Draft

Report on Safety Research

of High-Level Waste Management

Division of Environmental Safety Research,
Tokai Research Establishment, JAERI

3.3 Safety Tests of Solidified High-level Radioactive Wastes

High-level radioactive wastes arisen in reprocessing of nuclear fuels are planned to be solidified in glasses, and after being stored by means of engineered storage, they will be isolated in the geologic formations. A series of R & D are being performed based on the above plan.

In order to obtain the basic data for the safety evaluation of glass waste forms, several cold tests were carried out.

Synthetic rock and ceramics have been chosen out of various alternative waste forms. Fundamental studies on their properties are now in progress.

3.3.1. Recent work

(1) Thermal shock resistance of vitrified products

Thermal shock of vitrified products may occur in case that outsides of canisters are decontaminated by water or that a fire breaks out in a storage facility and water used to put out the fire comes into contact with the vitrified products. Owing to such thermal shock, cracks are initiated on the surface of vitrified products and the total surface area of vitrified products is increased. The increase of the surface area may result in the increase of the fractional release of radioactive elements from vitrified products and in the decrease of mechanical strength of vitrified products. Such a change of chemical and physical properties of vitrified products has a influence on safety of the storage of vitrified products. So the object of this work is to clarify the thermal shock behavior of vitrified products caused by several conditions anticipated in a facility.

The sample used was a borosilicate glass containing 14% of simulated high laval waste. The shape of specimens was a cylindrical rod. They had a height of 25mm and a diameter of 25mm. The specimen was heated at a given temperature for about 2 hrs with a electronic furnace and then dropped into water at room temperature. The temperature difference between the specimen and the water was varied from 50°C to 600°C. The specimens after quenching were observed by an optical microscope.

When the temperature difference was 50°C, we could not find any cracks on the surface of specimen with an optical microscope. At the temperature difference of 75°C, however, several cracks were observed on the surface of specimen, and deterioretion of physical properties under the influence of such crack formation might occur. The number of surface cracks increased with the temperature difference in the range of 75°C to 500°C. In this range of temperature difference, it was observed that several glass chips peeled off from the surface of specimens. When the temperature difference became 600°C, the number of cracks greatly increased. After quenching, the specimen was easily broken into many pieces, and the original cylindrical shape was destroyed. The crack distribution in the specimen was also observed on a fractured surface of the specimen at the temperature difference of 600°C. The number of cracks decreased stepwise in the inner part of the specimen.

In order to explain the experimental results described above, we have a plan to develop an analytical method which deals with the crack initiation and the crack distribution caused by thermal shock. Such an analytical method will also be useful for the estimation of thermal shock behavior of large vitrified products in canisters.

(2) Effect of the elements of platinum group on devitrification of a high-level waste glass

Devitrification of a high-level waste (HLW) glass may be responsible for deterioration of the properties of the glass. The elements of platinum group in a HLW are considered to act as nuclei in devitrification of the glass. Function of the elements for devitrification of a borosilicate glass containing a simulated HLW was studied.

A simulated HLW glass included 20 wt% simulated waste oxides. A reference glass excluded only the elements of platinum group from the composition of the simulated HLW glass. Thses glasses were melted at 1200 °C for 2 hours in an electric furnace and then annealed at 700 °C for 1000 hours. The devitrified glasses were subjected to optical photography and electron probe microanalysis.

In the simulated HLW glass, fine segregated materials of about ten micrometers in diameter were distributed all over the whole body. This phenomenon is presumed to show the distribution of nuclei of platinum group elements. In the reference glass, large clusters of dendroid segregated materials of about one millimeter in length grew. It was confirmed that the trunks of dendroid segregated materials had molybdenum-rich compositions and rare-earth elements deposited densely on knots of the materials.

Identification of chemical forms of the above-mentioned elements in the segregated materials is needed.

(3) Durability tests of package and structual material

The effects of gamma-rays irradiation on stress corrosion cracking (SCC) have been studied about candidate alloys as high-level waste canisters and overpacks. The purpose of this study is to evaluate integrity of these alloys in the view of safety evaluation for the interim storage and the final disposal systems. ~~These~~ The tested alloys were Type 304 SS, Type 304L SS, Type 309S SS, Incoloy 825, Inconel 600, Inconel 625 and SMA 50. The sensitized double U-bend type specimens were used in this test.

Type 304 SS, Type 304L SS and Type 309S SS have been found to be susceptible to SCC in boiling deionized water with gamma-rays irradiation but not without irradiation (Fig. 1). According to the electrochemical analysis, SCC in these alloys produced would be due to radiolysis of water with gamma-rays irradiation. Fractographic observation revealed that the cracking mode was completely intergranular.

The other alloys were not susceptible to SCC, but SMA 50 rusted over its surface.

In order to obtain the fundamental data of concretes which are used in the construction of storage facilities of HLW, the effects of gamma-rays irradiation on the properties of cocretes was evaluated.

Concretes were tested about the changes of weight, compressive strength, modulus of elasticity, Poisson's ratio, length changes and carbonation depth.

Consequently, it was concluded that the effect of gammmarays irradiation within the range of 1.0 x 10^9 R on the properties of cocrete was uncertain, and that the effect of heating was more distinguished than gamma-rays irradiation effect.

(4) Leaching behavior of glass waste form

Surface layer on the leached glass is liable to act as an important factor on leaching mechanism. In present work constituents of the layer was examined.

The investigated glass was a borosilicate glass (J-10) with 14 % waste oxides.[1] The leaching was carried out by using a quartz Soxhlet leach test apparatus for 200 days. The leachate solutions were analyzed by means of the inductively coupled plasma instrument (ICP) and the atomic absorption analysis. The leached sample was washed, and dried. The sample fixed in resin was polished. The leached surfaces were analyzed by optical microscopy, scanning electron microscopy (SEM) and electron probe micro analysis(EPMA).

The specific weight loss after leached for 200 days is about 63 %. Fractional releases of Cs, B, Mo, and Na are almost the same as the specific weight loss. But releases of Ba, Sr, and Ca are less than above value and these value are about 50 %. The values of Cr and Al are also small(~40 %).

The sample leached for 200 days is covered by a gel layer. It was found that this surface layer was composed of at least four distinct phases. The thickness of this layer is about 550 μm.

The relative concentrations in this layer are shown in Fig. 2 in the form of X-ray line scans, each element profile obtained at the arrow in a ~~micro-photograph~~ photomicrograph of the surface layer. Si increases with depth to the composition of the bulk glass.

Ca was found more at the inner portion in the layer. Al was found to be slightly depleted from the surface layer relative to the bulk glass but its concentration in the layer changes little. Na was found to be depleted from the surface layer, as were Cs and Mo. Contents of Fe and Ce are higher in the surface layer, particularly the outer portion, relative to the bulk glass. The other rare earth elements have the profiles similar to these of Fe and Ce.

Although the leaching mechanism strictly differ with leached elements, it is clarified from this experiment that the leaching mechanism can be considered by classifying the leached elements in-to at least three groups; first group elements (Na etc.) are depleted through the surface layer, second group (Si, Ca etc) are remained in the inner portion of layer, and third group(Fe, Ce etc.) are enriched in the outer portion of layer. On the basis of these facts, the research is in progress to establish leaching mechanism and to develop mathematical models of leaching.

(1) T. BANBA, et al., "Simulated HLLW compositions for cold test of waste management development." JAERI-M 82-088(1982)

(5) Examinations and studies for the high level waste forms to be returned from the oversea-reprocessing

The high level waste forms arisn from the oversea-reprocessing are scheduled to be returned to Japan from 1990. The following research items were studied :

① The studies of safety criteria for high level wastes management.

② Characterization of high level waste forms to be returned from the oversea-reprocessing.

The five years program has started since 1981, for the studies of safety of high level waste storage facilities. In this year, the main research items were settled to evaluate the safety of the facilities.

(6) Alternative waste forms

Two kinds of ceramics have been selected for alternative waste forms. One is zeolitic ceramics and the other is SYNROC. The former was prepared by hot-pressing and the results have been already reported[2]. The latter was prepared by hot-pressing plus air-sintering to examine its constituent phases and distribution of simulated high level waste elements by means of X-ray diffractometry and analytical electron microscopy, respectively. Only three phases, hollandite, $BaAl_2Ti_6O_{16}$, perovskite, $CaTiO_3$ and zirconolite, $CaZrTi_2O_7$ occurred except unidentified phase(s) which had a few very low peaks in an X-ray diffractogram. It is possible that such unidentified phases might have an influence on leaching behavior. The examination of distribution of elements gave the following notable features: Mo was found in perovskite instead of forming water-soluble Cs-molybdates, and more amount of rare earth elements was accommodated in perovskite than in ziconolite.

(2) T. BANBA, et al., "Ceramic Solidification Tests of High Level Wastes with Natural Zeolite (I) (Effects of Treatment Procedures and Additives", (in Japanese), JAERI-M 9193 (1980).

3.3.2 Future plan

The hot tests in WASTEF (Waste Safety Testing Facility) are being carried out to aim at performing following 3 items with the support of cold tests.

(1) Characterization by using radioisotope

Stability against α-radiation will be tested by using Cm-244, short lived α-nuclide. α-Disintegration of $1 \sim 3 \times 10^{18}$ is to be obtained in unit cubic centimeter of boro-silicate glass sample after 2 years storage.

Several property changes are to be measured for instance leachability, stored energy, He residue in glass, etc.

(2) Improvement of accuracy by using radioisotope

Behavior of nuclide on leaching, vapourizing and pulverizing will be investigated using Cs-134, Sr-85, Ru-106 and Mo-99.

(3) Demonstrative test

Heat generation by radioisotope and properties of real waste samples are to be measured to cirtify the justness of cold test and to confirm the applicability of the measuring methods.

3.4 Safety Evaluation Tests for Geological Disposal of High-level Waste Solids

An extensive R & D has been carried out for long term assessment about the geological disposal. The study is constituted by two kinds of approach that are examination by experiments and estimation by computer codes.

Experiments on nuclide behavior in rock formation have been carried out in laboratory and in field. The comparison between the both data should showed interesting results. Such approach will give a clue to establish a total system of the safety assessment in future.

Regarding to the estimation study by computer code, calculation about nuclide migration in geosphere has been started.

3. 4. 1 Recent Work

(1) In-situ measurement of heat transfer and thermal stress in schalstein rock mass

One of the most important impacts to the stability of a repository for the high level waste is the thermal disturbance from the decay heat.

Field tests to investigate the heat transfer in-situ have been conducted since the end of 1977 by JAERI.

The drift at the depth of 380 m under the earth surface in Akenobe Mine was used for the heater experiments.

The rock mass consists of a schalstein of Permian age.

Two 48 mmϕ holes for measuring water content (No. 10) and thermal stress (No. 9) and eight 29 mmΦ holes for electric heaters (No. 1, No. 2) and thermocouples (No. 3 ∼ No. 8) were drilled as shown in Fig. 3 .

The relation between observed temperature and calculated one at 2 kw heating was shown in Fig. 4 .

It was found that the heat flow by water was not only negligible but also predominant over the heat conductivity especially in the field of rich water content.

On the other hand, temperature distribution in the dry and under 100°C area could be estimated by heat conduction.

The value of the thermal conductivity of a rock mass in-situ was bigger than those of core samples in a laboratory and the water in a rock mass was liable to increase the thermal conductivity.

The behavior of water was investigated.

Temperature distribution and thermal stress were measured to clarify the mechanism of influence of water on heat transfer.

Concerning to the thermal deformation of the rock mass, the large compressive stress was observed in the deeper area and the tensile stress at the rock surface.

To support above experiments, temperature distribution in the near field of a repository and the stress around the tunnel were calculated.

(2) The development of nuclide migration code

For the assessment of the radionulide migration from the high-level waste repository, two mathematical approaches have been developed by many researchers. One is the "modelequation" method using transport equations which are solved by using finite-difference or finite-element techniques with appropriate initial and boundary conditions.

Another approach is the "direct-simulation" method which defines numerical structures of migration process. The direct-simulation method requires that an efficient bookkeeping structure is established to control the response of the numerical representations so that all physical constraint are satisfied. This method has following advantages;

(1) It is always mass conservative.

(2) There is no cumulative numerical dispersion.

(3) There is inherent numerical stability.

(4) It facilitates handling of multicomponent systems.

The phenomena taken into account are radioactive decay, convection and dispersion in the ground water and sorption and desorption in the geologic media.

Numerical calculations of ^{234}U decay chain in the single layered geologic media have been carried out, and reasonable results have been obtained in comparison with UCB code MGRATO3. The concentration profiles at the location of X = 500 m is shown in Fig. 5 .

(3) In-situ ion migration tests in fractures

Migration of nuclides was studied in a real environment in situ for safety assessment of the HLW geological disposal. Retardation of the migration is due to the sorption of ions on minerals, not only in a rock but also in a fracture. The latter one is usually clay minerals, so it must be more predominant because of its larger ion exchange capacity. Importance of in situ test is that natural and real structure of the fracture is only available there.

The test was carried out in the same place used for heat conductivity test described above. No. 9 hole was used for charging solutions containing ions. Double packers were inserted into the hole at about 18 cm and 68 cm depth. The solution was charged in at 4 kg/cm^2 pressure by compressor. Water appeared at position A-D on the wall of the drift, which was collected with funnels fixed on the wall. No leakage of the solution to the other testing holes was observed.

LiI, $CsNO_3$, $Ba(NO_3)_2$, $Sr(NO_3)_2$ and $ZrO(NO_3)_2$ were disolved in water to prepare the solutions of 50 - 100 $\mu g/ml$ of metal elements.

Typical break through curves are shown in Fig. 6 . Here C/Co is the ion concentration ratio between the effluent and initial solution. Low absorption coefficient of I^- is known by many workers. The C/Co of I^- after leveling off (0.85) shows the dilution by natural water, which agree with the values of Sr^{2+} and Cs^+. Slower effluent in position B defines the difference of retardation between Sr^{2+} and Cs^+ more clearly than in position C. The C/Co of Ba^{2+} is smaller than the one of Sr^{2+}. ZrO^{2+} was not detected or large fluctuation of the value was observed.

(4) Nuclide Sorption test

As a study of nuclide migration from high level waste, sorption of Cs on granite has been measured in the laboratory.

The apparent distribution coefficients (Kd) for Cs between solution and granite rock was measured by batch method under conditions with various initial concentrations of Cs from 1×10^{-5} μg/ml to 10 μg/ml with ^{137}Cs tracer. The Kd values increases from 50 ml/g to about 170 ml/g with decreasing final cesium concentration from 10^{-1} μg/ml to 10^{-6} μg/ml.

The dependence of Kd value on cesium concentration and also on contact time suggested that the sorption behavior was very complicated and the validation experiments in the geological formations were necessary.

(5) Hydrothermal interactions between simulated high level waste glass and natural rocks

Nuclide release from the repository is controlled by the hydrogeological and geochemical interactions. For the assessment of HLW geological disposal method, it is fundamental to investigate the interaction of potential waste forms with grounwater under repository conditions. Therefore, the interaction of a typical waste form and candidate repository host rocks were examined under the conditions of 300°C, 300 bars.

Starting materials were borosilicate glass, granite, basalt and de-ionised water. The glass contained approximately 20 wt% of simulated fission product oxides. The initial fluid : glass : rock mass ratio was 20 : 1 : 1. The run duration was 30 days.

Eleven elements were analyzed for the liquid phases from hydrothermally treated mixtures by AAs or ICP method. Silicon, sodium, iron, calcium, potassium, magnesium, manganese, molybdenum and cesium were detected but strontium included in simulated HLW glass at about 0.3 wt% as simulated radwaste element was not detected at all. Analyses of the solid phases by optical microscope and SEM revealed that a few new crystalline phases were formed during hydrothermal treatment. One of them was mainly composed of silicon, aluminium, sodium and cesium.

3.4.2 Future work

It is planned to develop the model for evaluating the nuclide migration in geologic media and underground water movement in the near field.

Hydrothermal interaction between high level waste production and rocks are studied in order to predict the long-term behavior in geologic media.

Experimental work including corrosion, leach and compatibility among the artificial barrier materials are planned to be measured under the γ-ray irradiation from ^{90}Sr and ^{137}Cs source.

By field test, sorption and diffusion of nuclides in rock mass will be measured in unsaturated water flow through fractured rock.

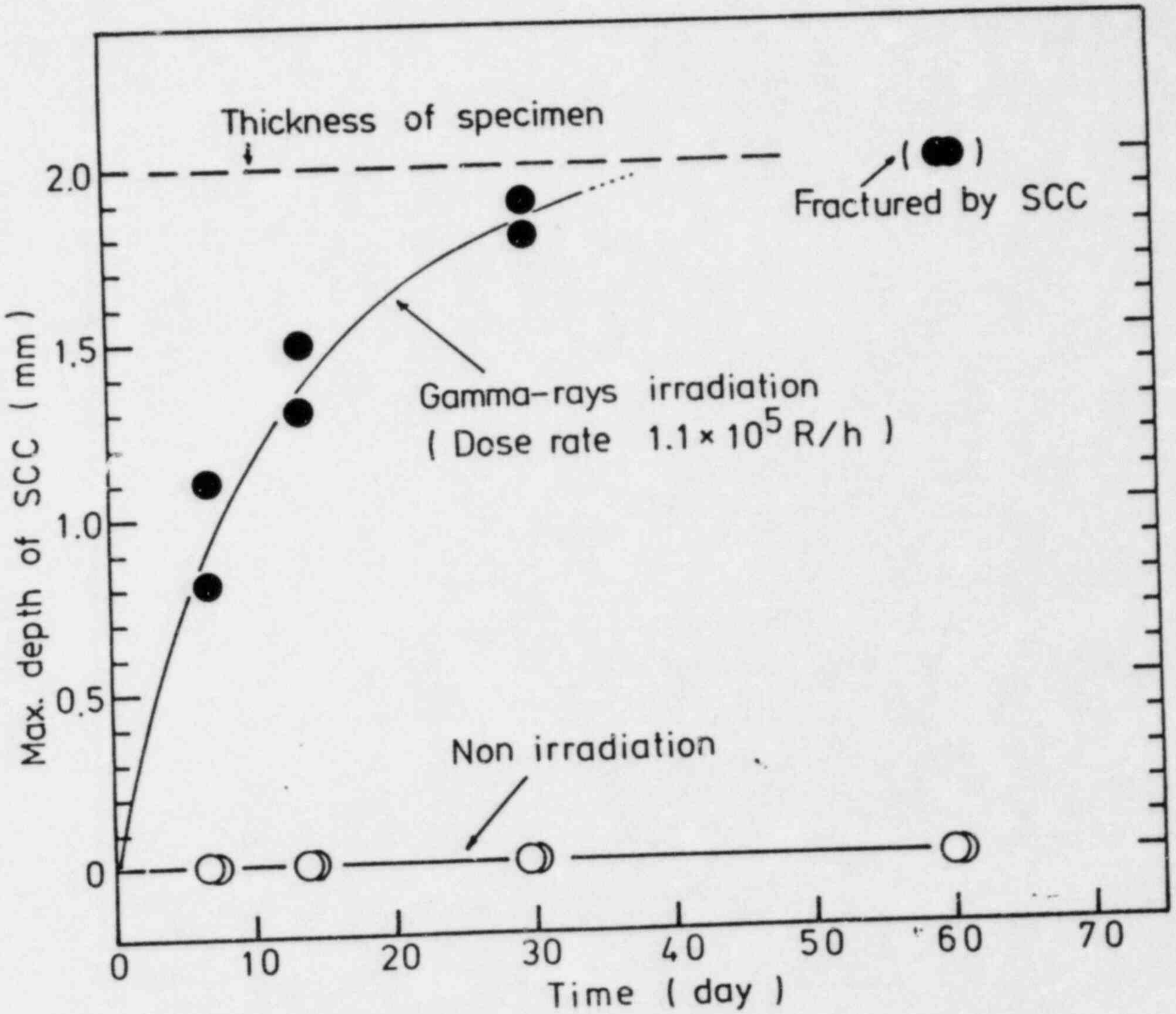


Fig. 1 Effects of gamma-rays irradiation on SCC failures of sensitized Type 304 ss.

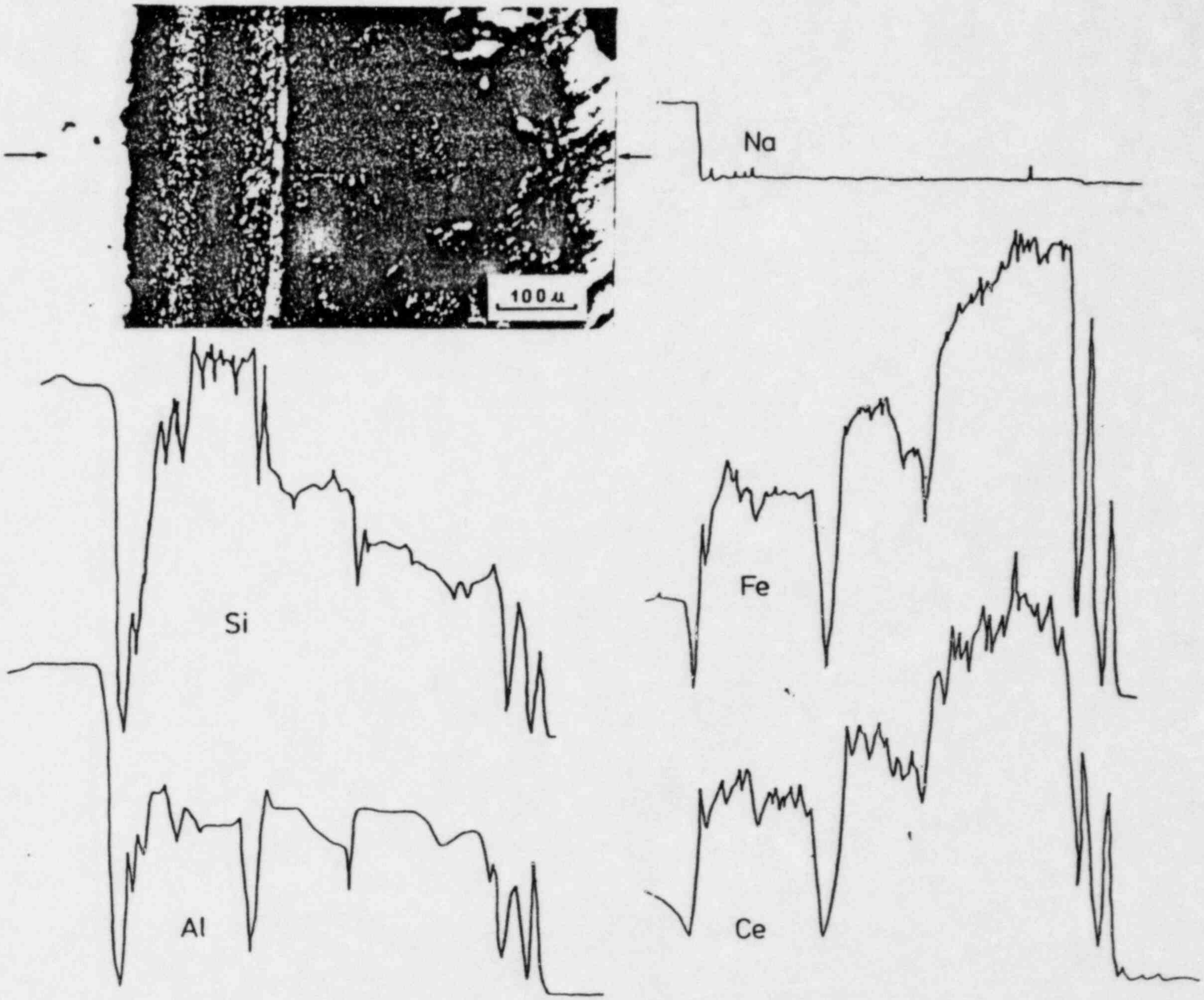


Fig. 2 X-ray line scans of surface layer of waste glass

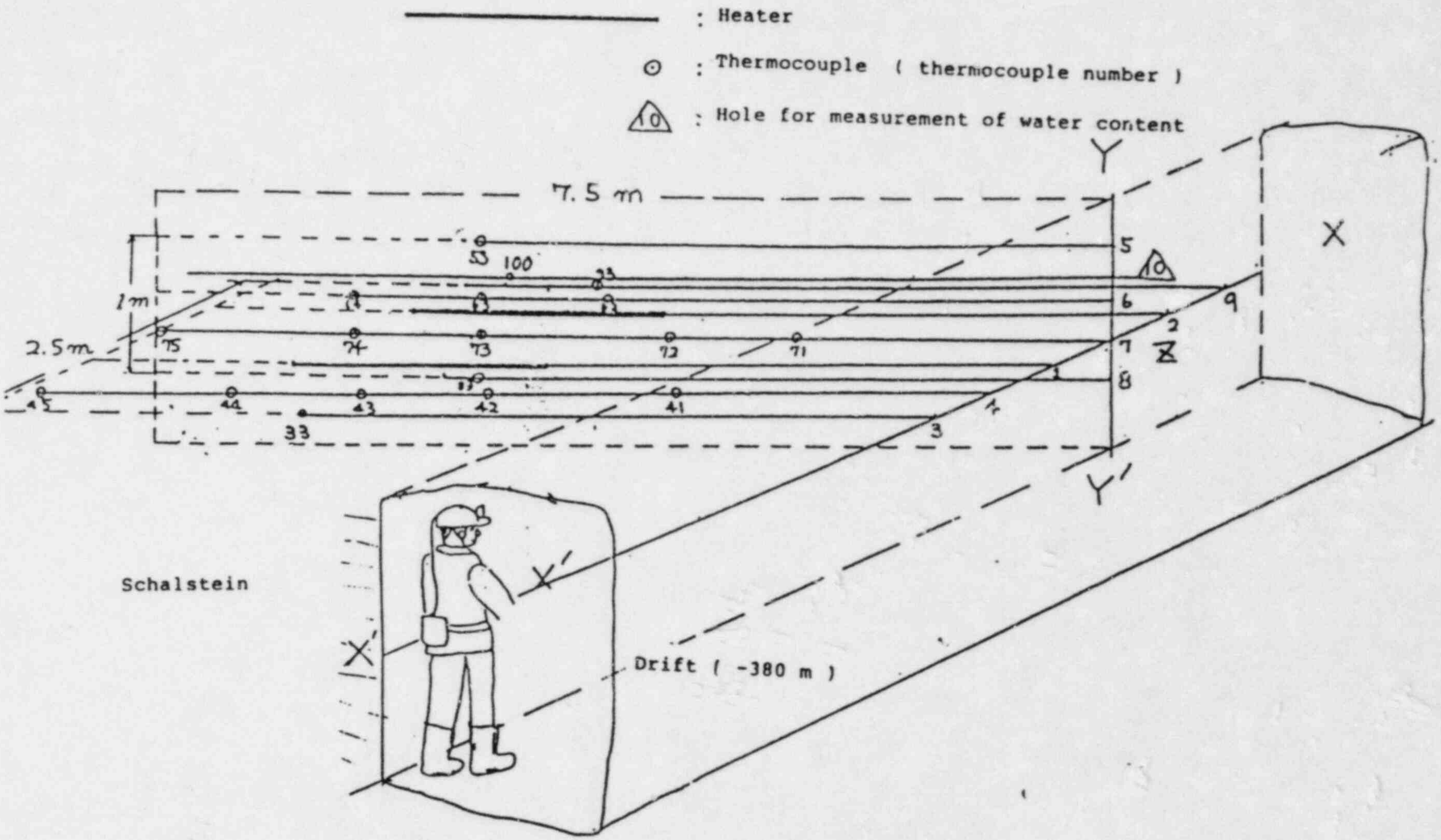


Fig. 3 Layout of heaters and thermocouples

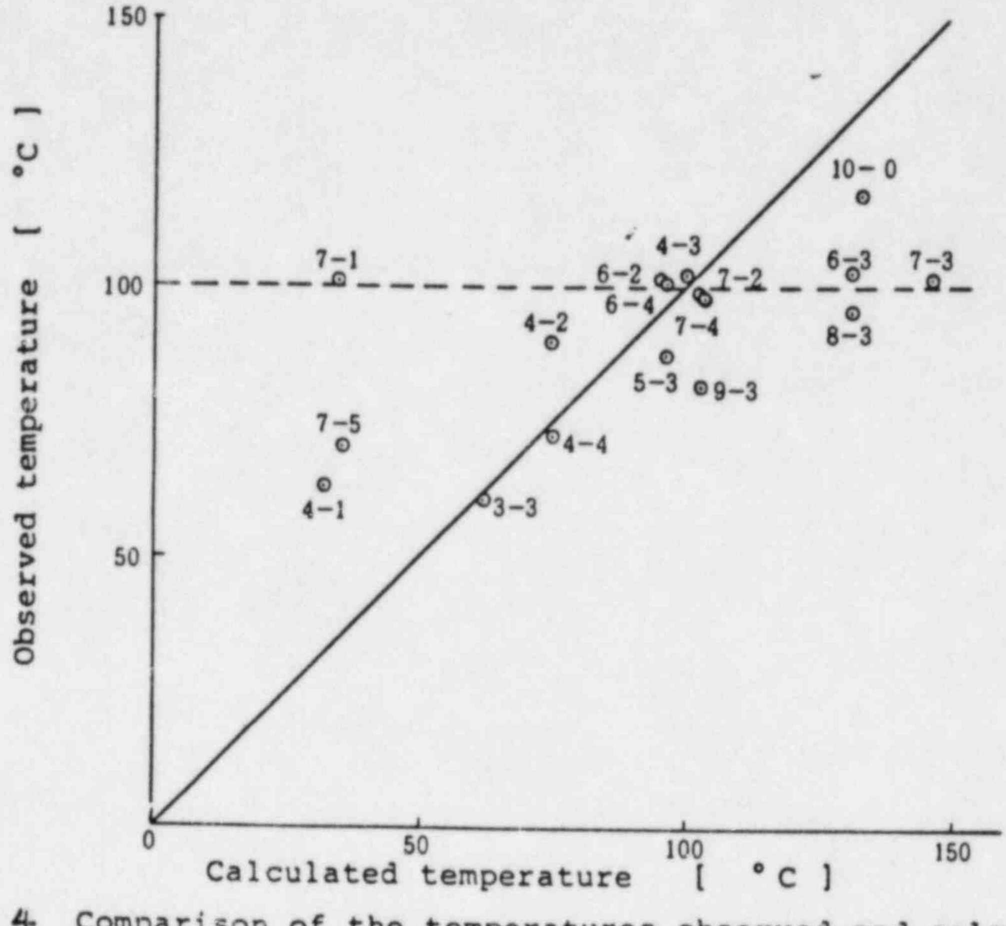


Fig. 4 Comparison of the temperatures observed and calculated after 45 days

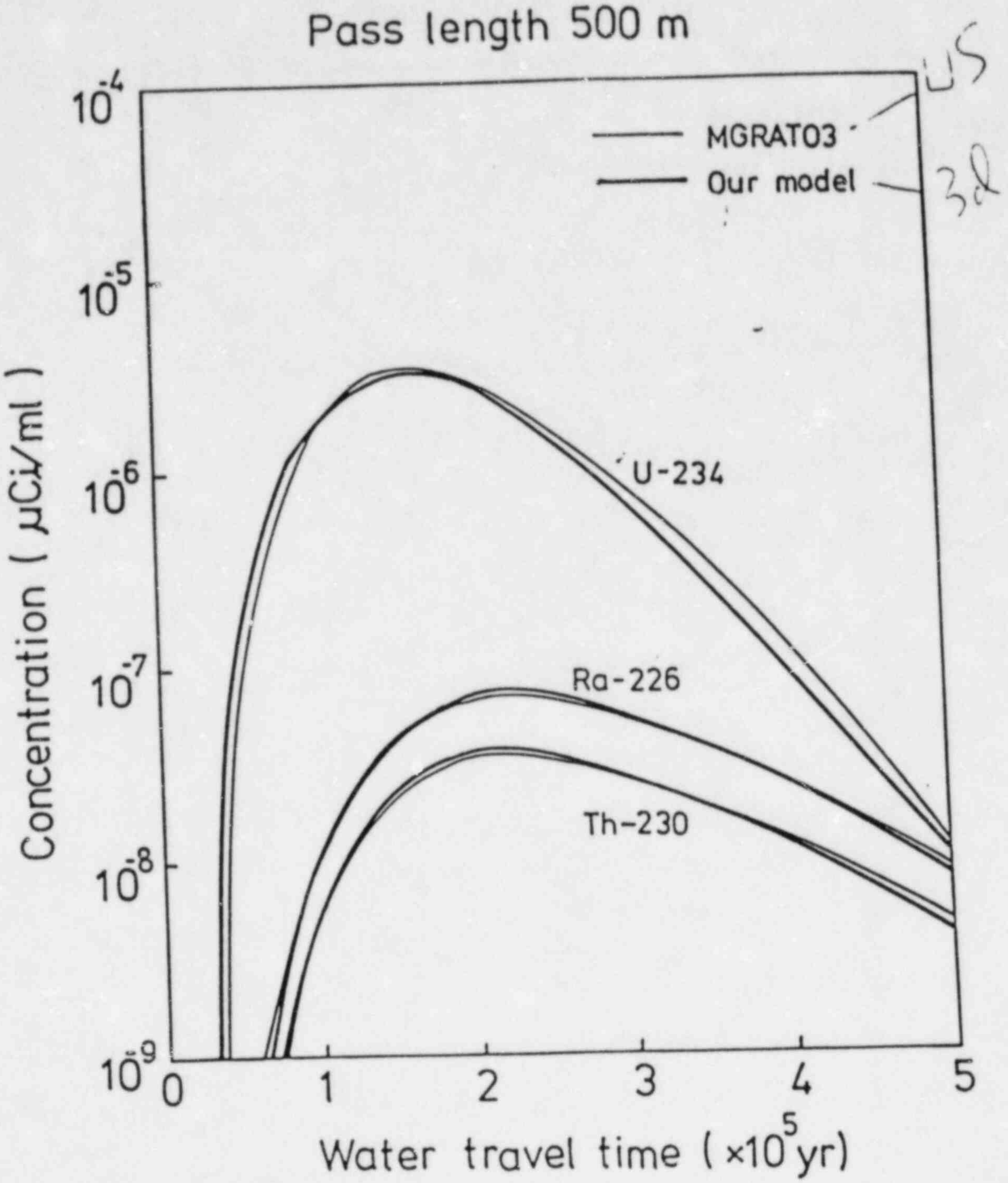
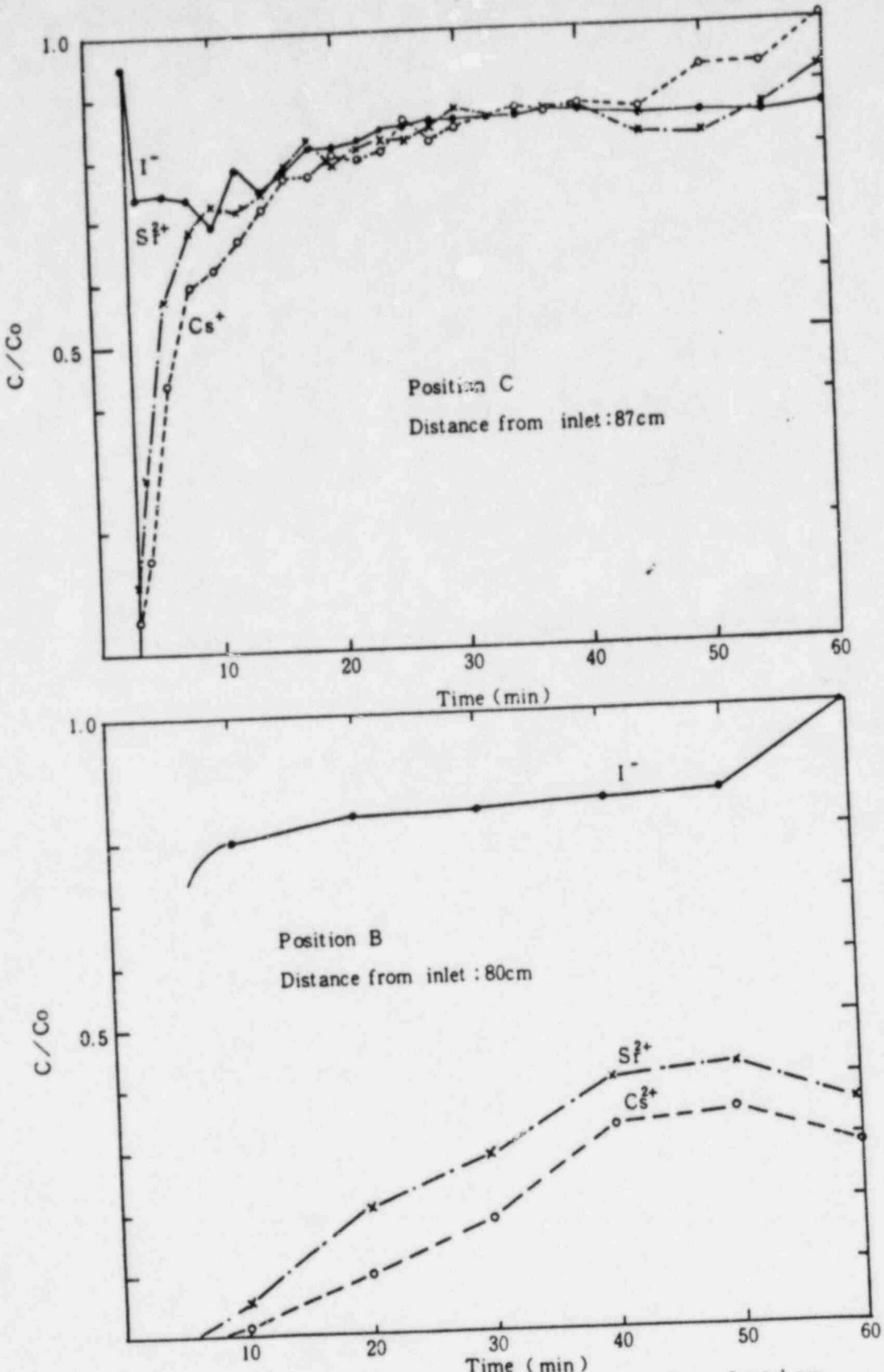


Fig.5 Concentration profiles at the location of X=500 m



C_0: Concentration of the charged solution

C : Concentration of effluent

Fig. 6 Break through curves of ion from fissures in rock mass

9.4 Hot operation tests of WASTEF

The Waste Safety Testing Facility (WASTEF) was planned to obtain useful data for safety assessment of high-level waste management by testing solidified HLW forms with actual wastes and relating materials under long term storage and disposal conditions. It involves five concrete cells and a lead cell to handle radioactivity upto 5×10^4Ci and to store upto 1×10^6Ci. (Fig. 1) After the facility was completed in Summer of 1981 and each apparatus and equipment in the cells was tested in the cold stage, the hot operation started using cesium-137 in the end of November 1982.

The vitrification apparatus, which is installed in the No. 2 cell, can provide vitrified forms of various volumes (usually 1ℓ, in maximum 5ℓ) and of various compositions for the safety tests. A vitrifying process takes approximately seven days through the steps of waste preparation, denitration with formic acid, evaporation, calcination after mixing with glass forming materials, vitrification in a metallic melter by heating upto 1300°C and cooling gradually in a canister. (Fig. 2) Fig. 3 shows a temperature distribution in the melter displayed graphically by a computer, which is installed for the treatment and control of data obtained in the facility. Such in-situ indications can promote effective and safe operations of the facility.

Fig. 4 also shows a temperature distribution in the storage test apparatus, which is installed in No. 1 cell to test safety of long term storage under usual and unusual conditions using 1ℓ vitrified form and canister.

Brief description of other main apparatus installed in the cells is as follows. Sample preparation appratus ; preparation of samples of various sizes for various tests, including three cutting machines, a core-drilling machine, polishing machine etc. Disposal test apparatus ; leaching vitrified forms of 1ℓ in underground water and measurement of reaction between the leachant and rocks. Characterization test apparatus ; measurement of foundamental characterization of vitrified forms including heat generation, thermal conductivity, leachability etc.

α - damage test apparatus ; measurement of long term α - damage on vitrified forms using short half-life nuclides as Curium-244.

In 1983, a full operation starts using radioisotopes over 10^3Ci, and after basic data have compiled using radioisotopes, tests with actual wastes will start in 1986.

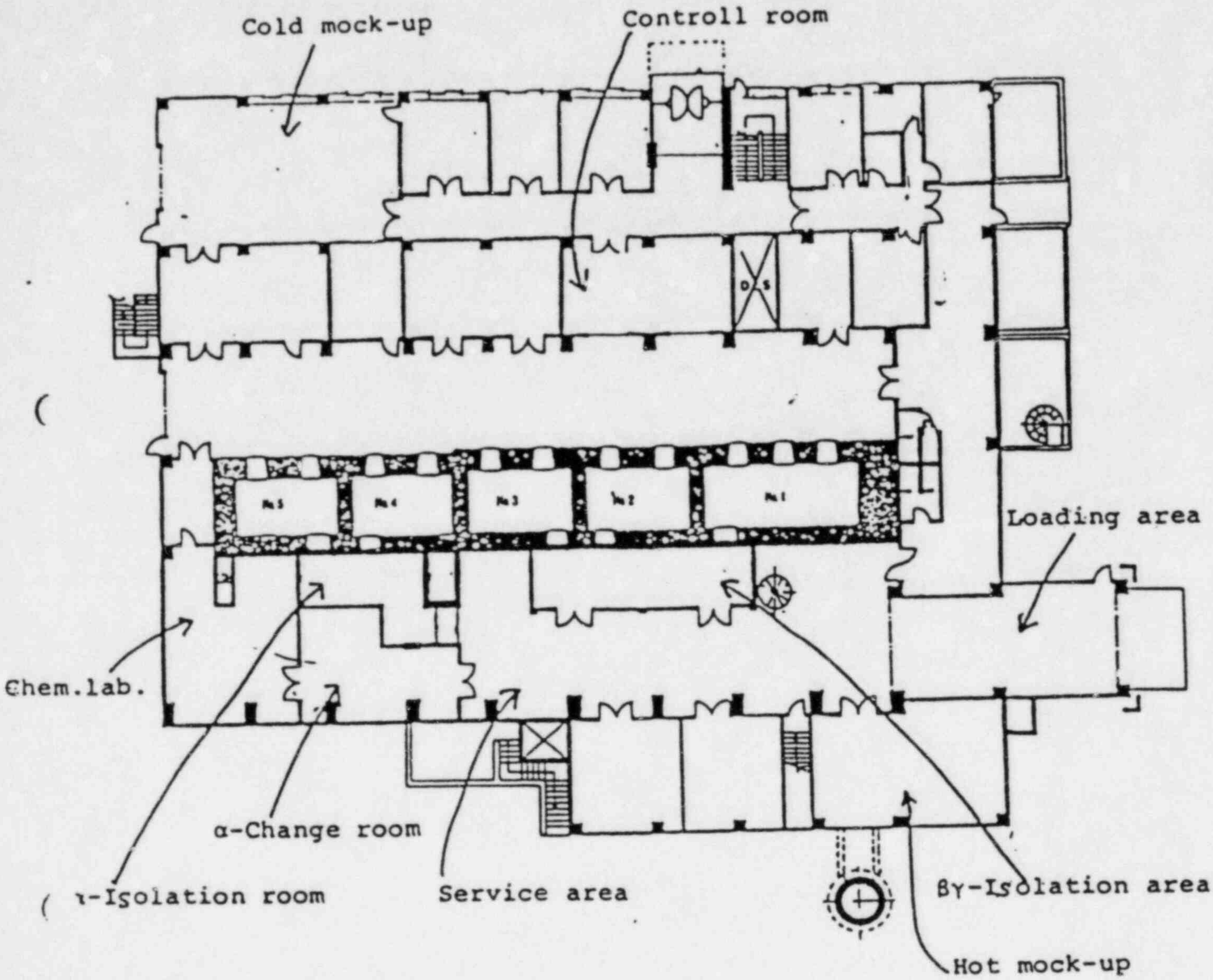


Fig. 1 Floor Plan (1st floor) of WASTEF

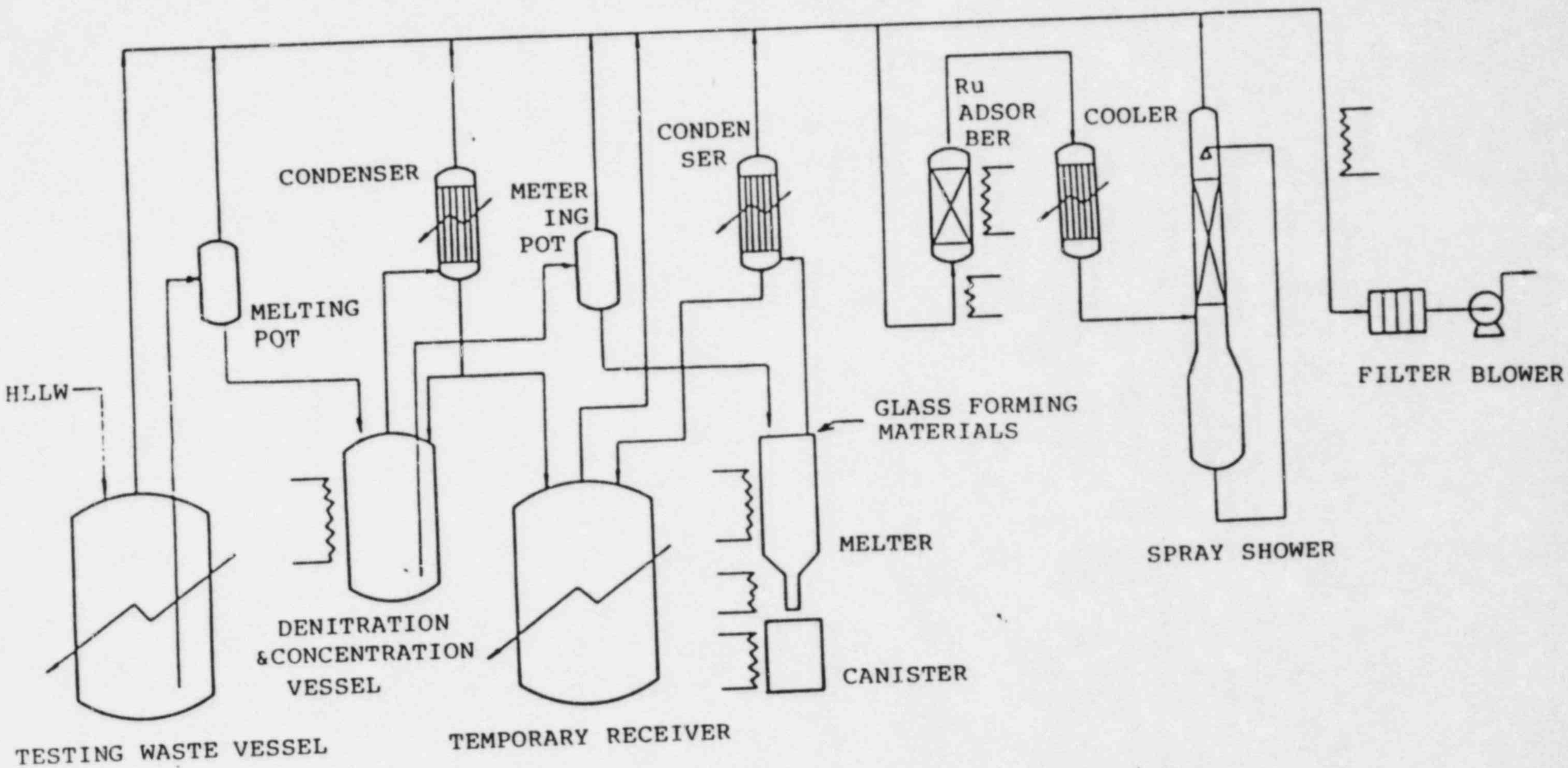


Fig. 2 Flow sheet of vitrification apparatus

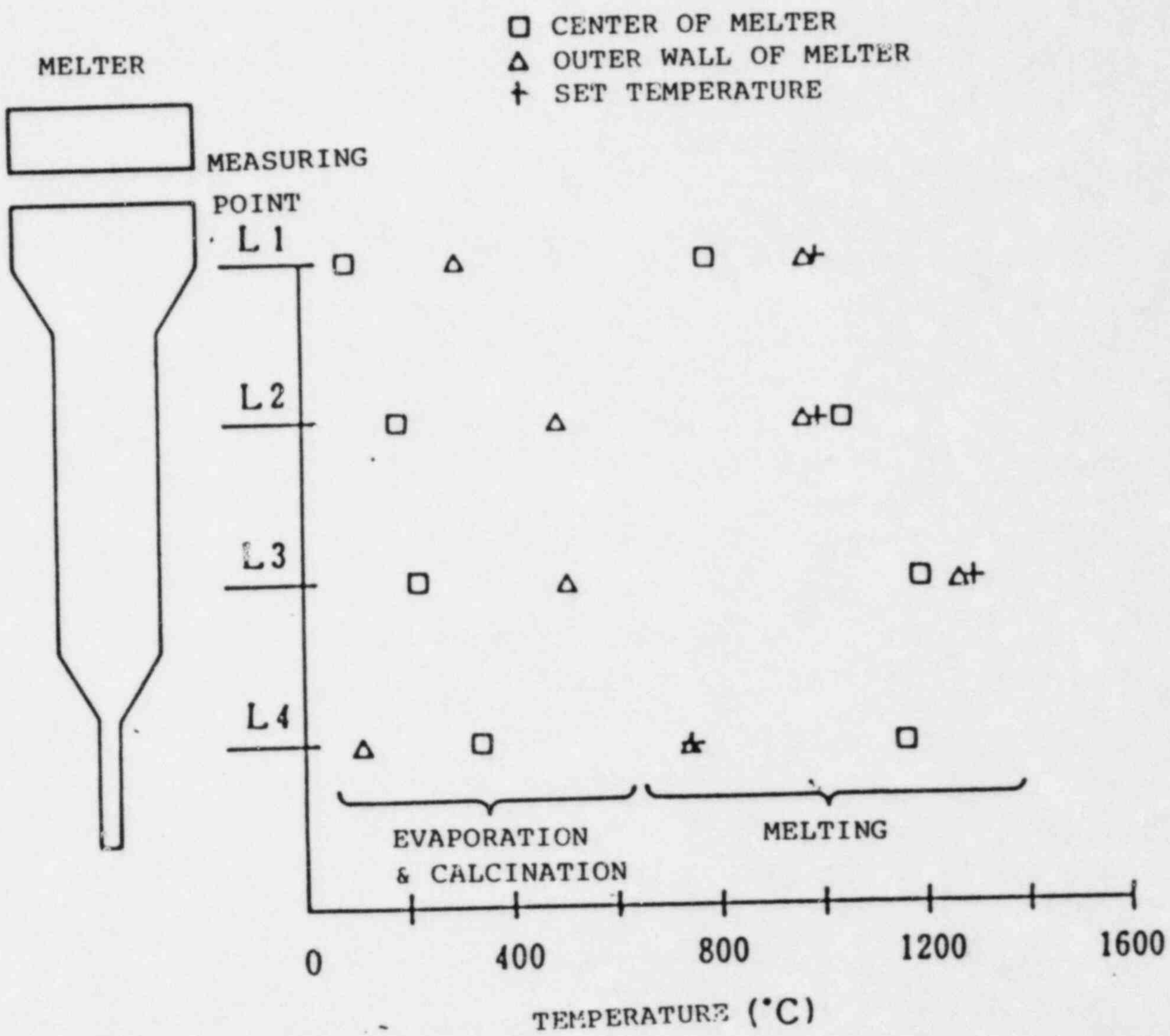
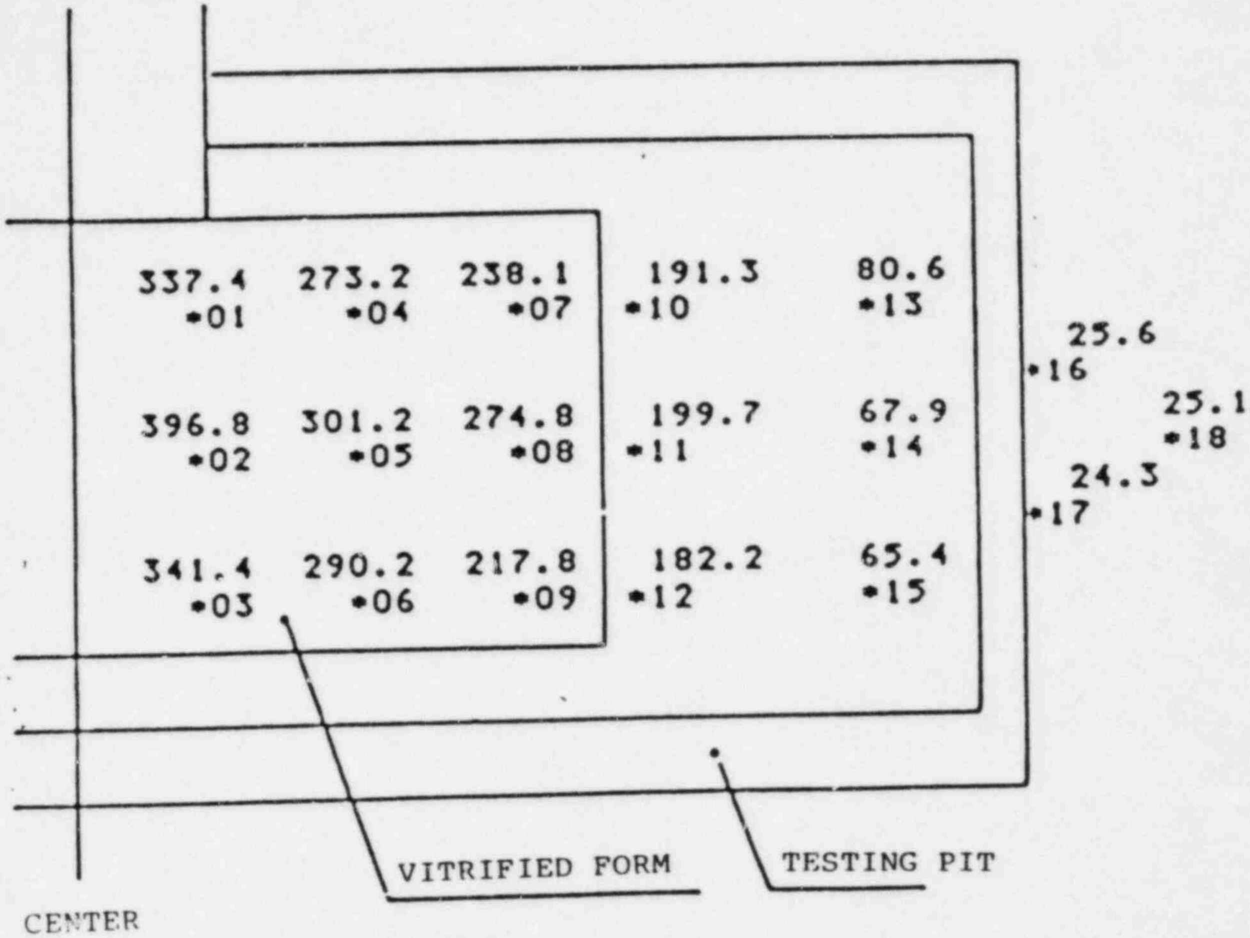


Fig. 3 Temperature di tribution of melter of vitrification apparatus



*01	VITRIFIED FORM	(inner-high)
*02	ditto	(inner-middle)
*03	ditto	(inner-low)
*04	ditto	(intermediate-high)
*05	ditto	(intermediate-middle)
*06	ditto	(intermediate-low)
*07	ditto	(outer-high)
*08	ditto	(outer middle)
*09	ditto	(outer-low)
*10	CANIATER SURFACE	(high)
*11	ditto	(middle)
*12	ditto	(low)
*13	IN TESTING PIT	(high)
*14	ditto	(middle)
*15	ditto	(low)
*16	TESTING PIT SURFACE	(high)
*17	ditto	(low)
*18	ATOMOSPHER IN THE CELL	

Fig. 4 Temperature distribution(°C) in case of inner heating of storage test apparatus

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2. Evaluation and Analysis of Environmental Radioactivity

1983

DIVISION OF ENVIRONMENTAL SAFETY RESEARCH

2. Evaluation and Analysis of Environmental Radioactivity

With regard to evaluation and analysis of environmental radiation, methods have been developed to evaluate exposure dose from nuclear facieties, by separating the natural radiation contributions from observed exposure data. In 1981 and 1982, measurements of cosmic ray exposure rate on the ground at different altitudes have been undertaken. Also, Monte Carlo calculations have been performed to clarify the gamma ray behavior in environment.

With regard to measurement and evaluation of radionuclides in natural environment, the purpose is to evaluate the exposure dose by radionuclides released from nuclear facilities. Accordingly, the accurate surveillance on background radioactivity was continued in order to accumlate the data of behavior of main radionuclides in natural environment. The depth distributon of radioniclides in soil and sea sediment was suveyed in 1981 and 1982.

With regard to study on diffusion of gaseous and liquid wastes, experimental investigation has been conducted for atmospheric diffusion in coastal area to develop computation models. In 1981 and 1982, a series of trajectory observation was carried out. A rawin-sonde attached to a constant level balloon was tracked by double radio theodolites. Mehtods of tracer experiments for diffusion in sea water, especially in a surf zone, has been developed. In 1981, aerial photographs of tracer dye were successfully taken using a radio-controlled model airplanes.

The development of computer systems of assessing environmental radiological consequences attributed to nuclear facilities aims at the generic assessment of whole and long effects on the public of radioactivity released into environment from the activities in the nuclear fuel cycle in Japan. The study includes the collection and analysis of environmental data which the systems need for the assessment. In

1981 to 1982, the population doses from the gaseous effluents released from all the LWRs were assessed, a system for the internal dose calculation was almost completed, and a regional atmospheric dispersion model was progressed. Efforts are being made to develop a data-base system for collection and analysis of the bioaccumulation of radioactivity by marine organisms.

Research and development of emergency monitoring and prediction code system for accidental release from nuclear facilities are undermentioned.

With regard to research and development of aerial gamma ray survey system, fundumental experiments, both on natural radiations and on controlled gamma rays from Co-60 and Cs-137 sources have been carried out. The experiments are also intended to test the suvey system characteristics in operation on a helicopter. An areal air sampling and radioactivity measuring system by a small aircraft was developed to estimate the concentration of radionuclides accidentally released into atmosphere. On the selective cllection of methyl iodide by inorganic adsorbents in the presence of noble fission gases, a test apparatus for iodine sampling cartriges was constructed. And weathering test is planned on the cartriges for long strage, in connection with its use for emergency. For a stack gas monitor valid for environmental exposure evaluation, the exposure rate-release rate gas monitoring method was developed.

With regard to construction of a code system of prediction of environmental cosequences at the nuclear accident, have been developed the first tentative version of code system AAA-I in 1982. The improvement and investigation of practical use of this system will be performed in 1983. For calculation of wind field, concentration and dose, two sub-systems have been prepared. One is a simple sub-system, utilizing the Gaussian plume and puff models, another is a realistic sub-system making use of the mass-consistent wind field model and the particle-diffusion model. In study about radiation shielding effects of buildings, the mass attenuation coefficients of various building materials for low energy gamma rays were obtained.

8.1 Evaluation and Analysis of Environmental Radiation

So far, there has been carried out the research on environmental radiation characteristics, on theoretical analysis of environmental gamma rays and on measuring instruments and methods. Also the evaluation mathods for environmental dose from nuclear facilities have been researched.

(1) Measurement and Analytical Evaluation of Environmental radiation

In order to clarify the environmental gamma ray behavior, a Monte Carlo transport calculation code was developed. The code is intended to calculate gamma ray transport effectively utilizing the geometical symmetry. The calculated results indicated good agreement with the experimental data performed at Nevada test site by F. Haywood et al.[1] Using the code, field experiments with the air-borne gamma-ray survey system described later are going to be simulated, and the data necessary for analysis of gamma ray transport will be accumulate.

(2) Study on Evaluation of Environmental Radiation

For last few years, measurement data on energy spectrum and exposure rate from natural gamma rays and cosmic rays have been collected. Many investigators have reported on vertical distribution of cosmic ray intencity making use of airplanes. However, there are only few data on the variation of cosmic ray intensity at various ground heights. The experiments were carried out using a NaI(Tl) scintillation spectrometer, a high-pressure argon ionization chamber and a spherical plastic ionization chamber at several mountains of different altitudes. The observed exposure from cosmic rays at the top of Mt. Fuji (3776m) indicated about 12μR/h higher value than the data of sea-level. Also, the data at Mt. Tsukuba (870m) and at Mt. Yamizo (1022m) are 1μR/h higher compared with the sea-level data. Fig. 8-1 shows the absorbed energy pulse height spectra obseved by a 3"φ spherical NaI(Tl) scintillation detecor at various altitudes.

1) F.F. Haywood, et al., CEX-62.14 (1964)

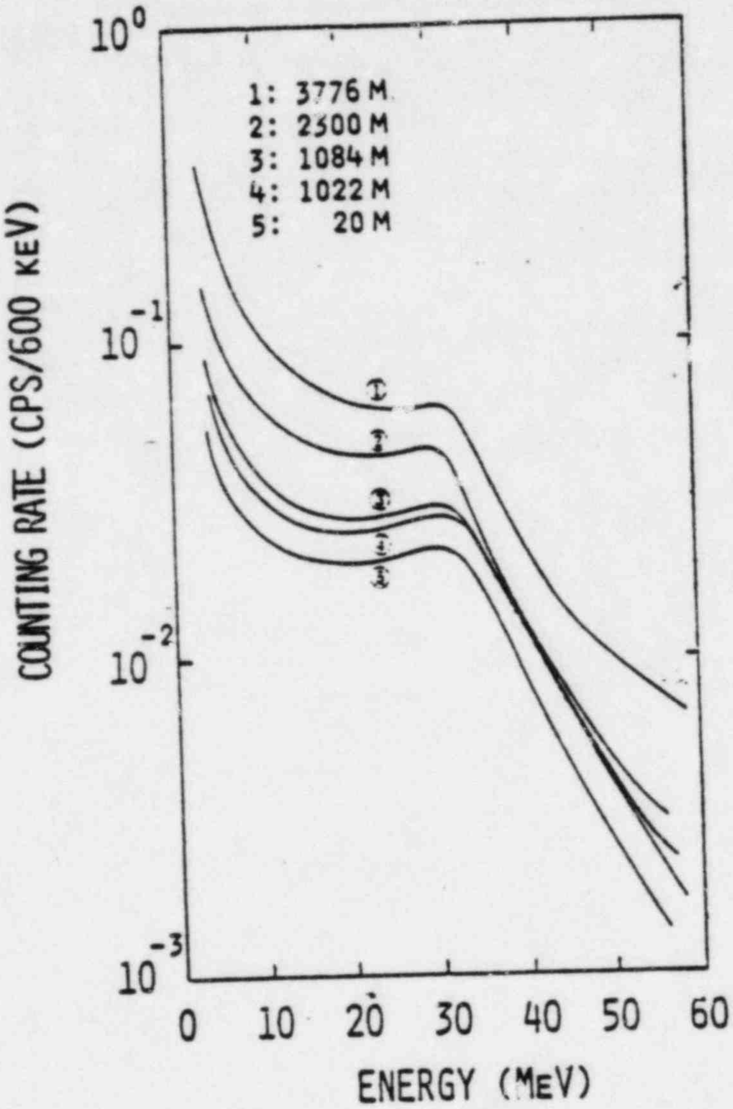

Fig. 8-1 Absorbed energy pulse height speccra observed by a 3"ϕ NaI(Tl) scintillation detector at various altitudes.

8.3 Study on Diffusion Model of Gaseous and Liquid Wastes.

(1) Atmospheric Diffusion

It is often assumed in the assessment of short range atmospheric diffusion that plume axis is straight line. The assumption may not be valid when travel distance reaches some 100km. The underlying topography and the local meteorological conditions such as alternation of land-sea breeze. may affect the trajectory of effluent. In order to study the medium range trajectory, trackings of constant volume ballon(tetroon) by double theodolites were conducted in 1979 at Tokai site. The maximum tracking distance with this method seemed to be 15km on the day of highest visibility. Various tracking methods using radio wave have been developed by many researchers for longer range air trajectory. Radar-transponder system is considered to be the best among them.

We have improved the radio wave method to save operation cost and man power required in radar-transponder system. In this method, low level rawin sonde is attached to tetroon, and tracked by two automatically tracking receiver systems set up several kilometers apart. Field experiment with this system was performed from October 6 to 23, in 1981. Tetroon adjusted to fly 300m above ground was released from a station in Tokai Establishment. Distance of tracking syotems was 7.5km. The position of tetroon was determined through the azimuth angles simultaneously measured by two receivers and the signal of atmospheric pressure from rawin sonde.

Fourteen runs of trajectory data were obtained maximum tracking distance of which was 27km. Systematic change of flow direction above Naka-River was noticeable. There included the tetroon motion arrive at sea breeze front and went aloft rapidly. The new observation method was confirmed to be effective for medium range trajectory study.

Simultaneous tracking of balloons adjusted to fly multi-levels is projected in order to make clear the effect of vertical wind shear.

(2) Diffusion in Sea Water

A tethered balloon system has been developed in order to take aerial photographs of dye-cloud in times of coastal diffusion experiment. The system has some deficiencies, that is, it cannot be used both under strong winds and under on-shore blowing winds. To recover the deficiencies, another method of aerial photography was developed.

A radio controlled airplane on the market was reconstructed to mount a 35mm camera, the shutter of which is handled through the channel ordinarily used to control wheel logs. Observation with this system was carried out from winter to summer in 1981. Performance of the new system was satisfactory. The program of surf zone diffusion experiment has been left off since 1981.

8

8.4 Research and Development of Emergency Monitoring and Prediction Code System for Accidental Release from Nuclear Facilities

(1) Research and Development of Aerial Gamma Ray Survey System

In 1980, the survey system equipped with the fundumental functions was constructed considering the required conditions for preliminary research. The system, orizinally developed for geological research, is able to measure low-level gamma rays from radioactive nuclides of K-40, Th-232, U-238 and their daughter elements effectively. The system was reconstructed for use of gamma ray survey in emergency situation. The air-borne parts consist of two NaI(Tl) scintillation detectors, one of which has 1024 $inch^3$ volume, two multichannal pulse height analyzers, a data acquisition instrument and an MT recorder. On the ground, a mini computer is employed to process the obtained data rapidly in situ. In order to identify the flying position, a video recorder searching a ground surface, MRS (Microwave Ranging System) and a radar altimeter are added to the survey system. In 1981, field experiments were performed to research the natural radiation property and the system characteristics. Fig. 8-7 shows the used helicopter in flight to measure natural radiations. In 1982, were carried out experiments making use of Co-60 and Cs-137 gamma ray sources in an air-over-ground geometry. Based on the experimental data, the research and developmant of the practical survey system, including the data process and analysis, will be cintinued.

Fig. 8-7 Helicopter equipped with the air-borne gamma ray survey system in flight to measure natural radiations.

(6) A Code System for the Prediction of Environmental Consequences at the Nuclear Accident

Several computer codes and the first tentative version of code-system, which controls the codes, have been developed for the prediction of environmental consequences at the nuclear accident.

(i) Code System

The code-system -I(

) have been developed in 1982. The general flow of system is shown in Fig.8-10. This code-system has a coversational software which orders sequential calculation of wind field, concentration, and dose distribution in the local(25 km) and regional(100 km) scales, by mediating colour display system connected with the computer FACOM-M200 in JAERI.

Calculation results are recorded in the data pool system developed in JAERI [1] and displayed graphically with the map according to user's order, for example, wind field, concentration or dose distribution isopleth and countermeasure area. The example output of the system is shown in Fig.8-11. This shows concentration isopleth displayed on the administrative bounds map. User can select several kind of maps, administrative bounds, coast line and place name etc..

According to the development of the data pool system, site characteristic, meteorological and regional input data essencial to the calculation are recorded in it. The site characteristic and regional data of fourteen nuclear sites in Japan are prepared and the meteorological data collection system are investigated.

As a future plan, the improvement of the system will be continued and the method of the practical use with mega minicomputer will be researched. Those plans will be finished at 1984.

(ii) Wind Field and Concentration Calculation Models

In 1981, wind field calculation code, WIND-04, and atmospheric dispersion calculation code, COARA, have been developed for the realistic simulation. The code, WIND-04, calculates three dimensional mass-consistent wind field with variational method from the observational data. The code, COARA, is capable of simulating time-dependent distribution of airborne effluents with Particle-diffusion method. With time- and Space-varing wind field obtained from WIND-04, complex dispersion study can be undertaken. These codes are included in the code-system, with simple calculation code, GPCONC, based on Gaussian plume model.

The calculation results are compared with the field experiments performed in JAERI for the verification of codes.

As the other work, the translation of the famous MATHEW/ADPIC and their auxiliary computer codes has been carried out from CDC-7600 computer version to FACOM M-200's.[2)] The codes consist of a part of Atmospheric Release Advisory Capability(ARAC) system of Lawrence Livermore National Laboratory. These codes are used in order to compare with the other codes developed in JAERI.

As a future plan, the improvements and capability evaluation of physical models will be performed.

References

(1) M. TOMIYAMA, et al., "Datapool; Its Concept and Facilities", JAERI-M 8715 (1980) (in Japanese)

(2) K. TAKAHASHI, et al., "Translation of ARAC computer codes", JAERI-M 82-040 (1982) (in Japanese)

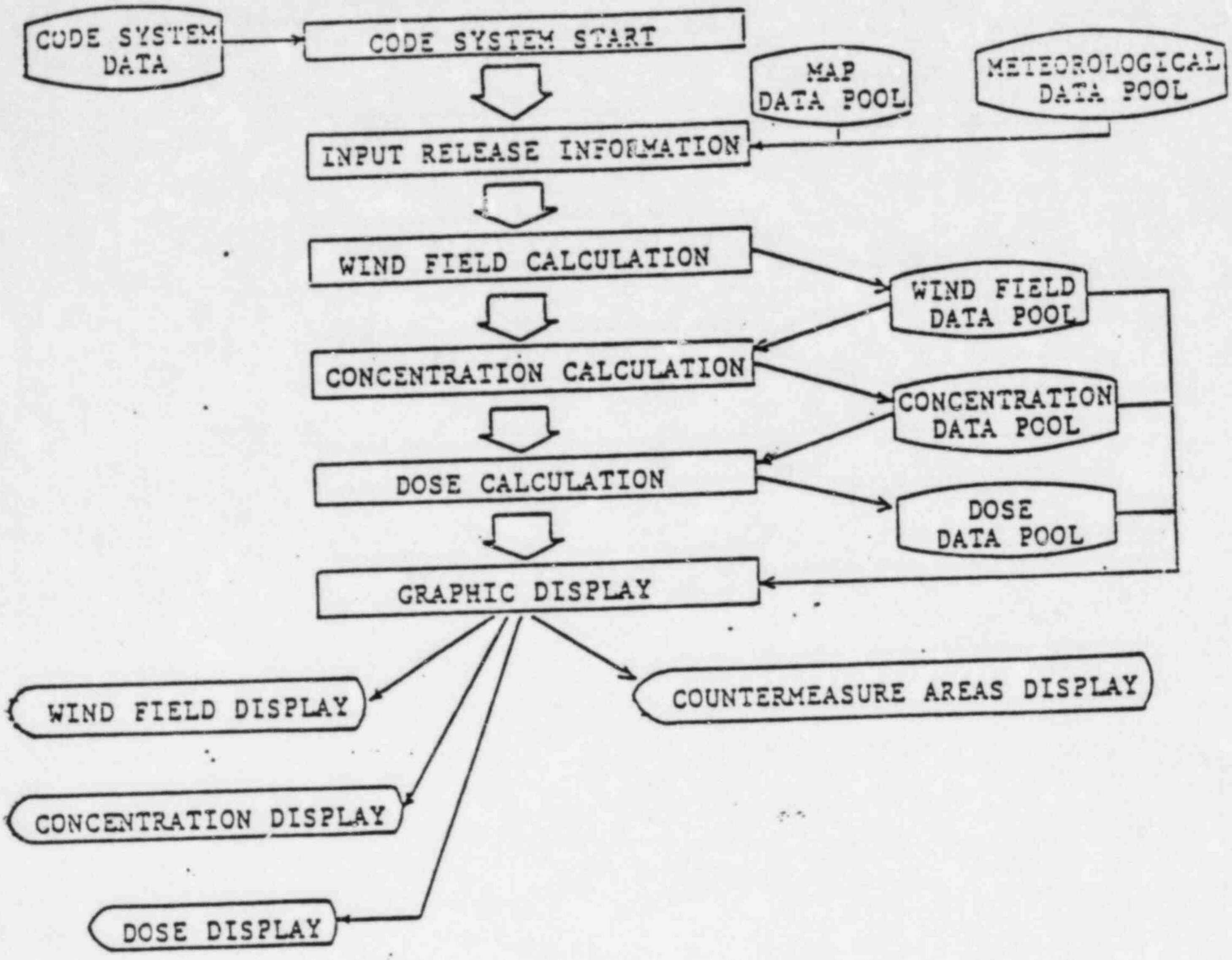

Fig. 8-10
~~FIG. 1~~ CONCEPTUAL DESIGN OF CODE SYSTEM

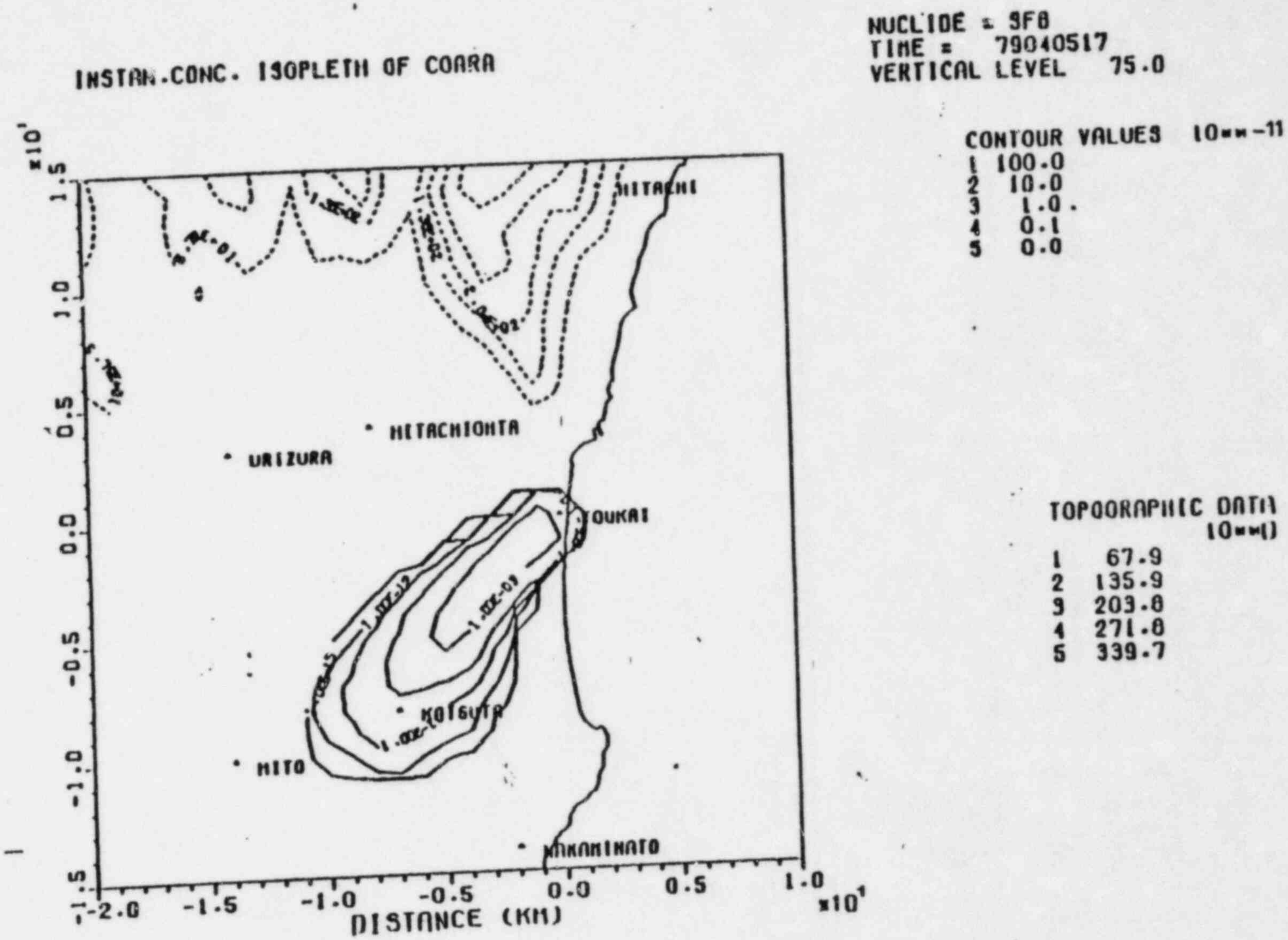

Fig. 8-11 The example of System output (Instantaneous Concentration Isopleth Ci/m³)

iii) Preparation for fast prediction of dispersion of radioactive materials

In order to meet the requirement of fast prediction of radiological consequences for the emergency planning, are needed some approximate estimations of the dispersion of radioactive materials released. For this purpose, two dispersion models, a puff model and a plume-segment model, which had been originally developed in USA were modefied for preparation in the system.

The puff model which was improved to be able to take into account the depletion of the materials in the atmosphere would be available for the fast estimation of the local dispersion up to about 20 kilometers from the release source. Real time meteorological data observed on the site are principally used for this model though the model was revised so that off-site data are also available.

For the plume-segment model which would be able to approximately calculate the regional dispersion up to about 200 kilometers, has been developed a subprogram to simply obtain distributions of the atmospheric stability and the wind field on the 2-dimensional grid points by interpolating the observed meteorological data.

iv) A subsystem for assessing environmental consequences

Since Japan Nuclear Safety Commission issued, in 1980, a guideline for the emergency preparedness in which action levels for evacuation, relocation and interdiction of ingestion were defined, local officials responsible for their emergency planning would perform appropriate countermeasures according to these action levels. For purpose to assist them for their action, this subsystem has been designed to calculate

a) the projected external doses to total body from the radioactive noble gases (cloudshine dose),

b) the projected doses to thyroids of adults and infants following inhalation of radioiodines in the atmosphere, and

c) the projected concentrations of the radioiodines on the ground and leafy vegetables.

These calculations are principally performed using concentrations of the above nuclides in air provided from another subsystem which predicts their dispersion on the basis of real time meteorological data. At the beginning of an accident, however, rough but quick prediction of the consequences would be required by the emergency operation center. For response to this requirement, the subsystem includes the calculation of the cloudshine dose based on the Gaussian plume model for the dispersion. The calculation is performed very quickly by interpolating the normalized doses stored in the system. Variation with time of the nuclide composition and effective gamma energy in the noble gases is taken into account.

Information of the iodine release is expected to be limited at the emergency situation. Presuming to get only the release amount of ^{131}I, the subsystem estimates the releases of the other iodines, considering change in their fractions with time.

(8) Field Experiment of Atmospheric Diffusion

A five year program of field diffusion experiment was examined and started in 1980, with the intention of obtaining the effective data to develope the practical models, which predict environmental concentration distribution of accidentally released radioactive materials. Because of difficulties in experimental method, the available field data had been scarce in vertical concentration distribution, so emphasis was laid on vertical diffusion experiment in the first two years. The first series of field experiment was conducted at Tokai site in 1980. An elevated line source was simulated using a helicopter which flys back and forth along a coast-line 150m above the sea surface. SF6 gas was used as an air tracer. At four stations of 0.7, 1.5, 2.6 and 3.7km inland from the shore-line, tethered balloons were set up to sample the air at seven altitudes. In spite of the experiment in autumn when the temperature difference between land and sea is minimum throughout a year, thermally induced boundary layer (TIBL) and its effect on concentration distribution were found.

In 1981, the experiment was conducted in mid-summer when TIBL develops markedly. A number of improvements were given to the tracer release technique, to the sampling net, and to the meteorological observation, making use of experience of preceding year. An off-shore station was added to abserve temperature profile over sea. TIBL with parabolic cross section was clearly abserved through temperature soundings. Concentration distribution in and above TIBL was measured. Data of 11 cases of vertical concentration distribution was accumulated, which is expected to contribute to the development of diffusion models for coastal site. The lay-out of the field experiment is shown in Fig. .

The future program of the field diffusion experiment is as follows. In 1982, emphasis is laid on horizontal diffusion. A continuous elevated point source is used. The sampling net is extended to 15km from source. In 1983, comprehensive diffusion experiment is intended, in which both vertical and horizontal distribution is measured to verify the prediction models developed by that time. In 1984, the last year of the program, field experiment at a complex terrain site will be supplemented to check the applicability of models to the complex site.