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MINUTES, SPECIALISTS MEETING ON SODIUM-WATER REACTIONS HELD
UNDER THE AUSPICES OF THE INTERNATIONAL ATOMIC ENERGY AGENCY
AT ARGONNE NATIONAL LABORATORY, NOVEMBER 5-6, 1968

October 15, 1969

Argonne National Laboratory
Argonne, Illinois 60439

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MINUTES
SPECIALISTS MEETING ON SODIUM-WATER REACTIONS

Held under the Auspices of the
International Atomic Energy Agency

At Argonne National Laboratory
on November 5-6, 1968

Prepared by: Kurt Goldmann

At Request of U. S. Atomic Energy Commission
UNC Project No. 2357

October 15, 1969

INTRODUCTION

A Specialists' Meeting on sodium-water reactions was held under the auspices of the International Atomic Energy Agency (IAEA) at Argonne National Laboratory (ANL) on November 5-6, 1968. The meeting was attended by representatives from France, Germany, Japan, the United Kingdom, and the United States. Mr. Dmitri A. Yashin, (USSR), representing the IAEA, was Secretary, and Mr. Ralph H. Jones, (USA) was Chairman. A complete list of participants is attached as Appendix A.

The United States was represented by the following team:

Ralph H. Jones - AEC/RDT, Chairman

Kurt Goldmann - UNC, Secretary

John A. Ford - APDA

Leonard Goodman - AI

Paul B. Probert - B&W

This document constitutes the minutes of the Specialists' Meeting for the United States team. The minutes have been reviewed by all team members and have been approved by Mr. Jones, Chairman.

Agenda items for the meeting were as follows:

1. Fundamental studies of reaction kinetics
2. Instrumentation for rig use
3. Sodium-water reaction tests
 - a) Large leaks
 - b) Small leaks
4. Origin of water leaks
5. Detection of leaks
6. Effects on steam generator design
7. Computer codes
8. Design criteria and methods
9. Design features to accommodate reaction
10. Relief systems
11. System recovery
12. Proposed tests
13. Simulation tests
14. Corrosion by molten Na-NaOH mixtures

The last two items were considered additional, to be discussed only if time was available. Time did run out after item 13.

The discussions were recorded and a transcript of the record tape was made available by AEC/RDT, to assist in preparing these minutes. Much of the detailed data which were presented are contained in papers and reports which are listed as References in Appendix B.

Summary reports of the Specialists' Meeting were presented by a panel to the International Conference on Sodium Technology and Large Fast Reactor Design at ANL on November 7. One participant from each country served on the panel and gave his appraisal of the Meeting and a summary report on a major topic. Mr. Yashin reviewed the history which led to the Specialists' Meeting. Mr. Jones was the moderator of the panel. The panel discussions are contained in the Proceedings of the Conference, ANL-7520, Part I, pp. 4-11. They are attached hereto as Appendix C, in lieu of a separate summary.

1. Fundamental Studies of Reaction Kinetics

Presentations by: Japan - Furukawa
 United States - Goldmann

On a molecular scale, reaction rates of sodium and water are on the order of microseconds. There is considerable controversy as to what the rate limiting steps of the reaction are, once the sodium and water have been brought into contact with each other. In practical systems, the rate limiting step is probably the time required to bring sodium and water into contact. These rates are milliseconds or larger.

Tests performed at UNC (NDA) ten years ago, demonstrated that the temperature and state of the sodium, i.e., whether it is solid, molten or ready to boil, has a strong effect on reaction rates. At temperatures near the melting point of sodium, the reaction rates were measured to be proportional to the surface area of a sodium drop. The rates increase drastically as the sodium temperature approaches saturation temperature. A deflagration stage is reached when the sodium starts boiling and the drop bursts into numerous droplets (creating a large sodium surface area for contact with water.) The foregoing observations were made in systems in which the sodium and water were blanketed with an inert gas. In the presence of oxygen, a 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.

Additives which increase the viscosity of the water lead to higher reaction rates. Surface reaction rates can be greatly reduced by covering the water surface with oil or other inhibitors.

A movie showed tests which were performed in Japan, in which large quantities of sodium and water were brought into contact in an ambient air environment. The tests were performed with water into sodium and sodium into water at various temperatures, including ice onto solid sodium. Explosions were observed when molten sodium was dropped into water. The explosions were attributed to "steam explosions." The possibility that the explosions were caused by hydrogen-oxygen reactions was raised.

It was concluded that the rate limiting step of a sodium-water reaction in a sodium heated steam generator was not dependent on the reaction kinetics but rather on the transport mechanisms of bringing the reactants together.

There is no work in progress or planned in regard to fundamental studies of reaction kinetics.

2. Instrumentation for Rig Use

Presentations by: France - Lions
 Germany - Mausbeck
 United Kingdom - Bray
 United States - Goodman

Thermocouples measurements are used to infer the velocity of the reaction front. Because of the high velocity, such

measurements are not useful in small equipment. Stainless steel sheathed, cromel-alumel couples, 0.25 mm diameter with a 20 msec response time are used in France. Experience in England showed that 0.5 mm thermocouples "just blew away" and a standard 1/8" sheathed, DFR thermocouple is now used. (Grounded hot junction). Four to five millimeter thermocouples are used in Germany.

Sodium front velocities are measured with a series of spark plugs which short one after another as the sodium reaches them.

A gamma ray level gage was used in one vessel test in Germany to get an understanding of whether the piston model of sodium movement is applicable or not.

Piezo-electric transducers are used to measure sodium pressure directly. Strain gages on external surfaces of the test equipment measure sodium pressures indirectly. In France, the Kistler piezo-electric transducer, type 601-A is in use. They have developed a technique for using it for pressures up to 2,000 atmospheres.

The Barton Cell is a similar transducer used in Germany. They use an adapter which is filled with an organic grease to maintain the transducer itself below its maximum permissible operating temperature of 240°C. The transducer has a frequency range up to 170 KHz; the preamplifier 100 KHz; the main amplifier 20 KHz, and a light-beam oscillograph with a range of 0 - 4,000 Hz. A cathode ray oscilloscope is used

to get better resolution. With the latter, pressure rise times of 5×10^{-5} sec were measured in sodium-water reaction tests. The equipment was checked during detonator tests for which pressure rise times of 5×10^{-6} sec and pressures of 7,500 atmospheres were measured.

Piezo-electric transducers which are used at AI are not specifically designed for cooling. Their backside was left open and nitrogen from a gas bottle was used to maintain a temperature of 500°F for a 900°F test.

The development of capacitance type pressure transducers in England has not been successful, and they are now going to use piezo-electric transducers. They also encountered difficulties in interpreting strain gage data because temperature and pressure transients arrived at the gages simultaneously.

It was noted that water flow rates were generally not measured and the question was asked, "Why not?" The best calculations are a long way from duplicating transient flow measurements. The state of the water, i.e., subcooled vs. two phase flow, make flow measurements difficult. For earlier tests in Germany, water flow was measured by observing its level in a container. A number of turbine type flow meters have been tried in England since 1961 - none successfully. A meter is being developed by Dr. J. J. Hunter of the National Engineering Laboratory in England, but it has also not been successful in sodium-water reaction tests. At AI a high response (~ 5 msec) flow meter was used but its location degraded the quality of the transient data obtained.

3. Sodium Water Reaction Tests

a) Large Leaks

Presentations by: France - Lions
Germany - Mausbeck
United Kingdom - Bray
United States - Ford, Goodman

Large leak tests have been conducted in France, Germany, the United Kingdom, and the United States to establish the effects of tube ruptures on the structural integrity of sodium-heated steam generators. Specific test objectives were stated as follows:

- a. Determine pressure rise vs. time at different locations.
- b. Determine discharge rates of reaction products.
- c. Determine whether a reaction can be contained in the steam generator shell.
- d. Determine the effects on connecting systems and associated plant equipment.
- e. Obtain test data in support of analytical model development.
- f. Establish conditions for large vs. small leaks (Under what conditions does the reaction become localized?).
- g. Develop clean-up and recovery procedures.

The test equipment generally consisted of tube bundles in shells, water in the tubes and sodium on the shell side. Cover gas spaces were provided in the shell above the sodium.

A tube burst was initiated several feet below the sodium level by admitting high pressure water into a tube which had a weak end cap, had a 4 inch long flat machined on one side, or by applying tension to a thinned section of a tube. In the largest test, 400 lb. of water at a pressure of 2,350 psi was discharged during a 20 sec period into 3,000 lb. of sodium which was preheated to 572°F. Diaphragms were installed in shell nozzles for pressure relief. The reaction products were piped through retention tanks or cyclone separators into the air or, at one installation, into a pool of water. Pressures and temperatures were measured as a function of time. The total number of tests which have been performed by the five organizations which reported on large leak tests are on the order of 100.

The measured reaction temperatures fall generally in a 750 - 950°C range, well below the theoretical values for complete reaction (whether $\text{H}_2\text{O} + \text{Na} \rightarrow \text{Na}_2\text{O} + \text{H}_2$ or $\text{NaOH} + \text{H}_2$ is assumed). Single peak measurements of 2,000°F and 1,300°C were reported (still below theoretical values).

Similarly, measured pressure peaks, generally in the 400-700 psi range fall well below values which were calculated with a proposed model. Measured pressure peaks in the AI tests were frequently equal to the initial water pressure (up to 2800 psi). The initial pressure peaks have a width of approximately 1 msec.

Correlations between predicted and measured pressure

histories were generally poor. It was stated that if the water pressure during the first few milliseconds is not accounted for properly, the calculated reaction pressures may vary by factors of 2 or 3. The contents of the reaction bubble are hydrogen and steam. Assumptions have to be made regarding the temperature in the bubble. Better agreement between test and calculations is hoped for in the French program by accounting for the compressibility of the sodium and the elasticity of the walls.

The test vessels were designed to withstand steady state pressures from 300-1500 psi. None showed any damage. In several instances, adjacent tubes were distorted. It is not known to what extent the loss of material strength due to higher temperatures contributed to the distortion. Generally there was no corrosive or erosive attack observed for the large scale tests, except in one test a secondary tube failure occurred as a result of wastage from the primary reaction.

It was noted that all tests were conducted with a single tube rupture and questions were raised as to whether it was possible for more than one tube to rupture simultaneously in a steam generator and cause higher pressures. It was the general consensus that truly simultaneous failures would be most unlikely. A subsequent rupture won't effect the initial peak pressures, which are highest, because not enough sodium is left for reaction and also the sodium column has already been accelerated and is moving out of the steam generator. It was pointed out that tube bundles with cross-flow configurations, might experience more tube bending as

a result of a reaction, but again it was considered unlikely that they would rupture because of ductility and supports. A question was raised as to the possibility of several tubes having been thinned (e.g., from vibration) to the point of almost rupturing and then experiencing simultaneous failures as a result of a change in plant operation (e.g., major thermal transient). The question was not answered.

The sodium was initially static in all tests. This was not expected to effect the results except that during the period of flow reversal in a real system, excessive pressures may be generated in the sodium inlet line to the steam generator. (Sodium velocity does play an important role in small leaks.)

At AI, it was noted that the water flow rates through tubes which had longitudinal splits were considerably lower than expected. For cross-section areas of the hole after the tube split of twice the flow area, water flow rates were only $\frac{1}{2}$ that observed with dead end ruptures.

The tests were observed and photographed by personnel who were permitted within 100 yards of the test equipment. A cyclone type separator achieved a 95% retention efficiency for reaction products. Combined with an open tank in series, the retention efficiency was boosted to over 99%. Discharge piping failed in several tests, because of improper design. In one test, the adjacent building wall was set on fire, presumably due to radiation from the hydrogen-oxygen flame in the discharge plume.

Special tests were run in Germany to establish the conditions for large and small leaks. A large leak was defined as one in which there is a piston-like movement of the sodium column away from the reaction zone; and a small leak was defined as one in which the reaction zone is more or less stationary. The heating in the surrounding reaction zone is quite different for the two different types of leaks. The hole size for a small leak was found to be 2 mm or less with 300 psi water and 3 mm or less with 900 psi water, under the conditions of the test.

b) Small Leaks

Presentations by: Germany - Mausbeck
United Kingdom - Bray
United States - Ford

Small leak tests were run at APDA to determine the extent of damage due to wastage. Water jets were directed through sodium against target tubes in typical tube bundle configurations. The target tube spacing was $\frac{1}{4}$ " and 1" from the orifice. Typical conditions were 2650 psig water and 10 psig, 600°F sodium.

Wastage rates increased and then decreased with increasing water flow rates through a range of 10^{-3} to 10^{-1} lb/sec. The maximum wastage rates occurred at water flows of approximately 10^{-2} lb/sec. The highest observed wastage rate was 3 mil/sec, for which the tube material was $2\frac{1}{4}\text{Cr-1Mo}$. Wastage rates for austenitic stainless steels were an order of magnitude lower and were even less for Incoloy 800. The wastage rates were constant with time at constant conditions. Tubes with maximum wall thickness of 0.120 in. were fully penetrated.

Appreciable wastage was observed only on target tubes. Low flow rates caused pits while higher rates produced doughnut shaped craters due to the cooling effect of the water at the center of the jet. On tests in which an array of thermocouples was the target, thermocouples at the center of the jet measured a temperature of 300°F below that of the sodium, while a thermocouple 0.030 in. away from the center measured 1300°F . The temperature gradient was $\sim 30,000^{\circ}\text{F/in.}$ Readings from closely spaced thermocouples remained quite steady during all runs, indicating that the reaction zones were stable and well defined. Sodium flow perpendicular to the water jet increased wastage rates over those observed with stagnant sodium. However, there was no correlation between wastage rates and sodium velocity.

A real pin hole failure developed in a $2\frac{1}{4}\text{Cr}-1\text{Mo}$. tube of a 5MW test steam generator in Germany. The failure was recognized by increasing pressures and hydrogen concentrations in the cover gas after 54 hours of operation. Upon draining the unit the leak could not be found and operation was started again. After an additional 14 hours there were further indications of a leak from a bubble detector. Also the rotational speed of the centrifugal pump increased while the flow almost stopped and actually reversed in some portion of the system. The leak was caused by a tube material defect adjacent to a weld. By varying weld procedures, it was subsequently demonstrated that similar defects, consisting of a hard spot and small cracks, could be duplicated by holding the weld ignition arc too long (3 sec). It may be difficult to detect such a defect during fabrication. During operation with 2 additional pin hole leaks, it was noted that a plugging meter detected the leaks before the bubble meter, because of the high solubility of hydrogen in sodium at high temperature. No damage other than a slight roughening of the annular tubes was observed in these tests.

A section of a tube which developed a pin hole leak in a large test facility was cut out and installed in a small leak test facility in the United Kingdom. It was impossible to keep the hole open. Other instances of self-healing were reported. It was suggested that weld-defect holes are more

prone to self-healing than artificial holes. At APDA capillary tubes as small as 0.004 in. diameter did not plug while 0.016 in. and 0.025 in. orifices in nozzles just upstream of rupture disks, utilized for reaction initiation, plugged in as little as 1.5 sec.

4. Origin of Water Leaks

Presentations by: United Kingdom - Watts
 United States - Ford, Probert

The following items were mentioned as most likely causes for water leaks:

Weld Defects

Vibration - fretting and fatigue failures

Corrosion - external and internal

Lack of Full Quality Assurance Coverage

To minimize the probability of a weld defect failure, the PFR steam generator uses single U tubes with welds only at the tube-tube sheet joints.

Assuring that there is no excessive vibration is a design problem which may require model testing.

Corrosion is an operational control problem. Several failures have been traced to improper cleaning procedures. Of special concern are stress corrosion. Every metal is susceptible to stress corrosion under some conditions, e.g., nitrates and caustics will attack ferritic steels while chlorides or caustics will attack austenitic stainless steels.

A recent failure was traced to an agent in a cleaner-passivating solution which has been used for years without apparent difficulty. Possibly the use of different cleaners in series caused the problem. It is being suggested not to put any chemicals into equipment so one doesn't have to be concerned about getting them out later. Equipment should be drainable.

From a quality assurance point of view, it is important that a planned, coordinated program between manufacturer, installer, and user be carried out.

Secondary tube damage due to overtemperature was discussed under item 9.

5. Leak Detection

Presentations by: France - Lions
Germany - Mausbeck
Japan - Furukawa
United Kingdom - Bray
United States - Ford, Goldmann

The detection of small leaks was of principal interest. The following instruments were mentioned for leak detection:

1. Hydrogen Detectors (in sodium and cover gas)
 - a. Diffusion types with:
 1. mass spectrometer
 2. conductivity cell
 3. moisture indicator
 - b. Electrochemical cell

2. Oxygen Meter
 - a. Electrochemical cell
3. Gross Impurity Meters
 - a. Plugging meter
 - b. Rhometer
 - c. Bubble meter
4. Noise Indicator
5. Pressure Indicator

Nickel diaphragms and tubes have generally been used satisfactorily for hydrogen diffusion type meters. Substitution of 316 SS for nickel was found acceptable in Japan, although diffusion rates were 2 to 3 times lower. Diffusion rates decrease with temperature.

More hydrogen was observed in the $2\frac{1}{4}\text{Cr-1Mo.}$, Phenix evaporator than in the higher temperature, austenitic stainless steel superheater. This apparent anomaly was explained by hydrogen production as a result of water corrosion of ferritic steels. Hydrogen was measured with a diffusion type detector. The detector was calibrated by injecting 1 ppm hydrogen directly into the NaK. The nickel diaphragm was held at 400°C . On the basis of this calibration, the hydrogen resulting from the corrosion on the water side of the tubes was estimated at 150 mg/hr following start-up after a non-operating period. After a few hours of operation this value decreased to 24 mg/hr and remained steady.

In the Fermi steam generators a large tube surface area is exposed in the cover gas space. Sodium frost on these tubes affects hydrogen concentration in the cover gas. For example, at no flow, a hydrogen concentration of 1,700 ppm was measured. This value decreased to a few hundred ppm after initiation of water flow through the tubes. It is presumed that sodium condensation and subsequent reaction with the hydrogen on the cold tube surfaces is the cause. Upon shut-down, the feedwater inlet tubes approached sodium temperatures again and hydrogen was released. This phenomenon complicates the detection of leaks by measuring hydrogen in the cover gas. However, the method still has merit for steady-state conditions.

Detection of a leak by measuring hydrogen in the cover gas is relatively slow because of the transport time through the sodium from the location of the leak to the cover gas space. Diffusion type meters have additional time lags due to the time required for measureable amounts of hydrogen to diffuse through the membrane.

Faster responses are expected with electrochemical cells. A limited amount of data was obtained at APDA on the response characteristics of an available oxygen meter following water injections into sodium. The response is instantaneous but the magnitude of the voltage changes is a function of variables such as sodium temperature and prior impurity levels, whose interactions are not understood. An electrochemical hydrogen meter is under development. Initial tests of laboratory models show promise.

Highest response rates should be obtainable from sound detectors. Tests conducted at APDA so far have shown difficulty in distinguishing between plant background noise and noise generated by small leaks.

For a leak rate corresponding to the highest observed wastage rates, none of the instruments except possibly the noise detector would be fast enough to prevent a tube burst, especially considering steam generator dump times of 1 minute. To improve this, it was suggested that detectors be placed in critical areas within the steam generator. Maintaining low cold trap temperatures and therefore, low initial oxygen and hydrogen content in the sodium, should assist in detecting leaks with oxygen or hydrogen detectors.

6. Test Results as Applied to Steam Generator Design

Presentations by: France - Lions
 United Kingdom - Bray
 United States - Goodman, Ford

Peak pressures do not necessarily occur in the reaction zone. Twenty to twenty-five percent higher pressures have been measured at other locations at which pressure waves following two different paths reinforced each other at AI.

The peak pressures measured in single tube failure tests may not be conservative for design. One series of tests was run at AI with 900°F sodium and 2800 psi, 250°F water with a single pressurized tube. Peak reaction pressures were 600 psi with a tube-split failure and 1000 psi with a dead-end failure.

With additional tubes connected via a manifold increasing the immediately available water volume 10 times, pressures in excess of 1600 psi were measured at most locations. For the latter test, the water temperature was also higher, 380°F.

More attention should be paid to the design of the secondary system, i.e., IHX, pumps, valves and piping to withstand pressure pulses resulting from sodium-water reactions in the steam generator. The IHX is particularly important because it interfaces with the radioactive primary system. Long pipe lines will probably help in attenuating pressures. However, increased L/D ratios between the location of a leak and pressure relief devices will increase peak pressures. It was stated that a system could probably take 10 times design pressure for 1/10 sec without failing, but nobody has performed any analysis on the effect of over-pressures lasting for 10-20 msec, which are typical.

Secondary leaks may occur as a result of erosion-corrosion or over temperature. A peak temperature in excess of 1300°C has been measured. To minimize the probability of a secondary leak, the design should provide for dumping to relieve water pressure as soon as a leak is detected.

Reaction products can sink to the bottom of the vessel and form a cement-like mass. In one series of tests in the U.K. their

density was 2, composition: 50% NaOH, 40% Na₂O and 10% NaH. The melting point was 300 - 350°C, but attempts at draining at 550°C were not successful, either due to high viscosity or insufficient preheat of drain lines. (See also items 9 and 11).

The difficulty of developing analytic models for predicting temperatures and pressures was demonstrated at APDA by the poor reproducibility of data from one sodium-water reaction test to another. For example, by applying a statistical analysis to a large number of tests, it was found that the maximum temperature had a probability range from 800 - 2400°F. Peak pressures varied from less than water pressure to several hundred psi at maximum probability to 800 psi above water pressure for the lowest probability. On the other hand, reproducibility of 4 out of 5 tests was reported by France.

It was stated that the prediction of pressures away from the reaction zone was strongly dependent on the compressibility of the sodium and the elasticity of the piping. These parameters would have to be included in the analytic models for the prediction of pressures at the IHX. Codes should be developed since one cannot set up full scale experiments for all geometries. No solution was offered to the problem of non-reproducibility of test data in the reaction zone, but this may not be of consequence to the prediction of pressure propagation around the system.

7. Computer Codes

Presentations by: France - Lions
Japan - Hori
United Kingdom - Bray
United States - Ford, Goodman

Since codes which attempt to predict the high frequency pressure oscillations in the reaction zone have not been successful, present efforts are directed towards models which utilize time averaged pressures and temperatures. These approaches rely on a piston model which basically assumes that the hydrogen, resulting from a sodium water reaction will form a spherical and eventually a cylindrical bubble which will expand and push the sodium out of the steam generator like a piston. Different organizations have made different assumptions in regard to water leak rates from ruptured tubes, temperatures within the bubble, presence of steam in the bubble, compressibility of sodium, etc.

Computer codes developed by APDA are described in References 29 and 30. They were used in the analysis of the B&W steam generator designs. The results of one set of calculations shows that the peak pressures from the simultaneous rupture of 10 and 676 tubes was 10 and 40 times higher, respectively, than the peak pressures resulting from the rupture of a single tube.

A single tube failure is expected to clear all sodium from the PFR evaporator within 1 sec. Additional tube failures are not expected within such a short time. The United Kingdom code models the complete secondary circuit with all bypasses.

The model which has been used in France's computer code is described in Reference 12. Simplifying assumptions include a) All the water and sodium react adiabatically and instantaneously to form hydrogen; b) The hydrogen expands isothermally at 1000°K ; and c) The sodium is incompressible. As mentioned before, the code is being modified to account for the compressibility of the sodium and the elasticity of the steam generator shell to obtain better agreement between predictions and test results. For the first few milliseconds, large differences between calculations and experiments were found if the calculations did not account for the initial water flow transient due to inertia effects. It is also suspected that better agreement would result if the initial water temperature transient were included.

With the assumption that the hydrogen generation rate rather than the water leak rate controls the hydrodynamic pressures, Hori presented the results of a study in which impulse was calculated for various assumed hydrogen generation patterns. These included step, ramp, sinusoidal, square wave, sawtooth and other changes. The results showed that the total impulse after 40 msec. is rather insensitive to the hydrogen generation pattern.

AI's computer model shows good correlation with experiments, if it is assumed that steam production at sodium temperature, rather than hydrogen generation, is the pressure source. Thus, water leak rate may be important.

8. Design Criteria and Methods

Presentations by: France - Lions
Germany - Gast
United Kingdom - Watts
United States - Goodman, Probert

Steam generators must be designed to operate safely, reliably and economically. The designer must assume that there will be at least one leak during the lifetime of the unit. The question as to how many occurrences the steam generator should be designed for was raised, but not answered. Reaction products must be relieved safely and in an acceptable manner to the community. A small leak should be detected and the system should be designed to dump water from the steam generator and isolate it before a large leak develops. The IHX must not fail as a result of overpressure from a sodium-water reaction, and especially if there was such a failure, pressure waves must not propagate through the primary system sodium to the reactor core, or destroy its decay heat removal system.

The steam generator shell must not rupture, but yielding is permissible. There is essentially no information on the yield strength of steels at high strain rates and high temperatures. Strain hardening at high strain rates will increase the yield strength so that the use of data obtained at low strain rates will probably be conservative. No accepted design procedures are available for handling non-uniform pressure

distributions and transient, unsymmetrical loadings. These uncertainties are presently overcome by using factors of safety and factors of ignorance.

Pressure drop calculations for the relief system are generally based on all liquid or all gas flow. Pressure drops are much higher for two-phase sodium-gas mixtures. Two-phase pressure drop factors should be applied. The pressure relief system should be designed so that no large amount of water or sodium can accumulate and suddenly be brought into contact with each other.

Provisions must be made in the design to clean-up after a sodium-water reaction and to make repairs to the steam generator and other components as necessary to get back to power in the shortest possible time.

9. Design Features to Accommodate Reactions

Presentations by: Germany - Mausbeck
United Kingdom - Bray
United States - Probert

The B&W full scale and prototype steam generator designs have shell liners to protect against thermal transients. The gas spaces between the liners and shells also serve as additional barriers against the effects of sodium-water reactions. A large gas space is provided above the sodium to serve as a gas cushion. Single wall tubes are used, rather

than double wall, since complications in fabrication and inspection are thought to make the latter less reliable. Two 20 in. rupture disks are provided on top for large failures. A 4 in. safety valve, with a rupture disk under it to protect it against sodium frosting, will relieve relatively minor pressure excursions. Calculations were made on the distortion of tubes and support structure, but the present transient stress analyses methods are considered to be inadequate. The shell was analyzed on an impulse-momentum and work-energy basis. The former was found to be more conservative. The allowable strain was assumed to be the strain corresponding to the maximum stress on a conventional stress-strain curve. The results showed that 20-40 tubes could fail simultaneously before the vessel would fail.

A unique pressure relief system, which has been developed for the KNK steam generator, was described. It consists of depressurizer vessels and parallel relief lines connected through suitable orifices in the sodium inlet and outlet lines of the steam generator in such a manner that components in the secondary system will not experience the high pressures resulting from a sodium-water reaction in the steam generator. Test data were presented for an excursion in which the pressure in the simulated steam generator reached 80 atmospheres, whereas the maximum pressure in the adjacent system was only 26 atmospheres. The maximum measured temperature in the depressurizing tank was 1300°C after 400 msec. The maximum

wall temperature rise was only 50°C. Spark plug type level gages, sodium pressure gages and a bubble meter are used for leak detection. The bubble meter is designed to detect a hydrogen bubble in the sodium, provided that sufficient water had reacted to exceed the hydrogen solubility. It operates on an electromagnetic induction principle.

A question was raised as to whether the equipment should be designed to withstand shock waves. There was considerable discussion relating to differences between pressure waves and shock waves. One opinion held that there can not be any shock waves because if shock waves are to be generated, the reaction front must travel faster than the velocity of sound in the reactants, and this was thought not to be possible. Another opinion held that liquid slugs can be accelerated by secondary reactions to high velocities and that such slugs can generate shock waves upon impact on structural members or other liquid columns. It was also pointed out that expansion shock waves are generated when a rupture disk breaks. At AI, the frequency of pressure oscillations in the reaction zone was observed to be 750 - 1000 cps, while the frequency 5 ft. above, and below, was 1500 - 2000 cps. The pressure traces were cusps, rather than sinusoidal. The frequencies were directly proportional to the water flow rates.

Various opinions were expressed on the question of whether the design should provide for dumping the sodium as well as the water upon detection of a leak. It would take a relatively long time to dump the large volume of sodium.

If no sodium isolation valves were provided, one would have to handle 100's of tons of sodium. It was pointed out that rupture disks installed below the sodium level will assure at least partial drainage. There is an advantage to retain the sodium for dissolving reaction products. It was suggested that the water side be pressurized with inert gas to keep sodium from causing secondary reactions on the water side. A bursting disc should be provided in the dump system to relieve pressure in case of a reaction. The PFR is designed for a 20 sec. water dump, no sodium dump, but isolation valves on the sodium side are provided.

Steam cleaning devices have been used successfully in the United Kingdom for the removal of reaction product "cement" (See Item 6). Steam lances with water injection dissolved and mechanically removed the cement from vessels. Attempts at removing cement from pipes have also been made, but it is probably easier to cut out the pipe. The work is done in air. A nitrogen blanket reduces the amount of fire, but it being a messy job either way, air is preferred.

Everyone indicated difficulties in purchasing diaphragms for under-sodium service with life time guarantees in excess of one year. How does one replace them?

A large pressure differential is set up in the sodium between the reaction zone and the discharge line, following a break of the rupture disk. This large pressure differential generates

extremely high sodium velocities. Tubes and supports must be designed to withstand cross flow velocities which may be generated in this manner. Forces would even be more severe if the flowing medium consisted of alternate gas flow and liquid slugs. Considerable damage to close-packed plates was reported by the United Kingdom and tube bending was observed by others.

The United Kingdom reported secondary tube damage due to overtemperature. Increase in tube diameter and thinning of the tube wall was noted. Metallographic examination confirmed that the mechanism was not erosion but rather a loss of strength of the tube due to overtemperature. There was no water flow through the tube to provide cooling. The test might, therefore, be considered representative of a steam generator leak at a time at which the feed water pump had failed, or when there was not water flow for other reasons.

10. Relief Systems

Presentations by: France - Lions
Germany - Mausbeck
United Kingdom - Watts
United States - Ford, Goodman

The relief system must be designed to minimize damage to the plant and to prevent any damage to the surrounding population. Protection against failure of the IHX is in Germany considered to be the primary function of the relief system, all others being secondary.

The relief system for the AI modular steam generator was described. Because of the high L/D ratios of the modules, rupture disks must be located under sodium. There is a disk at the center of each module and one in the header which connects the outlets of ten modules for each bank of steam generators. Relief system line sizes have been tentatively set at $1/3$ to $1/2$ the module diameters. The lines are routed to a containment tank or cyclone, with tangential inlets. A demister may be added. The stack is closed with a flapper valve. Spark plugs are located in the relief lines to determine which module failed, for shut-down purposes. The appropriate bank will be isolated first. Then the failed module can be isolated and the plant can be brought back to power.

In Germany, relief systems are provided with orifices (See item 9 and reference 15), to assure that neither pressure pulses nor quasi-constant pressures can reach the IHX. In the IHX, the primary sodium is on the shell side. The tubes are designed to withstand 35 atmospheres secondary sodium pressure before yielding.

The PFR has nine vessels, i.e., 3 sets of superheaters, evaporators and reheaters. Similar construction is employed for all units. Water flows inside the U tubes, sodium enters the vertical shells at the bottom and leaves through side nozzles near the top. Inert gas spaces are provided under the tube sheets. The design of the relief system had not been finalized. It was contemplated to locate one rupture disk

under the sodium near the top of each evaporator and rupture disks at the bottom of each superheater and reheater. The relief lines are joined to a common header into a sodium-hydrogen separator which is vented via a stack. A flap valve and flame arrester will be provided on top of the stack. An inert atmosphere purge is triggered after a leak. Since a 20 sec dump causes severe thermal shocks, they are working toward a design which will initiate a slow dump first upon an advanced warning signal.

A sodium-water reaction took place near the sodium surface in a Fermi steam generator and caused very little material transport into the relief system. This system has a 24 in. rupture disk set at 50-60 psig on each unit. Thirty inch lines connect each disk to internally vaned, centrifugal separators. Vent stacks are closed on top with rubber membranes. A nitrogen atmosphere is maintained in the whole system. The rupture disks have wires which interrupt an electric circuit upon a break and initiate automatic isolation and dump. The dump system has provisions for testing its operability without actually isolating and dumping, so that units will not get shocked each time the operability of the system is being checked. There are also parallel and back-up systems to provide complete redundancy for isolation and dumping. Further details are given in reference 22.

The secondary system of Phenix includes a separate buffer tank. This through-flow tank is connected to the sodium outlet of the steam generators. It is partially filled with inert gas. The tank may be eliminated in the final design if further calculations which account for the compressibility of sodium, show that it is not needed. Rupture disks are located at the inlets and outlets of each evaporator, superheater and reheater as well as on the buffer tank. Lines from the disks are headered into a single line into a hydrogen separator. A dump tank below the separator is also connected via valves to the secondary sodium system.

Some systems have relatively short lines from the rupture disks to a common relief header and a question was raised about the effects of a failure on the disks of the non-failed units. Reverse failures of disks have been experienced at AI in two instances. To prevent such ruptures, the disks used in the United Kingdom have vacuum supports which presumably are strong enough to hold the disk against any pressure which might develop in the common relief system. Reverse disks might be useful in this regard and are being tested in Germany. It was suggested that shock waves on the sodium side would rupture all disks anyway, however, estimates at AI indicate that disks of only adjacent modules would fail, i.e., in case of a leak in one module in a bank

of 10 modules only 4 out of 13 disks would fail. To eliminate the problem of rupture disks under sodium, it was suggested that they be installed on top of a stand-pipe in which a small volume of inert gas is trapped. However, this could introduce problems of level and sodium purity control and would increase the response time of the system.

The importance of coordinating relief system design with steam generator design was stressed. There is a question as to whether the steam generator manufacturer or the plant designer should have responsibility for the design of the relief system. In either case, close coordination between these two organizations is mandatory.

11. System Recovery

Presentations by: United Kingdom - Bray
United States - Ford, Goodman, Probert

Plant outage is very expensive. It is, therefore, of utmost importance to an operating utility that system recovery, following a sodium-water reaction, be accomplished correctly, safely, and in a minimum of time.

Written emergency procedures must be prepared in advance of any steam generator operation. System recovery procedures should also be prepared in advance, although it will not be possible to foresee and cover all possible circumstances.

Special tools and equipment, which are required for recovery operations, should be on hand - possibly having been supplied by the steam generator manufacturer.

There may be basic differences in recovery procedures for modular and large shell steam generators. For modular units, clean-up procedures may be confined to a few units if isolation valves are provided. Economic trade-off studies regarding optimum numbers of modules and valves are under way at AI. With a large shell unit, it will be more difficult to assess the extent of damage. Covers near the tops of the tubes can be taken off to test for leaks. If necessary, the tube bundle can be pulled for a visual inspection. Depending on damage, tubes could be plugged, groups of coils or complete tube bundles can be replaced. Because of cost, it would be desirable to have a procedure for handling minor damage without pulling the tube bundle.

Upon isolation and dump of the failed Fermi steam generator, the primary and secondary sodium pumps of the affected loops were stopped. On restart, it was found that the secondary sodium pump was frozen and it took several hundred ft-lb of torque to free it. The sodium was saturated with impurities. It took 22 days of cold trapping to remove the impurities to below saturation levels. This experience suggests the installation of cold traps in each secondary sodium loop with a capability of removing reaction products from the sodium in the system

and associated dump tank. Upon removal of the tube bundles, it was noted that the tube surfaces below the sodium level had been completely cleaned by the sodium, whereas massive deposits of reaction products were still retained on the tube surfaces in the cover gas space. these deposits were removed with alcohol.

There was considerable discussion concerning alcohol vs. steam-water cleaning. Of special concern is the production of sodium hydroxide which can cause stress corrosion. It was pointed out that hydroxide forms with alcohol or water. For the sodium-water reaction tests at AI, the system was flushed with water immediately following a test. The use of cold water caused thermal shocks which lead to surface cracks in pipe nozzles. Higher temperature water would be recommended. The problem of sodium entering the water side and causing corrosion was avoided at Fermi by pressurizing the water side with nitrogen.

The problems of removing massive solid reaction products had been discussed before. (See Items 6 and 9).

12. Future Tests

Presentations by: France - Lions
United Kingdom - Bray
United States - Goodman, Jones, Probert

Future tests are needed to provide data in the following areas:

1. What leak should the design be based on, i.e., leak size, shape, location, frequency?
2. What assumption should be made regarding a division of energy between thermal (heating) and pressure effects?
3. What is the spacial distribution of energy?
4. Better structural analyses methods for transients must be developed and verified experimentally.
5. Materials properties for high strain rates must be established.
6. Sophisticated calculation methods must be simplified to permit scoping of new designs for proposal purposes.
7. What are the criteria on which the design of relief systems should be based?
8. Data are needed for the design of fire fighting equipment and the development of fire fighting procedures for large sodium fires.
9. How can one estimate the corrosion and erosion effects of small leaks?

Better control over test variables is needed. Keeping the complexity of the test to a minimum would help. One should make sure that the instruments are capable of responding to the parameters to be measured. If the technology of instruments is not sufficiently advanced, the best available instruments should be used and their limitations should be considered in interpreting the test results.

A statement was made that for a modular steam generator, the effects during the first 100 msec. can perhaps be calculated with sufficient assurance for safety, whereas this may not be the case for large vessel designs. This led to the comment that a steam generator design should not be selected on the basis of whether or not it can be analyzed since one may end up in an economically non-competitive position. The use of conservative analytic models may be less expensive than the development of precise models. It was pointed out that a single tube-in-tube (KNK) design is representative of the simplest model but modules with more tubes approach the large vessel types and there is really no sharp dividing line.

The question of whether or not one can continue to operate with a known leak was discussed at length. Normal practice with fossil fueled boilers was cited as a reason for making plans for shut-down and repair as soon as a leak is detected. Experience with 2400 psi boilers is that a 1/16 to 1/8 inch hole will cause loss of water which can be detected with a level indicator in a feedwater storage tank. The leak will get larger and the limited inventory of water will force a shut-down within a day or two. Because of more sensitive leak detection methods, it is expected that a water leak in a sodium heated steam generator will be detected much earlier. It was pointed out that it may be difficult to distinguish between

a water leak and hydrogen in sodium from other sources. Experience at the SCTI was cited, where the maximum hydrogen level in the cover gas of the ALCO/BLH steam generator was originally limited to 400 ppm. This value was reached immediately after start-up. A new maximum level of 2000 ppm hydrogen was then specified and the actual level increased to 1800 ppm and then decreased again. The source of hydrogen was not attributed to a water leak.

13. Simulation Tests

Presentations by: Japan - Hori
 United States - Ford

Sodium-water reaction simulation tests, which have been performed with gas and water at APDA, were described and preliminary results were presented. The purpose of the tests was to verify the analytic model (mathematical representation and assumptions) for calculating pressures in the steam generator and to establish scaling factors.

The test section consisted of a water-filled, transparent vertical, cylindrical chamber with suitable gas injection equipment from the bottom. Pressures were measured with piezo-electric transducers during a rise time of 1/4 sec. and recorded with a visicorder oscilloscope having a 4000 cps. frequency response. A high speed motion picture camera was

used to observe and record the events in the test section at 4100 frames/sec. The gas supply reservoir had sufficient capacity to maintain gas pressure within 1% of the initial value for the duration of the tests (40 msec).

Three types of tests had been run when the program was terminated: valve characterization, optical distortion and time scale synchronization tests. Tests were run with helium and air at pressures of 40 to 100 psi. Water temperature ranged from 38 to 127°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 is suspected to be the cause.

The tests indicate that factors which had been omitted in the mathematical model, such as the inertia of the incoming gas, should be included to obtain better agreement with experimental results. The density of the gas appeared to have an effect. The calculated peak pressures for air were 32% higher while those for helium were only 5% higher than those observed during the tests.

Considerable mixing between gas and water was observed while the analytic model has assumed discrete gas and liquid regions. The bubbles contained up to 50% liquid by volume. The piston model was modified to account for slips between phases, but has not yet been used for comparisons with experimental results.

Some thought had been given in the United Kingdom to model the whole secondary system with non-reacting fluids. High temperature gasoline appeared to have the right properties for simulation, but was considered to be too hazardous for use.

Other non-reacting fluids such as glycerin and tetraethylene glycol have been considered at AI for simulation tests.

APPENDIX A

PARTICIPANTS IN IAEA SPECIALISTS' MEETING ON
SODIUM-WATER REACTIONS

Held at Argonne National Laboratory
November 5-6, 1968

IAEA

D. A. Yashin

FEDERAL REPUBLIC OF GERMANY

K. Gast, Karlsruhe
H. Mausbeck, Interatom

FRANCE

J. Leduc, CEA
N. Lions, CEA

JAPAN

K. Kurukawa, JAERI
M. Hori, JAERI

UNITED KINGDOM

J. A. Bray, UKAEA
M. J. Watts, UKAEA

UNITED STATES

J. Ford, APDA
K. Goldmann, UNC
L. Goodman, AI
R. Jones, USAEC
P. Probert, B&W

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APPENDIX C

REPRODUCED FROM THE PROCEEDINGS OF
THE INTERNATIONAL CONFERENCE ON SODIUM TECHNOLOGY AND
LARGE FAST REACTOR DESIGN

ANL-7520, PART 1

APPENDIX C

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

RALPH H. JONES, Moderator

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U. S. Atomic Energy Commission
Washington, D. C.*

Introduction

RALPH H. JONES

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 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 exposé 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 giving appraisals of the meeting and summary reports on the subjects covered.

Purpose of IAEA Specialists' Meeting on Sodium-Water Reactions

DMITRI A. YASHIN, Scientific Secretary

International Atomic Energy Agency

Let me briefly refer to the history which has led us to the Specialists' Meeting on Sodium-Water Reactions.

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.

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.

Appraisals of IAEA Specialists' Meeting by Participants

Ralph H. Jones, Meeting Chairman

It was gratifying to me to observe, at this IAEA Specialists' 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.

Reports on IAEA Specialists' Meeting

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Kazuo Furukawa, Japan

Although Japan has not yet done much development work on the sodium-heated steam generator except for a few fundamental researches concerning sodium-water reactions and water-leak detections, 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 knowledge in 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.

Kurt Goldmann, United States of America

It was a most valuable meeting.

Amongst safety problems in LMFBR 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 problem of sodium-water reaction 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 less by more than an order of magnitude than those spent on nuclear excursions.

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 than stainless steel.
8. Very few tests were made to determine the maximum pressure peak which a shell can withstand.

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

I. Statement of Problem

Ralph H. Jones, Specialists' Meeting Chairman

For acceptance in LMFBR plants a sodium-heated steam generator must be proven to be:

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

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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, and 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

Kazuo 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 a fundamental study conducted a few years ago. Following a literature survey, which was conducted

in 1959, their group carried on a small-scale study of chemical reaction between sodium and water in the absence of air, concluding that the reaction is controlled by the mixing rate of sodium and water.

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 many papers on the sodium-water reactions are 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 for the sodium-oxygen-hydrogen system, at a wide range of temperature and pressures, including solid and liquid regions as well as the gas region. 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, NaOH/Na, NaOH/Na₂O/Na, Na₂O/Na, Na₂O/Na₂O₂, 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 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 ms. 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 and APDA, in France by the CEA at Cadarache, in Germany by Interatom at Bensberg, and in the UK at Dounreay in Scotland. Nearly all the work was carried out with 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 there is inadequate control of water purity. 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 from $\frac{1}{16}$ to $\frac{3}{16}$ in. is involved, very rapid wastage rates have been observed for tube material in the path of the jet. Wastage rates of up to 3 mils/sec for 2¼ 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 in which 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 with 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 with orifice sizes up to $\frac{3}{4}$ in., the wastage effect has been completely absent in many tests. For example, in a series of 17 tests in a modular steam-generator mockup 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 3 sec 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 in.) 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 ft 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 with 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.

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 by hot water.

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

4. Leak Detection

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.

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.

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

5. Calculational 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 à l'Énergie 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 and sodium flow) are calculated to within 20% of the experimental values during the first 50 ms; 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.

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

On the other hand, Mr. Mori 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. 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 are 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 overpressures in any location of the loops and particularly in the intermediate heat exchangers.

6. Design of Steam Generators and Relief Systems

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 mixtures 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.

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 a relief system use rupture discs as a primary means of depressurization. The number and locations of the rupture discs vary, 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.

Participants in IAEA Specialists' Meeting on Sodium-Water Reactions

Held at Argonne National Laboratory
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Discussion

Mr. Isbin (Univ. of Minnesota): Did I understand, Dr. Goldmann, that you said that you were not able to determine what the safety criteria should be during these meeting days?

Mr. Goldmann: That's right.

Mr. Isbin: Can you add a few more remarks as to what the problems are?

Mr. Goldmann: To start with, we know that sodium and water do react rather violently with each other. The question arises, first of all, can you allow such a reaction to take place; I think the obvious answer is yes. You cannot possibly design and build a steam generator in which you can guarantee that there will not be such a reaction. You are then confronted with the problem of deciding to what extent must you contain a reaction in order to safeguard many things. Perhaps one of the very first things that has to be prevented, as was mentioned briefly by some of the other speakers, is the sodium-water reaction leading to a nuclear excursion in the primary side of the system. However, today there are

no statements in any of the designs of the steam generators or the secondary systems that say there must be a device that will assure that if there is reaction this reaction will not lead to a nuclear excursion. This is the most severe sort of thing that you certainly will have to guarantee some day.

Then there are other considerations, like what size of leak must you guard against. What is the probability for a single tube to leak or a whole bunch of tubes to leak all at the same time? What are the consequences to the surroundings in terms of how far you can get the reaction products spewed out into the atmosphere or the surrounding area? Such criteria have not been set, and I think it is very important to get working on this, to set them, and to realistically estimate what the cost is going to be to meet such criteria.

Mr. Eichelberger (AI): Dr. Goldmann mentioned detecting these leaks by measuring the hydrogen in the gas phase or in the sodium. Can you give us an idea of your detection levels in these media?

Mr. Goldmann: It appears that the normal hydrogen levels observed in cover-gas spaces of reactors are of the order of hundreds, possibly up to 2000 ppm. The problem is to identify a sudden change in hydrogen level due to a leak. The chances are that you will see a leak by a rate of change of hydrogen level rather than just a change of an absolute level. That was one of the perhaps significant experiences at APDA. In the sodium itself, the hydrogen levels are quite low, and I am really not prepared to give a number.

Mr. Jansson (W): I would like to know if the panel discussed the role of stress corrosion by hydroxide in the reactor system. Stainless steels are not exactly known to be very resistant to this. Especially at the point of a leak, you would expect a very high hydroxide concentration, and I would guess that you could get attack in the sodium loop as well as on the water side. The second question is to Dr. Bray. Is the reaction product with the melting point of 300-350°C sodium hydroxide, or does it also contain something else?

Mr. Bray: Perhaps I can answer the second question first. Yes, it is largely sodium hydroxide. It seems to be a mixture of sodium hydroxide, sodium oxide, and sodium hydride in varying proportions, according to the temperatures and the type of reaction. Stress corrosion of stainless steel was lightly touched upon in the meeting. I think that on the sodium side small amounts of sodium hydroxide will be removed by the conventional cleanup system provided in any sodium circuit. Caustic stress corrosion could arise during reclamation procedures, decontami-

nation, repair, and rebuilding. I think this is just a question of how much repair can be done to a damaged tube bundle, and how stringent can recovery procedures be made. Also, post-recovery inspection criteria have to be specified for a particular case.

Mr. Jones: I would like to ask that we amplify on that, since at the meeting we did cover another aspect of stress corrosion. John Ford, from APDA, could you say a few words about nitrite-nitrate corrosion?

Mr. Ford (APDA): What Mr. Jones is referring to is the corrosion on the inside of the steam-generator tubes, which was experienced at the Fermi plant due to improper cleaning procedures. This was stress corrosion of ferritic materials, 2.25 Cr-1 Mo, by a cleaning solution which was composed, among other things, of nitrites. The nitrites when exposed to air reacted to form nitrates; this, in combination with high temperature and some non-stress-relieved portions of the tubes, caused stress-corrosion cracking.

Mr. Jansson: May I ask if you do not think that stress corrosion could be a mechanism whereby a leak could grow and become more serious?

Mr. Bray: This question of leak growth is very difficult. Although it is purely a supposition, I wouldn't have thought that stress corrosion would be likely to affect leak growth because the sort of leak growth that we have seen is very rapid indeed. Very rapid wastage is associated with this kind of corrosion-erosion behavior. I would have thought that stress-corrosion attack of this type was more likely to be a long-term feature. This isn't to say that caustic stress corrosion in stainless steel components following decontamination couldn't be a problem if, for example, full washing procedures weren't followed, or if washing procedures weren't successful. I don't think this is a very real problem, for it is concerned with direct propagation in stainless steel tubes in operating heat exchangers with the tubes immersed in sodium anyway.

Mr. Isbin: With reference to the discussion of pressures, I believe by panelist Bray, the indication was that pressures up to 3000 psi were measured. This was discounted in part of your discussion in that you didn't find any damage. I am not sure whether you were implying, and perhaps the other panelists might want to amplify, are these pressures real, do you think? Are there measurement difficulties? Do you have a mechanism for generating such high pressures?

Mr. Bray: This puts me on a spot, as perhaps the weakest part of our experimental work has been in pressure measurement. With regard to pressure damage, we haven't seen very much. Pressure measurements have been made by Mr. Lions and by the Atomic International people. Mr. Goodman looks eager to answer this question. I wonder, would you like to say a word on this?

Mr. Goodman (AI): As far as our data are concerned, we have reason to believe the reality of the pressures. We have had ways of verifying the validity of the tapes so that we can be assured that the pressures were real. So far as not sustaining any damage is concerned, I can't speak with certainty about the other test programs, but in our test program we have designed a system to sustain pressures well in excess of 3000 psi. It came as no surprise that we saw no readily obvious accumulated damage after a series of tests. We were using schedule 80 pipe, and these tests only lasted for very short periods of time.

Mr. Lions: We measured peak pressures of about 100 bars in a shell of 10-cm diameter, 4 m in length, and 5 mm in thickness. After five runs, the shell was deformed but had not failed. That shell was of Type 316 stainless steel.

Mr. Probert (B&W): To make failure analyses on the basis of peak pressure is invalid. These analyses should be made on the basis of impulse-momentum calculations or work-energy type of calculations.

Mr. Mausbeck: We did some experiments in tube geometry. In this case we measured more than a thousand atmospheres but over a duration of less than 1 ms; the peaks lasted about 1/4 ms, so the energy stored in the peaks was small.

Mr. Latzko (Delft): Dr. Mausbeck mentioned that in all modular designs the rupture discs were mounted under sodium. Is that correct?

Mr. Mausbeck: The discs were submerged under sodium.

Mr. Latzko: Well, I would like to know what materials were used, what was the maximum number of hours obtained with the rupture discs under sodium, and how were they mounted?

Mr. Mausbeck: It wasn't really much discussed during the meeting. I can only add our experience, that we have had them under sodium for, I think, about 6000 hr now.

Mr. Bray: I would like to add that the material used for this sodium work was nickel.