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(Eds.) Editorial Committee on Nuclear Safety Research Results

日本原子力研究所
Japan Atomic Energy Research Institute

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Progress of Nuclear Safety Research – 2001

(Eds.) Editorial Committee on Nuclear Safety Research Results※

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JAERI is conducting nuclear safety research primarily at the Nuclear Safety Research Center in close cooperation with the related departments in accordance with the Long Term Plan for Development and Utilization of Nuclear Energy or the Safety Research Annual Plan issued by the Japanese government. The safety research at JAERI concerns the engineering safety of nuclear power plants and nuclear fuel cycle facilities, and radioactive waste management as well as advanced technology for safety improvement or assessment. Also, JAERI has conducted international collaboration to share the information on common global issues of nuclear safety.

This report summarizes the nuclear safety research activities of JAERI from April 1999 through March 2001.

Keywords: Nuclear Safety, Nuclear Power Plant, Nuclear Fuel Cycle, Radioactive Waste Management, Experiment, Analysis

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原子力安全性研究の進捗状況（2001 年度版）

日本原子力研究所東海研究所
安全性研究成果編集委員会編※

(2001 年 9 月 4 日受理)

日本原子力研究所は、国の定める原子力エネルギー開発・利用に関する長期計画や安全研究年次計画に沿って、安全性試験研究センターを中心に関連部門との密接な連携のもとで、原子力安全性研究を実施している。研究内容には、原子炉施設及び燃料サイクル施設の工学的安全性研究、放射性廃棄物安全性研究、安全性向上及び評価に関する先進技術の研究等が含まれる。また、国際協力により世界共通の原子力安全課題に関する情報の共有を図っている。

本報告書は、1999 年 4 月から 2001 年 3 月までの期間に日本原子力研究所において実施された原子力安全性研究を要約したものである。

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1. INTRODUCTION

To ensure the safe development and utilization of nuclear energy, the Japan Atomic Energy Research Institute (JAERI) is conducting nuclear safety research in close cooperation with nuclear utilities, venders, universities, and governmental research organizations. The safety research at JAERI concerns the safety of nuclear power plants, nuclear fuel cycle facilities, and radioactive waste management. Continuous efforts have been made by JAERI to provide technically sound and reliable data and information to the public. Although JAERI is a governmental research institution, it takes neutral positions on issues pertaining to governmental regulatory bodies and the nuclear industry.

The scope of this safety research is based on the Long Term Plan for Development and Utilization of Nuclear Energy issued by the Atomic Energy Commission and the Five-Year Plan for Nuclear Safety Research issued by the Nuclear Safety Commission. The Five-Year Plan from 1996 to 2000 specified research items on human factors, severe accidents, probabilistic safety assessments, fuel reprocessing, aging of nuclear power plants, and reliability of digital control systems to be performed by government organizations to ensure the safety of nuclear facilities. These items have been selected because they are needed to develop safety criteria and to establish a technical basis for licensing procedures.

Nuclear safety research at JAERI is conducted primarily at the Nuclear Safety Research Center of the Tokai Research Establishment of JAERI. Various items of safety research are performed by different research divisions within JAERI to optimize the use of facilities and the expertise of JAERI personnel. In addition to administrative offices and research divisions, technical advisory committees are established by JAERI. Japanese experts are invited to offer advice and comments to maximize the utilization of the research results and to improve the quality of research. The organizations engaged in nuclear safety research at JAERI are shown in **Figure 1-1**.

This report summarizes the nuclear safety research activities in JAERI from April 1999 through March 2001.

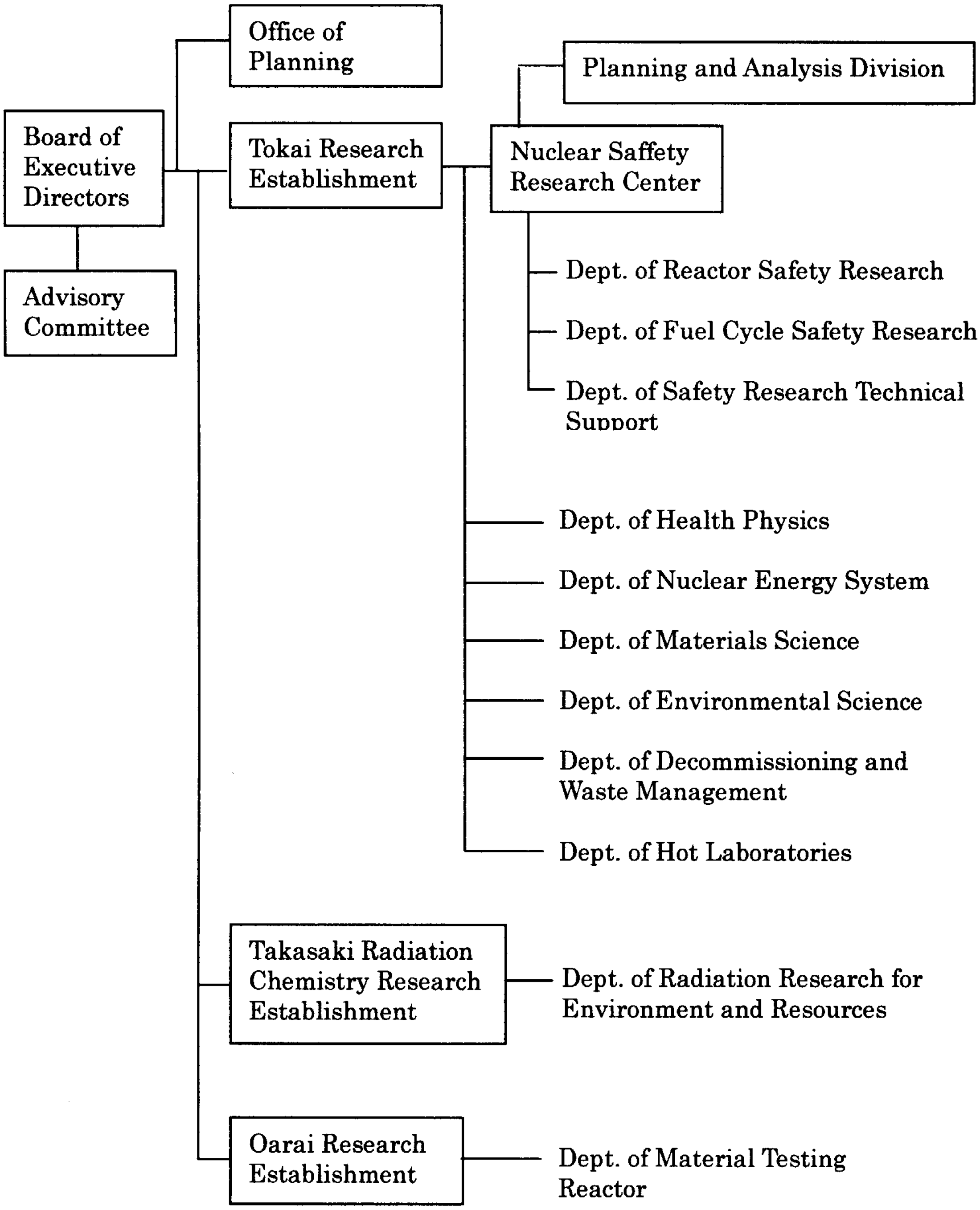

Fig. 1-1 Organizations engaged in nuclear safety research at JAERI

2. REACTOR SAFETY RESEARCH

Recent nuclear power generation in Japan shows good operation performance. The regulatory safety review, etc have been assuring the high safety of the nuclear power plants. According to the recent achievement and future demand in fuel burnup extension as well as MOX fuel utilization in light water reactors (LWRs) and long-term operation of the plants, importance of further reactor safety research has been still recognized by Nuclear Safety Commission, regulatory organizations and industry. The technical knowledge obtained from the research will be used for establishment or modification of the Japanese Nuclear Safety Evaluation Guidelines in timely manner.

The measures for preventing severe accident that brings catastrophic core damage have been of great importance since the TMI-2 accident in 1979. Through the comprehensive research and development over 20 years, accident management measures have been almost completed for the domestic LWRs. The proposed next-generation reactors in conceptional design are considered to adopt the equipments with passive safety feature. However, there are still the uncertainties in the evaluation of the external hazards such as earthquake, human factors and some phenomena during severe accidents. The JCO accident in Tokai-mura in 1999 has indicated that nuclear facility disaster countermeasures were not enough rationalized for an accident which influences the public and environment. When taking its serious impact on the public into consideration, continued efforts should be made to investigate the emergency countermeasures even if the occurrence of the severe accident is at quite a low level.

Deregulation in every field, in progress or impending, would bring changes in nuclear safety practice to achieve more rational regulatory strategy. “Probabilistic Safety Goals” and “Risk-Informed Regulation” have been widely recognized as beneficial ones for the purpose. To meet this mainstream, the efficient utilization of probabilistic safety assessment (PSA) methodology should be enhanced.

From the viewpoints described above, the activities of reactor safety research in JAERI include:

(1) Fuel behavior at high burnup under normal and accident condition
(2) Aging degradation and reliability of components and structures
(3) Thermal-hydraulic phenomena and accident management measures to prevent severe core damage
(4) Severe accident phenomena and accident management measures to mitigate such phenomena
(5) Safety assessment of nuclear power facilities

The outline of these activities and main results will be shown in the following chapters.

2.1 Fuel Safety Research

Fuel burnup extension and MOX fuel utilization in LWRs are being advanced in Japan for better economy of nuclear power generation and efficient utilization of uranium resources. In order to confirm the safety of the high burnup fuel and the MOX fuel as well as to obtain basic data important to the regulatory judgments, various experimental and analytical researches are carried out at JAERI on the fuel behaviors under normal operation, abnormal transient and accident conditions.

Fission product (FP) gas release, thermal property degradation, and rim structure formation become more significant in the fuel pellet as the burnup is extended. Cladding ductility is reduced by enhanced waterside corrosion, hydrogen absorption, and neutron irradiation. As for the MOX fuel, it is generally recognized that release rate of FP gas at high burnup is higher than that of the UO_2 fuel. For the purpose of examining the influences of these phenomena on the integrity of the fuel rod under normal operation and abnormal transient conditions, the fuel behavior is evaluated through post-irradiation examinations at the Reactor Fuel Examination Facility (RFEF), irradiation tests at the Japan Materials Testing Reactor (JMTR), and development of a fuel behavior analysis code FEMAXI.

A reactivity-initiated accident (RIA) is one of the postulated nuclear reactor accidents. It is characterized by an abnormal increase of the reactivity in the reactor core caused by control rod ejection and so on, which leads to a rapid reactor power increase. Integrity of the reactor fuel rods could be threatened by the mechanical and thermal loading due to the power burst and fuel temperature increase, with potential adverse effects to fuel rod integrity. Thus, fuel performance under the simulated RIA conditions has been studied extensively in the Nuclear Safety Research Reactor (NSRR) since 1975. The NSRR is a TRIGA-Annular Core Pulse Reactor, which can simulate a rapid power excursion of a peak power up to 23,000MW for several milliseconds. The test results with fresh fuels clarified the failure thresholds of fuel rod and the consequences of the failure under a wide range of conditions. These experimental data were utilized to establish the Japanese Nuclear Safety Evaluation Guideline for RIAs by the Nuclear Safety Commission in 1984. The NSRR project has extended its experimental capability to cover the tests with irradiated fuels in 1989. Short test fuel rods were re-fabricated from irradiated fuel rods from PWRs, BWRs and the proto-type Advanced Thermal Reactor (ATR) "Fugen". In addition, test fuel rods with higher uranium-235 enrichment of 10% and 20% were prepared and irradiated in the JMTR, in order to examine the performance of irradiated fuel at higher fuel enthalpies. Now, the research focuses on the behavior of the high burnup fuel rod. The project has covered the fuel burnups of 50GWd/tU for PWR fuels, 61GWd/tU for BWR fuels and 20GWd/tHM for ATR/MOX fuel by the end of FY 2000. The results obtained so far have revealed that the behavior of high burnup fuels under the RIA conditions is quite different from that of fresh fuels. Accumulation of fission products in the fuel pellets and the cladding embrittlement during the base irradiation largely influences the failure mode and its consequences. Thus, the behavior of the irradiated fuel differs depending strongly on the irradiation conditions, in addition to its initial design. For the expected fuel burnup extension, fuels with modified or new type cladding alloys would be used, which might show different irradiation behavior on the course of burnup accumulation. Their characteristics under the accident conditions should also be examined. In the MOX fuel, the locally higher Pu contents in a pellet, Pu spots, would cause locally higher thermal energy insertion under the RIA conditions. The density of accumulated fission products during the base irradiation is higher there. Consequently, this heterogeneity could cause the different fuel behavior under RIA conditions in comparison to UO_2 fuel. Therefore, it is essentially important to examine these high burnup effects and MOX effects under the RIA conditions at a wide range of fuel burunps covering the expected burnup level for the reactor safety.

A loss-of-coolant accident (LOCA) is another postulated accident. During a LOCA, it is generally estimated that the peak clad temperature would reach very high level between 1000 and 1400K, and the fuel cladding is exposed to high temperature steam for several minutes until emergency core cooling water quenches the fuel bundle. The fuel cladding is oxidized by steam and becomes brittle when severely oxidized. The embrittled cladding may fail by thermal shock caused by the quenching during a reflooding stage. Current LOCA criteria and knowledge on fuel behavior during a LOCA are mainly based on the results of experiments with un-irradiated claddings performed many years ago. During the reactor operation, fuel cladding is corroded, hydrogenated, and neutron irradiated, resulting in reduction of cladding ductility. In order to obtain the data evaluating the high burnup fuel behavior under a LOCA condition, systematic experimental and analytical studies are conducted on fuel cladding to evaluate oxidation kinetics, burst behavior and failure-bearing capability of oxidized cladding upon quenching.

The recent results from the studies on the fuel safety are briefly summarized below.

2.1.1 Research on the Evaluation of Fuel Behavior during Normal Operation and Abnormal Transients

In this study, re-irradiation test of spent fuel in the JMTR, PIE in hot laboratories, and development of an analysis code on the basis of the data obtained from the above tests and international collaborations which have been carried out for confirming the safety and integrity of high burnup fuel rods.

In the JMTR, re-irradiation tests have been performed on the fuel rod segments instrumented with center-line thermocouples and internal pressure sensor. The test results obtained from the gadolinia added fuel rods have shown that the thermal conductivity degradation was in good agreement to the value calculated by an empirical equation proposed by the Halden Reactor Project.

(1) Research on the thermal properties of high burnup fuel

Fuel burnup has been gradually extended to reduce the nuclear fuel cycle cost for LWRs. The thermal conductivity of fuel pellet is known to degrade with burnup due to the accumulation of solid FPs, introduction of lattice defects due to neutron irradiation and precipitation of small bubbles of fission gas in the matrix of pellet. The degradation of thermal conductivity increases the fuel temperature resulting in the increase of fission gas release as well as PCMI due to thermal expansion of pellet. Therefore, it is important to clarify the degree of degradation in order to evaluate the behavior of high burnup fuel. The thermal conductivity can be derived from the thermal diffusivity measurement data, however, the thermal diffusivity data on high burnup fuel are very limited. The thermal diffusivity of high burnup fuel was measured by means of laser flash method in the temperature range from room temperature up to 1800 K on the pellet samples irradiated for long period at the Halden Boiling Water Reactor (HBWR). An example of the results is shown in **Fig.2.1-1.** The thermal diffusivity of high burnup fuel is smaller than that of unirradiated fuel, especially at low temperature region. The thermal diffusivity gradually increased when the measurements were repeated with rising the maximum temperature. This increase can be attributed to the recovery of radiation defects(the disorder of crystal lattice due to irradiation). In the figure, the thermal diffusivity of simulated high burnup fuel (SIMFUEL) measured by Lucuta et al. is also depicted for comparison. In the SIMFUEL, main fission products corresponding to the burnup level, except for fission gases, are added to UO_2 to simulate the high burnup fuel. The thermal diffusivity of high burnup fuel is lower than that of the SIMFUEL at low temperature region. This difference may be attributed to the effects of radiation defects or fission gas bubbles, which are not included in the SIMFUEL. The difference of the sample density mostly accounted for the difference of thermal diffusivity among the samples.

The gadolinia added UO_2 fuel, which is used for the reduction of excess reactivity of fuel assembly at the first loading, has lower thermal conductivity than UO_2, and then gadolinia added fuel shows higher fuel center temperature than UO_2 at the same power level. It is known that the thermal conductivity decreases with burnup, as mentioned above. Therefore, it is important to clarify the thermal conductivity change of gadolinia added fuel with burnup to evaluate the behavior such as fission gas release and PCMI at high burnup. The thermal diffusivity of 6wt% gadolinia added fuel irradiated to 28 GWd/t at a commercial LWR was measured by means of laser flash method. The diffusivity of irradiated gadolinia added fuel was lower than that of unirradiated gadolinia added fuel, but the relative reduction of thermal diffusivity with burnup was smaller in the case of gadolinia fuel than that for UO_2

fuel. It is estimated that the effects of accumulation of soluble FPs and the accumulation of radiation defects on the thermal diffusivity were relatively reduced because the lattice of UO_2 is already disturbed to some extent by the addition of gadolinia.

(2) Development of computer codes for the analysis of fuel rod behavior in normal and abnormal transient conditions

The fuel rod behavior analysis code FEMAXI-IV, which was developed on the basis of the data obtained through the Halden Reactor Project has been released to the NUPEC (Nuclear Power Engineering Corporation) to be utilized as a cross-check tool for the safety licensing procedure. The next version FEMAXI-V was developed which is capable of analyzing the post-boiling transient of BWR rods and has models required to describe the fuel behavior of more than 50GWd/tU burnup. The source code, detailed description and input/output manual of the FEMAXI-V code have been released, and the code has been adopted in domestic and foreign organizations to be used as one of the tools for the fuel rod analysis. The FEMAXI-V code has been evaluated by the data which were obtained in the Halden Reactor Project and in the domestic commercial reactors. As a result, a reasonable agreement between measured data and calculations has been demonstrated for the fuel behavior from BOL to very high burnup region. Also, the evaluation of the code revealed that models are still to be developed to deal with pellet-clad bonding layer formation and the rim structure formation at the periphery of high burnup fuel pellet. Further improvements are in progress.

A neutronics burning analysis code PLUTON has been developed on the basis of the theoretical considerations. Evaluations of the code have resulted in a successful agreement between measured data and calculations. **Figure 2.1-2** shows a comparison of concentrations of generated Pu in the radial direction of pellet.

2.1.2 Research on fuel behavior under RIA conditions

Short fuel segments, which are irradiated in variety of reactors e.g. PWRs, BWRs, have been pulse irradiated in the Nuclear Safety Research Reactor (NSRR) to investigate fuel behavior under reactivity initiated accident (RIA) conditions. The test results have been used to establish the Japanese Safety Evaluation Guidelines for RIAs, which took account of the fuel burnup on the failure and its influence. In the last two years by the end of fiscal year 2000, seven pulse irradiation tests, making a total of 64 tests, with irradiated fuel at burnups up to 61GWd/tU have been conducted as illustrated in **Table 2.1-1**.

Characteristic cladding failure was observed in pulse irradiation tests of high burnup PWR fuel rods [1], [2]. The PWR fuel at burnup of 50GWd/tU failed with longitudinal cracks, at fuel enthalpies as low as 251J/g (60cal/g). The cracking of the Zircaloy-4 cladding occurred due to pellet cladding mechanical interaction (PCMI) in the early stage of transient before cladding temperature escalation occurred. Boiling water reactor fuel, on the other hand, generally showed smaller deformation than PWR and never indicated the hydride-assisted PCMI failure up to burnup of 56GWd/tU [(3), (4)]. In the recent study, higher burnup BWR fuel at 61GWd/tU with the current Japanese BWR fuel design, 8x8 type, from Fukushima Daini unit 2 was pulse irradiated in tests FK-6 through 9. Cladding failure occurred in three tests at peak fuel enthalpies of 62, 70 and 86cal/g for the first time in the BWR fuel tests [(5)]. It should be noted that, the rods were irradiated for 5 cycles with an extra cycle to have the highest burnup in the BWR fuels ever tested under the simulated RIA conditions. The appearance of the rod failed in Test FK-6 is shown in **Fig. 2.1-3**. Diagonal cracks were generated in the test fuel to separate the rod into three pieces, in addition to the longitudinal cracks which were common to the PWR fuel failure. Bonding of the fuel pellets to the cladding due to chemical interaction in the test fuel could enhance the axial loading and contribute to the characteristic fuel failure observed. The cross section of the BWR cladding failed in the test is compared with that of the PWR cladding in **Fig. 2.1-4**. The crack in the test FK-6 cladding seems to have propagated along hydride precipitates in most part of the outer half of the cladding in a zigzag manner. In the BWR cladding, hydrides look less localized at the outer surface and oriented more randomly than in the PWR cladding failed in test HBO-1 [(1)]. In the PWR cladding, brittle fracture started at the outer surface of the cladding where hydrides densely precipitated. Then the cracking propagated to the ductile inner region to cause the wall through opening. This failure mode is unique to high burnup fuel with extensive cladding embrittlement due to hydrogen absorption and oxidation in combination with neutron irradiation. Difference in the distribution of the hydride precipitates is likely due to higher treatment temperature for the BWR cladding. The BWR cladding was re-crystallized, while the one of PWR was stress-relieved. Even though the hydrogen concentration was smaller in the BWR cladding by a factor of about 2, the failure enthalpies in the BWR fuel tests are comparable with those of the PWR fuel due probably to the randomly oriented hydrides. Peak fuel enthalpies and enthalpy increase by the time of the failure in the NSRR tests with types of fuels are plotted in **Fig. 2.1-5** as a function of fuel burnup. In the figure, results of earlier US tests [(6)] and French CABRI tests [(7)] are plotted for comparison. A line of burnup dependent failure threshold by the PCMI failure, which is used in the safety evaluation guideline, is attached in the figure.

2.1.3 Study on high burnup fuel behavior under LOCA conditions

With a view to obtaining basic data to evaluate high burnup fuel rod behavior under a LOCA condition, a systematic research program is being conducted. The influences of corrosion, hydrogen absorption, radiation damage etc. during the reactor operation are mainly examined on oxidation kinetics, burst behavior and failure bearing capability of oxidized cladding upon quenching. Separate effects of pre-hydriding on the oxidation kinetics and the failure bearing capability have recently been examined with artificially hydrided cladding tubes. A computer code is also developed to analyze the fuel rod behaviors including the possible cladding failure upon quenching.

Low-tin Zircaloy-4 cladding tubes with hydrogen concentrations ranging 200 to 1600wtppm were subjected to the high temperature oxidation tests. The influence of pre-hydriding on the oxidation rate varied depending on oxidation temperature as well as hydrogen concentration. The largest influence was observed in the oxidation at 1223 and 1273 K. The weight gain ratio of pre-hydrided to as-received cladding, oxidized at 1273 K, is plotted as a function of hydrogen concentration in **Fig. 2.1-6**[8]. The figure shows that pre-hydriding generally enhances oxidation at this temperature and it is more remarkable at higher hydrogen concentrations. The present test result indicates about 9% increase of oxidation amount at maximum. Therefore, the enhancement of oxidation rate due to pre-hydriding is considered to be insignificant within postulated LOCA conditions and hydrogen concentrations in the high burnup fuel cladding.

The integral thermal shock tests simulate the LOCA sequences, namely heat-up, isothermal oxidation and quench by reflooding water, on segment test rods. The test rods are pre-pressurized to about 5MPa to rupture during the heat-up phase. Recent tests have been performed with pre-hydrided claddings to investigate the separate effect of pre-hydriding on failure-bearing capability on quenching, and under non-restraint and restraint conditions to investigate the effect of the possible axial loading which is generated by restraint of the axial cladding shrinkage on cooling. Failure maps obtained from the tests for both the as-received and the pre-hydrided cladding tubes are compared in **Fig. 2.1-7**[9]. The oxidation amount, ECR[1], of each data was calculated by the Baker-Just equation, and both the reduction in cladding wall thickness due to ballooning and double-sided oxidation after cladding rupture are taken into account in the calculation. The threshold value of ECR between failure and non-failure is about 60% for both the cladding tubes under no restraint condition. This indicates almost no influence of absorbed hydrogen on the threshold value. On the other hand, it is shown that restraining the shrinkage reduces the threshold ECR and the reduction is more remarkable for the pre-hydrided cladding. The restraint condition is possibly severer than actual loading conditions. Thus, the tests under controlled loading conditions are planned to perform.

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1 * ECR : Equivalent Cladding Reacted (Proportion of oxide layer thickness assuming that all of absorbed oxygen forms stoichiometric ZrO_2)

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Table 2.1-1 NSRR test matrix with irradiated fuel under RIA conditions

Test fuels	Fuel burnup (GWd/t) 10 20 30 40 50 60	Number of tests (FY1999-2000)
PWR		23 (2)
BWR		14 (4)
ATR/MOX		5 (1)
JMTR		22 (0)

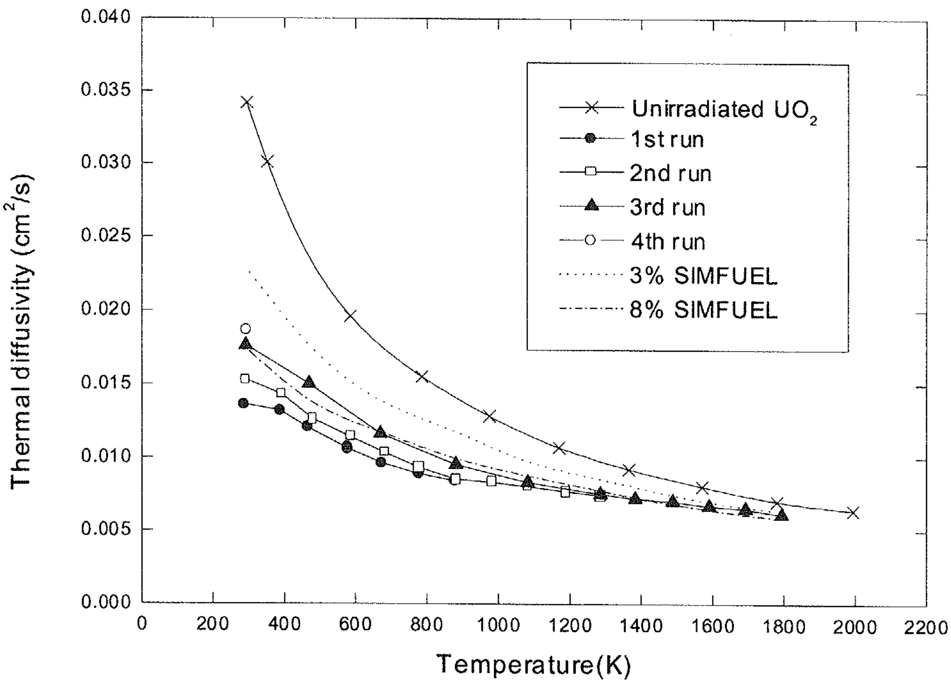

Fig. 2.1-1 Thermal diffusivity of high burnup UO_2 pellet (63MWd/kgU) irradiated at the HBWR.

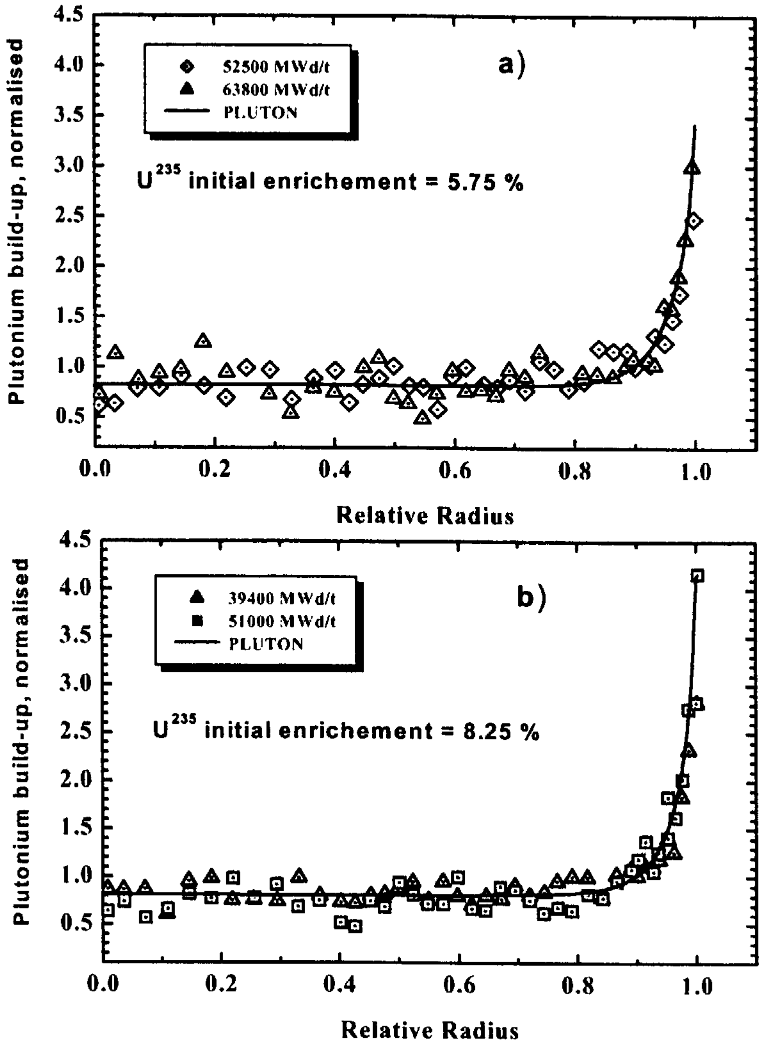

Fig. 2.1-2 Comparison of the radial total Pu concentration measured in different PWR-type fuels with that calculated by the PLUTON burnup model: cross-sectional burnup. a) 64000 MWd/t, b) 51 000 MWd/t.

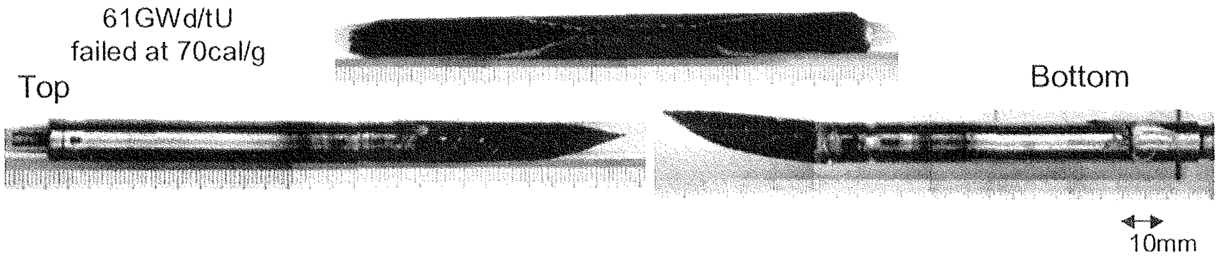

Fig. 2.1-3 Appearance of fuel rod failed in test FK-6.

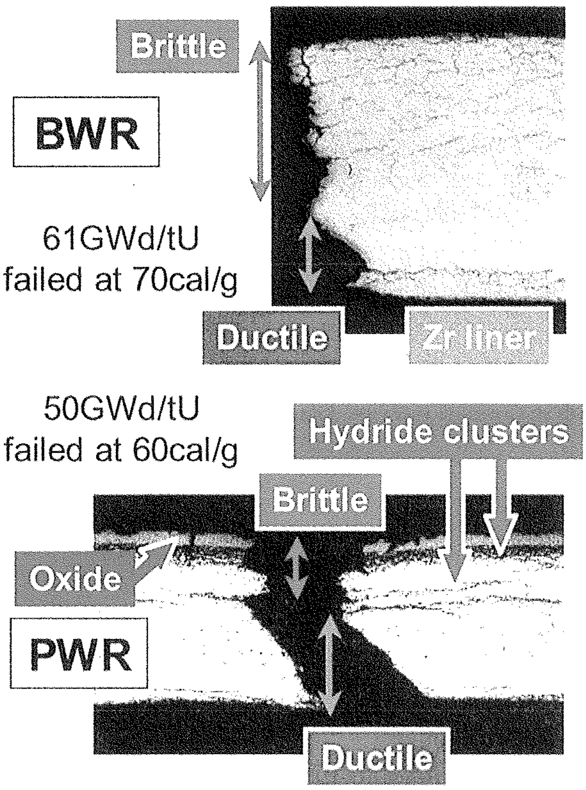

Fig. 2.1-4 Cross sections of BWR and PWR cladding failed in the NSRR/RIA tests.

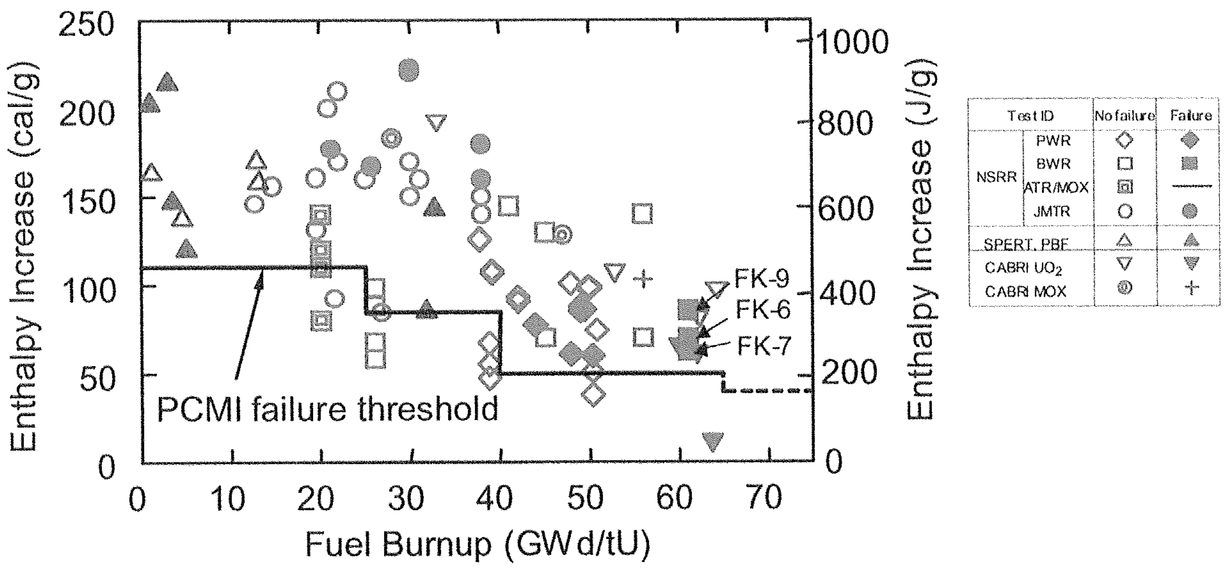

Fig. 2.1-5 Failure enthalpies in the NSRR tests as a function of fuel burnup in comparison with those observed in earlier US tests and French CABRI tests.

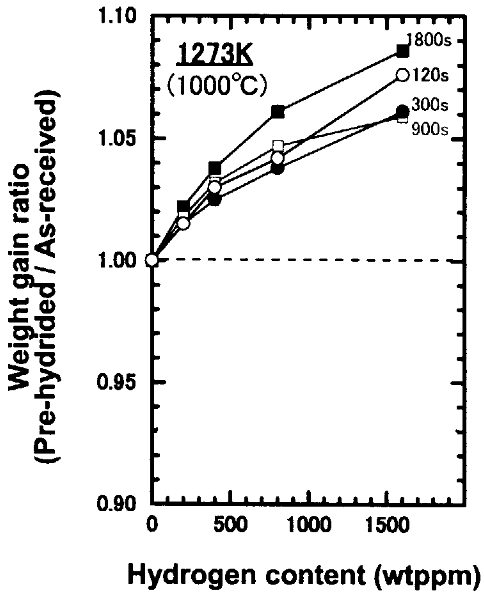

Fig. 2.1-6 Ratio of weight gain between pre-hydrided and as-received samples as a function of hydrogen content for the oxidation at 1273 K in steam.

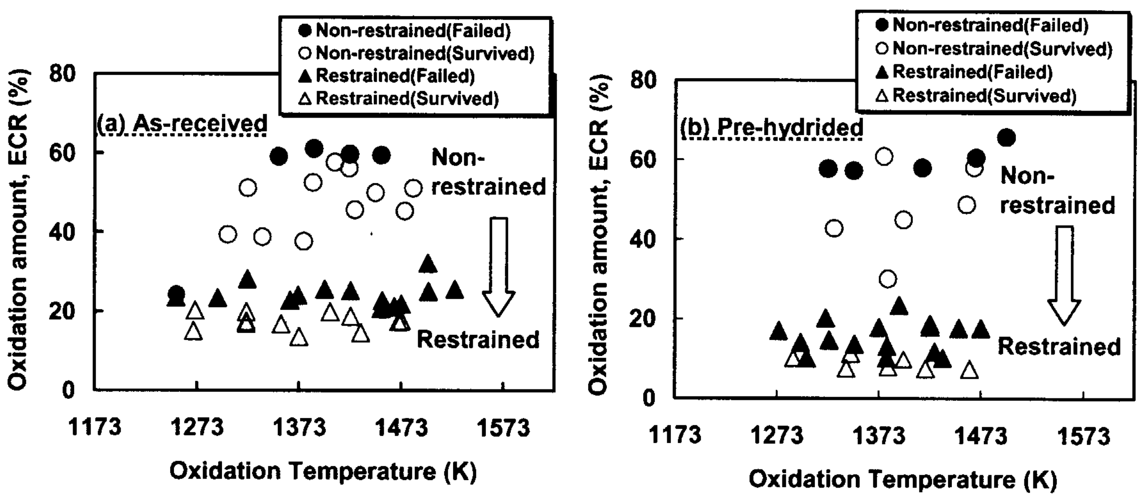

Fig.2.1-7 Failure maps relative to oxidation amount and oxidation temperature based on the test results under restraint condition; (a) as-received and (b) pre-hydrided cladding samples.

2.2 Aging Degradation and Reliability of Structures and Components

A few light water reactors (LWRs) in Japan began to operate in the early 1970's and have almost reached a 30-year operation. The LWRs, which over the years have proven themselves to be safe, have come to be considered a highly reliable type of reactor. The LWRs will be the mainstream of nuclear power generation for a considerably long period. It is, therefore, necessary for the aged reactors to further operate when many aging phenomena are readily manageable. The domestic countermeasures designated to cope with the aging of LWRs have been already taken expecting a 60-year operation. Particularly important is to maintain the integrity of the safety-related structures and components subjected to aging during long-term operation. Reactor vessel, internal core components, primary coolant piping, recirculation pump, electrical cables, containment structures and concrete structures have been chosen as safety-relates ones necessary for the devising of aging in the BWR plants in a report by Ministry of Economy, Trade and Industry (METI). Steam generators and pressurizer are included in the PWR plants. The relevant research and development have been comprehensively carried out among JAERI, national research organization and nuclear energy industry under the domestic coordination of METI.

At JAERI, the detection and evaluation of aging degradation under irradiation are investigated for structures and components which are important from the safety standpoint and difficult in replacing and repairing, e.g. reactor pressure vessel (RPV), internal core structures and electrical cables. We also plan to obtain such aging degradation data of the concrete structures.

For RPV, the ductile-to-brittle transition temperature (DBTT) should be lower than the operation temperature of the vessel to prevent the brittle fracture. In general, neutron exposure of the vessel steels reduces the ductility, resulting in the shift in DBTT to higher temperatures. This is called "irradiation embrittlement". To monitor this, surveillance capsules filled with standard specimens made of the material representative of the vessel steels have been installed in the plant in the vicinity of the core. They are periodically retrieved from the plant and tested. The results from Charpy impact test have been used for the estimation of fracture toughness of the vessel steels according to the present regulation of surveillance program, based on the assumption that the degree of irradiation embrittlement obtained from Charpy impact test is equivalent to that from fracture toughness test. Recently, many efforts have been made to develop fracture toughness testing method called "Master Curve" method and this has been accepted and introduced in regulations in EU and US. This method can determine the temperature dependence of fracture toughness using only 6 specimens. The research on this method to improve the accuracy in evaluating the irradiation embrittlement of the Japanese RPV has been conducted using JMTR in JAERI (1). Also necessary are the research on the effects of neutron flux, γ ray, phosphorus as impurity atom on irradiation embrittlement to precisely predict irradiation embrittlement in the long-term operation of RPV. The use of nondestructive evaluation method of irradiation embrittlement, which has been widely developed, will also contribute to the safety of RPV.

It is known that irradiation-induced stress corrosion cracking (IASCC) occurs in the austenite stainless steel for the internal core structures exposed to high fluence under tensile stress in high temperature water environment. Since cracking has been often observed in foreign aged LWRs, IASCC becomes one of the issues in domestic reactors. Understanding the mechanism(s) of IASCC, the controlling factors and expected crack growth rate are the area where we are concentrating on.

Probabilistic fracture mechanics (PFM) method has been recently highlighted to rationally incorporate the uncertainties arising from the materials and structures, unlike the conventional deterministic method. A failure probability is obtained through the PFM analysis by imputing the material strength, cracking size, load and so on as probabilistic distribution. We investigate how this PFM method is applied to the domestic regulation, in the meantime, develop a PFM code referring to a recent progress of fracture mechanics and computers. A PFM code PASCAL (PFM Analysis of Structural Components in Aged LWR) has been developed in JAERI and verified in some functions.

For other research related to the component integrity, one of important issues must be how to cope with earthquake. Recently, isolation technique has been applied to domestic and foreign nuclear plants to reduce a risk by earthquake. The isolation is classified into building, floor and components isolations. Among these, only components isolation is expected to apply to the existing components in the plants. A methodology and a computer code EBISA (Equipment Base Isolation System Analysis) for evaluating the effectiveness of the nuclear component with the base isolation were developed. A failure frequency of emergency transformer with base isolation, which was identified to be important by seismic probabilistic safety assessment, was evaluated by EBISA code.

The typical results of the researches described briefly described above in FY1999-2000 are shown in the following sections.

2.2.1 Examination on Irradiation Embrittlement of Reactor Pressure Vessels Steel by γ ray

The effect of γ -ray on the embrittlement has been examined to improve the accuracy of irradiation embrittlement prediction. Only fast neutron dose (E>1MeV) has been included in the present prediction of RPV steels where a ratio of γ ray to total displacement per atom (dpa) is several percentages. However, it has been pointed out that γ ray may effectively contribute to the irradiation embrittlement, because both a generation and survival rate of freely migrating defects (FMD) generated by simple atomic displacement via Compton scattering in γ ray irradiation would be higher than that by cascade in fast neutron irradiation at the same dpa. The FMD induces the Cu migration, resulting in the embrittlement of the steels. In the present study [2], Fe-Cu model alloys, which aim to extract the embrittling effect of Cu as an impurity atom in RPV steel, were irradiated by high-energy electron for simulating γ-rays irradiation and neutron in test reactor irradiation (JMTR), and irradiation-induced hardening as a measure of embrittlement was compared between these irradiations.

In **figure 2.2-1**, the electron or neutron-induced hardening in terms pf increase in Vickers hardness for the Fe-Cu model alloys with high and low Cu contents (0.6wt%, 0.02wt%) were plotted against dpa. The significant increase in hardening for Fe-0.6%Cu alloy was mainly attributed to Cu precipitation. The difference between electron and neutron irradiation hardening was found to be small on a per dpa basis. Although the result is contrary to what we expected as described above, practically dpa works well as a scaling parameter for the estimation of irradiation-induced embrittlement.

2.2.2 Nondestructive Evaluation of Irradiation Embrittlement

The magnetic interrogation method (MIM) has been proposed to measure nondestructively the degree of material degradation of RPV steel. This method relies on the fact that a change of magnetic coercivity due to neutron irradiation in the low-alloy ferritic steel of RPV can be correlated with material degradation. The measurements of coercivity, however, require the magnetization of inspection part to a level of magnetic saturation and this needs relatively high power electric supply. To solve this problem, an incremental permeability measurement method was newly proposed, by which the correlation between the magnetic property and the mechanical property can be measured and evaluated along with a hysteresis minor loop with relatively low-level magnetization. To investigate this method, we carried out magnetic measurement with a magnetic yoke and planar coils for evaluating changes of material properties of ferromagnetic structural steels [3].

Experimental setup is shown in **Figure 2.2-2**. The materials used in the present experiment were A533B steel which is a low-alloy steel used for RPV and of TYPE SUS410 which is a martensitic stainless steel. Test plates of the steels with different material properties caused by heat treatments were prepared. The planer coil has two windings; one is the primary for excitation and another the secondary for induction of output voltage. The planer coil was placed on a test plate with a magnetic yoke which induced a bias DC magnetic field, and excited with a small constant current of 25 Hz. Then the output voltages were measured with changing slowly the bias field by excitation of the magnetic yoke with a triangular wave current of 0.005 Hz. Output voltage characteristic of the secondary planar coil is shown in **Figure 2.2-3**. Hysteresis curve in the figure shows a butterfly shape. This characteristic reflects incremental permeability of a whole magnetic circuit, and can be considered as a reflection of the examined test steel provided that incremental permeability of the magnetic yoke is enough larger than that of the examined steel. The value of W, which is the half distance between two minimums in the figure, is considered as a reflection of coercivity of the examined steel. **Figure 2.2-4** shows a good relation between W and coercivity, Hc. In the same way, there is a good relation between W and Vickers hardness, Hv. Changes of output voltages with different test plates were correlated with their mechanical and magnetic properties. This indicates that the measuring method adopted in the present experiment could be applied for nondestructive evaluation of material degradation in RPV steels.

2.2.3 Development of Probabilistic Fracture Mechanics Code

Probabilistic Fracture Mechanics (PFM) has been used in the fields of reliability analysis, life extension assessment and risk management for important structural components as a promising and rational evaluation methodology. The probabilistic approaches are also being introduced into regulations and standards related to structural integrity such as Pressurized Thermal Shock (PTS), Leak Before Break (LBB) and etc in USA.

In JAERI, the PFM analysis code named PASCAL (PFM Analysis of Structural Components in Aging LWR) has been developed. This code can evaluate the conditional probability of crack initiation and failure of a pressure vessel under transient conditions such as a PTS and possesses many improved input functions in relative to those in existing codes. The probabilistic simulation methods used in this code are the importance sampling Monte Carlo (IMC) method for an infinite length surface crack and the stratified sampling Monte Carlo (SMC) method for a semi-elliptical surface crack. The initial crack size, chemical composition, neutron fluence, fracture toughness and ductile to brittle transition temperature are all treated as probabilistic variables.

In addition to the basic functions for PFM analysis, PASCAL has many original functions such as the fracture criterion of R6 method, the initial semi-elliptical surface crack with distributions of both depth and aspect ratio, the model to evaluate the decrease of upper shells fracture toughness, the model to evaluate the effect of thermal annealing, the equations of stress intensity factor, the models of semi-elliptical surface crack extension, and the procedures for optimized sampling. The accuracy and reliability of this code have been verified in some early analyses [4-8]. The results obtained for an international benchmarking problem are illustrated in **Figure 2.2-5**. In addition to these functions, some improvements in the reliability and efficiency of PFM analysis such as stress intensity factor solutions and computational techniques were performed. The conclusions are summarized below.

- For the stress intensity factor of an infinite length surface crack, solutions have been based on FEM analysis for both longitudinal and circumferential cracks. These solutions are accurate and can be used up to 95% of vessel wall. For the stress intensity factor of a semi-elliptical surface crack, the solution given by Newman and

Raju can be used even for a stress distribution up to a third order polynomial by a method using only a partial stress distribution in the cracked area and converting it into a linear distribution.

- For a RPV with overlay cladding, a convenient method has been proposed to calculate the stress intensity factor with sufficient accuracy for both infinite and semi-elliptical surface cracks in both longitudinal and circumferential directions.
- An optimum sampling and cell-dividing procedure is proposed for the SMC in the probabilistic simulation, and both the accuracy and efficiency can be improved by using this procedure.

Effort for improving the reliability and efficiency of the PFM analysis is being continued towards establishing a utility tool for integrity evaluation of aged LWR components. Some analysis results for practical problems will be carried out in the future.

2.2.4 Verification and Improvement of Evaluation Methodology of Component Base Isolation

The application of the base isolation technique to the seismic design of nuclear power plant components as well as buildings has been expected as an effective countermeasure to reduce the seismic force in components. A methodology and a computer code EBISA, for evaluating the effectiveness of the seismic base isolation of nuclear component, were developed. The code has been verified and updated by two kinds of base isolation test systems [9].

The ball bearings and air springs were installed on the test bed to observe the dynamic response under the natural earthquake motion (**Figure 2.2-6**). The behavior of base isolation system has been observed under several earthquakes. The time history data of acceleration and displacement in the earthquake occurred on March 26, 1999 were obtained. The maximum vertical acceleration of the test system reduced to 40% of the input acceleration. Also, three-dimensional response and base isolation of another system using multi-layer-rubber-bearings and coil springs have been investigated against various large earthquake motions by shaking table tests (**Figure 2.2-7**) . It was demonstrated that the vertical maximum accelerations response reduced to about 70% of input accelerations by oil dampers and countermeasures to reduce the rocking motion.

The data for improving the vibration models of EBISA code were also obtained through these observations and shaking table tests. The accuracy of vibration models was verified by simulating the observation and shaking table test results.

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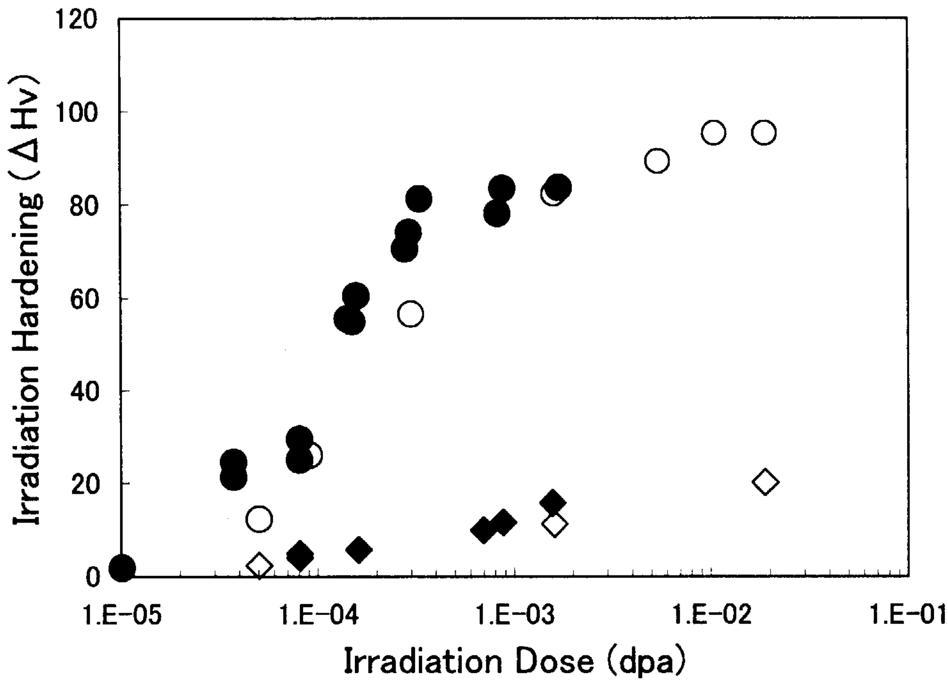


Fig. 2.2-1 Comparison of hardening by electron and neutron irradiation at 250℃.

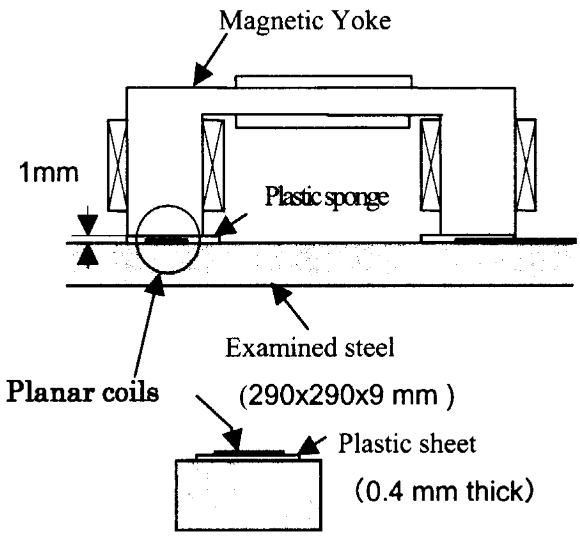


Fig. 2.2-2 Experimental set-up for nondestructive measurement of incremental permeability.

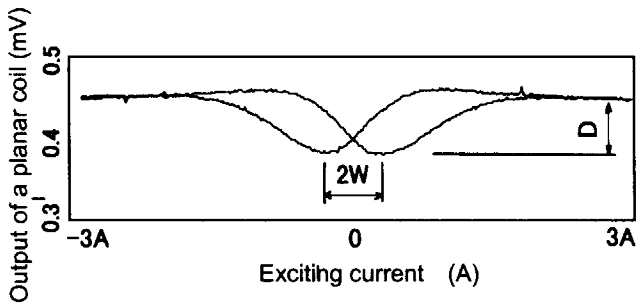


Fig. 2.2-3 X-Y trace curve of an output voltage of planar coil along a triangular excitation current of ±3 A for an A533B steel plate.

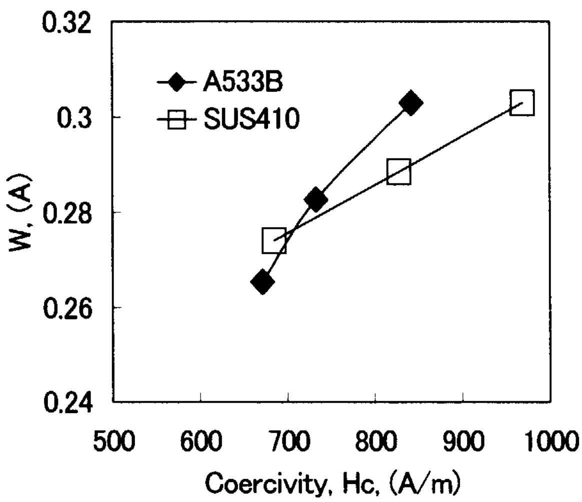


Fig. 2.2-4 Correlation between the peak position W of X-Y trace curve of planar coil output and the coercivity Hc.

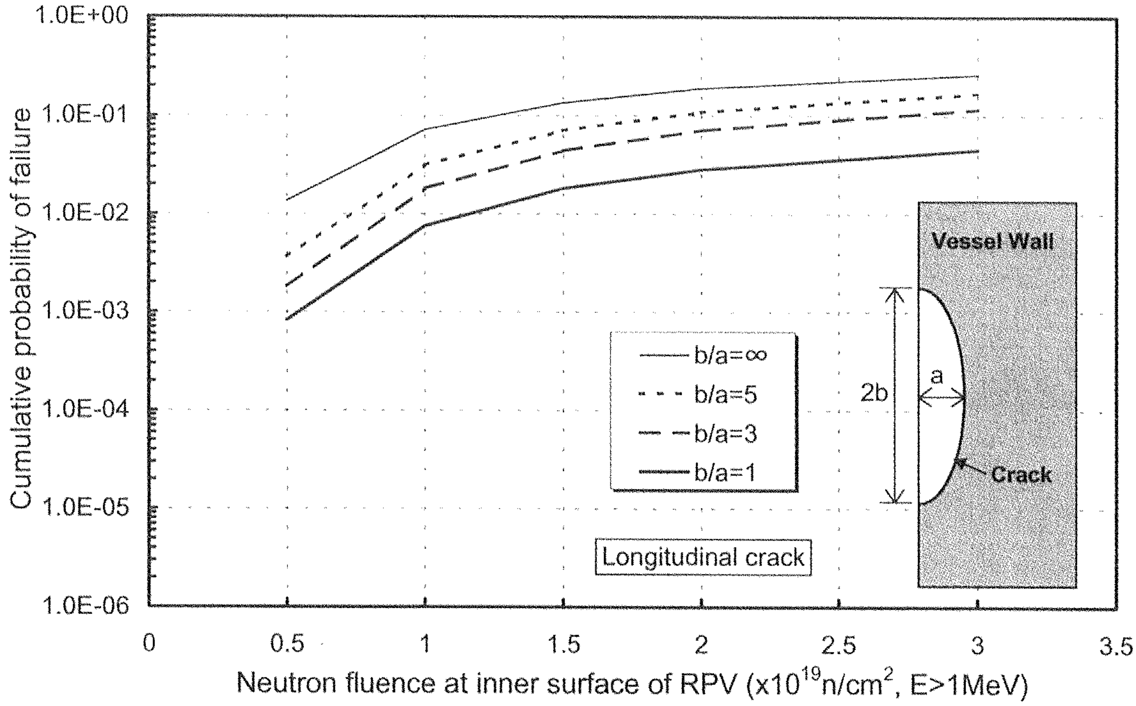
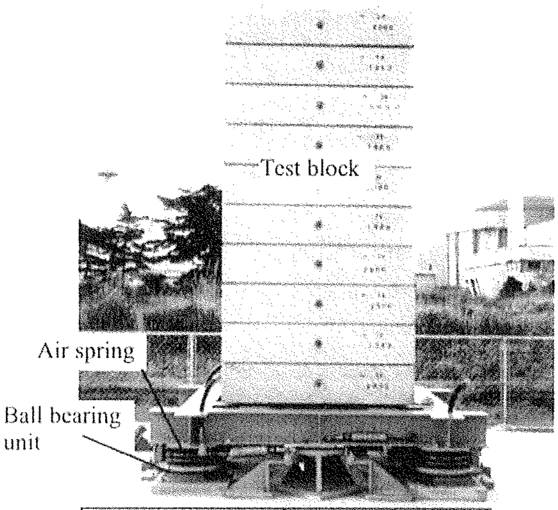
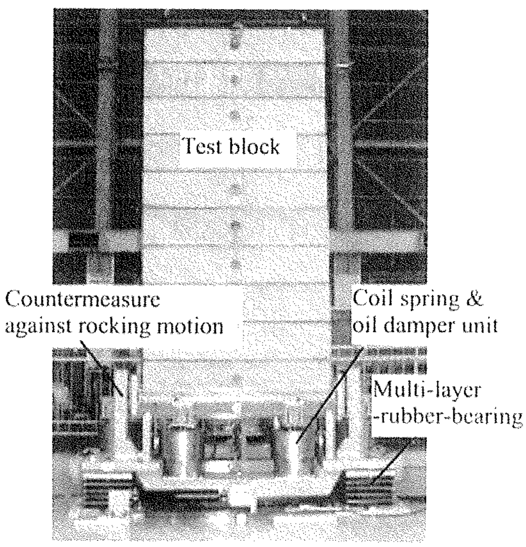


Fig. 2.2-5 Failure probabilities for different crack geometries under an international benchmark problem. Initial crack geometries were varied in terms of the aspect ratio (b/a) for longitudinal semi-elliptical crack at the inside surface of reactor pressure vessel wall.



	Horizontal	Vertical
Natural frequency (Hz)	0.33	1.0
Damping factor	0.17	0.05

Fig. 2.2-6 Three-dimensional base isolation test system using ball bearings and air springs installed on the field



	Horizontal	Vertical
Natural frequency (Hz)	0.4	1.2
Damping factor	0.23	0.35

Fig. 2.2-7 Three-dimensional base isolation test system using multi-rubber-bearings and coil springs

2.3 Reactor Thermal Hydraulics Safety Research

Based on a lesson from the Three-Mile Island Unit-2 (TMI-2) accident in 1979, accident management (AM) was proposed to perform in all the plants in Japan as a short-term countermeasure to prevent severe accident (SA) although the probability of SA to occur is extremely low. AM measures are categorized into two phases; Phase-I AM to provide counter-measures to prevent SA from occurring in case that safety features are in failure during an accident, Phase-II AM to mitigate the influences of SA. In JAERI, effectiveness of the Phase-I AM measures has been confirmed yet critical operations have been clarified in ROSA-V program using the Large Scale Test Facility (LSTF) the reactor-size PWR simulator being combined with parametric computer code analyses. The obtained results have been referred to by utilities as useful information for the consolidation of AM measures into plant operations during accidents. Furthermore, effectiveness of passive safety features for next-generation reactors is under investigation as a long-term scope to incorporate the AM measures/concepts into the reactor design for both PWR and BWR.

The ROSA-AP600 program final report was issued in March 2001 for LSTF confirmatory tests of a next-generation reactor AP600 design conducted under the ROSA-AP600 program agreement between JAERI and USNRC. The USNRC has extensively utilized the data to perform its review of the AP600 design characterized by passive safety features. Since vendors and utilities in Japan are planning to apply several passive safety features to next-generation reactors such as APWR+ and ABWR-II, wide ranges of passive safety concept are under investigation in ROSA-V program to timely provide analyses models and codes to confirm and improve predictive capabilities of safety analyses code.

Other than the ROSA-V program, BWR nuclear-thermal-hydraulic coupling experiment has been started in 1999 to clarify the mechanisms and influences of the flow and power instability in BWR core. Research on BWR nuclear-thermal-hydraulic coupling is underway by using THYNC loop furnished with innovative fast-response void meters to improve safety analyses methods against recent new design such as MOX fuel and high-burnup fuel, since safety margins of BWR core for stable operation would decrease. Developed three-dimensional, nuclear-thermal-hydraulic behavior code TRAC/SCKETCH has been utilized by MITI for safety analyses of the full-MOX Oma reactor.

ROSA-V/LSTF experiments have been performed to investigate accident management (AM) measures to prevent severe accidents and passive safety features for next-generation reactors. Furthermore, BWR nuclear-thermal-hydraulic coupling experiment has been started in 1999 to clarify the mechanisms and influences of the flow and power instability in BWR core.

2.3.1 Passive safety systems for the next generation PWRs

Several proposed designs for the next-generation reactors utilize passive components for their safety systems in order to simplify the systems and improve the reliability. To assess and optimize such passive safety system designs, their performance should be evaluated against various accident scenarios. The current computer codes for doing this job, however, may have large uncertainties in predicting phenomena occurring in passive systems, because the codes are not generally validated for the phenomena occurring in new passive components and operations, and those occurring under the thermal-hydraulic conditions that have not been expected to occur in the current-generation PWRs.

In order to improve the ability for the codes to evaluate the performance of passive safety systems for next generation PWRs, a thermal-hydraulic research program is being conducted using the ROSA/LSTF facility. **Table 2.3-1** shows the tests conducted in 1999 and 2000 for this project, which include tests investigating the primary depressurization system using the steam generator (SG) secondary side depressurization and the natural circulation (NC) SG heat transfer at low pressure for a long-term decay heat removal. Those safety systems using SGs are to be used in a next-generation PWR called APWR+ being developed by Mitsubishi. In the followings, results of a low-pressure natural circulation (NC) test are summarized. Since the NC data taken at nearly atmospheric pressure using a PWR simulator was almost non-existent, several new findings were obtained[1].

The test consists of several quasi-steady-state natural circulations at a different primary mass inventory, and a constant core power of 0.936 MW and SG secondary pressure of 0.14 MPa. This core power corresponds to 1.3% of the nominal power, which is equivalent to the decay heat level at 5000 seconds after scram. Primary mass was stepwisely discharged by approximately 5% from the pressure vessel bottom.

The NC flow behavior at high-pressure has been investigated at several PWR simulation test facilities, and is generally recognized that there are three heat transfer modes of single-phase NC, two-phase NC and reflux condensation, depending on the primary mass inventory. Similar behavior was observed in this test as shown in **Figure 2.3-1**. With decreasing the mass inventory, the NC flow rate first increased, peaked at ~90%, and decreased. The reflux condensation was observed for the mass inventory below ~40%. During the reflux condensation, the vapor produced in the core flows into the SG U-tubes, condenses in there, and returns to the core through both hot and cold legs.

During the two-phase NC, two flow behaviors were simultaneously observed among U-tubes: "two-phase condensing flow" and "two-phase stratification"(see **Fig.2.3-2**). The two-phase stratification was never observed in high pressure tests previously conducted at several PWR simulators and was caused by the SG secondary side temperature distribution. The typical temperature profile shown in **Figure.2.3-3** indicates that the temperatures in the upper region are saturated corresponding to the pressure at each elevation, peak at ~3m, and are subcooled in the lower part. The fluid is subcooled because of the circulation in the SG secondary side. That is, the liquid saturated at the separator flows into the downcomer, and enters the bottom of the boiling region. This temperature distribution causes condensation to occur in the top part and boiling to occur in the middle part of the U-tube so that a nearly-steady state vapor mass balance is achieved in the U-tube with the vertical two-phase stratification. This means that the net heat transfer does not occur through the U-tubes with the flow stagnation. The heat transfer occurs only through the U-tubes with the two-phase condensing flow. That is, the non-uniform behavior reduces the effective heat transfer area in the SG, and thus, cannot be negligible for the analysis of the low pressure NC. Since the model that can take into account the non-uniform flow behavior is not available for the current thermal-hydraulic analysis codes, the loop flow rate calculated by the RELAP5/MOD3 code was unrealistically oscillatory as shown in **Figure 2.3-4**. The model to deal with the non-uniform behavior is now being developed for the use in the one-dimensional thermal-hydraulic codes.

2.3.2 Passive containment cooling system for next generation BWR

A passive containment cooling system (PCCS) is under planning to use in a next-generation-type BWR for long-term cooling by condensing steam in heat exchangers especially during a severe accident when a containment spray system is in failure. **Figure 2.3-5** shows the schematic of a PCCS. Steam and non-condensable gas generated in lower drywell are driven to the PCCS heat exchanger by the pressure difference between the drywell and the wetwell, which is limited by the water head difference ΔP between the main steam vent line and the end of the PCCS drain line.

Japan Atomic Energy Research Institute (JAERI) and The Japan Atomic Power Company (JAPC) started a cooperative study to verify the performance of the PCCS using horizontal heat exchanger in 1998. Fundamental heat exchanger performance under postulated severe accident conditions was studied using a horizontal single U-tube test facility, which was constructed in the Large Scale Test Facility (LSTF). **Figure 2.3-6** shows the schematic of the facility. The test section was composed of horizontally leveled tube segments with a co-axial cylindrical cooling jacket. The tube outer diameter was 31.8 mm, and the total heat exchange length was 8 m. Saturated steam being mixed with non-condensable gas was provided to the tube primary via the test section inlet tank, and subcooled water (0.3 MPa, < 60 °C) was provided to the cooling.

Steam was condensed in ~6 m from the tube inlet under the postulated conditions; 0.7 MPa, 1% decay power, 1% Nitrogen gas partial pressure and U-bend inclination of –5 degree. The inlet gas velocity was ~19 m/s. **Figure 2.3-7** shows the primary and secondary fluid temperature and the tube outer surface temperature during this experiment.

Figure 2.3-8 compares the obtained inner wall condensation heat transfer coefficient with correlations by Dobson & Chato and Shah for annular flows and Chato for horizontal stratified flows at a pure steam inlet condition. Dobson-Chato correlation predicted well the data obtained under relative high quality conditions though some data was far higher than the predicted value probably because of the condensation enhancement by roll waves, which would agitate the liquid film. Shah correlation, on the other hand, greatly underpredicted the data. At the low quality region, the Chato correlation predicted well the data.

Studied phenomena included condensation heat transfer in the horizontal tube being degraded by non-condensable. **Figure 2.3-9** shows the gross heat removal of the condenser tube versus non-condensable gas partial pressure gas at the total pressure of 0.4 and 0.7 MPa by comparing the experimental results with the vertical condenser tube. The vertical tube data was taken in a saturated water pool under atmospheric pressure as the secondary condition. For the horizontal tube, the degradation was insignificant; by ~11 % for 0.7 MPa cases even at a high gas concentration of 20 %. The degradation was slightly enhanced as the total pressure decreased. As for the vertical case at 0.3 MPa, on the other hand, the degradation was significant; by ~40 % at the gas concentration of 10% primary because of small inlet gas velocity of ~3 m/s.

2.3.3 Coupled nuclear thermal hydraulic instability in BWR *

(1) Development of Thermal-Hydraulic and Neutronic Coupling loop (THYNC)[(2),(3)]

Experimental facility for coupled nuclear thermal hydraulic instability was developed. This is the first non-nuclear experimental facility capable to simulate thermal-hydraulic and neutronic coupling behavior in a BWR core. This experimental facility consists of three parallel 2x2-bundle test sections and one bypass 4x4-bundle test section, as shown in **Figure 2.3-10**.

(2) Development of void fraction meter[(4),(5)]

In order to perform thermal-hydraulic and neutronic coupling experiment, it is necessary to measure area-averaged void fraction in a rod bundle under high-temperature and high-pressure conditions with high-response and high accuracy. Hence, we developed a high-response conductance-type void fraction meter.

Figure 2.3-11 shows a result of calibration tests with steam-water two-phase flow in a rod bundle. The tests were performed under high-temperature and high-pressure conditions (7MPa, 290 deg. C). Void fraction measured with the developed void fraction meter (Y-axis) is compared to the void fraction converted from DP measurement (X-axis), which is obtained by neglecting frictional pressure loss. **Figure 2.3-11** shows both void fractions agrees well with

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* This work was carried out by JAERI under contract with the Ministry of Education, Culture Sports, Science and Technology of Japan.

each other, and indicates that the accuracy of the present void fraction meter is within 10%.

(3) Development of heater with dynamically-controlled axial power shape

Heater, which axial power shape can be dynamically controlled (bottom-peak, mid-peak, and top-peak), was developed. The axial power shape of nuclear fuel dynamically changes due to void feed back and Doppler effects. The effect of the dynamically-changing axial power shape on thermal hydraulics, specifically CHF etc. is investigated by using this heater.

(4) Channel stability experiment[3]

Thermal-hydraulic stability experiments were performed. Effect of channel number on channel instability threshold was investigated. The ratio of hydraulic resistance of channel outlet to channel inlet under single-phase flow Kout/Kin was set to be 2.6, which is slightly larger (or less stable condition) than the ratio of typical BWRs. Experimental result is shown in **Figure 2.3-12**. When the channel power was increased, channel instability occurred. Channel instability threshold was defined as the power when the standard deviation of the amplitude of flow oscillation became significantly large. When the channel power was increased further, "intermittent boiling transition" (Repetition of boiling transition (BT) and rewetting) took place. When the channel power was increased further, "Continuous boiling transition" (No rewetting) took place. This power was defined as BT threshold. The figure indicates that BT threshold under flow oscillation is much lower than that under constant flow rate. Channel instability and BT thresholds with two- and three-channels are slightly lower than those with one channel, though the difference is less than 15% of that with one channel.

(5) Thermal-hydraulic and neutronic coupling experiment[3],[6],[7]

Figure 2.3-13 shows a typical result of thermal-hydraulic and neutronic coupling experiment. This experiment was performed at 7MPa and 380kg/m^{2}s of channel mass flux (Bypass mass flow rate is about 50 times of channel mass flow.). The external reactivity was inserted at constant rate. Hence, the channel power increases. When the channel power exceeds 17kW/rod, channel mass flux starts to oscillate. After that, stationary oscillation (power --> void fraction --> flow rate--> power) is established.

It is known from the neutronic theory that the channel power is dependent on the initial channel power and the inserted external reactivity, as follows. If the external reactivity $\Delta\rho_{ext}$ is inserted stepwise, the channel power increases and then void fraction increases. Hence, the negative void reactivity $\Delta\rho_{void}$ is added. The channel power increases until the balance between inserted external reactivity and negative void reactivity is established, i.e. $\Delta\rho_{ext}+\Delta\rho_{void}=0$. Thus, the final channel power Q is dependent on $\Delta\rho_{ext}$. **Figure 2.3-14** shows that Q measured at the present experiment is almost equal to Q calculated with the neutronic theory. This indicates that thermal-hydraulic and neutronic coupling in is properly simulated in the present experiment.

Present experiment uses electrical heaters instead of nuclear fuels. Both have different thermal time constant τ, which should affect thermal-hydraulic and neutronic coupling behavior. In order to simulate heat conduction in nuclear fuels properly, additional thermal time constant τ_2 was incorporated in power control program in the experiment. **Figure 2.3-15** shows power transient under several additional time constant τ_2. This result was obtained under stepwise external reactivity insertion (40 cents). Due to stepwise reactivity insertion, channel power increases and finally becomes constant. The additional time constant τ_2 does not influence the final channel power, but it influences the channel power increase rate. The increase rate is higher and the thermal-hydraulics is more unstable with lower τ_2. In the figure, calculation result with TRAC code is compared. Calculated power increase rate and final channel power are almost equal to the experimental results, while amplitude of channel power oscillation is not calculated well.

(6) Code development

Three-dimensional thermal-hydraulic and neutronic coupling code TRAC-BF1/SKETCH-N[(7)], which has capabilities of fast-running and high-accuracy calculations, has been developed. Developmental assessment analyses of the code were performed so far using OECD/CSNI/NEA standard problems including BWR cold water injection benchmark[(8)] and Ringhals-1 stability experiment benchmark[(9),(10)], respectively. Further code assessment and improvement are now underway based on the THYNC experiment data.

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Table 2.3-1 Summary of LSTF tests conducted in 1999 to 2000

Run I.D.		Test Contents
PS-SD-07	April/99	Cold leg 2 inch break, Primary depressurization using SG secondary depressurization, Gravity-driven injection, Flashing-driven injection
ST-NC-21	May/99	Low pressure steady state reflux condensation, Effects of air accumulation in SG
TR-RH-07C	May/99	Residual heat removal (RHR) system failure, Gravity-driven injection, PV vent
PCCS tests	From June/99 To Jan./00	Passive containment cooling system using horizontal U-tubes
SB-HL-15	Feb./00	Accident management for a hot leg break, Effects of nitrogen gas accumulation
ST-NC-22	March/00	Low pressure steady state reflux condensation, Effects of air accumulation in SG
ST-NC-23	June/00	Low pressure steady state two-phase natural circulation, Effects of mass inventory in primary loop
PS-PD-01	Sep./00	Cold leg 3 inch break, Depressurization using primary ADS, Passive safety components (PRHR, CMT, ACC)
ST-NC-24	Nov./00	Low pressure steady state reflux condensation, Effects of air accumulation in SG

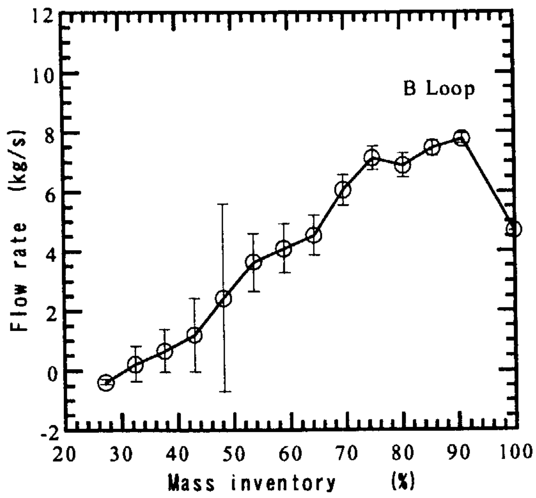

Fig. 2.3-1 Measured natural circulation flow rates vs. mass inventory.

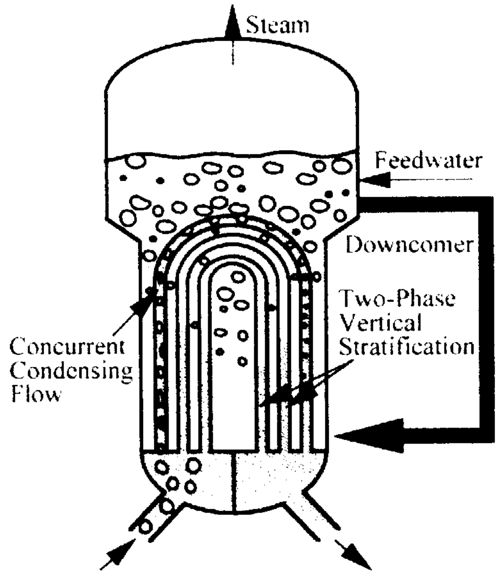

Fig. 2.3-2 Nonuniform flow behavior among U-tubes.

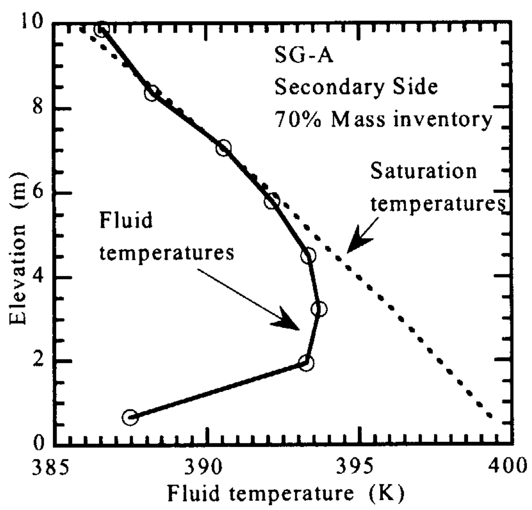

Fig. 2.3-3 SG secondary side temperatures.

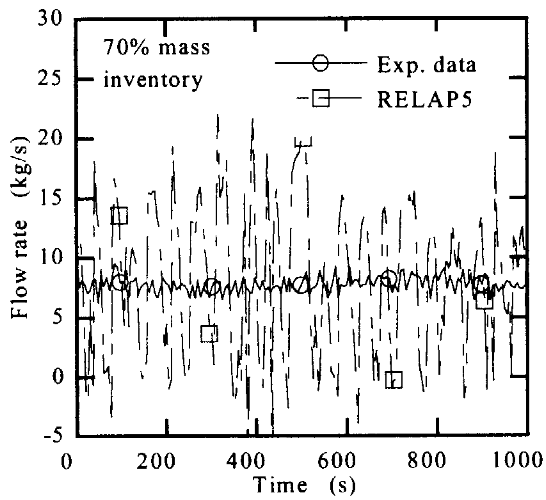

Fig. 2.3-4 Comparison of measured and calculated flow rates.

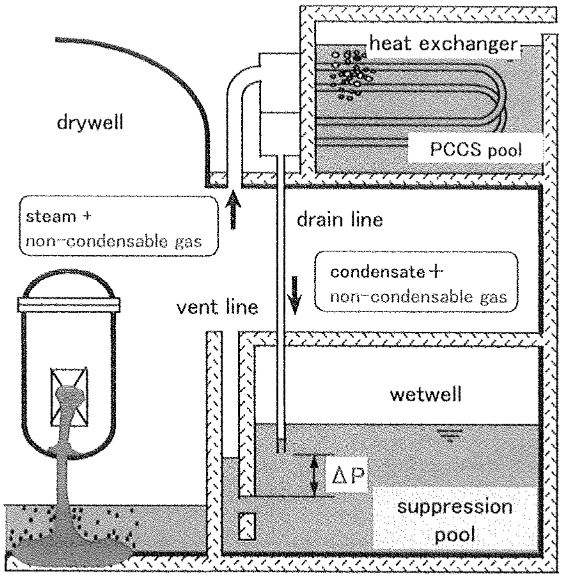

Fig. 2.3-5 Schematic of PCCS operation.

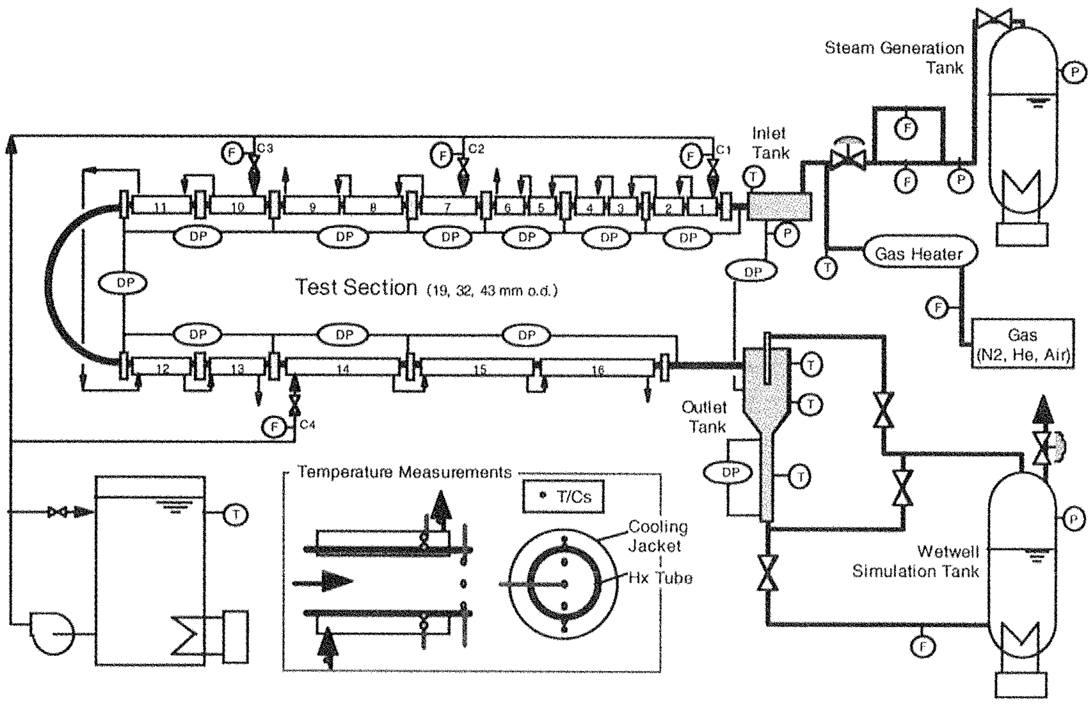

Fig. 2.3-6 Schematic of horizontal single U-tube test facility.

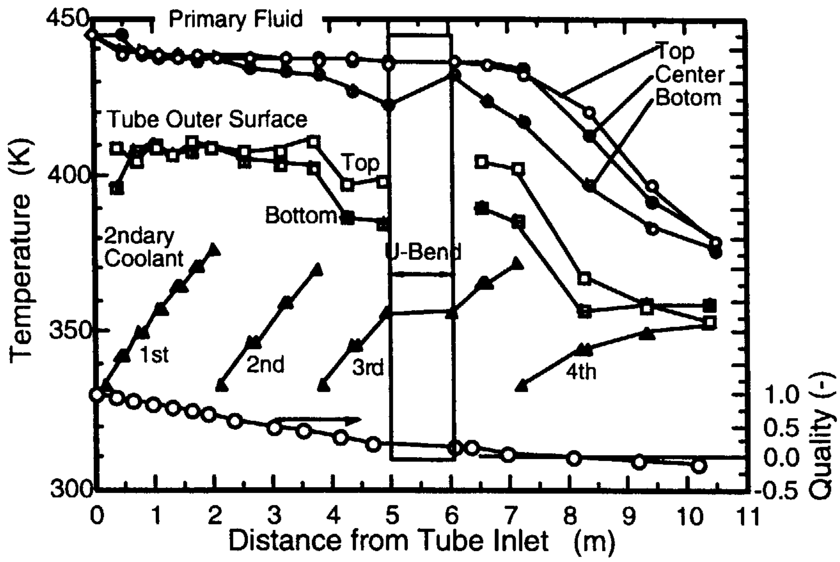

Fig. 2.3-7 Primary and secondary fluid temperature and outer tube surface temperature at nominal condition.

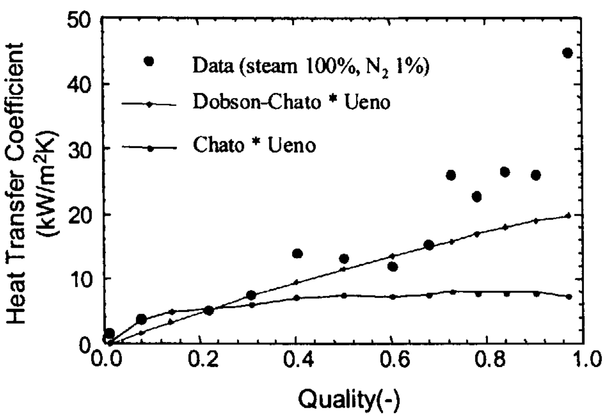

Fig. 2.3-8 Comparison of obtained inner wall condensation heat transfer : Coefficient with correlations by Dobson & Chato, Shah and Chato.

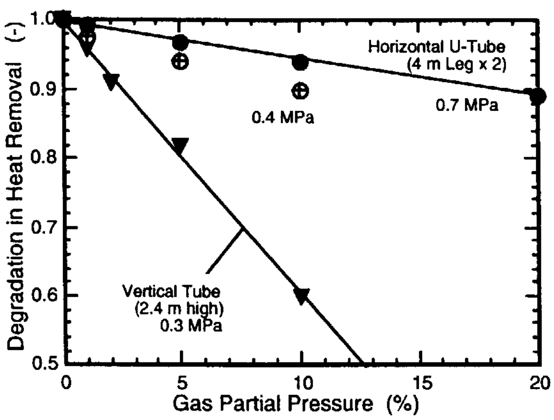

Fig. 2.3-9 Gross heat removal of the condenser tube versus non-condensable gas partial pressure.

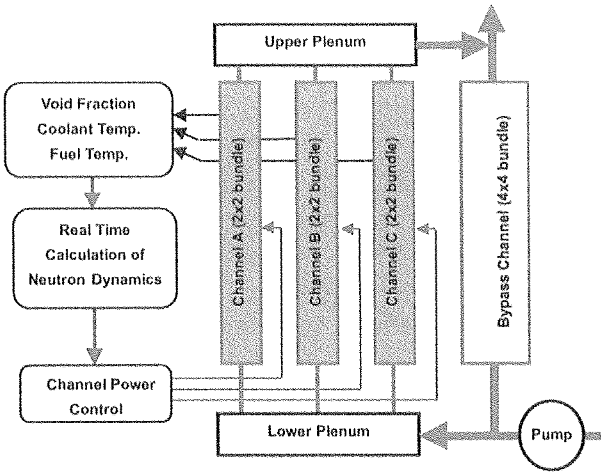

Fig. 2.3-10 Test section of Thermal-Hydraulic and Neutronic Coupling loop (THYNC).

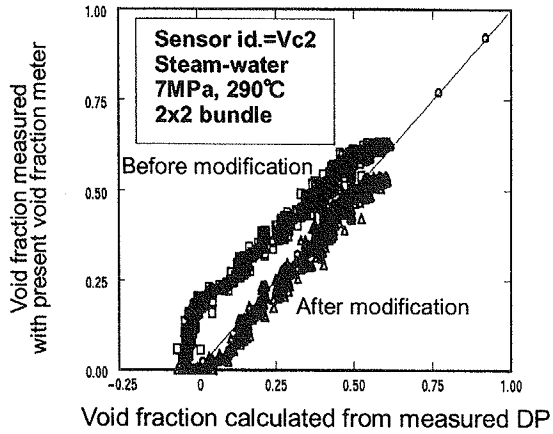

Fig. 2.3-11 Calibration result of present void fraction meter under high-pressure and high-temperature conditions (Before and after modification on water temperature effect).

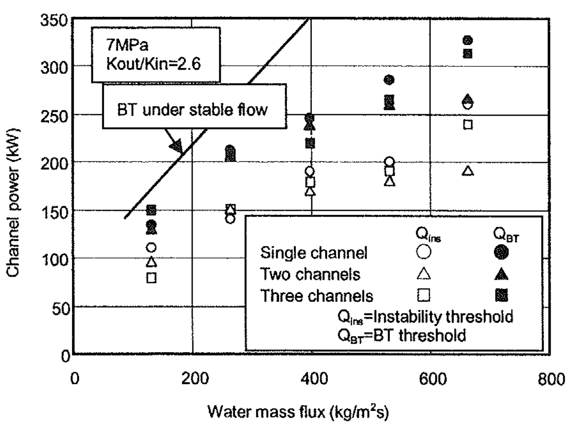

Fig. 2.3-12 Instability and BT thresholds of multi-channels (Effect of channel number).

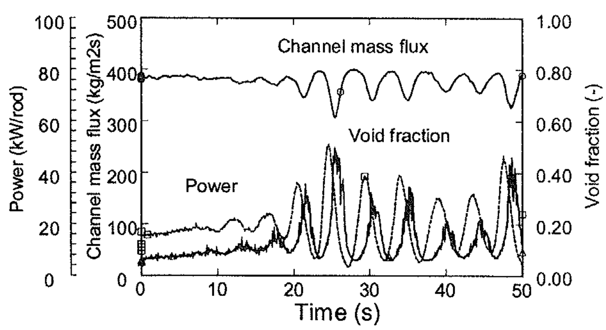

Fig. 2.3-13 Example of TH and neutronic coupling test (7MPa, Insertion of external reactivity at constant rate)

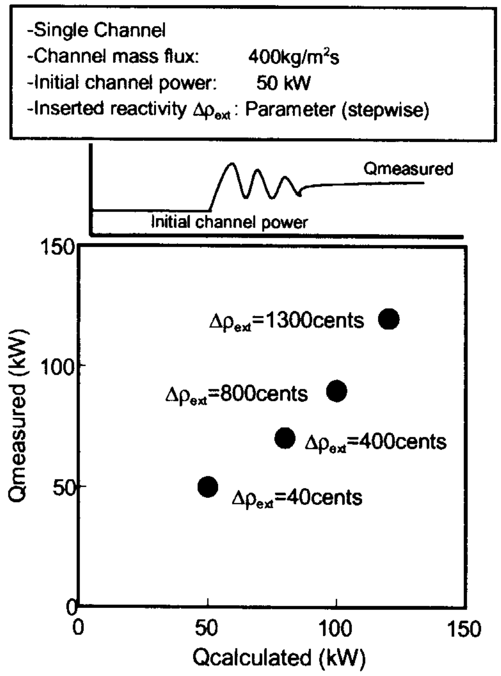

Fig. 2.3-14 Effect of $\Delta \rho_{ext}$ on TH and neutronic coupling.

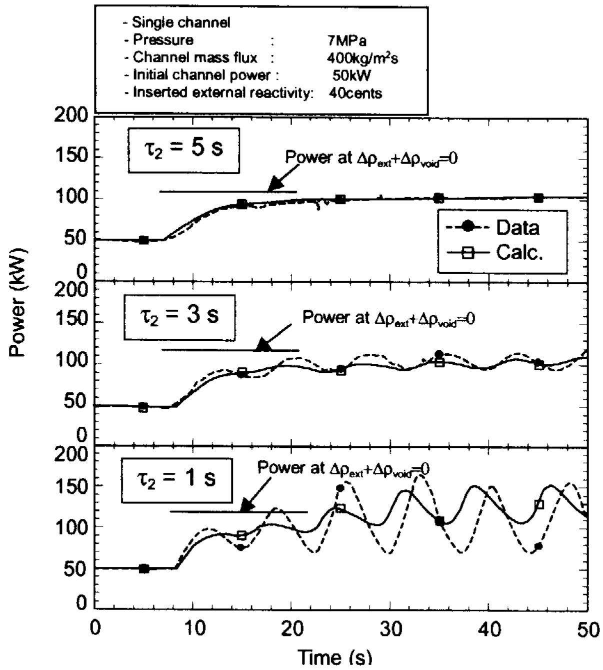

Fig. 2.3-15 Example of TH and neutronic coupling test.

-Incase of insertion of external reactivity (Stepwise)

-Effect of τ_2 on TH and neutronic coupling

2.4 Severe Accident Safety Research

After TMI-2 accident, a substantial increase of knowledge about phenomena that would occur during severe accident has been gained through great effort in would-wide research. Despite such a great advancement, there are still remaining issues and concerns which represent a high safety risk and for which insufficient knowledge-base has been acquired so far. The severe accident research in JAERI covers such remaining issues, considering work sharing among domestic and international researchers. The major severe accident phenomena that JAERI deals with are fuel-coolant interactions (FCI) including ex-vessel steam explosion and melt coolability, melt behavior, source term evaluation including FP release from irradiated core under pressures, containment and piping integrity. Obtained results will be utilized for risk evaluations, implementation of accident management measures and future reactor development.

2.4.1 Fuel-coolant interaction

During a severe accident of LWRs, there is a possibility to have a fuel-coolant interaction, or a steam explosion, if the molten core material falls down onto the lower head of the pressure vessel, ablates through the vessel wall and drops into a water pool in the containment vessel (ex-vessel steam explosion). If a steam explosion occurs, the mechanical loads caused from this violent phenomena can threaten the integrity of the containment vessel.

To estimate the impact of ex-vessel steam explosions, it is necessary to evaluate the molten core discharge conditions from the pressure vessel, the melt-coolant pre-mixing and the thermal-to-mechanical energy conversion during the explosive interaction. JAERI is developing a system of analytical codes for this purpose. **Figure 2.4-1** shows the phenomena in question and the concept of the code system. In this framework, CAMP code simulates the molten core convection and interaction with the pressure vessel and predicts the melt release conditions. JASMINE-pre and -pro codes simulate the melt-water mixing and explosive interaction to evaluate the mechanical energy output.

JASMINE consists of two separate modules -pre and -pro to simulate the two processes in a steam explosion which have different time scales and controlling physics. So called pre-mixing is a relatively long term (seconds) multiphase mixing between melt and coolant which establishes a certain multiphase mixture. The following propagation stage (milliseconds) dominated by shock wave propagation is the process where thermal-to-mechanical energy conversion takes place. Those two stages are modeled by JASMINE-pre and -pro, respectively. JASMINE-pre also involves the debris bed formation in its scope, which occurs when the melt stream or broken up particles settle on the containment floor, and which is important to predict the coolability of the core debris.

In FY1999 through 2000, a scoping analysis on a PWR ex-vessel steam explosion was performed. The premixing mass of the molten core was evaluated by the melt jet breakup model of JASMINE-pre. The result was arranged to set up the premixing condition described in **Figure 2.4-2**. With these initial and boundary conditions, the propagation process was simulated by JASMINE-pro assuming a triggering at the center of the premixing region. **Figure 2.4-3** shows the thermal-to-mechanical energy conversion ratio obtained by the calculation. The value ranged from 0.6 to 0.9%. The melt volume fraction had the most important influence, i.e. the increase of the melt in the premixing region caused more energy conversion in the range tested. Increase of the void fraction brought slight decrease of the conversion ratio. The triggering position did not much affect the result.

2.4.2 Experimental study on radionuclides release from irradiated fuel

The VEGA (Verification Experiments of radionuclides Gas/Aerosol release) program is being performed at JAERI since 1999 to clarify mechanisms of radionuclides release from irradiated fuel under severe accident conditions and to improve source term predictability by accumulating the release data from high burnup UO_2 or MOX fuel[1],[2]. The release of short life or low volatile nuclides of which data are not sufficient from the previous studies is mainly investigated by heating the fuel specimen up to 3250K in steam at elevated pressure up to 1.0MPa. One of special features is to investigate the effect of ambient pressure on the radionuclide release from fuel that has never been examined in previous studies.

The facility mainly consists of a steam/gas supply system, high-frequency induction furnace, thermal gradient tubes, aerosol/charcoal filters, noble gas trap, cascade impactor and on-line gamma ray measurement system (see **Fig. 2.4-4**). The test facility was installed into the beta/gamma concrete No.5 cell at the Reactor Fuel Examination Facility (RFEF) and completed in February 1999. Depending on the maximum temperature of induction furnace and the carrier gas species, different materials are used for the VEGA furnace. In the case of unoxidizing condition, tungsten (W) is used as crucible materials while ThO_2 is used for oxidizing condition.

Until the end of 2000, three VEGA tests were performed in inert He gas under conditions including the highest temperature and pressure among previous studies. In the first VEGA-1 test in September 1999, about 10g of fuel at 47GWd/tU without cladding was heated up to 2773K under atmospheric pressure to obtain reference data and to confirm the facility capabilities[3].

The second VEGA-2 test was conducted at 1.0MPa in April 2000 under the same conditions as VEGA-1 except for the system pressure. The on-line gamma ray measurement for fuel specimen showed that the fractional release of cesium (Cs) in VEGA-2 decreased by about 20% compared with that of VEGA-1 at 0.1MPa as shown in **Figure 2.4-5**. The influence of system pressure on radionuclide release was not taken into account in most of previous models since there had been few experimental data under high pressure condition. The obtained result is unique for developing a pressure dependent release model(4).

In the third VEGA-3 test performed in October 2000, the fuel temperature was raised up to 3123K. Almost 100% of volatile Cs was released at the time of fuel melting but no release of low-volatile Eu. It was also found that the diffusion coefficients of Cs in UO_2 grain obtained by VEGA results agreed mostly with the ORNL's results.

Fabrication study of ThO_2 furnace structures for oxidizing condition mostly finished3). Two tests per year using fuel re-irradiated at JAERI's research reactor and various kinds of carrier gas including steam will be initiated from FY2001. The test using MOX fuel will start from FY2002. In parallel with the experiments, a radionulcide release model that can treat the pressure effect will be developed and incorporated into the JAERI's source term analysis code, THALES-2.

2.4.3 Research on integrity of piping of reactor cooling system*

During a severe accident of a light water reactor (LWR), high temperature gases generated in degraded reactor core flow into piping of the reactor cooling system (RCS). The piping is subjected to a thermal load due to heat transfer from the high temperature gases together with an internal pressure load in high-pressure sequences, causing an increase in the failure probability of the RCS piping. Decay heat from fission products (FPs) deposited onto the piping surface might be added to the thermal load. The failure of and FP behavior in the RCS piping have a large impact on the accident progression and FP source term. In order to evaluate the structural integrity of the RCS piping during severe accidents, JAERI has conducted Wide Range Piping Integrity Demonstration (WIND) project since 1993, which includes researches on the failure of the RCS piping and FP transportation within the piping.

The research on the piping failure was composed of three activities; piping failure tests, post-test elasto-plastic creep analyses(5) and material property measurement(6)(7). Pipes made of typical LWR materials including stainless steel, carbon steel and Inconel were used as test sections in the piping failure tests. In those tests, history of displacement and time to creep failure of pipes were measured at a variety of temperature and internal pressure. The tests were used for the post-test creep analyses with ABAQUS code in order to validate the analytical approach developed in WIND project. Constitutive equations for creep strain and time to creep failure needed for the analyses were fabricated with the results of the material property measurement. Based on the findings of the piping failure tests and the post-test creep analyses, a simplified model was developed for the prediction of time to creep failure of the RCS piping. Piping failure is predicted to occur when a creep damage became unity in

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* This work was carried out by JAERI under an entrustment from the Ministry of Education, Culture Sports, Science and Technology of Japan.

the simplified model where the creep damage is accumulated in pipes as a function of time, temperature and Mises equivalent stress imposed on the pipes. The comparison between the predicted and measured time to creep failure is shown in **Figure 2.4-6**. It was found that the simplified model was capable of predicting the time to creep failure with an acceptable accuracy.

Concerning the aerosol transportation within the RCS piping, test series on aerosol deposition(8), revaporization(9) and interaction between FPs and structural materials(10), and related analyses for code validation(11)(12) have been performed. Cesium iodide (CsI) and cesium hydroxide (CsOH) were used as FP simulants in the tests. In some of the aerosol revaporization tests, CsI aerosol and vapor were introduced into the test section, where metaboric acid (HBO_2) was placed in advance on a floor area of the inner surface to investigate influences of boric acid included in primary coolant. The molecular ratio of iodine to cesium (I/Cs) deposited on the inner surface of the test section is shown in **Figure 2.4-7** for those revaporization tests. It was found that the ratio became smaller as the temperature of the test section wall increased, and the amount of iodine was significantly reduced at temperatures above 500°C. This result implies that CsI is decomposed by an interaction with boric acid at high temperatures when boric acid is retained on the inner surface of RCS structures during severe accidents.

2.4.4 Analysis of accident management measures

If the secondary system is depressurized during PWR severe accidents such as station blackout by stuck-opening of main steam safety valve (MSSV) or an accident management measure, the integrity of steam generator U-tube would be threatened at the earliest among the pipes of primary system. This is because the hot leg counter-current natural circulation (CCNC) of highly superheated steam and gas flow delivers decay heat of the core to the structures of primary system (see **Fig. 2.4-8**). When the steam generator tube rupture (SGTR) occurs in the above high-pressure sequence, it would cause the direct release of primary coolant including fission products (FP) to environment. Recent SCDAP/RELAP5 analyses by USNRC showed that the creep failure of pressurizer surge line would occur earlier than SGTR and results in release of the coolant into containment. However, the analyses did not consider the decay heat from deposited FP on the steam generator U-tube surface. The effect of decay heat on the steam generator U-tube integrity was investigated in the following two-step method.

The hot leg CCNC flow model used in the USNRC's calculation was validated in the first step through the analysis of LSTF simulation experiment in JAERI(13). The CCNC model of RELAP5 reproduced well the thermohydraulics observed in the hot leg cross section of the LSTF experiment as shown in **Figure2.4-9** and thus the model is considered to be reliable.

In the second step, an analytical study was performed with SCDAP/RELAP5 for TMLB' sequence of Surry plant with and without secondary system depressurization. The decay heat from deposited FP was calculated by JAERI's FP aerosol behavior analysis code, ART. The ART analysis showed that relatively large amount of FPs may deposit on the steam generator U-tube inlet mainly by thermophoresis. The SCDAP/RELAP5 analyses considering the FP decay heat predicted small safety margin for steam generator U-tube integrity during secondary system depressurization. Considering associated uncertainties in the analyses, the potential for SGTR cannot be ignored. Accordingly, this should be considered when the accident mitigation measure is decided for the high-pressure sequence with the secondary system depressurization[13].

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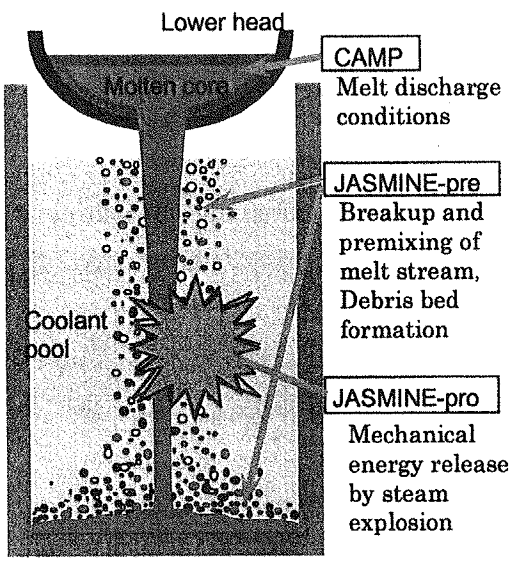

Fig. 2.4-1 Ex-vessel FCI phenomena and the code system developed at JAERI.

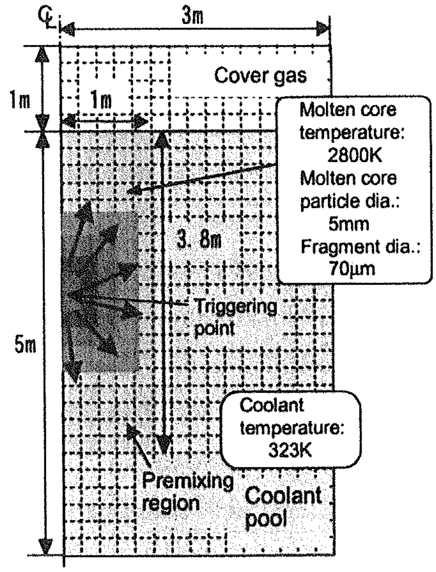

Fig. 2.4-2 Conditions of PWR ex-vessel steam explosion analysis.

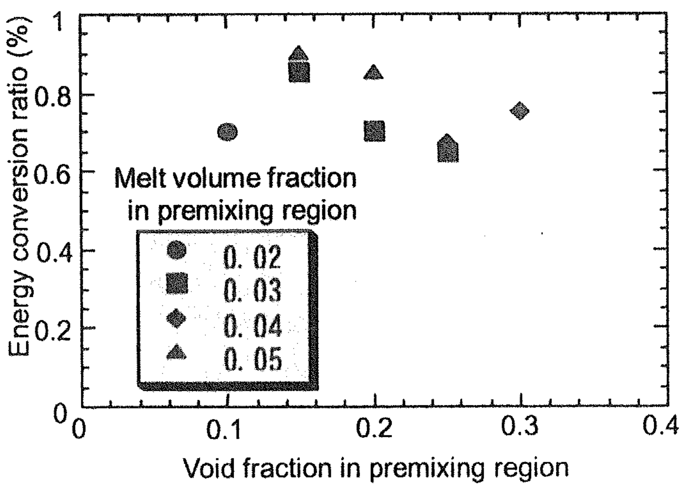

Fig. 2.4-3 Energy conversion ratio obtained by JASMINE-pro calculation.

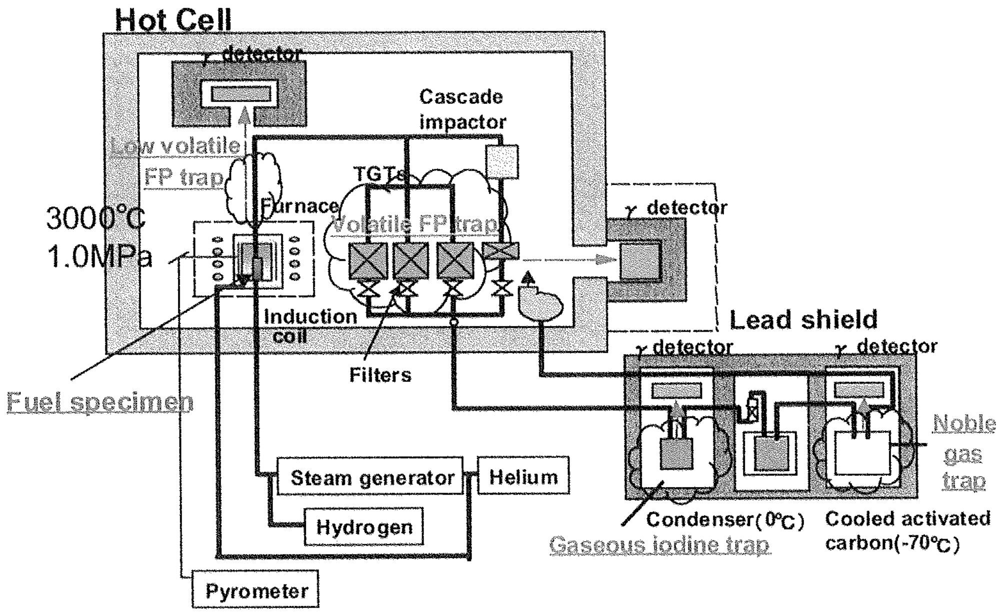

Fig. 2.4-4 Schematic of VEGA facility

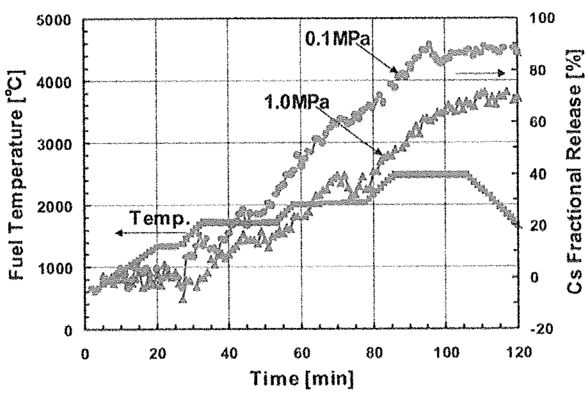

Fig. 2.4-5 Effect of system pressure on Cs fractional release under severe accident conditions.

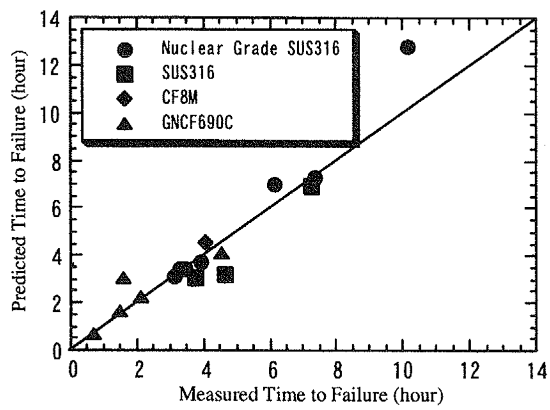

Fig. 2.4-6 Comparison of time to creep failure predicted by simplified model with test results.

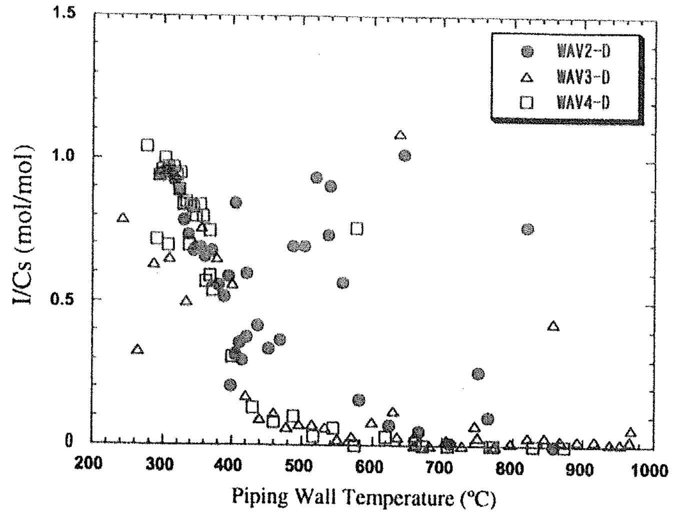

Fig. 2.4-7 Molecular ratio of iodine to cesium in aerosol revaporization tests with HBO_2 on inner surface of test section.

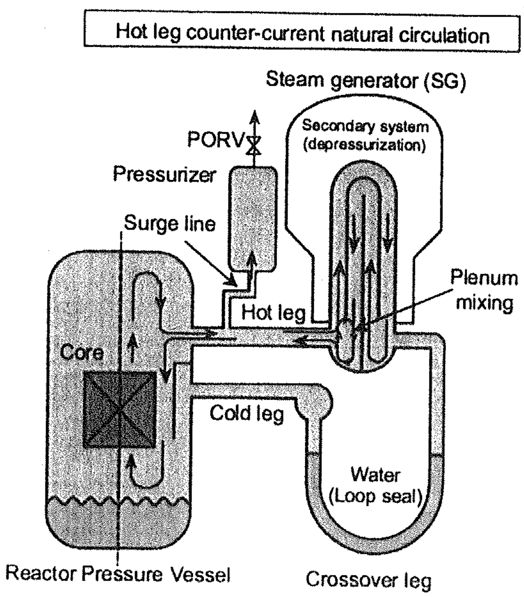

Fig. 2.4-8 Natural circulation flow patterns developed during PWR severe accidents.

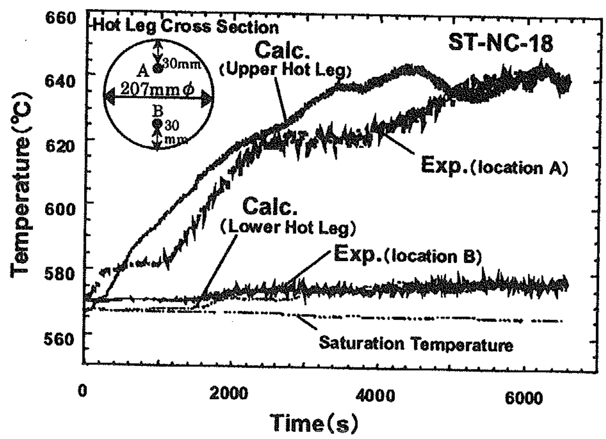

Fig. 2.4-9 Temperature distribution in hot leg cross section in case with CCNC.

2.5 Analytical Studies on Safety of Nuclear Facilities

The analytical studies on safety of nuclear facilities such as nuclear power plants (NPPs) and fuel cycle facilities cover five research areas shown in **Figure 2.5-1**. They aim at developing analytical methods and tools and obtaining insights useful for safety regulation through their application to various safety issues. The research activities in these areas are briefly summarized as follows:

The research on the Probabilistic Safety Assessment (PSA) aims at development of methodology to use PSA for safety management and regulation. Ongoing programs include the use of seismic hazard analysis for determining reference earthquakes for seismic design of NPPs, the application of seismic PSA to NPPs located on quarternary deposits, and the level 3 PSA of a BWR for internal events. These programs are intended to supply technical information for discussions on the revision of the seismic design guide and the safety goals in Japan.

The research on Human Factors has placed emphasis on the study of fundamental mechanisms and characteristics of human cognitive behavior including occurrence of erroneous actions to enhance operational safety and to improve man-machine interfaces of nuclear facilities.

The Analysis and Evaluation of Operating Experience Data aim at obtaining safety significant insights through the analysis and evaluation of incidents that occurred at nuclear facilities and to provide the lessons learned to regulatory bodies and the issues to be resolved to safety research programs.

In the research on Emergency Countermeasures, computerized systems have been developed aiming at enhancing the capability of countermeasures in a nuclear emergency. As well, a study has been carried out for effective information transfer to the public.

In addition, the Seismic Emergency Information System, which transmits earthquake information such as hypocenter, local seismic intensity and estimated consequences in a few minutes, has been developed to support decision making for mitigation of seismic disaster in general.

The first four research activities are closely related to nuclear safety regulation. Their products are expected to contribute to the regulatory bodies. The last one, however, deals with a generic issue, not specific to nuclear safety, pertinent to earthquakes as an application of methods and tools developed in seismic risk evaluation of NPPs.

In relation to the PSA study mentioned above, it is noted that the studies on phenomena during severe accidents and, prevention and mitigation of such accidents are being conducted as described in Sections 2.3 and 2.4. As for the deterministic safety evaluation, which is being applied in the safety review, development and improvement of computer codes are being conducted in the areas of plant thermal hydraulic behavior during accidents and structural integrity as described in Sections 2.2 and 2.3, respectively.

- Analytical Studies on Safety of Nuclear Facilities
  - Probabilistic Safety Assessment (PSA)
    - Level 3 PSA for internal events
    - Level 1 PSA for external events
  - Human Factors
    - Evaluation of man-machine systems
    - Human reliability analysis
    - Characteristics of human cognitive behavior
    - Methods to support human intellectual activities
  - Analysis and Evaluation of Operating Experience Data
    - Review and compilation of incidents
    - Analyses of safety significant events and implementation of computer codes
  - Emergency Countermeasures
    - Support system for ETAB*
    - Optimization of protective actions
    - Information transmission
  - Seismic Emergency Information System
    - Estimation of parameters of earthquakes
    - Dispatch of seismic information

*: the Emergency Technical Advisory Body under the Nuclear Safety Commission

Fig. 2.5-1 Major activities in the Analytical Studies on Safety of Nuclear Facilities.

2.5.1 Probabilistic Safety Assessment

A probabilistic safety assessment (PSA) provides a systematic and quantitative assessment of the overall safety of a nuclear facility. It is now widely recognized as a powerful tool for decision making in safety management and regulation of NPPs. PSA is performed in three levels: a level 1 PSA evaluates the frequency of core damage(CDF) by performing systems reliability analysis; level 2 PSA evaluates the containment failure probability and the source terms; and a level 3 PSA evaluates the off-site consequence. The research program on PSA at JAERI was initiated in 1980 and JAERI has developed methodology and computer codes for level 1 through level 3 PSA for internal initiating events and level 1 PSA for seismic events. A level 2 PSA for internal events and a seismic level 1 PSA for a reference BWR, which is called "the model plant" at JAERI, have been performed.

In the recent years, the research program on PSA at JAERI has been focused on two areas, the application of seismic PSA methodology for examination of issues related to seismic design and the level 3 PSA for internal events of the model plant.

(1) Application of Seismic PSA Methodology for Examination of Issues Related to Seismic Design

A seismic PSA of an NPP provides various useful information for design and safety management of NPPs. For example, it provides information on relative importance of various accident scenarios and failures of systems and components, which can be used for enhancing safety of existing NPPs and optimizing safety system design of future plants. JAERI has conducted, as a part of the Five Year Research Program for Nuclear Installations of the Nuclear Safety Commission of Japan, the following research subjects to extend the uses of seismic PSA for seismic risk management and design of NPPs.

(a) The use of seismic PSA for NPPs sited on Quaternary Deposits(1,2),

(b) The use of seismic hazard analysis for determining scenario earthquakes for seismic design(3),

(c) The use of seismic PSA for further reduction of seismic risk by more detailed understanding of important accident scenarios(4).

The following part of this section will be focused on the results from item (a). In the item (b), the probabilistic seismic hazard analysis was used to identify seismic sources which have dominant contribution to the seismic hazard at an NPP site and the merits of such probabilistic method for determining design earthquakes have been examined. In the item (c), since the loss of offsite power transients have been shown to be a dominant initiating event in PSAs of many plants, scenarios of transient events with loss of offsite power was studied in detail. The fragility of electric power grids outside NPPs was evaluated with the use of the data of equipment failure during the Kobe earthquake in Japan.

At present in Japan, all NPPs have been sited on bedrock which was formed in Tertiary or earlier era. In future, from viewpoint of expanding the available range of siting, it is expected that NPPs may be constructed on the ground of Quaternary Deposits (QDs).

The objectives of this study are to accommodate the procedure of seismic PSA to evaluation of NPPs sited onto QDs, and to formulate a probabilistic approach of seismic stability analysis for the ground of QDs.

Figure 2.5-2 shows the procedure of this study. Since JAERI has made a seismic PSA for the model plant, the site and plant data for the seismic PSA were used for this study as far as possible. Since the model plant was originally sited on Tertiary stratum, the ground condition was replaced by those of QDs of Diluvial Epoch for this study. The ground was a uniformly layered QDs with 415~715m/s in shear velocity.

In order to perform a probabilistic analysis for failure of the ground, a 2D-FEM model of QDs consisting of layers of gravel and mudstone was developed. Material property data for response and strength evaluation were assembled from limited data of observation in geological surveys of QDs and the variabilities of these data were statistically analyzed.

From a survey of existing works, it was recognized that the seismic stability evaluation for the ground of QDs should address several types of failure modes of the ground, namely, sliding on the bottom surface of building foundation, rotational slip between the clod of soil under the foundation and the surrounding soil stratum, cyclic mobility of dense gravelly soil deposits saturated with pore water, where the cyclic strength of soil shows a gradual decrease within a definite strain limit, residual settlement of the ground, and bearing capacity of the ground insufficient to support superstructure. Procedures of probabilistic stability evaluation were formulated for those failure modes using the safety factor (SF) based on mainly the outcome of a study by JSCE (Japan Society of Civil Engineers) [5]. The results from this analysis are shown in **Fig.2.5-3**. This figure shows the probability of occurrence of non-linear deformation of the ground due to each failure mode in terms of acceleration at the bedrock. It should be noted that initiation of non-linear deformation does not necessarily mean a collapse of the building or any other superstructure. Then the effect of each failure mode on the integrity of the superstructure was examined with consideration of the strain, deformation and change in boundary conditions of the building foundation.

In this analysis, the cyclic mobility was identified as a dominant failure mode for this model site, and, among material property data, the variability of liquefaction strength was identified to be a source of large uncertainty in probability of failure by cyclic mobility.

Through this study, it was confirmed that this probabilistic approach is useful to examine the relative importance of various failure modes of the ground. Since seismic stability of QDs has large uncertainty due to the random variability of material property, further study is necessary to improve material property data and to examine the effects of such uncertainty. It is planned to assess the effect of difference in ground response between Tertiary stratum and QDs, such as the difference in damping effects, on the component failure and core damage frequency. This analysis will clarify the merits/demerits of siting on QDs, and identify the factors that influence the seismic risk.

(2) Level 3 PSA for Internal Events of the Model Plant [6]

The objectives of the level 3 PSA of the model plant at JAERI are to find needs for further improvement of risk assessment methodology and to obtain a better understanding of the controlling factors of public risk. Other recent important objective is to supply inputs for discussions by the Nuclear Safety Commission on safety goal, effectiveness of emergency planning, siting criteria, etc. In order to obtain an estimated profile of consequences for various accident conditions, the consequence assessments have been made for postulated core damage accidents with source terms derived from a generic level 2 PSA of the model plant. The model plant is a BWR5 with Mark-II type containment, which is a typical nuclear power plant in Japan. It should be noted, however, that the analysis does not fully consider accident management measures currently under preparation in Japan.

Based on the results from accident progression analyses, the core damage sequences were categorized into five groups by the similarity in timing of containment overpressure failure after core melt initiation. The source terms used in the analysis have been calculated by a severe accident analysis code, THALES-2 for each represented sequence from these five core damage accident sequences. Each sequence was considered to have four different failure modes that included the drywell and wetwell failure modes, controlled release mode by containment venting, and accident termination mode by containment spray. **Figure 2.5-4** shows the event timings and source terms considered in the analysis. The off-site consequences have been assessed for 144 different weather sequences taking into account the countermeasures by an accident consequence assessment code, OSCAAR.

Figure 2.5-5 shows the conditional probabilities of both early fatality and late cancer fatality as a function of distance from the release point for various core damage sequences. Early fatalities appear only for a scrum failure sequence. Because of its threshold, the curve of early fatality is strongly dependent on the distance from the release point. The curves of late fatality depend strongly on containment failure mode and decrease with a less pronounced slope. Considering that the level 2 PSA for the model plant provided a CDF of 5×10^{-7} per reactor year and containment failure frequency of 1.7×10^{-7} per reactor year, the risks of both early fatality and late cancer fatality to an average individual in the vicinity of the plant would be much less than the quantitative health objectives of the USNRC [7]. The results also indicated that collective dose was dominated by the long-term exposure from the ground and food contamination and their magnitude was strongly dependent on releases of iodine and cesium. On the other hand, early doses were governed by both external irradiation and inhalation from the radioactive cloud. The level 3 PSA will be completed by adding analysis of other containment failure modes with taking into account uncertainties involved in the analysis.

2.5.2 Human Factors

The research project on human factors in nuclear facilities was initiated in FY1989. The research conducted includes the following subjects. (a) The research subject "Methodologies of Man-Machine System (MMS) Evaluation" consists of two items, i.e., development of methodologies for evaluation of MMS and development of a dynamic simulation model of MMS, JACOS. JACOS is to be utilized as a tool for evaluating a MMS such as a control panel or operational procedure. For the evaluation methodology, an integrated conceptual framework has been developed. (b) In the research subject "Human Reliability", a software tool JASPAHR was developed for supporting human reliability analysis in PSA. (c) The research subject "Characteristics of Human Cognitive Behavior", consists of studies on characteristics of human cognitive behavior with full scope reactor simulator experiments, small scale micro-world experiments and analyses of significant accidents. The JCO criticality accident has been analyzed as part of this activity. (d) In the research activity "Man-Machine Interface", an innovative man-machine interface system has been developed with extending the Ecological Interface Design (EID) concept[8].

The interface system developed consists of two workstations each of which has four CRT terminals and two large display units (100 inch wide) shown in **Fig. 2-5-6**. The CRT terminals for each work- station are placed adjacent to each other, so that operators can easily refer to the information on the two or more terminals in parallel. On the two large display units, graphical formats representing higher level of information at means-ends network, such as "Energy Balance of Reactor System" and "Mass Flow Distribution in Secondary System", are presented. There are two types of menu, namely a) static menu and b) dynamic menu. The static menu consists of five permanent buttons each of which corresponds to different types of graphic display formats. By clicking one of these buttons, a menu for corresponding type of graphical format is placed on the Information Display Area. As for the dynamic menu, a set of menu buttons are placed by clicking mouse at a "navigation cue" attached at the information items such as process parameters on a display page. Each navigation cue is associated with sets of graphical display formats, so that user can call an appropriate display pages on the Information Display Area by choosing one of the display formats among those linked to the navigation cue. With the dynamic menu, operators can also select the CRT terminals on which a display format chosen is placed. With this CRT selection function, operator can refer to the relevant information without changing or hiding the graphical display format being currently looked at. Adding to the graphical formats included in the existing simulator interface, five graphical display formats representing higher level information items at means-ends network were newly created. The display formats having been created are: (1) Energy Balance in Reactor System, (2) Mass Balance in Primary System, (3) P-T Diagram with the Functional Relation of Related Process Parameters, (4) Mass Flow Distribution in Secondary System, and (5) State Diagram of Steam Generator. Among these, the display format of P-T Diagram is an extension of the diagram included in the existing interface system by adding P-T diagram representing a state of pressurizer and the functional relation of process parameters related. One graphical display format representing "Energy Balance in Reactor System" among the graphical display formats newly created is shown in **Fig.2.5-7**.

In the research activity of analysis of significant accidents from the viewpoint of human factors, the JCO Criticality Accident has been conducted utilizing a framework based on the cognitive systems engineering[9]. In the framework, analysis is made integrally both from system viewpoint and actor's viewpoint. From the system viewpoint, the functional structure of the work system and the system requirements in terms of goal to be achieved are uncovered. On the other hand, from the actor's viewpoint, how the system function and working situation are perceived by the actors is uncovered. The characteristics of changing the procedures in JCO is regarded as the process in which the conflicts among the various constraints aiming at the goal had been resolved through degrading the safety requirements for criticality. The actors' migration toward unsafe region could be also regarded as the consequence of pressure of higher efficiency and gradient toward less workload, occupational risk and manipulation difficulty. Among all, the change into utilizing the pure uranyl nitrate storage tank was the most serious error in degrading the double criticality control strategy with mass and geometrical control into the single criticality control. It should be noticed that the option of utilizing the storage tank in the homogenizing process was not a unique solution. Based on the analysis of the accident, some countermeasures are derived to prevent recurrence of such an accident.

- Improvement of training and education method including actors' participation to some kind of criticality-related experiment and exercise utilizing a computer simulation program for criticality phenomena.
- Enhancement of risk perception of actors with introducing warning icon at the entrance of the facility and warning sign for each equipment.
- Implementation of information system to support actors for monitoring & prediction of

system state. With such information system, actors could comprehend the system state in terms of the safety boundary and could predict the effect of assumed actions. Furthermore, in order to support actors for developing an appropriate mental model, it would be effective to make such an appropriate information available regularly to them.

2.5.3 Analysis and Evaluation of Operating Experience Data

The primary sources of operating experience data for the analysis and evaluation of incidents are (a) the reports of the Incident Reporting System (IRS), (b) the reports of the International Nuclear Event Scale (INES), both of which have been jointly operated by the Organization of Economic Cooperation and the Development/Nuclear Energy Agency (OECD/NEA) and the International Atomic Energy Agency (IAEA), and (c) the Licensee Event Reports (LERs) and Generic Communications such as Information Notice of the U. S. Nuclear Regulatory Commission (USNRC). The IRS provides the regulatory bodies (and their related organizations) in the 31 member states with the information on only the incidents at nuclear power plants, while the INES provides the information for public communication on events at all types of nuclear facilities. Recent activities are summarized below.

(1) Review and compilation of incidents

About 200 incidents reported to the IRS in 1999 and 2000 were reviewed and compiled. Those included incidents involving adverse effects of fire protection system on the safety features, long-term unavailabilities of safety systems, loss of safety functions due to flooding, steam generator tube rupture (SGTR), primary system pipe crack and small leaks, and recurring events such as ECCS strainer clogging and power oscillation.

As well, the reports from INES have been translated into Japanese, disseminated to the relevant organizations such as the Nuclear Safety Commission of Japan (NSC), and opened to the public through the Internet. In these two years, approximately 70 INES reports we received from June 1, 1998 to December 31, 2000 were reviewed and translated. Also, a report containing these INES reports was published(10). During this period, two level 4 events took place in Japan (the JCO criticality accident) and in Egypt (radiation exposure due to lost sources) involving two or more fatalities. Relatively a large number of incidents involving lost sources were reported to the INES, about a half of which occurred in France.

In addition, twenty-one criticality accidents in nuclear fuel processing facilities in foreign countries were reviewed and analyzed to identify the overall trends of these accidents and the sequences and causes of accidents in terms of similarities to the JCO accident that occurred on September 30, 1999(11,12). These 21 accidents consist of seven in the United States, one in the United Kingdom and thirteen in Russia. Most of them occurred during the period from mid-1950's to mid-1960's, but one criticality accident took place in Russia in

1997. The observations from the review are summarized as follows:

- Twenty of the 21 accidents occurred with the fissile material in a liquid.
- Twenty of the 21 accidents occurred in vessels/tanks with unfavorable geometry.
- There were seven fatalities that were involved in five accidents.
- Three accidents involved a re-criticality condition caused by inadequate operator actions and two of them led to the death of the operators.
- Mechanisms of the criticality termination vary as follows: ejection or splashing of the solution at the time of power excursion, boiling or evaporation, addition of neutron poisons, or manual draining of solutions.
- In some cases, the problems similar to those observed in the JCO accident were identified: violations of procedures and/or technical specifications for improving work efficiencies, procedural changes without any application to and permission from the regulatory body, lack of understanding of criticality hazards, and complacency that a criticality accident would not occur.

(2) Accident Sequence Precursor Study

Since 1996, the research activity on the Accident Sequence Precursor (ASP) Study has been carried out to obtain the risk significant trends, to characterize risk insights useful for identifying plant vulnerabilities, to feed the lessons learned from the study back to plant operations, and to establish risk indicators for event assessment. This activity consists of the ASP analyses of actual events that have the potential of risk significance, the review of the ASP documents published by the USNRC and development of the event tree models for providing a more realistic ASP analyses. In these two years, the following studies were carried out.

- Event trees used in the USNRC's ASP Program are different from those constructed in the PSA and as well, some anomalies observed in the events could not be modeled in the ASP event trees because those were prepared as the standardized or generic models applicable to nuclear power plants with the similar design features. For the purpose of resolving these problems, thermal-hydraulic analyses were performed using the best estimate code, TRAC-PF1, for a variety of the SGTR-initiated sequences to develop a sophisticated event tree that could depict realistically the sequences involving actual failures observed(13).
- In order to derive such risk information from the results of the ASP analyses, the published ASP documents were reviewed and the trends were examined focusing on dominant sequences identified for precursor events identified(14). Also, the occurrence frequencies of the precursor events and annual core damage probabilities were estimated.

2.5.4 Emergency Countermeasures

The research on emergency countermeasures aims at enhancing capability of countermeasures in a nuclear emergency, where it is required to promptly obtain information on accident conditions, emergency countermeasures and so on, and to effectively transmit disaster information to the public. The research consists of three activities: 1) development of computerized support system for the Emergency Technical Advisory Body (ETAB) under the Nuclear Safety Commission, 2) research on optimizing protective actions in an emergency, and 3) research on information transmission in an emergency.

(1) Development of computerized support system for ETAB*

If an accident that results in an emergency occurs at a nuclear facility, the ETAB provides technical advice to the national government. To support the ETAB's activities by providing the ETAB with useful information promptly in an emergency, a computerized support system for the ETAB, COSTA[(15)], has been developed at JAERI since 1985. The COSTA system consists of several databases, the program for identifying plant status and predicting source terms, and the evacuation simulation program. The information provided by COSTA is summarized below:

a) nuclear power plant data of plant configuration and plant technical specifications, and environmental data of topography and population statistics,
b) incident data reported to the Incident Reporting System at the OECD/NEA,
c) severe accident analysis results consisting of a summary of the analysis, thermal-hydraulic and FP(fission products) behavior data calculated with computer codes,
d) emergency data transmitted from an affected plant,
e) plant status inferred, in which the emphasis is on the status of the integrity of the FP barriers (fuel cladding, primary cooling system, and containment vessel),
f) source terms (timing and amount of FP released to the environment) predicted,
g) environmental dose equivalent predicted based on the Gaussian plume model, and
h) evacuation simulation results including time required for evacuation and public dose averted by evacuation.

In 1999-2000, the COSTA system was improved to add the function of identifying plant status and predicting source terms for the advanced BWR and to make it easier and more practical to use. Then, the development of the COSTA system was completed in 2000, and the system was transferred to the Nuclear Safety Technology Center for practical use.

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* This work was carried out by JAERI under an entrustment from the Ministry of Education, Culture Sports, Science and Technology of Japan.

(2) Research on optimizing protective actions in an emergency*

A computerized system for optimizing off-site emergency countermeasures has been developed(16) to support planning of emergency preparedness by supplying useful information such as an effective action applicable to the early stage of an emergency. The system consists of a program for simulating protective actions (sheltering, evacuation and iodine prophylaxis), programs for evaluating health effects and economic consequences, a program for selecting optimized protective actions, and a database that includes information on analysis conditions and output data of these programs. The basic concept applied in the system is justification and optimization, that is, the protective action should be justified and reasonably optimized as to achieve the maximum net good. The system was constructed in 1999, and an analysis with the system was performed for a model site where various emergency situations were postulated by changing source terms, atmospheric conditions, timing of conducting protective actions and so on. Also, a mathematical model for estimating effectiveness of iodine prophylaxis was developed(17). **Figure 2.5-8** shows effectiveness of iodine prophylaxis calculated for Japanese and Caucasian. It is shown that more than 80 % of the internal radiation exposure dose to the thyroid gland (the thyroid dose) would be effectively averted by administration of stable iodine (KI) a few hours before an inhalation of radioactive iodine. More precisely, the effectiveness can differ as illustrated due to a difference in the iodine metabolism caused by dietary intake of iodine among Japanese and Caucasian. It should be noted that the thyroid dose is, in general, evaluated not only by the effectiveness but also by a dose conversion factor, for internally taken radioactive iodine, which is smaller in Japanese than in Caucasian.

(3) Research on information transmission in an emergency (18)

To study appropriate methods for disaster information transmission to the public in an emergency, a case study was performed for the JCO criticality accident which occurred on September 30, 1999 in Tokai-mura, Japan. In the criticality accident, the local authorities requested evacuation for the inhabitants within 350 meters of the JCO facility, and indoor sheltering for those within 10 kilometers of the facility. The questionnairing was conducted to know when, where and how the affected inhabitants received the information on protective actions requested from the local authorities and how they responded to the requests. Ninety-two inhabitants within 350 meters were interviewed directly. For the inhabitants within 10 kilometers, 1,298 questionnaires were collected from randomly sampled 7,000 inhabitants. The case study showed the following.

- It took two to three hours for all the inhabitants concerned to know the requests.

* This work was carried out by JAERI under an entrustment from the Ministry of Education, Culture Sports, Science and Technology of Japan.

- Information notification systems of the local authorities such as disaster prevention radio communications played important roles to transmit the information to the inhabitants in Tokai-mura. On the other hand, mass communication was most effective within 10 kilometers outside Tokai-mura.
- About eighty-five percents of the inhabitants within 350 meters evacuated. Here about thirty percents of them followed the directions of the local authorities and were transported by the public buses. On the other hand, about sixty percents of them did not follow the directions and evacuated by their own cars.

In conclusion of the case study, disaster prevention radio communication systems of local authorities are effective and should be arranged in other near field municipalities as well as Tokai-mura to enhance the capability of disaster information transmission.

2.5.5. Seismic Emergency Information System (19)

The Hyogo-ken Nanbu Earthquake that took place on January 17, 1995, has brought serious damages in structures and lifelines around the city of Kobe. It has been recognized through its experience that it is important to transmit accurate information on the earthquake and damages and to take appropriate initial countermeasures based on the information.

From this point of view, JAERI has conducted an R&D program since 1996 to develop a real-time earthquake information system within the framework of "Frontier Research Programs on Earthquake" under the auspices of the Science and Technology Agency (STA). A part of the program has been carried out under the joint program between JAERI and NIED (the National Research Institute for Earth Science and Disaster Prevention). The program has involved the development of methodologies to estimate a hypocenter, fault parameters and earthquake motion.

In order to examine the methodologies and performance of the system, a region around 30 km of the Tokai Research Establishment of JAERI was selected as the test field. A database has been developed for the soil in the test field. The database includes the parameters such as shear wave velocity and relationships between equivalent shear modulus vs. shear strain and damping ratio vs. shear strain in every 500 m^2 mesh. Existing seismic monitoring networks consisting of three seismometers were utilized as part of the system.

(1) Configuration and functions of the system.

This system consists of six subsystems as shown in **Fig.2.5-9**. The subsystem SS-1 transmits the earthquake wave data measured by the seismometers to SS-2, the earthquake motion data acquisition subsystem. The subsystem SS-2 extracts the typical noises from the earthquake wave data and calculates earthquake characteristic values. These processed data are transmitted to SS-3 and SS-4 to identify the hypocenter, fault parameters and earthquake motion parameters in 500 m^2 mesh. In SS-4, six different techniques are provided for estimating the distribution of earthquake motion parameters. Some of them are capable of considering the amplification characteristics of a local ground motion due to nonlinear features of the surface soil and irregularity of the bedrock. The subsystem SS-5 transmits the earthquake information to SS-6, the subsystem of users, through the Internet and facsimile.

(2) Application to disaster mitigation system

A prototype system to be used for disaster management and mitigation[2] has been developed as an application of the seismic emergency information system mentioned above. This system consists of a site at users and that at a host, the disaster information center, and is characterized by a feature of mutual data transmission between them. Namely, the host transmits the information on the earthquake and damages being estimated, and the users can not only request additional information but also input the information on, for example, actual damages near the user site. This communication can easily be achieved by applying a data condensing technique. It is also noted that a multi-dimensional geographical information display system DiMSYS[20] is used for data management in each site. The system has been tested in Tokai-mura under a hypothetical earthquake.

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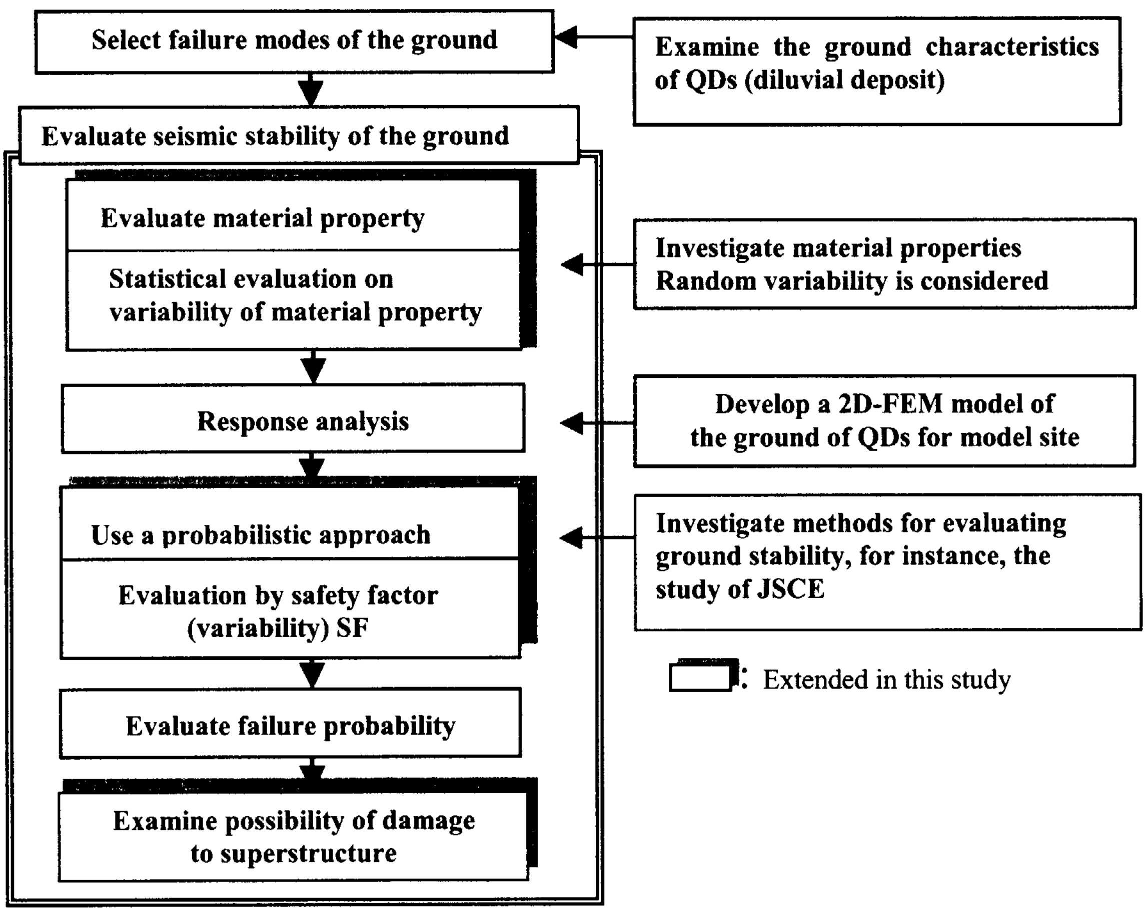


Fig.2.5-2 Procedure of probabilistic seismic stability evaluation for QDs[2] .

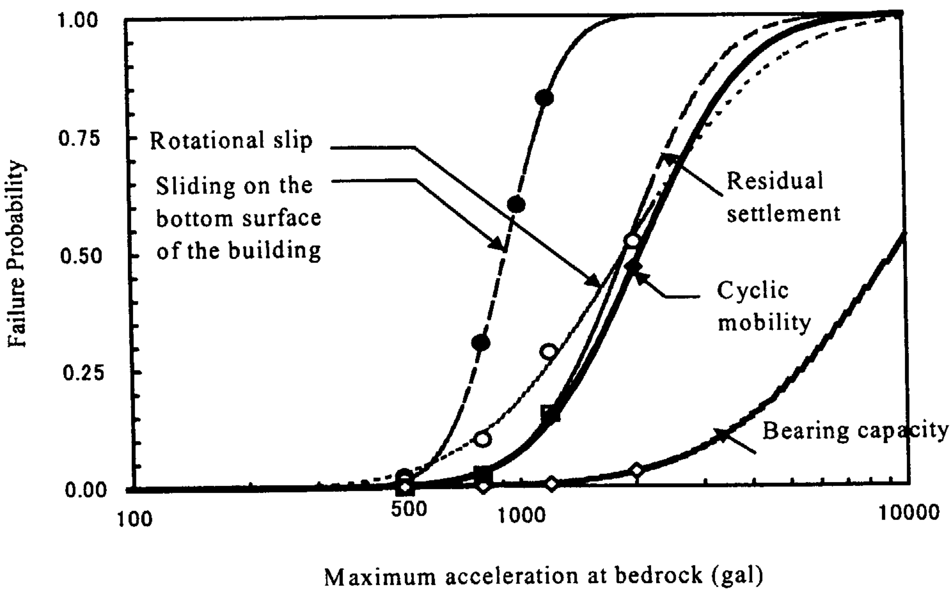


Fig. 2.5-3 Failure probability of the ground (2)
Failure is defined here by the occurrence of non-linear deformation and is not the failure of superstructure.

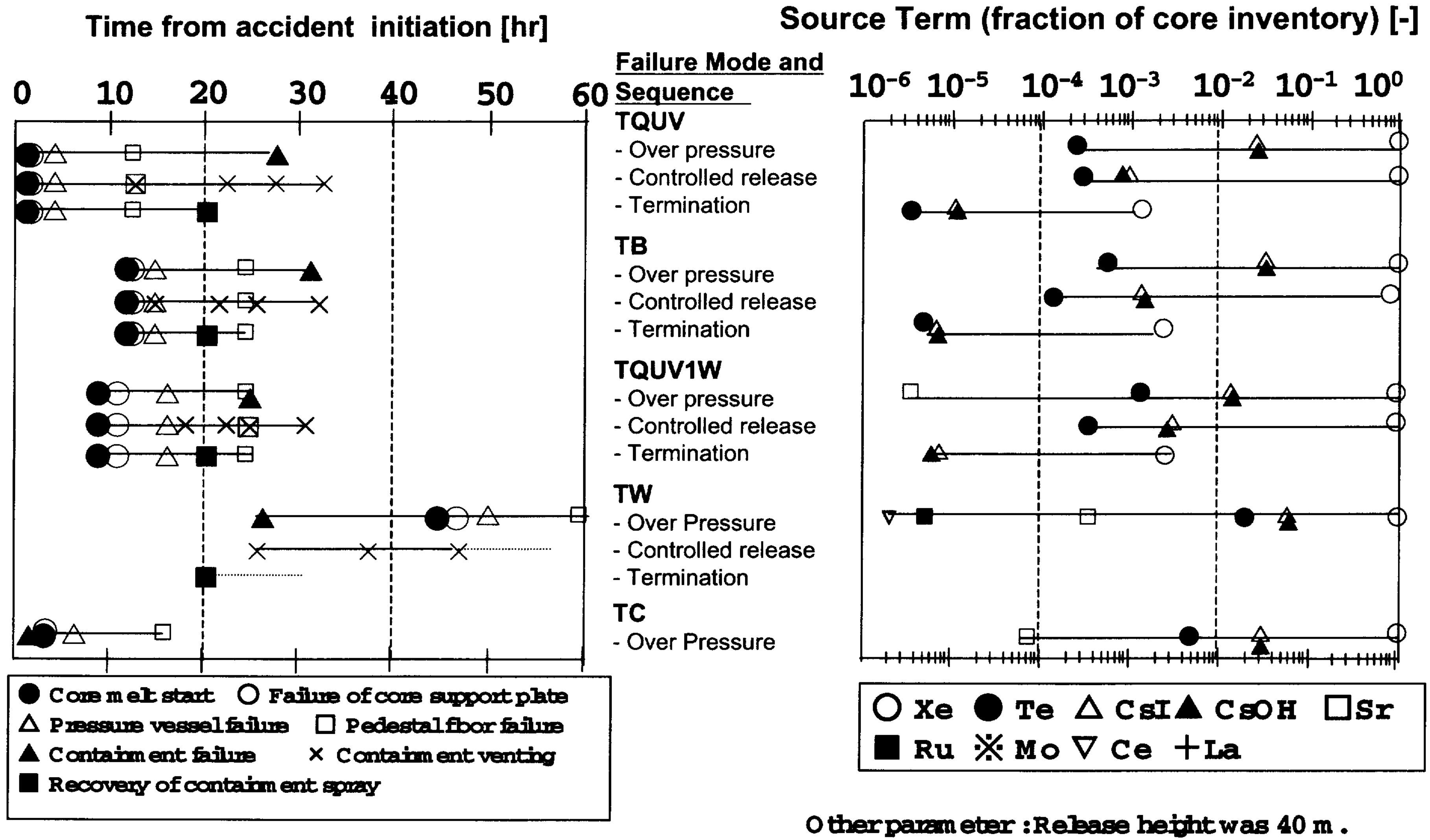


Fig. 2.5-4 Accident progressions and source terms for five sequence groups(6).
(TQUV: Loss of all injection systems, TB: Station blackout, TQUV1W: Transient followed by successful low pressure injection and failure of residual heat removal, TW: Transient followed by successful high pressure injection and failure of residual heat removal, TC: Transient followed by failure to scram)

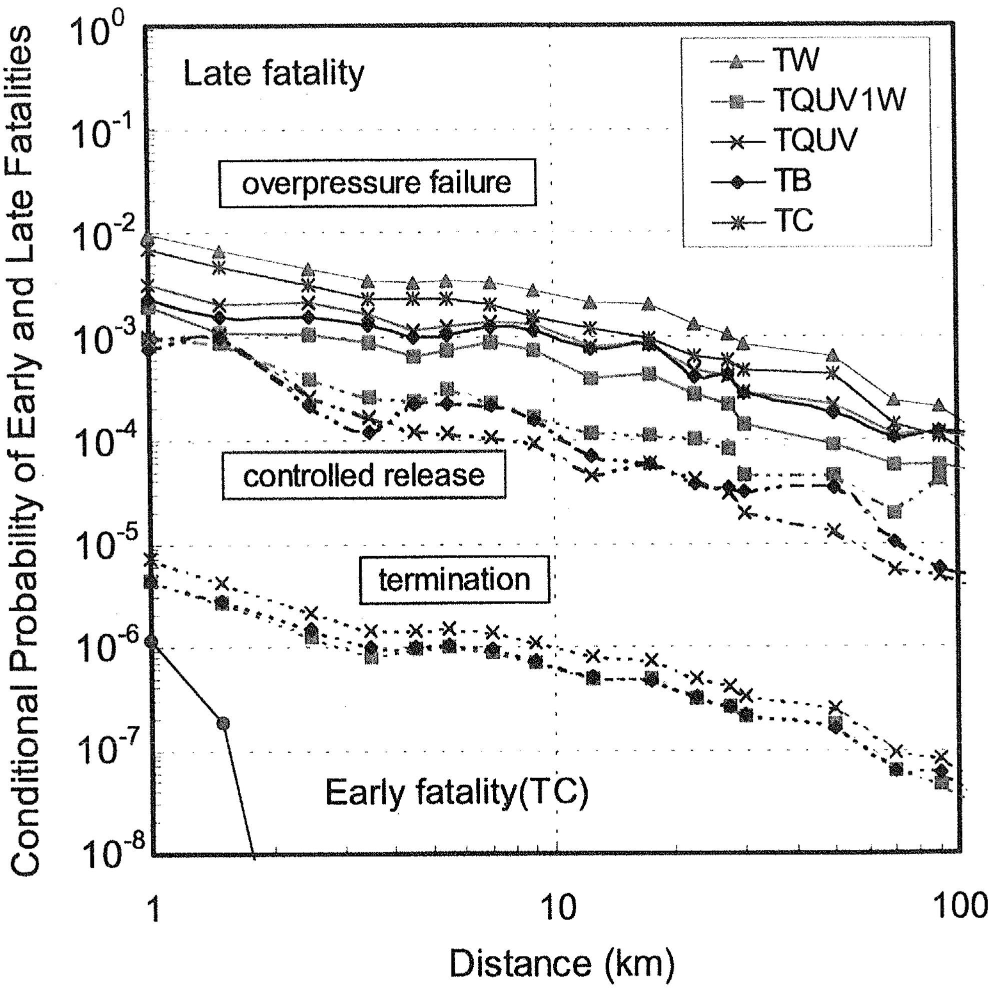


Fig. 2.5-5 Conditional probabilities of individual early and late fatalities with respect to distance from the release point (6).
(see Fig. 2.5-4 for abbreviations)

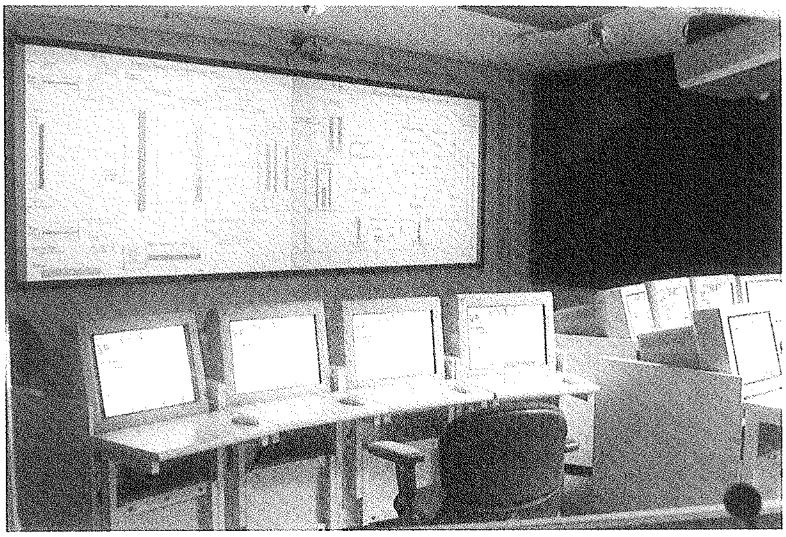


Fig.2.5-6 Overview of interface system developed with Ecological Interface Design concept

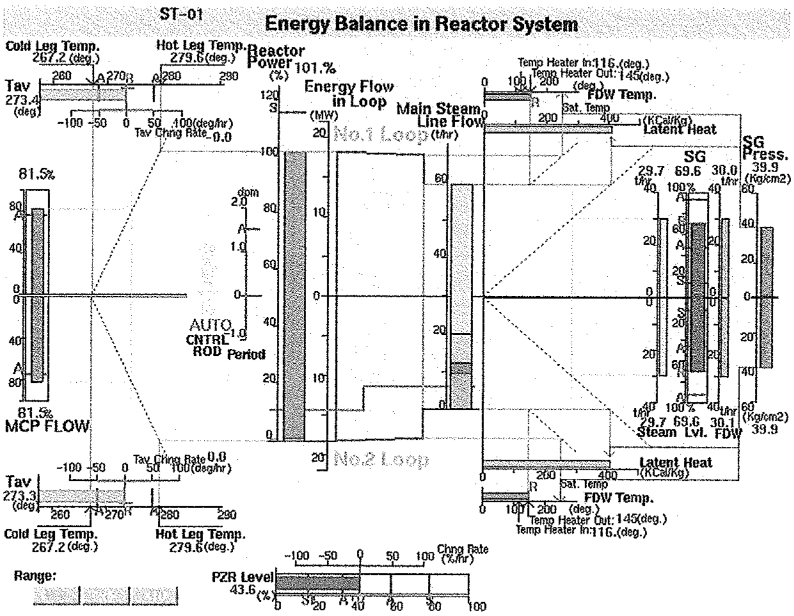
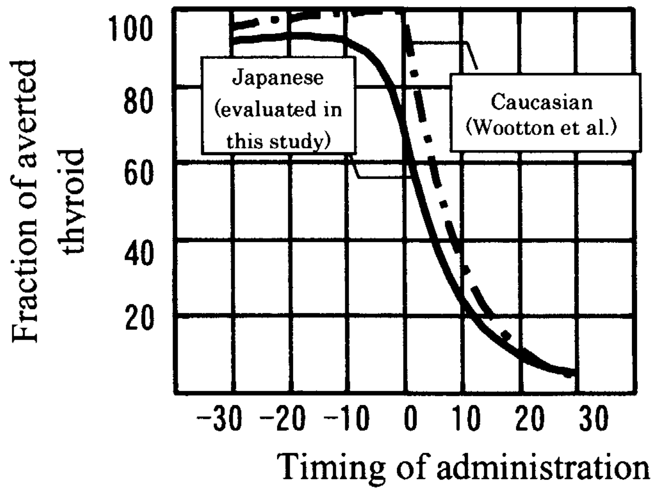


Fig. 2.5-7 Graphical display format representing higher level information

- Energy Balance in Rector System -



(Administration of stable iodine:130mg of KI for adult.
Inhalation of radioactive iodine: ^{131}I during 0-1 h)

Fig. 2.5-8 Effectiveness of iodine prophylaxis. (17)

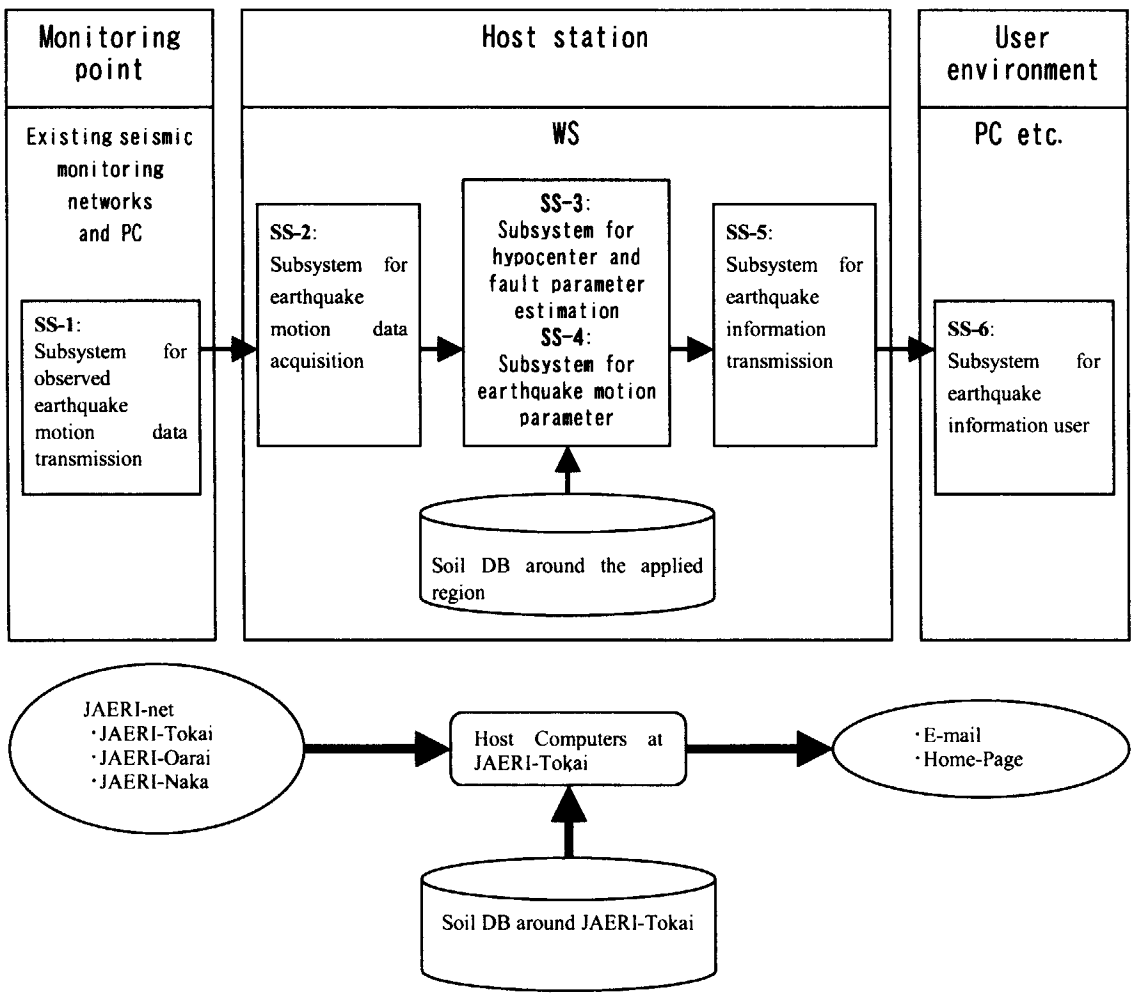


Fig.2.5-9 Configuration of the seismic emergency information system(21)

3. NUCLEAR FUEL CYCLE SAFETY RESEARCH

On reflection of the criticality accident occurred in the Tokai Plant of JCO Co, Ltd., hereafter called "JCO accident," the safety culture, i.e., the basic idea to secure nuclear safety, is claimed further to take root and spread throughout the nuclear industry. The safety is essentially important for the people to accept the policy of nuclear power utilization. Concrete preventive measures against criticality, fire and explosion accidents, and unplanned leakage of radioactivity into the environment are required to avoid any next accident like the JCO accident taking place. For JAERI, as a research institute on atomic energy of Japan, it is important to study the technological basis for securing nuclear safety and the phenomena of those accidents in details, and to work out how to cope with them for prevention and mitigation. Providing with idea and data useful for establishing safety regulation and technical criteria is required to recover the confidence of the people in Japan.

JAERI conducts safety researches on criticality safety, process safety and radioactive waste disposal safety, as tabulated in **Table 3-1**, to make a significant contribution toward securing the safety of nuclear fuel cycle system involving various nuclear fuel handling facilities. The study on anti-corrosion of commercial reprocessing equipment is also conducted as a safety research. The research for elucidation of the technical aspect of JCO accident continues.

The knowledge perfection has been pursued through data acquisition with cooperative experiments and information exchange, taking advantage of the international cooperation with IPSN (Institut de Protection et de Sûreté Nucléaire, France), CIRP (China Institute for Radiation Protection) and OECD/NEA (Nuclear Energy Agency), and the domestic collaboration with universities, public cooperation and private companies. Fairly great percent of the budget for the safety research has been provided as the special account of the government under contract with the regulatory offices of nuclear fuel and waste disposal related facilities.

Our research activities, mostly defined in *the five-year program of safety research on nuclear facilities and radioactive waste*, intend to support the activity of Nuclear Safety Commission. Some examples are for securing the safety operation of nuclear material handling facilities, and for securing the safety of plutonium utilization and geological disposal of the waste bearing long-lived nuclides from the long-term point of view.

Table 3-1 Safety Research on Fuel Cycle Facilities and Waste Disposal

Research Field	Subject Matter
Criticality Safety	Criticality characteristics and criticality accident phenomena of fuel solutions are studied with the experiments using critical experiment facilities STACY and TRACY. Burnup credit is studied to apply to spent fuel handling facilities such as reprocessing, storage and transportation pursuing the further advanced application technology.
Process Safety	Confinement of radioactive material in accidents is studied in focusing on the volatile nuclides like iodine generated by fission and on the dust by fire. The behavior of long-lived nuclides including Pu, Np, Tc and others is studied under the normal operation of commercial reprocessing flow-sheet. Advanced technology to control them more effectively is also a research subject.
Radioactive Waste Disposal Safety	Characteristics of the multiple engineered and natural barriers for isolation of nuclear waste are studied in focusing on the migration behavior and the chemical reaction of trans-uranium and mobile nuclides mainly with underground water. Long-term endurance of engineered barrier is also a research subject. Uncertainty of long-term safety analysis is studied to develop the safety assessment method of waste disposal into both the shallow land and the deep geological matrix.
Anti-Corrosion of Reprocessing Equipment	Corrosion controlling parameters and the prediction technology of material life are studied taking the fuel dissolver and the acid recovery evaporator as examples. Development of new anti-corrosion materials is also a research subject.

3.1 Criticality safety research *

3.1.1 Development of criticality safety control technology

In nuclear fuel processing facilities, solution state uranium and plutonium fuels are treated at various process stages. Criticality safety design is conducted for each fuel handing condition so that an inadvertent criticality has to be prevented in any cases. If several equipments are arrayed in a room or a cell, neutron interaction has an effect of increasing fission reaction. How to deal with this effect is usually discussed as a modeling method of criticality safety analysis.

On the other hand, from a view point of criticality safety control, development of a technique monitoring how far a fuel handling equipment is from criticality (degree of subcriticality) would enable us to detect fluctuation of process conditions, which can improve safety of the equipment.

In Japan Atomic Energy Research Institute, a fundamental research regarding critical conditions of solution fuels and reactivity effect of arrayed equipments is performed and development of criticality analysis systems as well as collection of systematic critical data are being carried out. In addition, an experimental study that is focused on a development of subcriticality measurement technique applicable in a nuclear fuel facility is performed.

(1) Neutron interaction effect among several equipments

A modeling technique for criticality safety analysis that has a neutron interaction effect among array of several equipments is rather complicated unlike a simple geometry. Thus, it is important to validate a reliability of calculation codes by using systematic criticality data.[1] For this purpose, we have commenced criticality experiments that use low-enriched uranyl nitrate solution fuel at STACY (static critical experiment facility) of Japan Atomic Energy Research Institute. The core configuration is shown in **Figure 3.1-1.** In this experiment, changing a distance between two slab core tanks (35 cm-thickness, 70 cm-width, 150 cm-height), we measured the critical solution heights and estimated the neutron interaction effects between two tanks.

As shown in **Figure 3.1-2**, as the distance between the core tanks becomes wider, the critical solution height increases and the neutron interaction effect decreases. Reactivity effects calculated by the code named SRAC-TWOTRAN and the JENDL-3.2

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* A portion of this work was performed under a contract from the Ministry of Education, Culture Sports, Science and Technology.

nuclear data library reproduce the measured values relatively well.

(2) Subcriticality experiment method

A real time measurement of subcriticality requires a short-time measurement of neutron flux distribution in a subcritical state and a quick detection of change due to the system fluctuation as well. For the pulsed neutron method, a detailed information on spatial flux distribution change is needed. A measurement system which installs a position sensitive proportional counter (PSPC) was developed under a collaborative work with Nagoya University, and this system was applied to experiments in the slab core tank (28-cm-thickness, 70 cm-width) and the 80 cm-diameter cylindrical core tank at the STACY facility. This counter shortens the measurement time compared with a conventional counter scanning method.

In the 80 cm-thickness cylindrical core tank, the PSPC was used for subcritical measurement experiment of the pulsed neutron method, and time-decay of neutron flux distribution was measured with the time-step of 50 msec. An example of measurement result is shown in **Figure 3.1-3**. Time-decay of neutron flux distribution can be measured at once with the PSPC. The PSPC will be applied to subcritical measurements such as pulsed neutron method in the future.

(3) International collaborations

To promote an effective experimental research on criticality safety, active international collaborations have been performed. The information on critical experimental data, criticality safety evaluation methods and so on have been exchanged with the IPSN (Institut de Protection et de Sûreté Nucléaire, France) in France.

In Valduc facility of IPSN, critical data of UO_2 powder that is necessary for criticality safety evaluation of uranium fabrication facilities were measured from 1983 through 1987 as MARACAS experiment project. This experiment used low moderated mixture of 5 wt.% enriched UO_2 and water (H/U=2～3). With the widely used Monte Carlo code MCNP and the evaluated nuclear data library JENDL-3.2, criticality analyses were performed to validate an applicability and a feasibility of the calculation technique. The outline of the calculation results is shown in **Figure 3.1-4**. The k_{eff} values calculated by MORET (French Monte Carlo code) and MCNP do not show the dependence of H/U, and these calculation methods give appropriate result even for low moderated systems.

As a multilateral international collaboration, taking part in the working group of the International Criticality Safety Benchmark Evaluation Project (ICSBEP) under the Working Party on Nuclear Criticality Safety (WPNCS) of OECD/NEA, we have obtained information of criticality experiments of each participating country and have started to evaluate subcritical experiment data.

Criticality analyses and sensitivity analyses of critical factor uncertainties were performed for the criticality data of 10 wt.% enriched uranyl nitrate solution which were obtained in basic slab core configurations water-reflected and unreflected, and basic cylindrical core configurations reflected by concrete, borated concrete, or polyethylene in the STACY. These results are included in the International Handbook of Evaluated Criticality Safety Benchmark Experiments by ICSBEP. [2-6]

In addition to ICSBEP, we contribute to the Expert Group on Burn-up Credit Criticality (EGBUC) under WPNCS and obtain a lot of valuable information to introduce burn-up credit into Japanese spent fuel handling facilities.

(4) Preparation of criticality safety data

Nuclear Criticality Safety Handbook was published in 1988 for reference to safety reviewing of the Rokkasho Reprocessing Plant (RRP), of which operation is scheduled to start in 2005. *Nuclear Criticality Safety Handbook, Version 2* was released in 1999, which will be expectedly used in criticality safety control of RRP.[7] Outstanding features of *Version 2* are as follows: a) It exemplified safety margins related to modeled dissolution and extraction processes of reprocessing plant; b) It described a simple method to evaluate the total fission number, which is directly connected to people's exposure, of a postulated criticality accident. c) It depicted basic design and idea for setting the alarm system that minimizes the disaster induced by a criticality accident. d) It included research findings on the modeling of fuel systems for safety evaluation.

In a reprocessing facility where nuclear fuel solution is processed, one could frequently observe a series of power peaks with the highest peak right after a criticality accident. The criticality alarm system (CAS) is designed to detect the first power peak and warn workers near the reacting material by sounding alarms immediately. Consequently, exposure of the workers would be minimized by an immediate and effective evacuation. Therefore in the design and installation of a CAS, it is necessary to estimate the magnitude of the first power peak and to set up the threshold point where the CAS initiates the alarm. Furthermore, it is necessary to estimate the level of potential exposure of workers in the case of accidents so as to decide the appropriateness of installing a CAS for a given compartment.

A simplified evaluation model to estimate the minimum scale of the first power peak during a criticality accident is derived by theoretical consideration of neutronic-thermal behavior for use in the design of a CAS to set up the threshold point triggering the alarm signal.[8] Another simplified evaluation model is in the same way derived to estimate the maximum scale of the first power peak for use in judgment of appropriateness for installation of a CAS. Both models are shown to have adequate margin in predicting the minimum and maximum scale of criticality accidents involving a high ^{235}U enriched uranium solution by comparing French CRAC experiment data as shown in Fig.3.1-5. As for those involving a low ^{235}U enriched uranium solution, these evaluation models are also verified by comparing together with TRACY experiment data.

For increasing burn-up, initial uranium enrichment should also be increased. The initial uranium enrichment of light-water-reactor fuels is limited to 5 wt.% now, however, it is observed to exceed 5 wt.% in the future. Therefore, criticality experiments applying 5 to 10 wt.%-enriched uranium fuels were analyzed with a combination of MVP code and JENDL3.2 library. The results of the calculated neutron multiplication factor (CNMF) are presented as a bar graph in **Figure 3.1-6**, which shows that the CNMF ranges from 1.02 to 0.99, proving the effectiveness of this calculation method.

3.1.2 Criticality Safety Evaluation of Irradiated Fuel

(1) Preparation of ORIGEN library based on JENDL3.2 [9]

A set of new ORIGEN2 libraries "ORLIBJ32" are prepared, based on the Japanese Evaluated Nuclear Data Library JENDL-3.2. The libraries contained one-grouped cross sections, variable actinide cross sections and decays, and fission yields. The libraries intend to calculate assembly-averaged isotopic compositions for LWR fuels, esp. 17×17-typed PWR fuel and 8×8- or 9×9-typed BWR fuel, and core-averaged isotopic compositions for FBR fuels in several types. LWR libraries were evaluated by comparing the calculations with the latest post irradiation examinations made in JAERI. The evaluation showed improved features of the libraries for many isotopes. See **Figure 3.1-7** for the case of ^{235}U. Evaluation of the FBR libraries was carried out by the comparison between new and old libraries of FBR. The calculated weights of several isotopes presented large differences, which results depend on the neutron spectrum of the reference system.

(2) Correction factors for safe-side prediction of isotopic composition [10]

The isotopic composition calculations were performed for 16 spent fuel samples from the Obrigheim PWR reactor and 55 spent fuel samples from 7 PWR reactors. Four methods were applied: the SAS2H module of the SCALE4.4 code system with 27, 44 and 238 group cross-section libraries and the SWAT code system with the 107 group cross-section library. The four kinds of calculation results were compared with the measured data. For convenience, the ratio of the measured to calculated values was used as a parameter. Based on the ratios for the combined 71 samples, the correction factors that should be multiplied to the calculated isotopic compositions were generated for a conservative estimate of the neutron multiplication factor of a system containing PWR spent fuel, taking burn-up credit into account. **Table 3.1-1** shows the obtained correction factors only for uranium and plutonium isotopes.

(3) Benchmark calculations comparison of irradiated BWR fuel [11]

Phase IIIA Benchmarks were conducted by the Expert Group on Burn-up Credit under the auspices of the Nuclear Energy Agency of the Organization for Economic Cooperation and Development (OECD/NEA). The benchmarks were intended to confirm the predictive capability of the current computer code and data library combinations for the neutron multiplication factor (k_{eff}) of a layer of irradiated BWR fuel assembly array model. In total 22 benchmark problems were proposed for calculations of k_{eff}. The effects of following parameters were investigated: cooling time, inclusion/exclusion of FP nuclides and axial burn-up profile, and inclusion of axial profile of void fraction or constant void fractions during burn-up. Axial profiles of fractional fission rates were further requested for five cases out of the 22 problems. Twenty-one sets of results were presented, contributed by 17 institutes from 9 countries. The relative dispersion of k_{eff} values calculated by the participants from the mean value was almost within the band of ± 1 %Δk/k. The deviations from the averaged fission rate profiles were found to be within ± 5 % for most cases. JAERI contributed the benchmark activity as the coordinator.

(4) Criticality data of curium isotopes [12]

Critical masses of three curium isotopes, ^{245}Cm, ^{246}Cm and ^{247}Cm, were calculated with a combination of the JENDL-3.2 and a continuous energy Monte Carlo neutron transport code, MCNP4A. The subcritical masses corresponding to the neutron multiplication factor k_{eff} = 0.9 and 0.8 were also computed in the same way. The subcritical masses that correspond to k_{eff} = 0.9 for ^{246}Cm metal and $^{246}CmO_2$ with a 30-cm-thick stainless steel reflector were computed as 25.2 kg and 41.8 kg, respectively. The minimum critical mass for ^{245}Cm was obtained as 65.6 g in a sphere of a homogeneous mixture of granulated ^{245}Cm metal and water surrounded by a fully thick water reflector, which was compared other results in **Table 3.1-2**. The corresponding quantity for ^{247}Cm was found to be 2.19 kg. The critical masses of ^{245}Cm, ^{246}Cm and ^{247}Cm metals were computed also for reference by replacing the JENDL-3.2 with the ENDF/B-VI; they were reduced by 23 %, 45 % and 2 %, respectively, from each corresponding value, which revealed a large dependence of the results on the evaluated nuclear data libraries. The calculations were made for revision of the ANSI/ANS-8.15, *the American National Standard for Nuclear Criticality Control of Special Actinide Elements.*

3.1.3 Thermo-Nuclear Behavior of Criticality Accidents

To evaluate the consequences of criticality accidents, such as the one occurred at the JCO company on the 30th September, 1999, it is necessary to understand the mechanism of the accident, and estimate the power, temperature and radiation profiles during the accident. At JAERI, a series of super-critical experiments have been conducted using TRACY(Transient Experiment Critical Facility) with 10 wt% ^{235}U enriched uranyl nitrate aqueous solution to simulate the criticality accident, and the characteristics of the accident is being investigated.

In this section, the results of reactivity analysis to evaluate the reactivity feedback mechanism of TRACY, and the activity relevant to the investigation of the JCO criticality accident are presented.

(1) Reactivity Analysis

For a criticality accident of fissile solution, the feedback reactivity consists of the temperature effect and the radiolytic gas void effect. The reactivity analysis of the TRACY super-critical experiments has been performed to evaluate those feedback reactivity effects. [13]

The experiments with step inserted reactivity of 0.4 to 2.9 dollars have been employed in the present analysis. The total feedback reactivity was obtained by subtracting the inserted reactivity from the total reactivity, which was given by the inverse kinetic analysis. Then, the void feedback reactivity was estimated by subtracting the temperature reactivity, which was estimated from the temperature change, from the total feedback reactivity.

Figure 3.1-8 shows the component of the feedback reactivity at the power peak obtained by the above procedure. It can be seen from the figure that the void effect is not negligible for the inserted reactivity over than 1.5 dollars. In addition, while the temperature feedback reactivity increases with inserted reactivity of 2.5 to 3.0 dollars, the void reactivity stays almost constant. The latter is caused by that the growth of the void is restricted by the pressure accompanied with the void production.

(2) Investigation of the JCO Criticality Accident

Static and kinetic analyses of the precipitation tank were performed with MCNP-4B and AGNES2 codes, respectively.[14] In the JCO accident, uranium solution was fed to an unsafe-shape vessel, and the criticality occurred. It is estimated that the maximum inserted reactivity is 3 dollars, and the insertion rate is 0.2 dollar/sec. This condition is almost same as the one of the TRACY ramp-feed experiment, in which the reactivity is inserted by continuous feed of fissile solution. Then, the power profile of the TRACY experiment will become similar to that of the JCO accident. Thus, the TRACY experiment is used as a reference for the evaluation of power behavior in the JCO accident.

To evaluate the power behavior in the accident, kinetic analyses using a one-point kinetics code AGNES2 and the quasi-steady state method were conducted. Those methods were validated using the TRACY experiment data. The AGNES2 code is a one-point kinetics code for the analysis of criticality accidents in fissile solution. This code can calculate the radiolytic gas void feedback effect as well as the temperature feedback one. The kinetics parameters such as the temperature and void coefficients of reactivity, the effective delayed neutron fraction and the prompt neutron lifetime were calculated using neutronic calculation codes.

The result of validation of the AGNES code for the TRACY ramp-feed experiment is shown in **Figure 3.1-9**. The experiment was performed with feeding the uranyl nitrate solution up to the solution level equivalent to the insertion reactivity of about 3 dollars. Due to the production and disappearance of the radiolytic gas void, the power oscillates as shown in the figure. The calculated power profile reproduces the experiment well, then it is concluded that the model and parameters used in the code are appropriate.

The power profiles in the first 200 seconds in the JCO accident calculated with the AGNES2 code are shown in **Figure 3.1-10**. For the calculation, the inserted reactivity of 1.5 and 3.0 dollars that are estimate as the minimum and the maximum one in our investigation was used. The circles in the figure show the power profile obtained from the measured data of a gamma-ray area monitor installed in the JCO site. It was confirmed through the TRACY experiment that the gamma-ray area monitor followed the nuclear power level. The open circles exist between the two calculated power profiles, it means that the calculation reproduces the accident. Although the first power peak varies with the inserted reactivity, the number of fission at the end of the first pulse is almost constant, that is about 5×10^{16} fissions. The power profile after 200 seconds largely depends on the capability of the cooling jacket that surrounded the accident tank. The mock-up test is being performed at JAERI to measure the cooling characteristics of the jacket.

3.1.4 Source term evaluation in criticality accidents

In criticality accident in the fuel solution, release of radiolysis gases and radioactive materials from the fuel solution to the gas phase will be expected, as shown in **Figure 3.1-11.** Since volatile nuclides, such as radioactive noble gases and iodine species, have high release ratio from the fuel solution to the gas phase, quantitative estimation of generation and release behavior of the nuclides are very important for evaluating public dose under the accident. Therefore, data concerning the release and transport of the radiolysis gases and radioactive nuclides has been acquired during simulation experiments in TRACY. (15) There is a vent-gas line connected to TRACY core tank and vent gas, including the gases and nuclides, circulates through the line. To observe release and transport behavior of the gases and nuclides, the line is equipped with gas concentration measurement devices for measuring radiolysis gases and some samplers for collecting the radioactive nuclides.

The plus and linear correlation were observed between the energy release from the fissions and the released molecules of the hydrogen (radiolysis gas) and the slope, in other word, G-value of the hydrogen was estimated about 0.8 molecules/100 eV.

The gas phase in the TRACY core tank is ventilated by vent-gas, constantly. To evaluate the release behavior of the radioactive iodine (^{131}I and ^{133}I) from the fuel solution to the gas phase, simultaneous equations representing the time variation of number of atoms of the nuclide in the fuel solution phase and the gas phase in the core tank were assumed by considering release coefficient of the nuclide and the ventilation constant. The time variations of number of atoms of the nuclide in the both phases were calculated and the release ratio (%), which was defined as the ratio total produced number of atoms to total released number of the atoms of the nuclides, was estimated by solving the equations simultaneously. (16) The calculated results are shown in **Figure 3.1-12.** At 4.5 h after the withdrawal of the transient rod from the solution, the release ratio of the iodine species were about 0.2%, in case that the transient rod was re-inserted to the solution just after transient criticality, and about 0.9%, in case that the rod was not re-inserted. Since, in the latter case, temperature of the fuel solution and stirring effect in the solution by generated radiolysis gas were held relatively high and large than the former case because of continuous criticality condition, it was attributed that the release ratio in the latter case was larger than the ratio in the former case.

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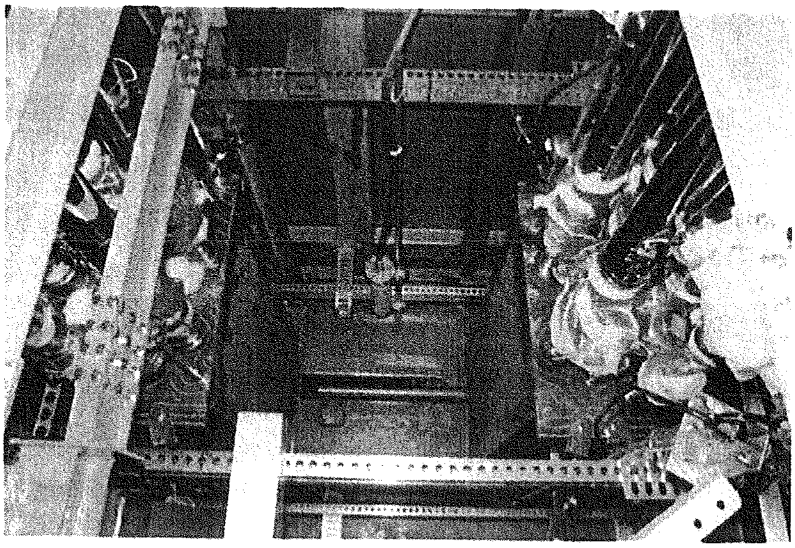

Fig. 3.1-1 Core configuration (Surface-to-surface distance: 100 cm).

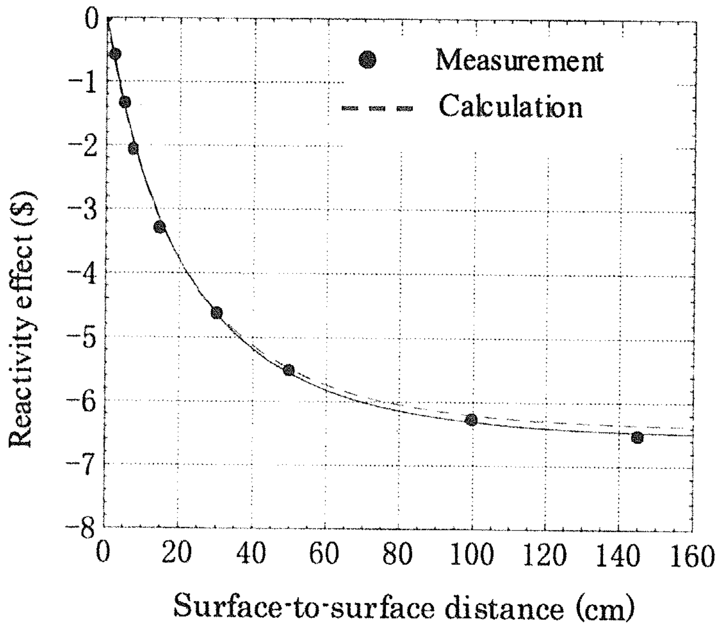

Fig. 3.1-2 Surface-to-surface distance vs. measured reactivity of the interaction system.

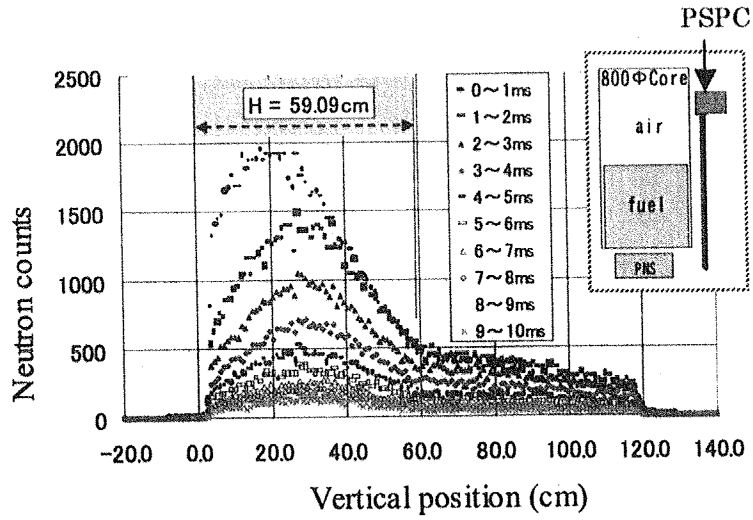

Fig 3.1-3 Neutron flux distribution change after pulse generation.

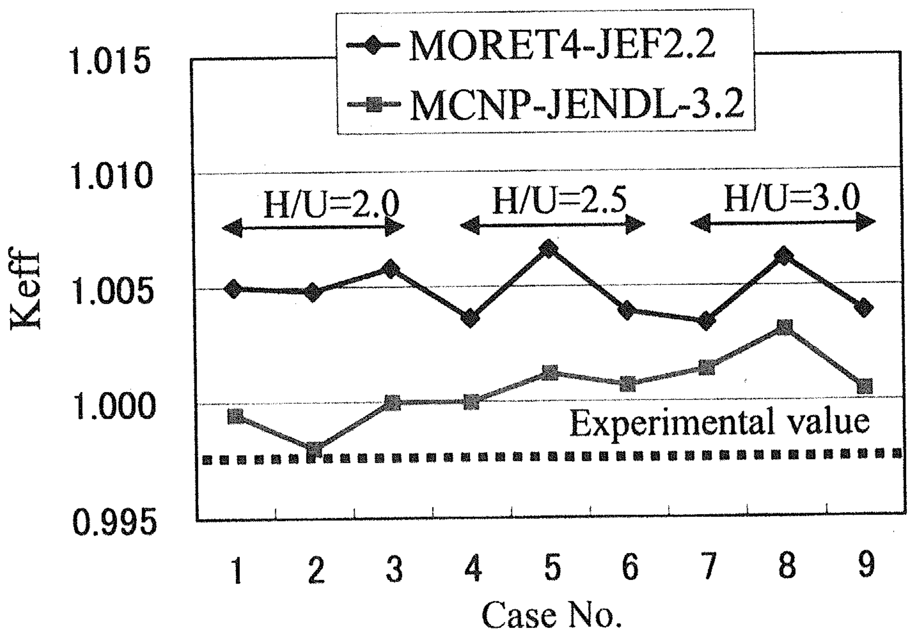

Fig. 3.1-4 Calculation results of MARACAS experiment.

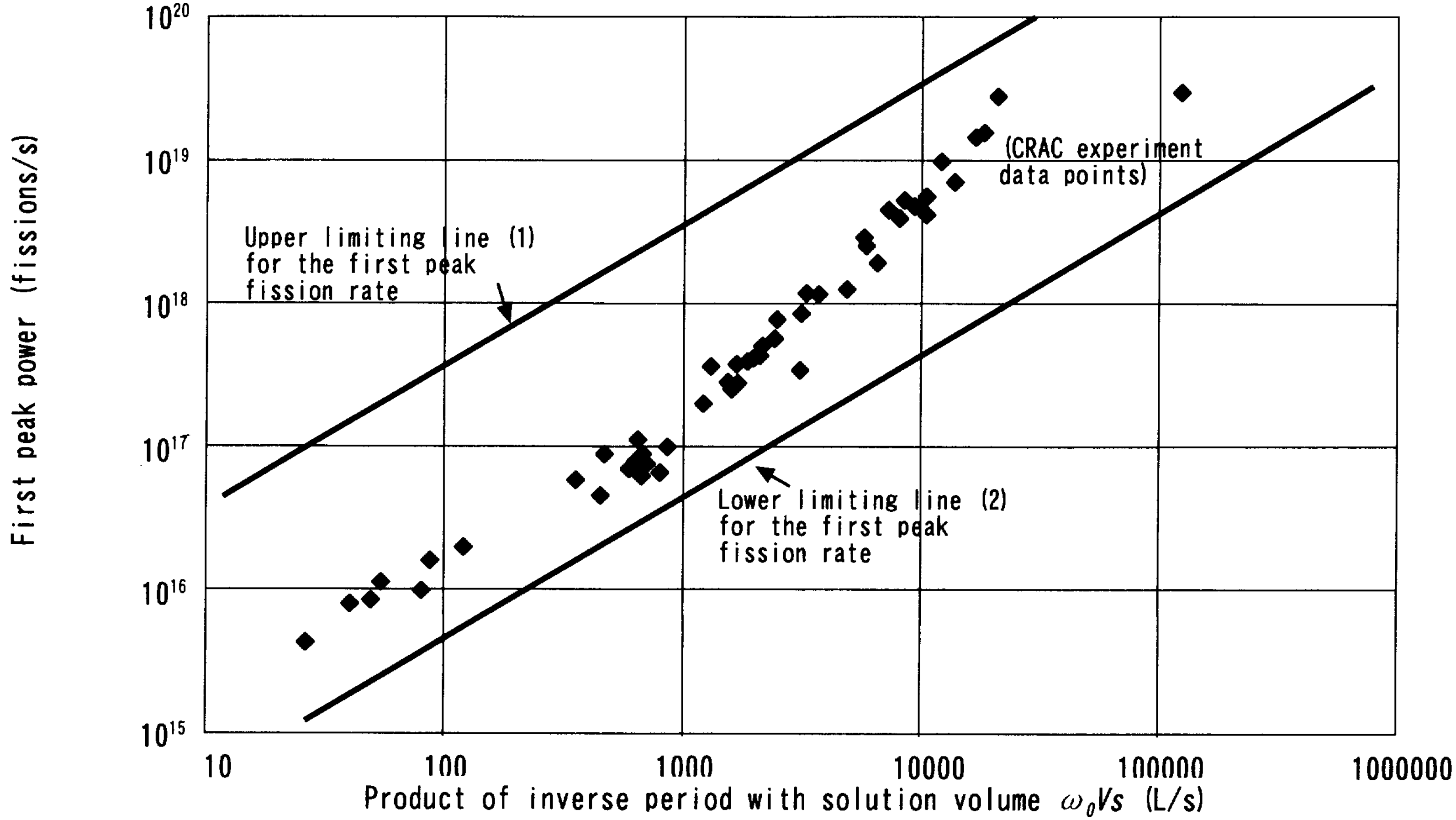

Fig.3.1-5 Comparison of the first power peak fission rates of CRAC experiments with the upper limiting line (1) $P_{MAX} = 3.3\text{x}10^{15}\omega_0 Vs$ and the lower limiting line (2) $P_{MIN} = 4.1\text{x}10^{13}\omega_0 Vs$.

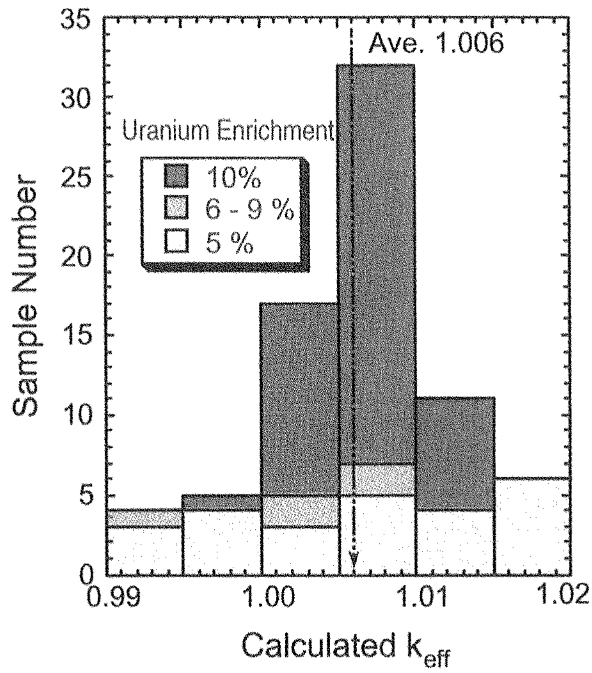

Fig. 3.1-6 Histogram of neutron multiplication factor obtained by analysis of critical experiments with 5-10 wt.% enriched uranium fuel using a combination of MVP code and JENDL3.2 library.

Table 3.1-1 Correction factors for uranium and plutonium isotopes

Nuclide	Tolerance Limit Factor	Data #	Correction Factor			
			SAS2H			SWAT
			27 G	44 G	238 G	107 G
U-234	2.292	25	0.7386	0.7548	0.7581	0.7472
U-235	1.987	71	1.1057	1.1047	1.0749	1.1091
U-236	1.987	71	0.9103	0.9116	0.9140	0.9519
U-238	2.005	65	0.9919	0.9918	0.9923	0.9920
Pu-238	2.060	51	0.7905	0.8404	0.8516	0.9830
Pu-239	1.987	71	1.0901	1.1166	1.0819	1.1104
Pu-240	1.987	71	0.9864	0.9433	0.9274	0.9407
Pu-241	1.987	71	1.1145	1.1858	1.1484	1.1984
Pu-242	1.999	67	0.8986	0.8166	0.8113	0.8882

Table 3.1-2 Comparison of the minimum critical masses of ^{245}Cm metal granules in a water mixture

Author	Nuclear Data Library	Computer Code	Bare		Water reflected	
			Cm mass [g]	Cm conc. [g/L]	Cm mass [g]	Cm conc. [g/L]
Present	JENDL-3.2	MCNP4A	138	10.0	65.6	12.1
Present	ENDF/B-VI	MCNP4A	117	8.1	54.9	11.6
Roussignol	JEF-1	DTF-IV			42.7	
Srinivasan	ENDL-82	DTF-IV	136	12	62	12

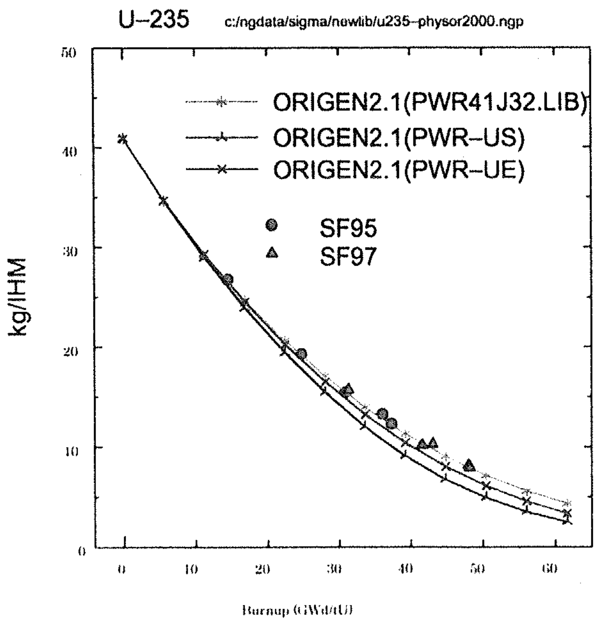

Fig. 3.1-7 Comparison between calculation results for ^{235}U using three libraries.

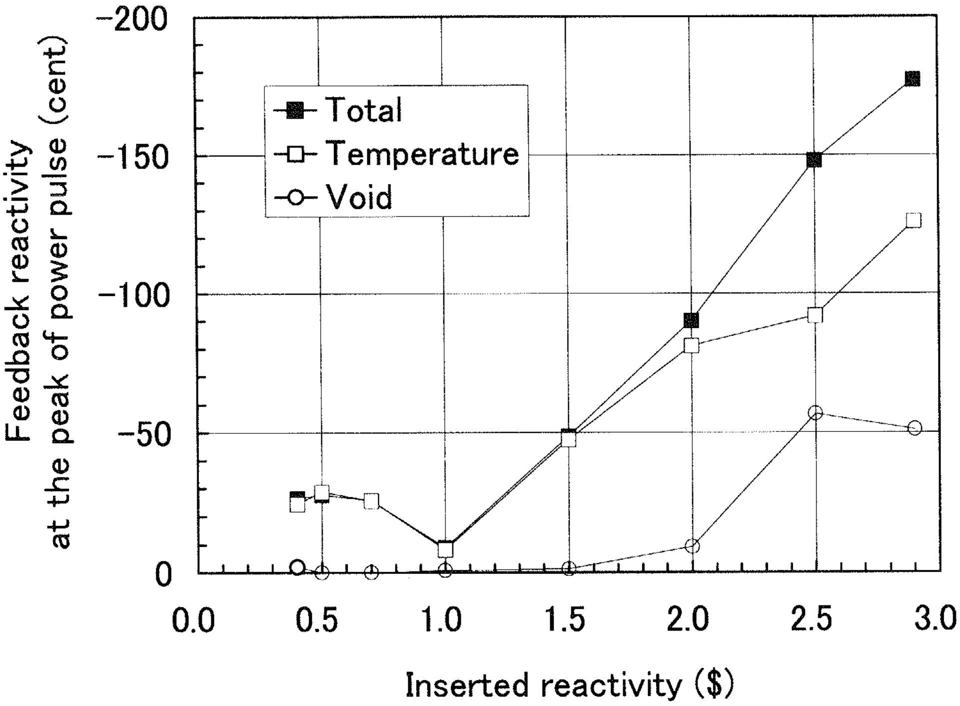

Fig.3.1-8 Component of feedback reactivity at the peak of power pulse
Contribution of void feedback reactivity increases with the inserted reactivity.

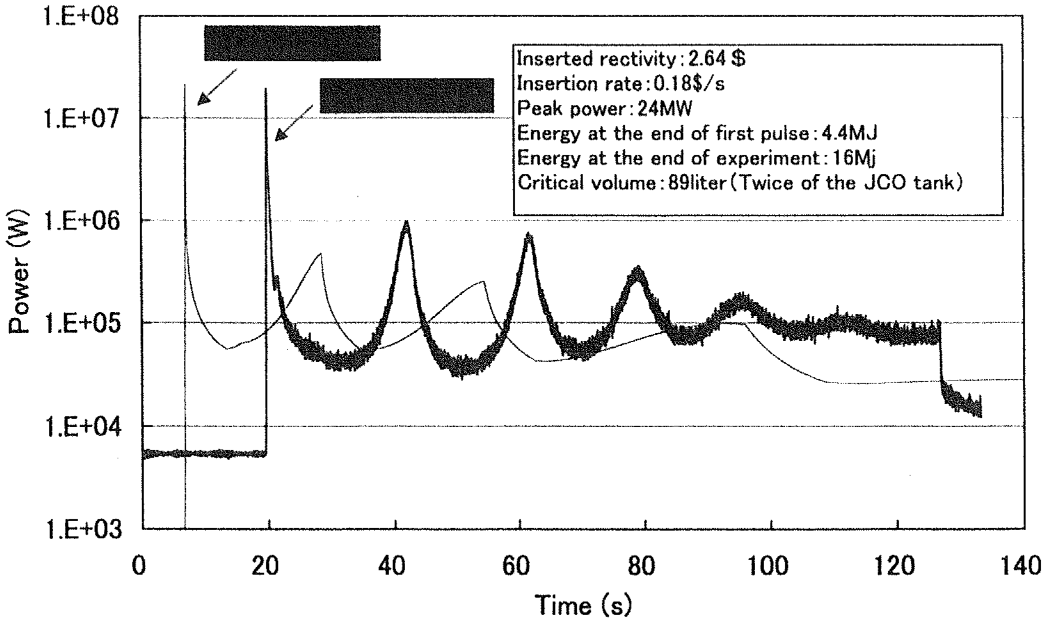

Fig. 3.1-9 Analysis of the TRACY experiment with AGNES2 code
Calculation/Experiment ratios [Peak power : 0.94, Released energy : 0.90]

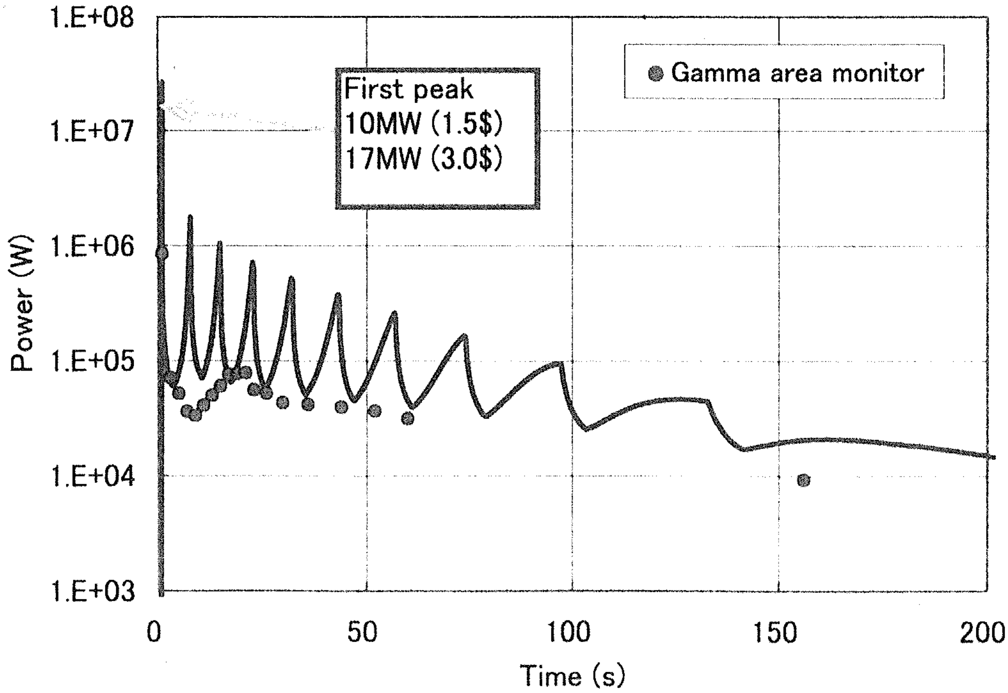

Fig. 3-1-10 Analysis of JCO Criticality Accident with AGNES2 Code

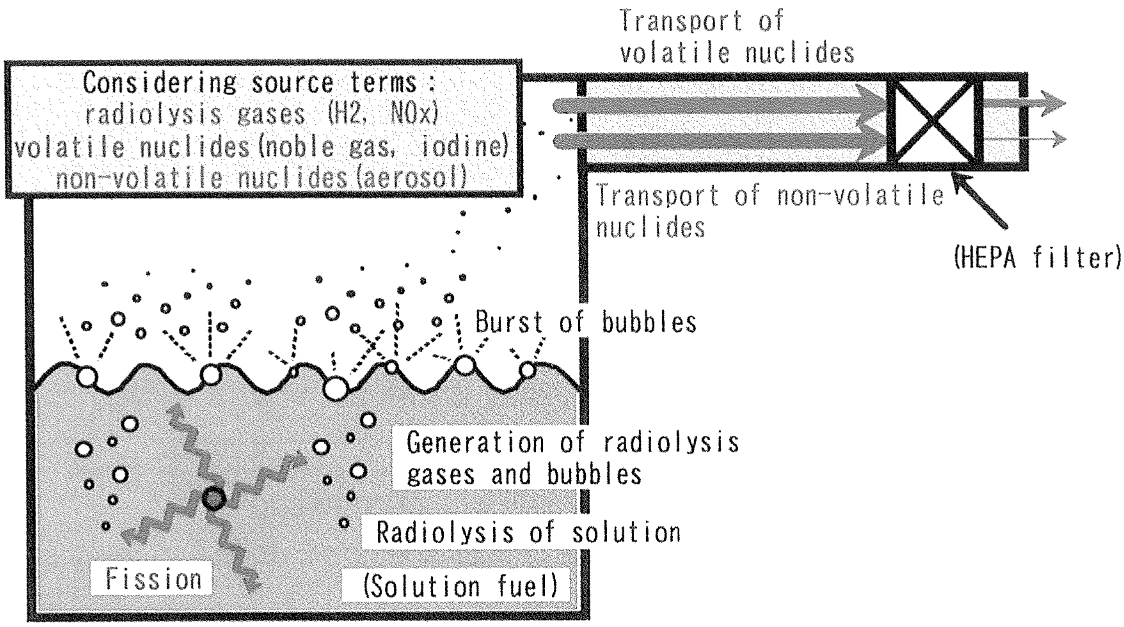

Fig 3.1-11 Sequence of release and transport of radiolysis gases and radioactive materials under the criticality accident in the fuel solution.

Radiolysis gases and radioactive materials are released from the fuel solution to the gas phase, transport in the ventilation system of the facility and are released to the outside of the facility under the criticality accident in the fuel solution.

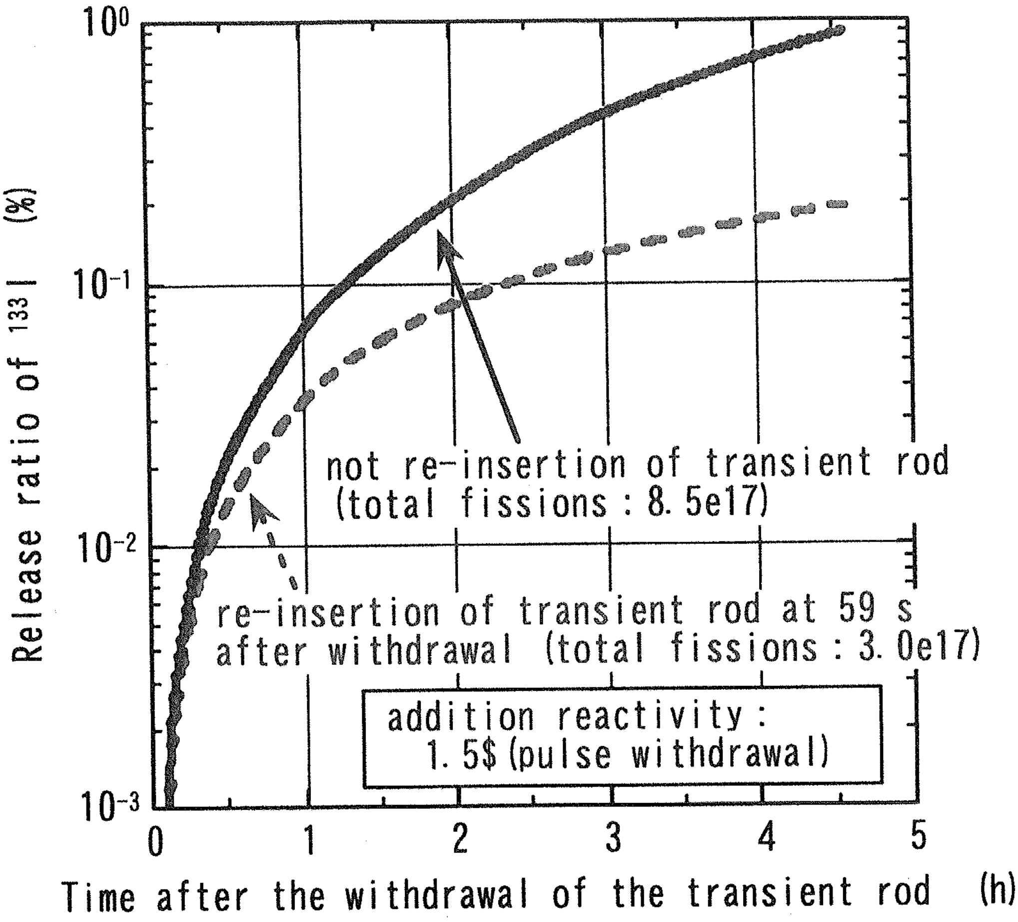

Fig 3.1-12 Estimation results of release ratio of radioactive iodine from fuel solution to gas phase.

At 4.5 h after the withdrawal of the transient rod from the solution, the release ratio of the iodine species were about 0.2%, in case that the transient rod was re-inserted to the solution just after transient criticality, and about 0.9%, in case that the rod was not re-inserted.

3.2 Process Safety Research

Present-day nuclear fuel reprocessing plants employ the PUREX process, which consists of spent fuel dissolution, off-gas treatment, separation by TBP solvent extraction and waste treatment. The process safety for reprocessing should be maintained under normal operation and radioactive materials should be confined within these plants even during fire or explosion accidents.

The chemical process for reprocessing is designed to control radioactive material safely within a confined process area. Although most of radioactive material is confined in the highly radioactive waste, some radioactive materials can easily be mixed into process products, and others like volatile radioactive materials are diffusive. Therefore, it is necessary to study the behavior of nuclides in the process as a whole to assess the safety margin of radioactivity confinement capability.

An advanced reprocessing concept that could enhance the confinement efficiency for long-lived nuclides in a simplified Purex process has been proposed. In the future the objective of this process is to enhance the control of long-lived nuclides in a simplified, future reprocessing plant that can treat high burn-up fuel and spent MOX fuel with a streamlined purification process. Flow sheet tests on the new process have been carried out using spent fuel.

To prove the adequacy of the safety evaluation of ventilation systems in fuel reprocessing plants, the following tasks have been performed to evaluate the confinement capability of radioactive materials under fire or explosion accident. The behavior of smoke arising from burning fire is studied. Not only liquid burnable material, organic solvent, but also solid material, cotton gloves and latex groves, was burnt and the adequacy of ventilation systems, especially the intactness of pre-filter and HEPA filter was examined.

3.2.1 Research on Confinement Safety of Current Reprocessing Process*

In 1998, the experiment with spent fuel began. In 1999 and 2000, spent fuel with a burn-up of 29 and 44 GW·d·t^{-1} was treated in this work. The research facility "NUCEF-BECKY" was utilized for these works. "BECKY" has a small-scale $\alpha\gamma$ cell equipped with key reprocessing equipment, such as a dissolver, extractors, off-gas treatment and effluent treatment systems.

(1) Studies on Dissolution Step, Iodine Stripping Step, and Iodine Removal from Dissolver Off-gas (DOG) during These Steps

As summarized in **Table 3.2-1**, batch-wise dissolution experiments using PWR spent fuel

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* A portion of this work was performed under a contract from the Ministry of Education, Culture Sports, Science and Technology.

with burn-up of 8, 29 or 44 GW·d·t^{-1} have been conducted[1-3]. In **Figure 3.2-1**, the concentration change of U or HNO_3 in dissolver solution is shown as a typical example of dissolution behavior. Although U, Pu, most of FPs were well dissolved, only some parts, 60-70 %, are supposed to be dissolved in the cases of Ru, Mo and Tc. Small amounts of Zr, Pd and Rh were included in insoluble residue, or precipitate, and the existence of crystalline precipitate of Mo and Zr was confirmed by X-ray diffraction pattern [1].

The iodine adsorbent column of AGS (silver nitrate impregnated silica gel) could effectively remove the iodine-129 from the DOG during the spent fuel dissolution [1, 2]. Iodine stripping using NO_2 could effectively remove iodine from the dissolver solution, and two-steps iodine stripping process using potassium iodate (KIO_3) could expel additional iodine from the solution [2]. As summarized in **Table 3.2-2**, the fraction of adsorbed iodine-129 accounted for 62 to 72 % of the total amount of iodine-129 estimated by ORIGEN2. These value were comparable to that determined in the dissolution test [2] of unirradiated UO_2 pellets using iodine-131 as tracer. In the spent fuel test [2], iodine-131, generated by spontaneous fission, was also found in the column.

(2) Studies on Extraction Step

Using respective dissolver solution of spent fuel with burn-up of 8 [4], 29 [1], or 44 GW·d·t^{-1}, flow sheet tests were conducted to simulate the first extraction cycle of a large-scale reprocessing plant, from co-decontamination to Pu barrier. Three miniature Plexiglas mixer-settler contactors in the $\alpha\gamma$ cell were utilized.

In the case of 29 GW·d·t^{-1} spent fuel test [1], two extractive runs, SFEX2-1 and 2-2, were performed to study the influence of the flow rate of main solvent, SFEX2-1 as a standard and SFEX2-2 with reduced flow rate. **Figure 3.2-2**, shows the concentration profiles of U, Pu, Np and nitric acid in the fist mixer-settler representing co-decontamination step and first FP scrubbing step in SEFX2-1, standard solvent flow rate. **Table 3.2-3** summarized the distribution behavior[1] of U, Pu, or Np among U product, Pu product and raffinate. In Run SFEX2-2, the decrease of the flow rate of the main solvent resulted in slight increase of U and Pu content in the raffinate. Tc, known as forming complexes with U, Pu, and Zr is extracted in the de-contamination step and scrubbed at second FP scrubbing steps [4].

(3) Process Simulation Technology

A code system, ARECS [5] (Advanced Reprocessing Evaluation Code System), was developed for the simulation of a chemical process system composed of dissolution (**Fig. 3.2-1**), off-gas treatment, and extraction (**Fig. 3.2-2**) process units. This code system is capable of simulating the behavior not only of U and Pu but also long-lived minor nuclides such as I-129, C-14, Np-237, and Tc-99 that are important in the safety analysis of the nuclear fuel cycle.

3.2.2 Development of Advanced Reprocessing Process

An advanced PUREX process, PARC (Partitioning Conundrum Key) process [2, 4, 6, 7], has been developed for reduction of radioactive wastes volume and minimization of radio toxicity based on the ALARA principle. The recovery technologies for long-lived nuclides are under development, focusing on C-14 and I-129 in the head-end, Np-237 and Tc-99 in the extraction step, and Am-241 in the effluents. PARC process widely utilizes "salt-free" technologies by using organic compounds as reagents. The new partitioning function in PARC process improves the safety and the economy of reprocessing by simplifying the process, while maintaining high separation performance to recycle plutonium in LWRs.

(1) Studies on Recovery Technology for C-14 and I-129 from DOG

AgS [1, 2], or mordenite [2] is employed to capture I-129 as I_2, or C14 as CO_2, respectively. As shown in **Figure 3.2-3**, the adsorbent consisting of hydrogenated-mordenite impregnated with sodium hydroxide [2] showed a capacity to be about 1.4 g-CO_2 / 50 g-adsorbent, which was about 4.3 times higher adsorption capacity than that of hydrogenated-mordenite for the carbon dioxide. Though the capacity was decreased with the increase in concentration of NO_x gas and relative humidity, good repeatability for CO_2 adsorption was obtained in the cold tests. In the spent fuel test [2], C-14 was released during dissolution with the release of Kr-85.

(2) Studies on Advanced Extraction Process

(i) Studied on Np / Tc Separation from U, Pu stream

The first extraction cycle of the PARC process corresponds to the co-decontamination of U/Pu partitioning in the PUREX process, and separates U, Pu, Np, and Tc. This cycle can eliminate the purification steps following the first extraction step in the Purex process, and can provide a much simpler process. Experiments of an advanced extraction process of PARC was conducted by using spent fuel of 8 [4, 6], and 44 GW·d·t^{-1}, in an $\alpha\gamma$ cell as well as **3.2.1**. It was demonstrated that; *n*-butyraldehyde [4, 6] was a selective reductant of Np(VI) in the presence of U(VI), Pu (IV) and Tc (Tc); *iso*-butyraldehyde effectively reduced Pu(IV) to Pu(III) in the U/Pu partitioning step; high acid scrubbing was effective for Tc separation. **(Figure 3.2-4)**

(ii) Studies on Actinide / Lanthanide Separation

Am^{3+}-selective adsorption was studied by using an adsorbent [8], porous silica-polymer composite material supporting nitrogen donor aromatic compound (2,6-bis (5, 6-dialkyl-1, 2, 4-triazin-3-yl) pyridine (Alkyl-BTP)), to develop an actinides / lanthanide separation process. In the case of Ethyl-BTP adsorbent, the value of 60 $ml{\cdot}g^{-1}$ was obtained as adsorption distribution ratio of Am^{3+}. In this case, the value of adsorption selectivity of Am^{3+} against Ce^{3+}, Eu^{3+} or Gd^{3+} was 390, 13, or 9.5, respectively. Changing the alkyl substituent gives much higher distribution ratio.

(3) Studies on "Salt-free" Technology

Butyl amine, a "salt-free" organic compound was employed as solvent washing reagent. It is confirmed that butyl amine compounds [6, 7], carbonate and oxalate, effectively removes HDBP from the used solvent. Butyraldehyde, also being a "salt-free" compound, was easily decomposed by bubbling air in warmed nitric acid solution. Butyl amine was successfully decomposed by Ag mediated electrochemical oxidation [6].

3.2.3 Source Term Evaluation in Fire and Explosion*

When fire accident occurs in the controlled area in the nuclear fuel reprocessing plant, there is the possibility that increase of differential pressure of ventilation filters and breakage of the filters are induced by clogging of smoke on the filters. If confinement capability for radioactive materials of the ventilation system is lowered by the breakage of the filters, increasing amount of the released radioactive materials from the facility to the outside causes enhancement of public dose. Therefore, verification of integrity of the ventilation filters under the accident is very important for guarantee on the safety of the facility. To evaluate the confinement capability of radioactive materials in the ventilation system in the facility under the fire accidents, data on the source terms, such as mass, energy and smoke release rate from burning materials, and clogging properties of the filters are needed. To obtain the information, the verification tests on safety of the cell ventilation system under the fire accidents and estimation with accident analysis code, CELVA-1D, have been examined and performed.

The block diagram of the experimental facility is shown in **Figure 3.2-5**. The mock combustible materials was burned in the first model cell and smoke, which was released from the materials, transported from the first model cell to the ventilation filters by way of the first duct, the second model cell, the second duct and the third duct. The smoke particle size distribution and concentration were measured by collecting the smoke from the first and third ducts by using cascade impactors and air filters. Furthermore, differential pressures of the ventilation filters were also measured. Up to the present, as the mock combustible materials, solid wastes (gloves made of rubber and cloth packed in carton box) and recovered solvent (dodecane and tri-butyl phosphate (TBP)), which are often used and wasted in the nuclear facilities, and bitumen/salt mixture [9] have been considered.

The example of the experimental results for investigation the clogging property of the ventilation filters is shown in **Figure 3.2-6**. In this experiment, gloves made of rubber packed in carton box were used as the mock combustible materials. The differential

* A portion of this work was performed under a contract from the Ministry of Education, Culture Sports, Science and Technology.

pressure of pre-filters strongly rose but the one of HEPA filters did not rise. This means that the pre-filter was useful for preventing the clogging of HEPA filters. The experimental results will be adopted by the safety analysis on the confinement capability of radioactive materials in the ventilation system of the nuclear fuel reprocessing plant under the fire accident.

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Table 3.2-1 Experimental conditions for respective dissolution run

Burn-up / $GW \cdot d \cdot t^{-1}$	Cooling time / y	Run ID	Amount of U element / g	HNO_3 concentration / $mol \cdot l^{-1}$ (volume / l)	Temperature / K
8	19.5	SFD1-1	436.05	5.00 (1.64)	373
		SFD1-2	443.01	4.12 (1.69)	373
		SFD1-3	439.90	3.34 (1.63)	373
29	17	SFD2-1	372.66	5.21 (1.57)	373
		SFD2-2	283.95	3.98 (1.44)	373
		SFD2-3	274.36	4.06 (1.29)	373
44	12.25	SFD3-1	332.66	5.26 (1.59)	373
		SFD3-2	332.48	4.68 (1.64)	373
		SFD3-3	350.04	3.93 (1.61)	373

Table 3.2-2 Summary of off-gas treatment experiments, amount of I-129 captured by the iodine recovery system, units in Bq

Burn-up / $GW \cdot d \cdot t^{-1}$	Origen2 calc.	Dissolution run 1	Dissolution run 2	Dissolution run 3	Iodine stripping	Total	Ratio for total / calc. / %
8	2.97×10^5	4.4×10^4	8.5×10^4	5.9×10^4	2.2×10^4	2.1×10^5	70.8
29	9.86×10^5	2.5×10^5	1.9×10^5	1.9×10^5	5.7×10^4	6.8×10^5	69.3
44	1.78×10^6	3.7×10^5	3.5×10^5	3.6×10^5	2.0×10^5	1.3×10^6	72.0

Table 3.2-3 The results of extraction experiment for 29 $GW \cdot d \cdot t^{-1}$ spent fuel solution, assessed distribution behavior of U, Pu, or Np among U product, Pu product and raffinate

Stream	Run ID.	Fraction of U / %	Fraction of Pu / %	Fraction of Np / %
U product*	SFEX2-1	99.5**	0.00	2.20
	SFEX2-2	102**	0.00	1.22
Pu product	SFEX2-1	0.53	103	1.80
	SFEX2-2	0.35	91.3	4.78
Raffinate	SFEX2-1	0.38 >	1.35	98.0***
	SFEX2-2	0.1 < < 0.41	1.66	103***

* This means U loaded solvent.

** The amount of U^{4+} supplied for Pu reductant is subtracted.

*** These values are assessed from the calculation results by ESSCAR code.

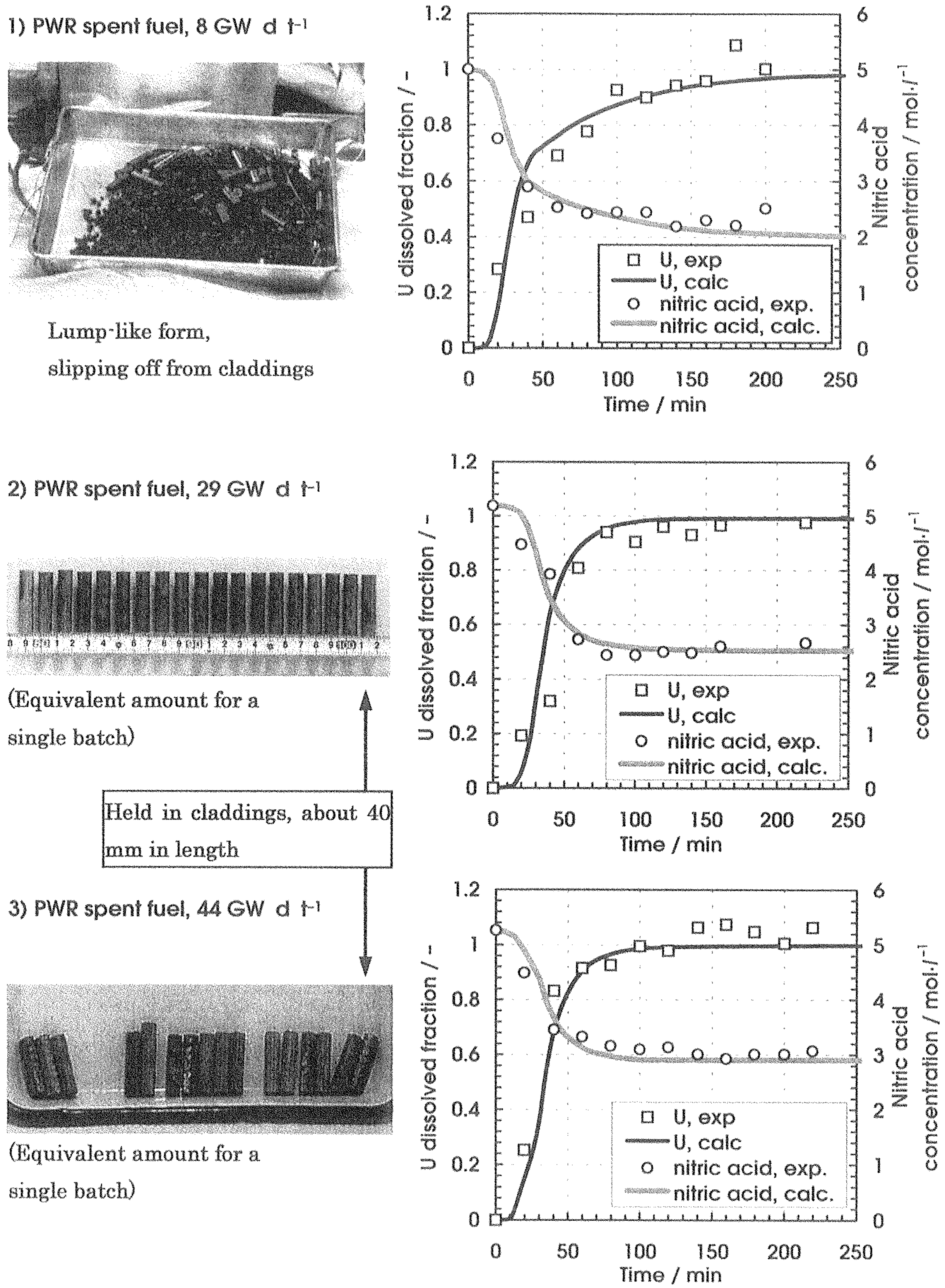

Fig. 3.2-1 Photographs of respective spent fuel used for dissolution, and dissolution curves. Plot symbols for experimental results, and lines for calculated results by ARECS simulation code system. Burn-up: 8, 29, or 44 GW·d·t^{-1}.

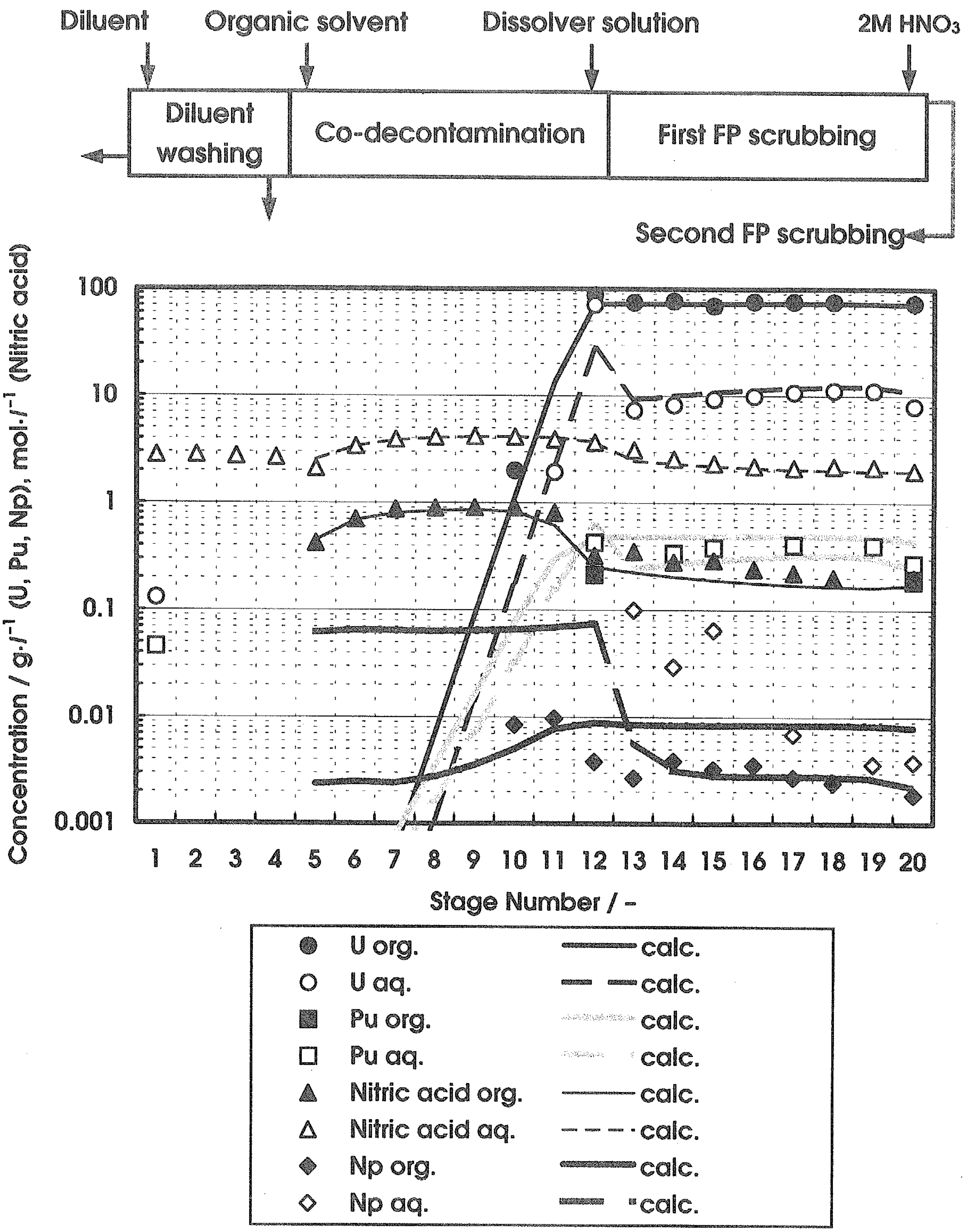

Fig. 3.2-2 The results of extraction experiment for 29 GW·d·t^{-1} spent fuel solution. The concentration profiles for U, Pu, and HNO_3 in organic or aqueous solution of each stage of the first extractor, representing co-decontamination and first FP scrubbing. Plot symbols for experimental results, and lines for calculated results by ARECS simulation code system.

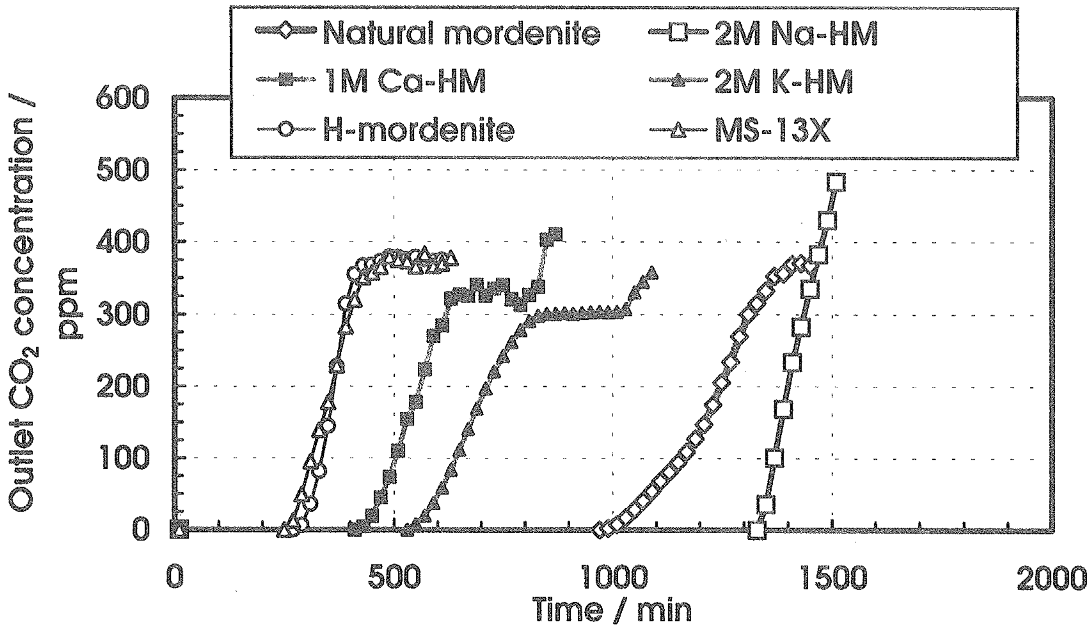

Fig. 3.2-3 The results of CO_2 recovery experiments. Breakthrough curve for each adsorbent, natural mordenite, hydrogenated mordenite (HM), sodium hydroxide (2 mol·l^{-1}) impregnated HM, potassium hydroxide (2 mol·l^{-1}) impregnated HM, calcium chloride impregnated HM, molecular sheaves (MS) 13X.

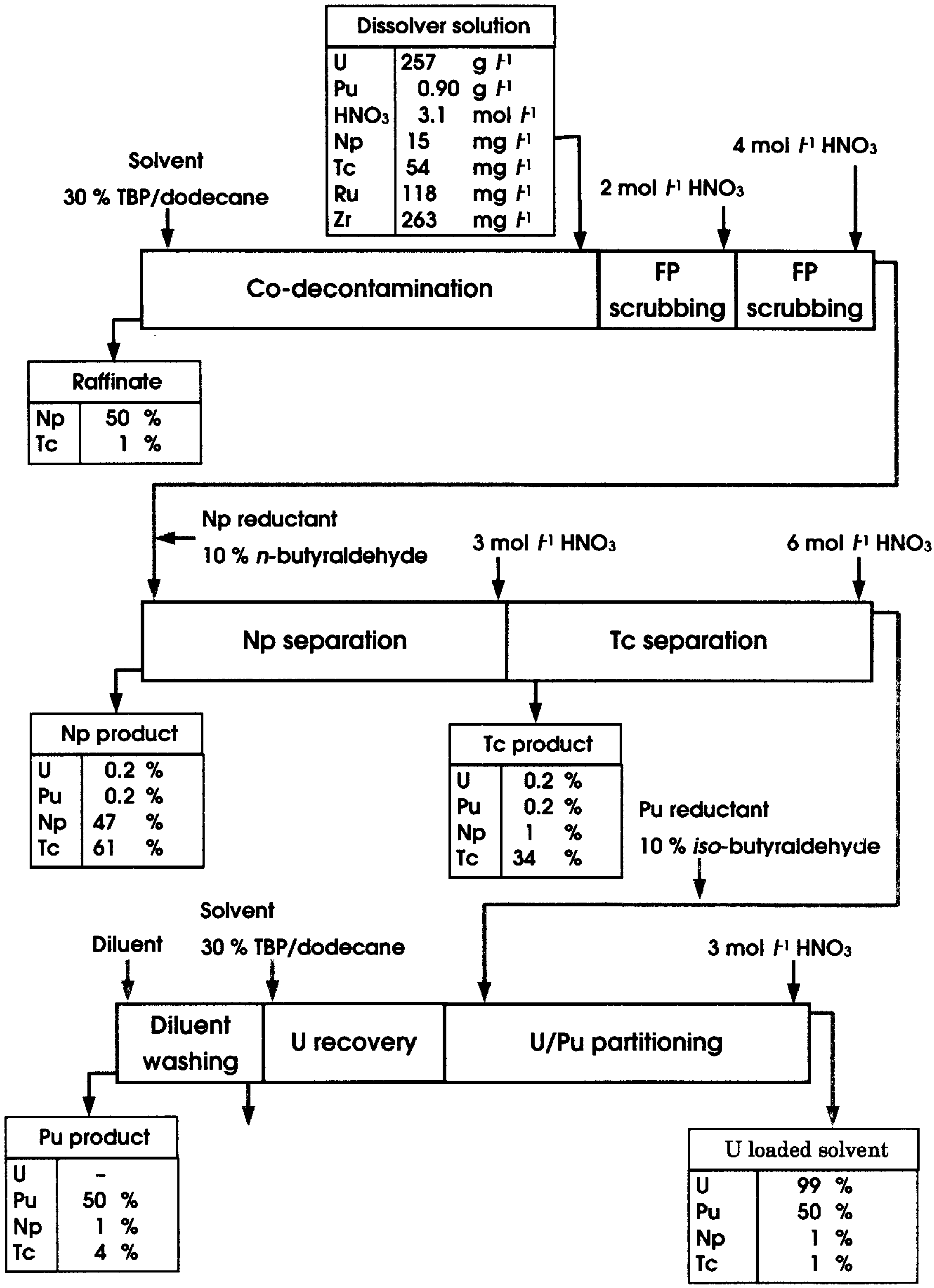

Fig. 3.2-4 The results of extraction experiment for 8 GW·d·t^{-1} spent fuel solution under PARC flow sheet. Distribution of U, Pu, Np, Tc in respective stream.

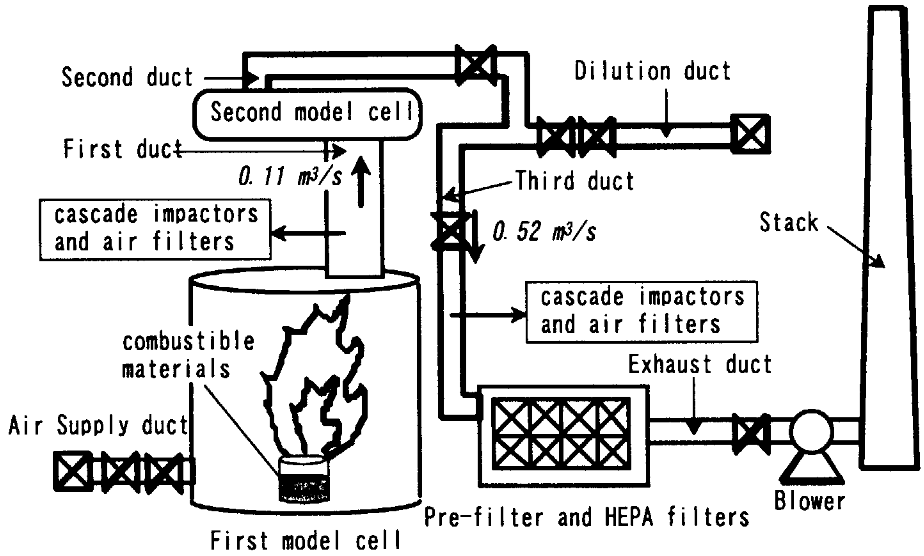

Fig. 3.2-5 The block diagram of the experimental facility. The combustible material was burned in the first model cell and smoke was transported from the first model cell to the ventilation filters. The smoke particle size distribution and concentration were measured by collecting the smoke from the first and third ducts by using cascade impactors and air filters.

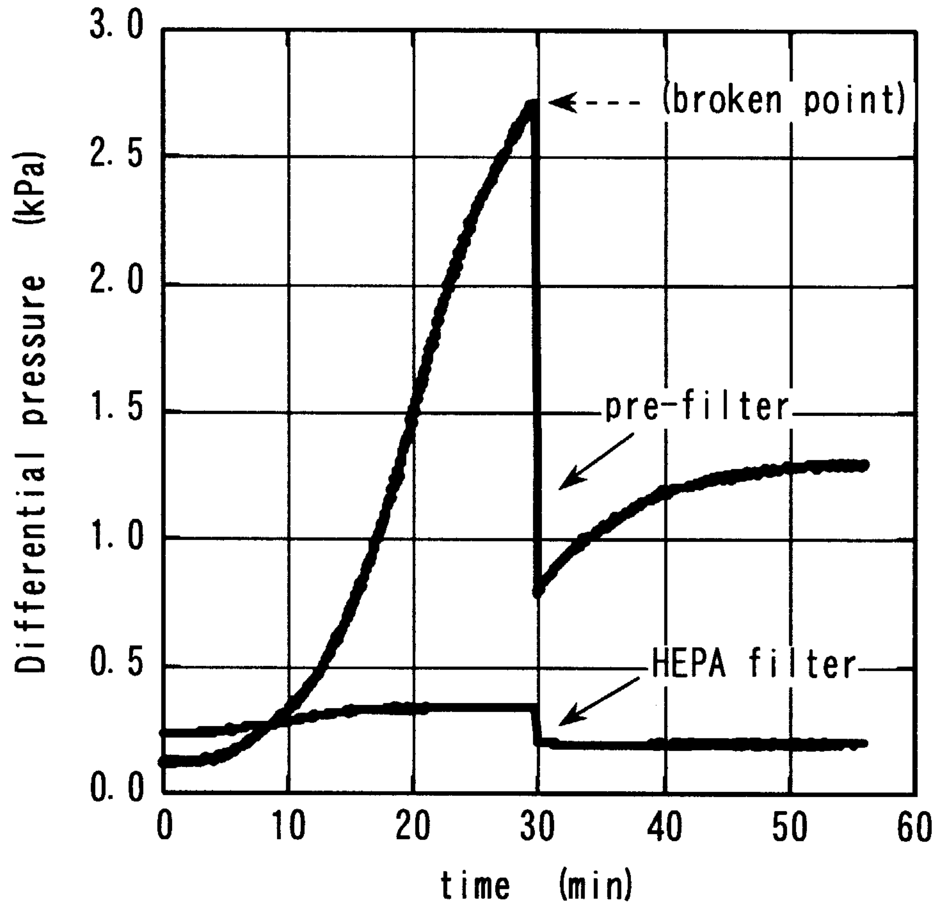

Fig. 3.2-6 Example of the experimental results for investigation the clogging property of the ventilation filters (combustible materials: gloves made of rubber packed in carton box). The differential pressure of pre-filters strongly rose but the one of HEPA filters did not rise. This means that the pre-filter was useful for preventing the clogging of HEPA filters.

3.3 Corrosion resistance of reprocessing component materials

The reliability of structural materials is considered to be one of the most important technological issues on the commercial reprocessing for high burn-up fuels. The long performance of equipment materials used in the commercial PUREX plant of RRP has been conducted by mock-up tests and laboratory tests.(1) It is clarified that the trans-passive corrosion of heat transfer tubes on evaporators made of austenitic stainless steels and the susceptibility to stress corrosion cracking (SCC) on a dissolver made of zirconium are critical issues on the reliability of reprocessing equipment.(2), (3) The susceptibility to these failures increases with amount of TRU and FP elements included in spent fuels, because Np, Pu, Ru, Pd act as strong oxidizers. As shown in **Fig 3.3-1**, the test solution for the mock-up of acid recovery evaporator modified with V instead of Np. The equivalence of corrosiveness of V to it of Np was clarified with corrosion tests under the heat flux control (**Fig 3.3-2**). The corrosion rate of heat transfer tubes of mock-up evaporator depended on the heat flux and the concentration of V in test solutions. Even if low temperatures, the corrosion rate of heat transfer tubes is higher than it of immersion tests at the same surface temperature, because of the formation of oxidizer ions due to the thermal decomposition of nitric acid (**Fig 3.3-3**). The susceptibility to SCC of zirconium was examined by slow strain rate tensile tests (SSRT) in boiling nitric acid solutions. The initiation and propagation of SCC was monitored quantitatively by means of the modified AE detection technique. The development of the modified alloys has been pursued for counter-measures against these problems (**Fig 3.3-4**). It has been found that the intergranular corrosion resistance of stainless steels is possible to completely improve by purifying the electron beam melting process and by modifying the metallographic structure through the thermo-mechanical treatment. The other counter measure is to inhibit the trans-passive corrosion by addition of oxide film former elements such as W and Si. The corrosion rate of these new alloys at heat transfer surfaces was markedly improved by inhibiting the trans-passive corrosion accompanied with heavy grain boundary attack (**Fig 3.3-5**).(4) It was also found that the susceptibility to SCC of Zr is possible to improve by addition of titanium. However, the addition of titanium decreases the corrosion resistance of Zr. We selected niobium alloys as alternative materials instead of zirconium. By addition of tungsten to the niobium, the corrosion resistance and the mechanical strength have been improved. This niobium alloy is possible to use in heavily corrosive nitric acid contaminated with fluorine. The difference in the corrosion resistance between of Zr and Nb-alloys depends on the chemical stability of each oxide film (MO_2 on Zr and M_2O_5 on Nb).(5) The resistance to SCC of Nb-5W alloy was also confirmed by SSRT.

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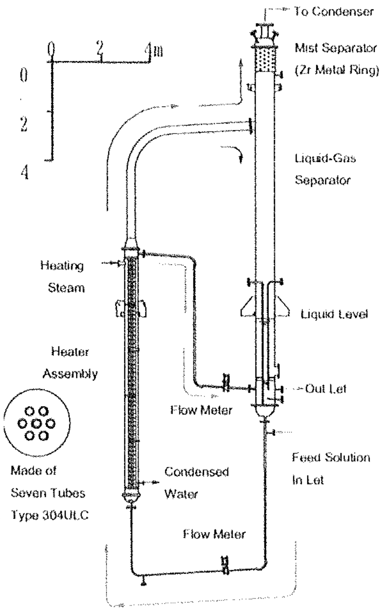
Heat conducting tubes	7
Chemical composition	9N HNO_3 (Metal Ion added)
Boiling point	68℃(at liquid surface)
Pressure	120torr(at mist-separator)
Operation	
Concentration ratio	40 times
Heating steam	100℃(saturation)
Material	type 304ULC



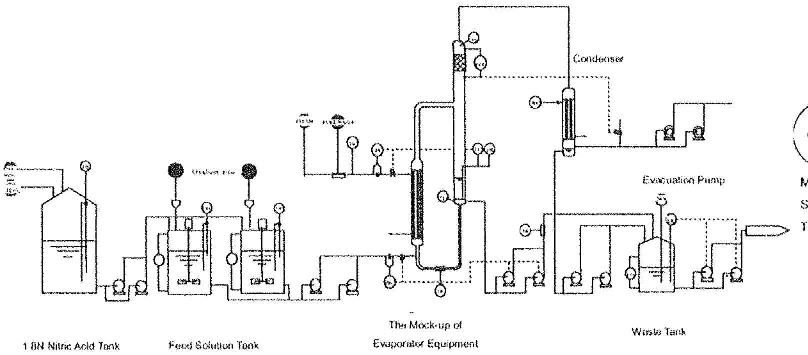


Fig. 3.3-1 The small-sized mock-up equipment simulated to acid recovery evaporator made of type 304ULC steel of the RRP.

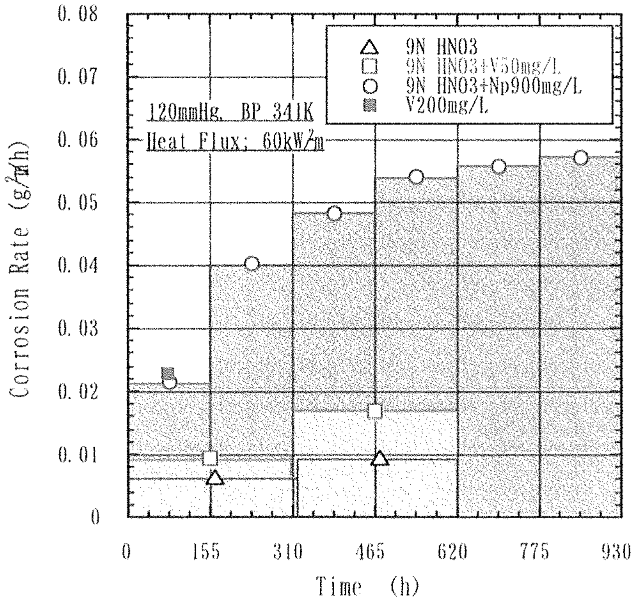


Fig.3.3-2 The effect of Np ions on corrosion of stainless steels under heat flux in boiling nitric acid solutions. The corrosiveness of Np

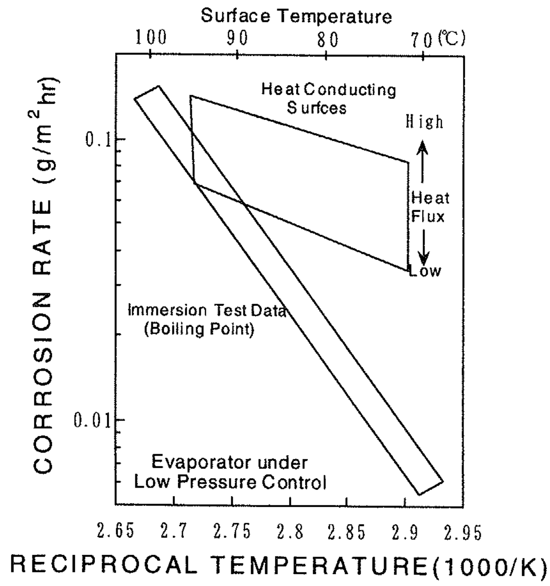


Fig. 3.3-3 The effect of heat flux on Arrhenius relationship of the corrosion rate plotted to the surface

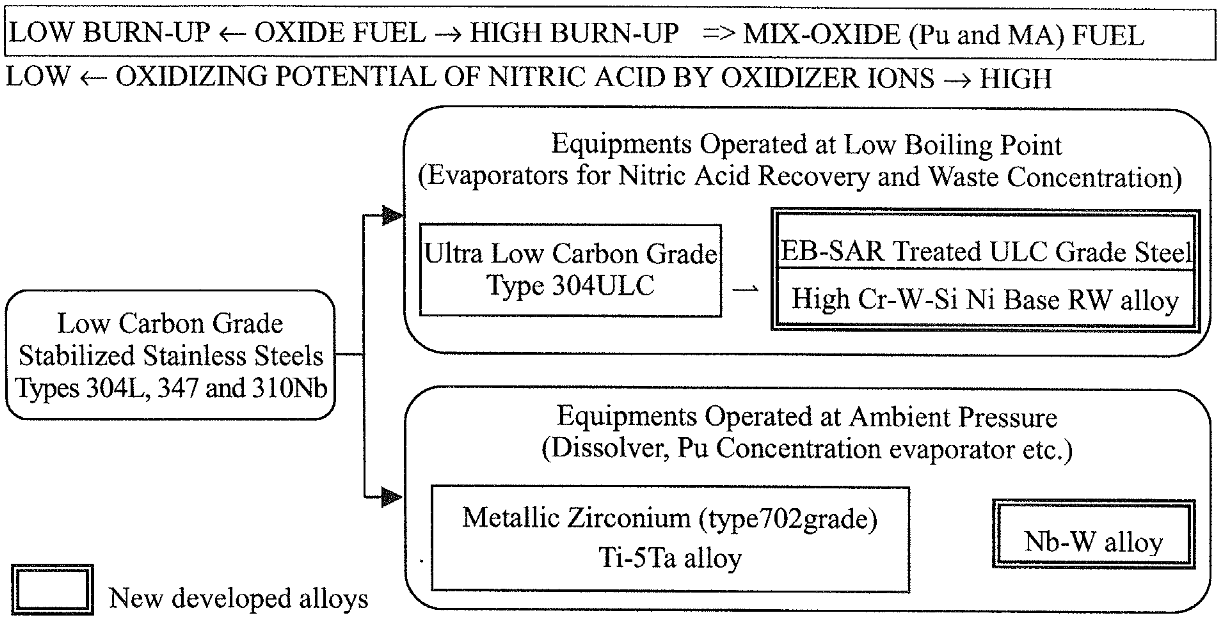


Fig. 3.3-4 The outline of historical development process of corrosion resistant materials against oxidizing nitric acid solutions

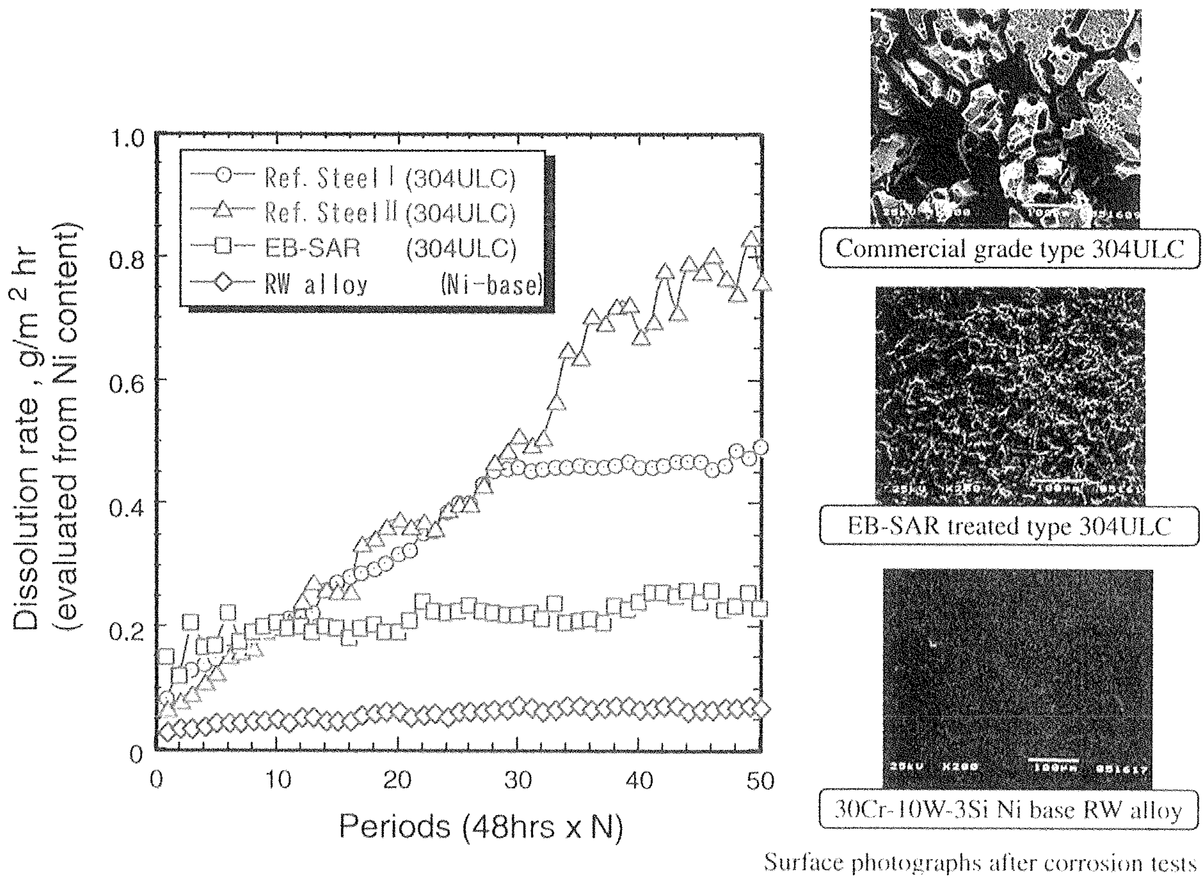


Fig. 3.3-5 The corrosion rate of new developed alloys and commercial heats of type 304UCL steels under heat flux control in nitric acid solutions.

3.4 Safety Research on Radioactive Waste Disposal

Radioactive waste is generated from nuclear power plants, nuclear fuel cycle facilities, radioisotope utilizing facilities such as hospitals, research institutes and other facilities. Since radioactive waste varies greatly in level of radioactivity and in the type of radioactive materials contained, arrangements should be made to classify the waste by method of disposal, regardless of the facility from which it originates, and take specific measures for its treatment/conditioning and disposal. Radioactive waste having relatively high radioactivity and long half-lives should be disposed of by a method capable of securing long-term safety, so that the living environment will not be affected. This type of radioactive waste requires "geological disposal". That is, burying the waste several hundred meters deep in stable underground zones, the natural barrier, which functions in combination with a system of multiple engineered barriers. On the other hand, radioactive waste, whose radioactivity attenuates to a sufficiently low level so that it would no longer be a risk to the living environment within a period for which institutional management is realistic, can be safely disposed of at relatively shallow depths.

In safety assessments of geological disposal and shallow land disposal systems for radioactive waste, it must be considered that groundwater will penetrate the waste disposal repository through time. However, the penetration of water may alter the performance capability of the engineered and natural barriers. Based on this concept, performance of engineered barriers, the waste forms, buffer and backfill materials, and the mobility of radionuclides in the natural barrier are important to evaluate the migration of radionuclides from the repository to the biosphere. In parallel, it is also necessary to establish the safety assessment methodology for the radioactive waste disposal system.

As for the engineered barriers, JAERI is carrying out full-scale leaching tests to evaluate performance of cement-solidified incombustible waste, fundamental leaching tests for bituminized waste and molten solidified waste [(1)(2)(3)], measurements of solubility of radionuclides with long half-lives[(4)] and study on the immobilization of radionuclides in cementitious materials[(5)]. Sorption/migration and diffusion behavior of radionuclides in the rock mass are being studied for the performance assessment of the natural barrier[(6)(7)(8)(9)]. For the development of safety assessment methodology, research activities are focused on safety and uncertainty analyses for shallow-land disposal of uranium wastes and radionuclide transport in heterogeneous aquifers[(10)(11)].

3.4.1 Leaching Behavior of Bituminized Waste Form*[(1)(2)]

Bituminized radioactive waste will be returned to Japan from French COGEMA

*This work was carried out by JAERI under an entrustment with the Ministry of Education, Culture Sports, Science and Technology of Japan.

reprocessing plants for spent nuclear fuel. This waste contains transuranium elements (TRU) and will possibly be buried in a deep underground repository as "TRU waste" in Japan. Quantitative evaluations on the leaching behavior of and the release of radionuclides from the buried bituminized waste form are required for the radiological safety assessment. The release of embedded elements under deep underground conditions has not been widely reported. We performed laboratory-scale leaching tests for radioactive and non-radioactive simulated COGEMA bituminized waste.

Leaching of organic carbon is indicative of bitumen matrix degradation. The leaching rate of soluble elements is higher than that of organic carbon as seen in Figure 3.4-1. Therefore, degradation of the bitumen matrix is thought to be less responsible for the performance of bituminized waste.

The release of elements such as sodium and cesium as soluble salts from the bituminized waste form was diffusion-controlled within leaching durations up to 200 days. Some of the samples were physically degraded by swelling. The concentrations of slightly soluble elements such as barium(II) and neptunium(IV) in the contacting water enriched in nitrates and organic matter can be predicted based on thermodynamic considerations for the elements.

3.4.2 Solubility-limited Concentrations of Selenium under Repository Conditions (4)

Performance assessment calculation for a high-level radioactive waste repository shows that selenium-79 (half-life 10^6 years) is one of the radionuclides that possibly dominates long-term radiological hazard. Release of ^{79}Se from a repository is expected to be limited by solubility of the compounds. Therefore, thermodynamic data of selenium are necessary for performance assessment calculations.

Since the chemical conditions under deep underground environments are likely to be anoxic, HSe^- is considered to be a dominant selenium species in groundwater. Selenium can be more soluble as polyselenide anions, Se_n^{2-} in alkaline solutions. Although the equilibrium constant for the formation of HSe^- is critical in estimating the solubility of selenium compounds, the previous value for the constant has a large uncertainty. The equilibrium constant of the reaction, $nSe(cr) + 2e^- = Se_n^{2-}$, has been calculated, but not determined directly. For this reason, we have performed solubility experiments of metallic selenium in aqueous solutions under anoxic conditions and pH between 5 and 13, to obtain equilibrium dissolution constants for metallic selenium.

The solubility of metallic selenium was measured in a mixture of 0.1M-NaCl and 0.05M-N_2H_4 under anoxic conditions ($O_2 < 1$ ppm) by both oversaturation and undersaturation methods. Equilibrium was attained in 40 days. The aqueous selenium species identified a re HSe^- at pH between 5 and 9, and Se_4^{2-} at pH between 10 and 13, using UV-V absorption spectrometry. The solid phase was identified as Se(cr) by X-ray diffraction. The equilibrium constants of

$$Se(cr) + H^+ + 2e^- = HSe^- \qquad \log K^0 = -6.5\pm0.5$$

and

$$4Se(cr) + 2e^- = Se_4^{2-} \qquad \log K^0 = -16.8\pm0.5$$

were determined. The standard molar free energy of formation of HSe^- and Se_4^{2-} was determined to be 37.1±2.9 and 95.9±2.9 kJ/mol, respectively.

Figure 3.4-2 shows that the solubility of selenium calculated from the equilibrium constants determined in this study is one order of magnitude higher than that predicted from previously reported thermodynamic data. In the higher pH region, selenium forms more soluble polyselenide anions, Se_4^{2-}, which was not assumed in the literatures. By using these results, the concentrations of selenium in natural groundwater and in alkaline, cement-contacting water can be reliably predicted.

3.4.3 Diffusive Mass Transfer in Rock Matrix[6]

When long-lived radionuclides are transported in groundwater through fractures in a rock mass surrounding a radioactive waste repository, diffusion into pores in the rock matrix and ensuing sorption onto mineral surfaces reduce the concentration of the nuclides transported through the geosphere. To take the matrix diffusion into account in evaluating the migration of radionuclides in deep geologic formations for the safety assessment of the repository, it is essential to understand the mechanisms of and quantify the diffusion of radionuclides into the rock matrix.

A pore diffusion model, in which diffusivity of an aqueous species in porous materials is assumed to be proportional to its diffusivity in bulk and to the geometric parameters of the pores, has often been applied to the diffusion of ions in rock matrix. However, the applicability of the model has not been verified. In this study, effective diffusivities of cationic and anionic species in granite from Inada, Ibaraki, Japan were determined by a through-diffusion method and the applicability of the pore diffusion model was examined.

The effective diffusivity (D_e) and distribution ratio (R_d) of $^{133}Ba^{2+}$ in Inada granite have been determined in triplicate using 0.1, 1 and 10 mol m^{-3} $BaCl_2$ solutions (Figure 3.4-3). The D_e value obtained for the $BaCl_2$ concentration of 10 mol m^{-3} agreed with the estimated value based on the pore diffusion model. The lower $BaCl_2$ concentrations yielded higher D_e and R_d values than those for 10 mol m^{-3}. The variation in D_e was not considered to be due to the speciation of barium in the solution nor to the variation in physical properties of the pore structure in the rock. Contribution to diffusion in an adsorbed state, surface diffusion, is considered to be responsible for the variation in D_e. This result is strongly indicative of diffusion in an adsorbed state.

The effective diffusivity of $^{125}I^-$ determined in Inada granite and in compacted bentonite was tabulated in Table 3.4-1. The D_e of I^- in compacted bentonite was smaller than the value estimated based on the pore diffusion model. This is because accessible pore space in the compacted bentonite is limited for anionic species due to electrostatic repulsion between the species and negatively charged surface of the pores. This was not the case for the diffusion of I^- in granite, where the size of the pore space is much larger than the range of the electrostatic repulsion so that the accessible pore space in granite for anionic species is the same as that for neutral and cationic species.

The results of this study showed that the diffusivity of divalent cations can be higher than the value expected from the pore-diffusion model because of the contribution of the surface diffusion. The surface diffusion may enhance diffusive mass transfer of species that are electrostatically adsorbed on solid-water interfaces.

3.4.4 Safety Analysis of Radioactive Nuclide Migration

(1)Safety and uncertainty analyses for shallow-land disposal of uranium wastes*(10)

Uranium wastes are characterized by the existence of long-lived radionuclides, the growth of daughters associated with uranium decay, the emanation of radon from wastes containing Ra-226 etc. To evaluate the performance of the waste disposal system over long timeframes, a deterministic approach is used for quantitative estimates of peak dose to an individual. However, uncertainties with respect to parameters, scenarios etc. are inherent in the long-term assessment for uranium waste disposal. For the next step in safety assessment, it is essential to estimate the uncertainties quantitatively. JAERI has developed the deterministic and probabilistic safety assessment system to estimate the long-term radiation effect owing to the shallow-land disposal of uranium wastes. Safety and uncertainty analyses of the system for disposal of uranium wastes were performed using both deterministic and probabilistic safety assessment models. The results of the deterministic analyses are shown in Figure 3.4-4. The dose calculation for a long residence time scenario is of great importance owing to the influence of daughters built up by uranium decay. The dose for a long residence time scenario is sensitive to the release condition of radionuclides from the facilities over long-term period. The parameter uncertainties for the important pathways in residence scenario are estimated from the probabilistic analyses using a statistical approach. The uncertainty analysis indicates that the influence of parameter uncertainty is the most significant in the estimation of the inhalation of radon gas (Figure 3.4-5).

(2) Radionuclide Transport in Heterogeneous Aquifer(11)

The results of Twin Lake tracer tests are used to develop a geostatistical approach based on the correlation between observed hydraulic conductivity values at different localtions. The geostatistical model assumes that the correlation strength of the values of hydraulic

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* This work was carried out by JAERI under an entrustment with the Ministry of Education, Culture Sports, Science and Technology of Japan.

conductivities between any two locations depends on the distances between these locations, and is expressed as an exponential function. Numerically, a matrix decomposition method is used. The groundwater flow field is calculated using the 3D-SEEP code using the realized spatial distribution of hydraulic conductivity that is calculated by the geostatistical model. The flow simulation results are used as input into the transport computer code, which in turn is used to simulate the tracer breakthrough curves at different observation wells. A random walk method is used for the radionuclide transport simulation. The simulated tracer plumes of tracer tests explain favorably the experimental tracer plumes (Figure 3.4-6).

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Table 3.4-1 Effective diffusivity (D_e) of species in Inada granite and in bentonite[a]

Media	Species	D_e ($m^2\ s^{-1}$)	Solution	D_v[b]($m^2\ s^{-1}$)
Granite	I^-	$(5.06 \pm 0.09) \times 10^{-13}$	10^2 mol m^{-3} SrI_2 → DDW[c]	1.14×10^{-9}
	I^-	$(7.98 \pm 0.07) \times 10^{-13}$	10^2 mol m^{-3} CsI → DDW	1.87×10^{-9}
	Cs^+	$(6.32 \pm 0.09) \times 10^{-13}$		
	HTO	$(7.7 \pm 1.4) \times 10^{-13}$	DDW	2.14×10^{-9}
	I^-	$(5.80 \pm 0.05) \times 10^{-13}$	10^2 mol m^{-3} KI	1.86×10^{-9}
	I^-	$(7.1 \pm 2.1) \times 10^{-13}$	10^{-1} - 10^1 mol m^{-3} $BaCl_2$	2.04×10^{-9}
Bentonite	HTO	$(2.12 \pm 0.36) \times 10^{-10}$	4.5 mol m^{-3} Na_2SO_4	2.14×10^{-9}
	I^-	$(8.00 \pm 0.15) \times 10^{-12}$		2.04×10^{-9}

(a) Kunigel V1 (density: 1.6×10^3 kg m^{-3}, calculated porosity: 0.41%).

(b) Diffusivity of the species in bulk of the solution.

(c) Distilled and deionized water

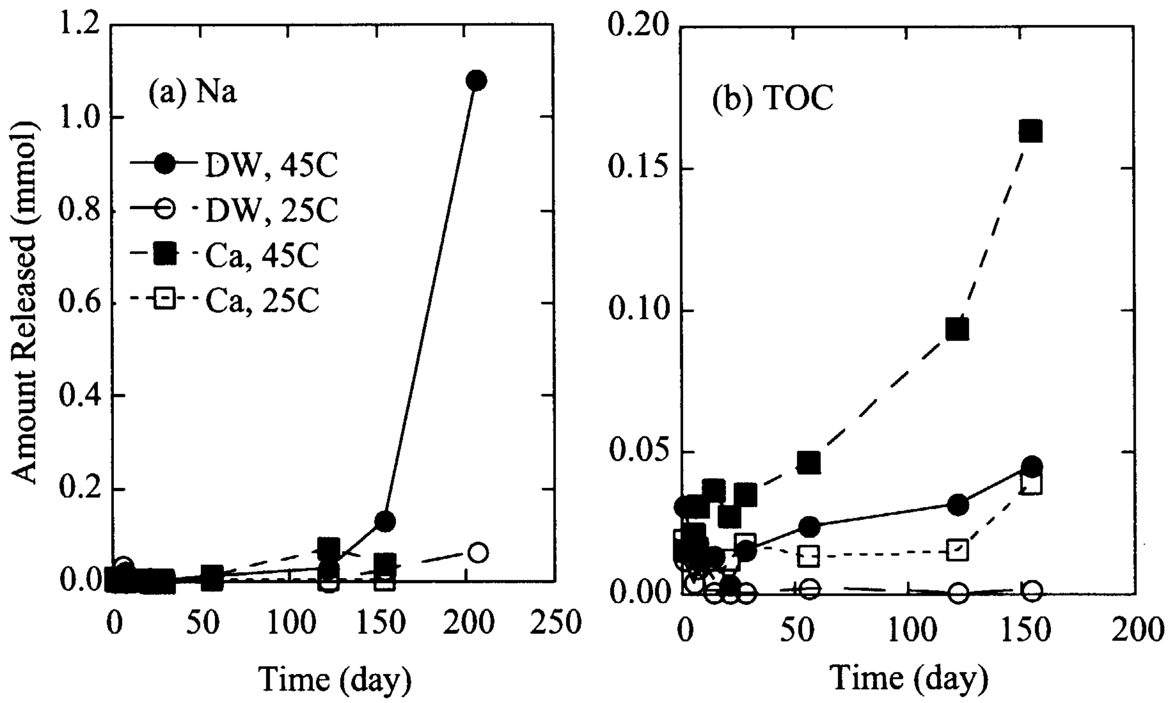

Fig. 3.4-1 The release of (a) Na and (b) organic carbon (TOC) from a COGEMA-type bituminized waste leached under atmospheric conditions: DW: Deionized water, Ca: 0.01 mol/L calcium hydroxide solution.

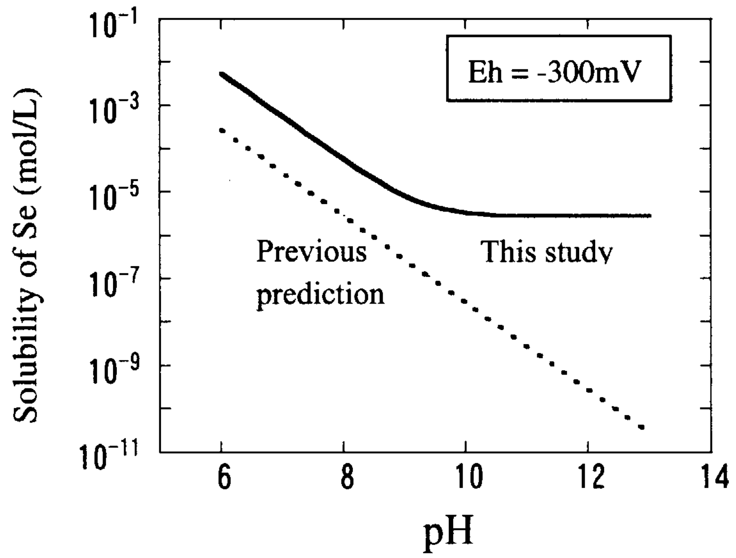

Fig. 3.4-2 The solubility of selenium under reducing condition(Eh = -300mV). The concentration of selenium estimated in this study is higher than that calculated by using previously reported equilibrium constant.

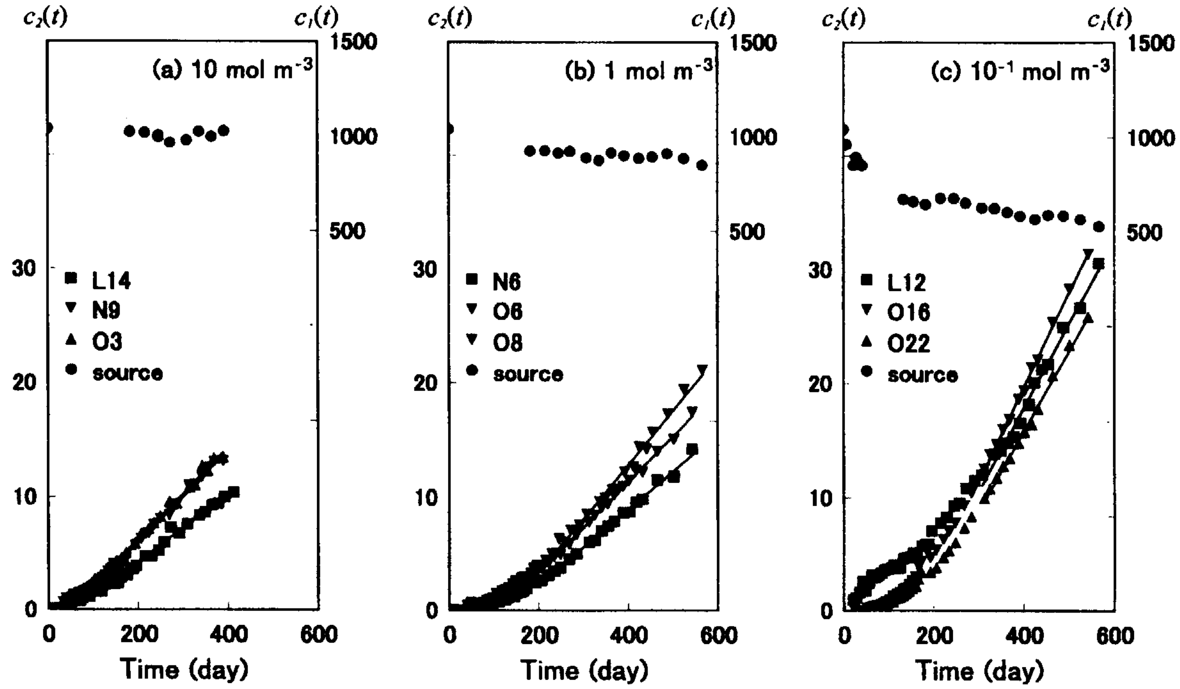

Fig. 3.4-3 Diffusion curves of ^{133}Ba through Inada granite samples obtained for 10 mol m^{-3} $BaCl_2$ solution (a), for 1 mol m^{-3} $BaCl_2$ solution (b) and for 10^{-1} mol m^{-3} $BaCl_2$ solution (c), where c_1 and c_2 denote concentration of ^{133}Ba in source reservoir and in measurement reservoir (MBq m^{-3}), respectively.

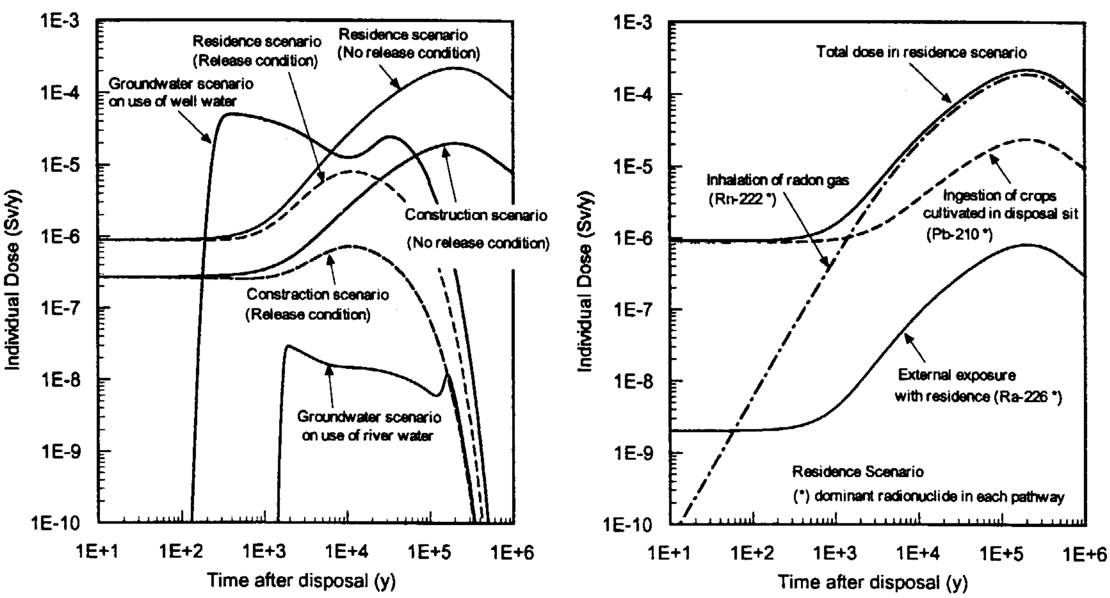

Fig. 3.4-4 Results of deterministic analyses: (a) Individual doses for four scenarios, (b) Individual doses for three exposure pathways in residence scenario under no release condition from the disposal facilities.

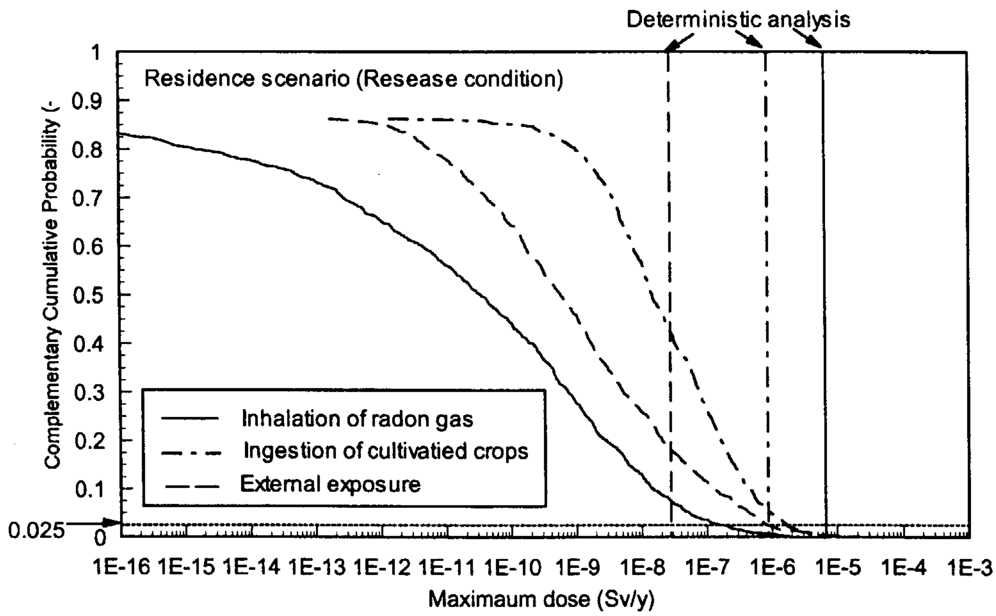

Fig. 3.4-5 The CCDFs of the maximum dose in residence scenario, derived from the probabilistic analysis for one thousand sets of sampling parameter

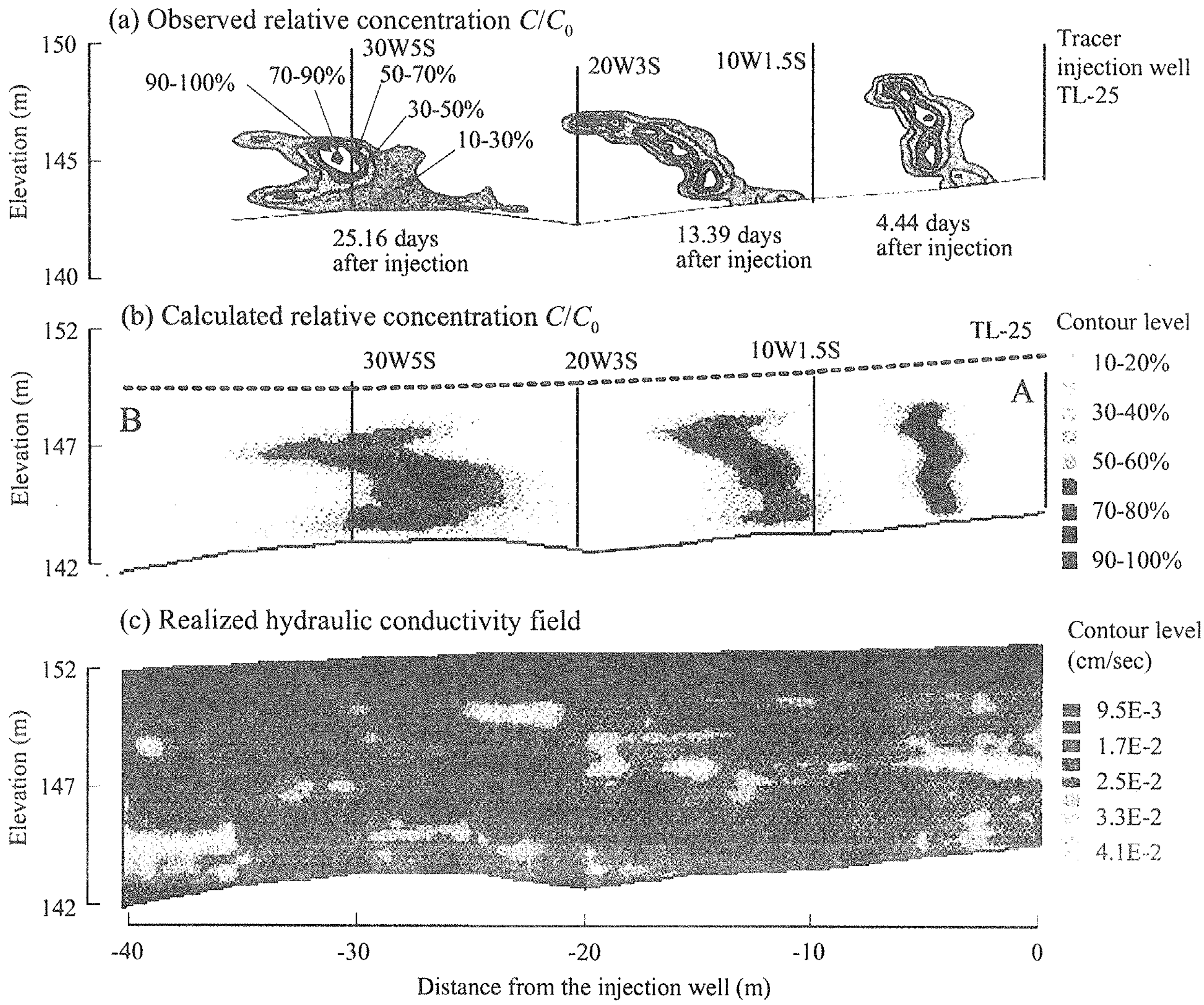

Fig. 3.4-6 Concentration distribution of tracer plume obtained from (a) observation, (b) calculation at 4.44, 13.39 and 25.16 days after injection and (c) one example of realized hydraulic conductivity (x component) used for calculation of (b). Illustrations of plumes of (a) and (b) are drawn by contours representing 10-20%, ..., and 90-100% of the relative maximum concentration.

4. TEST FACILITIES FOR SAFETY RESEARCH

Several large-scale test facilities are being utilized to perform nuclear safety research in JAERI.

For examinations using high-radioactive material, JAERI has three hot laboratories, the Reactor Fuel Examination Facility (RFEF) for the post-irradiation examination of LWR fuel, the Research Hot Laboratory (RHL) for the post-irradiation examination of nuclear fuels and materials, and the Waste Safety Testing Facility (WASTEF) for examinations concerning corrosion behavior of the nuclear plant material and disposal of radioactive wastes from nuclear research facilities and other plants. The outline and utilization of these facilities are shown in the following section 4.1.

For fundamental researches on the nuclear fuel cycle and the radioactive waste management, JAERI has the Nuclear Fuel Cycle Safety Engineering Research Facility (NUCEF). The facility consists of the Static Experiment Critical Facility (STACY), the Transient Experiment Critical Facility (TRACY) and the Back-end Fuel Cycle Key Elements Research Facility (BECKY). STACY and TRACY are critical experiment facilities for criticality safety research in the nuclear fuel cycle. BECKY is an R&D facility on chemical separation in reprocessing, partitioning of high-level radioactive waste, management of TRU-containing waste and basic chemistry on TRU elements. The outline and utilization of NUCEF are shown in the section 4.2.

Some other facilities in JAERI relating nuclear safety research are briefly described from section 2.1 to 2.4, which are the Nuclear Safety Research Reactor (NSRR) for research on fuel behavior under the reactivity initiated accident, the Large Scale Test Facility (LSTF) for research on thermal hydraulics behavior under the accidental condition of LWR, the Assessment of Loads and Performance of Containment in Hypothetical Accident (ALPHA) facility for experiments on containment vessel of LWR, and the Wide Range Piping Integrity Demonstration (WIND) facility for experiments on piping of LWR cooling system under the severe accident.

4.1 Hot Laboratories

There are three hot laboratolies for a nuclear safety research at Tokai Research Establishment of JAERI, which can handle a high-radioactive material. The post-irradiation examination of a nuclear fuel and a material is carried out in the RFEF and RHL, and the safety test concerning advancement of a waste forms and for study on corrosion behavior of the nuclear plant material is carried out in the WASTEF. Details of each facility are shown in 4.1.1 through 4.1.3.

4.1.1 Reactor Fuel Examination Facility, RFEF

The RFEF has been performing the Post-Irradiation Examinations (PIEs) of LWR fuels and materials for an inspection of the integrity and the reliability. There are a pool for storing the fuel assemblies, six beta-gamma concrete cells with three lead cells for PIEs of the irradiated LWR fuels and materials, and two alpha-gamma concrete cells with two lead cells for PIEs of the fuels containing plutonium in RFEF.

Main present works in RFEF are the PIEs of the high burn-up fuels including MOX fuels and the development of several new PIE apparatuses for obtaining the more detailed PIE data. Field emission type scanning electron microscope (FE-SEM)* and twin baskets type densitometer for small size specimens are under development in order to obtain the detailed information on the irradiated fuels and materials.

The images of small structures including rim structure on the high burn up pellets and small cracks in the oxide film layer in Zircaloy tubes can be probably obtained by FE-SEM. The FE-SEM is equipped with Energy Disperse Spectroscopy (EDS) for analyzing the elements in the observation area on the samples. FE-SEM (**Figure.4.1-1**) is installed in a radiation shielding box made by steel plates with a manipulator for the remote handling of the samples. The box has a gate for loading and withdrawing of samples into the box. The cold mock-up test is continued for confirming the way of the maintenances and that of the sample preparation. Twin baskets type densitometer is applied to measuring the mass-density of the small size specimens taken from the irradiated fuels. From the viewpoint of the safety research for the high burn-up LWR fuels, the density of the small fragments in the fuels is important information for the clarification of the thermal conductivity and for the determination of the swelling rate in the local area. It is difficult to measure the density of the small size specimens by the conventional densitometer weighing mass of the samples together with that of the basket in both air and liquid, because the inaccuracy caused by the

* This work was performed under the auspices of the Ministry of Education, Culture, Sports, Science and Technology of Japan.

large mass and buoyancy of the basket inhibited the precise mass measurement of the samples. Twin baskets system offsetting buoyancy and mass of the baskets are equipped in the new developed densitometer (**Figure.4.1-2**). The improved system achieved to measure the mass of the specimens in both air and liquid precisely, and furthermore makes it possible to perform the measurement by an once-through operation contributing to shorten and simplify the examination process. The present works are in the stage of cold mock up to optimize the measurement condition and to confirm the accuracy in the selected condition.

4.1.2 Research Hot Laboratory, RHL

Operation of the RHL was initiated in 1961; it was the first PIE facility in Japan, and many kinds of PIEs have been conducted for 39 years. These examinations have assisted R&D works of fuels and materials for LWRs, the High Temperature Engineering Test Reactor (HTTR), and a Nuclear Fusion Reactor. In the RHL, monitoring tests on fuels and structural materials had been performed for safe operation of the Magnox reactor at Tokai Power Station in the Japan Atomic Power Company (JAPCO) during 32 years.

The RHL is a comprehensive beta-gamma hot cell facility that includes ten concrete cells and thirty-eight lead cells with many types of apparatus. Main activity in recent years is a safety research for LWR. RHL has made PIE data on tensile, Charpy impact and plain strain fracture toughness tests for safety evaluation of LWR pressure vessel steel. PIEs of pulse-irradiated fuels in NSRR have also been carried out to evaluate RIA for spent fuel.

4.1.3 Waste Safety Testing Facility, WASTEF

The facility was established in 1981 to investigate the safety storage and disposal of high-level wastes from reprocessing of spent fuel. It is equipped with three beta-gamma concrete cells, two alpha-gamma concrete cells, one lead cell, and six glove boxes. In the hot cells, manipulators are used for handling hot samples and operating apparatuses.

At present, WASTEF conducts several experiments using the hot cells. At first, a diffusion test of ^{238}Pu and ^{237}Np in bentonite has been carried our under the simulated environment of a disposal site since 1999. We also started various new examinations mentioned below. Experiments on mineral synthesis have been performed under a simulated disposal environment. Further, preparation technique of americium and curium nitrides from their oxides has been developed using carbon thermal reduction method. Recently, we achieved the preparation of americium and curium nitrides.

In addition, the heating surface corrosion test* has been conducted under a simulated condition, which clarifies effects of the spent fuel solution and TRU elements on corrosiveness of materials used in the dissolving tank and the acid recovery evaporation tank at a reprocessing plant. **Figure 4.1-3** shows the heating surface corrosion test apparatus.

In 2000, two devices (one is for detecting impurities of nitrides and the other is for studying decomposition reaction of nitrides at high temperature) were installed in a maintenance box over the concrete cell. A research in depth on how to use the nitrides as transmutation fuel was launched.

An in-cell apparatus for IASCC tests of irradiated material test specimens was installed in the beta-gamma hot cell. A SSRT test apparatus, which can create a high-temperature and high-pressure corrosive environment, will be installed in 2001 and a long-term experiment will be carried out by automatic and remote control. Moreover, an in-cell fabrication apparatus of small-sized waste products solidified by melting will be installed to investigate their barrier performance in the present fiscal year.

* This examination was performed under the auspices of the Ministry of Education, Culture, Sports, Science and Technology.

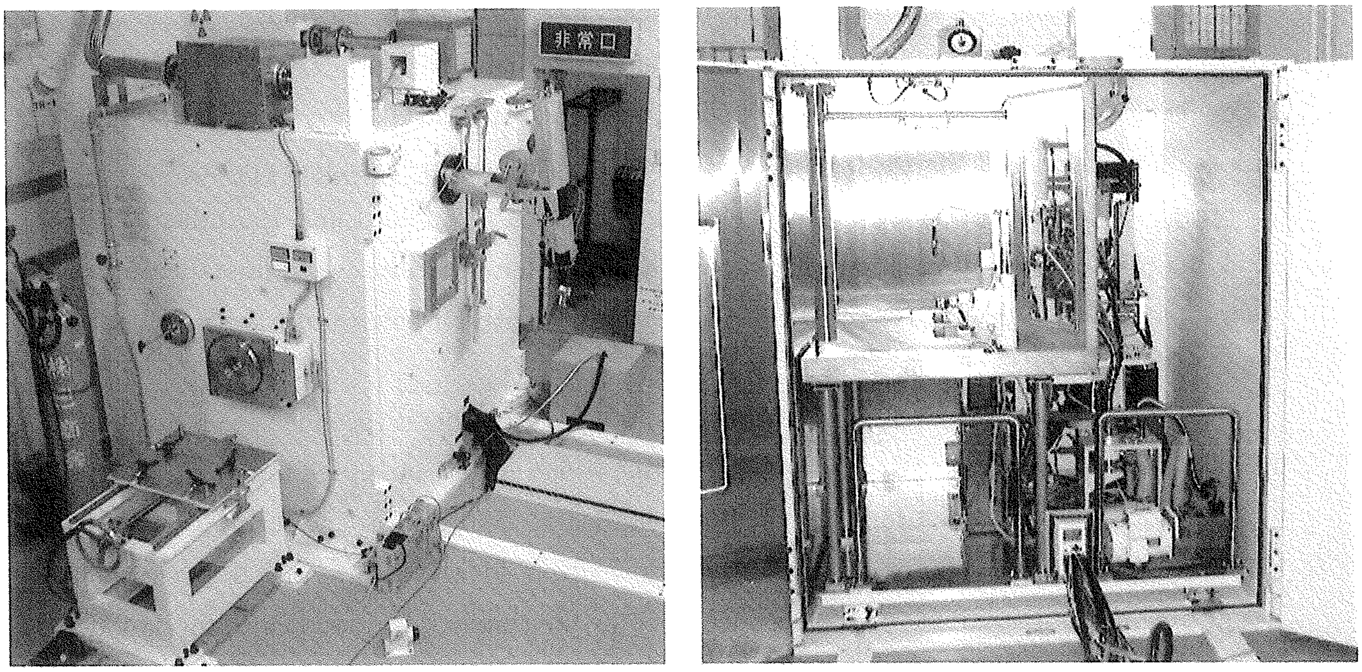

Fig 4.1-1 Photographs of appearance of FE-SEM with shield box.

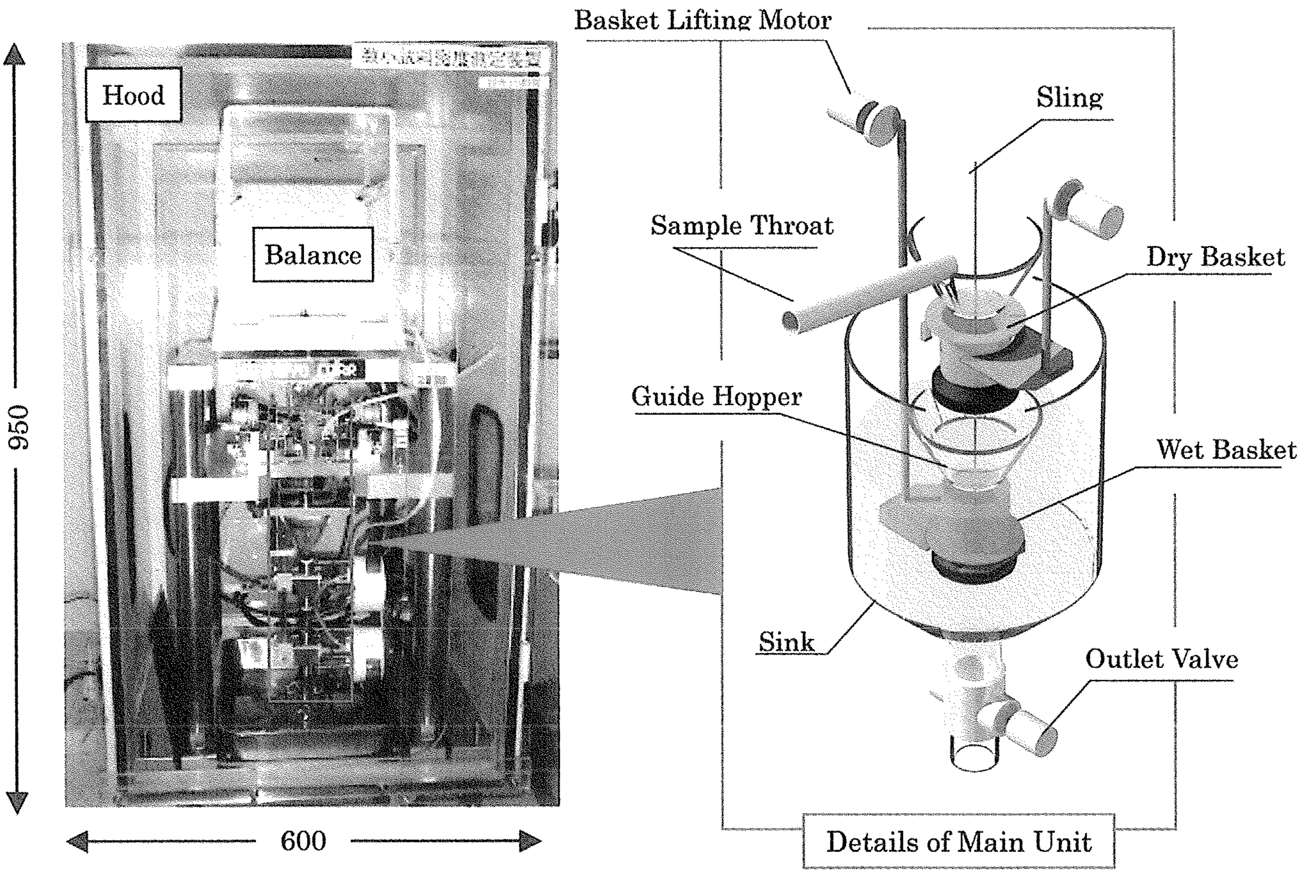

Fig. 4.1-2 Photograph and Schematic Drawing of Micro-Sample Densitometer.

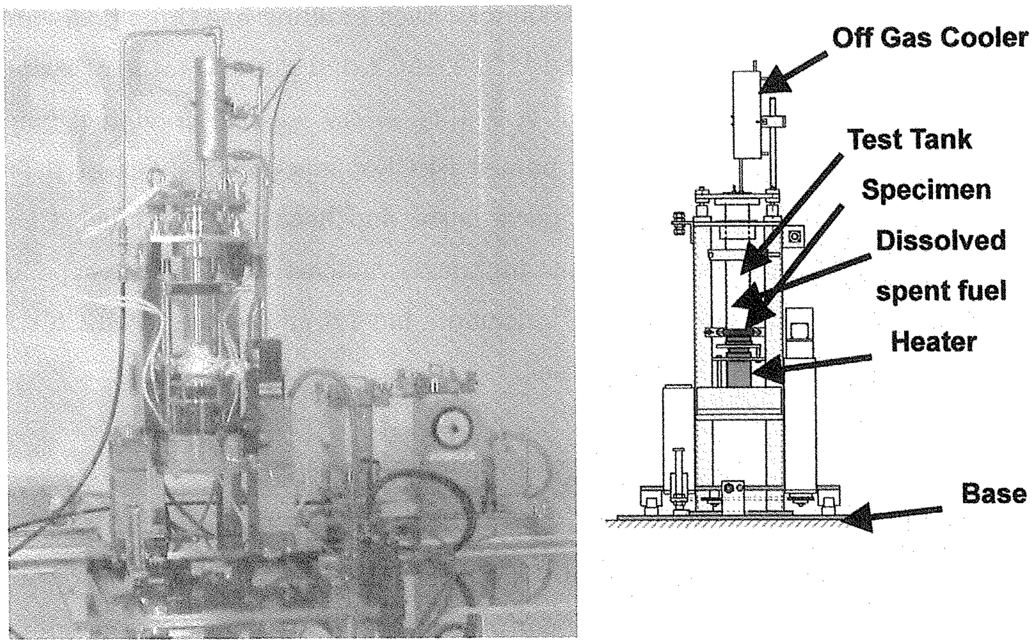

Fig. 4.1-3 Heating surface corrosion test apparatus.

4.2 Nuclear Fuel Cycle Safety Engineering Research Facility (NUCEF)*

NUCEF is a large-scale facility for fundamental researches on safety and technological improvement in the field of nuclear fuel cycle and radioactive waste management.

Figure 4.2-1 shows a cutaway view of NUCEF. The facility consists of two buildings, Experiment Buildings A (9,500 m^2) and B (8,000 m^2). Each building has three floors and one basement. The Experiment Building A has STACY, TRACY and a fuel treatment system for STACY and TRACY, where the research on criticality safety with solution fuel has been carried out. The Experimental Building B, named BECKY, is a complex research facility, where $\alpha\gamma$ cells with high-density concrete, glove boxes and hoods are installed. Chemical separation in reprocessing, partitioning of the high-level radioactive waste, management of the TRU waste, and basic chemistry on TRU elements have been studied in BECKY. **Table 4.2-1** shows major specifications of the facilities.

4.2.1 STACY and TRACY

In STACY, criticality experiments using two slab-shaped core tanks and 10% enriched uranyl nitrate solution have been performed since 2000. The dimension of the core tanks is 35cm in thickness, 70 cm in width and 150 cm in height. The data on neutron interaction effect have been obtained by changing the distance between two core tanks up to 150 cm, and placing neutron-isolating materials, such as polyethylene and concrete, and neutron absorbers, between those two core tanks. **Figure 4.2-2** shows the core system for the experiments of the neutron interaction effect.

In TRACY, experiments on supercriticality of 10% enriched uranyl nitrate solution have been carried out using a 50cm-diameter cylindrical core tank. In 2000, an observation system(1) with a radiation-resistive optical fiber scope was installed in TRACY to comprehend solution fuel behavior under supercritical conditions. The behavior of the solution fuel and radiolytic gas voids under supercritical conditions was clearly observed by this system. **Figure 4.2-3** shows the behavior of solution fuel under a supercritical condition by feeding of the solution fuel. For the evaluation of absorbed doses at a criticality accident, experiments on radiation dosimetry has been carried out by using thermoluminescent and alanine dosimeters. In the experiments, the absorbed doses of neutron and gamma ray were measured up to 900Gy under criticality accident conditions(2).

To support the above experiments in STACY and TRACY, the composition of fuel solution

* A portion of this work was performed under a contract with the Ministry of Education, Culture, Sports, Science and Technology.

has been adjusted in the fuel treatment system. The concentrations of uranium and nitric acid, and the amounts of impurities and fission products in the solution fuel have been determined in an analytical laboratory. The solution fuel containing 6% enriched uranium was prepared for the experiments of a heterogeneous core system of STACY in the fuel treatment system.

In accordance with a future program of the criticality experiments using plutonium solution fuel in STACY, the equipment was installed for handling and dissolution of mixed oxide (MOX) powder, which is difficult to dissolve in nitric acid solution. **Figure 4.2-4** shows a photograph of the MOX dissolver using a silver mediated electrochemical oxidation method.

4.2.2 BECKY

Experiments on reprocessing were performed three times using 1kg of LWR spent fuel in the $\alpha\gamma$ cells since 1998. The spent fuels having range of burn-ups from 8000 to 44000 MWd/t were used for the experiments. In addition, partitioning tests were carried out two times in the $\alpha\gamma$ cells, using 10 L (5 TBq) of the high-level liquid waste solution arising from the reprocessing experiments. **Figure 4.2-5** shows a photograph of equipment for the reprocessing experiment.

For studies on waste management, atmosphere controlled glove boxes filled with argon gas were installed in 1999, where experiments on bituminized waste form and thermodynamic data of nuclides contained in the radioactive wastes have been carried out.

Studies on detection of small amount of fissionable materials in the TRU-containing waste have been also carried out using a D-T pulsed neutron source. Other experiments on TRU waste management and TRU chemistry have been performed in BECKY.

References

(1) Ogawa, K., et al., "Development of Solution Behavior Observation System under Criticality Accident Conditions in TRACY", J. Nucl. Sci. Technol., *37*(12), 1088 (2000).

(2) Sono, H., et al., "Measurement and Analysis of Neutron and Gamma-Ray Doses on Criticality Accident of Low-enriched Uranyl Nitrate Solution using Tissue-equivalent Dosimeters at the TRACY Facility", *Proc. ANS Int. Topl. Mtg. Advances in Reactor Physics and Computation into the Next Millennium (PHYSOR2000),* May 7-12, Pittsburgh, U.S (2000).

Table 4.2-1 Major specifications of facilities in NUCEF

STACY and TRACY Critical Experiment Facility		
	STACY	TRACY
Thermal power	Max. 200W	Max. 10kW (delayed critical operation) Max. 5000MW (transient operation)
Excess reactivity	Max. 0.8$	Max. 0.8$ (delayed critical operation) Max. 3$ (transient operation)
Maximum fuel inventory	Uranium nitrate solution 4 or 6 % enrichment 500kgU 10% enrichment 150kgU Plutonium nitrate solution 60kgPu UO_2 fuel rod 5 % enrichment 400kgU	Uranium nitrate solution 10% enrichment 150kgU
Reactivity control method	Feed and drainage of solution (safety rods for shut down)	Feed and drainage of solution and withdrawal of transient rod
First Criticality	February 1995	December 1995
Back-end Fuel Cycle Key Elements Research Facility(BECKY)		
	Research on chemical separation in nuclear fuel cycle	Research on TRU waste management
Research Themes	-Improvement of reprocessing process -Development of partitioning process -Related fundamental chemistry on TRU	-Development of TRU measuring technique -Development of TRU disposal method
Experiment methods	Laboratory scale experiments using actual spent fuel and high-level radioactive liquid waste	Laboratory scale experiments using artificial TRU waste
Major radioactive materials and their maximum handling quantities	-Spent fuel (Max. 45000 MWd/t) 3 kg/year pellets and solution etc. -High-level radioactive liquid waste 2 liters 185 TBq/year	-Plutonium solid, solution etc. -Radioisotopes (TRU, Cs, Sr etc.) solid, solution etc. 2.56 TBq/day
Major equipment	-$\alpha\gamma$ cells -Glove-boxes	-Atmosphere controlled glove-boxes -Glove boxes -D-T Pulsed neutron source

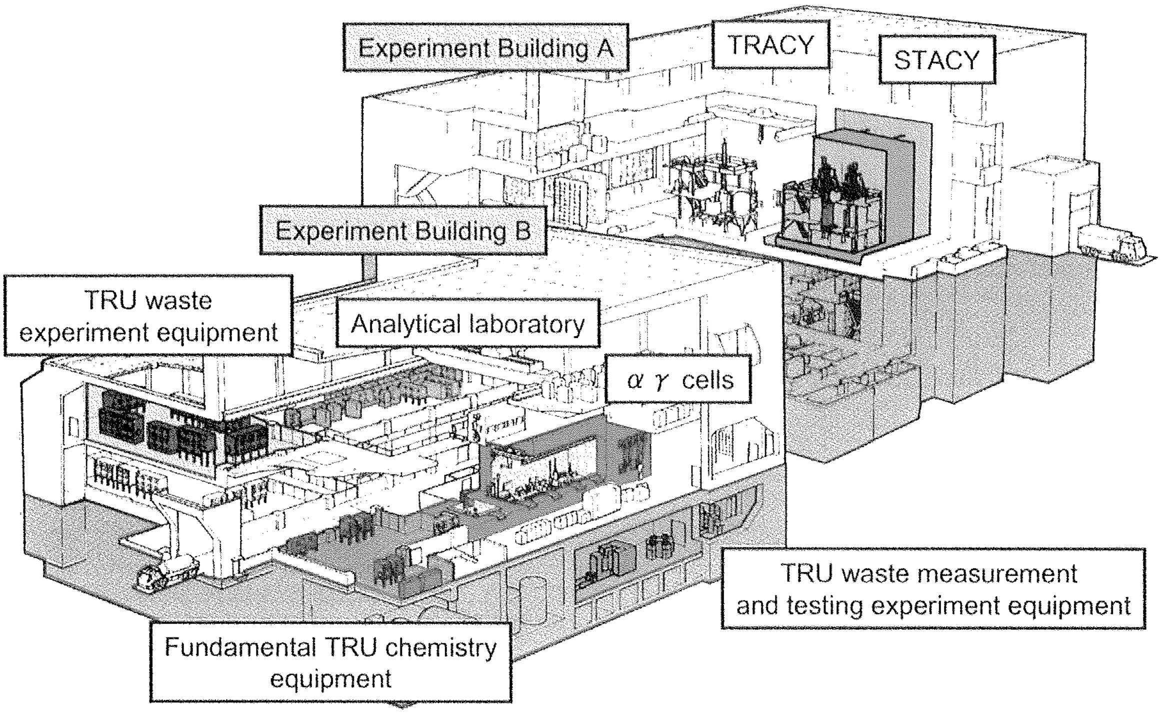

Fig. 4.2-1 Cutaway view of NUCEF.

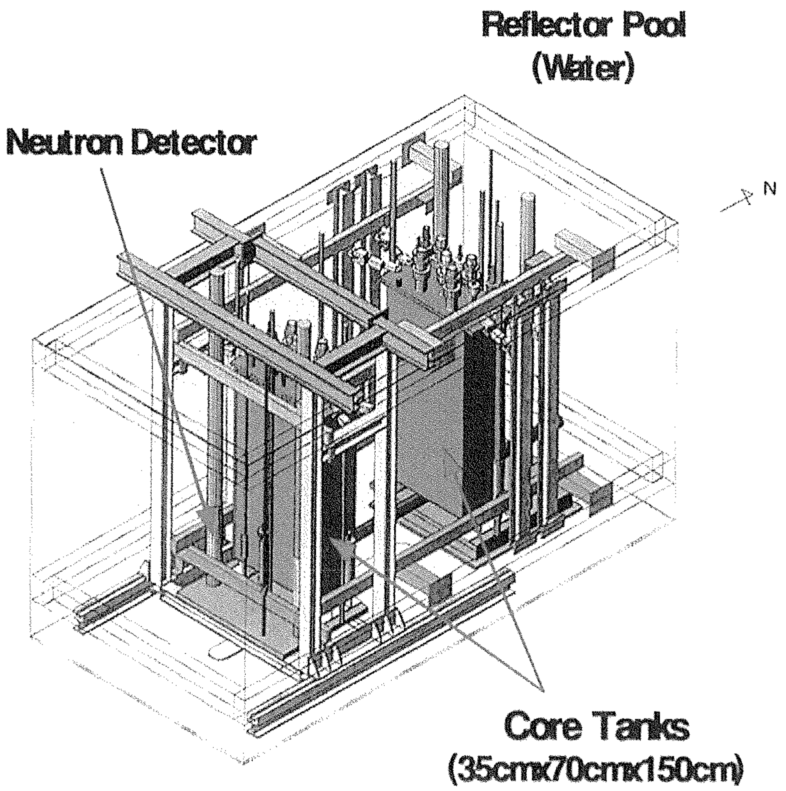

Fig. 4.2-2 Core system of STACY for neutron interaction study.

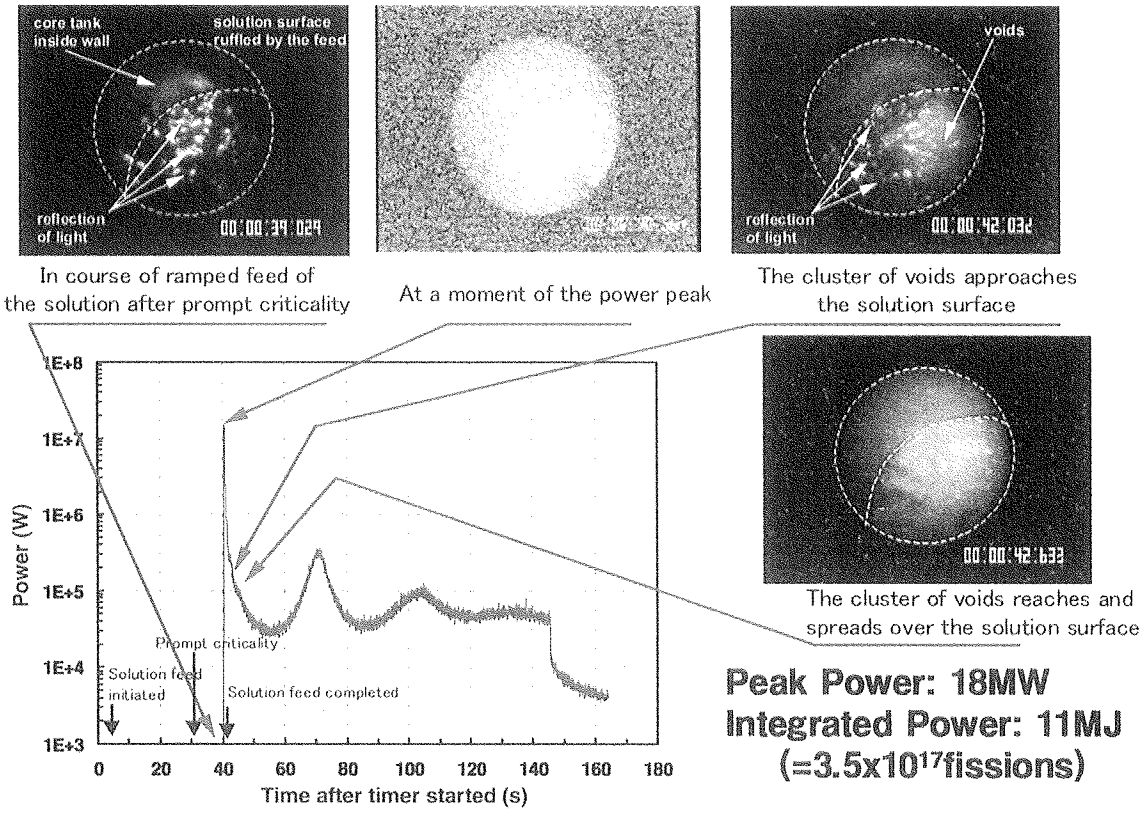

Fig. 4.2-3 Behavior of solution fuel under the super criticality condition (1.5 $, 14 c/s).

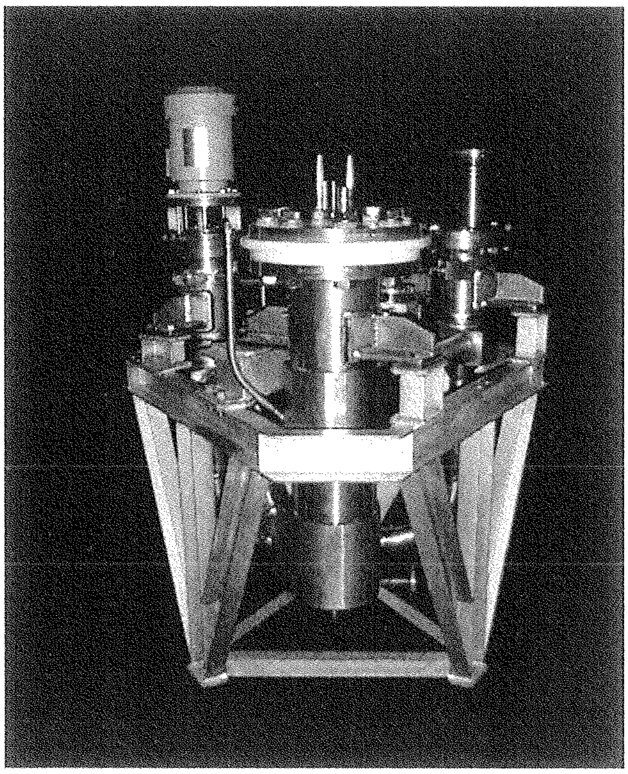

Fig. 4.2-4 MOX dissolver using silver mediated electrochemical oxidation method.

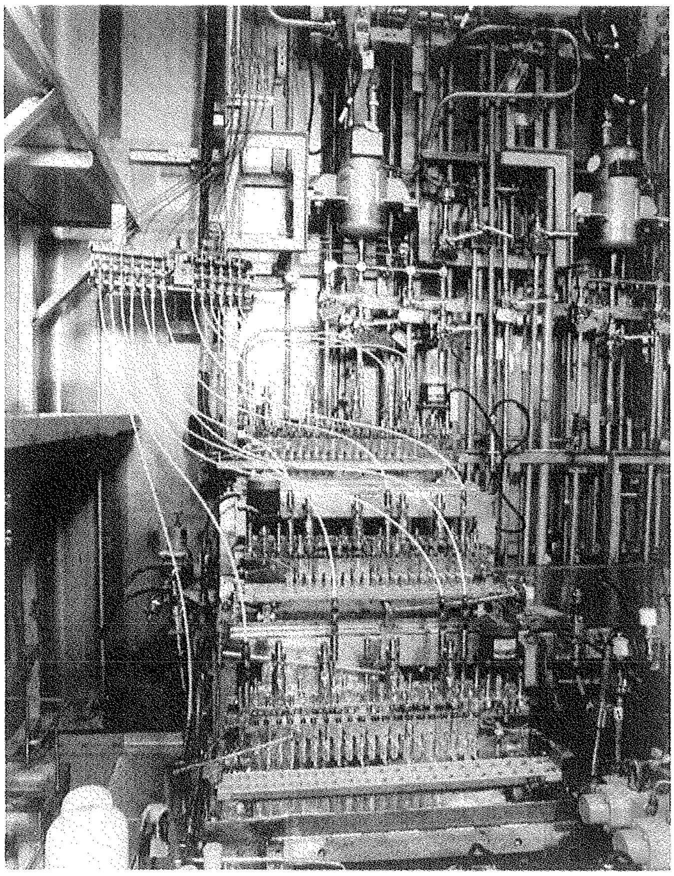

Fig. 4.2-5 Equipment for reprocessing test in α–γ cell of BECKY.

5. INTERNATIONAL COLLABORATION

JAERl recognizes that improving the safety of nuclear facilities is a common global goal and is necessary to achieve public acceptance of nuclear energy. With this impetus, JAERl has assigned international collaboration a significant role in its nuclear safety research activities. This collaborative effort allows the sharing of data and information among the participating countries by conducting cooperative research to obtain scientific and technical knowledge. International collaborative research activities are listed in Table 5－1. 1n addition, JAERl actively takes a responsible role by providing technical experts and assistance to the international community through the International Atomic Energy Agency (IAEA) and the Organization for Economic Cooperation and Development (OECD) to enhance and further ensure the safety of nuclear facilities around the world.

Table 5-1 International cooperative agreements in the field of nuclear safety research

Items	Participants (Host Organization)	Place	Period	Objectives
OECD Halden Reactor Project	24 Organizations from 20 Countries (OECD & Norwegian Institute for Energiteknikk)	Norway/Halden	Apr. 1967-Dec. 1999	Research on performance and reliability of fuel. Man-machine communications.
Cooperative Research Program in the Nuclear Safety Area and other Related Areas	JAERI KAERI	Korea/Taejon	June 1994-July 2001	Probabilistic safety assessment, Human factors, Fuel behabior in accidental conditions, Low-level waste disposal, Real-time dose assessment for power plant accidents.
JPDR Pressure Vessel Program	JAERI ORNL	Japan/Tokai	Apr. 1994-Apr. 1999	Investigation of the aging degradation of JPDR pressure vessel material.
ROSA-AP600 Program	JAERI USNRC	Japan/Tokai	Oct. 1992-Oct. 1999	Confirmatory thermal-hydraulic testing on AP600 advanced passive reactor design.
CSARP	26 Organizations from 18 Countries (USNRC)	USA/Albuquerque	Jan. 1993-Dec. 2000	Research on severe fuel damage and fission product source term under severe-accident (SA) connditions.
ACE Analysis Program	14 Organizations from 11 Countries (EPRI)	USA/Argonne	Mar. 1994-Dec. 2000	Fission product behavior in containment under SA conditions, Molten core/ concrete interactions, Evaluation of accident management measures.
RASPLAV	24 Organizations from 17 Countries (USNRC)	Russia/Kurchatov	July 1994-June 2000	Integrity of pressure vessel lower head.
Cooperation Agreement in the Field of Nuclear Safety and Protection	JAERI IPSN	France/Cadarache	June 1994-Sep. 2002	Reactivity initiated accidents (RIA) and severe accident analytical experiments, Criticality safety.
Agreement for Cooperative Research on Maintenance of Pressure Vessel Integrity	16 Organizations from 14 Countries (IAEA)	Japan/Tokai France/	May 1997-Mar. 2000	Development of surveillance specimen reconstitution technique.

Table 5-1 International cooperative agreements in the field of nuclear safety research (Continued)

Agreement for Cooperative Research on Radiological Safety for LLW	JAERI CIRP	China/Shanxi Province	Aug.1995- July. 2001	Cooperative research on assessment method of radiological safety for shallow land disposal of LLW.
Implementing Agreement for Cooperation in the Field of Radioactive Waste and Spent Fuel Management	JAERI CEA	Japan France	Nov. 1999 Sep. 2002	Research on MA (Minor Actinides) and log-lived FP (Fission Products) partitioning Migration behavior database for long-lived nuclides Storage-Containers for storage Pyrochemical processes
Co-ordinated Research Programme, "Improving Long term Safety Assessment Methodologies for Near Surface Radioactive Waste Disposal Facilities (ISAM)"	30 Organizations from 29 Countries (IAEA)	Austria/Vienna	Nov. 1997- Oct. 2000	To provide a critical evaluation of the approaches and tools currently used in the post closure safety assessment of proposed and existing near-surface radioactive waste disposal facilities

国際単位系（SI）と換算表

表1　SI基本単位および補助単位

量	名　称	記　号
長さ	メートル	m
質量	キログラム	kg
時間	秒	s
電流	アンペア	A
熱力学温度	ケルビン	K
物質量	モル	mol
光度	カンデラ	cd
平面角	ラジアン	rad
立体角	ステラジアン	sr

表2　SIと併用される単位

名　称	記　号
分，時，日	min, h, d
度，分，秒	°, ′, ″
リットル	l, L
トン	t
電子ボルト	eV
原子質量単位	u

1 eV=1.60218×10^{-19}J

1 u=1.66054×10^{-27}kg

表5　SI接頭語

倍数	接頭語	記　号
10^{18}	エクサ	E
10^{15}	ペタ	P
10^{12}	テラ	T
10^{9}	ギガ	G
10^{6}	メガ	M
10^{3}	キロ	k
10^{2}	ヘクト	h
10^{1}	デカ	da
10^{-1}	デシ	d
10^{-2}	センチ	c
10^{-3}	ミリ	m
10^{-6}	マイクロ	μ
10^{-9}	ナノ	n
10^{-12}	ピコ	p
10^{-15}	フェムト	f
10^{-18}	アト	a

表3　固有の名称をもつSI組立単位

量	名　称	記号	他のSI単位による表現
周波数	ヘルツ	Hz	s^{-1}
力	ニュートン	N	m·kg/s^2
圧力，応力	パスカル	Pa	N/m^2
エネルギー,仕事,熱量	ジュール	J	N·m
工率，放射束	ワット	W	J/s
電気量，電荷	クーロン	C	A·s
電位，電圧，起電力	ボルト	V	W/A
静電容量	ファラド	F	C/V
電気抵抗	オーム	Ω	V/A
コンダクタンス	ジーメンス	S	A/V
磁束	ウェーバ	Wb	V·s
磁束密度	テスラ	T	Wb/m^2
インダクタンス	ヘンリー	H	Wb/A
セルシウス温度	セルシウス度	℃	
光束	ルーメン	lm	cd·sr
照度	ルクス	lx	lm/m^2
放射能	ベクレル	Bq	s^{-1}
吸収線量	グレイ	Gy	J/kg
線量等量	シーベルト	Sv	J/kg

表4　SIと共に暫定的に維持される単位

名　称	記　号
オングストローム	Å
バーン	b
バール	bar
ガル	Gal
キュリー	Ci
レントゲン	R
ラド	rad
レム	rem

1 Å=0.1nm=10^{-10}m

1 b=100fm²=10^{-28}m²

1 bar=0.1MPa=10^5Pa

1 Gal=1cm/s²=10^{-2}m/s²

1 Ci=3.7×10^{10}Bq

1 R=2.58×10^{-4}C/kg

1 rad=1cGy=10^{-2}Gy

1 rem=1cSv=10^{-2}Sv

（注）

1．表1－5は「国際単位系」第5版，国際度量衡局1985年刊行による。ただし，1 eVおよび1 uの値はCODATAの1986年推奨値によった。

2．表4には海里，ノット，アール，ヘクタールも含まれているが日常の単位なのでここでは省略した。

3．barは，JISでは流体の圧力を表わす場合に限り表2のカテゴリーに分類されている。

4．EC閣僚理事会指令ではbar，barnおよび「血圧の単位」mmHgを表2のカテゴリーに入れている。

換　　算　　表

力	N(=10^5dyn)	kgf	lbf
	1	0.101972	0.224809
	9.80665	1	2.20462
	4.44822	0.453592	1

粘　度　1 Pa·s(N·s/m²)=10 P(ポアズ)(g/(cm·s))

動粘度　1m²/s=10^4St(ストークス)(cm²/s)

圧力	MPa(=10bar)	kgf/cm²	atm	mmHg(Torr)	lbf/in²(psi)
	1	10.1972	9.86923	7.50062×10^3	145.038
	0.0980665	1	0.967841	735.559	14.2233
	0.101325	1.03323	1	760	14.6959
	1.33322×10^{-4}	1.35951×10^{-3}	1.31579×10^{-3}	1	1.93368×10^{-2}
	6.89476×10^{-3}	7.03070×10^{-2}	6.80460×10^{-2}	51.7149	1

エネルギー・仕事・熱量	J(=10^7 erg)	kgf·m	kW·h	cal(計量法)	Btu	ft·lbf	eV
	1	0.101972	2.77778×10^{-7}	0.238889	9.47813×10^{-4}	0.737562	6.24150×10^{18}
	9.80665	1	2.72407×10^{-6}	2.34270	9.29487×10^{-3}	7.23301	6.12082×10^{19}
	3.6×10^6	3.67098×10^5	1	8.59999×10^5	3412.13	2.65522×10^6	2.24694×10^{25}
	4.18605	0.426858	1.16279×10^{-6}	1	3.96759×10^{-3}	3.08747	2.61272×10^{19}
	1055.06	107.586	2.93072×10^{-4}	252.042	1	778.172	6.58515×10^{21}
	1.35582	0.138255	3.76616×10^{-7}	0.323890	1.28506×10^{-3}	1	8.46233×10^{18}
	1.60218×10^{-19}	1.63377×10^{-20}	4.45050×10^{-26}	3.82743×10^{-20}	1.51857×10^{-22}	1.18171×10^{-19}	1

1 cal＝4.18605J（計量法）

＝4.184J（熱化学）

＝4.1855J（15℃）

＝4.1868J（国際蒸気表）

仕事率　1 PS(仏馬力)

＝75 kgf·m/s

＝735.499W

放射能	Bq	Ci
	1	2.70270×10^{-11}
	3.7×10^{10}	1

吸収線量	Gy	rad
	1	100
	0.01	1

照射線量	C/kg	R
	1	3876
	2.58×10^{-4}	1

線量当量	Sv	rem
	1	100
	0.01	1

（86年12月26日現在）

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