 MW(e) plant will be from approximately \$15 000 to \$50 000 per day just for the loss in capacity alone, depending somewhat on the availability of less efficient units in the system to pick up the lost capacity.

Quick assessment of damage will require carefully worked out procedures and good access to the steam generator for inspection and testing. This dictates some features of the design such as handholes in the water and steam headers for inspection of the inlet and outlet ends of the tube circuits. Experience at the Fermi plant has shown that having the tube sheets horizontal has been very helpful for leak detection. With the tube sheets horizontal it is possible to flood the water side of the steam generator, pressurize the sodium side of the steam generator with nitrogen and look for bubbles coming from the leaking tube circuits. In the case where major damage to the heating surface is suspected it would be desirable to have the tube bundles removable so that additional inspection can be made and repair work carried on if needed.

12. PROPOSED FURTHER ACTIVITY

MR. PROBERT: Some of the areas that require additional development work are as follows:

- (a) Better analysis methods and experimental confirmation;
- (b) Materials properties such as true stress, true strain, strain hardening exponent, for very rapid rates of loading at high temperatures;
- (c) Design criteria and design of sodium hydroxide fume traps for use with the vent system;

- (d) Design criteria for fire protection vaults, isolation of one steam generator from the other and fire fighting devices;
- (e) Safe maintenance procedures including the criteria for design of equipment. Requirements for isolation valves and rules for operating them, freeze seals, whether freeze seals are permissible, procedures for establishing and maintaining a freeze seal, instrumentation required and limits to location and operations performed near them.

13. SIMULATION TESTS

MR. FORD: Under the sodium-water reaction programme sponsored by the USAEC at APDA, simulant tests utilizing gas and water were conducted. The purpose of these tests was to verify the analytical models used for calculating the pressure in a steam generator during a sodium-water reaction and to establish scaling factors between various size models. The test section for these tests consisted of a water-filled transparent vertical cylindrical chamber with a gas reservoir connected to its base through a quick-opening solenoid valve. Pressures in the test chamber were measured with piezoelectric transducers and recorded on a Visicorder oscilloscope having a 4000 cycle per second frequency response. Also, motion pictures of the tests were taken with a high-speed camera with the capability of 4100 frames per second. The gas supply reservoir had sufficient capacity to maintain the gas pressure within 1% of the initial value for the duration of the tests (40 milliseconds). Three types of tests had been run when the programme was terminated; valve-characterization tests, optical-distortion tests and time-scale synchronization tests. These tests were run with helium and air at pressures of from 40-100 psi and with water temperatures ranging from 38-120°F. The observed flow changed greatly in the 40-60 psi pressure range and became nearly independent of upstream pressure at 100 psi. Critical flow of the gas entering the cylinder was suspected to be the cause of this phenomena.

Test results indicated that modification of the mathematical model to include the inertia of the incoming gas should be made to obtain better agreement between it and the experimental results. Also, the gas density appeared to have a significant effect on the results. The calculated peak pressures for air were 32% higher than the pressure measured experimentally, while those for helium were within 5% of the observed pressures during the test. Analysis of the motion pictures taken during the test showed that

considerable mixing between the gas and water took place. The gas bubble contained up to 50% liquid by volume. The mathematical model assumes discrete gas and liquid phases; however, the piston model was modified to account for slip between the phases, but no comparison has been made between this model and the experimental results. These tests are summarized in Reference 38.

MR. HORI:

Effect of slip between gas and liquid on piston model calculation for sodium-water reactions

For the analysis of sodium-water reactions, the simple piston model is often used. In usual two-phase flow, there is slip between the two fluids (usually gas travels faster than liquid). When slip occurs, the gas phase penetrates the liquid phase, and the liquid velocity of this two-phase mixed region becomes less than that of the liquid single-phase region. Therefore, the average velocity of the liquid column will become less than the velocity of the liquid front.

In the following, the average velocity of liquid column is related to the slip ratio and void fraction in the transient flow case. Using this relation, the corrections of the acceleration term and the frictional pressure drop term for slip are derived.

In this derivation, the slip ratio and void fraction are assumed to be constant.

The derived equations (1), (2), (6) and (7) can replace the corresponding terms in the equation of motion for the piston model to correct for the effect of slip between liquid and gas.

Correction of acceleration term

$$\text{For } \int_0^t V_f dt < \frac{X_{fo}}{S-1} \quad (\text{Figure 24(b)})$$

$$\begin{aligned} \frac{d(V_f)_{avg}}{dt} &= \frac{d}{dt} \left\{ V_f \left[1 - \frac{(S-1) \int_0^t V_f dt}{X_{fo}} \frac{a(S-1)}{1-a} \right] \right\} \\ &= \left[1 - \frac{(S-1) \int_0^t V_f dt}{X_{fo}} - \frac{a(S-1)}{(1-a)} \right] \frac{dV_f}{dt} - \left[\frac{(S-1)^2}{1-a} \right] V_f^2 \quad (1) \end{aligned}$$

$$\text{For } \int_0^t V_f dt > \frac{X_{fo}}{S-1}$$

(Figure 24(c))

$$\frac{d(V_f)_{avg}}{dt} = \left(\frac{1 - aS}{1 - a} \right) \frac{dV_g}{dt} \quad (2)$$

For slip flow, $\frac{d(V_f)_{avg}}{dt}$ defined by equation (1) or (2) should be used instead of $\frac{DV_f}{dt}$ for the acceleration term in the equation of motion of the piston model.

Correction of friction term

$$\text{For } \int_0^t V_f dt < \frac{X_{fo}}{S-1}$$

(Figure 24(b))

In the single-phase region

$$\Delta P_{SP} = \frac{4f (X_f - X_g) \rho_f (V_f)^2}{2g_c D} \quad (3)$$

In the two-phase region

$$\Delta P_{TP} = \frac{4f' (X_g - X'_f) \rho_f (V'_f)^2}{2 g_c D} \quad (4)$$

Where f' is a function of

$$\frac{DV'_f \rho_f}{\mu f} \quad (5)$$

For the entire region

$$\begin{aligned} \Delta P &= \Delta P_{SP} + \Delta P_{TP} \\ &= \frac{4 \rho_f}{2g_c D} \left\{ f \left[X_{fo} - (S-1) \int_0^t V_f dt \right] + \right. \\ &\quad \left. f' \frac{(S-1)}{(1-a)} \left(\frac{1-aS}{1-a} \right)^2 \int_0^t V_f dt \right\} V_f^2 \quad (6) \end{aligned}$$

For $\int_0^t V_f dt > \frac{X_{fo}}{S-1}$ (Figure 24(c))

The entire region is in two-phase flow

$$\begin{aligned} \Delta P &= \Delta P_{TP} = \frac{4f' \rho_f}{2g_c D} (X_g - X'_f) (V'_f)^2 \\ &= \frac{4f' \rho_f}{2g_c D} \frac{X_{fo}}{1-a} \left(\frac{1-aS}{1-a} \right)^2 V_g^2 \quad (7) \end{aligned}$$

Equation (6) or (7) should be used for the calculation of frictional pressure drop in slip flow.

A sample calculation in a simple condition of the two-phase transient flow was made to investigate the effect of the slip using the relation derived above.

The calculation model is as follows: (see Fig. 24)

A constant pressure gas is to accelerate a column of liquid, of which physical properties are assumed to be the same as sodium.

Figure 25 is the plot of the velocity of liquid surface with time for the 8.034 kg/cm^2 gas pressure.

It can be seen from the graphs that the higher slip ratio gives the higher increase rate of the liquid level. This shows that in slip flow the part of the liquid which is not yet penetrated by gas, achieves a higher velocity to compensate for the liquid velocity of the two-phase mixed region. As the slip ratio and void fraction becomes larger, the liquid velocity of the two-phase mixed region becomes less.

Here, we could only estimate qualitatively the effect of slip in a constant leak Na-H₂O reaction case. And that coincides with the commonly obtained estimation that the pressure of the generated hydrogen gas would be lower for the higher slip ratio and void fraction and after the gas reaches the surface the gas pressure would be relieved, partially or totally, depending on the slip ratio and void fraction, by the channeling of gas through the liquid.

D	Diameter of the column, ft
f	Fanning friction factor for liquid flow
f'	Fanning friction factor for two-phase flow
g_c	Conversion factor, 32.2 ft-lbm/lbf-sec ²
P	Pressure, psia
ΔP	Pressure drop, psi
ΔP_{SP}	Single-phase frictional pressure drop, psi
ΔP_{TP}	Two-phase frictional pressure drop, psi
S	Slip ratio ($= V'_g/V'_f$); (different from the usual definition of slip ratio)
t	Time, sec
V_g	Superficial velocity of gas (volumetric flow rate of gas/the cross-sectional area of the column), ft/sec
V_f	Superficial velocity of liquid ($= V_g$); (volumetric flow rate of liquid in the liquid single-phase region/the cross-sectional area of the column), ft/sec
$(V_f)_{avg}$	Weight average velocity of the liquid column, ft/sec
V'_g	Velocity of gas in the two-phase region, ft/sec
V'_f	Velocity of liquid in the two-phase region, ft/sec
X_f	Height of upper surface of the liquid, ft
X'_f	Height of lower surface of the two-phase region, ft
X_{fo}	Height of upper surface of the liquid column at $t=0$, ft
X_g	Height of upper surface of the two-phase region (or height of lower surface of the liquid single-phase region), ft
α	Void fraction (cross-sectional area occupied by gas flow/cross-sectional area of the column)
ρ_f	Density of the liquid, lbm/ft ³
μ_f	Liquid viscosity, lbm/ft-sec

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INTERNATIONAL ATOMIC ENERGY AGENCY

INTERNATIONAL WORKING GROUP ON FAST REACTORS

Specialist Meeting on Sodium-Water Reactions

Argonne, USA, 5 - 6 November 1968

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International Atomic Energy Agency
INTERNATIONAL WORKING GROUP ON FAST REACTORS
Specialist Meeting on Sodium-Water Reactions
Argonne, USA, 5-6 November 1968

AGENDA

1. Opening of the meeting
 2. Fundamental studies of reaction kinetics
 3. Instrumentation for rig use.
 4. Sodium-water reaction tests and evaluation of results.
 - Large leaks
 - Small leaks

Small-scale tests of corrosion/erosion, metal wastage, etc.,
in presence of reacting sodium and water.
 5. Origin of water leak failure and its protection.
 6. Methods for detecting steam generator leaks and development
of instrumentation for detection of sodium-water reactions
in an actual steam generator.
 7. Test results and conclusions from test programmes applicable
to steam generator designs.
 8. Computing codes for the effect of a sodium-water reaction
in a steam generating plant. Comparison with experiment.
Influence of Geometry.
 9. Design of steam generators to accommodate reactions.
 - Methods for calculating effects of reaction on
steam generator structure and secondary system.
 - Description of particular steam generators and
associated sodium-water reaction effluent system,
including ancillary circuits, instrumentation, etc., e.g.
isolation and dump systems for water or sodium, hydrogen
detection equipment; specialized components of an ef-
fluent system, e.g. bursting discs, flame arrestor,
separators, demisters.
 - Design philosophy, i.e., analysis of reaction; postu-
lated sequence of events (usually pessimistic) leading
to particular design of steam generator, effluent
system, etc.
 - Experience of sodium-water reactions in steam generators.
 10. Design and effectiveness of relief systems.
 11. Methods for system recovery following a steam generator leak.
 12. Proposed further tests, i.e. regions of concern for which satis-
factory results have not yet been obtained.
 13. Simulation tests of reactions using other fluids.
- Closing of the meeting.

Reports to the International Conference
on the November 5 and 6, 1968 IAEA Specialists' Meeting on
Sodium-Water Reactions

Session ST-1

November 7, 1968

Bldg. 360

Moderator: Ralph H. Jones
U. S. Atomic Energy Commission

Introduction

Ralph H. Jones, USAEC
Specialists' Meeting Chairman

Gentlemen, I am happy to see the turnout for our rather
ad hoc panelist program.

As Chairman of the IAEA Specialist Meeting here at ANL on
Sodium-Water Reactions, I will hasten to explain several things.
First, the Specialist Meeting was held on Tuesday and Wednesday
of this week. Second, the Specialists representing each country
participating in the meeting were endowed with their assignments
on this panel just yesterday. Third, the Panelists prepared their
statements last night and we have had no time to compare notes.

With this in mind, I believe we can proceed with what should
be a very interesting expose' on a most interesting subject. Our
plan is to have an official IAEA statement by Mr. Yashin followed
by reports by myself and representatives of each participating
country on our appraisals of the meeting and summary reports on
the subjects covered.

Following the Panelists' reports, we will try to answer questions from the audience.

I would like to introduce Mr. Yashin of the International Atomic Energy Agency.

Purpose of IAEA Specialists' Meeting on Sodium-Water Reactions

Dmitri A. Yashin, Scientific Secretary
International Atomic Energy Agency

Let me shortly refer to the history which has led us to the Specialists' Meeting on Sodium-Water Reactions held here the last two days.

About two years ago the idea of coordination of international meetings on the problems of fast reactors was initiated in the International Atomic Energy Agency. This idea was supported by a number of countries. As a result, the International Working Group on Fast Reactors was set up in March of this year under IAEA auspices to serve primarily as a collective adviser for the purposes of coordination of international meetings in the field and better exchange of information between the countries involved in the program. In particular, Dr. Wensch has made a great contribution in organizing the Group and identifying its terms of reference. Now he is the U. S. representative to this Group.

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The Working Group has agreed that in addition to topical conferences, it would be most helpful to organize small meetings of specialists on specific aspects of fast reactors. Such meetings can provide an opportunity for thorough review and detailed discussions of a particular problem, which would be impossible to envisage within a large conference.

The idea of the meeting of specialists on sodium-water reactions, the first of its kind, has been unanimously supported by all members of the International Working Group on Fast Reactors.

The program of the meeting drawn up in conjunction with participating countries was too large for two days' discussion but the high activity of participants and excellent chairmanship by Mr. R. Jones have made it possible to cover all the items.

In conclusion, I would like on behalf of the International Atomic Energy Agency to thank once again U. S. authorities and Argonne staff for the possibility of holding the meeting at Argonne.

Thank you.

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Appraisals of IAEA Specialists' Meeting by Participants

Ralph H. Jones, Meeting Chairman

It was gratifying to me to observe, at this IAEA Specialist Meeting, that all of the participating countries are actively working on the sodium-water reaction problem. From this I conclude that there is international acceptance of responsibility to find the necessary solutions to reactions occurring in sodium-heated steam generators.

In general, all participating countries are pursuing the single-wall tube steam generator as the best approach for the LMFBR plants. All recognize that this approach is contingent upon safely accommodating a sodium-water reaction in the steam generator. This must be accomplished without communicating the trouble to the reactor primary system.

I conclude that the international exchange of ideas at this Specialist Meeting has been of service in assuring the success of the sodium-heated single-wall steam generator.

Kazuo Furukawa, Japan

Although Japan has not yet worked much on the sodium-heated steam generator development except for a few fundamental researches concerning sodium-water reactions, we are now starting a

big project of Fast Breeder Development to meet the National power demands. Therefore, we have heartily appreciated coming to this up-to-date Specialists' Meeting of IAEA, which was very useful to get the guiding knowledges on this field.

We are hoping for further international cooperation.

J. Alan Bray, United Kingdom

I would like to begin by saying how much I enjoyed meeting other members in the field of sodium-water reactions at the Panel. People engaged in this kind of safety study necessarily tend to plow a rather lonely furrow, and it is good to be able to meet now and then. We have had an interesting and fruitful exchange of information and viewpoints this week.

Dr. Kurt Goldmann, United States of America

It was a most valuable meeting.

Amongst safety problems in LMFB power plants, I consider the problem of containing a sodium-water reaction in a safe and economic manner only second to the problem of containing a nuclear excursion. Admittedly the sodium-water reaction problem is an order of magnitude less severe than the nuclear excursion problem, but it seems to me that the efforts spent on sodium-water reactions are more than an order of magnitude less than those spent on nuclear excursions.

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The meeting during the last two days represented a big step forward in the direction of rectifying this unbalance by giving recognition to the importance of this problem. If one cannot safely contain a sodium-water reaction in an economic manner, the utilities will not buy any LMFBR's. While we have reasonable assurance that sodium-water reactions can be contained safely, we are far from knowing what safety design criteria should be applied to LMFBR plants and what the costs of meeting such criteria will be.

For two days, we discussed all aspects of sodium-water reactions. Personnel from major installations around the world, which are actively engaged in this work, were represented. I would estimate that during these two days we collectively advanced the state-of-the-art to a point which might have taken years to reach on an isolated basis. Encouragement should be given to continue the free exchange of information in this special field in the future.

N. Lions, France

1. Discussions were very open.
2. There is a general agreement on instrumentation.
3. It seems that a secondary failure can occur in pool type steam generators; less secondary failures are expected in the case of modular steam generators.
4. A leak of a size less than the total rupture of one tube may be more destructive than the full-tube leak.
5. No test has been run with more than one intentionally ruptured tube.
6. For detection of sodium-water reactions, development is being carried out on an electrolytic device, in addition to the nickel membrane detector which is generally used.
7. Ferritic steel appears to suffer more damage by small water leaks.
8. Very few tests were made to determine the maximum pressure peak which a shell can withstand.

Dr. H. Mausbeck, Federal Republic of Germany

For all the different steam generator designs, the study of the sodium water reaction leads to many similarities in test work and mathematical description. Therefore, this meeting was highly valuable, since it gave a broad exchange of results and

problems. I think it can be said that the meeting offered to a great extent the status of the technology in this field.

Especially, many areas of common understanding and interest were found, like:

- Definition of a maximum steam generator failure
- Codes for pressure propagation through the steam generator and the secondary system
- Wastage of adjacent tubes or construction elements by tube failures, especially by small leaks
- Necessity of measurement and identification of small leaks

Summary Topical Reports on Sodium-Water Reactions .

1. Statement of Problem

Ralph H. Jones, USAEC, Specialists' Meeting Chairman

The study of sodium-heated steam generator for LMFBR plants. For acceptance in this application, a steam generator must be proven to be: -

First	--	SAFE
Second	--	RELIABLE, and
Third	--	ECONOMICAL

To fulfill the first requirement - SAFETY - the LMFBR steam generator must have: -

1. An undebatably high degree of assurance of safety to the general public.
2. Nearly the same degree of assurance that a reaction in the steam generator will not be communicated to the primary reactor system.
3. An assurance of safety to plant personnel equal to, or better than, that in conventional steam plants.
4. A high degree of assurance that a minimum of damage is imposed on the steam generator plant.

The second requirement - RELIABILITY - reflects the application of the LMFBR plant in the electric utility environment. In this commercial application - DEPENDABILITY - AVAILABILITY - RELIABILITY - spell the success or failure of the energy source, regardless of its origin. To meet its demands we must look into the realm of the designer/fabricator. Here the sodium-water technologist must be able to translate his data into a usable, understandable form for use by the designer in arriving at a safe and reliable steam generator design. The designer/fabricator must have a knowledgeable awareness of the consequences of steam generator failures. Only in this knowledgeable way can he institute the proper engineering design and quality assurance programs to fit the end-product application.

Last, but ultimately not least, the safe and reliable steam generator must not impair the economic acceptability of the LMFBR plant in the electric utility market.

It is this spectrum of goals that we are faced with solving today. The technologists, designers and fabricators that can work together to produce the SAFEST, MOST reliable, and MOST economical sodium-heated steam generator have SOLVED THE PROBLEM.

With this in mind--why has there been so little enthusiasm shown by industry to get in to help solve the problem?

2. Fundamental Studies of Sodium-Water Reactions

Kzauro Furukawa, Japan

The topic "Fundamental Studies on the Sodium-Water Reactions" has received rather little attention in this meeting.

However, some comments can be classified into five categories as follows:

(1) Fundamental studies of reaction kinetics

Two presentations were made by UNC (United Nuclear Corporation) and by JAERI (Japan Atomic Energy Research Institute). UNC presented the status of their fundamental study which was conducted a few years ago. Following a literature survey, which was conducted in 1959, their group carried on a small scale study on chemical reaction between sodium and water with absence of air, concluding that the mechanism of the reaction is controlled by mixing the rate of sodium and water, and so on.

JAERI presented a movie showing a series of experiments on sodium-water reactions which was conducted at Tokai Research Establishment in 1967. Sodium of various forms, solid and liquid, was brought into contact with water of various states, steam, boiling, room temperature or ice. The 23 experiments were made

in air changing the reactant weights and contact methods. These showed the very wide variety of reaction modes, from the case of a most violent explosion to the case of no reaction.

Already there are many papers on the sodium-water reactions reported in the literature. However, in order to understand the problems associated with fast reactors, it is essential to continue to accumulate knowledge and information on these reactions.

(2) Thermodynamical properties of sodium-oxygen-hydrogen system

To deepen our knowledge of sodium-water reactions, there is a need for compilation of thermodynamic properties -- particularly the sodium-oxygen-hydrogen system, at a wide range of temperature and pressures, including solid and liquid regions as well as the gas region. However, our knowledge in this field is still limited and was not discussed here.

(3) Corrosion and erosion studies

As another fundamental research field, there are the corrosion and erosion studies of structural materials in several prepared environments which have a relation

to the small-water-leak wastage; for example, NaOH , Na_2O , NaOH / Na , $\text{NaOH} / \text{Na}_2\text{O} / \text{Na}$, $\text{Na}_2\text{O} / \text{Na}$ and others under changing temperatures, pressures and states, and also under changing flow velocities. These were not discussed in this meeting due to the limited time available.

(4) Fluid dynamics

A most important problem in fluid-dynamics concerned with the sodium-water reaction is the transient two-phase flow dynamics of sodium and the products of reaction, including steam and hydrogen. A fundamental research on this subject was reported by APDA. They used air or helium instead of steam and hydrogen, and water instead of sodium.

The reaction simulation method, using a combination of fluid media other than of the sodium and water, is considered to be a very effective tool for the analysis of sodium-water reactions in a steam generator. By this method, the chemical reaction is separated from the hydrodynamics, and the two-phase flow phenomena can be visually observed by using high speed motion pictures.

(5) Instrumentation

Reactions between sodium and water take place in less than 1 milli-second. Because of the very rapid reaction rate, it is difficult to apply ordinary instrumentation in experimental facilities. Therefore, very fast response pressure, temperature and structural stress sensing devices are needed. These devices must be immersed in liquid sodium, which multiplies the difficulties. In this meeting some new improvements and refinements in transducers, thermocouples, strain gages and water flowmeters received some comments.

This was a summary of the "Fundamental Studies" in sodium-water reactions, and I hope for further activity in this field.

3. Large and Small Leak Tests

J. Alan Bray, United Kingdom

The meeting considered the results of a considerable amount of test work carried out in the USA by Atomics International & APDA, in France by the CEA at Cadarache, in Germany by Interatom at Bensberg, and at the U.K. at Dounreay in Scotland. Nearly all the work was carried out on models representing particular designs of steam raising units. The tests have all tended to strengthen confidence in the intrinsic overall safety of single wall sodium-water heat exchangers.

Damaging sodium-water reactions in liquid metal heated steam generators may be divided broadly into two main types:

- a) Small water leaks which may arise from manufacturing defects such as welding faults or flaws in tube material.
- b) Large water leaks resulting from the complete rupture of one or more water-tubes. These might occur from fretting or vibration damage or possibly from water side corrosion if inadequate control of water purity was present. Sodium-side corrosion resulting from unnoticed small leaks could also bring about complete tube failures.

In tests dealing with small leaks where an orifice size of $1/16"$ to $3/16"$ is involved, very rapid wastage rates have been observed on tube material in the path of the jet. Wastage rates of up to 3 mils/sec on 2 1/4 Cr - 1 Mo steels have been reported in the APDA work, and similar rapid metal removal has been seen in tests at Dounreay. For a given spacing there seems to be an "optimum" rate at which maximum wastage occurs. There appears to be some relationship between the nickel content of alloys and their resistance to this kind of attack. Materials such as Incoloy 800 have suffered much less attack than stainless steels which, in turn, have behaved better than the ferritic steels. In an interesting but unexplained phenomenon, APDA has found that the rate of attack in a system where sodium is flowing is increased by a factor of 10 over a static system.

The experiments seem to suggest that these rapid wastage rates are associated with the formation of a definite jet of water-steam fluid in the sodium. Where small leak tests have been carried out on tube-in-tube models, no large wastage has been reported and it appears that the reaction bubble in this situation rapidly fills the cross section of the tube and pushes the reacting sodium water interfaces apart so that an in-sodium jet does not form.

Where large "full bore" water injection experiments have been carried out using orifice sizes up to $3/4$ ", the wastage effect has been completely absent in many tests. For example, in a series of 17 tests in a modular steam generator mock-up carried out by Atomics International, no wastage was observed. However, in some of the Dounreay tests secondary leaks in pressurized target tubes surrounding the injection point did occur within as short a time as three seconds after initiation of the primary burst. The reason for these failures again could be associated with a jet. In the Atomics International module the shell was of comparatively small diameter (8") and the reaction bubble would soon fill the cross section as in the tube-in-tube tests. The Dounreay tests were made in vessels of 2 to 3 feet in diameter, and the reaction bubble could have become detached from the jet itself, thus allowing the rapidly moving stream of reaction product particles at the sodium-water interface to bring about wastage of adjacent tubes.

Temperatures in the reaction zone of up to 1300°C were measured, and pressure peaks of up to 3000 psi were observed for short periods. Some pressure damage such as tube bowing in the reaction zone was reported, but no failures of the outer shell of any of the test rigs occurred. In one series of large scale tests at Dounreay a vessel with a design pressure of 300 psi was used for ten tests without any detectable deformation or other damage. It is obvious that adequate pressure relief devices were used in all of the tests as damage to equipment due to shock or pressure waves seemed insignificant in all the work reported. Some data from AI indicated that transient pressures in adjacent modules may be as high as, or even somewhat higher, than in the reaction zone.

In large leak tests exact duplication of reaction conditions proved difficult, and good reproduction of results was not obtained. However, in small leak work a greater degree of reproducibility was possible. The same is true of tests on comparatively small modules of simple geometry.

Some experiments had brought about failure of adjacent pressurized tubes by high temperatures produced in the reaction zone, but as these tests did not have water flow they may not have been realistic.

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Large scale sodium-water tests highlighted the problems of dismantling and maintenance due to the formation of quantities of reaction products. These had a density about twice that of sodium and therefore settled at the bottom of the test vessel. There was no true melting point, but the material became liquid between 300 and 350°C. A typical analysis was NaOH 50%, Na₂O 40%, and NaH 10%. When cold, the material was rock hard and formed an effective cement, binding the components together. It could be removed using hot water.

In general, it was concluded that economic and safe operation of sodium-heated steam generators was entirely possible.

4. Leak Detection

Dr. Kurt Goldmann, United States of America

I have been asked to summarize specifically what we know about leak detection. I would like to do this in two parts:

- a) Detection of Large Leaks
- b) Detection of Small Leaks

First to the large leaks. You can't miss with the detection of large leaks. Pressure gages on the sodium side and in the cover gas spaces will give immediate indication of large leaks. Continuity wires on rupture disks and shorting wires downstream of them will signal the rupture of a disk. Flow and level indicators may give additional confirmation of a large leak.

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The second problem of detecting a small leak is much more severe. Since the leak is from the water to the sodium side (because of pressure differentials) we have to look for a leak on the sodium side. Sodium-water reaction chemistry tells us that we should look for hydrogen, sodium hydride, sodium hydroxide or sodium oxide in the sodium, depending on temperature.

Since a small leak is < 0.1 lb/sec of water and a typical sodium flow rate in a secondary loop may be 10,000 lb/sec we are looking for a change of impurity level in the sodium of < 10 ppm.

The most commonly used means for detecting a small leak is a hydrogen detector. Such detectors sense hydrogen in the cover gas or after diffusion from sodium through a membrane, which is frequently made of nickel. The membrane may be a flat disk, a small tube or a bellows. The measurement may be made with a mass spectrometer, gas chromatograph thermal conductivity cell or moisture indicator (after catalytic conversion of hydrogen into water). All of these devices require "measurable" amounts of hydrogen to diffuse through the membrane or into the cover gas. The variable amounts of hydrogen normally present in the cover gas make it difficult to recognize a genuine leak. A hydrogen meter, based on electrochemical principles, is under development and should not have some of these shortcomings.

The electrochemical oxygen meter has not yet been used to detect small leaks, but it should meet all requirements. The rhometer might, similarly, be useful.

Plugging meters have been used successfully, but their indication is not instantaneous.

If the leak is large enough to exceed the hydrogen saturation limit of the sodium, hydrogen bubbles form. A hydrogen bubble meter, consisting of an induction coil in suitable geometry, has been used with success in Germany.

APDA has tried to sense a leak by sound detection, but efforts to date were hampered by background noise.

5. Calculation Methods and Applications to Steam Generator Designs

N. Lions, France

Two types of steam generators are now in operation or in design. The first type uses modules containing a few tubes in a small diameter shell. The second type consists of a large diameter shell surrounding many tubes; this is the pool-type steam generator. After the symposium, held during the two last days, we can see that there is pretty good correlation between tests and calculations for the modular steam generator. This is reported in Dr. Mausbeck's paper presented at the symposium and in a paper of the French "Commissariat a l' Energie Atomique" presented at the "Aix-en-Provence Conference" in 1967. In both cases, calculations are made by using a computation model, called "Piston-model", in which water diffusion through hydrogen, which was previously produced, is neglected. The main parameters of the phenomena (pressure pattern, sodium flow) are calculated to within 20% of the experimental values during the first 50 milli-seconds; after this time period, correlations are not so good.

The reproducibility of results of successive experiments was tested in France. It was found to be fairly good when the rupture of the water tube is reproducible.

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Other correlations were shown to fit the test data obtained by Atomics International for another modular steam generator.

On the other hand, Mr. Hori from JAERI presented an interesting theoretical study concerning the slip between gas and liquid for the "piston-model".

Calculations were performed by APDA and UKAEA for the pool-type steam generators; but to date, the results of these calculations have not been checked by actual large-leak experiments.

As a conclusion, we shall say that one must be prudent when using computations of sodium-water reaction effects on steam generators and secondary sodium loops. The calculations are specific for the particular steam generator which was tested and cannot be applied to another type of steam generator.

When the calculations were in good accord with tests for a specific steam generator (tested with a part of the device at full scale), they can be used for the real steam generator and the secondary sodium loops of the reactor to predict the over pressures in any location of the loops and particularly in the intermediate heat exchangers.

6. Design of Steam Generators and Relief Systems

Dr. H. Mausbeck, Federal Republic of Germany

Among the problems covered during the meeting I should mention briefly the question of pressure relief in the case of a tube failure. This question is of common interest for all of the various designs of steam generators. At the moment, all steam generators under consideration are of the single-wall tube, once-through design; they can generally be divided into modular and tube-in-shell types.

The problem of pressure relief is mainly dictated by the formation of hydrogen during the reaction. The entrance of high pressure water/steam into the sodium system contributes to a minor extent. The design of the pressure-relief system influences the design pressure for the secondary sodium system. The questions to be solved can be arranged according to their importance in the following sequence:

- Protection of the intermediate heat exchanger, since it is the safety interface between the primary and secondary loops. Therefore, its failure could cause core destruction-- this means a nuclear incident.
- Protection of the secondary loop and the steam generator itself to avoid secondary damage and large sodium spills.

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- Relief of the reaction products, the excess sodium, and the following steam and water entering through the leak, must be carried out without damage to the immediate surroundings and without risk to the general public.

All the different designs for relief system use rupture discs as a primary means of depressurization. The number and location of the rupture discs varies, depending on steam generator design, expected reaction pressures, and size of the relief lines.

In all the modular designs the rupture discs are submerged in sodium. In case of the tube and shell designs, one often finds a gas space below the discs. In both cases, a most interesting question arises as to how to ensure an adequate and guaranteed lifetime of the rupture device. Additional test work is probably necessary in all cases. The relief line(s) leads to one or more reaction products separation tanks to ensure that only the gaseous products are released to the atmosphere; these are mainly hydrogen and steam.

As a consequence of the high entrance velocities into the reaction products separation tank, some sodium, sodium oxide, or sodium hydroxide escape with the gaseous products. Motion pictures shown at the meeting indicated the importance of an adequate design of the separation system. I conclude, from the material presented, that:

1. The reaction products separation tank has to have a good separation efficiency, as is the case with a cyclone-type device; there is some indication that a separating device with two tanks in series improves the separation.
2. As to the measured data: A retention efficiency of about 95% was reported for a cyclone and more than 99% for two tanks in series with a water spray system in the second one.
3. With respect to the location of rupture discs, two different locations were reported:
 - Directly at the steam generator unit; this gives the fastest pressure relief, but the action of isolation valves is normally needed.
 - Behind additional depressurizer tanks; this has enough additional sodium available, not to depend on valve actions during the pressure relief process.

In respect to the reliability, both of these systems should be adequate.

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Fig.1.

FIG. 2

REACTION $\text{Na-H}_2\text{O}$

Pression dans la bulle d'hydrogène

Essai GLF 1A

FIG. 3

ESSAI GLF 1B

FIG. 4

ESSAI GLF 2A

Pression dans la bulle d'hydrogène

Fig. 5

FIG. 6

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Fig. 8

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FIG. 11

COMPARISON OF RIG 10, TUBE WASTAGE DATA

FIG. 12

COMPARISON OF RIG 10 AND RIG 43 TUBE WASTAGE DATA

FIG. 13

Conditions: Bulk Sodium Temperature, F ~ 600
Injection Water Temperature, F ~ 600
Weight of Injection Water, lb 0.11
Injection Water Flowrate, lb/sec 0.0033
Capillary Diameter (Nominal ID), in. 0.008

- Needle-Like Jet Water Injection
- ⊖ Diffuse Spray Water Injection or Reaction Away from Area of Thermocouple Hot Junctions
- Unexplained Low Temperatures

RIG 43 MAXIMUM METAL SURFACE TEMPERATURE STUDY

FIG. 14

RESPONSES OF OXYGEN METER 1 AND OXYGEN METER 3B BEFORE, DURING, AND AFTER AN INJECTION OF SODIUM HYDROXIDE TO COLD-TRAPPED SODIUM

FIG. 15

- Hydrogen Sensor Voltage Response to H₂ Concentration and Temperature

(PRET-7)

Hydrogen Generation Rate Q (m^3/sec at 1 atm)

Bubble Pressure P (Kg/cm^2 abs.)

Impulse $I = \int_0^t P \, dt$ ($\times 10^{-2} \text{ Kg-sec}/\text{cm}^2$)

Fig. 16 SENSUOLOAL WAVE - II

Period = 10 m sec

Ramp charge at $t=0$

Fig. 17

FULL SIZE STEAM GENERATOR

Fig. 19

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BUBBLE PRESSURE FOR Na-WATER REACTIONS IN THE FULL SIZE UNIT

Fig. 20

STEAM GENERATION SYSTEM FLOW DIAGRAM

FIG 21

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FIG. 22

FIG.23.

Fig. 24. Model of Flow for Slip Analysis

Velocity of Liquid Surface

TABLE 1

Na-H₂O REACTION TESTS IN AIR (temp.: 10°C; humidity: 50%)

JAERI

Na (cm ³)				H ₂ O (cm ³)					Reaction phenomena						
solid	liquid			solid	liquid			steam	explosion		firing	melting		noise (horns)	
	ca. 150°C	burn- ing start	ignit- ing		ice	20°C	50°C	boil- ing	300°C	strong	weak	(no explo.)	fume	no fume	
1.3x5				←		100							X		Remaining 3/4 Na
23				←		100							X		Remaining 4/5 Na
20				←				100					X		Reacting more severely than the case of water
100				dropping		20							X		Deforming by melting
80				dropping				2					X		Fuming stronger than the case of water
	50			dropping		2							X		Scattering Na
	50			dropping		10						X			Unstable reaction, remining 1/2 Na
			50	dropping		10					X				Scattering Na to the distance of 3m
			50	dropping		10					X				Same as the above, and melting the Aluminum dish
			50	dropping			10			X				75	H ₂ -O ₂ detonation
		50		←		100				X				95	H ₂ -O ₂ detonation
			50	←		100				X					Scattering Na to the distance of 5m
	50			←				100		X				95	Detonation, and fuming with steam
		50		←				100		X				100	Scattering Na to the distance of 7m
			50	←				100		X					Reacting weakly than the above case
20				←	3								X		Oxidizing Na surface after ice melt
			50	←	3						X		X	85	Scattering the sodium & ice pieces
100				←					X				X		Applying the dry (superheated) steam cleaner
		50										X			Remaining dark muddy material (m.p. 400°C)
30				→				1000					X		Starting the reaction after 2 sec. of contact and scattering half of Na
	10			→		1000					X				No fire
	10			→				1000		X					No fire
	100			→		1				X					Steam-explosion

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SAFETY TESTS FOR STEAM GENERATORS HEATED BY LIQUID METAL
LINEAR GENERATOR WITH NEST OF TUBES (G L F)

TABLE 2

Test No	STEAM GENERATOR						SODIUM CIRCUIT				SAFETY SYSTEM				RESULTS																OBSERVATIONS																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																												
	Type	Ring	Nest of tubes	Starting conditions			Draining	Volume nitrogen blanket				Starting conditions				Blanketing (rupture pressure)				Max temp	Average temp 30 m	Flow rate through venting apparatus	Hydrogen pressure	Sodium flow rate				Rate of displacement of reaction		Blanketing of energy		Blanketing rupture flow																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																											
				Water pressure	Water temperature	Na temperature		Expansion tank	Heat exchanger	Temp No	Na flow rate	Pressure	#1	#2	#3	#4	calculation	test	calculation					test	calculation	test	calculation	test	calculation	test		#1	#2	#3	#4																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																								
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				bar	°C	°C	m ³	m ³	m ³	%	°C	kg/s	bar	bar	bar	bar	bar	°C	°C	kg/s	bar	bar	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s	kg/s

TABLE 3

TABLE 3.

SODIUM/WATER REACTION TESTS ON THE NOAH AND SUPER-NOAH RIG

NO.	TYPE	MATERIAL	SODIUM SYSTEM					WATER SYSTEM					DAMAGE
			Quantity (lb)	Temperature (°C)			Press. lb/in ²	Temp. °C	Press. lb/in ²	Orifice Dial (in.) and Type	Injection time(sec.)	Estimated amount Injection (lb.)	
				Initial	Peak*	Final							
1	1/6th scale model of 250 MW steam generator	Mild Steel and 18/8/1	680	350	575	500	5	315	2000	1/8 in. vertical	2.0	2	Severe corrosion of M.S. plate and pressurised water line (18/8/1). Slight mechanical damage. Water line (0.020 in. wall) failed
2	Full-size tube in above model	Mild Steel and 18/8/1	720	300	600	350	5	300	800	0.59 vertical	4.5	30-40	No observable corrosion/erosion. Mod. mechanical damage.
3	Full-size tube in above model	Mild Steel and 18/8/1	650	270	540	420	5	300	800	0.59 vertical	4.5	30-40	No corrosion/erosion observed. Only slight mechanical damage.
4	Pressurised tubes	2 1/4 Cr 1 Mo	650	300	510	400	5	300	2000	0.59 vertical with	4.5	30-40	No corrosion/erosion detected. Slight bowing of pressurised tubes.
5	Pressurised tubes	2 1/4 Cr 1 Mo	700	310	625	430	5	300	2000	0.59 vertical with	4.5	40-60	A pressurised tube failed after 4.5 sec. Moderate mechanical damage corrosion/erosion contributed to tube failure. Metal examination indicated temps. of 900°C.
6	Pressurised tubes. Pre-weakened	9 Cr 1 Mo	700	300	1000	400	5	300	1900	0.59 at rt. angle to tubes	4.5	20	Only slight bowing of tubes occurred.
7	Pressurised tubes	2 1/4 Cr 1 Mo	700	310	1060	425	5	300	1900	0.59 at rt. angle to tubes	9.0	40-60	Similar to reaction No. 6
8	Pressurised tubes. Pre-weakened	9 Cr 1 Mo	700	325	1025	550	5	300	0	0.59 at rt. angle	9.0	40-60	Pressurised tube failed after 2.5 sec. Damage and corrosion/erosion similar to reaction No. 5.
9	Corrosion test	Mild Steel	700	315	1060	390	5	300	2000	2 x 1/8 in. 2 x 1/16 in.	9.0	30-40	Negligible corrosion/erosion. Slight bending of specimen wires.
10	Pressurised tubes	9 Cr 1 Mo	700	315	946	580	5	300	2000	0.59 at 45° angle to tubes	4.5	30-40	Pressurised tube failed after 1.5 sec. Little corrosion/erosion. High local temps. and pressures cause failure.
11	Corrosion test	Mild Steel	700	(a) 370	895	390	5	300	1900	1/16 in. at rt. angle to tubes	(a) 30.0 (b) 30.0	15-20	Moderate amount of corrosion/erosion and pitting occurred to a pattern typified by a 'bunsen flame' jet in test (a).
12	Pressurised tube	316 s.s.	700	300	1325	450	5	300	2000	0.59 at rt. angle to tubes	9.0	30-40	Only slight corrosion/erosion causing polishing of tubes surface. Tubes swelling occurred and bowing.
13	Pressurised tubes	9 Cr 1 Mo	700	300	335	335	5	300	2000	Pin hole defect at rt. angle	3600	2-3	Some polishing of tubes in reaction zone
14	Pressurised tubes	316 s.s.	700	300	1190	440	5	300	2000	0.59 at rt. angle	9.0	40-60	Tube swelling and surface polishing and bowing were observed similar to reaction No. 12
15	Pressurised tubes (using Argon not water)	9 Cr 1 Mo	700	375	1000	480	5	300	2000	0.59 at rt. angle to tube	9.0	40-60	Some polishing of tubes observed but no other damage
16	Pressurised tubes	2 1/4 Cr 1 Mo	3200	300	1300	580	5	300		0.59	20	350	Pressurised tube failed 2.5 secs after injection ceased.

* = Maximum temperature recorded on immersed thermocouples. Actual peak transient temperature must have exceeded these values

TABLE 4
SODIUM-WATER TUBE THINNING TEST DATA

RUN NO. DATE	1 6-4-65	2 6-14-65	3 6-22-65	4 6-25-65	5 7-15-65	6 7-21-65	7 7-24-65	8 7-28-65	9 8-2-65
Orifice Diameter, in.	0.016	3/32	0.016	3/32	0.016	0.025	0.025	0.025	3/32
Planned Rate of H ₂ O Injection, lb/sec	0.04	1.0	0.04	1.0	0.04	0.1	0.1	0.1	1.0
Amount of H ₂ O Injected, lb	1 - 2 (Est.)	4.0	--	4.0	1.0	--	--	4.0	4.0
Duration of Injection, sec	~ 180	--	~ 16	3.7	~ 15	~ 1.5	~ 2.4	~ 49	3.7
Initial Temp of Na & H ₂ O, F	350	350	350	350	350	350	350	350	350
Initial Press. of H ₂ O, psig	3000	--	2950	2950	2950	2950	2950	2950	2950
Tube Size, in.	5/8	5/8	5/8	5/8	1.0	5/8	1.0	1.0	1.0
Maximum Wastage, mils									
Tube No. 11	0.6	0.5	1.3	0	0.3	0.1	2.3	4.5	1.6
12	0.9	0.2	1.7	0.2	2.7	0.2	1.9	2.0	2.6
22	4.9	3.3	1.9	0.4	1.8	0	--	1.3	0.7
23	23.7	28.4	2.6	0.8	1.7	0.4	0	0	3.4
24	6.1	1.0	1.5	0.3	0.3	0.4	0.4	2.2	1.8
32	0.9	0.8	1.6	0.1	0.3	0.1	1.1	3.5	0.5
33	2.6	0.2	2.1	0.1	3.5	0.1	0.2	0.9	0.5
34	0.5	0.3	1.5	0.1	1.1	0.9	1.7	0.3	1.6
Maximum Weight Loss, grams									
Tube No. 11	0.0656	0.0602	0.0599	0.1008	1.2	0.0224	0.3	0.2	0.1
12	0.0978	0.0686	0.0543	0.1323	0.5	0.0334	0.2	-0.1	0
22	0.3226	--	--	0.0756	--	--	--	--	--
23	1.2721	0.3776	0.0334	0.0920	--	0.0597	1.2	0	0.2
24	0.4098	0.4229	0.1742	0.1271	--	0.0823	0.6	0.2	0
32	0.0707	0.0900	0.0465	0.0807	0.4	0.0209	0.4	1.1	0.1
33	0.0669	0.0905	0.0381	0.0703	0.1	0.0156	0.1	0.6	-0.1
34	0.0395	0.0608	0.0333	--	0.4	0.0213	0.2	-0.2	0.5
Maximum Temperature in Reaction Zone, F	1000	1057	1286	2014	640	530	413	1261	1551
Inside Adjacent Tube, No. 22	--	426	456	--	450	384	No Ch.	710	510

Table 5 Comparison of Pressure and Impulse

Hydrogen Generation Pattern	Period or Time Const. (m sec)	Maximum Pressure (Kg/cm ²)	Impulse at 40 m sec (Kg.sec/cm ²)	Relative Impulse at 40 m sec
1. Step Change	---	∞ (at t=0)	0.114	1.000
2. Ramp Change	---	2.31 (Constant)	0.0925	0.82
3. Sinusoidal - I Step Change at t=0	10	∞ (at t=0)	0.121	1.06
	20	∞ (at t=0)	0.128	1.12
4. Sinusoidal - II Ramp Change at t=0	5	11.2 (at t=2 ms)	0.116	1.02
	10	7.93 (at t=4 ms)	0.118	1.04
	20	5.61 (at t=8 ms)	0.121	1.06
5. Square Wave	5	∞ (at t=0)	0.119	1.04
	10	∞ (at t=0)	0.125	1.10
	20	∞ (at t=0)	0.134	1.18
6. Sawtoothed - I	10	6.54 (at t=0-5 ms)	0.117	1.03
	20	4.62 (at t=0-10 ms)	0.118	1.04
7. Sawtoothed - II	5	∞ (at t=0)	0.117	1.03
	10	∞ (at t=0)	0.121	1.06
	20	∞ (at t=0)	0.128	1.12
8. Step Input - Linear Decrease		∞ (at t=0)	0.138	1.21
9. Exponential Increase	0.435	15.8 (at t=0)	0.113	0.99
	0.868	11.2 (at t=0)	0.113	0.99
	1.736	8.02 (at t=0)	0.112	0.98

* Calculation Condition

$$t_1 = 0.040 \text{ (sec)}$$

$$\int_0^{t_1} Q(t) dt = 0.040 \text{ (m}^3\text{)}$$

$$\frac{P_a g_c A^2}{m} = 10.0$$

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Frictional ΔP , head, cover gas pressure neglected.