C.1. Sodium-Water Reaction Studies for MONJU Steam Generators

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ABSTRACT

The R & D results of the PNC's sodium-water reaction project are reviewed. The purposes of the project with the specific object for each test rig and computer code are given. The test items which should be investigated for the safety evaluation of the MONJU steam generators are discussed, and the status of the PNC's work on each item is described.

The results on the small-leak wastage measurement are shown and the improved experimental equations to predict the wastage rate from the leak rate and the sodium temperature are given. The preliminary results on the wastage of tube bundle in the intermediate leak range are shown. The depth and the area of the wastage and also the wastage rate for each tube are shown graphically.

The measured peak value of the initial pressure spike for the large leak is shown. The scatter of the data and its causes are discussed. The bubble growth rate estimated from the void probe measurement is presented. The results of the simulation experiment on the pressure wave propagation to the secondary circuit are given, comparing them with the prediction by the one-dimensional computer codes SWAC-5K and SWAC-5H.

# 1. INTRODUCTION

## 1.1 The PNC Sodium-Water Reaction Project

The sodium-water reaction project (SWAT Project) started from 1969 as one of the projects in the sodium-heated steam generator development program of Power Reactor and Nuclear Fuel Development Corporation.

Purpose of SWAT project is to study the sodium-water reactions relative to the development of safe, reliable, and economic sodium-heated steam generators for the prototype fast breeder reactor "Monju".

This project includes the development of :

(1) Data needed to estimate the extent of pressure pulse generated in the steam generator and to determine subsequent mechanical effect on the steam generator structure and related system during a large leak of water into sodium.

(2) Data needed to estimate the extent of tube wastage in the steam generator during a leak of water into sodium.

(3) Instrument systems needed to detect a small leak of water into sodium, of which response should be sensitive, fast and reliable enough to prevent the leak propagation to other tubes.

(4) Computer code needed to predict the effect of a large leak of water into sodium to the integrity of the steam generator and related systems.

The PNC's test rigs for the sodium-water reaction studies are summarized in the Appendix 1.

The PNC's computer codes for the sodium-water reaction analysis are summarized in the Appendix 2.

## 1.2 Safety Evaluation of MONJU Steam Generator and Necessary Test Items

During the each stage of MONJU plant design, the safety evaluation of the steam generators and its related system has been conducted using the most reasonable models and values estimated at each stage. These models and values are to be verified and updated by the sodium-water reaction tests and analyses which have been conducted in parallel with the plant design.

In the design and the safety evaluation of the MONJU plant, it is assumed that the maximum sodium-water reaction accident is the one caused by the water leak from the double-ended ruptures of four tubes, where the initial failure is assumed to be less than one tube rupture and the secondary failure to be less than three tube ruptures.

Even in the maximum accident, the boundary wall between the primary and secondary circuit in the intermediate heat exchanger should withstand the pressure propagated from the steam generator, and also the integrity of the steam generator shell and the secondary circuit components should be kept.

For the safety evaluation of the MONJU steam generators, the following items are to be demonstrated and confirmed by the sodium-water reaction tests and analyses:

(1) The confirmation of the maximum scale of the accident

It is necessary to demonstrate that the maximum. accident caused by the initial failure of less than one tube rupture is less than four tube ruptures.

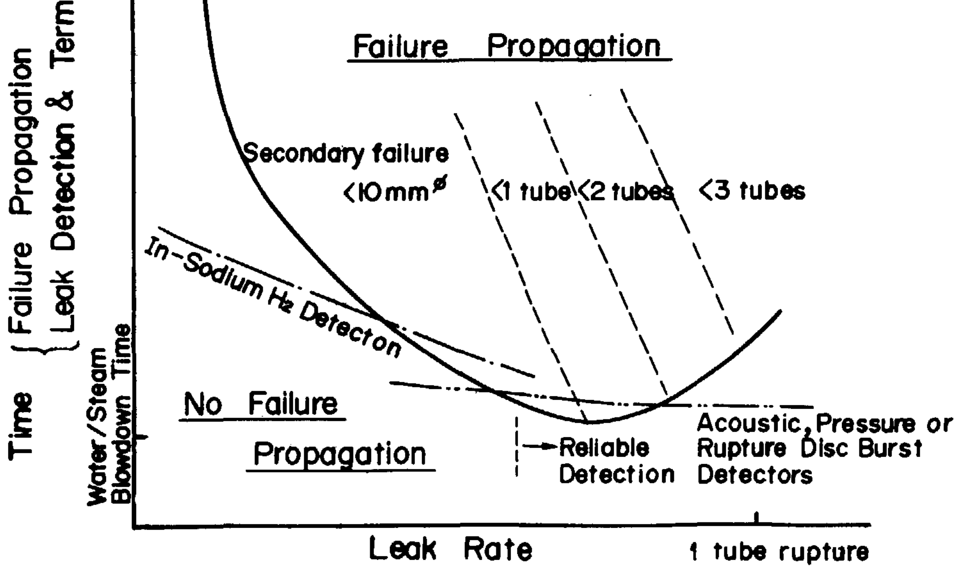
For the above purpose, it is needed to obtain the data on the following items.

(i) The time of the failure propagation and the extent of the secondary failure for the initial failure of less than one tube rupture.

(ii) The time of the leak detection and termination for the initial failure of less than one tube rupture.

From the above data, the maximum scale of the accident could be estimated as shown in Fig. 1.

Fig. 1 ESTIMATED RELATION BETWEEN THE LEAK RATE, AND THE TIME AND THE EXTENT OF THE SECONDARY FAILURE AND THE TIME OF THE LEAK DETECTION/TERMINATION. (SCHEMATIC)




(2) The confirmation of the effect of the maximum accident

It is necessary to demonstrate that the effect of the maximum accident (four tube ruptures) on the steam generators, the intermediate heat exchanger and other secondary circuit components is within the tolerable limit.

For the above purpose, it is needed to obtain the data on the following items.

(i) The sodium-water reaction pressure and temperature in the steam generator for the leak rate corresponding to the one plus three tube ruptures. Specifically, the initial pressure spike, strain of the steam generator shell, quasi-static pressure, performance of pressure relief system, maximum temperature of steam generator shell etc.

(ii) The propagation of pressure to the secondary circuit and the intermediate heat exchanger for the leak rate corresponding to the one plus three tube ruptures.

1.3 Status of PNC's Sodium-Water Reaction Tests and Analyses

The sodium-water reaction tests by the SWAT rigs and the analyses by the SWAC codes are progressing. Followings are the summaries of the present status.

(1)-(i) The time of the failure propagation and the extent of the secondary failure.

a) For D 0.7 mm (D = Diameter of leak hole)

The quantitative results have been obtained by the SWAT-2 tests.

b) For D 7 mm

The preliminary results have been obtained by the SWAT-1 tests.

c) For D < One tube rupture

The tests are scheduled to be made by SWAT-3.

(1)-(ii) The time of the leak detection and termination.

a) Leak detection

The sensitivity and the response time have been measured using the SWAT-2 rig for the in-sodium hydrogen meter, in-argon hydrogen meter, continuous plugging meter and acoustic sensors.

b) Leak termination

The SWAC-11 code is being developed to analyze the blow-down phenomena for predicting the change of leak rate and the time of leak termination.

(2)-(i) The pressure and temperature in the steam generator.

The quantitative results have been obtained by the SWAT-1 tests. Large scale tests are scheduled to be made using SWAT-3.

(2)-(ii) The propagation of pressure to the secondary circuit.

The computer codes (SWAC-5K and SWAC-5H) were developed, and were compared and verified by the SWAT-1B tests.

## 2. WASTAGE IN THE SMALL AND INTERMEDIATE LEAKS

### 2.1 Wastage Results in the Small Leak[1]

Small leak wastage tests with steam injection into sodium have been performed using the SWAT-2 test rig.

Figure 2 shows the target tube assembly which simulates the heat transfer tube bundle of the MONJU steam generators. For the tests with injection direction "A", the target tube stack was pulled up after each steam injection to conduct the wastage tests successively. For the test with injection directions "B" and "C", both target and dummy tubes were placed horizontally. In the tube penetration tests, the target tube was pressurized by 100 atg argon gas.

The following test conditions were kept constant throughout all of the wastage tests:

| | |
|---|---|
| Sodium velocity past target, | 0.24 m/sec |
| Spacing between nozzle and target, | 17.5 mm |
| Target tube diameter, | 26.5 mm |
| Head of sodium above injection point, | 780 mm |

The objective of tests, steam injection directions and number of tests conducted are summarized in Table 1. Although the direction of steam injection is paraller to the sodium flow in Test Series I and perpendicular to the sodium flow in Test Series II, there were no appreciable differences between the data observed for the two cases.

Table 1 Summary of Test Series

| Test Series | Objective | Injection Direction | Number of Test |
|---|---|---|---|
| I | Wastage Rate | Vertical B | 2 |
| | | Tangential C | 2 |
| II | Wastage Rate | Vertical A | 32 |
| III | Penetration | Vertical B | 3 |
| | | Tangential C | 1 |

A, B, C : see Figure 2

The tests were conducted over an injection steam pressure range of 20 ~ 166 atg. The injection pressure had no significant effect on the wastage rate if the data were correlated by the leak rate. The steam quantity ranged from 7 to 120 g and the injection durations from 5 to 100 sec. The leak rate were from 0.1 to 6 g/sec.

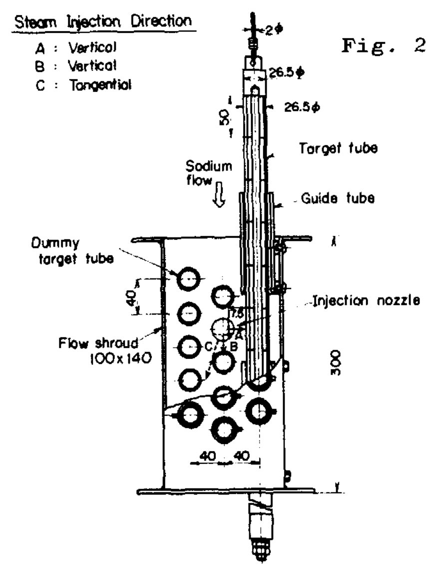



Fig. 2 CROSS-SECTION OF TARGET TUBE ASSEMBLY

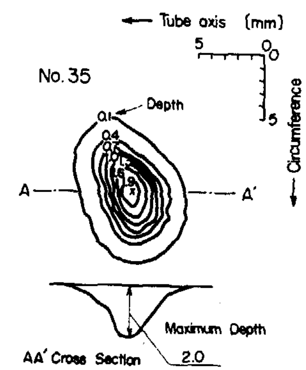



Fig. 3 TYPICAL CONTOUR MAP OF WASTAGE

Figure 3 shows a typical contour map of wastage measured on the target tube. Most of the data showed the symmetric maps like this example.

Figure 4 shows the effects of steam leak rates on the wastage rates with the parameters of sodium temperature and target material. The temperatures of sodium and steam were regulated to be almost equal and they were varied over a range of 270 to 540°C. As can be seen from the figure the wastage rate increases with the leak rate in the range tested but has a tendency to indicate a certain peak value.

The wastage rates by the tangential injection were lower than those for the vertical injection.

In each tube penetration test (Test Series III), the wastage rate which is determined from the thickness of tube wall and the time of penetration is smaller than the estimated value from the data of Test Series I and II. This result may indicate that the larger the depth of wastage, the harder it becomes for the target wall to be wasted.

The improved experimental equations are derived, using the least square method, to give the relationship between the wastage rate, and the leak rate and sodium temperature.

For $2\frac{1}{4}$Cr-1Mo steel

$$W_L = 463 \exp\left[ - \left\{ 0.135 \left(\ell n \frac{G}{34}\right)^2 + 5330/T \right\} \right]$$

For stainless steel

$$W_L = 8470 \exp\left[ - \left\{ 0.0602 \left(\ell n \frac{G}{1240}\right)^2 + 7520/T \right\} \right]$$

where $W_L$ : wastage rate [mm/sec]
G : leak rate [g/sec]
T : sodium temperature [°K.]

Figure 5 shows the comparison between the wastage rates measured and the ones calculated using the above equations.



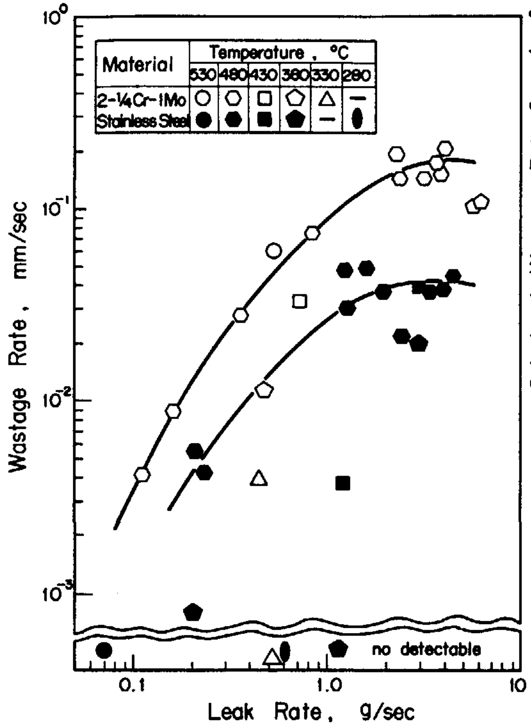



Fig. 4

EFFECT OF LEAK RATE, MATERIAL AND SODIUM TEMPERATURE ON WASTAGE RATE

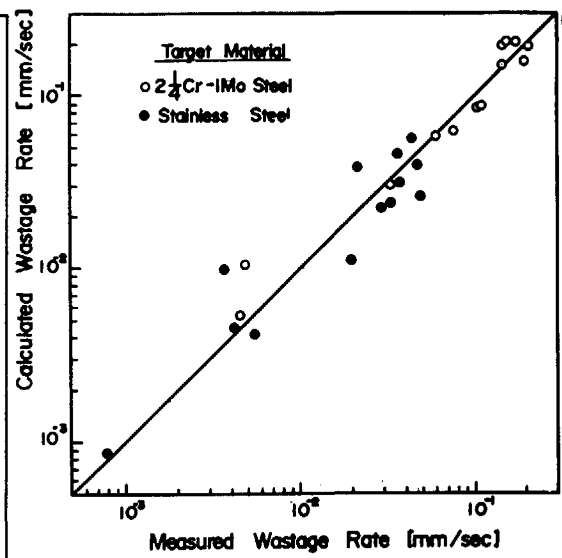



Fig. 5

COMPARISON OF CALCULATED AND MEASURED WASTAGE RATE

2.2 Wastage Results in the Intermediate Leak[2]

Wastage phenomena of a Cr-Mo steel tube bundle in the intermediate leak range with leak hole diameter of 1.0 to 2.5 mm were observed using SWAT-1 test rig. The wastage depth distribution in the tube bundle was measured, and the wastage rates at various position were obtained.

The target tube assembly consists of seven tubes with a bank pitch of 40 mm, which is similar to the MONJU steam generator tube arrangement. The tubes are of 27 mm O.D., 3.8 mm thickness, 120 mm long, $2\frac{1}{4}$Cr-1Mo steel. The tests were conducted with the leak hole diameters of 1.0, 1.77 and 2.5 mm .

The injected water was 125 kg/cm$^2$G in pressure and 325°C in temperature. The injection duration was 203 sec

(1.0 mm ), 71.5 sec (1.77 mm ) and 47.5 sec (2.5 mm ). The estimated leak rates were 20, 52 and 128 g/sec.

The tests were conducted in the stagnant sodium with temperature of 330°C.

The test results show that all the tubes in the assembly had some extent of wastage in the three tests conducted. Figure 6 shows the example of the wastage distribution for the case of 2.5 mm leak hole. The tube thickness is doubled in Fig. 6 for the convenience of illustration.

In all cases, the tube which was directly hit by the water jet had the highest wastage rate. The maximum wastage rates measured were $3.5 \times 10^{-2}$ mm/sec for the case of 2.5 mm hole.

The penetration of the tube wall by the wastage was occurred in the case of 1.0 mm leak hole. The directly hit tube had a penetration hole of 4.8 mmϕ.

Comparison of these wastage results with those of APDA's small leak test(3) is shown in Fig. 7.

The results of temperature measurement show that high temperature of more than 800°C had continued where the tubes were severely wasted.

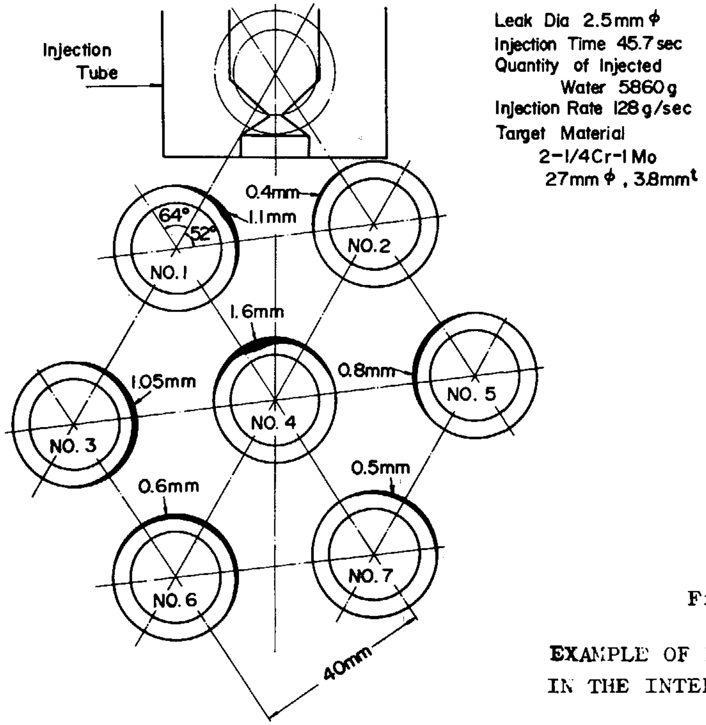



Fig. 6

EXAMPLE OF MULTIPLE WASTAGE IN THE INTERMEDIATE LEAK

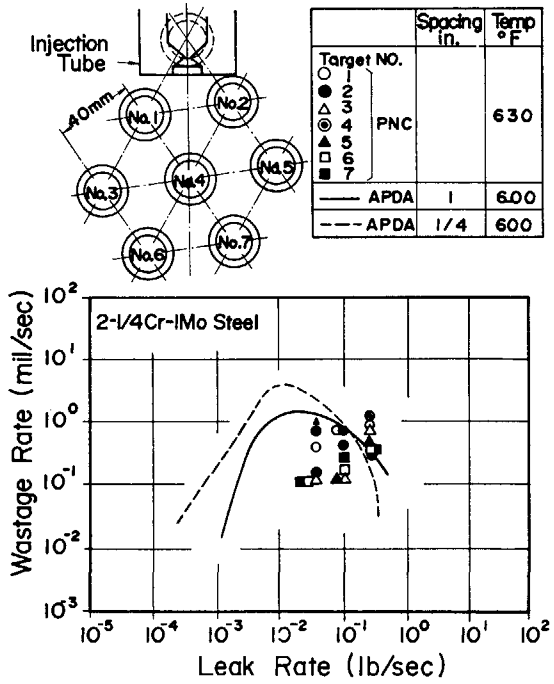



Fig. 7 COMPARISON OF MULTIPLE WASTAGE DATA WITH APDA DATA

## 3. REACTION PRESSURE AND ITS PROPAGATION

### 3.1 Initial Pressure Spike

The pressure generated by large leak have been investigated using the SWAT-1 test rig(4) (Fig. 8).

Just after the water injection, the initial pressure spike was usually observed, of which width was about 1 msec or larger. The peak values of initial pressure spiked are correlated with the injection rates as shown in Fig. 9. The scatter of data is probably caused by both the difficulty of controlling the initial injection condition and the difficulty in the pressure measurement technique in high temperature. The peak values shown in Fig. 9 are the pressure measured at the dead-end of the pressure taps (pressure guide tubes) of 216 mm long. The calculation of the pressure at the wall from the tap-end pressure can be performed numerically[5] or graphically[6] as shown in Fig. 10 (b). This calculation method was verified by the SWAT-1A simulation experiment using gun-powder explosion in water, as shown in Fig. 10 (a). The peak pressure at the wall is about 80 % of the peak pressure at the tap end for the SWAT-1 test cases. It was also found in the SWAT-1A experiment that a very small amount of gas in the pressure tap affect the measured pressure, for instance about 0.15 $cm^3$ of gas made the peak pressure at the tap end 3.5 times of that at the wall.[6]

Strain gauges were attached to the outside wall of the reaction vessel. The equivalent internal pressure could be calculated from the strain measurement using the equation of the thin-wall cylinder strained under a steady internal pressure. The equivalent internal pressures corresponding to the peak strain values are from 2 to 10 $kg/cm^2$ in the SWAT-1 tests.

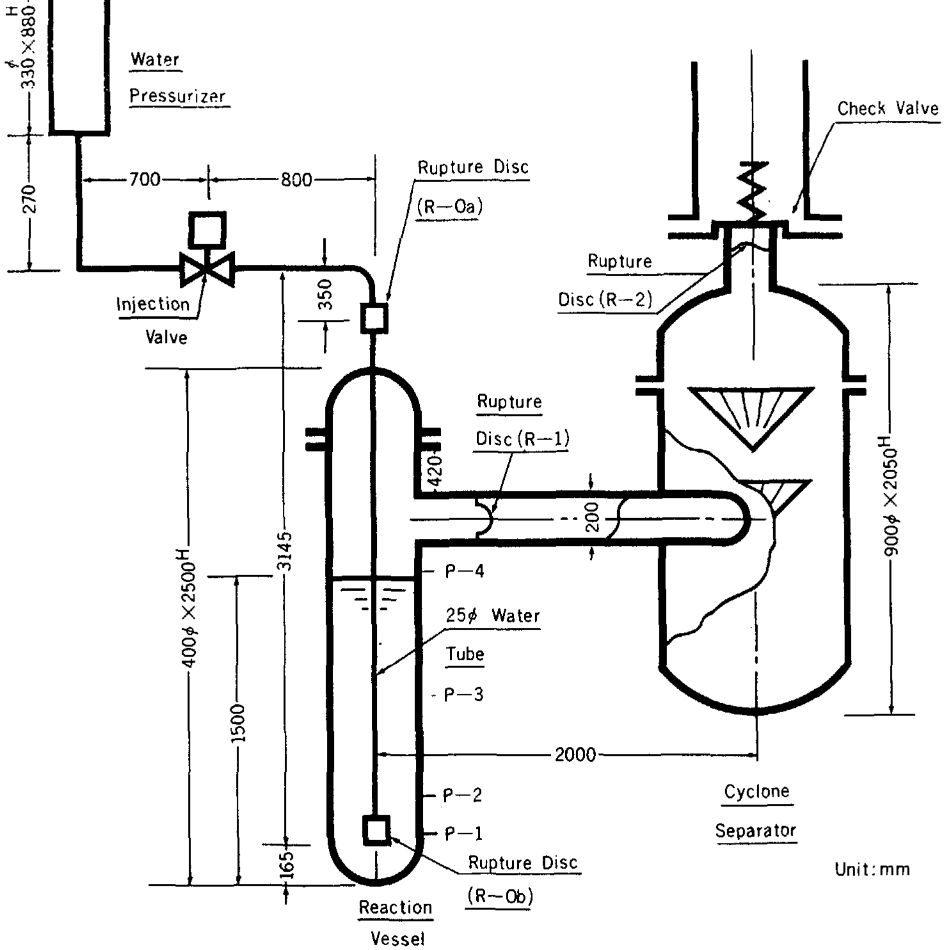
It is necessary to improve the pressure measuring technique and also to analyze the pressure generating mechanism at the leakage inception, before concluding on the effect of the initial pressure spike on the steam generator structure.

Some insight into the process of pressure generation could be obtained from the initial bubble growth data shown in Fig. 11. These data were obtained in the SWAT-1 tests using the resistance type void sensing probes around the injection point. The bubble shape is assumed to be a right cylinder, and the equivalent radius of the sphere having the same volume as the cylinder is calculated and shown in Fig. 11. It should be noted that the diameter of leak hole or the leak rate does not affect the bubble volume. The time of the peak pressure reached is usually about 1 msec, and the radius at this time is 3 ~ 4 cm.

### 3.2 Pressure-Wave Propagation in Secondary Circuit[7]

To investigate the validity of the codes, a series of tests were performed using the 1/12.5 scaled down model of MONJU secondary loop, which consists of steam generator,

Fig. 8 SCHEMATIC DIAGRAM OF SWAT-1 TEST RIG




reheater, intermediate heat exchanger, overflow column, and valves. Since water is easy to handle, it was used as the propagation media instead of sodium. And the pressure of around 2 to 7 atm. was inflicted on the free surface of the steam generator as an input.

Figure 12 shows the experimental loop, simulating the MONJU secondary loop system. The loop and each component were designed with reference to the primary conceptual design of MONJU. The dimension of the model was about 1/12.5 of Monju plant. The components were arranged so that the flow diagram of the experiment was the same as the actual plant. The pressure measuring points are marked by PG-NO. in Fig. 12.

In Fig. 13, the intermediate heat exchanger model is compared with its full scale design. In this model the

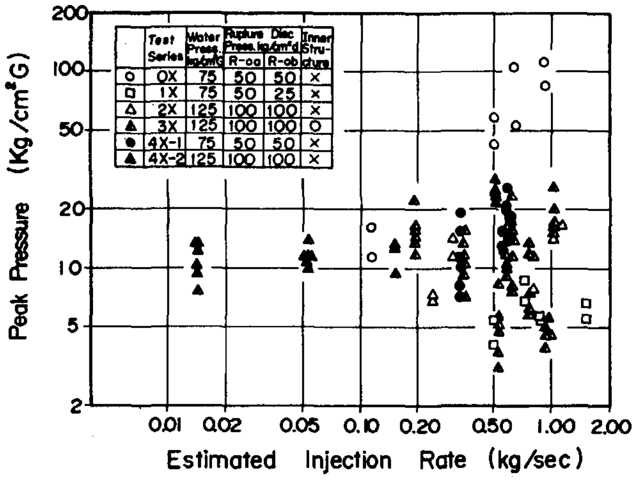



Fig. 9 PEAK VALUE OF INITIAL PRESSURE SPIKE (SWAT-1)

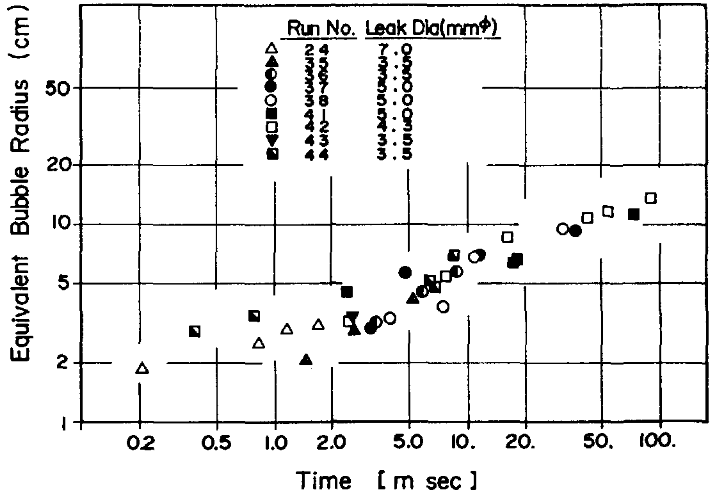



Fig. 11 INITIAL BUBBLE GROWTH OF SODIUM-WATER REACTION (SWAT-1)

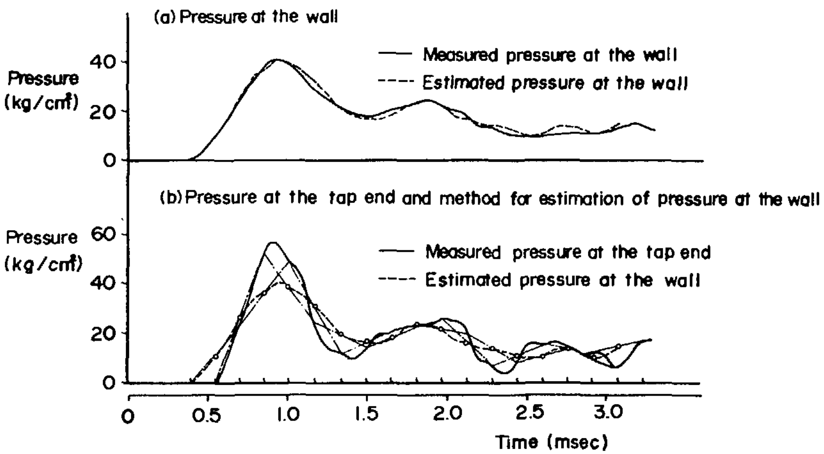



Fig. 10 COMPARISON OF PRESSURE AT TAP-END AND AT WALL, AND METHOD OF ESTIMATION OF PRESSURE AT WALL (SWAT-1A)(6)

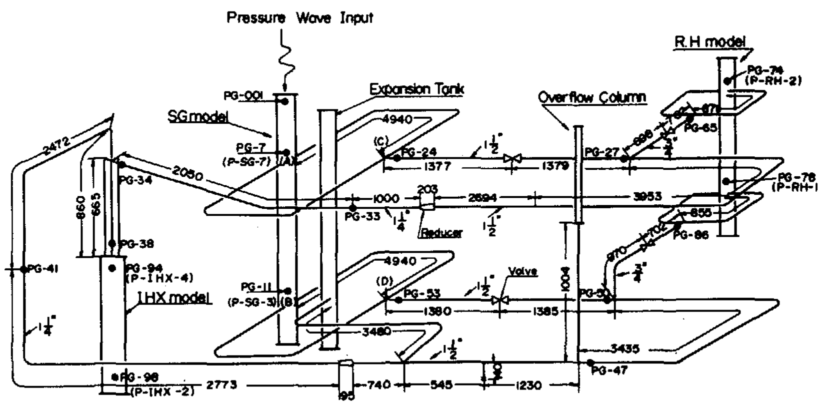
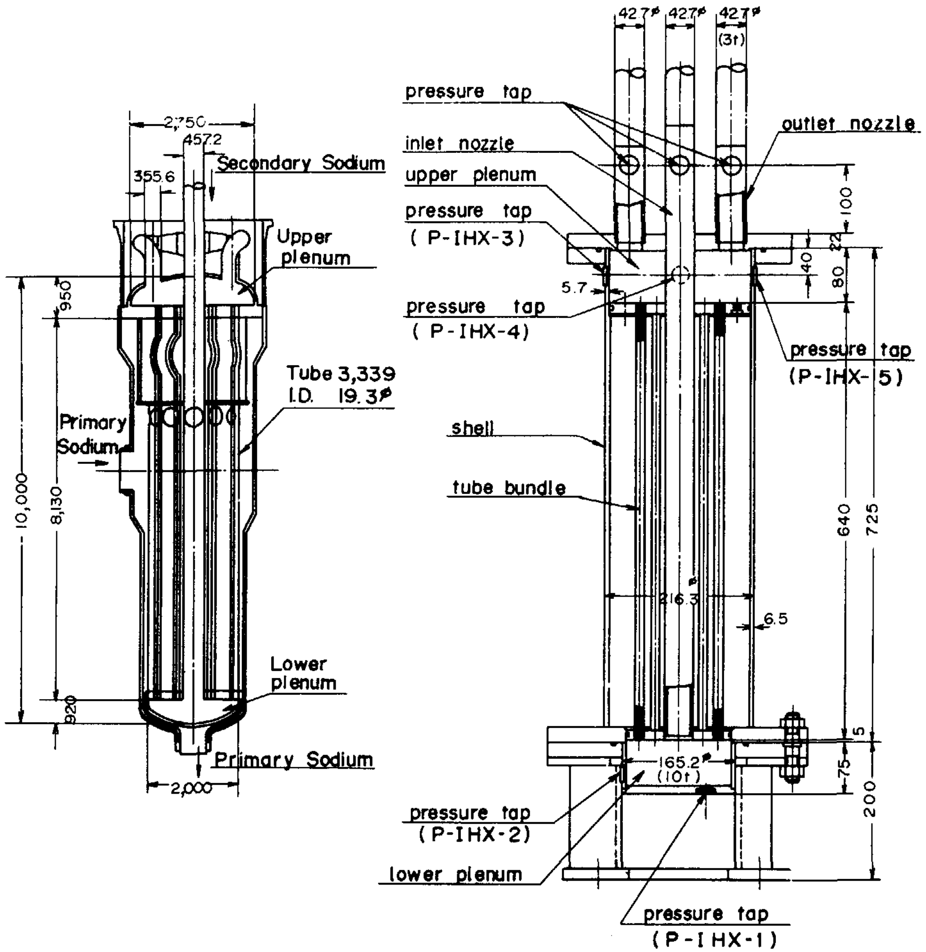



Fig. 12 TEST SECTION OF THE SIMULATED SECONDARY LOOP OF MONJU




(a) MONJU I.H.X. (Primary design)   (b) I.H.X. scale down model

Fig. 13 MONJU IHX AND ITS SCALED DOWN MODEL

secondary sodium flow path was faithfully simulated.

The comparison of the pressure at major points in the loop is also shown in Fig. 14. Here, the attenuation of the pressure and the propagation time are easily understood. In the main piping the attenuation of the pressure is not much, but as shown in Fig. 14 in the IHX the pressure decreases to 1/2 ~ 1/4 of the SG pressure, and in the reheater the pressure decreases to 1/50 ~ 1/100 of the SG pressure.

Two computer codes SWAC-5K(8) and SWAC-5H(9) were developed to analyse the pressure wave propagation in the secondary circuit of the fast reactor.

The SWAC-5K uses the method of characteristics and the SWAC-5H uses the wave superposition method. In the both codes, the one-dimensional model and the constant wave velocity were assumed.

The two codes were compared each other and with the experimental data, but no significant difference was found between the two.

To investigate the validity of the codes, the measured pressure at each point of the loop were compared with the calculated pressure which uses the pressure at the inlet and outlet nozzle of the steam generator as the input.

In Fig. 15, the pressure change at the branch of the outlet piping of reheater is shown. The calculated results agree well with the experimental results. The lines a-a, b-b, and c-c, shows the time that the reflected wave return from the reheater, the overflow column and the expansion tank, respectively. Even after the reflected waves arrived from these components, the both results agreed well. This demonstrates the modeling is suitable for the purpose.

Data obtained in the upper plenum of the IHX were compared with the calculated ones in Fig. 16. In the IHX the calculated results are always higher than the experimental results.

The difference between them can be attributed to the fact that in IHX the flow path is divided into many small tubes but the codes treat the path one-dimensionally in a single channel.

The results using the one-dimensional model give a higher pressure than the experimental results and this fact is suitable for the designing purpose and for the safety analysis.

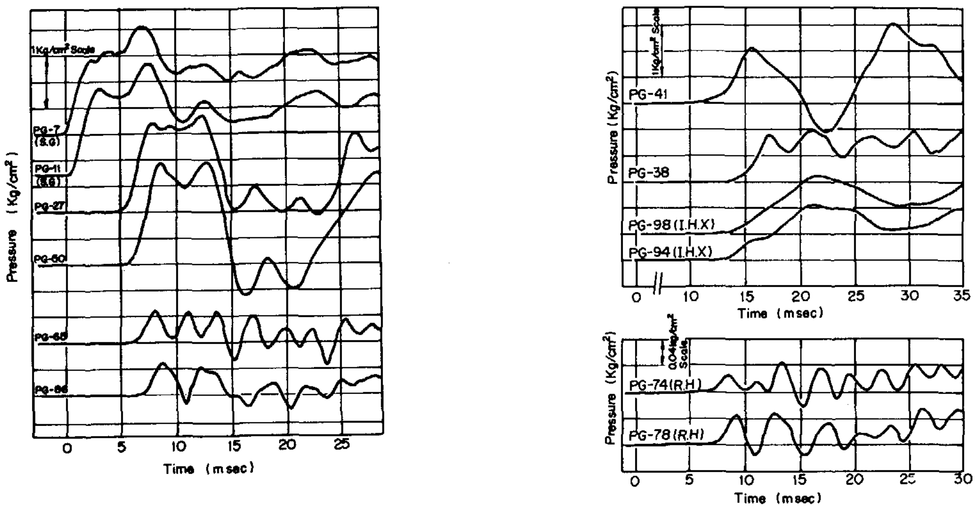



Fig. 14 THE COMPARISON OF THE PRESSURE VARIATION AT THE MAJOR POINTS OF THE LOOP

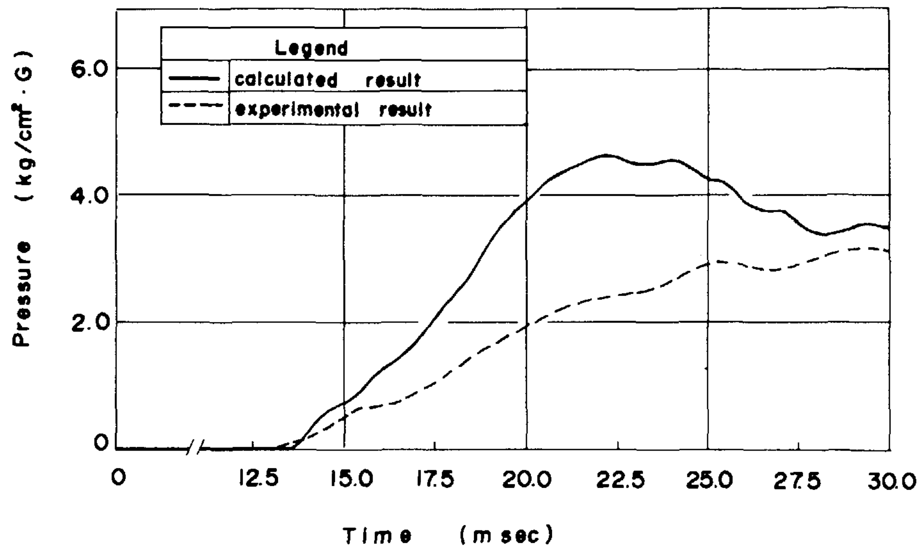



Fig. 16 COMPARISON OF THE EXPERIMENTAL AND THE CALCULATED RESULT AT PG-94, UPPER PLENUM OF IHX

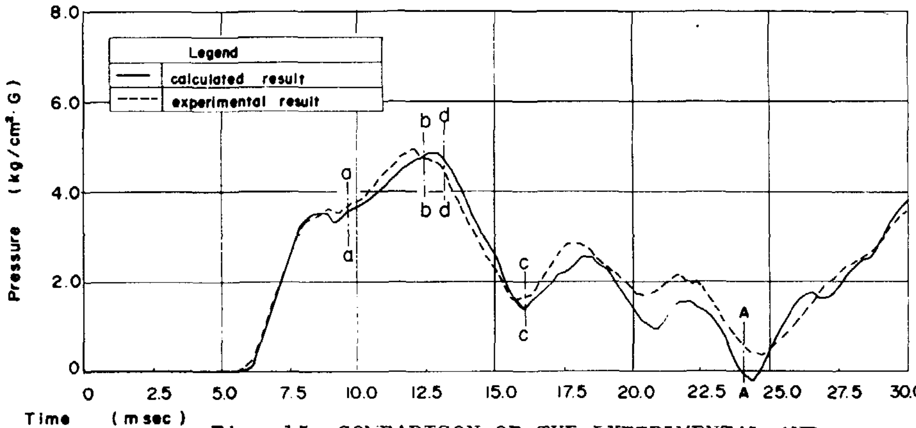



Fig. 15 COMPARISON OF THE EXPERIMENTAL AND THE CALCULATED RESULT AT PG-50, BRANCH POINT OF RH

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APPENDIX 1

PNC's Test Rigs for Sodium-Water Reaction Study

1. Large Leak Sodium-Water Reaction Test Rig (SWAT-1)

| | | |
|---|---|---|
| Purpose | : | Large leak sodium-water reaction study for the 'Monju' steam generator |
| Scale | : | About 1/8 scale model of 'Monju' SG |
| Status | : | In operation since Nov., 1970 |



2. Pressure Behavior Simulation Test Rig (SWAT-1A)

| | | |
|---|---|---|
| Purpose | : | Study of pressure behavior inside the vessel |
| Scale | : | About 1/8 scale model of 'Monju' SG. Using water as the fluid and gun-powder explosion as the pressure source. |
| Status | : | Test completed (September, 1973 ~ May, 1974). In co-operation with Central Research Institute of Electric Power Industries. |

3. Pressure Wave Transmission Test Loop (SWAT-1B)

| | | |
|---|---|---|
| Purpose | : | Study of pressure wave transmission in the 'Monju' secondary circuit |
| Scale | : | About 1/12.5 scale model of 'Monju' SG, RH, IHX, pipings and other components. Using water in stead of sodium. |
| Status | : | Test completed (August, 1971 ~ September, 1972) |

4. Small Leak Sodium Water Reaction Test Loop (SWAT-2)

| | | |
|---|---|---|
| Purpose | : | Small Leak sodium-water reaction study —— measurement of wastage and development of leak detector —— for the 'Monju' SG system. |
| Scale | : | Target-tube assembly consist of about 15 actual diameter tube of 'Monju' SG |
| Status | : | In operation since April, 1972 |

5. Steam Generator Safety Test Facility (SWAT-3)

| | | |
|---|---|---|
| Purpose | : | Over-all safety test of 'Monju' SG and secondary circuit for possible sodium-water reaction |
| Scale | : | About 1/2.5 scale model of 'Monju' SG |
| Status | : | In construction (To start operation from March, 1975) |

APPENDIX 2



PNC's Computor Codes for Sodium-Water Reaction Analysis

1. One-Dimensional Sodium-Water Reaction Analysis Code (SWAC-1)

| | | |
|---|---|---|
| Model | : | One dimensional piston-model. Compressibility of Water and Sodium considered. |
| Developed by | : | Kawasaki Heavy Industries, Ltd. |
| Status | : | Completed |

2. 'Monju' Secondary System Leak Analysis Code (SWAC-2)

| | | |
|---|---|---|
| Model | : | Hydrogen transport through the circuit in piston-flow (Modification of the APDA code reported in APDA-255). |
| Developed by | : | Central Research Institute of Electric Power Industry and PNC |
| Status | : | Completed |

3. Three-Dimensional Sodium-Water Reaction Analysis Code (SWAC-3)

| | | |
|---|---|---|
| Model | : | Three dimensional non-mixing model. |
| Developed by | : | Mitsubishi Atomic Power Industries, Inc. |
| Status | : | Completed |

4. Pressure Wave Transmission Analysis Code (SWAC-5K)

| | | |
|---|---|---|
| Model | : | One dimensional with branches. Method of characteristics used for fluid compressibility effect. |
| Developed by | : | Kawasaki Heavy Industries, Ltd. |
| Status | : | Completed |

5. Pressure Wave Transmission Analysis Code (SWAC-5H)

| | | |
|---|---|---|
| Model | : | One dimensional with branches. Wave superposition method used for fluid compressibility effect. |
| Developed by | : | Hitachi Co., Ltd. |
| Status | : | Completed |

6. Sphere-Cylinder Model Sodium-Water Reaction Analysis Code (SWAC-7)

| | | |
|---|---|---|
| Model | : | Combination of spherical bubble growth and one-dimensional (cylindrical) bubble growth. Compressibility of water and sodium considered. |
| Developed by | : | Kawasaki Heavy Industries, Ltd. |
| Status | : | Completed |

7. Pressure Relief & Quasi-static Pressure Analysis Code (SWAC-9)

| | | |
|---|---|---|
| Model | : | Two-phase flow in relief line. Critical flow effect considered. |
| Developed by | : | Ishikawajima-Harima Heavy Industries, Ltd. |
| Status | : | In preparation |

8. Water Leakage Rate Analysis Code (SWAC-11)

| | | |
|---|---|---|
| Model | : | Compressible water/steam flow in heat transfer tubes. Blow-down effect considered. |
| Developed by | : | PNC and Century Research Corporation |
| Status | : | In preparation |