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(APRIL 1, 2000 – MARCH 31, 2001)

March 2002

Department of Nuclear Energy System

日本原子力研究所
Japan Atomic Energy Research Institute
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Japan Atomic Energy Research Institute
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This report summarizes the research and development activities in the Department of Nuclear Energy System during the fiscal year of 2000 (April 1, 2000 - March 31, 2001).

The Department has been organized from April 1998. The main research activity is aimed to build the basis of the development of future nuclear energy systems. The research activities of the Department cover basic nuclear and atomic & molecular data evaluation, conceptual design of a reduced-moderation water reactor, reactor physics experiments and development of the reactor analysis codes, experiment and analysis of thermal-hydrodynamics, energy system analysis and assessment, development of advanced materials for a reactor, lifetime reliability assessment on structural material, development of advanced nuclear fuel, study of nuclear transmutation systems, design of a marine reactor and the research for a nuclear ship system. The maintenance and operation of reactor engineering facilities belonging to the Department are undertaken.

The activities of the research committees to which the Department takes a role of secretariat are also summarized in this report.

Keywords: Nuclear Energy System Department Annual Report, Nuclear Data, Advanced Reactor, Reactor Physics, Thermal-hydraulics, Energy System Analysis, Advanced Material, Advanced Nuclear Fuel, Nuclear Transmutation, Marine Reactor

Board of Editors for Annual Report:
平成12年度エネルギーシステム研究部年報

日本原子力研究所東海研究所
エネルギーシステム研究部

（2002年1月24日受理）

本報告書は、平成12年度におけるエネルギーシステム研究部の研究活動状況をとりまとめたものである。

エネルギーシステム研究部は、平成10年度より新たに編成された部であり、将来型炉等新たなエネルギーシステムを視野に入れた基礎基盤的な研究を進めている。平成12年度のエネルギーシステム研究部の研究分野は、核及び原子分子データの評価、低減速スペクトル炉の概念設計研究、炉物理実験及び炉特性解析コードの開発、伝熱流動実験及び解析、原子力エネルギーの経済性評価、原子炉用新材料の開発及び経年挙動の研究、新型燃料の研究、核変換システムの研究、船用炉及び原子力船舶システムの研究等に亘っている。この他、エネルギーシステム研究部の各種炉工学施設の維持・管理も行っている。

本報告では、エネルギーシステム研究部が運営を担当する研究委員会の活動報告もとりまとめられている。

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Preface

The research activities of the Department of Nuclear Energy System, Japan Atomic Energy Research Institute, during the fiscal year 2000 (April 1, 2000 – March 31, 2001) are presented in this report. The Department is expected to develop an advanced nuclear energy system and to establish the related fundamental technologies for the system. The research activities cover such field as reactor physics, thermal-hydraulics, material science including research for advanced fuel and R&D for nuclear ship.

The total number of permanent staff working in the Department as of March 31, 2001 was 125 including the clerical service staff. The Department was funded from JAERI expenditure amounting to 808 million yen for FY 2000, excluding nuclear fuel cost and personnel expense. About 205 million yen was provided by the research contracts with external organizations; Science and Technology Agency (STA) for demonstration of reliability of new materials used in nuclear fuel reprocessing plant, Japan Nuclear Cycle Development Institute (JNC) for improvement of the evaluated nuclear data library for minor actinides, the Institute of Applied Energy for the development of the Reduced Moderation Water Reactor with passive safety equipment, the Institute of research and innovation (IRI) for research on IASCC data preparation for maintenance rule of aging BWR plants, and Mitsubishi Research Institute Inc. (MRI) for investigation of the development and research on cladding materials. The Department has served as the secretariat of Japanese Nuclear Data Center Committee, the Research Committee on Reactor Physics, the Research Committee on Marine Reactors and the Research Committee on Partitioning and Transmutation System.

The research activities have been conducted in eleven research groups with the support of two divisions.

Nuclear Data Center

We have two research themes; one is for nuclear data and the other is for atomic and molecular (A&M) data. As to the first one, the research activities are targeted to the JENDL (Japanese Evaluated Nuclear Data Libraries, including Special Purpose Files as well as General Purpose File) developments needed in nuclear developments and nuclear data measurements to enrich the fundamental database. As the reevaluation work for JENDL-3.3 has almost completed, file editing and their benchmark tests have been progressed. Wide range of benchmark test has been conducted with the aide of Japanese Nuclear Data Committee (JNDC) so as to be released in the next fiscal year. Considerable progresses are also made for JENDL high-energy file. A new Chart of
Nuclides (Version 2000) was published. Fundamental database having been developed so far are available through our home page (http://wwwndc.tokai.jaeri.go.jp). As to the A&M data, main efforts were devoted to compilation and evaluation of atomic and molecular collision data for JAEMDL-5. Due to the restructuring of research themes, this theme is deceased at the end of September 2000. Data maintenance for A&M data will be given by the Plasma Analytical Laboratory of Naka Research Establishment hereafter.

This center has also a function of the National Data Center as disseminating the nuclear data to Japanese customers, contacting foreign or international centers to exchange the nuclear data information, and coordinating international collaboration to enhance Research and Developments of nuclear data. To coordinate the development of JENDL, this center serves as a secretariat of Japanese Nuclear Data Committee (JNDC). We are going to hold an International Conference on Nuclear Data for Science and Technology (ND2001) at October 2001 in Tsukuba City. Preparation work has been progressed.

**Research Group for Energy System Assessment**

In the analysis of energy systems, i) a macro-economy model was developed and modified in order to analyze economic impacts of nuclear phase-out in Japan, ii) a preliminary analysis was made on impacts of nuclear phase-out by using a stand-alone energy model, iii) experience curves were investigated on photovoltaic power systems in order to assess future roles of dispersed generation technologies in the long-term evolution of electric power systems. In the analysis of nuclear fuel cycle systems, i) an economic assessment was made on plutonium recycling technologies focusing on economics of reduced moderation water reactors, ii) a database was developed on spent fuel reprocessing wastes of future nuclear power reactors.

**Research Group for Advanced Reactor System**

Main research activities of this group are the studies of reduced-moderation water reactor (RMWR), which include core and system design studies, and basic thermal hydraulic investigation. Another research areas are developments of diagnosis system of nuclear power plants based on the neural networks, and new reactor instrumentation by utilizing optical fiber.

**Research Group for Reactor Physics**

Research field of this group covers very wide range of fundamental studies on reactor physics and their applications to nuclear fuel cycle systems. Main areas are development of calculation code systems, reactor physics experiments using critical assemblies FCA and TCA, new reactor concepts
and non-destructive assay of TRU content in radioactive waste package. The code development work has been continued for high speed and high accuracy Monte Carlo codes and nodal codes to construct a comprehensive code system for reactor core design including thermal-hydraulics, core management and kinetics. The experimental studies have been concentrated on the reduced-moderation water reactors, accelerator-driven sub-critical systems, gas-cooled fast reactors and so on.

Research Group for Thermal and Fluid Engineering

Design studies were widely performed to check the feasibility of the proposed concepts of reduced moderation water reactors (RMWRs) under normal operation and several abnormal events, as well as a model experiment for critical heat flux (CHF) in a tight lattice core. Performance of the ITER vacuum vessel pressure suppression system during an ingress-of-coolant event (ICE) was investigated quantitatively using an integrated ICE test facility.

Research Group for Reactor Structural Materials

This research group carries out irradiation assisted stress corrosion cracking (IASCC) study and development of material performance database. The group has aimed at clarifying the IASCC mechanism in in-core structural materials like stainless steels, at developing the method of evaluation and prediction of IASCC behavior, and at developing the countermeasures to IASCC. The group has also aimed at constructing the material database networking system.

Research Group for Compatible Materials

R&D on the advanced materials have been extensively carried out aiming at the development of the advanced materials on the basis of the results of the material evaluation. The mechanism of SCC on zirconium and local attacks of steels in nitric acids were analyzed and modeled. On the demonstration test of material performance for Rokkasho reprocessing plant sponsored by the STA, the acid recovery evaporator and dissolver mock-ups were operated and evaluated the corrosion-resistance.

Research Group for Advanced Fuel

Activities of the research group are focused on R&D on transuranium fuels from the viewpoint of realizing flexible fuel cycle system in future. Plutonium rock-like oxide fuel has been developed for burning excess plutonium in existing LWRs, followed by direct disposal of spent fuel as a short-term subject. As a long-term subject, nitride fuel has been investigated for
both transmutation of long-lived minor actinides and advanced fast reactors, being coupled with pyrochemical reprocessing by molten salt electrorefining.

Research Group for Nuclear Transmutation System

As to a conceptual design of an accelerator driven transmutation system (ADS), the transmutation capabilities were confirmed for the minor actinides in the core region and the long-lived fission products such as I-129 in the blanket region. The transient analyses of the ADS plant have been also made for beam trip events in the proton accelerator. Static corrosion tests in liquid lead-bismuth were performed in a basic research program of the materials of the ADS. Furthermore, the international collaboration has been progressed for the participation in MEGAPIE experiments, which is the project for demonstration of a lead-bismuth spallation target for 1MW of beam power at PSI, Switzerland.

Advanced Marine Reactor Laboratory

Two concept designs of very small-scale and highly compact reactors with 1.25 MWt for an under-sea research vessel and with 1 MWt for heat supply at an office building have completed. They include designs of the core design for a long life operation, the radiation shielding, the safety system, the steam generator, and so on. Design study on a heat supply system by siting a 100 MWt reactor underground deeply at a big city has commenced.

Nuclear Ship System Laboratory

Hybrid automatic operation system for the advanced marine reactor capable of smooth transfer from a normal operation to an abnormal operation has been developed on the base of operator’s knowledge. The nuclear engineering simulation system of the advanced marine reactor, which has been used for designing of the automatic operation system was improved on hardware such as simulator’s control system, graphical tool, CRT and so on.

Reactor Engineering Facility Operation Division

This division operated three large-scale engineering facilities; FCA, TCA and Heat Transfer Fluid Flow Test Facilities in accordance with each experiment program and maintained in the monthly or the annual inspection. Consequently safety operations of these facilities were achieved and contributed sufficiently to the execution of each experimental study. Furthermore the maintenance work and decommissioning for VHTRC was carried out.
The Department is involved in the following project-oriented program in JAERI;

(1) Design Studies of Advanced Reactors,
(2) Development of High Temperature Gas-Cooled Reactor,
(3) Engineering Research for a Fusion Reactor.

The activities of the Department in FY 2000 have contributed to the essential progress in the field of engineering.

Masa-aki Ochiai,
Director,
Department of Nuclear Energy System
October 31, 2001
1. Nuclear Data, and Atomic and Molecular Data

Both projects for JENDL-3.3 General Purpose File and JENDL Special Purpose Files have been progressed significantly with the cooperation of JNDC (Japanese Nuclear Data Committee) during this period.

As to the JENDL-3.3 General Purpose File project, evaluation work is almost finished and final file-editing is going on for the release of JENDL-3.3 planned within the FY2001. So much feedback information from integral tests has been made. Re-evaluation work for Np-237 and Am-241 have been made to substitute for JENDL-3.3 and JENDL Actinide File. Also reevaluation for Na and complete new evaluation for Er isotopes (total of 6 isotopes) have been made. Er isotopes evaluation are especially made for high burn-up applications for advanced nuclear reactors. Sensitivity analysis for Pu-238 bare system based on JENDL-3.2 and ENDF/B-VI data has been made for the critical mass values and importance in evaluations for elastic cross sections are shown compared with other important reaction cross sections such as fission or capture.

For the Intermediate Energy File, which is important for the Accelerator Driven System in the Neutron Science Research, much effort is made for the file developments for JENDL High Energy File, JENDL PKA/KERMA File and JENDL Photonuclear Data File. The evaluation work for the neutron and proton files up to 3GeV has been made.

Mass chain evaluation has been continued within the framework of the international cooperation for the ENSDF File. The evaluation of mass chain A=121 has been made in the year of 2000 out of 12 our responsible mass chains. A new version of Nuclear Chart (Chart of Nuclides 2000: JAERI) has been published and distributed by JAERI Nuclear Data Center. The data are also accessed from our home page (http://wwwndc.tokai.jaeri.go.jp).

For the preparation of group constants, the libraries for ORIGEN2 Code based on JENDL-3.2 has been produced for the specific MOX fuels utilized in Japan for LWRs. Good performances on the LWRs were confirmed by analyzing post irradiation experiments (PIE) performed in European PWRs and the SAXTON reactors in USA.

As to the atomic and molecular (A&M) data, angle differential radiative recombination cross sections in the K, L and M-shells for one-electron systems are calculated by the exact relativistic model and their characteristics are discussed. The research works for A&M data keeping brilliant histories more than 25 years were deceased on 30 September 2000 in JAERI NDC due to the restructuring of research theme in our Department.
1.1 Evaluation of Neutron Nuclear Data for $^{237}$Np and $^{241}$Am

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Nuclides of $^{237}$Np and $^{241}$Am are the most important ones among minor actinides. In order to recommend the evaluated nuclear data for JENDL-3.3 and JENDL Actinide File under construction, re-evaluation work was performed for these two nuclides. The data stored in the current version of JENDL-3.2 were carefully compared with available experimental data and recent evaluated data. For the quantities whose drawbacks were found, new evaluation was made or the most reliable evaluated data were adopted. Figures 1.1.1 and 1.1.2 show the cross-section data of $^{237}$Np and $^{241}$Am. The cross sections in the energy range from 0.3 to 500 eV for $^{237}$Np and from 0.3 to 150 eV for $^{241}$Am are average values in suitable intervals in the figures.

For the data of $^{237}$Np, the resonance parameters were improved by considering new experiments. The fission cross section was evaluated by using available experimental data and the least-squares fitting code GMA. Other cross sections were improved by adopting reliable evaluated data. The fission spectra and the number of delayed neutrons per fission were also improved.

For the data of $^{241}$Am, the resonance parameters were improved by considering the fission cross section measured at the Kyoto University and thermal cross sections. The fission cross section of $^{241}$Am was evaluated with the GMA code. The others improved are the capture, inelastic scattering, (n,2n) and (n,3n) reaction cross sections, energy distributions and number of neutrons per fission.
Fig. 1.1.1  Cross sections of $^{237}$Np

Fig. 1.1.2  Cross sections of $^{241}$Am
1.2 Evaluation of Sodium and Erbium Data for JENDL-3.3

K. Shibata and Medium-Heavy Nuclide Data Evaluation Working Group* of Japanese Nuclear Data Committee
(e-mail: shibata@ndc.tokai.jaer.go.jp)

Neutron nuclear data for $^{23}$Na and $^{162, 164, 166, 167, 168, 170}$Er have been evaluated for JENDL-3.3 in the energy region from $10^{-5}$ eV to 20 MeV. Nuclear model calculations were mainly adopted in these evaluations.

Most of the reaction cross sections of $^{23}$Na were calculated by using the multi-step statistical model code TNG. The Reference Input Parameter Library (RIPL), which was developed at IAEA, was used to prepare data required as input to TNG. The neutron optical model parameters were adjusted so as to reproduce measured total cross sections with the least-squares method. We adopted Perey’s and Huizenga-Igo’s optical model parameters for protons and α-particles, respectively. As for nuclear level density, the composite formula was adopted for sodium isotopes, while the constant temperature model was used for fluorine and neon isotopes. The direct process for inelastic scattering was taken into account with DWBA calculations. The calculated $^{23}$Na(n,n$'$γ) cross sections are in good agreement with the experimental data measured by Dickens et al. as seen in Fig. 1.2.1. The (n,2n) reactions cross sections are illustrated in Fig. 1.2.2. It is found from the figure that the ENDF/B-VI data are much larger than the present evaluation and the JENDL-3.2 data above 14 MeV.

The resolved resonance parameters of erbium isotopes were taken from the compilation of Sukhoruchkin et al. Negative resonances were assumed for $^{166, 168, 170}$Er. Unresolved resonance parameters were evaluated only for $^{167}$Er. Reaction cross sections of $^{162, 164, 166, 167, 168, 170}$Er were evaluated using the statistical model codes EGNASH and CASTHY. The neutron optical model parameters were determined so as to reproduce measured total cross sections of elemental erbium. Erbium is regarded as a candidate for burnable poison, and thus its neutron capture cross sections are important from a viewpoint of reactor physics. Neutron capture cross sections of $^{166, 167, 168}$Er were measured by Harun-ar-Rashid et al. These measurements were taken into account in the present evaluation. Figure 1.2.3 shows the capture cross sections of $^{166, 167}$Er.

* Members are K. Shibata, T. Asami (Data Engineering, Inc.), Y. Harima (CRC), M. Igashira (Tokyo Institute of Technology), H. Kitazawa (National Defense Academy), T. Watanabe (Kawasaki Heavy Industry), Y. Watanabe (Kyushu University), N. Yamamuro (Tokyo Institute of Technology).
Fig. 1.2.1 $^{23}$Na(n,n'γ) cross sections.

Fig. 1.2.2 $^{23}$Na(n,2n) cross sections
The evaluated data were compiled in the ENDF-6 format and will be made available as JENDL-3.3.

References:
1.3 Integral Test of JENDL-3.2 for Bare $^{238}$Pu Metal System

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Estimation of critical mass of $^{238}$Pu was carried out by K.R. Yates $^1$ as part of revision work of ANSI/ANS-8.15-1981 "Nuclear Criticality Control of Special Actinide Elements" $^2$. The results indicated that the critical mass estimated by using JENDL-3.2 nuclear data library gave relatively smaller value than that estimated by ENDF/B-VI. To clarify the difference of the critical mass calculated from the two nuclear data libraries, integral test of $^{238}$Pu cross section data for bare-spherical metal system was carried out.

The integral test was carried out for 100% $^{238}$Pu metal sphere without the reflector. Radius of the system was determined to be 4.956 cm from critical mass, 10.07 kg, which was reported in the reference $^1$, and density of $^{238}$Pu metal, 19.75 g/cm$^3$. The MCNP4B2 code $^3$ and continuous energy libraries generated from JENDL-3.2 and ENDF/B-VI were used for the present calculation. All the MCNP4B2 calculations for the present integral test used 150 generations, 20,000 neutrons/generation, 50 generations skipped.

Table 1.3.1 shows $k_{\text{eff}}$ and $k_{\text{inf}}$ of the $^{238}$Pu metal system calculated from JENDL-3.2 and ENDF/B-VI. Approximately 5% difference can be seen in the calculated $k_{\text{eff}}$. To investigate sensitivity of cross sections to the multiplication factors, the multiplication factors were evaluated using pseudo nuclear data library that one of cross section data such as fission, capture, elastic scattering cross sections, fission spectrum, and v-p in JENDL-3.2 was replaced with data in ENDF/B-VI. The calculated results are also shown in Table 1.3.1. From the calculation of $k_{\text{eff}}$ by using the generated libraries, it was found that the discrepancy of $k_{\text{eff}}$ was mainly attributed to large elastic scattering cross section values stored in JENDL-3.2 compared to the ENDF/B-VI data as shown in Fig.1.3.1. Contrary to the $k_{\text{eff}}$ calculation, it can be said that the capture cross section is rather sensitive to the $k_{\text{inf}}$ value and the elastic scattering cross section has no sensitivity to the $k_{\text{inf}}$.

In nuclear data evaluation, the elastic scattering cross section is often used as an adjustable cross section to retain consistency between total cross section and sum of partial cross sections. The present study indicated that the elastic scattering cross section also has sensitivity to $k_{\text{eff}}$ of the fast-bare-metal critical system and is required to be evaluated in higher accuracy.
References


Table 1.3.1 Calculated multiplication factors by using various nuclear data libraries

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<th>%Δk/k</th>
<th>kinf</th>
<th>%Δk/k</th>
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<td>-</td>
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<td>2.762</td>
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Replaced cross section

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<td>Capture</td>
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<td>2.794</td>
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<tr>
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<td>Elastic scattering</td>
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Fig.1.3.1 Comparison of elastic scattering cross section in JENDL-3.2 and ENDF/B-VI
1.4 Evaluation and Compilation of Nuclear Structure and Decay Data in 2000

J. Katakura and ENSDF Group*
(E-mail: katakura@ndc.tokai.jaeri.go.jp)

The international network on nuclear structure and decay data evaluation aims at complete and continuous nuclear structure and decay data evaluation of all mass chains. The evaluated data are compiled as ENSDF (Evaluated Nuclear Structure Data File) file. The data file is maintained and distributed by National Nuclear Data Center, Brookhaven National Laboratory (BNL), U.S.A. As a member of the network, Japanese group, whose data evaluation center is Nuclear Data Center, Japan Atomic Energy Research Institute, has responsibility for evaluating 12 mass chains with A=118-129.

In the fiscal year of 2000 (from April 2000 to March 2001), the evaluations of A=121 mass chain was published in Nuclear Data Sheets 1). The evaluation includes all experimental data available after the last evaluations, 1990 for A=121. The published data sets of A=121 are listed in table 1.4.1. These data sets are evaluated based on the all available measured data relating to A=121 mass. Each data set excepting “Adopted Levels, Gammas” or “Adopted Levels” data sets is the evaluated data set from the same kind of measurements categorized in the name of the data set. The “Adopted Levels, Gammas” or “Adopted Levels” data sets are compiled from the each data set in the same nuclide.

The new data from the last evaluation of other mass chains to which Japanese group has the responsibility are reviewed and being prepared for update of the last mass chain evaluation.

References:


*Members are H. Iimura, M. Oshima, S. Ohya, J. Katakura, M. Kambe, K. Kitao, T. Tamura and Y. Tendow
Table 1.4.1  Evaluated data sets of A=121

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<td>$^{122}$Te(t,α)</td>
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<tr>
<td>$^{121}$Ag</td>
<td>Adopted Levels</td>
<td></td>
<td>$^{123}$Sb(p,t)</td>
</tr>
<tr>
<td>$^{121}$Cd</td>
<td>Adopted Levels, Gammas</td>
<td>$^{121}$Te</td>
<td>Adopted Levels, Gammas</td>
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<tr>
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<td></td>
<td>$^{121}$Te IT Decay</td>
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<tr>
<td></td>
<td>$^{121}$Cd β$^{-}$ Decay (8.3 s)</td>
<td></td>
<td>$^{121}$I e Decay</td>
</tr>
<tr>
<td></td>
<td>$^{121}$Cd β$^{-}$ Decay (13.5)</td>
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<td>$^{114}$Cd($^{11}$B,p3nγ)</td>
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<td></td>
<td>$^{121}$In IT Decay (3.88 min)</td>
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<td></td>
<td>$^{124}$Sn(p,α)</td>
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<td>$^{120}$Te(d,p)</td>
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<td>$^{122}$Te(p,d)</td>
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<td>$^{119}$Sn(t,p)</td>
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<td>$^{121}$Sb(d,d'), (α,α')</td>
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<td>$^{121}$Cs e Decay (155 s)</td>
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<td></td>
<td>$^{121}$Ba e Decay</td>
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</table>
1.5 ORIGEN2 Libraries for LWR-MOX Fuels Based on JENDL-3.2

K. Suyama, M. Onoue, H. Matsumoto, A. Sasahara and J. Katakura
(E-mail: katakura@ndc.tokai.jaeri.go.jp)

A set of ORIGEN2 libraries for LWR-MOX fuels has been developed based on
JENDL-3.2 1). In the compilation of the libraries SWAT code system 2) was used. The
SWAT code system is the code system composed of general purpose neutronics code
system SRAC95 3) and point burnup code ORIGEN2 4).

Enrichment of Pu and Pu content of MOX fuels were treated as parameters for
PWR fuels. In addition to them void ratio was also treated as a parameter for BWR
fuels. These parameters were selected considering the specification of the MOX fuels
which is expected to be used in nuclear power reactors in Japan. Then five libraries for
PWR and seven libraries for BWR were compiled. These libraries are listed in Table
1.5.1. A breakdown of the Pu contents of PWR and BWR MOX fules in Table 1.5.1 are
given in detail in Tables 1.5.2 and 1.5.3. As seen in Table 1.5.1, assumed burnups are
60 GWD/t for PWR and 50 GWD/t for BWR.

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<th>Burnup (GWD/t)</th>
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<th>Pu Enrichment (wt%)</th>
<th>Pu Contents</th>
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The verification of the compiled libraries were performed by the analyses of
isotopic composition data taken in post irradiation examinations (PIE) for 14 × 14
Table 1.5.2  Pu contents for PWR MOX fuel

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<th>Pu Contents</th>
<th>Composition 1</th>
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<td>(%) Pu-241</td>
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<td>Pu-242</td>
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<td>Am-241</td>
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Table 1.5.3  Pu contents for BWR MOX fuel

<table>
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<tr>
<td>Ratio Pu-241</td>
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</tr>
<tr>
<td>(%) Pu-242</td>
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<td>5.59</td>
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<tr>
<td>Am-241</td>
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MOX spent fuels used in European PWR reactor. Nuclide concentrations of actinides and fission products were compared between the measurements and the calculation using new libraries. Even though the specification of the irradiated MOX fuels was different from that used for making ORIGEN2 libraries, the comparison showed that the calculated nuclide concentrations were within at most 20 % of the PIE data. The effects of the parameters used for making ORIGEN2 libraries on the calculated results were examined by analyzing the PIE data of SAXTON reactor, USA. From these analyses, the effects of the different parameters to the calculated results were found to be not so large and the Pu contents showed to be consistent within 10 %.

The new libraries for MOX fuels compiled in the present work have been included in ORLIBJ32 which has already released as ORIGEN2 libraries for LWR fuels for Japanese nuclear reactors in 1999, and it is available from NEA Data Bank as NEA-1642 (ZZ-ORLIBJ32 1.1).
References:


1.6 Angle-Differential Cross Sections for Radiative Recombination and the Photoelectric Effect in the K, L, and M-Shells of One-Electron Systems Calculated within an Exact Relativistic Description

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(E-mail: ichihara@ndc.tokai.jaeri.go.jp)

Radiative recombination (RR) is one of the dominant atomic collision processes occurring in the fusion plasma. In this reaction, an electron is captured into an empty atomic shell of the bare nucleus and a photon is emitted simultaneously. In the accelerator experiments, the photoelectric effect for the high-Z hydrogen-like ions have been studied by the inverse reaction, RR, where these ions can not be easily used as targets. Thus, the RR cross sections are needed over a wide range of nuclear charges and electron energies to investigate the fusion plasmas as well as the photoelectric effect.

We have carried out a tabulation of angle-differential RR cross sections for nuclear charges $Z=18,36,54,66, 79, 82$, and 92. The cross sections have been calculated for RR into the K, L, and M (1s$_{1/2}$, 2s$_{1/2}$, 2p$_{1/2}$, 2p$_{3/2}$, 3s$_{1/2}$, 3p$_{1/2}$, 3p$_{3/2}$, 3d$_{3/2}$, 3d$_{5/2}$) shells in the electron incident energy range of 1.0 keV to 1.5 MeV. Cross sections are obtained from complete relativistic calculations using exact Coulomb-Dirac wave functions for the bound and for the continuum states modified for the finite nuclear size. They also include all multipole orders of the photon field. The tabulated cross sections are accurate to three digits and should be considered as the results of benchmark calculations.

Differential cross sections per vacancy for RR given in the rest frame of the electron are tabulated as a function of the photon emission angle with respect to the ion motion towards the electron. These cross sections can directly be applied to the accelerator experiments. Corresponding photoelectric cross sections can be derived from the RR cross sections by the principle of the detailed balance. Then the photoelectric cross sections in the rest frame of the ion are obtained by the Lorentz transformation. Moreover, the dependence of RR cross sections on the projectile energy in MeV/u can be obtained by a simple scaling for accelerator experiments, and the dependence of the photoelectric cross sections on the photon incident energy in keV can be obtained by using elementary formulas.

* Bereich Theoretische Physik, Hahn-Meitner-Institut Berlin
Figure 1.6.1 shows the K-RR angle-differential cross sections of Z=92 for various electron energies ($T_e$), where the solid line represents the cross section and the dashed line represents the contribution of magnetic spin-flip transitions. It can be seen in Fig. 1.6.1 that the maximum of cross section gradually shifts to forward angles as the energy increases. Nonzero cross sections at 0° and 180° can be unambiguously assigned to magnetic spin-flip transitions because of angular momentum conservation. The spin-flip effect can not be considered in the non-relativistic calculation. The relativistic spin-flip effect for the K capture at forward angles is still appreciable at the lowest energy of 1.0 keV for Z=92 owing to highly relativistic 1s_{1/2} wave function. On the other hand, for lower charges (Z ≤ 54) and low energies ($T_e < 100$ keV), we observed that the angular distributions for the 1s_{1/2} states closely follow a $\sin^2 \theta$ shape, which is in accord with the non-relativistic calculation.

From the present calculations we confirmed that the angular distributions for the 1s_{1/2}, 2s_{1/2}, and 3s_{3/2} states have similar shapes. Regarding higher-angular momentum states, we confirmed that the angular distributions for the p_{1/2} states and for the p_{3/2} states, respectively, are similar in the L and in the M shell. Their maxima, as well as those of the d_{3/2} and the d_{5/2} states shift to backward angles as the energy increases. Furthermore, the effect in the angular distribution of the spin-orbit splitting between the p_{1/2} and p_{3/2} states as well as between the d_{3/2} and d_{5/2} states increases both with the energy and with the nuclear charge number Z.

Finally, from a comparison of cross section curves, we found that the angle-differential cross sections increase rapidly with the nuclear charge, but the shape changes slowly with the charge number Z. Therefore, with the seven tabulated charge numbers covering most of the Periodic Table, good approximations to a shape of the angular distributions for other values of Z should be given by a suitable interpolation procedure. The absolute values can then be obtained by using the total cross sections, which are tabulated for all integer values of Z. 3)

References:
2) Ichihara, A. and Eichler, J.: Atomic Data and Nuclear Data Tables, in press.
Fig 1.6.1 Angle-differential cross sections for K-RR of Z=92 for various electron energies $T_e$
2. Reactor Physics

In order to vectorize the characteristics method for heterogeneous neutron transport calculations, two vectorization algorithms, "Odd-even Sweep" and "Independent Sequential Sweep" methods have been studied. It has been found that a speed-up factor of more than 10 is achievable by the both algorithms and that the "Independent Sequential Sweep" method is superior from a viewpoint of faster convergence and smaller required memory size.

The benchmark test has been carried out on the STACY critical experiments for the JENDL-3.3 evaluation. The first evaluation of $^{14}$N(n,p) cross section gives overestimate of $k_{eff}$ by 0.3%Δk. This overestimation is resolved by re-evaluation of $^{14}$N(n,p) cross section.

New functions, simulation of Feynman-α experiment and point detector estimator, have been added to the MVP continuous energy Monte Carlo code. Analyses of the reactor noise experiment at the FCA XIX-3 assembly have been made for the verification test of the former function. As for the point detector estimator, we have analyzed the experiment on neutron and gamma-ray doses in critical accident performed at the TRACY facility.

At the TCA, applicability of the source multiplication method to large perturbation reactivity measurement has been studied at a wide range of subcriticality level varied from −0.2 to 10$. Considerably large reactivities from 0.4 to 12$, introduced by replacement of UO$_2$ fuel rods with ThO$_2$ ones, could be measured by setting a $^{252}$Cf source and a $^3$He detector at appropriate location. A series of rod-drop experiments have been made in order to investigate a correction method for the space and neutron energy spectrum effects on the measurements. Correction factors, which are obtained by taking account of neutron flux change before and after the rod drop evaluated by the one-dimensional, one-group diffusion approximation, can reduce those effects on the measurements.

A feasibility study of the nitride fuel He-GCFR has been made. It has been shown that the excellent thermal characteristics of nitride fuel are useful for high performance GCFR and that the reactor shutdown by a control rod and an alternative coolant system are necessary to terminate the coolant depressurization accident without a failure of fuel pins. Concerning the ULOF transient, a self actuated shut down system (SASS) can effectively decrease the maximum temperature of cladding and keep cladding and fuel at the safe level. In the depressurization transient, the cladding temperature does not exceed the design criterion of 900°C when the final containment pressure is kept higher than 0.4MPa, which can be realized in the same size of reactor vessel as the present LWR.
2.1 Fast Vector Computation Algorithms for the Characteristics Method

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We have studied fast vector computation for the characteristics method\(^1\) for the neutron transport calculation in a heterogeneous geometry. In this study, we have developed two vector computation algorithms, an "Odd-Even Sweep (OES)" method and an "Independent Sequential Sweep (ISS)" method and investigated their efficiency to a realistic assembly.

In the characteristics method, a number of parallel straight lines are drawn for discrete directions on a system. Neutron propagation is calculated along a segment across medium as

\[
\phi_{i,k}^\text{out}(\Omega) = \phi_{i,k}^\text{in}(\Omega) \exp(-\Sigma_i s_{i,k}) + Q_i(\Omega) \frac{1 - \exp(-\Sigma_i s_{i,k})}{\Sigma_i},
\]

(1)

where \(\phi_{i,k}^\text{in}(\Omega)\) and \(\phi_{i,k}^\text{out}(\Omega)\) are an incoming and outgoing angular neutron fluxes at segment \(k\) in region \(i\), respectively, \(\Sigma_i\) the total cross section of region \(i\), \(s_{i,k}\) the segment length, and \(Q_i(\Omega)\) the neutron source of region \(i\) in the angular direction \(\Omega\). The calculation of the neutron propagation is carried forward along the line from an entering external boundary to an exiting external boundary, and it is done backward with an appropriate boundary condition. Once the calculation of the neutron propagation is finished on all the segments, then region average angular fluxes and sources are updated with the angular fluxes on all the segments. These processes are alternately repeated until the region average angular fluxes and sources are converged within given criteria. As shown in Eq. (1), an outgoing neutron flux calculated with use of a known incoming neutron flux becomes an incoming neutron flux for the spatially next segment. Thus, an outgoing neutron flux of a segment is recursively used for the calculation of that of the spatially next segment.

For fast vector computation, it is important to achieve long vector lengths and avoid recursive operations. In the OES method, segments are classified into odd and even ones as shown in Fig. 2.1.1. The neutron propagation is calculated first for the odd segments and then for the even ones. Even if the incoming fluxes for the odd segments are simultaneously updated, each of them does not influence on another one because they are not spatially next to each other. In the ISS method, segments are classified by their order numbers of spatial
position from the entering external boundary as shown in the figure. The calculation of the neutron propagation begins with the segments of the first position, then proceeds to those of the second one and continues up to those of the last one. The calculation of the neutron propagation is done only for one or less segment on each line in a DO-loop, therefore when an incoming neutron flux of a segment is updated, it does not influence on that of any segment on each line in the DO-loop. Vector lengths in the OES method will be greater than those in the ISS method, therefore the OES method is expected to have an advantage in the speed-up ratio of vector to scalar computations. However, the OES method needs a larger computer memory size because of the greater vector lengths. The method needs several sweeps until updated data near the entering boundary influences on the one near the exiting boundary. On the other hand, the ISS method needs only one sweep until updated data near the entering boundary influences on that near the exiting boundary. Therefore, the ISS method is expected to show a faster convergence than the OES method.

We have investigated efficiency of the vector computation by a typical PWR assembly with an octant symmetry. Absorber rods are inserted. The number of regions and external boundary surfaces are 155 and 130, respectively. We prepared tracking data with 8 uniformly spaced azimuthal angles and a line spacing of about 0.01 cm. The numbers of lines and segments on a horizontal plane is 12,344 and 236,020, respectively. The calculation was done with 4 polar angles and with 3-group constants. Convergence criteria for the region average angular fluxes and $k_{eff}$ are set to $10^{-5}$. The average vector length for the neutron propagation by the OES method is 118,010. The vector length for the external boundary treatment is 12,344. The memory size in the OES method is about 60 mega bytes. The average vector length for the neutron propagation by the ISS method is 5,131. The average vector length for
the exiting external boundary is 268. The memory size in the ISS method is about 12 mega bytes. The ISS method can decrease the memory size by a factor of 5 compared with the OES method.

To investigate efficiency of the vector computation, the present program was compiled in both vector and scalar modes and the same problem was solved on the same computer. The used computer is FUJITSU FACOM VPP-5000. Table 2.1.1 summarizes the computation time necessary for an outer iteration which includes 5 inner iterations. For both the OES and ISS methods, the computation time per outer iteration is about 6 seconds by the scalar computation and about 0.4 second by the vector computation. With use of the OES or ISS method, the vector computation is about 15 times faster than the scalar computation. A difference of the speed-up factors between the OES and ISS methods is little. The expected advantage of the OES method in the computation time by the long vector length is not observed in the realistic assembly calculation. The numbers of outer iterations by the OES and ISS methods are 169 and 111, respectively. The ISS method shows a faster convergence than the OES method. Consequently, it was found that the ISS method is superior to the OES method as a fast vector computation algorithm for the characteristics method.

<table>
<thead>
<tr>
<th></th>
<th>OES method</th>
<th>ISS method</th>
</tr>
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<tbody>
<tr>
<td>CPU time by scalar</td>
<td>6.33</td>
<td>6.09</td>
</tr>
<tr>
<td>computation (sec)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>CPU time by vector</td>
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<td>0.44</td>
</tr>
<tr>
<td>computation (sec)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Speed-up factor to</td>
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<td>13.9</td>
</tr>
<tr>
<td>scalar calculation</td>
<td></td>
<td></td>
</tr>
<tr>
<td>No. of outer</td>
<td>169</td>
<td>111</td>
</tr>
<tr>
<td>iterations</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

Reference:
2.2 Re-evaluation of (n,p) Cross Section of $^{14}\text{N}$ for JENDL-3.3 and Its Effect to Critical Benchmark Test of STACY Experiments Using 10wt% Enriched Uranyl Nitrate Solution

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Overestimation of criticality for the enriched uranyl nitrate solution system is one of the problems of the JENDL-3.2 file. It is expected that the JENDL-3.3\textsuperscript{1}) dissolves this problem. Table 2.2.1 shows the results of critical benchmark calculations with major nuclear data files for the STACY experiment using 10wt.% enriched uranyl nitrate solution fuel in a water reflected cylinder tank of 60cm diameter. The calculations were performed with a continuous energy Monte Carlo code MVP\textsuperscript{2}). The details of the experiments and the benchmark model are described in the International Benchmark Handbook\textsuperscript{3}). As shown in Table 2.2.1, only the JENDL-3.2 gives an overestimation of the criticality by more than 0.7%Δk. In the recent benchmark test with the preliminary evaluation for JENDL-3.3, the overestimation was mitigated mainly by re-evaluation of $^{235}\text{U}$ cross section, but overestimation of about 0.3%Δk is obstinately remaining\textsuperscript{4}).

In order to clarify the reasons of the remaining overestimation, MVP calculations for the STACY benchmark were carried out with the MVP library based on JENDL-3.2 and JEF-2.2. By changing a nuclide-wise library data one by one from JENDL-3.2 to JEF-2.2, it was found that $^{235}\text{U}$ and $^{14}\text{N}$ have large contributions to the overestimation. Their contributions are about 0.4%Δk and 0.2%Δk, respectively. Contributions of other nuclides were less than 0.05%Δk. As the overestimation caused by $^{235}\text{U}$ has been successfully resolved in the preliminary JENDL-3.3, further investigation was carried out on the cross section of $^{14}\text{N}$.

Figure 2.2.1 shows the energy-dependent difference of net absorption rate of $^{14}\text{N}$ calculated by MVP, when the data of $^{14}\text{N}$ is replaced by that of JEF-2.2 in the JENDL-3.2 library. It is observed that the difference of about 90% is caused in the thermal energy region. The contributing reaction type is (n, p), because the (n, p) cross section of $^{14}\text{N}$ is about 30 times larger than (n, γ) cross section in the thermal energy region. Table 2.2.2 shows the 1/ν cross section values of $^{14}\text{N}(n,p)$ at 0.0253eV of JENDL-3.2, JEF-2.2 and ENDF/B-VI. The value of JENDL-3.2 is smaller than those of JEF-2.2 and ENDF/B-VI by about 2-3%. The value of the preliminary JENDL-3.3 used in the previous benchmark test\textsuperscript{4}) was the same as
that of JENDL-3.2. Thus, $^{14}$N cross section was re-evaluated and the resultant (n,p) cross section at 0.0253eV was set to 1.830 barn.

The STACY benchmark calculations were carried out with the modified preliminary JENDL-3.3. The benchmark calculations were performed for the seven critical cores with different fuel concentrations and different core heights. The specifications of these cores are described in Ref.3). Table 2.2.3 shows the C/E values with and without new evaluation of $^{14}$N(n,p) cross section for JENDL-3.3. The remaining overestimation for the criticality of the STACY cores has been successfully resolved.

References:

<table>
<thead>
<tr>
<th>Table 2.2.1</th>
<th>Result of critical benchmark test for the STACY experiment (Run 29) with continuous energy Monte Carlo code MVP and different nuclear data files</th>
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<tr>
<td><strong>Nuclear Data File</strong></td>
<td><strong>JENDL-3.2</strong></td>
</tr>
<tr>
<td>C/E</td>
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</tr>
</tbody>
</table>

Geometry model ('detailed mode') and experimental values are taken from Ref.3).
Statistical errors (1σ) of MVP calculations are within 0.00010.

<table>
<thead>
<tr>
<th>Table 2.2.2</th>
<th>Comparison of $^{14}$N(n,p) cross sections at 0.0253eV</th>
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</thead>
<tbody>
<tr>
<td><strong>Nuclear Data File</strong></td>
<td><strong>JENDL-3.2</strong></td>
</tr>
<tr>
<td><strong>Cross Section</strong></td>
<td>1.7800 barn</td>
</tr>
</tbody>
</table>
Table 2.2.3 Result of STACY benchmark with MVP and evaluation for JENDL-3.3

(Effect of re-evaluated $^{14}$N(n,p) cross section)

<table>
<thead>
<tr>
<th>Run No.</th>
<th>Uranium Concentration (g/l)</th>
<th>Critical Height (cm)</th>
<th>C/E after re-evaluation of $^{14}$N(n,p)</th>
<th>C/E before re-evaluation of $^{14}$N(n,p)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>310.1</td>
<td>41.53</td>
<td>0.9994</td>
<td>1.0025</td>
</tr>
<tr>
<td>29</td>
<td>290.4</td>
<td>46.70</td>
<td>1.0013</td>
<td>1.0035</td>
</tr>
<tr>
<td>33</td>
<td>270.0</td>
<td>52.93</td>
<td>0.9990</td>
<td>1.0015</td>
</tr>
<tr>
<td>34</td>
<td>253.6</td>
<td>64.85</td>
<td>1.0012</td>
<td>1.0037</td>
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<td>46</td>
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<td>78.56</td>
<td>1.0018</td>
<td>1.0040</td>
</tr>
<tr>
<td>51</td>
<td>233.2</td>
<td>95.50</td>
<td>1.0010</td>
<td>1.0035</td>
</tr>
<tr>
<td>54</td>
<td>225.3</td>
<td>130.33</td>
<td>1.0011</td>
<td>1.0034</td>
</tr>
<tr>
<td>Average</td>
<td></td>
<td></td>
<td>1.0007</td>
<td>1.0032</td>
</tr>
</tbody>
</table>

Geometry model ('detailed mode) and experimental values are taken from Ref.3).
Statistical errors (1σ) of MVP calculations are within 0.00015.

![Graph](image)

Fig.2.2.1 Difference of net absorption of $^{14}$N when cross section data of $^{14}$N is replaced by JEF-2.2 in JENDL-3.2
2.3 Simulation of Feynman-α Experiment by Using a Continuous-Energy Monte Carlo Code MVP

Y. Nagaya and T. Mori
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Recently the Feynman-α method has attracted attention to the subcriticality measurement for an accelerator-driven reactor and the Monte Carlo method has been applied to analyze the measurement\(^1\). In the present work, we have implemented the function of the Feynman-α simulation into a continuous-energy Monte Carlo code MVP\(^2\) for general purposes.

In the procedure of the Feynman-α simulation by MVP, the number of histories per batch is constant and a mean value and its variance are estimated from the statistical processing in each batch. Source particles are generated uniformly in time. Then, the time spectrum is scored for the fundamental gate width and the variance-to-mean value for each gate width is estimated by bunching the spectrum.

Since the correlation of particles must be estimated in the Feynman-α method, the analog Monte Carlo simulation must be performed. Thus, we modify MVP so that the number of fission neutrons is sampled from the Terrel’s distribution and the number of neutrons can be scored in the analog manner for the absorption and scattering reactions. In addition, we added a new source option that can simulate neutrons generated simultaneously at a time from a spontaneous fission source like Cf-252.

In order to verify the new function, we analyzed the reactor noise experiment performed at the FCA XIX-3 assembly. The assembly is one of the cores for the benchmark experiments of the effective delayed neutron fraction and the fraction was measured by the covariance-to-mean method. Thus we re-analyzed the experimental data by the Feynman-α method.

Figure 2.3.1 shows the geometry of the assembly. The assembly consists of three regions; core, soft blanket, outer blanket. The core is constructed from plutonium and stainless steel plates only. Since it is characterized by the plutonium core, the delayed neutron fraction is very small. The measured fraction has been estimated to be 251 pcm\(^3\). Three levels of subcriticality were set up by adjusting two pairs of control rods. The control rod positions
were 10.5, 14.0 and 17.5 cm from the core center. The BF$_3$ counters are located in the soft blanket region as shown in Fig. 2.3.1.

We modeled the geometry as built in the MVP calculations. The counters were modeled by diluting B-10 uniformly in the counter regions. Since we analyzed the reactor noise experiment in the time domain where the effect of delayed neutrons can be neglected, we generate prompt fission neutrons only. Namely, we used $\nu_p$ instead of $\nu_t$. The time spectra were tallied in the counter regions and the variance-to-mean values ($Y$ values) were obtained with the new function of MVP. The values versus the gate widths were fitted by the least square method and then we obtained the prompt neutron decay constant ($\alpha$ value);

$$\alpha = \frac{B - \rho}{\Lambda}.$$ 

Figure 2.3.2 shows the calculated results of the $\alpha$ value for each level of subcriticality. The calculated results are added for 3 other levels of subcriticality. The second to fourth cross marks from the left side correspond to the experimental results. The large discrepancy can be found in the levels of subcriticality between the calculated and experimental results. This is caused by the uncertainty of nuclear data. Therefore, we compare the critical prompt decay constant or the generation time obtained from the slope of the fitted line. The experimental and calculated constants are 7325 (2.7%) and 12921 (7.9%), respectively. On the other hand, the experimental and calculated generation time are $3.43 \times 10^{-7}$ and $4.63 \times 10^{-7}$, respectively. There still exist large discrepancies and the cause of them is currently under investigation.

References:
Fig. 2.3.1 Geometry of FCA XIX-3 assembly

Fig. 2.3.2 Prompt neutron attenuation constant versus subcriticality
2.4 Development of Fast Point Detector Estimator for Continuous Energy Monte Carlo Code MVP

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The point detector estimator is widely used in experimental analyses and shielding calculations by the Monte Carlo method. This estimator requires much computation time as well known. In the present work, the vectorization method of the point detector estimator has been studied and implemented into a vectorized continuous energy Monte Carlo code MVP\textsuperscript{1)} to extend the applicability of the code. The developed point detector estimator is available with all the geometry description capabilities of MVP except the reflection or periodic boundary condition.

A point detector estimator is a deterministic estimate of fluxes or responses at a point in space and evaluates contributions from every source and collision event during the random walk. This estimator consists of two steps. The first step is to generate pseudo particles toward the detector at source and collision points. The second is to calculate attenuation factors of particles passing through materials by tracking the particles without collision and tally the contribution to the detector. In order to increase the vectorization efficiency, the MVP code adopts the simulation method\textsuperscript{2)} similar to that used in a multi-group code GMVP, which is summarized as follows:

(1) One pseudo particle at most is produced for one real particle and one point detector in a pseudo particle generation event.

(2) The tracking procedure for pseudo particles is separated from that for real particles and treated as a child event of the source generation and collision events; that is, the tracking process of pseudo particles is started up from these events.

(3) The tracking process of pseudo particles is started when the number of queued pseudo particles exceeds a preset value, and it is interrupted if the number of pseudo particles becomes smaller than another preset value.

(4) The stack-driven zone-selection method\textsuperscript{1)}, which is a standard algorithm for real particles, is also applied to the tracking of pseudo particles.
In order to verify the implemented point detector estimator, we have analyzed the experiment on neutron and gamma-ray doses in critical accidents carried out at the TRACY facility\textsuperscript{3)}. Figure 2.4.1 shows the experimental arrangement and detector positions. Geometry model and calculation procedure are similar to those in Ref. 3). The calculation was performed as an eigenvalue problem of neutron and photon transport. The core tank, fuel solution and concrete walls were modeled, and vertical distribution of absorbed doses on the surface of the core tank and horizontal one in the reactor room were evaluated by using point detector estimators and track length estimators. The total numbers of histories and batches are 1 million and 100, respectively. The JENDL-3.2 library was used in the calculation. The kerma factors were taken from Ref. 3). Figure 2.4.2 shows the comparison between the calculated and measured results. The results of the point detector estimators show good agreement with those of the track length estimators. The calculated results are in agreement with the measured ones within 30\% for both gamma-ray absorbed doses by DLTB dosimeters and (neutron+gamma-ray) absorbed dose by alanine dosimeters.

References:
Fig. 2.4.1 Arrangement of dosimeters in the TRACY absorbed dose measurements on solution criticality accidents (taken from Ref.3))

Fig. 2.4.2 Comparison of absorbed doses between measurements and calculations by MVP
2.5 Determination of Large Perturbation Reactivity by Neutron Source Multiplication Method

T. Suzuki, S. Kohmoto*, K. Murakami, M. Kurosawa, S. Fujisaki and Y. Hoshi
(Tel: 029-282-5223, Fax: 029-282-5878)

Measurements of large reactivity effects of neutron absorber materials, void formation, temperature raise, etc. generally require re-construction of experimental cores to attain the critical state in both cases before and after the perturbation. This causes some difficulty to deduce the reactivity value of the perturbation, because the effect of re-construction must be corrected. Besides, an unexpected change in the arrangement of perturbation materials at the critical state has a possibility to result in a reactivity insertion trouble. For the purpose to avoid these problems, we are developing experimental methods to determine large perturbation reactivities at a subcritical state using the Tank-type Critical Assembly (TCA).

The neutron source multiplication method is the subject of our present attention for its simplicity and wide applications. It is required for this method that the subcriticality of the reference core before perturbation must be known correctly, and the change in counting efficiency of neutron detectors before and after perturbation is as small as possible. The former requirement is easily answered at tank-type reactors by controlling the core height with a sufficient accuracy to a considerably high subcritical state. The latter requirement is just the point of investigation, because the computational correction for the effect is desirable to be less than about 10%.

To find out a good arrangement of the neutron source and detector in the source multiplication method, a reference core was constructed by arraying 2.6%-enriched UO₂ rods into a 17x17 square core with the lattice pitch of 19.56 mm, and four cases of perturbations were studied where the central 1, 5, 9 and 25 UO₂ rods were replaced by ThO₂ rods as shown in Fig. 2.5.1. A⁹²⁵Cf source and a ³He detector were located with the right angle against the core center on the horizontal plane, and their positions in the vertical direction were selected to ride on the nodes of the first and the second harmonics, respectively, to reduce the effect of higher modes.

*Nagoya University
The reactivity effects of perturbation were obtained by
\[
\delta \rho = \rho - \rho_0 = \rho_0[(CR_\rho/CR)f-1]
\]
(2.5.1)
where \( \rho_0, \rho \): reactivities of reference and perturbed cores,

\( CR_\rho, CR \): count rates for reference and perturbed cores,

and the correction factor \( f \) for the change in counting efficiency was assumed to be unity. The \( \rho_0 \) was varied from about \(-0.2\) to \(-10\) dollars by lowering the water level from the critical level. The results of \( \delta \rho \) are listed in Table 2.5.1 with the reference values which were measured from the change in critical water level for a smaller \( \delta \rho \), or by the exponential experiment \(^\dagger\) for a larger \( \delta \rho \). As seen in the table, agreement between the results of the source multiplication and the reference methods is satisfactory. Therefore, the present arrangement method of neutron source and detector could be applied to the similar cores with a considerably large perturbation.

Reference:

<table>
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<th>9</th>
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<td>4.42</td>
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<td>10.09</td>
<td>0.38</td>
<td>2.22</td>
<td>4.61</td>
<td>12.1</td>
</tr>
</tbody>
</table>

Ref. 0.41 ± 0.03*  2.25 ± 0.15**  4.3 ± 0.2**  11.7 ± 0.3**

*Statistical error in neutron counting is less than 1%.

**From the change in critical water level.

**By the exponential experiment.

Fig. 2.5.1
Experimental arrangement

- 36 -
2.6 Space Dependence Correction for Rod Drop Experiments

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(Tel: 029-282-5223, Fax: 029-282-5878)

The rod-drop method is often used for measuring the shut-down margin of nuclear reactors, however, it is known that the measured reactivity worth of control/safety rods differs with the detector position and the detector response on neutron energy. Figure 2.6.1 shows the results of the integral counting method measured by using thermal neutron detectors for free drops of a boron safety-sheet in an early day of reactor experiments at the Tank-type Critical Assembly (TCA)\textsuperscript{11}. One can see strong space dependence of the measured reactivity worth varying about five times with the detector positions.

In order to investigate a correction method for the space and neutron spectrum dependence, a series of rod-drop experiments was performed with a similar core to the previous one as shown in Fig. 2.6.2. The safety sheet made of an Al-covered 0.7 mm-t Cd plate (total thickness: 2 mm) was dropped into the critical core of which water level was about 90 cm from the lower end of active fuel zone. It took 0.22 s for the head of the safety sheet to move from the water surface to the end position below the lower end of core by about 3 cm. The reactivity worth of the safety sheet $\rho_{\text{RD}}$ was obtained from the integral counting method based on the one-point kinetics model by using bare or 0.5 mm-t Cd covered $^3$He counters, which mainly detect thermal or epithermal neutrons, at the positions A, B, C and F shown in Fig. 2.6.2. The results are given in Table 2.6.1. The statistical error in neutron counting was less than 1%.

The correction factor $f$ for obtaining the true reactivity $\rho$ from the measured $\rho_{\text{RD}}$ was defined according to Ref. (2) under the one-dimensional and one-group approximations as

$$
 f = \frac{\rho}{\rho_{\text{RD}}} = \frac{\int_{-X}^{X} \phi o \phi \text{d}x}{\int_{-X}^{X} \phi^2 \text{d}x} \cdot \frac{\phi(XD)}{\phi(oXD)}, \tag{2.6.1}
$$

where $\phi_0, \phi$: neutron fluxes before and after the rod drop.

*Nagoya University, **Mitsubishi Heavy Industries, LTD.
X, xD: core half width and detector position in x axis perpendicular to the safety sheet.

The neutron fluxes were expressed as

\[ \phi_0 = \cos B_0 x, \quad B_0 = \pi / (2a), \quad a = X + \lambda, \quad \lambda: \text{extrapolation distance} (7.2 \text{ cm}), \quad (2.6.2) \]

\[ \phi = A \sin B (a + x) + C \exp \nu (x - x_0), \quad (x \leq x_0), \quad x_0: \text{position of safety sheet}, \]

\[ = E \sin B (a - x) + G \exp \nu (x_0 - x), \quad (x \geq x_0), \quad (2.6.3) \]

where squares of B0 and B are the bucklings of asymptotic distributions of \( \phi_0 \) and \( \phi \), and \( \nu \) is the inverse of relaxation length for the local distribution around the safety sheet. The constant A was determined so that the ratio of integral in Eq. (2.6.1) is unity, and E was obtained from the continuity of \( \phi \) at \( x = x_0 \). The amplitudes of local distributions C and G were given by assuming the black absorber condition at \( x = x_0 \) for thermal neutrons, but they were neglected for epithermal neutrons. To obtain the value of B, the reactivity worth of the safety sheet was measured by the exponential experiment\(^3\) as \(-4.4 \pm 0.1 \text{ dollars}, \) and the following relation was utilized;

\[ \rho = K (B_0^2 - B^2), \quad K: \text{buckling coefficient of reactivity} (4.288 \text{ $\text{\$ cm}^2$}). \quad (2.6.4) \]

The resultant values of \( f \) are shown in Table 2.6.1 with the corrected \( \rho \), and the variations of \( f \) are illustrated in Fig. 2.6.3. One can see in the table that the space and neutron spectrum dependence of reactivity worth almost disappears, and every \( \rho \)’s agree fairly well to that from the exponential experiment.

References:

<table>
<thead>
<tr>
<th>Table 2.6.1 Results of rod-drop experiments and corrections</th>
</tr>
</thead>
<tbody>
<tr>
<td>Det. pos.</td>
</tr>
<tr>
<td>-----------</td>
</tr>
<tr>
<td>A</td>
</tr>
<tr>
<td>B</td>
</tr>
<tr>
<td>C</td>
</tr>
<tr>
<td>F</td>
</tr>
</tbody>
</table>

*1: Bare detector, 2: Cd-covered one. **Values at the core edge near F.
Fig. 2.6.1 Space dependence in rod-drop experiments\textsuperscript{1)}

Detector position  Cd-Al safety sheet  2.8%UO\textsubscript{2} (19.56mm pitch)

Fig. 2.6.2 Experimental core and detector positions

(1) for thermal neutrons
(2) for epithermal neutrons

Fig. 2.6.3 Neutron flux distributions before and after rod-drop $\phi_0$, $\phi$
and correction factors $f$
2.7 A Concept of Helium Gas-Cooled Nitride-Fuel Fast Breeder Reactor

S. Iijima, S. Okajima, T. Yamane and T. Osugi
(E-mail: iijima@fca001.tokai.jaeri.go.jp)

The helium gas-cooled fast breeder reactor (GCFR)\(^1\) has an excellent breeding performance, because of the very small interaction between neutrons and the helium coolant. The high breeding ratio in the active core reduces the reactivity swing over the cycle. In addition to the breeding performance, the low coolant void reactivity brings some advantages in the core design. On the other hand, disadvantages of the GCFR include the high operating pressure of the coolant and the safety problems arising from the possibility of coolant depressurization accidents. A nitride fuel has the excellent thermal characteristics that are a potential in a core design of high performance fast reactor. A feasibility study of the nitride fuel GCFR has been made on the nuclear characteristics evaluation and on the severe accident analyses.

The main design parameters of 1000 MW electric power GCFR are shown in Table 2.7.1. The active core region with two enrichment zones consists of 271 hexagonal fuel assemblies. A diameter of active core is 3.3 m. The radial blanket is arranged around the active core. The axial blanket is also arranged above and below the active core. The active core height is 1.4 m. The (Pu, U)N fuel pellets of 5.35 mm in diameter are inserted into the ferritic steel cladding fuel pins. The maximum cladding temperature is 700 °C in a nominal power operation. A leakage of fission products from the fuel pellets is reduced over the cycle, because a temperature of nitride fuel becomes lower than that of MOX fuel in the same linear heat rate. The cooling system consists of the six loops and a helium gas pressure is ~9 MPa. The entire cooling system and the reactor core is housed within a prestressed concrete reactor vessel (PCRV).

It was shown in the thermal characteristics calculation that the fins increasing a cladding surface size of fuel pin by two times were necessary to keep a maximum cladding temperature under 700 °C in a nominal power operation. The transient calculations were made in the accidents for a coolant depressurization with a time constant ranging 100, 200 and 300 sec. In the severe accidents, the reactor shutdown by the control rod and the operation of alternative coolant system having 3 % capacity of the main circulator were necessary in order to keep the cladding temperature under the ferritic steel melting point (~1200 °C).
It was shown in this study that the excellent thermal characteristics of nitride fuel were useful on the core design of high performance gas-cooled fast breeder reactor. In the severe accident, the reactor shutdown by a control rod and an alternative coolant system were necessary to terminate the coolant depressurization accident without a failure of fuel pin.

Table 2.7.1 Main design parameters for 1000 MWe gas-cooled fast breeder reactor

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Thermal power</td>
<td>2600 MW</td>
</tr>
<tr>
<td>Electric power</td>
<td>1000 MW</td>
</tr>
<tr>
<td>Core: two enriched zone (inner / outer)</td>
<td>Pu enrichment (16.0 / 23.0 w%)</td>
</tr>
<tr>
<td>Core height</td>
<td>1.4 m</td>
</tr>
<tr>
<td>Core diameter</td>
<td>3.3 m</td>
</tr>
<tr>
<td>Fuel (closed pin type)</td>
<td>(Pu, U) N</td>
</tr>
<tr>
<td>Fuel pellet diameter</td>
<td>5.35 mm</td>
</tr>
<tr>
<td>Maximum linear heat rate</td>
<td>400 W / cm</td>
</tr>
<tr>
<td>Coolant (Pressure)</td>
<td>He (9 MPa)</td>
</tr>
<tr>
<td>Inlet and outlet temperature</td>
<td>250 °C / 550 °C</td>
</tr>
<tr>
<td>Breeding ratio</td>
<td>1.2</td>
</tr>
</tbody>
</table>

Reference:
2.8 Analyses of ULOF Transient and Depressurization Transient for Helium-Gas-Cooled Nitride-Fuel Fast Reactor

S. Okajima, T. Yamane, T. Osugi, S. Iijima, N. Nakano*, D. Sadahiro* and H. Ozaki*
(E-mail: okajima@fca001.tokai.jaeri.go.jp)

In a design study of helium-gas-cooled nitride-fuel fast reactor (He-GCFR), an accident analysis is important from a viewpoint of safety. Two typical accidents for He-GCFR, the unprotected loss-of-flow (ULOFS) transient and the depressurization transient in a case of a 10 cm diameter hole, were analyzed.

ULOFS Transient

In the ULOFS transient, a sequence of events was assumed as follows. All circular pumps for helium gas coolant stopped, the flow rate of the coolant decreased and the reactor shutdown was in failure. The VESPAR code was used in the analysis. It calculates the time dependent reactor power behavior by a one-point reactor kinetic equation and also calculates the temperature distributions in a fuel pin. Both negative feedback effects, Doppler effect and the fuel expansion effect, were considered in the calculation.

The calculation results are shown in Fig. 2.8.1. The fuel temperature increased as the flow rate decreased. Consequently, the reactor power decreased due to the negative feedback effect. The cladding temperature increased to the maximum point of 1,513 °C and then decreased to 1,454 °C. These temperatures were much higher than the allowable temperature of 900 °C.

To decrease the temperatures, we considered installing the self actuated shut down system (SASS) in the reactor. Fig. 2.8.2 shows the results of the ULOF analysis with the SASS. In this case, the cladding temperature increased up to 831 °C which is lower than the allowable temperature. It is clear that the SASS was effective to decrease the maximum temperature of the cladding and consequently the cladding and the fuel will be kept safe.

Depressurization Transient

The loss of helium gas pressure was induced by a hole that was accidentally made in pre-stressed concrete reactor vessel (PCRV) of the reactor. In this analysis, following specifications were adopted referring to the AGR: the area of the hole is 0.03 m², the time constant for depressurization is 100 sec and a reactor trip is actuated when the helium gas pressure dropped to 65% level of normal operation pressure. The cladding temperature was

* Fuji Electric Co. Ltd.
predicted as a parameter of the final containment pressure levels by analyzing the hottest fuel pin.

The calculation results in the case of 0.5 MPa of the containment backpressure are shown in Fig. 2.8.3. The cladding temperature increased up to 870 °C and suddenly decreased due to the drop of reactor power. Then it increased again to the second peak temperature of 760 °C in 520 sec after the commencement of the transient and gradually decreased to 670 °C. Figure 2.8.4 shows the maximum temperature of the cladding related to the containment backpressure in the PCRV. If the minimum containment backpressure of 0.4 MPa can be kept in the PCRV, the cladding temperature will not exceed the allowable temperature of 900 °C.

To realize the minimum containment backpressure of 0.4 MPa, the structure of the reactor containment was studied. The reactor vessel with 50m in diameter and 75m in height is sufficient to achieve the minimum containment backpressure of 0.4 MPa. The size of the vessel is the same as that of the present LWR vessel.

Conclusion

Both typical accidents, the ULOF transient and the depressurization transient, were analyzed to investigate the safety characteristics for the He-GCFR. In the ULOF transient, if the SASS was installed in the reactor, it was effective to decrease the maximum temperature of the cladding and consequently the cladding and the fuel will be kept safe. In the depressurization transient, when the containment backpressure is kept higher than 0.4 MPa, the cladding temperature will not exceed the allowable temperature of 900 °C. This pressure level can be realized in the same size of the reactor vessel as the present LWR.

Reference

Fig. 2.8.1 Calculation result of ULOF transient without SASS

Fig. 2.8.2 Calculation result of ULOF transient with SASS

Fig. 2.8.3 Calculation result of depressurization transient

Fig. 2.8.4 Maximum temperature of the cladding related to the containment backpressure in the PCRV
3. Advanced Reactor System Studies

The conceptual design study of Reduced-Moderation Water Reactor (RMWR) is being conducted. Design and investigation on four types of RMWR cores were performed. 1) The high conversion BWR type core: Further investigation on core characteristics under the situation of multiple recycling of plutonium have been made especially under the reprocessing scheme with relatively low decontamination factors for fission products and minor actinides. Result of the investigation shows the feasibility of the present core concept for multiple recycling. 2) The long operation cycle BWR type core: The characteristics of the flow reduction transient has been investigated using the precise model, in which void reactivity coefficient of void tube assembly is treated individually to the fuel assembly. Evaluation result shows that the void tube assembly has sufficient response characteristics to maintain safety of the core. 3) The BWR type core without blanket: For making the core configuration much closer to ABWR, control rod follower is removed in this design. 4) The PWR type core with light water coolant: A concept of seed-blanket fuel assembly type core has been investigated to improve the core performance. Other topics related to the RMWRs are as followings: Safety analyses for TCA critical experiments were performed. The safety was confirmed through the evaluation of reactivity limits. Also the size of the buffer region for eliminating the effect of UO₂ driver region is evaluated to assure the precise experiment in MOX fuel test region. The MOX fuel rod structural design and investigation for critical experiment have been performed and mock-up fuel rods were manufactured as samples. Sensitivity on channel and core reactivity stability for high conversion ratio BWR type core has been evaluated. The results showed similar tendency to the current BWR. Cooperation project for research and developments of RMWR with passive safety features have been performed under the funding from innovative and viable nuclear energy technology development project. Summarized description for the results in FY2000 is presented.

For the development of a multi-parameter measuring system using optical technique for primary coolant circuits in nuclear power plants, the irradiation effects on measurement system in high gamma-ray environment was evaluated by experiment. Neuro-expert monitoring system for nuclear power plant utilizing neural network and rule-based real-time expert system has been developed and superior performance over the conventional alarm system was demonstrated through the evaluation using plant simulator.
3.1 Conceptual Designing of Reduced-Moderation Water Reactor (1)

Study on High Conversion Ratio BWR Type Core —

T.Okubo, T.Iwamura, R.Takeda*1, K.Yamamoto*2 and H.Okada*3

(E-mail: okubo@hems.jaeri.go.jp)

A BWR type reduced-moderation water reactor (RMWR) aiming at the high conversion ratio of 1.1 has been studied in order to investigate the attainable upper limit for the conversion ratio. A concept has been established achieving the conversion ratio of 1.1 and the negative void reactivity coefficients 1). On this concept, further investigation has been performed as in the following. Namely, the core characteristics under the situation of multiple recycling of plutonium (Pu) have been investigated, especially under the advanced reprocessing scheme with relatively low decontamination factors (DFs) for fission products (FPs) and minor actinides (MAs). The core transition characteristics from the initial one to the equilibrium have been also studied. Furthermore, improvements of core performances such as the power output, the burn-up, the operation cycle length have been tried to be performed.

On multiple recycling of Pu, the effects of residual MAs and FPs in the MOX fuel on the core characteristics have been investigated 2) for some cases, which are based on the advanced reprocessing schemes with low DFs proposed for the fast breeder reactor fuel cycle. The effects of FPs in the MOX fuel have been evaluated to request more fissile Pu in the core but not to change the other core characteristics for both assumed cases of the advanced PUREX scheme of the PUREX+SETFICS+IE and the dry reprocessing scheme of the RIIAR. On the effects of MAs in MOX fuel, their effects on the void reactivity coefficient have been found to be large. That is, this makes the void reactivity coefficient become positive, and hence, must be compensated by changing the core design with reduced upper blanket length. This, in turn, results in reduction of the conversion ratio. As a multiple recycling situation, a case assuming the dry reprocessing scheme with DFs around 10 in average for FPs and 1 for MAs was adopted, and the equilibrium multiple recycling condition was evaluated. In this

*1 Hitachi, Ltd., *2 The Japan Atomic Power Company, *3 Tokyo Electric Power Company
condition, the amounts of MAs and FPs in the MOX fuel were evaluated to be 1.8 and 1.0 wt%, respectively. The TRU composition is presented in Table 3.1.1. In addition to the increase in MAs, $^{240}$Pu is also increased. They tend to make the void reactivity coefficient positive. Therefore, the core design must be modified to increase MOX length to keep the criticality. Also, especially the upper blanket was to be significantly reduced to keep the void reactivity coefficient negative. The reduction of the upper blanket, however, reduces the conversion ratio as shown in Table 3.1.2. Although the conversion ratio becomes lower for the multiple recycling of Pu under the advanced reprocessing scheme with relatively low DFs, the present core concept has been evaluated to be still feasible enough as a RMWR.

On the improvements of core performances such as the power output, the burn-up, the operation cycle length, some modification of the core design have been conducted. Since the core pressure drop for the present design is about 1/6 of that in an current ABWR, the natural circulation cooling is expected to be possible. Therefore, the core area can be extended by about 10 % even in the same size of the pressure vessel, by removing the internal pumps adopted in the previous design. Also, the rod diameter was reduced by about 6 % with keeping the linear power density. Through these modifications, the power output of 1,356 MWe can be achieved. To increase the burn-up up to 60 GWD/t from 45 GWD/t, the length of MOX parts was increased by about 4 %, and the upper blanket was shortened to make sure the void reactivity coefficient be negative at the increased burn-up. For the extended operation cycle to 24 months from 14 months, the number of refueling batch was reduced to 3.4 from 4.5. As a result, the fissile Pu conversion ratio was slightly reduced to 1.05 from 1.06, resulting from the shortened upper blanket design. The safety during the major transients and accidents has also been confirmed for this design.

References:
Table 3.1.1 TRU composition for multiple recycling case

<table>
<thead>
<tr>
<th>Item</th>
<th>Recycling case</th>
<th>Standard case</th>
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</thead>
<tbody>
<tr>
<td>Origin core</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Discharge burn-up (GWd/t)</td>
<td>45</td>
<td>45</td>
</tr>
<tr>
<td>TRU Composition (wt%)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>$^{237}$Np</td>
<td>0.5</td>
<td>0.0</td>
</tr>
<tr>
<td>$^{238}$Pu</td>
<td>2.4</td>
<td>2.7</td>
</tr>
<tr>
<td>$^{239}$Pu</td>
<td>50.6</td>
<td>47.9</td>
</tr>
<tr>
<td>$^{240}$Pu</td>
<td>34.0</td>
<td>30.3</td>
</tr>
<tr>
<td>$^{241}$Pu</td>
<td>4.1</td>
<td>9.6</td>
</tr>
<tr>
<td>$^{242}$Pu</td>
<td>3.2</td>
<td>8.5</td>
</tr>
<tr>
<td>$^{241}$Am</td>
<td>3.6</td>
<td>1.0</td>
</tr>
<tr>
<td>$^{242}$Am</td>
<td>0.1</td>
<td>0.0</td>
</tr>
<tr>
<td>$^{243}$Am</td>
<td>0.9</td>
<td>0.0</td>
</tr>
<tr>
<td>$^{244}$Cm</td>
<td>0.5</td>
<td>0.0</td>
</tr>
<tr>
<td>$^{245}$Cm</td>
<td>0.1</td>
<td>0.0</td>
</tr>
<tr>
<td>Total</td>
<td>100.0</td>
<td>100.0</td>
</tr>
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</table>

Table 3.1.2 Major core dimensions and characteristics for multiple recycling case

<table>
<thead>
<tr>
<th>Item</th>
<th>Recycling case</th>
<th>Standard case</th>
</tr>
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<tbody>
<tr>
<td>Electric power output (MWe)</td>
<td>1,100</td>
<td>1,100</td>
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<tr>
<td>Discharge burn-up for core part (GWd/t)</td>
<td>45</td>
<td>45</td>
</tr>
<tr>
<td>Core height (m)</td>
<td>0.84</td>
<td>0.68</td>
</tr>
<tr>
<td>Core mass flow rate (10^4 t/h)</td>
<td>1.4</td>
<td>1.3</td>
</tr>
<tr>
<td>Core exit quality (%)</td>
<td>51</td>
<td>55</td>
</tr>
<tr>
<td>Core average void fraction (%)</td>
<td>68</td>
<td>70</td>
</tr>
<tr>
<td>Core average fissile Pu content (%)</td>
<td>9.4</td>
<td>10.2</td>
</tr>
<tr>
<td>Loaded fissile Pu (t)</td>
<td>13.7</td>
<td>12.1</td>
</tr>
<tr>
<td>Fissile Pu conversion ratio (%)</td>
<td>1.02</td>
<td>1.06</td>
</tr>
<tr>
<td>Maximum linear power density (kW/m)</td>
<td>52.5</td>
<td>55.8</td>
</tr>
<tr>
<td>MCPR</td>
<td>1.3</td>
<td>1.3</td>
</tr>
<tr>
<td>Void reactivity coefficient (10^4 Δ k/k/% void)</td>
<td>-0.5</td>
<td>-1</td>
</tr>
<tr>
<td>Operation cycle length (EFPM)</td>
<td>14</td>
<td>14</td>
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<tr>
<td>Core axial fissile Pu enrichment distribution</td>
<td></td>
<td></td>
</tr>
<tr>
<td>wt%</td>
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</tr>
<tr>
<td>mm</td>
<td>DU 130</td>
<td>DU 330</td>
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<td>18</td>
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<td>18</td>
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<tr>
<td>mm</td>
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<td>18</td>
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<td>18</td>
</tr>
<tr>
<td>mm</td>
<td>DU 150</td>
<td>DU 200</td>
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<tr>
<td>Amount of MA/FP in MOX (wt%)</td>
<td>1.8 / 1.0</td>
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</tr>
</tbody>
</table>
3.2 Conceptual Designing of Reduced-Moderation Water Reactor (2)

- Study on Long Operation Cycle BWR Type Core -

T.Okubo, T.Shirakawa, T.Iwamura, T.Yokoyama*1, K.Yamamoto*2 and H.Okada*3
(E-mail: okubo@hems.jaeri.go.jp)

A BWR type reduced-moderation water reactor (RMWR) has been studied, aiming at the long operation cycle more than two years and the high burn-up around 60 GWD/t, under the high conversion ratio more than 1.0. The reactor with such a long operation cycle and high burn-up is mainly expected to be beneficial in reduction of radioactive waste production as well as the economical aspects. In the previous study 1), a basic core design concept has been established achieving the design goals mentioned above under the negative void reactivity coefficient, by newly introducing a special assembly named the void tube assembly, which makes the total core void reactivity coefficient negative canceling the positive value in the fuel assembly part. On this concept, further investigation has been performed as in the following. Namely, the characteristics during the flow reduction transient have been investigated more precisely than before, taking individual void reactivity coefficient separately into account for the fuel assembly and for the void tube assembly. The core transition characteristics from the initial loading to the equilibrium have been also studied. Furthermore, an improvement of the core performance in the conversion ratio has been conducted.

On the flow reduction transient, this core has been considered to have a unique transient response. That is, since the response time for void formation in the void tube assembly is longer than in the fuel assembly, the void reactivity coefficient is expected to be positive during the initial period of the flow reduction transient. With some delay, the void tube assemblies start to give the negative void reactivity coefficient. Therefore, the precise quantitative evaluation on this transient has been considered to be necessary. In the present evaluation, the void reactivity coefficient for the void tube assembly is treated independently from that for the fuel assembly, in order to present the transient behavior described above.

*1 Toshiba, Co., *2 The Japan Atomic Power Company, *3 Tokyo Electric Power Company
The evaluated results are presented in Figs. 3.2.1 and 3.2.2 for the transient, in which 30 % of the recirculation pumps are assumed to be tripped. The core power and flow rate are assumed to be at 102 % and 111 % of the rated values, respectively. These values give the severest results, based on some parameter survey on them.

The major parameters are presented in Fig. 3.2.1 during the initial period. Due to the flow reduction, values for the neutron flux and the average rod surface heat flux increase, and hence, $\Delta \text{MCPR}$ (Minimum Critical Power Ratio) starts to decrease. MCPR is evaluated with the Arai's correlation $^2$. The neutron flux, however, starts to decrease in one or two seconds. Therefore, $\Delta \text{MCPR}$ starts to recover correspondingly to it. The maximum value of $\Delta \text{MCPR}$ is 0.12 as shown in the figure.

Reactivity changes are presented in Fig. 3.2.2. During the initial one or two seconds, the void and the net reactivities are positive and the Doppler reactivity is slightly negative. After two seconds, however, they become negative and do not become positive after that. This means that the response in the void tube assemblies during the flow reduction transient is much faster than expected and fast enough to make the void reactivity coefficient negative in one or two seconds. Due to this reactivity response, the neutron flux starts to decrease in one or two seconds as mentioned above and shown in Fig. 3.2.1.

Based on this evaluation, the response of the void tube assembly during the flow reduction transient is found to be not too slow to make $\Delta \text{MCPR}$ a large value. The maximum value of $\Delta \text{MCPR}$ is 0.12 and this shows the flow reduction transient is not serious in this core from the safety point of view.

References:


Fig. 3.2.1  Major parameters for transient behavior

Fig. 3.2.2  Reactivity data for Doppler, void and net values during transient
3.3 Conceptual Designing of Reduced-Moderation Water Reactor (3)
— Study on BWR Type Core without Blanket —

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Our design object is a simple core without blanket with conversion ratio more than 1.0 and negative void reactivity coefficient, keeping current ABWR square fuel bundle configuration. In the previous study\(^1\), the core design with conversion ratio more than 1.0 and fuel burn-up of 25.6GWD/t has been proposed. In the present study, we try to make the core configuration much closer to ABWR by removing control rod follower.

In this design, basic fuel bundle configurations are the same as the previous design; BWR type 9x9 square lattice bundle, introducing large diameter fuel pin and axial distribution of PuO\(_2\) enrichments to reduce axial power peaking. In the low PuO\(_2\) enrichment region \(k_\infty\) is less than 1.0 and the void reactivity coefficient has negative value. In the present study, the differences of the design are as follows; we have removed follower on control rods to simplify the core configuration. This brings such penalty that local peaking factor becomes higher at control rod water gap side fuel pin. Therefore, we decreased PuO\(_2\) enrichment at this pin. Besides, we decreased core average linear power by raising core height from 2.6m to 2.7m. We have thinned cladding thickness by removing liner to raise conversion ratio. Liner protects cladding from thermal stress due to power change. Liner is needless to dull power change for Reduced-Moderation.

The fuel bundle geometry of the core is shown in Fig. 3.3.1. The PuO\(_2\) enrichment distribution is shown in Fig. 3.3.2.

Table 3.3.1 gives major dimensions and characteristics of the core. Figure 3.3.3 shows the calculation results of the effective multiplication factor \(k_{eff}\) through the equilibrium cycle. The void reactivity coefficient at the beginning of cycle is negative and almost equal to zero at the end of cycle. The maximum linear power density is 47.8kW/m at the end of cycle with all control rods withdrawn. This value is lower than for BWR/4 7x7 fuel bundle. The linear power density is rather high without control rods at the beginning of the cycle. But, it is able to be decreased by control rod.

Neutronics calculations for 2-dimensional XY fuel assembly lattice and 3-dimensional XYZ core burn-up were performed by using JAERI's general purpose neutronics code system SRAC95\(^2\).

Although possibility of core design without the control rod follower has been demonstrated through this study, the conversion ratio and the burn-up have been calculated to
become slightly lower than in the previous study. We’ll study to raise burn-up.

Reference:

Table 3.3.1 Major dimensions and characteristics of the core

<table>
<thead>
<tr>
<th>Item</th>
<th>Previous study</th>
<th>Present study</th>
</tr>
</thead>
<tbody>
<tr>
<td>Thermal power output (MWt)</td>
<td>3,926</td>
<td>←</td>
</tr>
<tr>
<td>Electric power output (MWe)</td>
<td>1,356</td>
<td>←</td>
</tr>
<tr>
<td>Fuel lattice arrangement</td>
<td>9 x 9 square</td>
<td>←</td>
</tr>
<tr>
<td>Fuel rod length (m)</td>
<td>2.6</td>
<td>2.7</td>
</tr>
<tr>
<td>Fuel assembly pitch (mm)</td>
<td>155</td>
<td>←</td>
</tr>
<tr>
<td>Gap width between rods (mm)</td>
<td>1.0</td>
<td>←</td>
</tr>
<tr>
<td>Fuel outer diameter (mm)</td>
<td>15.2</td>
<td>←</td>
</tr>
<tr>
<td>Fuel cladding thickness (mm)</td>
<td>0.75</td>
<td>0.5</td>
</tr>
<tr>
<td>Average fissile Pu content (wt%)</td>
<td>6.9</td>
<td>6.0</td>
</tr>
<tr>
<td>Number of refueling batch</td>
<td>4.27</td>
<td>4.27</td>
</tr>
<tr>
<td>Cycle burn-up (GWd/t)</td>
<td>6</td>
<td>5.5</td>
</tr>
<tr>
<td>Operation cycle length (EFPM)</td>
<td>12.6</td>
<td>13.3</td>
</tr>
<tr>
<td>Discharge burn-up (GWd/t)</td>
<td>25.6</td>
<td>23.1</td>
</tr>
<tr>
<td>Conversion ratio</td>
<td>1.03</td>
<td>1.0</td>
</tr>
<tr>
<td>Void reactivity coefficient</td>
<td>-1.4</td>
<td>-3.0</td>
</tr>
<tr>
<td>(10^{-6}d/k/△%void)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Maximum linear power density at EOC* (kW/m)</td>
<td>48.4</td>
<td>47.8</td>
</tr>
</tbody>
</table>

* EOC: End of cycle
Fig. 3.3.1  Cross section of fuel rod assembly

Fig. 3.3.2  Fuel bundle PuO₂ enrichment

Fig. 3.3.3  Calculated $k_{\text{eff}}$ for equilibrium cycle
3.4 Conceptual Designing of Reduced-Moderation Water Reactor (4)
-- Study on PWR Type Core with Light Water Coolant --

S. Shimada*1, T. Okubo, T. Iwamura, K. Hibi*2, K. Yamamoto*3 and H. Okada*4
(E-mail: shimada@popsrvr.tokai.jaeri.go.jp)

As a light water cooled reduced-moderation PWR type core, a concept with seed-blanket fuel assemblies has been investigated 1). As shown in Fig.3.4.1, seed fuel pins are surrounded by blanket fuels. The neutronics characteristics of fuel assembly was investigated by the continuous-energy Monte Carlo codes MVP 2) and MVP-BURN 3) as design tools. Continuous work 4) to improve the core characteristics has been performed mainly using by the SRAC 5) code, and MVP-BURN was used for confirmation of accuracy of the SRAC calculation.

The major aims of the design are,

1) Conversion ratio is more than 1.0
2) Void reactivity coefficient must be negative

Based on SRAC, the assembly shown in Fig.3.4.1 was used in the calculation. The axial configuration of fuel assembly is shown in Fig.3.4.2.

The main parameters are shown in Table 3.4.1 for the core design by SRAC. The effective multiplication factor and conversion ratio at the equilibrium cycle are shown in Fig.3.4.3. The above two aims of the design are satisfied under 3-batch refueling scheme. The control rod pattern shown in Fig.3.4.4 was determined by taking account of shut-down margin and reactor operation. In reactor operation the core power distribution must be flat to satisfy the design power peaking limits, so that chemical boron was used to control the power swing. The control rods were divided into 3 groups for reactor startup as shown in Fig.3.4.4. Using the control rods with the chemical boron for reactor operation, the power capability was confirmed as shown in Table 3.4.1.
References


<table>
<thead>
<tr>
<th>Item</th>
<th>Core based on SRAC</th>
</tr>
</thead>
<tbody>
<tr>
<td>Cycle Length</td>
<td>24EFPM ×3</td>
</tr>
<tr>
<td>Assembly Specification</td>
<td>Pu-238/239/240/241/242/Am-241</td>
</tr>
<tr>
<td>Pu composition</td>
<td>2.7/47.9/30.3/9.6/8.5/1.0wt%</td>
</tr>
<tr>
<td>Fuel or Rod Outer Dia.</td>
<td>12.0mm</td>
</tr>
<tr>
<td>Fuel or Rod Gap width</td>
<td>1.0mm</td>
</tr>
<tr>
<td>Pu f. Enrichment</td>
<td>17.0wt%/19.8wt%</td>
</tr>
<tr>
<td>No. of pin of seed / blanket</td>
<td>427/786</td>
</tr>
<tr>
<td>Core Specification</td>
<td>468mm</td>
</tr>
<tr>
<td>Assembly Pitch</td>
<td>6.28m</td>
</tr>
<tr>
<td>Effective Core Dia.</td>
<td></td>
</tr>
<tr>
<td>Conversion Ratio</td>
<td>1.05</td>
</tr>
<tr>
<td>Max. void reactivity coeff.</td>
<td>-1.7% Δ ρ /total void</td>
</tr>
<tr>
<td>Max. Linear Heat Rate</td>
<td>31kw/m</td>
</tr>
<tr>
<td>Discharge Burn-up</td>
<td>16 GWd/t</td>
</tr>
<tr>
<td>Puf. Loading/Discharge</td>
<td>2.88ton/2.99ton</td>
</tr>
</tbody>
</table>
Fig.3.4.1 Configuration of seed-blanket assembly chosen for SRAC

Fig.3.4.2 Axial configuration

Fig.3.4.3 Effective multiplication factor and conversion ratio with burn-up for equilibrium cycle

Fig.3.4.4 Control rod pattern
3.5 Study on Safety Analyses for TCA Critical Experiments

S. Shimada*1, T. Okubo, T. Iwamura, K. Yamamoto*2 and H. Okada*3

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To verify the feasibility of the Reduced-Moderation Water Reactor (RMWR), critical experiments using TCA (Tank Type Critical Assembly) at JAERI have been investigated.1,2) The fuel specification was determined considering current design for BWR and PWR type RMWR2). The core was divided into two regions by inserting an inner tank to avoid criticality only in MOX fuel region 2). To determine the number of fuel rods and the size of the inner tank, preliminary calculations were performed by using the SRAC code 3). The number of fuel rods for critical experiments was temporally determined.2)

Taking account of these previous calculations, more realistic calculations for experiments have been performed. The test region is composed with MOX fuel region and surrounding buffer region in the inner tank, and driver fuels are loaded in the outer region. The criticality of the test region is controlled only by changing the water level of outer region. The number of fuel rods and the size of the inner tank were determined by the present calculations. Safety parameters for licensing shown in Table 3.5.1 were calculated and then it is confirmed that the experiments would be performed safely by dropping the water in the driver region and/or safety plates in the driver region. Criticality and control rod worth were calculated for the many configurations of cylindrical and rectangular core by SRAC. Void reactivity, reactivity with water level, boron reactivity, moderator temperature coefficients and neutron flux distribution were calculated for the critical configurations. Calculations using five safety plates with one rod stuck were performed for cylindrical core by MVP 4) and rectangle core by SRAC, and confirmed sub-criticality. The reactivity change with the water level of the driver region for cylindrical core shown in Fig.3.5.1, effective multiplication factor$(k_{\text{eff}})$ vs. enriched boron concentration shown in Fig.3.5.2 and temperature coefficients shown in Fig.3.5.3 are typical results and useful for performing the experiments. The distribution of fast to thermal flux ratio is shown in Fig.3.5.4, which shows the buffer of 10cm is enough to remove the effect of driver region. Kinetic parameters and

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*3 Tokyo Electric Power Company
reaction rates were also calculated for the critical cores.

The conclusions obtained from this study are

(1) Revised numbers of fuel rods are proposed as shown in Table 3.5.2.
(2) Diameter of inner tank is 60cm. Size of square tank should be 100cmx40cm.
(3) UO$_2$ region of 10cm is enough as the buffer.
(4) The sub-criticality of test region is assured by draining the water in driver region.

The rate of water level increase must be less than 0.1cm/s.
The speed of control rod withdrawal must be less than 0.1cm/s.
Five safety plates are enough to make reactor sub-critical of 1.0%k/k with one rod stuck.

References

<table>
<thead>
<tr>
<th>Maximum excess reactivity</th>
<th>MOX Core</th>
<th>$&lt;0.4% \Delta k/k$</th>
</tr>
</thead>
<tbody>
<tr>
<td>Maximum possible excess reactivity</td>
<td>MOX Core</td>
<td>$&lt;0.4% \Delta k/k$</td>
</tr>
<tr>
<td>Reactivity inserted by the material (absorber, void etc.)</td>
<td>$&lt;5% \Delta k/k$</td>
<td></td>
</tr>
</tbody>
</table>

**Reactivity worth**

- Reactivity Control Worth: Reactivity should be negligible small at the complete void condition
- Reactivity insertion rate: $<2 \times 10^{-2} \% \Delta k/k/sec$ when core is near critical ($>0.99$)

**Safety plate**

- Reactivity control worth: Effective multiplication factor should be less than 0.99 when safety rods inserted at the condition of one rod stuck.
- Reactivity insertion rate: Not available

**Control rods used in the experiments**

- Control rod worth: $<0.4\% \Delta k/k$
- Reactivity insertion rate: $<2 \times 10^{-2} \% \Delta k/k/sec$
Table 3.5.2 Proposed number of fuel rods MOX critical experiments

<table>
<thead>
<tr>
<th>MOX fuels</th>
<th>5</th>
<th>10</th>
<th>15</th>
<th>20</th>
</tr>
</thead>
<tbody>
<tr>
<td>Pu' enrichment (wt%)</td>
<td>0</td>
<td>1300</td>
<td>1700</td>
<td>500</td>
</tr>
<tr>
<td>Number of rods</td>
<td>3000</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Depleted UO2 fuels for Blanket/Buffer region</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Number of rods</td>
<td>3000</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Driver fuels (4.9wt% UO2)</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Number of rods</td>
<td>3000</td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

Fig. 3.5.1 Reactivity increase with water level in the driver region (Cylindrical core without blanket)

Fig. 3.5.2 keff as a function of 90% enriched boron concentration (Rectangular core without blanket, control rods out)

Fig. 3.5.3 keff with core temperature(Cylindrical core without blanket)

Fig. 3.5.4 Fast to thermal neutron flux ratio ($\phi_f / \phi_t$) (Rectangular core, Pu-fis 15wt%, length of MOX = 80cm, without blanket)
3.6 MOX Fuel Rod Structural Design for RMWR Critical Experiment using TCA

S. Fujiwara, T. Okubo, T. Iwamura, K. Yamamoto*1 and H. Okada*2

(E-mail: fsetsuo@popsvr.tokai.jaeri.go.jp)

To verify the feasibility of the Reduced-Moderation Water Reactor (RMWR), critical experiments using TCA (Tank Type Critical Assembly) at JAERI have been planned and investigated1). The fuel specification was determined considering current design for the RMWRs. The core of the critical experiments using TCA is divided into two regions by inserting an inner tank to avoid the criticality only in the MOX fuel region itself. The test region is composed with the MOX fuel region and surrounding buffer region in the inner tank, and driver fuels are located in the outer region. Only by changing the water level of the outer region, the criticality is controlled.

In this study, the MOX fuel rod structural design of the critical experiments has been investigated, and 6 of the mock-up MOX fuel rod were manufactured as samples. One of the 6 rods is the cut model to see the inner structure of the rod. The structural design of the MOX fuel rod at the critical experiments using TCA is shown in Fig.3.6.1.

![Diagram of MOX fuel rod structure](image)

**Fig.3.6.1** The basic structural design of TCA MOX fuel rod

The unique characteristic points of the fuel rods at the critical experiments using TCA are as follows.

1) In the critical experiments using TCA, one by one handling of fuel rods is needed to change core configuration. So there are no fuel assemblies.

2) In the critical experiments, heat release of the fuel rods is negligibly small. So it is not

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necessary to consider the heat loads to the design.

(3) In the critical experiments, the fuel rods are put in normal room environment. Therefore the environment with high temperature or high pressure is not necessary to be considered.

(4) In the critical experiments, the fuel rods are not practically burned up. So there is no release of Fission Products (FP) gas.

In consideration of the point (1), the fuel rods in TCA are supported directly from the support plate in the inner tank, and it is basically necessary to avoid putting support materials in core region to keep the high accuracy of the critical experiment. So the axial support structure (upper and lower support) shown in Fig.3.6.2 is selected.

In the critical experiments of the RMWR, tight lattice arrangement of the fuel rods (rod pitch \( \leq 13\text{mm} \)) is needed. Therefore, the porous type upper and lower support plates are designed. Since, in general, the strength of the porous plates would be reduced by the holes, the design shown in the right hand side of Fig.3.6.2 is selected for the connection by the small hexagonal plates.

At the beginning of the tight lattice arrangement of the fuel rods, the fuel rods are inserted to the porous lower support plates. In the next process, as shown in the right hand side of Fig.3.6.2, a set of the 7 fuel rods is tied up with the small hexagonal plate using 7 holes. The neighbor small hexagonal plates are connected each other. By using this method, the one by one fuel rod handling to change core configuration of the critical experiments is possible.

In consideration of the points (2) and (3), the initial pressurization of the fuel rods by inner gas is not needed. Atmospheric pressure inert gas (He or Ar) is put in the fuel rods only to keep the MOX fuel pellets in inert gas environment.

In consideration of the point (4), there is no need to design the gas plenum for FP gas in the fuel rods for the experiments using TCA. But the space only for the buffer material is
needed to keep the axial stability of the fuel stack. Aluminum (Al) that have small cross-section of neutron absorption is selected for the buffer material of fuel rods to keep the high accuracy of the critical experiment. For the buffer material, considering the deterioration of the buffering function during transportation, the pellet type of Al buffer is selected, because the woolen type or mesh type of Al is not appropriate. And to keep the high accuracy of the critical experiment, steel spring is not adopted for the large neutron absorption cross-section.

In this study, the MOX fuel rod strength has been investigated, and the limitation and the guideline for the protection from the damage of the fuel rod has been obtained\(^2\) (see Table 3.6.1.). The fuel rod heat release is negligibly small and the fuel rods are under the ambient condition during the fuel rod handling and during the critical experiments. Therefore the expected loads to the fuel rods are limited to the mechanical loads. Since one by one fuel rod handling is used in the critical experiments, the possible loads to the fuel rods are limited to the direct loads to the rods.

<table>
<thead>
<tr>
<th>Load mode</th>
<th>Assumed abnormal condition during fuel handling</th>
<th>Destruction mode of fuel cladding</th>
<th>Limitation item</th>
<th>Limitation value</th>
<th>Decision criteria of limitation value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Bending</td>
<td>Bending during fuel handling</td>
<td>Destruction of fuel cladding</td>
<td>Displacement</td>
<td>378 mm</td>
<td>bending force = destruction force of Zr fuel cladding</td>
</tr>
<tr>
<td>Twist</td>
<td>Twist during fuel handling</td>
<td>Shearing destruction of fuel cladding</td>
<td>Twist angle</td>
<td>65°</td>
<td>shearing force = destruction force of Zr fuel cladding x 1/2</td>
</tr>
<tr>
<td>Axial compression</td>
<td>Abnormal axial tension during fuel</td>
<td>Bend and break</td>
<td>Axial compression force</td>
<td>19.3 kgf</td>
<td>axial compression force = bending load by compression</td>
</tr>
<tr>
<td>Axial tension</td>
<td>Abnormal axial tension during fuel handling</td>
<td>Ductility destruction of fuel cladding</td>
<td>Axial tension force</td>
<td>900 kgf</td>
<td>axial tension force = destruction force of Zr fuel cladding</td>
</tr>
<tr>
<td>Local compression</td>
<td>Collision with other object during fuel</td>
<td>Destruction by local compression</td>
<td>Local compression force</td>
<td>4.6 kgf</td>
<td>local compression force = destruction force of Zr fuel cladding</td>
</tr>
<tr>
<td>Impact of fuel drop accident</td>
<td>Rod drop accident during fuel handling</td>
<td>Destruction of fuel cladding</td>
<td>Height of fuel drop</td>
<td>2.3 m</td>
<td>impact force = destruction force of Zr fuel cladding</td>
</tr>
</tbody>
</table>

References


3.7 Sensitivity Analyses on Channel and Core Stability for High Conversion Ratio
BWR Type RMWR Core

N. Ishikawa, F. Araya and T. Iwamura
(E-mail: ishikawa@clsu3a0.tokai.jaeri.go.jp)

The BWR type Reduced-Moderation Water Reactor (RMWR) has specific features such as tight lattice core, and operating at low flow rate and high void fraction condition to improve the conversion ratio. To investigate the stability characteristics of RMWR, sensitivity analyses on channel and core reactivity stability were performed for major parameters (e.g., power, flow rate) utilizing the transient analysis code RETRAN02\(^1\). Major specification of the high conversion ratio BWR type RMWR\(^2\) is listed in Table 3.7.1 together with ABWR and axial power distribution of the core is depicted in Fig. 3.7.1.

Channel stability concerns the variation of flow rate of fuel bundle induced by disturbance. Since the BWR core consists of hundreds of fuel bundles, it is thought that the pressure drop across the core remains constant even though the flow rate perturbation is introduced into one channel. The thermal-hydraulic feedback mechanism acts on the channel flow such that the pressure drop across the core remains constant in spite of the flow rate perturbation. On the other hand, the core reactivity stability may be induced by the coupled response of the core thermal-hydraulics and the neutron kinetics. The primary effect is due to the core void reactivity feedback effect on core power. Both stabilities depend on the magnitude of feedback gain and phase lag of the feedback loop. If the feedback gain and/or phase lag becomes large, the stability is degraded.

As for sensitivity analyses on channel stability, channel power, flow rate and inlet orifice loss coefficient were chosen for major parameters. A channel flow rate perturbation is introduced by adding the pressure disturbance at lower plenum. The amplitude of the disturbance is 5% of the core pressure drop. The response of the channel flow rate is shown in Fig.3.7.2 for the power levels of 100% (nominal), 110% and 120%, respectively. The stability is measured by decay ratio (DR), which is defined as the ratio of the magnitude of the second overshoot to the first overshoot resulting from a step perturbation. The decay ratio is expressed in terms of damping factor $\zeta$ as
The damping factor $\zeta$ is obtained by fitting the response to a transfer function. The results of the sensitivity analyses on the channel stability are tabulated in Table 3.7.2. The results show a similar tendency to the current BWR.

For the sensitivity analyses on the core reactivity stability, the void reactivity coefficient and inlet orifice loss coefficient are as parameters. Figure 3.7.3 shows responses of the core power with respect to the void reactivity coefficients when the reactivity disturbance of 10[cent] was added. The void reactivity coefficients are set to 5 times and 1/5 times larger than the nominal one. The responses settle to stationary value faster as the absolute values of void reactivity coefficients become larger. However, no oscillation is observed in the present analyses. For inlet orifice loss coefficient, the response is almost same as the nominal one. It is confirmed that the core reactivity stability is maintained satisfactorily within the results of the sensitivity analyses.

References


<table>
<thead>
<tr>
<th>Table 3.7.1 Major specifications of high conversion ratio BWR type RMWR in comparison with ABWR.</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Items</strong></td>
</tr>
<tr>
<td>Electric power output [MWe]</td>
</tr>
<tr>
<td>Reactor pressure [MPa]</td>
</tr>
<tr>
<td>No. of fuel bundles</td>
</tr>
<tr>
<td>Core height [m]</td>
</tr>
<tr>
<td>Core mass flow rate [kg/sec]</td>
</tr>
<tr>
<td>Core exit quality [%]</td>
</tr>
<tr>
<td>Core average void fraction [%]</td>
</tr>
<tr>
<td>Fissile Pu conversion ratio</td>
</tr>
<tr>
<td>Void reactivity coefficient $[10^4 \Delta k/k%\text{void}]$</td>
</tr>
</tbody>
</table>
Fig. 3.7.1 Axial power distribution.

Table 3.7.2 Results of the sensitivity analyses on channel stability.

<table>
<thead>
<tr>
<th>Parameters</th>
<th>Condition</th>
<th>Decay Ratio</th>
</tr>
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<tbody>
<tr>
<td>Nominal case</td>
<td></td>
<td>0.003</td>
</tr>
<tr>
<td>Channel power</td>
<td>110%</td>
<td>0.070</td>
</tr>
<tr>
<td></td>
<td>120%</td>
<td>0.081</td>
</tr>
<tr>
<td>Channel flow rate</td>
<td>90%</td>
<td>0.054</td>
</tr>
<tr>
<td></td>
<td>80%</td>
<td>0.061</td>
</tr>
<tr>
<td>Inlet orifice loss coefficient</td>
<td>0.5 times</td>
<td>0.010</td>
</tr>
<tr>
<td></td>
<td>0.3 times</td>
<td>0.028</td>
</tr>
</tbody>
</table>

Fig.3.7.2 Channel flow response.  
Fig.3.7.3 Core power response.
3.8  Research and Development on Reduced-Moderation Light Water Reactor with Passive Safety Features


(E-mail: iwamura@hems.jaeri.go.jp)

In order to ensure sustainable energy supply in Japan, research and developments of reduced-moderation light water reactor (RMWR) with passive safety features have been performed by Japan Atomic Energy Research Institute (JAERI) in collaboration with The Japan Atomic Power Company (JAPC), Hitachi Ltd. and Tokyo Institute of Technology (TTI) under the funding from innovative and viable nuclear energy technology (IVNET) development project. The collaboration project was initiated in FY2000 and will be completed in FY2002. In FY2000, following results were obtained.

The reactor core consists of MOX fuel assemblies with tight lattice arrangement to increase the conversion ratio by reducing the moderation of neutron energy. The major specification of core design is shown in Table 3.8.1. The core design of 330MWe output with the operational cycle of 26 months was accomplished. A breeding ratio of 1.01, negative void coefficient and natural circulation cooling of the core were realized under the discharged burn-up of 60GWD/t. The core consists of 282 of hexagonal fuel bundles, which each have 217 of fuel rods with outer diameter of 13.0mm arranged

<table>
<thead>
<tr>
<th>Item</th>
<th>Unit</th>
<th>Design value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Electric output</td>
<td>MWe</td>
<td>330</td>
</tr>
<tr>
<td>Core radius</td>
<td>m</td>
<td>2.07</td>
</tr>
<tr>
<td>Core average burn-up</td>
<td>GWdt</td>
<td>60</td>
</tr>
<tr>
<td>Core height</td>
<td>m</td>
<td>0.74'</td>
</tr>
<tr>
<td>Core exit quality</td>
<td>%</td>
<td>52</td>
</tr>
<tr>
<td>Core average void fraction</td>
<td>%</td>
<td>88</td>
</tr>
<tr>
<td>Core pressure drop</td>
<td>MPa</td>
<td>0.04</td>
</tr>
<tr>
<td>Core average PuF content</td>
<td>%</td>
<td>10.7</td>
</tr>
<tr>
<td>PuF conversion ratio</td>
<td></td>
<td>1.01</td>
</tr>
<tr>
<td>Max. linear power density</td>
<td>kW/m</td>
<td>42.7</td>
</tr>
<tr>
<td>Void reactivity coefficient</td>
<td>10^-4 k/s/%void</td>
<td>0.5</td>
</tr>
<tr>
<td>Operation cycle length</td>
<td></td>
<td>26</td>
</tr>
</tbody>
</table>

1. Additionally, there are upper and lower blankets of 28.0 and 24.0 cm

Fig.3.8.1  Concept of core design

* Hitachi Ltd., **The Japan Atomic Power Company, ***Tokyo Institute of Technology
in triangular lattice of 1.3mm in gap width. As shown in Fig.3.8.1, the core part was shortened to 0.22m high and two core parts were piled up with an internal blanket region. Adding the upper and lower blanket regions, the total axial length becomes 1.21m.

Passive safety components were adopted to eliminate human factors from safety systems as much as possible and realize transparent safety. A small-scale natural circulation BWR was selected to meet the requirements for lower initial capital cost and distributed site. Figures 3.8.2 and 3.8.3 show the concepts of reactor pressure vessel and passive safety system, respectively. For the prevention of severe accidents, used passive systems include the automatic depressurization system, the gravity-driven safety injection system, and the passive containment cooling system (PCCS) utilizing horizontal heat exchangers submerged in a cooling pool. For the mitigation of severe accidents, used components include a newly proposed injection system called “siphon injection pipe” and a reactor vessel design that enables the retention of molten fuel in the reactor vessel by cooling the outside of the vessel (In-Vessel Retention). The proposed system also has advantages in the earthquake resisting due to the use of the PCCS pool that can be placed on a level lower than the containment top elevation such as the ground level.

Fig.3.8.2 Pressure vessel of the small-scaled RMWR

Fig.3.8.3 Safety system concept
The thermal-hydraulic feasibility studies including the characteristics of boiling transition (BT) and stability are needed because of the specific characteristics such as the tight lattice core, lower flow rate, high void fraction, axially double piled core, lower void reactivity coefficient, and natural circulation operation. As for the stability, the mechanisms of thermal-hydraulic instabilities that may appear in a natural circulation BWR were described, and the database for geysering instability were established for assessment with computer codes. Experimental calculations were performed for assessment of the applicability of the computer code TRAC-BF1 to the geysering instability. As shown in Fig.3.8.4, stable flow limitation of geysering can precisely be predicted with TRAC-BF1. The sensitivity calculations were performed on the channel and core stability to investigate the characteristics of the RMWR. The results showed that the core stability characteristics with respect to major parameters are similar to those of the current BWRs.

The R&D activities including critical experiments, safety analysis of MOX fuel and evaluation of MOX fuel reprocessing technologies are planned within the framework of the collaboration project.
3.9 Development of New Reactor Instrumentation System Using Optical Techniques - Multi-parameter Measuring System for Primary Coolant Circuit -

T. Kakuta, H. Yamagishi, T. Iwamura, and M. Urakami*
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A distributed multi-parameter measuring system based on optical fiber sensing technique was developed for use as a primary coolant monitor in nuclear power plants. It could be deployment for a stress-strain and a temperature monitors. The system has been evaluated in high gamma-ray radiation fields. The combined effects of stress-strain and radiation on the optical fiber sensor were demonstrated.

Feasibility of time-domain based measuring method.

A Raman scattering and a Brillouin scattering effects in the optical fibers, which depend on temperature and stress-strain, were used for this sensing technique. The system uses the principle of optical time-domain reflectometry (OTDR) where a laser pulse is launched in one end of optical fiber and the back-scattering signals are then detected. The sensor consists of a continuous length of optical fiber. The location of a temperature and a strain-induced change in scattering properties is found by the propagation delay of injected laser pulse (see Fig.3.9.1).

*The Japan Atomic Power Company
**System configuration.**

The measurement system is illustrated in Fig.3.9.2. The system is used working wavelength of 1550nm laser pulse and single-mode (SM) type optical fiber as a sensor for measuring parameters. In general, optical fibers are showing good radiation resistance in such working wavelength. Also, SM type optical fibers have good transmission characteristics for long distance over than 10km.

**Evaluation tests.**

Evaluation tests in the high gamma-ray environment were carried out in a cobalt-60 gamma-ray irradiation facility. A gamma-ray was irradiated up to a $10^6$Gy, with a dose rate of a few Gy/s.

For example, a result of distributed stress-strain measurement in the high gamma-ray field by Brillouin scattering OTDR is shown in Fig. 3.9.3. The strain of 0.3% in 4m was applied to the optical fiber at 2.95km from the measuring system. A gamma-ray irradiation at a $10^5$Gy did not affect a strain dependence of a measured value. Similar results were obtained in temperature measurements by Raman scattering OTDR.

Results of both parameters are indicating that the optical fiber sensor could survive a gamma-ray dose of $10^6$Gy at temperatures up to 600K. Thus, the multi-parameter measuring system could be used in primary coolant circuits in nuclear power plants. There, a gamma ray dose rate will be in the range of a few Gy/hr and optical fiber sensor will survive for a life of plants.
3.10 Neuro-Expert Monitoring System for Nuclear Power Plants

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This study presents a hybrid monitoring system for nuclear reactor utilizing neural networks and a rule-based real-time expert system¹. Main purpose of nuclear reactor monitoring is to identify the current status of the operational plant using process signals. The structure of this monitoring system is shown in Fig. 3.10.1. The simulator is a self-contained system designed to train power plant personnel in the general principles of a typical pressurized water reactor (PWR). The 87 analog signals are digitized and sent to neuro-expert monitoring system every two seconds. Out of these, 22 most significant plant signals are used for the plant modeling by neural networks. The advisory displays show the status of NPP diagnosed by neuro-expert system.

The simulator is manufactured on the basis of an existing 822 MWe power plant, Surry-1, USA. There are three primary coolant loops with three typical U-tube steam generators (SGs). The conventional alarm system is attached to the panel of the simulator. Therefore, operators can react to the events occurring as the actual NPP.

The auto-associative feedforward neural network is selected for plant modeling. The neural network has three layers: input, one hidden and output layer. The three layers are composed of 22 input nodes, 25 hidden nodes and 22 output nodes, respectively. The output signals are supposed to be the same as the input signals at the next time step. The merit of auto-associative network is that any unknown plant conditions, the inputs patterns in this case, never learned before always cause large deviation between measured and predicted. Therefore, it can easily be recognized as “something new”. The basic principle of the anomaly detection is to monitor the deviation between process signals measured from the actual plant and the corresponding values predicted by the neural network.

The expert system is used as a decision agent that works on the information space of both the neural network and the human operators. The information of other sensory signals is also fed to the expert system, together with the output patterns by the neural network. A major advantage of expert system is to process a lot of operator's knowledge and to derive useful

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information for complex decision environments. The expert system can treat almost all known correlation between plant status patterns and operation modes as a priori set of rules.

The graphical advisory displays are programmed by Java language and the monitoring results can be displayed on any computers connected to the Internet. Java is useful also to construct user-friendly man-machine interface. On this display, the schematic plant figures are shown with some important signals, the trend graph of the reactor outlet temperature signal and the messages from the expert system. The error messages and the diagnostic results are popped up on the screen when the monitoring system detects anomalies. The table of all measured signals and the trend graphs are also displayed on different pages.

The test results of the two malfunction cases during 2.5% per minute power decrease operation showed almost the same detection time and error signals. Figure 3.10.2 shows the feedwater pressure in the malfunction case of "Partial Loss of Feedwater". The solid line in the figure indicates the "measured" signals from the simulator. The dash-dot line represents the predicted values by the neural network. The diamonds indicate the deviation between measured signals and predicted values. The horizontal dotted lines show the fault severity levels for monitoring. When the deviation is in the range between two horizontal lines, the plant condition is considered normal; otherwise the monitoring system alerts anomalies.

The deviations were extremely small during normal power decreasing operation before the anomaly. Loss of feedwater (0.5%/hr) was introduced at 300 seconds point in Fig. 3.10.2. The monitoring system detected the fault of feedwater pressure signal 4 second after the malfunction started. However, the error disappeared after 430 second because of the recovery of feedwater pressure by the control system. In such a case, the operator without neuro-expert monitoring system might not recognize the anomaly for long time. The steam flow signal in another malfunction case is shown in Fig. 3.10.3. The neural network detected the anomaly 2 seconds after the malfunction started. The response of the monitoring system was similar to one during steady state in both cases. Therefore, the information of detection signal and time is very useful to identify anomalies by rule-based expert system at the next stage.

During the tests, the rule-based expert system correctly recognized the operation modes, and diagnosed the plant status, because the detection patterns by the neural network are dependent on the kind of anomalies, and independent on the amount of anomaly or operation modes.

From the testing results, it was shown that the neural network in the monitoring
system successfully modeled the plant dynamics and detected the symptoms of anomalies earlier than the conventional alarm system. The real-time expert system also worked satisfactorily in diagnosing and displaying the system status by using the outputs of neural networks and a priori knowledge base. The advisory display also worked satisfactorily.

Reference

Fig. 3.10.1 Monitoring system overview

Fig.3.10.2 Monitoring result of "Partial Loss of Feedwater"

Fig.3.10.3 Monitoring result of "Leakage of Atmospheric Steam Dump"
4. Thermal and Fluid Engineering

Advanced computer programs have been developed and several tests were performed to develop thermal hydraulic analyses method in next-generation nuclear systems such as reduced moderation water reactors (RMWRs), fusion reactor, and a target with high heat flux.

Design studies were performed to check the feasibility of the proposed concept of RMWRs under several abnormal events in FY 2000. An analytical evaluation was performed for PWR type reactor against a large break loss-of-coolant accident. The predicted peak clad temperature was lower than the design guideline by about 250 K. A critical heat flux (CHF) test was performed using the tight-lattice fuel assembly including seven heater rods arranged in the triangular array with 0.6 mm rod gap size. Although the test was performed under typical PWR conditions, dryout type CHF was predominant in the test as well as the previous ones with 1.0 or 1.5 mm gap size. It seems that the critical-boiling length approach is effective for critical power predictions of the tight-lattice bundle. To support the design work of RMWRs, advanced computer programs have been developed and verified in FY 2000. A multidimensional two-phase flow analysis code based on the two-fluid model (we named ACE-3D) has been improved by the new turbulent model for the bubbly flow. The code predicted excellently the phase distribution in various pipes regardless of their diameter. A mechanistic boiling transition model started to develop as well as the two-phase flow simulation code with interface tracking (we named TPFIT code).

A series of experiments on thermal-hydraulic safety in a fusion reactor was performed to get validation data for safety analysis codes of fusion reactors as one of International Thermonuclear Experimental Reactors (ITER) subtasks. In FY 2000, the test facility for the simulation of the ingress of coolant event (ICE) in the ITER was remodeled to simulate the new ITER vacuum vessel pressure suppression system. Several experimental and analytical works were performed for ICE and dust mobilization behavior during loss-of-vacuum events.

A model experiment of CHF was also performed to understand the burnout under subcooled conditions with extremely high heat flux. A new correlation on CHF was proposed and point of net vapor generation was investigated based on the void fraction dataset obtained by high-frame-rate neutron radiography.
4.1 Study on Cooling of RMWR PWR type Core during the Reflood Phase

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At present, reduced-moderation water reactor (RMWR) PWR type core with tight lattice core is developed as one candidate for the RMWR concepts\(^1\). To achieve a conversion ratio greater than unity, the tight lattice core is used. Under such tight configuration, the core thermal margin becomes smaller and should be evaluated in a normal operation and in the case of accidents. In this study, we have performed a feasibility evaluation on core cooling under the large break LOCA (LB-LOCA) accident of the RMWR PWR type system.

The electrical output of the reactor system is 1000MWe, and heavy water is used as the primary coolant and the moderator. The outline of the reactor vessel is shown in Fig. 4.1.1. The pressure vessel diameter is bigger in comparison with a current PWR and core height is smaller than current one. In the lower part of the core, the space of 2.3 m was set up in consideration of the adoption of the control rod follower. The present design adopts seed assemblies and several blanket assemblies. There are two types of seed assemblies.

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Fig. 4.1.1 Cross section of RMWR pressure vessel
One has internal, upper and lower axial blanket regions, and the other has no internal axial blanket region. In the blanket regions (internal, upper and lower axial blanket regions and blanket assemblies), power density is lower than that of the seed regions. There are the channel boxes to each fuel assemblies in order to restrict the movement of the coolant and adjust the flow rate in each assembly².

Figure 4.1.2 shows input nodarization of RMWR system. The current 4-loop PWR system is used as basis of the system of RMWR PWR type core. However, to fit into the bigger pressure vessel volume than that of current one (about 1.5 times), we set up the capacity of the pressurizer and also accumulator (1.5 times and 3 times of the current PWR respectively). In the simulation, The REFLA-TRAC cord is used. The reactor core is modeled with 6 one-dimensional components. Each three types of the assemblies are represented by the 2 one-dimensional components, one is maximum heat output assembly and the others have averaged value of heat output.

The temperature change of the clad surface of the maximum heat output assembly is shown in Fig. 4.1.3. Although the clad temperature is reached about 1200K in the position of 1.7 m from the lower core support plate, it is sufficiently lower than the design guideline of the current water reactor. The core cooling of the RMWR PWR type core in the case of
LOCA accidents is able to maintain by designing pressurizer capacity etc. appropriately.

References:
1) Hibi, K. et al., Conceptual designing of reduced-moderation water reactors (2) -Design for PWR-type reactors-, Proc. 8th Int. Conf. of Nuclear Engineering (ICONE8), Baltimore, April 2000.

Fig. 4.1.3 Clad temperature of the maximum heat output assembly
4.2 Development of 3-Dimensional Two-Phase Flow Simulation Code with interface tracking

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To improvement and establishment of exist two phase flow simulation code, detailed information is needed. However, it is difficult to obtain the detailed information of the flow fields, such as the bubble, the liquid drop or liquid film flow etc., by the experimental procedure. Also, the cord that gives more detailed information than the exist codes is requested to model the solid-fluid interaction and/or to understand the boiling phenomenon. So, we are trying to develop the two-phase flow simulation code with interface tracking (we named TPFIT). In this report, outline of the code is explained and examples of the cord results are shown.

The CIP (Cubic Interpolated Pseudo-particle) method\textsuperscript{1) that can treat the both uncompressible and compressible fluid was adopted in this code. The ICCG method is used to solve the Poisson equation of pressure. The staggered mesh and Cartesian coordinates are used. To estimate surface tension, the CSF model\textsuperscript{2)} is incorporated.

In order to show the potential of the TPFIT, two examples are carried. The examples of the simulation result are shown in Figs. 4.2.1 and 4.2.2. In the simulations, two-dimensional grid (70×100) and air and water properties are used. The example 1 is the case that 2 bubbles (lower bubble and upper bubble diameters are 3 and 4 cm respectively) are arranged in the water pool.

\begin{figure}[h]
\centering
\begin{subfigure}{0.3\textwidth}
\includegraphics[width=\textwidth]{fig1.png}
\caption{0 sec.}
\end{subfigure}
\begin{subfigure}{0.3\textwidth}
\includegraphics[width=\textwidth]{fig2.png}
\caption{0.05 sec.}
\end{subfigure}
\begin{subfigure}{0.3\textwidth}
\includegraphics[width=\textwidth]{fig3.png}
\caption{0.09 sec.}
\end{subfigure}
\begin{subfigure}{0.3\textwidth}
\includegraphics[width=\textwidth]{fig4.png}
\caption{0.12 sec.}
\end{subfigure}
\caption{4.2.1 Example (1) Air bubbles in the pool}
\end{figure}
Both bubbles are moving to the upper direction by the buoyancy force. Moving velocity of the lower bubble is larger than that of upper one, because of effects of the velocity filed of the water. So, the lower bubble comes into collision with upper bubble. Then the lower and upper bubbles become one large bubble. Finally, the bubble breaks away from the pool surface.

Figure 4.2.2 shows results of the air injection to the water pool. The air is injected to the water pool through the 2 cm diameter nozzle at 10 m/s. In the simulation, irregular interval grid is used, to get the detailed information around the nozzle exit. The velocity field around the nozzle exit is vibrating right and left direction no periodically.

References:
2) Brackbill, J., U. et al. J. Computational Physics, 100, 335 (1992)
4.3 Model Development for Bubble Turbulent Diffusion and Bubble Diameter in Large Vertical Pipes

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Multi-dimensional analyses have been expected recently with expanding computation resources for gas-liquid two-phase flow analyses in advanced nuclear systems (passive safety systems, natural-circulation-type reactors, etc.). However, the applicability of previous constitutive equations for multi-dimensional analyses has not been fully investigated especially for the effects of flow path scale because the equations have been assessed for small-scale experiments. In this study, we analyzed the applicability by the multi-dimensional two-fluid model code ACE-3D using data in 38 mm and 200 mm diameter pipes. We clarified a key-parameter to model the scale effects and developed new models for the effects on phase distribution. The scale effects can be classified by the relative relationship between bubble diameter $d_b$ and turbulent length scale $l_T$. Bubble-induced turbulence is increased under that $d_b$ is smaller than $l_T$ and bubble coalescence is predominated rather than breakup under that $l_T$ is about three times larger than $d_b$ and under higher void fraction. Based on these findings, we established new models for bubble turbulent diffusion and bubble diameter.\(^1\)

Figure 4.3.1 compares the developed model with the database of 200 mm pipe ($r$: radial position from pipe center, $R$: radius of pipe). The wall-peak phase distribution changes to the core-peak one with superficial air velocity $J_G$. The developed model can trace the trend. The core-peak prediction under $J_G = 0.26$ m/s was attained by a bubble coalescence in the model. Since some large coalesced bubbles were observed also in the experiment\(^2\), the model corresponds well to the physical situation.

Figure 4.3.2 shows the comparison for 200 mm and 38 mm pipes. In 200 mm pipe, the superficial water velocity $J_L$ is lower than that in Fig. 4.3.1. A bubble coalescence was observed in the experiment and the void fraction near the pipe center is higher than that near the wall. In the present model calculation, the bubble coalescence was also attained under higher void fraction due to the lower $J_L$ and the void fraction is well predicted. In 38 mm pipe, the wall-peak is still maintained in the experiment even under the lower $J_L$ although the difference of void fraction between regions near the center and the wall becomes smaller with decreasing $J_L$. The present model can predict the tendency.
Figure 4.3.3 shows the comparison for 480 mm pipe. We selected two databases: one is using the sinter tube (grain size: 50 μm) and the other is the nozzle (0.07 m inner diameter) as an air injection device. In the case of using the nozzle device, the top of nozzle was located at the center of test section bottom. Left figure corresponds to the sinter case and right one to the nozzle case. The location of sinter tube is at about 0.6 in r/R. The developed model predicts well the phase distribution at different elevations irrespective of the air injection method.

From this study, we have successfully modeled the effects of flow path scale on the phase distribution. Since the present models are based on the local flow parameters, we can expect a high applicability to different geometries. However, the assessments against steam-water systems are indispensable to apply to design and safety analyses of several nuclear systems. We now try to apply the present models to steam-water two-phase flow in 155 mm vertical pipe and we recognize that the applicability is also high. We have a plan to publish the assessment results in future.

References

![Graph showing phase distribution](image)

**Fig. 4.3.1** Prediction of phase distribution in 200 mm pipe
Fig. 4.3.2 Assessment for different flow rate conditions in 200 mm and 38 mm pipes

Fig. 4.3.3 Assessment for 480 mm pipe
4.4 Development of Mechanistic Boiling Transition Model
- Master Plan and Development of Flow Model -

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Design studies of reduced-moderation water reactors are being carried out at JAERI as one candidate for future reactors. The reactor adopts a tight lattice core with about 1 mm gap between adjacent fuel rods. To optimize the thermal design, boiling transition (BT) in such a tight lattice core is one of most important phenomena to be evaluated, but effects of the gap and spacer configuration have not been fully investigated. To evaluate the feasibility and to optimize the thermal design, a full-scale bundle test is required but several systematic full-scale tests are difficult to perform during an initial design phase. Thus, we made a plan to develop a mechanistic BT model to evaluate the geometry effects by a numerical simulation. This section indicates the master plan for the BT model development and describes an outline of developed flow model.

Figure 4.4.1 shows the master plan. Multi-dimensional thermal-hydraulic model for annular-dispersed flow is constructed and the function to treat a general curvilinear coordinate is installed to simulate the rod bundle including grid spacers. The validity of entrainment and droplet dispersion models is investigated using air-water experimental data in a vertical pipe and the applicability is verified by experimental data in a flow path with obstacles and in a rod bundle. We also have a plan to obtain database for rod bundle with heat transfer by neutron radiography CT and by full-pressure tests.

We have constructed a flow model for annular-dispersed flow based on multi-dimensional two-fluid model. Figure 4.4.2 shows the outline of the flow model. As the turbulent model in the dispersed droplet region, k-ε type one for gas-particle flow is applied with some modifications. The existence of water film on the wall is assumed and the boundary between the film and the droplet regions, where the entrainment is considered, is judged by the void fraction. The friction model on the film surface and the boundary conditions for the turbulent model on the same location are established using the air-water experimental data in a vertical pipe.

Figure 4.4.3 indicates an example for the simulation of air-water mixer. Air is supplied at the bottom of calculation domain with superficial velocity 20 m/s and water is at the side wall with 0.4 m/s. Droplets are generated at the location of water injection and at
the film surface. The droplets are diffused towards the pipe center. We will develop the flow and the heat transfer models using the vertical pipe data and verify the applicability using several databases including rod bundles in future.

References

![Diagram](image-url)

**Fig. 4.4.1** Master plan for development of BT model

![Diagram](image-url)

**Fig. 4.4.2** Outline of flow model
Fig. 4.4.3  Simulation of air-water mixer - water mass velocity and contour of void fraction
4.5 Critical Heat Flux Experiments of Tight Lattice Core for Reduced Moderation Water Reactor

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It is important to evaluate the thermal margin of the Reduced-Moderation Water Reactor (RMWR) which consists of tight lattice fuel assemblies with gap clearance around 1.0 mm. Critical heat flux (CHF) experiments were performed to construct a database for the evaluation with three tight lattice bundles simulating RMWR core. Table 4.5.1 shows the experimental conditions. The pressure was set to 15.5MPa, considering the PWR-type RMWRs. The power distribution was chopped cosine shape.

The previous reports presented that the combination of the subchannel analysis code COBRA-VI-I and the KfK correlation gives conservative predictions for the local CHF\(^1\). The present paper reports further considerations.

The following facts observed in the experiments imply that the phenomena are not DNB but dryout even in the PWR pressure condition.

- For most of the present data, CHF occurs at the outlet with relatively lower heat flux.
- The temperature rises at CHF were milder than that observed at the typical DNB.
- The bundle-average equilibrium outlet quality is over 0.1, much higher than that of the current PWRs.

Those led us to analyze the data with approaches commonly used for film dryout in BWR. In these circumstances, upstream profile has a significant effect on CHF. There are several methods of correlating boiling transition data for axially non-uniform heat fluxes. Two widely accepted methods for analyzing BWR are the Tong F-factor correlation and the critical quality-boiling length approach. Since the latter is easy to handle, we adopted it in the present study.

Data were plotted in Fig.4.5.1 in the critical quality vs. boiling length plane. Critical quality was obtained with the bundle-average equilibrium heat balance equation. The data
indicate that critical quality is indeed a function of boiling length. The trend of inlet temperature and quality contend with that obtained in dryout.

There are a lot of critical quality correlations that extend to the steam qualities of the present tests. With those correlations, we can predict the bundle critical power, not the local CHF. Three versions of these correlations have been examined. The first is CISE correlation. The second is Biasi correlation. This is originally a heat flux-quality correlation, and was translated to critical quality-boiling length type\(^2\). The third is modified Biasi correlation proposed by Arai et al\(^3\). It should be noted that the pressure of our tests exceeds the range of validity of Biasi correlation.

Among the three correlations, Biasi’s one gives the best predictions for the critical power. The results are shown in Fig. 4.5.2-4.5.4. For the gap clearance of 1.5 mm, it gives excellent predictions (<10%). For the cases of 0.6 and 1.0 mm, it gives conservative predictions. The difference of critical power between the analysis with Biasi correlation and the experiments was smaller than that of local CHF with the combination of the KfK correlation and the COBRA-VI-I. CISE and modified Biasi correlation give lower values than Biasi’s predictions for all the cases.

The critical quality-boiling length approach may be effective for critical power predictions of the tight lattice bundle. As for the differences between the analysis and the experiment in the cases of gap clearance lower than 1.0 mm, more precise analysis is needed.

References:


Table 4.5.1 Experimental conditions

<table>
<thead>
<tr>
<th>Gap clearance (mm)</th>
<th>Mass velocity (kg/m²s)</th>
<th>Inlet temperature (K)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1.5</td>
<td>1,100~4,400</td>
<td>533~593</td>
</tr>
<tr>
<td>1.0</td>
<td>1,100~3,100</td>
<td>513~593</td>
</tr>
<tr>
<td>0.6</td>
<td>1,375~4,585</td>
<td>513~593</td>
</tr>
</tbody>
</table>

Fig. 4.5.1 Relation between critical quality and boiling length

Fig. 4.5.2 Comparison of Biasi prediction and data (gap=1.5mm)

Fig. 4.5.3 Comparison of Biasi prediction and data (gap=1.0mm)

Fig. 4.5.4 Comparison of Biasi prediction and data (gap=0.6mm)
4.6 Critical Heat Flux Correlation for Subcooled Boiling Flow in Narrow Channels

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The purpose of this study is to conduct the critical heat flux (CHF) correlation for narrow channels. The CHF of subcooled flow boiling of water in narrow rectangular channels under the atmospheric pressure was measured parametrically. Experimental test channels were rectangular channels heated from one side with the channel gap of 0.2 – 3.0mm, channel width of 7 - 22 mm, and heated length of 50 - 200mm. First, the CHF correlation for the one-side heated rectangular channels was proposed with investigating the various system parameter effects on CHF. Effects of the channel gap and width on CHF and critical quality were shown in Figs. 4.6.1 and 4.6.2, respectively. The following knowledge was obtained under present experimental conditions.

1. CHF do not decrease remarkably with decreasing the channel gap.
2. Effects of the heated width on CHF and critical quality are small.
3. Critical quality decrease with decreasing the heated length.
4. Critical quality increases with increasing the inlet water temperature.
5. In case of tubes, the dimensionless CHF parameter $K^*$ tends to become higher value in comparison with that for one-side heated narrow rectangular channels.

Next, applicability of the correlation to the both-side heated rectangular channel, half-circumferentially heated tube, and full-circumferentially heated tubes was examined. Figure 4.6.3 shows the correlation between the dimensionless CHF parameter $K^*$, which is defined using Boiling number, viscous force and surface tension, and the exit thermal quality at the burnout. It was found that the dimensionless parameter $K^*$ becomes higher value in tube. New CHF correlation for narrow rectangular channels and small-diameter tubes was proposed using the critical quality, dimensionless CHF parameter and heated perimeter ratio as follows.

$$K^* = \frac{q_{CHF}}{G \cdot h_R} \left( \frac{G \cdot \nu}{\sigma} \right)^{0.5} = C_1 \left( \nu_{e,CHF} + C_2 \right)$$

$$C_1 = 6.9 \left( \frac{P_a}{P_v} \right)^2 - 10 \left( \frac{P_a}{P_v} \right) + 2 \times 10^{-3}$$

$$C_2 = -0.75 \left( \frac{P_a}{P_v} \right)^2 + 0.9 \left( \frac{P_a}{P_v} \right) - 0.28$$

Figure 4.6.4 shows the comparison of the dataset taken in the various heating channels.
with the prediction by the new correlation. Calculation accuracy of the correlation is ±45% (Maximum 10 times better in comparison with the existing CHF prediction methods which were proposed for the full-circumferentially heated tubes).

Detailed results were published in the JSME paper1).

References

Fig. 4.6.1 Effect of the channel gap on CHF and critical quality

Fig. 4.6.2 Effect of the channel width on CHF and critical quality

Fig. 4.6.3 Correlation between the dimensionless CHF parameter $K'$ and critical quality

Fig. 4.6.4 Comparison of the calculated CHF by the new correlation with the experimental CHF data
4.7 Study on Vapor Generation Point and CHF Model Using Void Fraction Database by Neutron Radiography

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Point of net vapor generation (PNVG) was investigated based on the void fraction dataset obtained by high-frame-rate neutron radiography. The test channels used in the experiment were rectangular channels heated from one side with channel gap of 3 and 5 mm, channel width of 30 mm, and heated length of 100 mm. Measured instantaneous void fraction and vapor generation point were shown in Figs. 4.7.1 and 4.7.2, respectively.

In this study, we discuss on (1) the determination of the instantaneous and time-averaged PNVG, (2) the effects of system parameters on PNVG, (3) the applicability of existing PNVG correlations to the channel with short heated length, and (4) the effect of the PNVG in CHF model. The following results were obtained:

(a) From the comparison between instantaneous PNVG and time-averaged PNVG, the thermal equilibrium quality at the instantaneous PNVG varied in the range within ±0.002 from the thermal equilibrium quality at time-averaged PNVG.

(b) Effects of the heat flux, channel gap, mass velocity, and inlet water temperature on the thermal equilibrium quality at time-averaged PNVG were weak under the present conditions (Figure 4.7.3).

(c) From the comparison of existing PNVG correlations and the measured values, the correlations tended to underestimate the thermal equilibrium quality at PNVG.

(d) By changing the PNVG correlation in Katto’s CHF model\textsuperscript{1} from Saha-Zuber’s correlation\textsuperscript{2} to an empirical correlation developed by the measured PNVG in the present experiment, the prediction accuracy of modified Katto’s model was improved from C/E\textsuperscript{0.5} = 3 to C/E\textsuperscript{0.5} = 1.2, where C/E indicates the ratio of prediction to experimental values. (Figures 4.7.4 and 4.7.5).

Detailed results were published in the papers\textsuperscript{3,4}.

References


Fig. 4.7.1 Instantaneous void fraction measured by high-frame-rate neutron radiography

Fig. 4.7.2 Distribution of the vapor generation points

Fig. 4.7.3 Void fraction profile along the flow channel (thermal quality)

Fig. 4.7.4 Measured time-averaged void fraction distribution and calculation of void fraction profile by the present PNVG correlation and homogeneous flow model.

Fig. 4.7.5 Comparison of modified CHF model with experimental data
4.8 Remodeling of an Integrated ICE Test Facility

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An integrated ingress-of-coolant event (ICE) test facility \(^1\) was constructed to investigate the performance of a vacuum vessel pressure suppression system (VVPSS) in the International Thermonuclear Experimental Reactor (ITER) and a series of VVPSS performance tests were carried out using the integrated ICE test facility, and then, pressure rise and two-phase flow characteristics were clarified quantitatively.

Figure 4.8.1 shows a concept of the VVPSS in ITER-FDR \(^2\). The water injected from the cooling tubes into plasma-facing component (PFC) flows through the divertor slits to the bottom of a vacuum vessel (VV) and the accumulated water in the VV goes through a relief pipe to a suppression tank (ST). At this time a great amount of vapor generates due to the flashing under vacuum and boiling heat transfer from the plasma-facing surfaces, and then, the pressure inside the PFC and VV increases. Because of the pressurization a couple of rapture disks which are settled at the relief pipe are broken and the water under high temperature and vapor flow into the ST. The ST initially holds water under low temperature and pressure (around 25°C and 2300 Pa). Therefore, water under high temperature and vapor can be cooled down and condensed inside the ST, and consequently, the pressure in the ITER can be decreased.

However, it was found from the design study of the ITER VVPSS that the efficiency of condensation in the ST is not high because the water temperature in the ST becomes high due to the mixture of water and vapor. Then, the ITER VVPSS design was remodeled to upgrade the condensation performance in the ST. Figure 4.8.2 shows a concept of new ITER VVPSS design \(^3\). Three relief pipes are installed at the upper region of the VV and connected to the ST and a drain tank (DT) is settled at the bottom of the VV with a drain pipe. This new VVPSS design concept using the ST and DT was considered to enhance the condensation performance in the ST by separating the vapor and water from the mixture at the ICE.

To simulate the new ITER VVPSS design the integrated ICE test facility was upgraded. Figure 4.8.3 shows an appearance of the integrated ICE test facility. It simulates the ITER structural components with a small-scale model of 1/1600 and mainly consists of a plasma chamber (PC), divertor, VV, relief pipes, ST, DT and drain pipe. Three relief pipes were
installed to the top of the PC and connected to the ST. The DT was connected to the bottom of the VV through a drain pipe. An electric valve is settled to each relief pipe and drain pipe. Dimensions of the relief pipe such as the diameter and length were determined as the pressure drop in the relief pipe between PC and ST is equal to that in ITER. Similarly, dimensions of the drain pipe were also determined.

References:
Fig. 4.8.2 A concept of new ITER-VVPSS design

Fig. 4.8.3 Upgraded integrated ICE test facility
4.9 Visualization of Water-Vapor Two-Phase Flows in Fusion Reactors during Ingress-of-Coolant Events

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Water-vapor two-phase flow configurations in a plasma chamber (PC) and vacuum vessel (VV) of an integrated ingress-of-coolant event (ICE test facility were observed visually using CCD (Charge-Coupled Device) cameras to clarify thermal-hydraulics in fusion reactors under the ICE phenomenon. On the other hand, numerical analyses with the TRAC-PF1 code, which is a safety analysis code for light water reactors, were carried out to understand the two-phase flow behavior during the ICE and the prediction accuracy of the TRAC-PF1 code was verified based on the results of the integrated ICE tests. Figure 4.9.1 shows an analytical model simulating the integrated ICE test facility. The test facility mainly consists of a boiler, PC, VV, divertor, suppression tank (ST), relief pipe and water injection nozzles.

Figure 4.9.2 shows the initial condition inside the PC and VV just before the flow visualization experiment. The left column shows the observation results in the PC and VV. The injected water into the PC goes through the divertor to the VV. The water accumulated in the VV flows through the relief pipe into the ST after the magnet valve is opened.

Figure 4.9.3 shows the experimental and analytical results during the injection of water into the PC from the three sets of water injection nozzles. The experimental results are shown at the left-hand side. The inside of the PC is occupied with vapor due to flashing as soon as the water is injected into the PC, and at the same time, the injected water drops down to the VV through the slits at the divertor. On the other hand, the analytical results regarding to the void fraction and velocity vector are represented at the right-hand side in Fig.4.9.2. At the color bar of the void fraction the red is 1 and means 100% vapor and the blue is 0 and means 100% water. Similarly, at the color bar of the velocity vector the blue shows 0 m/s and the red shows 20 m/s. It was clear from the analytical result that the injected water into the PC is accumulated on the divertor and then goes down to the bottom of the VV through the divertor slits. Furthermore, the configuration of convection of vapor inside the PC and VV due to the flashing was clarified quantitatively.

After the pressure in the PC reaches 0.15 MPa and the magnetic valve at the relief pipe is opened, the accumulated water in the bottom of the VV flows to the ST through the relief pipe. As a result of this, the void fraction in the VV rises up gradually to 1. This
matter can be verified well by the void fraction distribution in the analytical result shown in Fig. 4.9.4. After that the pressure inside the test facility decreases due to the condensation in the ST and the remaining water inside the PC and VV evaporates again because of the decrease of saturation pressure.

![Analytical model of the integrated ICE test facility by the TRAC-PF1](image1)

Fig.4.9.1 Analytical model of the integrated ICE test facility by the TRAC-PF1

![Initial condition in the PC and VV just before the flow visualization experiment](image2)

Fig.4.9.2 Initial condition in the PC and VV just before the flow visualization experiment
Fig.4.9.3 Experimental and analytical results during the injection of water

Fig.4.9.4 Experimental and analytical results after the VV is connected to the ST
4.10 Numerical Predictions of Air Ingress and Dust Mobilization Behavior in Fusion Reactors during Loss-of-Vacuum Events

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Air ingress and dust mobilization characteristics in ITER (International Thermonuclear Experimental Reactor) during a loss-of-vacuum (LOVA) event were analyzed numerically by three dimensional simulations using a newly developed thermal-hydraulic analysis code. Physical models on the motion of dust were considered to resolve the dust mobilization conveying by the fluid. Air ingress behavior through a breach at the LOVA event was calculated by using compressible Navier-Stokes equations.

An analytical model in Fig.4.10.1 simulates the dimensions of the actual ITER design. In Fig.4.10.1 the analytical region corresponds to the inside of the cryostat in ITER. The no-slip and adiabatic boundary conditions were adopted for the top, bottom and side walls in the analytical region. The VV was put at the center of the analytical region and the no-slip and adiabatic boundary conditions were set for all the walls inside and outside the VV. The breach was positioned at the top of the VV and two kinds of breach size were provided: 0.5 and 0.05 m². Here, the breach size of 0.5 m² simulates the estimated large break case in ITER, and similarly, that of 0.05 m² corresponds to the estimated small break case.

The input conditions for the fluid were: initial pressure inside the VV 1000 Pa; wall temperature in the VV 200°C; and external fluid outside the VV is air of 0.1 MPa and 20°C. On the other hand, the dust particle conditions were: diameter 0.1 mm; density 7900 kg/m³ (i.e., Stainless steel); initial velocity 0 m/s; and total particle number 10⁴. The initial position of the dust is at the floor of the VV. In the present numerical analysis the wall restitution coefficient was assumed 0.5 because of no exact data and the input conditions were set up based on an assumption that the fluid is in the continuum region (i.e., Knudsen number<10⁻²).

Figure 4.10.2(b) shows the predicted results on the fraction of mobilized dust in the VV when the breach size is 0.05 m². Here, the solid line represents the fraction of mobilized dust existing in the breach side of the VV and the dashed line is that existing in the opposite side. Each position of the breach side and opposite side in the VV can be seen in Fig.4.10.2(a). The fraction of mobilized dust was specified as the ratio between the mobilized dust number and initial dust number. In the present numerical simulations the total dust number is 10⁴ and each half of 10⁴ was initially put on the floor in the breach side and opposite side in the VV,
respectively. The dust mobilization is recognized after about 0.05 s from the beginning of the air ingress into the VV. The dust in the breach side moves to the opposite side due to the ingress air and the fraction of mobilized dust in the opposite side increases from 0 to 100%.

Figure 4.10.3 shows the predicted results on the fraction of mobilized dust in the VV when the breach size is 0.5 m². Here, the solid and dashed lines represent the results in the breach and opposite sides and those trends are almost the same as Fig.4.10.2(b).

Figures 4.10.4 and 4.10.5 show the predicted results of the dust-air two-phase flow when the breach size is 0.5 m² and 1 and 30 s later from the beginning of the air ingress, and then, (a) and (b) in Figs. 4.10.4-5 present the horizontal and vertical views on the mobilized dust. The stream from the breach is stronger with increasing the breach size and the dust mobilization behavior is accelerated in comparison with the case that the breach size is small. A buoyancy-driven exchange flow through the breach occurs due to the density difference based on the temperature difference between the inside and outside of the VV after the pressure in the VV is equal to atmosphere and then a small amount of the dust is released through the breach to the outside of the VV conveying by the exchange flow. In case of the small-break case the release of the mobilized dust through the breach to the outside of the VV did not occur. On the other hand, in case of the large-break case around 8% in the total dust number was released to the outside.

References:

Fig.4.10.1 Analytical region and boundary conditions
Fig. 4.10.2 Fraction of mobilized dust in the VV with the breach size of 0.05 m²

Fig. 4.10.3 Predicted results on the fraction of mobilized dust in the VV with the breach size of 0.5 m²

Fig. 4.10.5 Predicted dust mobilization after 30 s from the ingress of air
5. Energy System Analysis and Assessment

Development of analytical tools and application studies have been made in order to analyze the role of nuclear energy in Japan's future energy systems and to make assessment of new nuclear technologies and systems from such viewpoints as reducing uranium resource utilization, improving economics, or decreasing environment risks. Major activities during the fiscal year 2000 are summarized below.

In the development of analytical tools, a macro-economy model was developed and improved to make a study on economic impacts of nuclear phase-out in Japan. Since the energy model to be linked is of a linear optimization type, the Turnpike model was selected as the basic framework of the macro-economy model. In addition to this, the development of a detailed simulation model of nuclear fuel cycle systems was initiated.

In the study on power reactors and nuclear fuel cycle systems, an analysis was made on economics of plutonium recycling in Japan. Levelised electric power generation costs were calculated for enriched uranium light water reactors, fully mixed oxide fueled light water reactors, reduced moderation water reactors (RMWRs), and fast breeder reactors. It was found from this analysis that, although plutonium recycling is generally expensive, higher fuel burn-up and lowering of reprocessing service prices will substantially improve the economics of plutonium recycling technologies, in particular, of RMWRs. Beside this analysis, in order to analyze back-end systems of nuclear fuel cycle, a database was developed on reprocessing wastes of future nuclear reactors including the hazard index data of minor actinides in high level radioactive wastes.

In the study on energy systems, a preliminary analysis was made on the impacts of nuclear phase-out in Japan by using a stand-alone energy model. By comparing the costs of the two scenarios, nuclear expansion and phase-out, it was indicated that the additional energy system costs by nuclear phase-out are substantial, and that the analysis by the linked model will be useful in order to understand the economic implication of the additional costs. In addition to this, an investigation was made on experience curves of decentralized power systems. By focusing on solar photovoltaic power (PV) systems historical data on system prices were compiled. It was found that a major part of the observed reduction of the system prices can be attributed to the decrease of prices of other components than PV modules, so that the reduction of PV module prices will be important in the future.
5.1 An Analysis on the Economics of Plutonium Recycling

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The development of breeder reactors and fuel cycle technologies is being made in order to realize breeding cycle of plutonium. However, it is expected that these technologies might be too expensive to utilize on a commercial basis, therefore the improvement in economics of plutonium recycling is becoming an important subject for the future research and development. In this study\textsuperscript{11}, economics of plutonium recycling technologies was analyzed for comparison with enriched uranium LWRs and to find measures to improve it.

Levelised costs of generating electricity were calculated in order to analyze the economics of nuclear power reactor systems with enriched uranium LWRs (EU-LWR), fully MOX fueled LWRs (FM-LWR), reduced moderation water reactors (RMWR), or sodium-cooled fast breeder reactors (FBR). For RMWR and FBR two reactor types with different fuel burn-up were considered. Major assumptions are shown in Table 5.1.1. Characteristics of the nuclear power plants were set by assuming that reactor plants will be in operation around the years 2020-2030. Spent fuel reprocessing was assumed for all reactor types including EU-LWR. Cooling time of spent fuel before reprocessing was set as 4 years for EU-LWR and FM-LWR, and 2 years for RMWR and FBR. It was also assumed that geological disposal of vitrified high level wastes is made 40 years after reprocessing of spent fuel.

Levelised fuel cycle costs of the above nuclear power systems are compared in Fig. 5.1.1. The costs of EU-LWR are given for the two cases; a reference case using conventional uranium resources and a high cost case assuming the use of uranium from seawater. The costs of plutonium utilization reactors are also given for the two cases with different reprocessing service prices; a reference case with 390,000 yen/ton and a comparison case with 200,000 yen/ton. It is obvious from this figure that the fuel cycle costs of plutonium utilization reactors are very sensitive to the prices of reprocessing, a substantial reduction is expected when fuel burn-up is made higher and/or reprocessing prices are lowered. Since fuel fabrication and spent fuel reprocessing account for more than 90% of the fuel cycle costs, they can be reduced by at largest 26% if reprocessing service prices can be lowered to the half.

Total levelised generation costs are compared in Fig. 5.1.2. Lifetime extension of nuclear power plants from 40 years to 60 years will reduce the generation costs, although by only 5%. Reprocessing prices are much more sensitive to the generation costs; if lowered to the half, the generation costs will be decreased by 12% at largest.

The generation costs of plutonium utilization reactors change significantly depending
on fuel burn-up. The generation costs of RMWR-2 (higher burn-up) are much lower than RMWR-1 (lower burn-up), only about 10% higher than the costs of EU-LWR when reprocessing prices of MOX fuel is of the comparison case.

In summary, it was confirmed that the electric power generation costs of plutonium utilization systems are generally higher than those of enriched uranium LWR systems, and higher fuel burn-up and lowering of spent fuel reprocessing prices are necessary in order to improve their economic competitiveness.

Reference

Table 5.1.1 Major Assumptions for Analysis

1. Nuclear plant specifications

<table>
<thead>
<tr>
<th></th>
<th>Unit</th>
<th>EU-LWR</th>
<th>FM-LWR</th>
<th>RMWR-1</th>
<th>RMWR-2</th>
<th>FBR-1</th>
<th>FBR-2</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fuel burn-up</td>
<td>GWe/t</td>
<td>45</td>
<td>33</td>
<td>45</td>
<td>70</td>
<td>100</td>
<td>150</td>
</tr>
<tr>
<td>Load factor</td>
<td>%</td>
<td>90</td>
<td>86.7</td>
<td>90</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Life of plant</td>
<td>Year</td>
<td>40 or 60</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Fuel inventory</td>
<td>t/GWe/Year</td>
<td>21.1</td>
<td>29.9</td>
<td>32.8</td>
<td>18.5</td>
<td>C</td>
<td>12.9</td>
</tr>
<tr>
<td>Construction costs</td>
<td>¥/kWe</td>
<td>200,000</td>
<td>220,000</td>
<td>300,000</td>
<td>8</td>
<td>7.4</td>
<td>8</td>
</tr>
<tr>
<td>O&amp;M costs</td>
<td>Annual O&amp;M costs are made to be corresponding to 5.3% of construction costs.</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Decommissioning costs</td>
<td>Decommissioning and decommissioning waste disposal costs are made to be corresponding to 20% of the construction costs.</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

C:Core  B:Blanket

2. Front-end component and back-end service unit prices

<table>
<thead>
<tr>
<th></th>
<th>Unit</th>
<th>EU-LWR</th>
<th>FM-LWR</th>
<th>RBWR-1</th>
<th>RBWR-2</th>
<th>FBR-1</th>
<th>FBR-2</th>
</tr>
</thead>
<tbody>
<tr>
<td>Uranium purchase</td>
<td>$/\text{tU}_3\text{O}_8</td>
<td>17.16</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Uranium purchase (high cost)</td>
<td>¥/kg · U</td>
<td>28,000</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Conversion to UF_6</td>
<td>$/kg · U</td>
<td>6</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Enrichment</td>
<td>$/kg · SWU</td>
<td>125</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Fabrication</td>
<td>MOX fuel</td>
<td>¥/kg · HM</td>
<td>-</td>
<td>200,000</td>
<td>280,000</td>
<td>230,000</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Radial blanket</td>
<td>¥/kg · HM</td>
<td>-</td>
<td>80,000</td>
<td>80,000</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Transport</td>
<td>MOX fuel</td>
<td>¥/kg · HM</td>
<td>-</td>
<td>200,000</td>
<td>200,000 or 390,000</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>Radial blanket</td>
<td>¥/kg · HM</td>
<td>-</td>
<td>200,000</td>
<td>200,000</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Reprocessing</td>
<td>VHLW disposal</td>
<td>¥/canister</td>
<td>76,000,000</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>Uranium credit</td>
<td>$/kg · U</td>
<td>0</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>Plutonium credit</td>
<td>¥/g · Pu</td>
<td>0</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>
Fig. 5.1.1 Comparison of levelised fuel cycle costs

Fig. 5.1.2 Comparison of levelised generating costs
5.2 Development of Database on Reprocessing Wastes of Future Power Reactors

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In order to study the evolution of future nuclear energy systems in Japan by using the JALTES model\(^1\), the database for spent fuel reprocessing wastes of future nuclear power plants was developed. The reactor types selected for the database are reduced moderation water reactors (RMWRs) of PWR type using heavy water as the primary coolant, sodium-cooled fast breeder reactors and present PWRs in comparison with the formers. RMWRs are expected either to take a complementary role to FBRs, or to substitute FBRs, since it would take long time to reduce high capital costs and to enhance maintainability of FBRs. The current development goal of RMWR is its early introduction into nuclear electric generation system with capability of plutonium breeding based on the present water reactor technology. Introduction of the first of the kind of RMWR is expected to be around the 2020s.

As a summary of the database on spent fuel reprocessing wastes of the future nuclear power plants, annual fuel discharge and annual arising of vitrified high level wastes (VHLW) are described in Table 5.2.2. It was assumed that for reprocessing of PWR fuels, the PUREX process was applied that removes almost all minor actinides (MAs) in spent fuel (SF) then transfers them to high level liquid wastes (HLLW). For reprocessing of the future reactor fuels, advanced reprocessing scheme was assumed that recycles almost all MAs into its fuel material. The transition rates of MAs and fission products (FPs) from SF to HLLW assumed for the preprocessing schemes are summarize in Table 5.2.1. These data are set on the safety side referring to the Rokkasho licensing documentations\(^2\). VHLW is produced by denitrification of HLLW and addition of grass constituents and is contained in stainless steel canisters. The quantity of FPs and MAs contained in VHLW is limited by two constraints, based on their decay heat levels and quantity of oxide. The decay heat levels decrease with time and the quantity of oxides remains constant with time as shown in the table. Therefore it is strongly recommended to wait vitrification of HLLW until the decay heat levels decrease to the level where the constraint by decay heat is comparable to the one by quantity of oxides. It will take about 4 to 5 years after reprocessing that decay heat reaches to that level in the case of future reactor fuel. Annual amount of fuel discharge of RMWRs tends to be larger than that of FBRs, on the contrary VHLW per ton of heavy metal for RMWRs is smaller than that for FBRs, as a result annual arising of VHLW of RMWRs (a product of annual fuel discharge and VHLW per ton of heavy metal) is nearly the same quantity with FBR.

As another summary of the database that characterizes their wastes, ALI (annual limit of intake) ingestion hazard index per ton of heavy metal is shown in Fig. 5.2.1. The curves of
the hazard index of FBRs and RMWRs in long term are very close to each other. The hazard index of the PWRs, especially in the case of MOX, is larger than that of future reactors because larger amount of MA is transferred to VHLW on reprocessing. MA is major contributors to the hazard index.

By using this database in future JALTES studies, the role of RMWRs in Japan will be clarified from the viewpoints of accumulation of radioactive wastes and costs of nuclear energy systems.

References

Fig. 5.2.1 ALI Ingestion radiotoxicity hazard index of VHLW
Table 5.2.1  Assumed Transition Rate of nuclides to HLLW on reprocessing

<table>
<thead>
<tr>
<th>Reprocessing Method</th>
<th>transition rate to HLLW (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>U</td>
</tr>
<tr>
<td>HDwoMA (conventional)</td>
<td>0.5</td>
</tr>
<tr>
<td>LDwiMA (advanced)</td>
<td>0.5</td>
</tr>
</tbody>
</table>

Note) HDwoMA : Reprocessing with High Decontamination factor and without Minor Actinide recycling
LDwiMA : Reprocessing with Low Decontamination factor and with Minor Actinide recycling

Table 5.2.2  Arisings of VHLW from Reprocessing of Reactor Fuel

<table>
<thead>
<tr>
<th>Reactor</th>
<th>PWR</th>
<th>PWR</th>
<th>RMWR*</th>
<th>RMWR**</th>
<th>FBR</th>
<th>FBR</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fuel Type</td>
<td>UO2</td>
<td>MOX</td>
<td>MOX</td>
<td>MOX</td>
<td>MOX</td>
<td>MOX</td>
</tr>
<tr>
<td>Discharge Burnup (core average) (GWd/ thM)</td>
<td>50</td>
<td>50</td>
<td>50*</td>
<td>70*</td>
<td>100</td>
<td>150</td>
</tr>
<tr>
<td>Discharge Burnup (stuck average) (GWd/ thM)</td>
<td>50</td>
<td>50</td>
<td>26*</td>
<td>37*</td>
<td>60*</td>
<td>90*</td>
</tr>
<tr>
<td>Annual Fuel Discharge (thM/y/1GWes)</td>
<td>19</td>
<td>19</td>
<td>26*</td>
<td>19*</td>
<td>12*</td>
<td>8*</td>
</tr>
<tr>
<td>Reprocessing Method</td>
<td>HDwoMA</td>
<td>HDwoMA</td>
<td>LDwiMA</td>
<td>LDwiMA</td>
<td>LDwiMA</td>
<td></td>
</tr>
<tr>
<td>Decay Heat</td>
<td>after 2yrs</td>
<td>8.93E+03(3.9)</td>
<td>1.07E+04(4.7)</td>
<td>1.73E+03(0.75)</td>
<td>2.01E+03(0.88)</td>
<td>6.76E+03(2.9)</td>
</tr>
<tr>
<td></td>
<td>after 4yrs</td>
<td>4.39E+03(1.9)</td>
<td>5.92E+03(2.6)</td>
<td>6.63E+02(0.29)</td>
<td>8.39E+02(0.37)</td>
<td>2.77E+03(1.2)</td>
</tr>
<tr>
<td></td>
<td>after 10yrs</td>
<td>2.25E+03(0.98)</td>
<td>4.17E+03(1.8)</td>
<td>2.77E+02(0.12)</td>
<td>3.82E+02(0.17)</td>
<td>1.17E+03(0.51)</td>
</tr>
<tr>
<td>Oxide contents after 2−10yr (g/thM)</td>
<td>6.94E+04 [1.1]</td>
<td>6.83E+04 [1.1]</td>
<td>1.97E+04 [0.32]</td>
<td>2.61E+04 [0.42]</td>
<td>6.70E+04 [1.1]</td>
<td>9.79E+04 [1.6]</td>
</tr>
</tbody>
</table>

*\(a\) : Pressurized water reactor type using heavy water as the primary coolant.  *\(b\) : Average of core fuel and internal blanket.
*\(c\) : Average of core, internal blanket and axial blanket.  *\(d\) : Average of core and axial blanket.  *\(e\) : Excluding radial blanket.
*\(f\) : Time after discharge from a reactor.  *\(g\) : Time required decay heat decreases below the level that is assessed on oxide's constraint.

Notes: ( ) indicates a number of VHLWs assessed on decay heat that corresponds to the decay heat of ton of heavy metal divided by the decay heat limit of VHLW (decay heat ≤ 2.3kW). [-] indicates a number of VHLWs assessed on oxide quantity that corresponds to the oxide quantity contained in ton of heavy metal divided by the oxide contents limit in VHLW (oxides of FPs and actinides ≤ 15wt%).
5.3 Model Development to Analyse Economic Impacts of Nuclear Phase-out in Japan

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In order to analyse long-term macro-economic impacts of nuclear phase-out in Japan, an energy-economy interaction model is being developed. The model is made up by linking an energy system optimization model, MARKAL\textsuperscript{1)}, with a macro-economic analysis model\textsuperscript{2)} based on the turnpike theory. The outline of the linked model is shown in Fig.5.3.1. An economic part of the model is described especially focusing on the two aspects of the impacts by phase-out of nuclear energy; increases of fossil fuel imports which will increase net income transfer to foreign countries and increases of prices of goods and services which will enhance changes of an industry structure, i.e. shifting to non-energy intensive industries.

The outline of the macro-economy model is as follows:

(1) Objective function: Maximization of real consumption.

(To be changed to maximization of ‘real consumption – energy cost’ when linked with the energy model.)

(2) Framework: Since the energy model is of a linear optimization type, a turnpike economic model has been selected to describe the rest of economy in a consistent manner with the energy sector. The model can determine optimal economic growth paths, i.e. optimum levels of production, consumption, investment, etc. at each year during the planning time period that make real consumption maximum subject to given changes in technology progress and labor force supply in the future. The turnpike-type model has following features.

- It can provide optimum investment levels and sectoral production levels.
- Arbitrariness in selecting future economic targets can be removed since optimal growth paths are insensitive to future targets.
- Production function is of a Leontief type and prices are not included explicitly in the basic model, therefore effects of price changes should be appropriately added.

(3) Time period: 1995 – 2050

* Mitsubishi Research Institute
(4) Constraints: Two major constraints are given to the model, technology progress and labor force changes. The former is given in terms of the changes in input-output coefficients. An example is provided in the study by the Economic Planning Agency\(^2\). The latter one includes both changes in the number of labor force and in its productivity. Since the number of labor force will decrease after some time period in the future, the increase of productivity will have a key role in the future economic growth.

(5) Sector classification: The model consists of 24 sectors, i.e. 22 industry sectors, producers of government services, and producers of private non-profit services to households. When linking with the energy model, an appropriate interface will be developed in order to be compatible with sector classification in the energy model.

References


Fig. 5.3.1 Basic concept of linking the MARKAL energy model with the macro-economic model
5.4 Impacts of Nuclear Phase-out in Japan – Preliminary Analysis by the Energy Model

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An input database of the energy system optimization model MARKAL was developed in order to build an energy-economy interaction model to apply for the analysis of possible economic impacts by nuclear phase-out in Japan. The energy model consists of 26 energy sources, 30 technologies for electric power generation and/or district heating, 63 technologies for conversion, storage, and distribution of energy, 26 energy demand categories, and more than 100 end-use technologies. The time period was set tentatively as 1990-2030.

In advance to the analysis with the linked model, a preliminary analysis was made by using only the energy model. For this purpose, demand for energy service was assumed for each of demand categories by assuming economic growth rates of 2% per annum until the year 2010 and 1.6% per annum during 2010-2030. It was also assumed that recovery and disposal of CO₂ will be feasible from the year 2015, and can be implemented at the annual scale of 80 million ton of CO₂ in 2030.

Two scenarios were analyzed; a reference scenario and a phase-out scenario. In the reference scenario (A) nuclear power capacity was expanded from 45GWe in 2000 to 75GWe in 2020 and 80GWe in 2030. In the phase-out scenario (B) it was assumed that no nuclear investment will be made after 2000, and that existing plants will be decommissioned according to their lifetime of 30 years. As shown in Table 5.4.1, three analytical cases were defined for each of these two scenarios A and B; case 1 with minimum system cost, case 2 with CO₂ emission constraints, case 3 with minimum CO₂ emissions. In the case A2, penalty was given to the CO₂ emissions, while in the case B2 the amount of cumulative CO₂ emissions was constrained not to exceed that of A2.

Primary energy supply in the case A2 and B2 is shown in Fig. 5.4.1 and Fig. 5.4.2, respectively. In the case A2, nuclear energy accounts for 18% of total primary supply and 37% of electricity generation in 2030. On the other hand, natural gas and renewable energy are used substantially in the case B2 with the share of natural gas 40% in 2030. The CO₂ disposal option is used up to its upper limit in this case.

Annual emissions of CO₂ in the above 6 cases are compared in Fig. 5.4.3. With the emission constraints in the case B2, A2 and B2 have the same amount of cumulative emissions in the results. However, the annual emissions of B2 are controlled quite low after 2020 in order to compensate large amount of emissions during the time period of 2005-2020 when capacity of nuclear power generation is reduced rapidly. In addition, it is obvious from this figure that the amount of CO₂ emissions in the case B2 is very close to the limit of reduction when nuclear energy is not used, while further reduction can be made in the case A2 if higher penalty is given to the emissions of CO₂.

Finally, energy costs were compared between the case A2 and B2. As given in Table 5.4.2,
annual energy system costs in the case B2 are significantly larger than those in the case A2 by 4.1 trillion yen in 2020, and 8.7 trillion yen in 2030. If converted into electricity generation costs, they correspond to the increases of generation costs by 2.9 yen/kWh and 5.5 yen/kWh.

Although these additional costs will pull up the production costs of industries, they do not represent the net effect on gross domestic products (GDP). Table 5.4.2 also shows the decomposition of the total costs into fuel import costs and technology costs (construction costs and O&M costs of energy facilities and equipments). Additional fuel import costs imply the increase of income transfer to foreign countries, hence the loss of GDP, while most of additional technology costs will be paid to non-energy industries inside the country and will not affect GDP directly. Accordingly, in order to understand the economic implication of these additional costs, it is necessary to make analysis by using the energy-economy linked model.

Table 5.4.1 Scenarios and cases for the preliminary study

<table>
<thead>
<tr>
<th>Case Scenario</th>
<th>Minimum Cost</th>
<th>CO₂ Penalty**</th>
<th>CO₂ Minimum</th>
</tr>
</thead>
<tbody>
<tr>
<td>Reference</td>
<td>A1</td>
<td>A2</td>
<td>A3</td>
</tr>
<tr>
<td>Nuclear Phase-out*</td>
<td>B1</td>
<td>B2***</td>
<td>B3</td>
</tr>
</tbody>
</table>

* No investment after 2000. 30 years lifetime for existing plants.
** Penalty: 10,000 yen / tC in 2005, 25,000 yen / tC in 2030.
***Cumulative CO₂ emission of B2 is constrained lower than A2.

Table 5.4.2 Costs of nuclear phase-out in Japan

<table>
<thead>
<tr>
<th>Year</th>
<th>2020</th>
<th>2030</th>
</tr>
</thead>
<tbody>
<tr>
<td>Total Cost (X) (per kWh)</td>
<td>4.1 TYen (2.9 Yen)</td>
<td>8.7 TYen (5.5 Yen)</td>
</tr>
<tr>
<td>Fuel Import Cost (Y)</td>
<td>1.4 TYen</td>
<td>2.8 TYen</td>
</tr>
<tr>
<td>Technology Cost (X-Y)</td>
<td>2.7 TYen</td>
<td>5.9 TYen</td>
</tr>
</tbody>
</table>

(Increases of costs in B2 as compared with A2.)
Fig. 5.4.1  Primary energy supply in the case A2

Fig. 5.4.2  Primary energy supply in the case B2

Fig. 5.4.3  Annual emissions of carbon dioxide in each analytical case
5.5 An Investigation of Experience Curves of Solar Photovoltaic Power Systems

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In order to assess the role of new energy technologies, studies have been made so far in JAERI on Japan's long-term energy systems by using the system optimization model MARKAL. Since the model is of a linear optimization type, characteristic data of energy technologies in the model have been specified as fixed exogenous assumptions in these studies. However, it is known that in the actual world technology progress is dynamical, i.e. reduction of technology costs depends on experiences rather than time. Therefore, in order to provide more realistic competition of technologies, it is suggested to include experience curves of technologies in the model together with the utilization of non-linear optimization techniques. Since decentralized power technologies will most enjoy benefit of experiences, solar photovoltaic power (PV) systems have been selected for a preliminary investigation.

According to the data by New Energy Sub-committee, Advisory Committee for Natural Resources and Energy, Ministry of Economy, Trade and Industry\(^1\), experience curves of residential PV systems and non-residential systems are as shown in Fig. 5.5.1. The progress ratio (PR) of learning for the residential systems is 0.89, and that for the non-residential systems is 0.58, much lower than the former. (Note: PR is defined as the rate of price reduction at each doubling of cumulative installed capacity.)

The composition of the prices of the PV systems has been investigated in order to make better comparison of the above two systems. The prices of the PV systems decomposed to main components are shown in Fig. 5.5.2 and Fig. 5.5.3. Fig. 5.5.2 shows that the price of residential systems has lowered due to the decrease in the prices of other components than a PV module\(^2,3,4\). As a result the ratio of the module price to the total system price has increased to 68% at the cumulative capacity of 210MW. Fig. 5.5.3 shows the prices of the PV systems for public facilities averaged for 10 - 30kW units\(^3,4\). The PR of the total system is 0.52, close to the value for the average of the non-residential systems. A similar trend is observed as the residential systems, although relative prices of inverters and ancillary equipments are higher. The ratio of the module price to the total system price increased to 40% at the cumulative capacity of 4.9MW. These data show that the module price will play an important roll in future prices of the PV systems.

Fig. 5.5.4 shows the experience curves of the module price plotted by referring to the paper by Tsuchiya\(^5\) and the report by the New Energy and Industrial Technology Development Organization (NEDO)\(^3\). As shown in this figure progress ratio of the module
is 0.82, same as the reported PR value in the paper by United States Department of Energy (USDOE). The major PV module in Japan has shifted from an amorphous type in 1990 to a crystalline type in 1998, while crystalline silicon has been dominant in the PV module market in the United States. Since Japan and United States have experienced same progress despite the difference of module types in both countries, the above PR value would be a useful indicator for estimating future price reduction of the PV module even though dominant module types might be changed with time.

![Experience curves of mean price of PV system price](image)

**Fig. 5.5.1** Experience curves of mean price of PV system price

![Experience curves of component price of residential use PV system](image)

**Fig. 5.5.2** Experience curves of component price of residential use PV system (case for standard 3kw PV system)
Fig. 5.5.3 Experience curve of component price of PV system installed under NEDO field test program for public facilities

Fig. 5.5.4 Experience curve of PV module price in Japan

References
2) Home page of New Energy Foundation(http://www.solar.nef.or.jp/josei/kw_bunpu.htm).
6. Reactor Structural Materials

Research items carried out were classified into two categories, i.e., irradiation assisted stress corrosion cracking (IASCC) study, and development of material performance database.

In the field of IASCC study, development of in-situ observation technique during SCC test was initiated aiming at evaluation of the SCC behavior isolating the crack initiation stage from the crack growth stage for irradiated stainless steel. In-pile crack initiation detection technique, in-pile crack length measurement techniques, in-pile strain measurement techniques, etc. also have been tried to be developed aiming at performing in-pile IASCC tests using JMTR.

As a part of ITER (International Thermonuclear Experimental Reactor) task, IASCC behavior for type 316LN-IG (ITER grade) stainless steel specimens in both conditions of solution-annealed and joined by solid hot-isostatic-pressing procedure was also examined. As the results of a series of SSRT tests and SEM examinations, it was confirmed that IASCC susceptibility of 316LN-IG was quite low in ITER operating condition (temperature; 423 K) and ITER vacuum vessel baking condition (513 K) after neutron irradiation at 473 K to 1 dpa.

In the field of material performance database, JAERI material performance database (JMPD) and the distributed database named "Data-Free-Way" have been developed. Based on the experience of development of JMPD, Data-Free-Way was constructed under the collaboration of JAERI, the National Research Institute for Metals, the Japan Nuclear Cycle Development Institute and the Japan Science and Technology Corporation in order to share fresh and stimulating information as well as accumulated information for the development of advanced materials, for the design of structural components, etc.

The JMPD system was improved by adding the tables where details of neutron irradiation induced stress relaxation data can be described, and by improving the easiness of the output item selection. As an attempt concerning the knowledge base construction, the format of describing the quantitative finding extracted from the data base was examined, and two dimensional graph book was prepared.
6.1 Development of In-situ Observation Technique during SCC Test in High Temperature Water for Irradiated Materials

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As one of the analytic techniques of stress corrosion cracking (SCC) behavior, in-situ observations of specimen during slow strain rate tensile (SSRT) test are reported\(^{1,2}\). In most cases, however, unirradiated materials were observed as the specimens. From a view point of the evaluation of irradiation assisted stress corrosion cracking (IASCC) behavior, which is one of the key issues for the life management of light water reactor (LWR) core components, in-situ observation technique for irradiated material must be developed. Hence development of in-situ observation technique during SSRT test was initiated aiming at evaluation of the IASCC behavior isolating the crack initiation stage from the crack growth stage for irradiated stainless steel.

A SSRT testing apparatus for irradiated materials was developed and set up in a hot cell of Waste Safety Testing Facility (WASTEF). Schematic flow diagram of the apparatus is shown in Fig. 6.1.1. The apparatus consists of a water make-up section, a pressurized high temperature water circulation system, a tensile testing system with an autoclave, a monitoring/purification system, a hydrogen peroxide injection system, and an observation/recording system. The autoclave has a sapphire window for in-situ observation of the specimen. Dissolved oxygen and hydrogen in the circulation water are controlled in the water treatment system. In addition, it is possible to inject hydrogen peroxide into the circulation water during SSRT test. Main specification of the apparatus is listed in Table 6.1.1.

At present, a trial run of the apparatus is being made, and the observation/recording system has been improved for the usage in the hot cell. The in-situ observation technique during SCC test in high temperature water for irradiated materials will be completed in the next fiscal year.
Reference


<table>
<thead>
<tr>
<th>Autoclave capacity</th>
<th>1.77 ℓ</th>
</tr>
</thead>
<tbody>
<tr>
<td>Autoclave material</td>
<td>Type 316L stainless steel</td>
</tr>
<tr>
<td>Control range</td>
<td></td>
</tr>
<tr>
<td>Load</td>
<td>Max. 20 kN</td>
</tr>
<tr>
<td>Temperature</td>
<td>RT-573 K</td>
</tr>
<tr>
<td>Pressure</td>
<td>Max. 10 MPa</td>
</tr>
<tr>
<td>Flow rate</td>
<td>Max. 30 ℓ /h</td>
</tr>
<tr>
<td>Dissolved oxygen (DO)</td>
<td>10ppb - 32ppm</td>
</tr>
<tr>
<td>Dissolved hydrogen (DH)</td>
<td>Max. 2.8ppm</td>
</tr>
</tbody>
</table>
Fig. 6.1.1 Schematic flow diagram of environmentally assisted cracking test apparatus for irradiated materials in high temperature water
(A) Water make up section, (B) Pressurized high temperature water circulation system, (C) Tensile testing system, (D) Monitoring/purification system, (E) Hydrogen peroxide injection system and (F) Observation/recording system.
6.2 Feasibility Study for In-pile SCC test using Bent-Beam Method in High Temperature Water

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Irradiation assisted stress corrosion cracking (IASCC) is caused by the synergistic effects of irradiation, stress and corrosion by high temperature water. It is, therefore, essential to perform in-pile SCC tests, which are material tests under the conditions simulating those of actual LWR operation, in order to clarify the precise mechanism of the phenomenon, though mainly out-of-pile SCC tests for irradiated materials have been carried out in this research field. There are, however, many difficulties to perform in-pile SCC tests. Performing in-pile SCC tests, essential techniques must be developed. Hence as a part of development of the essential techniques for in-pile SCC tests, four-point bent beam in-pile SCC tests were performed on sensitized austenitic stainless steels in an irradiation capsule using Japan Materials Testing Reactor (JMTR). Four-point bent beam SCC test technique was adopted among some SCC test methods for the following reasons in the present study.

1. Relatively many test units can be installed in the limited test space in the case of bent beam test method.
2. Independent load or strain level can be applied for each specimen in the case of bent beam test method.
3. Initial load or displacement can be applied at a constant value in longer portion in the four-point bent beam test method than that in the three-point one.

Figure 6.2.1 shows the concept of irradiation capsule for in-pile SCC test at JMTR. Seven kinds of type 304 stainless steels were irradiated. Pressurized high temperature water flows downwards through the outside of the inner tube within the capsule from the upper plug, and it is heated with the bottom heater and flows out while contacting the specimens. Specimens were inserted at seven portions as shown in Fig. 6.2.1. At each portion, three loaded and three unloaded specimens were positioned. A thermocouple (TC) was inserted at each portion to measure the temperature of water very close to the specimen surface. At JMTR, an irradiation technique named the saturation temperature capsule (SATCAP) was developed for the irradiation of materials in high temperature water. In this study, the capsule for in-pile SCC test was connected to this saturation temperature facility, in which pure water of 7.2MPa was fed into the capsule at 600cc/min. As shown in Fig. 6.2.1 a good agreement is
observed between the measured and calculated temperature values. The water conditions such as electrical conductivity, pH and dissolved oxygen were measured at the inlet and the outlet of the capsule. The capsule was loaded into the fuel region of the JMTR core, and it was irradiated for 600 hours, which is one cycle of JMTR operation. In this in-pile test, the fast neutron fluence of the specimen at peak flux position was $3 \times 10^{24} \text{n/m}^2 (E>1\text{MeV})$.

Configuration of the specimen and the holder are shown in Fig. 6.2.2. A Penetration hole with a diameter of 1 mm was drilled at the center of some specimens. A stress concentration factor at the circumference of the penetration hole is 2.2. The fixed bending stress of the specimen was set up to the stress equivalent to 90% or 110% of the yield stress level that was evaluated by the tensile test for each material at room temperature.

To detect crack initiation during in-pile test, electric resistances on three specimens were monitored using MI cables. Changes in electric resistances between terminals attached on specimens are shown in Fig. 6.2.3. At 270 hours, the electric resistances measured on the specimen connecting with No.12 terminal increased gradually. It is considered that the wire or terminal of No12 cable failed or could not keep contact with the specimen and the potential difference measured on the specimen using MI11-14 was caused by the failure of No12 cable.

Specimens and hermetic plugs of MI cables were examined using an optical microscope and SEM after the test. The typical SEM photo is shown in Fig. 6.2.4 which is the portion around penetration hole. No crack was found in any specimens.

To confirm the integrity of the hermetic plug of MI cable, the plugs were examined with SEM. A junction part of the hermetic plug of MI2 that seemed to be failed during the irradiation is shown in Fig. 6.2.5. The thickness of Ni coating at this part was 20 μm in this plug. At the end of ceramic tube of hermetic plug, Ni coating showed cracking. During the in-pile test, the radiation dose from water increased after 430 hours. The reason of this increasing is considered that the silver on the plug dissolved out from the brazing parts into high temperature water and it was activated in core. On the other hand, the hermetic plug coated with Ni in thickness of 200 μm showed better durability.

References

Fig. 6.2.1 Concept of the capsule for in-pile SCC test at JMTR

Fig. 6.2.2 Configuration of specimen and holder

Fig. 6.2.3 Changes of electric resistances between terminals attached on specimens

Fig. 6.2.4 SEM observation micrograph of the specimen with a penetration hole

Fig. 6.2.5 SEM observation of the junction part of the hermetic plug on MI2
6.3 Development of In-pile Uniaxial Constant Loading SCC Test Technique in High Temperature Water

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Irradiation assisted stress corrosion cracking (IASCC) is caused by the synergistic effects of irradiation, stress and corrosion by high temperature water. It is, therefore, essential to perform in-pile SCC tests, which are material tests under the conditions simulating those of actual LWR operation, in order to clarify the precise mechanism of the phenomenon, though mainly out-of-pile SCC tests for irradiated materials have been carried out in this research field. There are, however, many difficulties to perform in-pile SCC tests. Performing in-pile SCC tests, essential techniques must be developed. Hence as a part of development of the essential techniques for in-pile SCC tests, we have embarked on development of the test technique which enables us to obtain the information concerning the effect of such parameters as applied stress level, water chemistry, irradiation conditions, etc. on the crack initiation behavior. Although it is difficult to detect the crack initiation in in-pile SCC tests, the crack initiation can be evaluated by the detection of specimen rupture if the cross section area of the specimen is small enough. Therefore, we adopted the uniaxial constant loading (UCL) test with small tensile specimens. The concept of UCL test unit is shown Fig. 6.3.1. Pressurized high temperature water flows outside of this unit. Loading level is controlled by controlling the pressure difference between the internal helium gas of the bellows and the external water. When the internal gas pressure of the bellows is less than the external water pressure, the specimen is pulled with bellows' contracting in the axial direction. At the inside of the bellows, there are contact points with some gap. When the specimen is ruptured, the bellows contracts over the gap and the points keep in touch. The detection concept of the specimen rupture is shown in Fig. 6.3.2. The electric resistance between these contact points is remotely monitored. The specimen rupture will be indicated as the reduction of electric resistance. In this year, some prototype test units were designed and a relationship data between gas pressure and deflection of bellows was obtained. Figure 6.3.3 shows the relationship using a spring (K=1kg/cm) instead of test specimen at RT. A good linear relationship between gas pressure and deflection of bellows was obtained.
Fig. 6.3.1 Concept of UCL test unit

Fig. 6.3.2 Detection concept of specimen rupture

Fig. 6.3.3 Relationship between gas pressure and deflection of bellows with a spring (K=1kg/cm)

\[ y = 0.8041x + 0.048 \]
6.4 Development of Crack Monitoring Technique in High Temperature Water

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As comparison between direct current potential drop (DCPD) and alternative current potential drop (ACPD) methods, the DCPD method is determined to adopt in the in-pile crack growth tests, because the ACPD method is greatly affected by the position of measurement cables relative to the DCPD method and large current can be applied in the DCPD method.

In order to evaluate the influence of the several noises on the monitoring of the crack length and long time stability in the measurement by the DCPD method, crack growth test was performed in the high temperature water for the trial crack-monitoring test. General view of CT specimen used in this study is shown in Fig. 6.4.1. In this test, four MI cables plugged between the core and the sheath tube were used and spot-welded to the top of the CT specimen as shown in Fig. 6.4.1. Moreover the sheath tubes of four MI cables were grounded in the position that is 100mm far from the top of the CT specimen. Material of the CT specimen used in this study is type 403 stainless steel.

Experimental conditions are as follows: 17.6 – 22 MPa m$^{1/2}$ in stress intensity factor, 288°C in test temperature, 10MPa in water pressure, 8ppm in dissolved oxygen and 0.15μS/cm in outlet electric conductivity. The measurement procedure of the DCPD method is four terminals method and current is 1A in this test. Fig. 6.4.2 shows relationship between potential drop of DCPD method, crack length and experimental time. Although the crack growth test was conducted for about 700 hours, it was found that the level of the noise was

Fig. 6.4.1 General view of CT specimen attached four MI cables
not large and long time stability in the measurement was enough for the monitoring of the crack length by the DCPD method. Potential drop stably increased with increasing of the crack length in this test. After the test, the crack tip of the CT specimen was examined with the optical microscope. About 0.15mm crack was observed on this CT specimen surface. Therefore it is confirmed that the DCPD method is applicable to the in-pile crack growth test for long time.

Since the length of MI cables used in the DCPD method is about 20m and the temperature distribution is found in the CT specimen in the in-pile crack growth test, the applied current might be very small in the DCPD method and the temperature compensation might be necessary for the CT specimen. Therefore we will adopt the six terminals DCPD method, because it is easy to compensate the temperature of the specimen and two terminals might be used as the back up of the DCPD measurement.

Future work will be necessary to conduct the crack growth test by the six terminals DCPD method and to evaluate the sensitivity of potential drop for small current to the crack length.

Reference:

Fig. 6.4.2 Relation between potential drop and crack length in DCPD method
6.5 High temperature water loop facility for the functional tests of in-pile IASCC test equipments under high flow rate

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Irradiation assisted stress corrosion cracking (IASCC) can be reproduced on alloys irradiated over the threshold neutron fluence level by post irradiation examinations (PIEs). However, it is considered that IASCC reproduced by PIEs on the irradiated alloys must be carefully distinguished from IASCC that occurs in the core under simultaneous effects of neutron radiation, stress and high temperature water environment. From this point of view, the in-pile material testing is regarded as a critical experiment for a study of IASCC, though there are various technical hurdles to attain the in-pile IASCC testing. Under a program to develop the testing techniques, a high temperature water loop facility was designed and installed at Tokai Research Establishment mainly for the functional tests of in-pile testing units that will be fabricated into the irradiation capsule.

The loop facility has been principally designed in order to simulate the environment in BWR, and especially it was designed to obtain a high flow rate in test sections because in the irradiation capsule the test equipment will be used in a high temperature and high flow rate water. The loop facility will be also utilized for the environmentally assisted cracking (EAC) test and corrosion test for in-core materials such as austenic stainless steels. In addition the specimens for in-pile testing will be stressed in the irradiation capsule with a loading system equipped metal bellows as described in section 6.3 of this report. The following requirements were, therefore, needed in the design of loop facility:

- increase flow rate to simulate high flow rate of coolant water in irradiation capsules,
- control dissolved oxygen/hydrogen concentrations to nominal values in BWR core,
- minimize pressure fluctuation due to pressurization by pumps.

Fig. 6.5.1 shows a flow diagram of the facility that consists of a water make-up section, a pressurization and heating section, a test section and a water purification section. Since the loop facility is designed to attain the BWR simulated condition, the maximum operational temperature and pressure are 300 °C and 10 MPa. In the water make-up section,
concentrations of dissolved oxygen (DO) and hydrogen (DH) are controlled by bubbling of oxygen, hydrogen and nitrogen gases in a water make-up tank. High purity water from the make-up section is pressurized by a high pressure pump. To minimize pressure fluctuation of supplied water, the three plungers type pump was employed and at the outlet piping from the pump three accumulator tanks were equipped to absorb the small pressure pulsation by pumping. Pressurized water is flowing with a flow rate of 30 litter/hour through the heat exchanger and then heated up with an electric pre-heater. At the test section, in two pressure tubes with an inner diameter of 49.5 mm and an effective length of about 700 mm, water temperature is controlled at 288 °C and the in-pile testing units are tested. A flow rate of water at the test section is increased by a non-seal pump that can increase a total flow rate up to 600 litter/hour in the test section. The water purification section consists of filters, ion exchangers and water chemistry monitoring equipments. DO, DH and pH of water at the outlet of the test section and at the water make-up section are monitored. Purified water is returned to the water make-up tank and circulated.

Fig. 6.5.1 Flow diagram of high temperature water loop facility.
6.6 SCC Susceptibility and Mechanical Properties of HIPed Joint of Type 316LN Stainless Steel for ITER Applications

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In the engineering design activity (EDA) for international thermonuclear experimental reactor (ITER), a module type structure has been proposed for the first wall/shielding blanket component. As a candidate material and a fabrication of the module, type 316LN stainless steel ITER Grade (316LN-IG SS) and the hot isostatic pressing (HIP) technique are being considered, respectively. However, it is reported that stress corrosion cracking (SCC) of 316 SS were initiated in corrosive environments\cite{1,2}, such as high temperature water including chloride. To evaluate the integrity and susceptibility to SCC of HIPed 316LN-IG SS, tensile tests in vacuum and slow strain rate tests (SSRT) in high temperature water were performed. Since an initiation of SCC would be accelerated by introducing an artificial crevice to the specimen\cite{3}, SSRT with a crevice condition were performed to examine the effects of crevice on SCC behavior.

Two plates of solution-annealed 316LN-IG SS were mechanically polished, and pressed isostatically for 2 hours in argon gas at 1323 K under pressure of 150 MPa. For tensile tests and SSRT, sheet type specimens with 7.62 mm in gage length, 1.52 mm in width and 0.76 mm in thickness, and round bar type specimens with 24 mm in gage length and 4 mm in diameter were machined from the HIPed plates and an as-received plate. Hereafter, the former and the later are called the HIPed joint and the as-received specimen, respectively. Tensile properties of the HIPed joint and the as-received specimens were examined in vacuum at room temperature to 573 K. SSRT and creviced SSRT were performed at 423 K and 513 K in oxygenated high temperature water. The applied strain rates were $1 \times 10^{-6}$ s$^{-1}$ and $2-10 \times 10^{-7}$ s$^{-1}$, respectively. To simulate a crevice condition, the specimen was wound by graphite-fiber wool and covered with 316 SS. After the HIPing process and these tests, metallographic observations were carried out.

The optical micrographs of the HIPed joint and the as-received specimens after the HIPing process are shown in Fig. 6.6.1. There is no defect and imperfect joint at the HIPed interface. Since the grain size of HIPed specimen was equivalent to that of as-received
specimen, grain growth by the HIPing process did not occur. The ultimate tensile strength (UTS) from this study and from the literature are shown in Fig. 6.6.2. In the temperature range of this study, UTS decreased with increasing the temperature. For the HIPed joint specimens, fracture initiated at the region except the HIPed interface, and the fracture surfaces showed only ductile feature, although the inclusions, i.e. Al₂O₃ and SiO₂, were observed at the HIPed interface. SSRT of the HIPed joint specimen performed in oxygenated 423 K water showed no SCC, although M₇₃C₆ type precipitates were observed at the interface. The stress-strain behavior after SSRT was equivalent to that after tensile test. Therefore, it is concluded that the HIPing process did not affect the tensile properties and SCC susceptibility in 423 K water. The maximum stresses after SSRT were equivalent to the UTS from tensile tests although specimens for SSRT under creviced condition were thermally sensitized at 1033 K for 100 hours. As seen in Fig. 6.6.3 (a), a lot of TG cracks were observed on the region wound by graphite-fiber wool and it suggested that the initiation of SCC was accelerated under the crevice condition. Fig. 6.6.3 (b) indicates that fracture did not occur at the HIP interface. Although it is well known that IGSCC occurs easily for the thermally sensitized stainless steel in oxygenated water, no IGSCC was observed after the creviced SSRT in 513 K water. In addition it is noted that IG cracking was not initiated at the tip of TG cracks, i.e. natural crevice. Since no IGSCC was observed in any specimens, it is concluded that initiation of IGSCC under the crevice condition was difficult for 316LN-IG SS even after HIPing process.

Reference:
Fig. 6.6.1 Optical micrograph of the HIPed joint and as-received specimen: (a) the HIPed joint specimen and (b) as-received specimen

Fig. 6.6.2 UTS of the HIPed joint and as-received 316LN-IG SSs

Fig. 6.6.3 Thermally sensitized HIPed joint specimen after creviced SSRT: (a) SEM photograph of TG cracks on a region wound by graphite-fiber wool and (b) Position of the HIPed interface
6.7 IASCC susceptibility of HIPed 316LN-IG irradiated at 473 K to 1 dpa

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Irradiation assisted stress corrosion cracking (IASCC) susceptibility of type 316LN-IG stainless steel was examined using a slow strain tensile test (SSRT) after neutron irradiation. The solution-annealed (SA) 316LN-IG plates were joined by solid hot-isostatic-pressing (HIPing) procedure. The detail of HIPed materials was presented in section 6.6 of this report1). The sheet type tensile specimens, 7.86 mm long x 1.52 mm wide x 0.76 mm thick in gage section, were cut out from the SA plate and HIPed interface. Hereafter, the former and the later are called SA and HIPed specimen, respectively. Microstructures of the specimens were examined using optical microscope, scanning electron microscope (SEM) and transmission electron microscope (TEM). The specimens were irradiated in High Flux Isotope Reactor (HFIR) at Oak Ridge National Laboratory. The estimated irradiation temperature is about 473 K. The calculated fast and thermal neutron fluences were 1.6x10^{25} n/m^2 (E>0.1MeV) and 2.4x10^{25} (E<0.5eV), respectively. The corresponding dose is about 1 dpa. After the irradiation, tensile tests in vacuum and slow strain rate test (SSRT) in oxygenated (dissolved oxygen, DO = 10 wt.ppm) water were performed at 423 and 513 K with strain rates of 2-10x10^{-7} s^{-1}. The specimens were examined with SEM after the tests.

Figs. 6.7.1(a) and (b) show the engineering stress-strain curves of irradiated specimens after tensile tests in vacuum. The curves of unirradiated specimens are also indicated in the figures. After irradiation, 0.2% proof stress and ultimate tensile strength increased and uniform strain and total strain decreased. The curves of HIPed specimens were identical with those of SA specimens. Fracture did not occur at HIPed interface, as seen in Fig. 6.7.2. All the specimens failed in a ductile manner. The same results were observed on unirradiated HIPed and SA specimens. Therefore, effect of HIPing process on tensile properties was negligible after the irradiation.

Figs. 6.7.3(a) and (b) show the stress-strain curves after SSRT of irradiated SA and HIPed specimens at 423 and 513 K, respectively. The curves after tensile tests are also indicated in the figures. The stress-strain behavior after SSRT in 423 and 523 K water were identical with those after tensile tests in vacuum. The difference in the behavior was not observed between SA and HIPed specimens. Fig.6.7.4 shows the fracture surface and fracture portion of HIPed specimen after SSRT in 513 K water. The only dimple pattern was observed on the surface. All the specimens failed in a ductile manner, and did not show any IASCC susceptibility. Fracture also occurred at the region except HIPed interface. There is no
difference in the SSRT behavior between SA and HIPed specimens. From these results, it was confirmed that IASCC susceptibility of HIPed 316LN-IG was quite low in ITER operating condition (temperature; 423 K) and ITER vacuum vessel baking condition (513 K).

Okada et al.\textsuperscript{21} reported that irradiation promoted the Cr depletion at grain boundaries even in slightly sensitized materials. It is known that Cr depletion increases the IASCC susceptibility\textsuperscript{3}. At the HIPed interface, M\textsubscript{23}C\textsubscript{6} type precipitates existed before irradiation\textsuperscript{1}, although the density of the precipitates was too low to thermally sensitize the HIPed materials. If nucleus of the precipitates was formed during the HIPing process, they may grow and attribute the sensitization due to the Cr depletion around them during irradiation. However, HIPed specimen did not show the IASCC susceptibility in 423 and 513 K water. Therefore it is concluded from these results that in 423 and 513 K water, HIPing process did not affect IASCC susceptibility after neutron irradiation at 473 K to 1 dpa.

Reference:
1) Nakano, J., et al., in this report.

![Graphs](image-url)

Fig. 6.7.1 Engineering stress-strain curves of irradiated specimens after tensile test.
Fig. 6.7.2  Fractured portion of HIPed specimen after tensile test at 513 K.

(a) At 423 K  
(b) At 513 K  
Fig. 6.7.3  Engineering stress-strain curves of irradiated specimens after SSRT.

Fig. 6.7.4  Fracture surface and fractured portion of HIPed specimen after SSRT in 513 K water.
6.8 Development of Analytical Method for Microstructure Observation of Oxide Film on Stainless Steel

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Irradiation assisted stress corrosion cracking (IASCC) of austenitic stainless steel has been studied as main concern of an aging problem of light water reactor (LWR) materials. In former studies of stainless steel, oxide film, especially passive film consists of Cr oxide is considered to retain its anticorrosive property. Thus thickness, microstructure and chemical composition of oxide film should be studied to understand IASCC phenomenon. However, there had been no applicable fabrication method to obtain cross section of oxide film as a thin film specimen for transmission electron microscope (TEM) observation. In former study, it was tried with ion milling but not successful\textsuperscript{11}. The aim of this work is to develop fabrication method for microstructure observation of oxide film. And not only microstructure observation but also chemical composition analysis is aimed to figure out the structure of oxide film.

Solution annealed type 304 stainless steel (type 304 SS) and Si doped type 304 stainless steel (Si doped SS) had been chosen as specimen materials. Chemical compositions are given in Tables 6.8.1 and 6.8.2, respectively. The shape of the specimen used in this study is a disk with 3mm in diameter and 0.3mm in thickness. And the surface of the specimen was polished to make it flat and fine. These specimens were exposed to pressurized high temperature (85kgf/cm\textsuperscript{2}, 288\textdegree C) water for 24 h for oxide film formation. Dissolved oxygen (DO) content was controlled as 32ppm (saturation content). After the exposure test to high temperature water, Ni for type 304 SS, and Cu for Si doped SS specimens were plated, respectively, to protect the surface of oxygen film while fabrication of observation specimens. To observe cross-section of oxide film using field emission gun scanning electron microscope (FE-SEM), the specimens were cut and the cross-section were polished. After FE-SEM observation, the thin film specimen for TEM observation and chemical analysis using energy dispersed X-ray spectrometer (EDS) was fabricated with focused ion beam (FIB) using spattering with Ga\textsuperscript{+} ion beam. Fabrication procedure is shown in Fig. 6.8.1. First, small piece of specimen (about 10\mu m x 10\mu m) was picked up. In this case it would include the cross section of oxide film. And, bond it on mesh sheet with W deposition, and fabricate it to a thin film with FIB.

Figure 6.8.2 shows the cross-sectional view of the oxide film formed on type 304
SS and Si doped SS specimens with FE-SEM. Oxide film was formed with lump shaped precipitates, and the thickness was 500nm for type 304 SS, and 200nm for Si doped SS specimens, respectively. However passive film was not figured out from these observations.

Figure 6.8.3 shows the cross-sectional view of the oxide film formed on type 304 SS with FE-TEM. It was revealed that oxide film was formed with about 100nm diameter lump shaped precipitates, however passive film was not observed clearly. And EDS analysis was not successful because Pt, used as coating before plating, and Ni, used as plating, disturbed the analysis of the other elements. Thus, for Si doped SS specimen, Cu coating and Cu plating were used, and EDS analysis was successful. Figure 6.8.4 (a) shows the cross-sectional view of the oxide film formed on Si doped SS specimen with FE-TEM. From the result of EDS analysis of the oxide film, the structure of the oxide film is considered as drawn in Fig. 6.8.4(b). Passive film consisting of Cr oxide was formed on the alloy, and Fe oxide precipitates was formed on it, and the surface of them was covered with Si oxide film.

In this work, fabrication of specimen for FE-TEM examination was successful with FIB. And the microstructure observation and the chemical analysis were conducted to figure out the structure of the oxide film on stainless steel. In further study of IASCC, the method developed in this work will be useful.

Reference:

Table 6.8.1 Chemical composition of type 304 stainless steel (mass %)

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Table 6.8.2 Chemical composition of Si doped type 304 stainless steel (mass %)

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Fig. 6.8.1 Thin film specimen fabrication procedure.
Fig. 6.8.2 Cross-sectional view of oxide film formed on specimens with FE-SEM.

Fig. 6.8.3 Cross-sectional view of oxide film on type 304 stainless steel with FE-TEM.

Fig. 6.8.4 (a) Cross-sectional view of oxide film on Si doped type 304 stainless steel with FE-TEM. (b) Structure of oxide film expected from the result of EDS analysis.
6.9 AFM Evaluation for Grain Boundary Corrosion Behavior of Ion Irradiated Stainless Steel

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Irradiation assisted stress corrosion cracking (IASCC) of austenitic stainless steel has been studied as main concern of an aging problem of light water reactor (LWR) materials. Corrosion behavior at the grain boundary for neutron irradiated materials should be studied to understand IASCC phenomenon. In former studies, to evaluate grain boundary corrosion behavior of sensitized materials, electrochemical potentiokinetic reactivation (EPR) method had been applied. It was, however, found that EPR method is not applicable for irradiated material, because irradiated material would contain some precipitates and irradiation damages in the grains, thus not only grain boundary portion but also inside of the grain would be corroded during the test. The aim of this work is to develop the experimental method to evaluate the grain boundary corrosion behavior of irradiated materials using atomic force microscope (AFM).

High purity Fe-18Cr-12Ni alloy was used in the present study. The chemical composition of the alloy is shown in Table 6.9.1. Specimens were solution annealed at 1050 °C for 30 min, and sensitized at 600 °C for 5 min. Ni ion was irradiated for the solution annealed specimen with 12MeV Ni$^{3+}$ ion up to a fluence level of 3.8x10$^{19}$ ion/m$^2$ at 573K using tandem accelerator at Takasaki ion accelerators for advanced radiation application (TIARA). The peak damage level was estimated to be 1 dpa. The shape of the specimen used in this study is a disk with about 3 mm in diameter and 0.3 mm in thickness. And the surface of specimen was polished to flatten, and finally electrochemically polished to get the flat and fine surface. And as shown in Fig. 6.9.1, the surface portion of 1 μm was removed from the irradiated specimen. Thus the peak region of irradiation damage was revealed on the surface of specimen.

For the AFM measurement of grain boundary corrosion, it is desirable that the surface of the specimen is corroded as shown in Fig. 6.9.2. That means most surface is remained flat, and only grain boundary is corroded a little. Aiming at obtaining the corroded specimen mentioned above, potentiostatic corrosion test was performed at 150mV, in 1N H$_2$SO$_4$ + 0.01mol/l KSCN solution at 30 °C. On this potential, most of the surface is expected to remain flat and only grain boundary would be corroded. Because the passive film would be formed and protect the surface except on grain boundary where there is Cr
depletion zone after irradiation or thermal sensitization.

Figure 6.9.3 shows the results of AFM observation after 1000 s potentionstatic corrosion test for (a) solution annealed, (b) thermally sensitized, and (c) ion irradiated specimens. Solution annealed specimen was not corroded at all. On thermally sensitized specimen, only grain boundaries were corroded. And on ion irradiated specimen, grain boundaries and some part of the surface except on grain boundary were corroded. The results shows that 1000 s is too long for irradiated specimen. Thus 500 s potentionstatic corrosion test was conducted for ion irradiated specimen as the second run. Figure 6.9.4 shows the result of the second run. Most surface was remained flat, and the measurement of grain boundary corrosion depth and width were successful. From the result of line analysis by AFM, the depth and the width of grain boundary corroded portion was estimated to be 70 nm and about 500 nm, respectively. In former studies, Cr depletion width along grain boundary was estimated to be a few nm\(^1\). The result obtained in the present study is much wider than that in the former studies. This suggests that not only Cr depletion area but also surrounding area was corroded in the present study. Further trials will be run for the optimization of the corrosion test conditions which are corrosion time, corrosion solution etc.

Reference:

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Table 6.9.1 Chemical composition of specimen (mass %)

Fig. 6.9.1 Specimen fabrication after ion irradiation.

Fig. 6.9.2 Ideal surface for AFM measurement
Fig. 6.9.3 AFM observation results after potentiostatic corrosion test for 1000 s.

Fig. 6.9.4 AFM observation result after potentiostatic corrosion test for 500 s for ion irradiated specimen
6.10 Present Status and Improvement of Function of JAERI Material Performance Database (JMPD)

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Fundamental studies of structural materials have been performed at Japan Atomic Energy Research Institute (JAERI) regarding practical applications for nuclear plants. For the evaluation of reliability and safety of structural materials, various kinds of material data, which are fatigue crack growth, creep, tensile, low-cycle fatigue, slow strain rate testing (SSRT) etc., have been accumulated through the above-mentioned research activity. The JAERI Material Performance Database (JMPD) was developed with a view to utilizing such material performance data efficiently\textsuperscript{1-3}. This paper will describe the interface of the JMPD which has been improved and the future direction of the JMPD.

Users can start the data search from any screens of reference, material, test type, description, or test condition which are prepared in the JMPD. Using the new interface, the searched results can be saved as the file. Users, therefore, can easily extract various kinds of graphic output using commercial graphic software. The data stored in the JMPD by the end of March 2001 are listed in Fig.6.10.1, in which the data from more than 11,600 test pieces are prepared for data evaluation.

Figure 6.10.2 shows the flow diagram of data retrieval in the JMPD. As for improvement of user interfaces, graphic function and modification function of the data were added in the JMPD. In Fig. 6.10.2, dark images are newly developed or modified interfaces.

After retrieval results were shown in the screen, users can easily extract various kinds of graphic output by new graphic function of the JMPD. An example of graphic output in the JMPD is shown in Fig. 6.10.3. This graph shows creep characteristics of Alloy 800H at high temperatures. The temperature dependence of creep strength for Alloy 800H is shown by different symbols in this figure. Symbols can be set up in the different screen for colors, geometries and paints. Therefore users can estimate several characteristics based on the factual data by using the graphic function in the JMPD.

Open-published data were also stored for several characteristics of structural materials in the JMPD and then there is possibility that overlapped data were stored in the JMPD. As for the other additional function, extraction and modification program for the
overlapped data was developed. Original data were not directly investigated and modified as considering the safety of database. Main menu of data extraction and modification screen in the JMPD is shown in Fig. 6.10.4. This screen is composed of menu bars, file names and contents of investigated data, and the list of identical data. In order to investigate the identical data for several characteristics of structural materials, the set-up screen of investigated terms and tolerance range for determination of identical data is added in this function. Finally users can delete and modify the data in this function.

As for the future direction of the JMPD, the following important points will be emphasized.

(1) Enrichment of the stored data: The stored data will be enriched in both of the core and the surrounding portions of the JMPD.

(2) Improvement of user-friendliness: Following the progress of the computational technology, both of hardware and software of the JMPD will be improved.

(3) Construction and enrichment of data networking system: A data networking system is essential for the efficient utilization of accumulated material data. Hence, the distributed database system named "Data-Free-Way" was constructed under the collaboration of the JAERI, the National Institute for Materials Science (formerly National Research Institute for Metals), the Japan Nuclear Cycle Development Institute and the Japan Science and Technology Corporation \(^{(31-6)}\). The most part of the JMPD is submitted to the Data-Free-Way. It is desirable that the number of the joining organizations to the data networking system increases.

(4) Construction of knowledge-base: It is desirable that in addition to the fact material data quantitative knowledge extracted from the fact database is stored. Recently, development has been made for a new system of a distributed knowledge base on the basis of factual Data-Free-Way stated above. A method of "seamless links" by applying eXtensible Markup Language (XML) via the Internet is adopted in this system.

References:


7. **Research on Cladding Materials for New Power Reactors**

The development of new cladding materials for MOX fuels used in the advanced reactors aiming at the ultra-high burn-up more than 100GWD/t and the fast neutron spectrum tailoring have been carried out. The stable austenitic Fe-(20~25)Cr-(25~35)Ni alloy with niobium alloy lining was selected for the most promising candidate based on the evaluation of major properties required for the MOX fuel cladding. The major problems on practical use of the candidate are considered to be the high thermal neutron adsorption and susceptibility to IASCC. The former is possible to improve up to the same level as Zircaloy claddings, because of the hard neutron spectra of MOX fuels and thinning up to nearly half thickness due to the high mechanical strength. The IASCC of fuel claddings made of meta-stable austenitic type 304 stainless steel had been experienced in light water reactors prior to the commercial use of Zircalloys. The susceptibility to IASCC of candidate Fe-Cr-Ni alloy was attempted to improve the sufficient austenite phase stability under heavy irradiation by applying the advanced steel making process and the chemical composition adjustment. The candidate alloy has the fine grain structure with ultra low impurities through the electron beam melting and thermomechanical treatment so-called the SAR process. The practical applicability of the alloy into cladding pipes was confirmed with the demonstration test by using the same commercial production process as Zircaloy claddings. The suppression in the micro-structural evolution of candidate alloys during heavy irradiation was clarified by triple ion beam irradiation tests. The grain boundary degradation of candidate alloys was not observed in any SSRT results and IGC tests at additional thermal histories that type 304 stainless steel was heavily sensitized. The applicability of niobium alloy for preventing PCI was clarified with the evaluation of the trapping effect of hydrogen, compatibilities to Fe-Cr-Ni alloys and MOX fuels. The ductility and corrosion resistance of niobium alloy in coolants are improved with the purification process by electron beam melting and alloying of molybdenum. The acceleration mechanism of interfacial reactions under heavy irradiation was analyzed with based on the excitation model by low energy electrons. The chemical reactivity of Xe and irradiation assisted oxidation of zirconium alloys were clarified with theoretical analysis and RF plasma simulation tests.
7.1 Corrosion behavior in high-temperature and high-pressure water of high purity Niobium alloys and austenitic stainless steels


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In the upgrading of the power reactor, ultra-high burnup extension of fuel (>100GWD/t) seems to be promising for economy, waste reduction and energy resource security. The realization of the new technology is dependent on the development of high-performance cladding tube. High purity niobium alloys and high stable austenitic stainless steels are developed as a candidate cladding material from the point of neutron economy, irradiation resistance, mechanical property, radioactivity and water-corrosion resistance of practical alloys. Corrosion behavior of the candidate materials for high-performance fuel cladding was examined in high-temperature and high-pressure water. The results were compared with the corrosion behavior of Zircaloy Intergranular corrosion

Strauss test was carried out for thermal aging and strained aging austenitic stainless steels (25Cr-35Ni-Ta, 25Cr-20Ni-Ta) produced in 1999 in order to confirm the intergranular corrosion resistance. The thermal aging was carried out at 600, 650 and 700°C for 15, 100 and 1000 h. The strain aging of 50% cold-worked specimens was carried out at 400, 450 and 500°C which produced the low-temperature sensitization by the austenite phase stability lowering for 15, 100 and 1000 h. Strauss test followed for JIS G-0575 was carried out. The specimen was put in boiling sulfate-sulfuric acid with copper plate for 100 h, and then the weight change was measured. The result is shown in Fig.7.1.1. The intergranular corrosion could not be recognized under all conditions except for type 316 stainless steel of the comparison material. Figure 7.1.2 shows the relationships between Ni+Cr equivalent and weight loss of the specimens which conducted the thermal aging (650°C×15 h) or the strained aging (450°C×100 h). The results of austenitic stainless steel (25Cr-35Ni-Ti) produced in 2000 include in Fig.7.1.2.

* The Japan Atomic Power Company
By the increase in the Ni+Cr equivalent, the corrosion resistance for intergranular corrosion is improved.

**Hardness**

The hardness of the specimens was measured in order to grasp microstructure change of the materials by the thermal or strained aging processing. The hardness test was carried out at a load of 5 kgf for 15 second hold using Vickers test machine. Figure 7.1.3 shows the results in austenitic stainless steel produced in 2000. In the figure, the hardness of each specimen before the thermal aging processing was also shown. There was no hardness change in either heat history. It seems that the microstructural change with the thermal aging processing such as precipitation, phase transformation have not been produced in the specimen. It is considered that the materials developed for high-performance fuel cladding have sufficient phase stability in the cold test which simulated irradiation effect by thermal aging and strained aging because of high austenite phase stability by suitable composition, super-purification and SAR processing.

**Corrosion resistance in high-temperature and high-pressure water**

Testing condition in high-temperature and high-pressure water is a temperature range of 280-420°C, a pressure of 23MPa, a dissolved oxygen concentration of 1ppm and an immersion time of 100 h. The materials tested are high stable austenitic stainless steels (25Cr-35Ni-Ta, 25Cr-20Ni-Ta, 25Cr-35Ni-Ti), high-pure niobium alloys (Nb, Nb-5W, Nb-10Mo), Zircaloy 2 and Zircaloy 4. The weight change of the specimens is shown in Fig.7.1.4. All stainless steels formed the oxide film of light color on the surface. The corrosion resistance of all stainless steels was better than Zircaloy 2 and 4 in all temperature range. The corrosion resistance of niobium was improved by the addition of tungsten or molybdenum. The niobium alloy added the 10% molybdenum showed corrosion resistance equivalent to Zircalloys.

**References**

Fig. 7.1.1 Results of Strauss test.

Fig. 7.1.2 Relationships between Ni+Cr equivalent and weight loss of austenitic stainless steels.

Fig. 7.1.3 Vickers hardness of stainless steel produced in 2000 after thermal aging or strained aging.

Fig. 7.1.4 Weight change of the specimens in high-temperature and high-pressure water.

- 149 -
7.2 Oxidation Enhanced by Low Energy Plasma at Pressure Boundary Materials of Nuclear Equipment

T. Saburi and K. Kiuchi
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One of the major subjects on atomic energy materials is considered to be the reliability of pressure boundary materials used in heavily corrosive nuclear environments. The oxidation rate of heat transfer materials such as fuel claddings in high temperature water and reprocessing device materials in nitric acid are accelerated one order higher than that obtained by immersion tests. The acceleration mechanisms is assumed to depend on the flux of oxygen activated by the thermal and irradiation decomposition in vapor phase at the heat transfer surface in aqueous environments. It is required to clarify the oxidation mechanism enhanced by chemically active oxygen for understanding the degradation mechanism of structural materials in those environments. With above viewpoint, we have developed the low energy plasma test device as a source of excited species for simulating and analyzing these accelerated oxidation mechanisms. The excited species generated in oxygen atmosphere was analyzed with the theoretical reaction model based on the empirical data obtained by RF plasma testing devices. The electrons are easy to accelerate the oxidation with linear dependency as expected by the direct excitation of the adsorbed oxygen. At the practical conditions, the oxidation would be controlled with incident flux of exited species and the bias effect in relation to the surface potentials like irradiation effects. The corrosion test was conducted using Cahn type microbalance apparatus as shown in Fig.7.2.1. Molybdenum, tungsten, titanium, nickel are selected as testing materials with respect to the representative characteristics of the thermodynamic and semiconductor properties of surface oxide films in relation to oxidation, and the results are shown in Fig.7.2.3. The exited species on the present test condition was evaluated theoretically. Figure 7.2.2 shows exited species theoretically evaluated on the present test condition. At the higher oxygen pressure, the major exited species are oxygen atoms. The corrosion testing apparatus with low energy plasma device is schematically shown in Fig.7.2.4. The incident flux with either particles of ions and electrons with negative charge or positive ions are possible to discriminate by applying the bias voltage between samples and plasma. The acceleration effect was evaluated with comparing the
plasma oxidation data with the thermal data obtained from oxidation tests. For example, titanium forms three different crystal structures of TiO₂ depending on the oxidation conditions. These oxides, namely, rutile, anatase and brookite are characterized with the formation conditions as followings: corrosion deposits in aqueous environments, oxide films rapidly formed at high oxidizing potential and natural minerals under high pressure. Therefore, the oxidation mechanism is elucidated with examining the crystal structure of oxide films formed. The specimen surface was examined by means of several surface analyses like ESCA, SIMS and Laser-Raman spectroscopy. The result is compared with that of oxide film formed on heat transfer surfaces in the nitric acid environments simulated to the spent fuel reprocessing equipment.

References
Fig.7.2.3 Oxidation example in 5Pa oxygen.

Fig.7.2.4 Plasma Oxidation Device.
7.3 Chemical Reactivity of Rare Gases Evaluated by Low Energy Plasma Excitation

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Radioactive heavy rare gases like xenon and krypton are formed as major fission products with the high abundance dependent on the burn-up ratio. The reliability of light-water reactor fuel elements is affected with the release behavior of these gases produced in oxide fuels\(^1\). The accumulation of these gases into a gap between cladding and fuels decreases the thermal conductivity and enhances the fuel temperature\(^1\). The extra gas release from oxide fuels would be enhanced vice-versa. Therefore, the chemical form and thermodynamic stability of heavy rare gases like xenon in oxide fuels is very important for understanding the gas release mechanism. Although, xenon and krypton are considered to rare gases elements with the closed shell with the high chemical stability, the chemical reactivity under heavy neutron irradiation like fuel elements would be controlled with the excitation behavior accompanied with radiochemical effects. Generally, the excitation cross-section on these rare gases has the maximum yield at low energy electrons with a few ten eV lower than the ionization yield\(^2\). The flux of low energy electrons is proportional to the flux of particles with the high energy. Electrons in relation to excitations are formed mainly by three mechanisms, namely, Compton scattering, electrons pair formation and photo-excitations. On the other hand, the formation of chemical compounds between heavy rare gases like xenon and fluorine was already reported\(^3\). The ionization energy of heavy rare gases elements is nearly equal to that of oxygen. It is expected to the high chemical reactivity between excited rare gases and oxygen in oxide fuels atmosphere under heavy irradiation. Moreover, the effective means for chemically stabilizing heavy rare gases is very important with respect to the waste management for minimizing the radioactive gas release from nuclear power plants.

In the present study, the chemical reactivity of heavy rare gases like xenon exited under heavy irradiation was examined with the low energy plasma excitation method by using devices driven two types of RF excitation sources as shown in Fig.7.3.1.

The chemical reactivity of heavy rare gases like xenon was examined fundamentally for understanding the gas release mechanism as fission products in nuclear oxide fuel elements under heavy irradiation and for searching the adequate waste management method as radioactive wastes. The electron density in low energy plasma was examined in two types of RF devices, namely, the IC. type and CC. type. The excitation behavior in mono-rare gases of He, Ne, Ar, Kr and Xe was examined as functions of gases pressure and RF input energy.
The results are summarized as follows:

1) The surface temperature of oxide fuels is one of the important factors on the gas release of radioactive heavy elements like xenon contained in fission products. The chemical form and stability of these gases have an important role on the gas release mechanism (Fig.7.3.2).

2) The electron density of heavy rare gases like xenon has the highest excitation yield in a wide pressure range under the low energy plasma discharge (Fig.7.3.3). The electron density increases with proportional to the RF output power at 100Pa at the maximum yield on the IC. type (Fig.7.3.4).

3) The electron density in the low energy plasma is dependent on the ionization energy of each rare gas (Fig.7.3.5).

4) The markedly high excitation efficiency is seen in Xe, Kr, Ar in the IC. type devices. It is interpreted with the excitation of adsorbed heavy rare gases by the low energy electrons with the high excitation yield.

References:


Fig.7.3.1 The experimental apparatus used in the low energy plasma excitation tests.
Fig.7.3.2  The mass fraction of the fission products calculated by ORIGEN-II. (PWR fuel; 4.5wt% enriched, Burn-up:45GWD/t, Cooling time:4years)

Fig.7.3.4  The effect of RF output power dependency on the electron density of rare gases obtained by the IC. type test.

Fig.7.3.3  The gas pressure dependency of the electron density for rare gases obtained by the IC. type test.

Fig.7.3.5  Difference in relationship between electron densities and the ionization energy obtained in both CC. type and IC. type test.
7.4 Effect of ion irradiation on corrosion behavior of austenitic stainless steel

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It is believed that inverse segregation of Cr and segregation of impurities (such as P, S, and Si) in grain boundary by the irradiation mainly caused IASCC of the austenitic stainless steel. As a quantitative evaluation method of the grain boundary, electro-chemical potentiokinetic reactivation (EPR) test, potentiostatic electrolytic test are being tried in the sensitized stainless steel, the neutron irradiated material and the ion irradiated material1-4). However, it is difficult to accurately evaluate the intergranular corrosion in the irradiated material because corrosion resistance in grain itself deteriorates by introduced radiation defects. Then, the development of ISACC resistance evaluation technique, which consists of the ion irradiation and the residual stress by the indentation and the electrochemical corrosion, is advanced. In this report, the change in surface appearance of ion irradiated Type 316 stainless steel (SUS316) which is the comparison material of the candidate cladding material was examined to confirm the capability of this technique.

Specimen is disk of 3mm in diameter with 0.2mm thickness. The electrochemically polished specimens were irradiated in triple (12MeV Ni^{+}, 750keV He^{+} and 290keV H^{+}) ion beam modes at a temperature of 300°C. The TRIM code was used to compute the implanted ion concentration and the displacement dose as a function of depth beneath the specimen surface. The displacement damage in the specimen is mainly attributed to Ni^{+} ion implantation. The peak dose is about 22dpa around 2mm. The peak values of concentration of Ni, He and H atoms are about 6000appm, 260appm and 2500appm, respectively. The peak positions of implanted He^{+} and H^{+} ions are controlled so that the effect of implanted Ni^{+} ions can be neglected. The He/dpa and H/dpa ratios of the specimens are about 24 and 230 at the depth of approximately 1.3mm. In case of the full MOX-ABWR of burn-up 100GWD/t, helium and hydrogen which are created by the nuclear transformation reaction are 120appm and 1100appm for the candidate material, respectively. Displacement damage becomes about 50dpa. The corrosion test specimen was produced by the cross section method5) in order to obtain information of the depth direction in the ion irradiated region. The test specimen which drove the indenter near the damage region was also produced.

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The analytical result of residual stress which arises around the indentation of the corrosion test specimen is shown in Fig.7.4.1. The stress component is represented the direction of y. The part of oblique line in Fig.7.4.1 shows a tensile state. The part of the triangular dotted line is a position of the indentation, and the part of the dotted line of left side is the damage region. It is proven that the tensile stress exists in the surface of damage region. The corrosion test carried out in 10% oxalic acid at room temperature (refer to JISG0571). The appearance of the corrosion surface was examined by scanning electron microscope (SEM) and atomic force microscope (AFM).

Surface appearance and surface roughness of the specimen after the corrosion are shown in Fig.7.5.2. The upper part of SEM image is the nickel-plated part, and the lower side is the test specimen. Three grains including grain boundary are in the test specimen side, and there is a corrosion-resistant difference by the crystal orientation\(^6\). Corrosion part of about 2mm corresponded to the damage region was observed in all grains. In line 1 of Fig.7.4.2, the difference in depth (difference in the corrosion rate) between damage region and non-damage was 50〜90nm. Line 2 and 3 which stride over grain are measurement result by AFM of damage region and non-damage region. Though the intergranular corrosion was able to be confirmed in the non-damage, the confirmation was not possible for the intergranular corrosion in the damage region. The results of the corroded specimen with indentation by SEM and AFM are shown in Fig.7.4.3. By the steeper pile-up around the indentation, it was not possible to confirm for the remarkable corrosion in residual stress division of the damage region.

After ion irradiated SUS316 with indentation was corroded, the surface appearance was examined to confirm the capability of this technique, and following results were obtained.
1) The difference in corrosion resistance between non-damage region and damage region was able to be confirmed from the SEM image.
2) The difference in depth between non-damage region and damage region was 50〜90nm. The difference could quantitatively be grasped by AFM.
3) The intergranular corrosion of non-damage region was able to be confirmed by AFM.

References
2) T.Inazumi et al., Corrosion, 46(1991)786
4) T. Tsukada et al., JAERI-M 92-169 (1992)
6) S. Yamaguchi, J. Appl. Phys., 22(1951) 983

Fig. 7.4.1 Analytical result of residual stress which arises around the indentation.

Fig. 7.4.2 Surface appearance and surface roughness of the specimen after the corrosion by SEM and AFM.

Fig. 7.4.3 Results of the corroded specimen with indentation by SEM and AFM.
7.5 Fatigue Behavior of Partial Penetration Weldment of Austenitic Stainless Steel for ITER Vacuum Vessel

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In a fusion experimental reactor, the high temperature plasma is confined and controlled inside a torus vacuum vessel (VV), which contains and supports in-vessel components such as blankets and divertors. The VV is the most important component in view of the fact that it is the first barrier to prevent tritium release from the reactor. As for the VV, low cycle fatigue caused by the cyclic electromagnetic and thermal stresses is identified to be the dominant failure modes. The VV of ITER is designed adopting a double-walled structure made of 316LN (0.06-0.08 % nitrogen) stainless steel and thereby composed of inner and outer shells of 40 - 60 mm in thickness jointed by welded stiffening ribs, and filled with shielding plates between the both shells to prevent neutron. The VV is also divided into 20 sectors, which are jointed by field welding at central plane of ports. Since it is not practicable to weld the outer shells from the inside of double-wall, the weldment is outside welding of butt joint and is also difficult to inspect its penetration bead, which may contain incomplete penetrations (IP). In this study, a weldment with an artificial IP existed in the weld root was manufactured to investigate the effect of the IP on fatigue behavior. Fatigue test of the weldment was performed using as-welded large specimens to explore its fatigue strength. Moreover, fatigue lives of the weldment were evaluated based on elastic-plastic fracture mechanics using finite element method (FEM).

A plate of 40 mm thick 316L stainless steel was employed in this study. The weldment was one side welding of butt joint using narrow-gap tungsten inert gas welding as shown in Table 7.5.1. The depth of IP was approximately 3 mm without root opening gap as shown in Fig.7.5.1. Fatigue
specimen of the weldment was machined with the IP located at its center. The fatigue test was performed at room temperature using hydraulic fatigue machine with stress ratio $R = 0$. Crack growth were also measured to obtain fatigue crack propagation rates of the weldment. In this study, J integral range $\Delta J$ was used as a correlation parameter of the fatigue crack propagation rates. To calculate $\Delta J$ at each individual crack length, elastic-plastic finite element analyses were conducted.

The fatigue specimen of weldment fractured at the IP and the crack propagated through the weld metal perpendicularly to the loading direction. Crack propagation curves are shown in Fig.7.5.2. When the crack initiation was defined as 0.1 mm of crack growth from the tip of IP, ratios of the crack initiation lives to the total fatigue lives were less than 3% for both specimens. Namely, most of total fatigue lives were crack propagation lives. Since the tip of IP was very sharp, the IP behaved and propagated like a crack. The fatigue strength of the weldment was shown in Fig.7.5.3 making comparison with that of smooth specimen. The fatigue strength of the weldment was considerably lower than that of smooth specimen and fatigue strength reduction factor $K_t$ was approximately 5. In order to evaluate fatigue lives of the weldment, integration of fatigue crack propagation rates was numerically conducted substituting the relation between $\Delta J$ and crack length, which was obtained by the elastic-plastic FEM analyses. The results of calculation for the weldment used in the fatigue test and a weldment contained IP of 1 mm depth are also exhibited in Fig.7.5.3. The calculated fatigue lives for the fatigue specimens were corresponding to the experimental fatigue lives. As for the weldment contained IP of 1 mm depth, the fatigue strength reduction factor $K_t$ was about 4. The fatigue strength of weldment with the IP was largely decreased comparing with that of the smooth specimen even if the IP was small.

References
Table 7.5.1  Welding processes and conditions.

<table>
<thead>
<tr>
<th>Groove and welding process</th>
<th>Welding conditions</th>
</tr>
</thead>
<tbody>
<tr>
<td>8° grinding, 8 passes</td>
<td>TIG welding</td>
</tr>
<tr>
<td>3.8</td>
<td>Welding current</td>
</tr>
<tr>
<td>70−555mm/min</td>
<td>180−375A</td>
</tr>
<tr>
<td></td>
<td>Welding voltage</td>
</tr>
<tr>
<td></td>
<td>13−18.5V</td>
</tr>
<tr>
<td></td>
<td>Welding speed</td>
</tr>
<tr>
<td></td>
<td>70−555mm/min</td>
</tr>
<tr>
<td>Filler Metal: Y316L</td>
<td></td>
</tr>
</tbody>
</table>

Fig. 7.5.1 Macro-structure of weldment.

Fig. 7.5.2 Crack propagation curves of weldment.

Fig. 7.5.3 Comparison of experimental fatigue lives with evaluated fatigue lives.
8. Research on Reprocessing Materials Applied for Commercial Reprocessing Plants

The development of life prediction technologies, corrosion resistance alloys and corrosion monitoring methods applied to the Purex type commercial reprocessing equipment has been carried out. On the life prediction study in relation to the Science and Technology Agency projects for the safety research, the reliability of an evaporator for nitric acid recovery and a dissolver used in the Rokkasho Reprocessing Plant was evaluated by the small scale mock-up tests, laboratory tests for evaluating the controlling factors and computer simulation methods for establishing the database system. The third ISI after operated nearly twenty thousand hours was pursued for evaluating the corrosion behavior of heat transfer tubes of the evaporator mock-up made of R-304ULC stainless steel operated at the low boiling point. The acceleration behavior of corrosion due to the initiation and propagation of grain-boundary attacks was founded. The distribution of wall thinning rate along the vertical direction of tubes was analyzed with the computer simulation of empirical data by using the thermo-fluid dynamics codes and two dimensional graphic simulation methods. The time dependent corrosion behavior was divided into two controlling steps, namely, the anodic dissolution at short time and the default of grains after long time. The equality in corrosiveness between V^{5+} and Np^{6+} on heat transfer surfaces was proved by corrosion tests using radio isotopes in WASTEF. The reliability of zirconium used in devices operated at a normal pressure was examined with respect to the susceptibility to stress corrosion cracking. The enhancement effect of susceptibility to stress corrosion cracking was clarified by the slow strain rate tensile test. The applicability of acoustic emission method for stress corrosion cracking was quantitatively evaluated.

This research is sponsored by the Science & Technology Agency under contract of the “Demonstration Tests for Evaluating Reliability of New Materials Used in Reprocessing Plant”.

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8.1 Operation and In-service Inspection of Demonstration Testing Equipments for Acid Recovery Evaporator and Dissolver in 2000

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The demonstration testing equipments for the Rokkasho Reprocessing Plant (RRP) have been operated at Ehime Works of Sumitomo Chemical Co., Ltd. The first operation of the acid recovery evaporator and the dissolver were started on March 27, 1998 and March 23, 1999, respectively. The corrosion behavior of the demonstration equipments was evaluated by in-service inspection (ISI). The ISI of them were carried out from August 21 to September 1, 2000. The ISI of the acid recovery evaporator was third, and that of the dissolver was second.

**Acid Recovery Evaporator**

(1) The third Operation

The acid recovery evaporator was operated for 4410 h as following conditions. The chemical composition of nitric acid solution in it was controlled as following: 9 N HNO₃, 1200mg/l Fe³⁺, 300mg/l Cr⁺⁷, 5mg/l Ru⁺³ and 50mg/l V⁺⁵ for simulating the solution of the reprocessing plant. The flow rates at a steam temperature of 100°C is 178 kg/h, with the circulation flow rate of 860 kg/h in the thermo-siphon tube. The by-pass flow rate from the top of the heat conducting tubes is 21 kg/h, the flow rate of concentrate is 5.4 kg/h and the flow rate of distilled acid is 172.6 kg/h. The feed solution is concentrated 40 times by thermo-siphon. The steam temperature of the evaporator was varied for estimating the testing conditions at steady-state operation. It was controlled at 98, 96, 94 and 92°C for a short time. The relationship between the feed flow rate and steam temperature is shown in Fig.8.1.1. With decreasing of the feed flow rate from 178 to 112 kg/h, the circulation flow rate was decreased from 860 to 790 kg/h, and the flow rate in by-pass tube was increased from 21 to 667 kg/h.

(2) The third ISI

One hundred eighty grams of deposit was found inside of the bottom cap of gas-liquid separator. In the second ISI, it was 176 g in 7180 hours operation. These deposits were grains detached from the inside of heat conducting tubes due to grain-boundary attack. The thickness of 7 heat conducting tubes was measured by the ultrasonic immersion measurement
using a 10 MHz UT sensor. The accuracy of measurements is within ±0.01 mm. The thinning of tubes was observed in two different positions. One is nearly 1 m apart from the bottom tube plate and the other is the upper of it as shown in Fig.8.1.2. The maximum corrosion rate of the tubes is higher than 0.1 mm/y. The corrosion at the low position of heat conducting tube is governed with the surface temperature of it. On the other hand, the corrosion at the upper position is enhanced by the oxidizing ions like V^{5+} formed by the evaporation and thermal-decomposition of nitric acid at heat conducting surfaces.

**Dissolver**

(1) The second Operation

The dissolver was operated for 4650 h as following conditions. The steam temperature is 165°C under 0.71MPa. The steam flow rate is ca. 30kg/h. The nitric acid evaporated from the dissolver is cooled in a condenser and dropped in the dissolver. The testing solution is 3N nitric acid solution contains 142g/l Al(NO₃)₃, 1200mg/l V^{5+}, 1000mg/l Ce^{4+} and 1500mg/l Ru^{3+} for simulating the dissolver solution of the reprocessing plant. The solution was replaced every three months. The compositional change of it was little.

(2) The second ISI

After 4,650 hours operation, the inside of the dissolver was covered with adhered film and some deposits. They were a mixture of ruthenium oxide and vanadium oxide deposited from the test solution. Radiographic test and dye penetrating test were carried out along the welding lines for detecting defects such as cracking. There were no defects in tested parts. The thickness of the dissolver was measured by ultrasonic testing using a 10 MHz sensor. The wall thinning was not observed.
Fig. 8.1.1 Relationship between the feed flow rates and steam temperatures obtained by testing equipment for evaluating the operation parameters of the evaporator.

Fig. 8.1.2 Corrosion rates of 7 heat conducting tubes obtained by UT measurements.
8.2 Inspection of Heat Transfer Tubes after Mock-up Tests of Miniaturized Apparatus for the Acid Recovery Evaporator

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The demonstration test for the acid recovery evaporator used as the major equipment of Rokkasho Reprocessing Plant (RRP) has been carried out since April in 1998\textsuperscript{11}. The mock-up miniature equipment of the acid recovery evaporator has been used in the demonstration test. The total time of the test using the mock-up equipment reached about two and half years in March 2001, which corresponds to about 20,000 hours. After the test finished, four from among the seven heat transfer tubes, which were made of RSUS304ULC stainless steel (34mm in outer diameter, 4.5 mm in thickness and 4.7 m in length), were drawn out from the equipment. For one among them, the corrosion level and the mechanical properties were evaluated.

On the corrosion level, the residual thickness of the tube was measured with an ultrasonic apparatus and by an optical method. Both measurements showed good agreement. The corrosion level depends on the positions in the longitudinal direction of the tube. The corrosion level showed the maximum values near the upper tube sheet and at the position of 900 mm from the lower tube sheet. The value of the residual thickness was 4.4 mm. Intergranular corrosion was observed in the inner surface of the heat transfer tube as shown in Fig.8.2.1. The corrosion depth at the grain boundary was statistically shown to be about one grain from the surface as shown in Fig.8.2.2.

In order to evaluate mechanical properties of it, various ordinary tests for the tube were carried. They were tensile, roughness, bend, hardness flattering and flaring test. No change in mechanical properties was observed in comparison with those measured before the demonstration test. Propagation of intergranular cracks in the inner surface of
the specimen was found after flattering test.

Reference


Fig.8.2.1  SEM photograph of inner surface of tested tube.

Fig.8.2.2  Cross-sectional view of the inside of tested tube.
8.3 Development of Ultrasonic Thickness Measuring System

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The demonstration test for evaluating reliability of the acid recovery evaporator in Rokkasho Reprocessing Plant has been carried out in JAERI. As a nondestructive measurement of the thickness of heat transfer tubes in an acid recovery evaporator and short tubes, we have developed an ultrasonic thickness measuring system using immersion method with high resolution. The system can measure and record tube thickness automatically with a personal computer. The results obtained by the system are coincident with the results obtained by a destructive method using an optical microscope.

A block diagram of the ultrasonic thickness measuring system is shown in Fig. 8.3.1. An ultrasonic thickness measurement unit (Panametric Type 5215) was used for the thickness measurement of heat transfer tubes. An ultrasonic pickup probe having the frequency at 10MHz (Panametric M312-SUOS-1) was also used. The ultrasonic thickness measurement unit dealt with the signal detected by the probe, and the voltage corresponding with thickness is output. The voltage is transformed into digital values by an A/D converter in a computer. Figure 8.3.2 is shown overview of the ultrasonic thickness measuring system.

The thickness of heat transfer tubes measured by this system is shown in Fig. 8.3.3. The results obtained by ultrasonic thickness measurements conducted by Sumitomo Chemical Co., Ltd. and the results obtained by optical thickness measurements are also shown in this figure. The thickness measured by the system has the some points where measured value changes greatly. It is due to surface scratches that produced by pulling up heat transfer tubes. The thickness measured by this system, however, agreed well with those measured by other methods. It was confirmed that the accuracy of thickness measured by this system is below 0.01 mm.
Fig.8.3.1 Block diagram of ultrasonic thickness measurement system.

Fig.8.3.2 Overview of ultrasonic thickness measuring system.
Fig. 8.3.3 Thickness of heat transfer tubes measured by three methods.
8.4 Corrosion of Stainless Steel in Nitric Acid Solution Containing Neptunium

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The previous report 1) conferring corrosion of type 304ULC stainless steel (304 ULC SS) was presented that the corrosiveness of neptunium ion in pure nitric acid may be equal to that of vanadium ion. In the present study to verify the corrosiveness of neptunium ion in nitric acid solution containing various metallic ions (i.e., iron, chromium and ruthenium ions), the corrosion of 304ULC SS in neptunium nitrate solution under heat-transfer condition was investigated by weight loss measurement obtained by ICP analysis.

The test specimens were tested for 400 h in the 900 mg/L of neptunium nitrate solution and for 600 h in 9 mol/dm³ nitric acid solution containing 200 mg/L vanadium ion, respectively. Table 8.4.1 shows the chemical composition of testing solutions. The solution was renewed every 300 h. The ratio of solution volume to surface area of specimen was about 18 mL/cm². The inner pressure of test-cell was controlled under 160 hPa by a vacuum controller and the heat-flux of specimen was 60 kW/m² during the tests.

As shown in Fig.8.4.1, corrosion rates increased with test time. The corrosion rate of type 304ULC SS in the testing solution with neptunium content of 900 mg/L was slightly lower than that in the nitric acid solution containing 200 mg/L vanadium ion. The amount of neptunium (900mg) or vanadium (200mg) corresponds to 3.8 mmol. According to this result, the assumption that corrosiveness of vanadium ion in nitric acid solution is equal to that of neptunium ion may be suitable. Figure.8.4.2 shows the specimen surface after corrosion tests. Intergranular corrosion was preferentially occurred in both nitrate solutions. The corrosion mechanism should be the same in both nitrate solutions. From above results, we will change the testing solution using a mock-up test for simulating an acid recovery evaporator in April 2001. The chemical composition of testing solution is the same as solution (A) in Table 8.4.1.

<table>
<thead>
<tr>
<th>Solution</th>
<th>HNO₃(mol/L)</th>
<th>Fe</th>
<th>Cr</th>
<th>V</th>
<th>Np</th>
<th>Ru</th>
</tr>
</thead>
<tbody>
<tr>
<td>(A)</td>
<td>9</td>
<td>1200</td>
<td>300</td>
<td>200</td>
<td>-</td>
<td>5</td>
</tr>
<tr>
<td>(B)</td>
<td>9</td>
<td>1200</td>
<td>300</td>
<td>-</td>
<td>900</td>
<td>5</td>
</tr>
</tbody>
</table>

Table 8.4.1 Chemical composition of testing solutions (mg/L).
Fig. 8.4.1 Corrosion rates of type 304ULC SS under heat transfer condition.

Solution (A); V200mg/L            Solution (B); Np900mg/L

Fig. 8.4.2 SEM photograph of specimens after corrosion tests.

Reference
8.5 Corrosion of Zirconium in Spent Fuel Dissolved Solution under heat conducting condition

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Zirconium has been used as a structural material of a spent fuel dissolver in a reprocessing plant. The corrosion resistance of it has been mainly evaluated by the corrosion tests in nitric acid solutions simulating a spent fuel dissolved solution due to the difficulty such as a preparation and handling of spent fuel dissolved solution.

In this study, we have carried out corrosion tests at WASTEF in JAERI after preparing spent fuel dissolved solution. The corrosion resistance of zirconium was evaluated under heat conducting condition. As a result, it has indicated that base and weld metals of zirconium have excellent corrosion resistance even in fuel dissolved solution as well as in pure nitric acid solution.

Table 8.5.1 shows the chemical composition of zirconium plate used for corrosion tests under heat transfer condition. Welded specimen and base metal specimen with 70 mm diameter and 6 mm thickness were prepared from it. The specimen was set in a corrosion cell filled in spent fuel dissolved solution and heated by electric heater with a heat flux of 70 kW/m². Figure 8.5.1 shows the instrument for corrosion tests. The spent fuel dissolved solution has been prepared using PWR type spent fuel (45GWD/t). Table 8.5.2 shows the chemical composition of the spent fuel dissolved solutions used in the present study. The solution had been boiling for 1000 h during the tests. From the weight loss measurement, corrosion rate was calculated. A reference corrosion test was conducted using 6N HNO₃ solution. After corrosion tests, the specimen surface was observed with a scanning electron microscope (SEM).

Figure 8.5.2 gives the corrosion rates of zirconium. In spent fuel dissolve solution the corrosion rate of base metal is 2.3 mg/m²·h and that of the welded specimen showed 0.6 mg/m²·h. The highest corrosion rate is almost the same as in 6N HNO₃. Zirconium has excellent corrosion resistance in boiling spent fuel dissolved solution. Figure 8.5.3 shows the surface morphology of zirconium. Localized corrosion was not observed in each specimen.
Table 8.5.1  Chemical composition of zirconium tested (wt%).

<table>
<thead>
<tr>
<th>Zr</th>
<th>Hf</th>
<th>Fe+Cr</th>
<th>C</th>
<th>H</th>
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<th>O</th>
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<td>99.2</td>
<td>0.4</td>
<td>0.07</td>
<td>0.02</td>
<td>0.0004</td>
<td>0.0035</td>
<td>0.14</td>
</tr>
</tbody>
</table>

Table 8.5.2  Chemical composition of spent fuel dissolved solutions.

<table>
<thead>
<tr>
<th>Solution</th>
<th>H⁺ (mol/L)</th>
<th>NO₃⁻ (mol/L)</th>
<th>U+Pu (g/L)</th>
<th>Volume (mL)</th>
</tr>
</thead>
<tbody>
<tr>
<td>(A)</td>
<td>3.96</td>
<td>5.96</td>
<td>238</td>
<td>290</td>
</tr>
<tr>
<td>(B)</td>
<td>3.83</td>
<td>5.90</td>
<td>246</td>
<td>300</td>
</tr>
</tbody>
</table>

Fig.8.5.1  Instrument for corrosion test under heat conducting condition.
Fig. 8.5.2 Corrosion rates of zirconium under heat transfer condition. 
SF solution: spent fuel dissolved solution

Fig. 8.5.3 SEM photographs of specimens after corrosion tests.
8.6 SCC Monitoring of Zirconium in Boiling Nitric Acid by Acoustic Emission Method

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The materials used in commercial reprocessing plants require the good corrosion resistance against boiling nitric acid solutions including oxidants like TRU and FP. Zirconium has been used as one of the materials applied to equipment operated in a normal pressure like a dissolver and a Pu evaporator, because zirconium shows the excellent corrosion resistance in heavily oxidizing nitric acid solutions. However, it has the high susceptibility to transgranular type stress corrosion cracking (TGSCC) in nitric acid solutions. On this study, the practical applicability of Acoustic Emission (AE) method for monitoring the SCC of zirconium was examined using the slow strain rate tensile test (SSRT).

Figures 8.6.1 shows the stress-time curves obtained by in boiling nitric acid and in silicone oil at 388K, respectively. Comparing the results in nitric acid with that in silicone oil, the time to failure decreased with increasing the concentration of nitric acid as expected from the previous report. The reduction rate of rupture time in nitric acid to that in silicone oil was 19%, 21%, 22% and 25% in 3N, 6N, 8N and 12N HNO₃, respectively. The sensitivity to SCC increased with increasing the concentration of nitric acid. Figure 8.6.2 shows the histograms of AE hits obtained in boiling nitric acids. In the present analyzing condition, the AE events due to the SCC could be detected in nitric acids by eliminating the plastic deformation events. The background noise was detected in a wide range of stress–time curve, even if the low stress region in 8N and 12N HNO₃. It might be the noise caused from the formation and annihilation of boiling voids. The difference in the initial AE counting rate between 8N and 12N HNO₃ would

Fig. 8.6.1 Stress-time curves of SSRT.
be dependent on the boiling condition as the background noise level.

In 12N HNO₃, the rising time of AE hits is about 35 h and the stress level is 215 MPa. In 8N HNO₃, the rising time of AE hits is about 52 h and the stress level is 305 MPa. In 6N HNO₃, there is two peaks on rising time. The first rising time of AE hits is about 57 h and the stress level is 293 MPa. In 3N HNO₃ the rising time of AE hits is about 109 h and the stress level is 420MPa. This stress level is nearly the ultimate tensile stress. The rising time is clearly shortened by increasing the acidity. The stress level producing AE event due to SCC in 12N HNO₃ was decreased up to 0.2 % proof stress level of zirconium at 373K. In 6N HNO₃ there are two peaks of AE hits rising. The second peak is arising at the tensile stress. The AE hits behavior is similar to its in 3N HNO₃. It is suggested that the difference in the

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**Fig. 8.6.2** Histograms of AE hits rate of SSRT.

---

**Fig. 8.6.3** Photographs and the area ratio of quasi-cleavage fracture after SSRT.
AE generating mechanism depends on the acidity of nitric acid solutions. The photographs of the fracture surface after SSRT are shown in Fig.8.6.3. The surface morphology in silicone oil showed the metallic luster and ductile fracture. On the other hand, that in boiling nitric acids showed black color and quasi-cleavage type fracture along circumference of notch groove. Figure 8.6.4 shows the relationship among the susceptibility to SCC, the area ratio of the quasi-cleavage type fracture and the nitric acid concentration. The area ratio of the quasi-cleavage type fracture increase with increasing the nitric acid concentration: 38%, 38%, 65% and 80% in 3N, 6N, 8N and 12N HNO₃, respectively. It depends on the susceptibility to SCC. The fracture surface area by SCC was coincident with the generation rate of AE signals. It is clear that the susceptibility to SCC increases with increasing the concentration of nitric acids more than 6N.

The applicability of the AE method for monitoring the transgranular type SCC of zirconium in boiling nitric acid solutions was examined and following results were obtained.

(1) The AE method is one of the most promising methods for clarifying the crack initiation and propagation of SCC of zirconium.

(2) Both the AE hits and the susceptibility to SCC increased with increasing the concentration of nitric acid. The stress level for generating the AE event due to the SCC was decreased up to the 0.2% proof stress in boiling 12N HNO₃.

References

1) K.Kiuchi: Genshiryoku Gakkai-Shi, 31 (1989), 229. [In Japanese]
8.7 Database System for Analyzing Reliability of Spent Fuel Reprocessing Plant

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The reliability of acid recovery evaporators and dissolvers of the Rokkasho Reprocessing Plant (RRP) has been evaluated with data obtained by the demonstration tests using the small-scale mock-up equipment. In this research, numerical analysis using several computational codes has been also conducted by use data obtained from the demonstration tests and laboratory tests.

The data obtained from mock-up tests and numerical analysis have been utilized for constructing database system which should be applied to the life prediction of RRP. The concept of the database system integrated evaluation codes based on the thermo-fluid dynamics and corrosion kinetics\(^1\) is shown in Fig. 8.7.1. In this figure, operation parameters for an evaporator, such as steam temperature, feed flow rate etc., are given to the computational codes coupled with database for providing the evaporation and corrosion data and predicting the corrosion rates and locations. Current contents of the database system are as follows:

1) In-service inspection data of the mock-up; wall thinning rates.
2) Operation data of mock-up; flow rate, steam temperature etc.
3) Simulation results for the mock-up based on thermo-fluid analysis; tube temperature, heat-flux, flow velocity, finite-element creep analysis for the mock-up of dissolver and graphic analysis for the propagation behavior of grain-boundary attack.

An example of the data browsed from the database system is shown in Fig.8.7.2. It represents corrosion rates extracted from the operation data of the mock-up evaporator. The metal dissolution rate corresponding to the average corrosion rate is about 0.075 \text{g/m}^2\cdot\text{hr} in 16,000 hours.

Reference


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Fig. 8.7.1  Schematic model of database system for an evaporator.

Fig. 8.7.2  Display of data on Metal dissolution rates in the database system.
8.8 Reliability Analysis of Spent Fuel Dissolver

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On the Purex type reprocessing plant, spent fuels are dissolved in boiling nitric acid. A continuous rotary dissolver is used in the Rokkasho Reprocessing Plant (RRP). It is made of zirconium with excellent corrosion resistance to nitric acid solutions containing a large amount of TRU and FP. The long performance of zirconium is considered to be affected by the stress corrosion cracking (SCC) susceptibility, which is increased both at the heat transfer surface with a high oxidation potential and at the heat affected zone on TIG welding joints.

According to this background, the demonstration test using a small-scale dissolver mock-up has been carried out since 1999. On this study, the thermo-stress and creep analyses using three-dimensional finite-element structural dynamics code have proceeded. The analysis models were constructed for evaluating both a practical dissolver and the mock-up by using heat-conducting shell elements. In order to ensure the applicability of the analysis models, major parameters such as the heat transfer coefficients for dry area, wet area and insoluble sediments were estimated.

Figures 8.8.1 and 8.8.2 show the simulation results of surface temperature and stress distribution (MISES stress) at an initial stage of the operation. Initial temperatures of the steam and liquid are 438K and 380K, respectively. The significant difference in temperature can be seen between the boiling surface of steam jacket and no-boiling surface in liquids, producing the high stress concentration of about 84 MPa in each end of the jacket and welding joints.

Figures of 8.8.3 and 8.8.4 demonstrate the effect of operation time on MISES stress due to the plastic deformation by low temperature creep. Figure 8.8.3 shows that the amount of deformation is about 0.6 mm after 4 years operation of practical scale equipment. Figure 8.8.4 shows the distribution of a residual stress. It is suggested that the maximum stress is reduced by the creep deformation up to about 48 MPa in weld-line area, and the high stress concentration in each end of the steam jacket was disappeared. Comparing Fig.8.8.4 to

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Fig. 8.8.2, it is suggested that the creep deformation reduces the residual stress and controls the initiation of SCC on TIG weld joints.

Fig. 8.8.1 Temperature at initial stage.

Fig. 8.8.2 MISES stress at initial stage.

Fig. 8.4.3 Plastic deformation after 4 years.

Fig. 8.8.4 MISES stress after 4 years.
9. Rock-Like Oxide Fuel for Plutonium Burning in LWRs

Research on plutonium rock-like oxide (ROX) fuel and its once-through burning in LWRs has been carried out in order to establish a new option for the optimum use of excess plutonium. Features of the ROX-LWR system are almost complete burning of plutonium and the direct disposal of spent ROX fuels without reprocessing.

To evaluate irradiation behavior of the ROX fuel, five ROX fuels, including new particle-dispersed ROX (PD-ROX) fuels were irradiated in JRR-3. These fuels were prepared using 20 % -enriched uranium instead of plutonium for easier fuel handling. Destructive post-irradiation examinations were performed on these fuels for such items as the puncture test and ceramography. The main results are: (1) Yttria stabilized zirconia (YSZ) single-phase fuel showed an excellent irradiation behavior; i.e. low fission gas release (<3%), negligible swelling and no appreciable restructuring. (2) The PD-ROX fuels showed lower swelling and higher fission gas release than those of powder mixture fuels. (3) Spinel matrix fuels showed some amounts of spinel decomposition and restructuring. Burnup analyses on the fuels were also carried out to evaluate burnup, linear power, fuel temperature, and isotopic abundance and production yield of fission Xe and Kr gases. Calculated results on isotopic abundance and production yield of fission gases showed a good agreement with those of the measurement.

Transient behavior of PD-ROX fuels was investigated in the Nuclear Safety Research Reactor. A short test rod containing 14 PD-ROX pellets, where about 300 μm YSZ particles were dispersed in the spinel matrix, were irradiated by a short pulse simulated a reactivity initiated accident (RIA) condition. The PD-ROX fuel rod showed very similar behavior to the powder mixture YSZ/spinel composite fuel rod. The threshold energy of the fuel pin failure for the ROX fuel seemed to be about 10 GJ/m³ and slightly higher than that of the UO₂ fuel.

Analyses of RIA and loss of coolant accident (LOCA) were performed for thorium based ROX (Th-ROX) fueled PWR using EUREKA-2 and RETRAN2 codes, respectively. Because reactor physics parameters of a Th-ROX fueled core were nearly the same as those of the UO₂ fuel core, no severe consequences were found to occur in either RIA or LOCA event. The Th-ROX fuel temperature under the RIA condition was expected to be 1500 K (lower than that expected for the UO₂ fuel by about 500 K), while the cladding temperature under the LOCA event was expected to be 1140 K (slightly higher than that expected for the UO₂ fuel).
9.1 Post-Irradiation Examination of Rock-Like Oxide Fuels (2) -Puncture Test-

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To evaluate irradiation behavior of the ROX fuel, irradiation experiment was carried out using 20% of enriched uranium instead of plutonium. Five kinds of ROX fuels were irradiated and several post-irradiation examinations were carried out. The results of profilometry, puncture test and fission gas analysis are described and discussed.

Five fuels were prepared; a single phase fuel of an yttria-stabilized zirconia containing $\text{UO}_2$ (U-YSZ), two particle-dispersed type fuels of U-YSZ particles in spinel or corundum matrix, two homogeneously-blended type fuels of U-YSZ and spinel or corundum. These fuels were named Z, SD, CD, SH and CH, respectively, and were irradiated in JRR-3. The compositions of these fuels and irradiation conditions were shown in previous paper\(^1\).

Estimated linear power and temperature of each fuel are listed in Table 9.1.1. It must be mentioned that the temperature of each fuel is remarkably higher than that of irradiation condition in LWRs. The estimated burnups are about 0.1 MWD*cm\(^3\), which is corresponding to about 10GWD•t\(^{-1}\) of LWR $\text{UO}_2$ fuel.

<table>
<thead>
<tr>
<th>Fuels</th>
<th>Z</th>
<th>SD</th>
<th>CD</th>
<th>SH</th>
<th>CH</th>
</tr>
</thead>
<tbody>
<tr>
<td>Linear power / kW•m(^{-1})</td>
<td>13.9</td>
<td>23.0</td>
<td>24.9</td>
<td>23.4</td>
<td>20.7</td>
</tr>
<tr>
<td>Temperature at Surface of pellet / K</td>
<td>990</td>
<td>1250</td>
<td>1300</td>
<td>1440</td>
<td>1290</td>
</tr>
<tr>
<td>Center of pellet / K</td>
<td>1490</td>
<td>1740</td>
<td>1820</td>
<td>1940</td>
<td>1730</td>
</tr>
</tbody>
</table>

The irradiation capsule was disassembled and five fuel pins were taken out after the irradiation. The stainless steel cladding surface where fuel pellets were loaded was partially discolored by oxidation. Any change in axial pellet stack length was not recognized from the comparison of X-ray photographs taken before and after irradiation. The precise diameters were measured from four directions of 45 degrees intervals. The measured values from any direction were almost the same. The average diameter of 4 directions for fuel Z, SD and SH
are shown in Fig. 9.1.1. No significant diameter change was observed except for the SH fuel. The average diameter increase of the SH fuel was about 30 μm at the fuel stack region. It is considered that the diameter increase of the pin was caused by the swelling of the pellet, since the quantity of diameter change was larger than that estimated from thermal expansion. Nevertheless the diameter change of the SH fuel pin was below 0.5 % of the diameter. The other fuels did not cause the shape change of the fuel pin, which suggest that the swelling of the fuels were less than the initial gap of about 100 μm.

![Diameter changes for Z, SD and SH fuels](image)

**Fig. 9.1.1 Diameter changes for Z, SD and SH fuels**

Fission product gas release is one of the key parameters for fuel irradiation performance. Gases in the fuel pin were collected and analyzed through the puncture test. Measured isotopic abundance of Kr and Xe using mass spectrometry were compared with the calculated results. Total yield and isotopic abundance of Kr and Xe were obtained from the burnup calculation using the SRAC95 code. The results of measured FP gas quantity and calculated total FP gas yield are shown in Table 9.1.2. The released Xe/Kr gas ratios were measured to be about 7.2, which agreed well with the calculated results. The smaller Xe/Kr ratio of the Z fuel might come from experimental difficulties because very small amount of gas was available for the determination.

The release rate of FP gas is the ratio of the measured and calculated FP gas volumes. They are also shown in Table 9.1.2. The Z fuel showed low FP gas release rate. The irradiation temperature of Z fuel was lower than that of other fuels. It is considered the lower irradiation temperature is an important factor for the lower FP gas release than other fuels.
The FP gas release rates of the particle dispersed type fuels (SD, CD) were higher than those of the homogeneously-blended type fuels (SH, CH). The microscopic crack generation may cause the high gas release rate of particle dispersed type fuels. Since thermal expansion of the U-YSZ particle is higher than that of the matrix, the cracks are generated in the matrix near the grain boundary. The grain size of particle dispersed type fuel was one order of magnitude larger than that of homogeneously-blended type fuel. So the thermal stresses concentrated, and cracks were generated easily. The produced FP gas might be escaped through the cracks generated in the grain boundary of U-YSZ particle. So, it is necessary for particle dispersed type fuels to fabricate the fuels which can relax the thermal stress generated by difference of thermal expansion coefficients between the U-YSZ particle and the matrix.

In addition, the FP gas release rates of spinel-based fuels (SD, SH) were higher than those of corundum-based fuels (CD, CH). Because of high irradiation temperature and large temperature gradient, the decomposition and restructuring of the spinel was occurred and this leads to the high FP gas release (see chapter 9.2). The restructuring of the spinel and the large amount of gas release will be avoided by lowering the irradiation temperature, because spinel has essentially high irradiation resistance to neutron and high gas retention ability.

### Table 9.1.2 Measured and calculated Xe and Kr volume and FP gas release rate

<table>
<thead>
<tr>
<th>Fuels</th>
<th>Z</th>
<th>SD</th>
<th>CD</th>
<th>SH</th>
<th>CH</th>
</tr>
</thead>
<tbody>
<tr>
<td>Measured volume of</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>released FP gas / cm³</td>
<td>Kr</td>
<td>0.004</td>
<td>0.083</td>
<td>0.050</td>
<td>0.050</td>
</tr>
<tr>
<td></td>
<td>Xe</td>
<td>0.021</td>
<td>0.601</td>
<td>0.364</td>
<td>0.359</td>
</tr>
<tr>
<td>Xe/Kr ratio</td>
<td></td>
<td>5.4</td>
<td>7.3</td>
<td>7.2</td>
<td>7.0</td>
</tr>
<tr>
<td>Calculated volume of</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>produced FP gas / cm³</td>
<td>Kr</td>
<td>0.131</td>
<td>0.219</td>
<td>0.234</td>
<td>0.229</td>
</tr>
<tr>
<td></td>
<td>Xe</td>
<td>0.936</td>
<td>1.57</td>
<td>1.68</td>
<td>1.65</td>
</tr>
<tr>
<td>Xe/Kr ratio</td>
<td></td>
<td>7.1</td>
<td>7.2</td>
<td>7.1</td>
<td>7.2</td>
</tr>
<tr>
<td>FP gas release rate / %</td>
<td></td>
<td>2.4</td>
<td>38.2</td>
<td>21.6</td>
<td>21.8</td>
</tr>
</tbody>
</table>

Reference:

9.2 Post-Irradiation Examination of Rock-like Oxide Fuels (3) -Ceramogrophic examination-

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The confirmation of the rock-like oxide (ROX) fuel irradiation behavior by ceramographic examination is very important because the ROX fuel is a once-through type fuel and is to be disposed without reprocessing after used in the conventional light water reactors. Five kinds of fuels; a single phase (Z) fuel of UO₂ and yttria-stabilized zirconia solid solution (YSZ), two particle-dispersed type fuels with spinel (SD) or corundum (CD) matrix, and two homogeneously-blended type fuels with spinel (SH) or corundum (CH), were prepared using 20% enriched U instead of Pu. The diameter of YSZ particles in SD and CD fuels was about 250μm. The fuels were irradiated in JRR-3 for about 100 days. The composition of SD fuel and its results of burnup calculation by SRAC95 code system are listed in Table 9.2.1. The estimated burnups are about 0.1MWD·cm⁻³, which is corresponding to about 10GWD·t⁻¹ of LWR UO₂ fuel.

| Table 9.2.1. SD fuel composition in mol% and estimated irradiation condition |
|-----------------|----------------|----------------|----------------|----------------|
| Composition     | Stabilized zirconia | UO₂          | AlO₁₅      | MgO           |
| (mol%)          |                  |              |             |               |
| 11.90           | 19.71           | 45.60        | 22.79       |

<table>
<thead>
<tr>
<th>Linear power (kW·m⁻¹)</th>
<th>Temperature (K) at</th>
<th>Burnup 235U (%)</th>
<th>(MWD·cm⁻³)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Surface</td>
<td>Center</td>
<td></td>
</tr>
<tr>
<td>23.0</td>
<td>1250</td>
<td>1740</td>
<td>23.28</td>
</tr>
</tbody>
</table>

The ceramography of the polished fuel specimens using X-ray diffraction analysis (XRD), scanning electron microscopy (SEM) and electron probe micro-analysis (EPMA) were carried out. Figure 9.2.1 show SEM images of SD fuels and line profiles of U, Zr, Al and Mg. From the SEM appearance, about 0.5mm diameter central holes existed and the YSZ particles of the SD fuel were concentrated to the center of the pellet, and the particles were deformed. The fuel structures and components were varied concentrically, and lots of pores were generated in center and middle regions. However, there is little change for the appearance of before and after irradiation fuel in the outer region. A magnified image of the
YSZ particle and matrix boundary on the SD fuels is also shown in Figure 9.2.1. In the surface area of the particle, gray layer of about 10 μm thickness was observed. The layer was composed of fine particles (<1μm in diameter) of spinel and YSZ. From the line profile results, it was confirmed that the irradiated SD fuel pellet comprised roughly three regions. The center region consisted of only YSZ (about 0.5 - 1.2 mm from the center of the pellet), the middle region consisted of YSZ and corundum (about 1.2 - 1.7 mm) and the outer region consisted of YSZ and spinel (about 1.7 mm surface). It was also revealed that Al in the middle region existed not as spinel but as corundum by comparing the distribution of Al and Mg and that Mg oxide segregated in the gap of the pellet and the cladding. The SH fuel pellet also consisted of similar structures; YSZ particle region, YSZ particle and corundum region, and YSZ particle and spinel region in addition to the MgO segregation.

It is considered that the occurrence of corundum phase in the middle region was caused by the decomposition of spinel by the fission damage and vaporization of MgO. The resistance of spinel to neutron irradiation is high, but that to fission fragment is not sufficient. In addition, vapor pressure of MgO is highest among the main cation oxides in the SD fuel as shown in Figure 9.2.2. Because of the high irradiation temperature, the large temperature gradient (~200K/mm) and irradiation field as well as the high vapor pressure, MgO produced by decomposition of the spinel was vaporized, and moved to the gap and deposited there at lower temperature. By the vaporization of the MgO, the spinel was transformed into the corundum. For the central hole occurrence, voids migration generated by the MgO vaporization led to void concentration and the central hole generation. The YSZ particles and corundum existed in the center area were pushed out by the central hole generation, and amorphous and semifluidic corundum would migrate faster than YSZ particles to the middle region, and as a result, YSZ were aggregated in the center area.

From the temperature distribution estimation in the SD fuel pellet at the beginning of irradiation, it is expected that the average irradiation temperature was over 1700K, where the vaporization of MgO occurred, in the central and the middle regions. On the other hand, the estimated temperature was much below 1700K in the outer region where the damaged area by fission fragments was limited to 10μm gray layer around the YSZ particles. Therefore, when the irradiation temperature is sufficiently lower than 1700K, the radiation damage of spinel-based fuel will be limited to the gray layer throughout the fuel pellet.
For SEM-EPMA results of Z, CD and CH fuels, there was no significant change compared with non-irradiated ones, however several radial cracks occurred and central region of the pellet revealed slightly porous microstructure, as observed in the LWR-UO₂ fuels. From XRD, deformation of the diffraction peak and phase separation of the fluorite phase as well as the corundum amorphization was not observed for the Z, CD and CH fuels. However it is known that the neutron resistance of corundum is inferior to that of spinel and corundum becomes amorphous by fission fragments, it is considerable that the damaged crystal structure recrystallizes because of the high irradiation temperature to anneal the amorphous corundum.

Figure 9.2.1
SEM images and line profile of U, Zr, Al and Mg. The white line indicates the line profile position and the right image shows the magnified one at the white square in the left image.

Figure 9.2.2
Vapor pressure of the main oxides
9.3 Burnup Analyses of Rock-Like Fuels Irradiated in the JRR-3

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Burnup analysis of Rock-Like (ROX) fuels has been done. Uranium based ROX fuels were irradiated at the Japan Research Reactor No.3 (JRR-3). After the irradiation, several post irradiation examinations (PIEs) were performed. Calculations were performed with the SRAC code system. Main obtained results are burnup, linear power, isotope abundance of Xenon and Krypton, and production ratio of Xenon to Krypton. The last two items were compared with measured data. Calculated values showed good agreement with those of the measurement.

Fuel and Irradiation

Five kinds of uranium based ROX fuels named SH, SD, CH, CD and Z were fabricated.\(^\text{13}\) SH and CH are homogenous fuels they are macroscopically homogenous mixture of yttoria-stabilized zirconia (YSZ) and an inert matrix material such as spinel or corundum. SD and CD are particle dispersed type fuels which consist of YSZ particles and an inert matrix material. Z fuel is an YSZ single phase fuel.

The fuels have been irradiated in the BR-1 irradiation hole located in the beryllium reflector of the JRR-3 for about 98 equivalent full power day (EFPD). After the irradiation, several PIEs were performed.\(^\text{21}\)

Calculation

First step of the calculations, neutron spectrum of the irradiation hole was calculated by the SRAC code system\(^\text{19}\). Using PIJ in the SRAC, that is a module based on the collision probability method, effective homogenized cross sections of the fuel element of the JRR-3 were generated. Number of energy group was 91. After the cell calculation, a whole core diffusion calculation in 2 dimension and 91 energy group was done using the cross sections. CITATION module in the SRAC system was used for the calculation. From the calculation, the 91 group neutron energy spectrum was obtained. It was used as the fixed boundary source for the burnup calculation.
PIJ was used for the burnup calculation. Fig. 9.3.1 shows the calculation model. Not only a ROX fuel and capsule structures but also a cobalt fluence monitor is included in it. Vertical direction was treated as infinity. Narrow gaps between capsule structures were smeared. Using the model, a fixed source problem was solved in 91 energy group. The neutron spectrum already obtained was used as the outer boundary neutron source into the capsule. A fine group flux distribution and effective cross sections of the ROX fuel were obtained from the calculation and they were collapsed into 1 energy group. Nuclide depletion calculation was performed using the 1 group cross sections and the flux. A power level of the calculation was normalized to have a same reaction rate of the fluence monitor as that of the measurement. Nuclide densities of the fuels were calculated by this calculation and used for the next step fixed source calculation. Repeating these steps, results of the burnup calculations were obtained.

Results

Table 9.3.1 shows the calculated burnup and liner power of the ROX fuels. Burnup of the fuels are 21~24 % U-235. They are equivalent to burnup of 10~12GWh/t of present light water reactors. Measurements of burnup will be carried out and results will be compared with calculated values. Linear power was used to estimate the fuel temperature that is important to stability of the fuel.

Table 9.3.2 shows abundance of Xe and Kr. Calculated values agree with measured values well. Table 9.3.3 shows production ratio of Xe to Kr (Xe/Kr). Calculated Xe/Kr values also agree with measured values except Z fuel. Measured value of Z fuel is smaller than those of the other fuels. It may have large measurement error because sampled gas volume of Z fuel is small and is not enough.

References

Table 9.3.1 Calculated Burnup and Linear Power

<table>
<thead>
<tr>
<th>ROX Name</th>
<th>CH</th>
<th>CD</th>
<th>SH</th>
<th>SD</th>
<th>Z</th>
</tr>
</thead>
<tbody>
<tr>
<td>Burnup</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>(% FIMA)</td>
<td>4.18</td>
<td>4.07</td>
<td>4.23</td>
<td>3.63</td>
<td>3.66</td>
</tr>
<tr>
<td>(% U-235)</td>
<td>23.89</td>
<td>23.28</td>
<td>24.15</td>
<td>20.87</td>
<td>21.01</td>
</tr>
<tr>
<td>Linear Power (MW/cm)</td>
<td>2.07x10^4</td>
<td>2.49x10^4</td>
<td>2.34x10^4</td>
<td>2.30x10^4</td>
<td>1.39x10^4</td>
</tr>
</tbody>
</table>

Table 9.3.2 Abundance of Xe and Kr after Irradiation (%)

<table>
<thead>
<tr>
<th>Element</th>
<th>Xe</th>
<th>Kr</th>
</tr>
</thead>
<tbody>
<tr>
<td>Mass Number</td>
<td></td>
<td></td>
</tr>
<tr>
<td>CH</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Meas.</td>
<td>130</td>
<td>83</td>
</tr>
<tr>
<td>Cal.</td>
<td>0.9</td>
<td>1.5</td>
</tr>
<tr>
<td>CD</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Meas.</td>
<td>0.0</td>
<td>10.3</td>
</tr>
<tr>
<td>Cal.</td>
<td>0.0</td>
<td>10.4</td>
</tr>
<tr>
<td>SH</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Meas.</td>
<td>0.0</td>
<td>10.4</td>
</tr>
<tr>
<td>Cal.</td>
<td>0.0</td>
<td>10.4</td>
</tr>
<tr>
<td>SD</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Meas.</td>
<td>0.1</td>
<td>10.4</td>
</tr>
<tr>
<td>Cal.</td>
<td>0.0</td>
<td>10.4</td>
</tr>
<tr>
<td>Z</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Meas.</td>
<td>0.2</td>
<td>10.9</td>
</tr>
<tr>
<td>Cal.</td>
<td>0.0</td>
<td>10.5</td>
</tr>
</tbody>
</table>

Meas.: Measurement, Cal.: Calculation

Table 9.3.3 Production Ratio of Xe to Kr (Xe/Kr)

<table>
<thead>
<tr>
<th>ROX Name</th>
<th>CH</th>
<th>CD</th>
<th>SH</th>
<th>SD</th>
<th>Z</th>
</tr>
</thead>
<tbody>
<tr>
<td>Measurement</td>
<td>7.1</td>
<td>7.2</td>
<td>7.3</td>
<td>7.2</td>
<td>5.4</td>
</tr>
<tr>
<td>Calculation</td>
<td>7.1</td>
<td>7.2</td>
<td>7.2</td>
<td>7.2</td>
<td>7.1</td>
</tr>
</tbody>
</table>

Fig. 9.3.1 Calculation Model of Irradiation Capsule

\( 40.0 \text{ mm} \)

Sample Case

Outer Pipe

Heat Transfer Media

Inner Pipe

Fluence Monitor

Clad Tube

U-ROX
9.4 Behavior of Particle-Dispersed ROX Fuel under RIA conditions

K. Kusagaya, T. Nakamura, H. Uetsuka and T. Yamashita
(E-mail: yamasita@popsvr.tokai.jaeri.go.jp)

Rock-like oxide (ROX) fuels which contain yttria stabilized zirconia (YSZ) as a host phase for fissile Pu can transmute Pu with high rate. However, their small Doppler feedback may cause a large fuel enthalpy deposite under a reactivity initiated accident (RIA) resulting in a severe fuel damage. Fuel behavior must be clarified to assess the safety under RIA conditions, which will supply valuable information to optimize the ROX fuel specifications and core characteristics. Pulse irradiation tests simulating RIA conditions were performed for three ROX fuels at the Nuclear Safety Research Reactor (NSRR) in JAERI. Those results on the powder mixed YSZ/spinel composite (SH) fuel and the YSZ single phase (Z) fuel were reported earlier\textsuperscript{1,2}, and the behavior of particle-dispersed YSZ/spinel (SD) fuel is described here.

A short fuel rod with a 17x17 PWR type design, containing 14 ROX pellets, was pulse irradiated in an experimental capsule filled with water. The specifications of the test fuel rod are listed in Table 9.4.1. Particle-dispersed test fuel pellets were fabricated using enriched uranium instead of plutonium for easier fuel handling. The irradiation test was carried out twice at different peak fuel enthalpies (E\textsubscript{p}) of 8.9 and 12.2 GJ/m\textsuperscript{3} by natural pulse operation of half widths of 5.1 and 6.6 ms at ambient temperature and pressure. Surface temperature of cladding, capsule pressure and water hammer were continuously monitored by various sensors during the pulse test. Ceramographs on several cross sections of the test rod were also taken after the test to investigate a change of micro-structures.

<table>
<thead>
<tr>
<th>Element</th>
<th>Over/ Stack length</th>
<th>279mm/ 135 mm</th>
</tr>
</thead>
<tbody>
<tr>
<td>Pellet</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Material composition</td>
<td>UO\textsubscript{2}:YSZ:MgAl\textsubscript{2}O\textsubscript{4} = 37:20:43 mol%</td>
<td></td>
</tr>
<tr>
<td></td>
<td>YSZ = ZrO\textsubscript{2}:Y\textsubscript{2}O\textsubscript{3} = 81: 19 mol%</td>
<td></td>
</tr>
<tr>
<td></td>
<td>235\textsuperscript{U} enrichment</td>
<td>20%</td>
</tr>
<tr>
<td></td>
<td>Diameter/ Length</td>
<td>8.05 mm/ 9 mm (P/C radial gap = 0.085 mm)</td>
</tr>
<tr>
<td>Cladding</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Material</td>
<td>Zry-4</td>
<td></td>
</tr>
<tr>
<td>Outer/ Inner diameter</td>
<td>9.5 mm/ 8.22 mm (thickness = 0.64 mm)</td>
<td></td>
</tr>
</tbody>
</table>
The cladding surface temperature ($T_{CS}$) reached the maximum during 1~2 seconds after a reactivity pulse was inserted and increased with increasing $E_p$ values. Observed maximum temperatures at cladding surface are plotted in Fig. 9.4.1 against inserted $E_p$ together with those for SH (large circles), Z (triangles) and UO$_2$ (small circles) fuels. The relation between $T_{CS}$ and $E_p$ for the SD fuels seems to be similar to that for the SH fuels and can be represented well by FRAP-T6 code$^3$ (a solid line). The cladding surface temperature for the SD and SH fuels shows a lower value by about 200 K than those for the Z and UO$_2$ fuels, which may be attributed to the lower melting temperature (2210 K) for the spinel/YSZ mixture than those for the Z (2820 K) and the UO$_2$ (3110 K) fuels.

![Graph](image)

Fig. 9.4.1 Relation between cladding surface temperature and peak fuel enthalpy

Rod failure was observed at $E_p = 12.2$ GJ/m$^3$ for the particle-dispersed fuel. Test results are summarized in Fig. 9.4.2 for all ROX fuels. Open marks show that the fuel rods were intact, while filled marks show that the fuel rods were failed. The horizontal line of 9.6 GJ/m$^3$ indicates the current rod failure threshold for the UO$_2$ fuel rod. The rod failure threshold for the ROX fuel seems to be above 10 GJ/m$^3$ and slightly higher that that of the UO$_2$ fuel.

Fuel failure behavior for the ROX fuel is quite different from that of the UO$_2$ fuel. The failure of the UO$_2$ fuel occurs during the quenching. A thinner part of the once molten and embrittled cladding was broken mainly due to thermal stress and the fuel itself remains solid. A large mechanical energy is released by the reaction between hot UO$_2$ fuel and coolant water. On the other hand, the failure of the ROX fuel occurred due to cladding burst when the
cladding temperature peaked. From ceramographs of the cross section of the failed rods, little indication of cladding melt was observed for YSZ/spinel (the SH and SD) fuel rods and the cladding damage was quite limited except for the burst opening. Lower fuel melting temperature (2210 K) of the SD and SH fuels than that (3110 K) of the UO$_2$ fuel should have contributed to the different failure modes. A significant fraction, of about 50 to 70%, of partly molten fuel was ejected out through the burst opening into the surrounding water. However, generation of pressure pulse or water hammer due to molten fuel/water interaction was not detected. Relatively mild fuel ejection during high temperature burst might have prevented the mechanical energy generation.

References:

9.5 Accidents Analysis of Thoria Based Rock-like Oxide Fuel PWR

H. Akie and R. Okawa*

(E-mail : akie@mike.tokai.jaeri.go.jp)

For the improvement of severe transient behaviors of a zirconia based ROX (Zr-ROX : PuO₂-(Zr,Y)O₂-MgAl₂O₄) PWR core, it was found effective to use the additive of 8-25 mol% UO₂ or ThO₂ in Zr-ROX fuel.¹ These additives improve not only small negative reactivity coefficients but also a large power peaking factor of a Zr-ROX core. At the same time of Zr-ROX, there has been proposed another type of ROX fuel based on thoria (Th-ROX : PuO₂-ThO₂-MgAl₂O₄)², which contains more than 40 mol% of ThO₂. A Th-ROX fueled PWR core is expected to also have favorable physics and transient characteristics.

Reactor physics parameters of a Th-ROX fueled PWR core have already been estimated³. In Table 9.5.1 are shown some results for weapons-grade and reactor-grade Pu (W-Pu and R-Pu) Th-ROX cores at BOC. In this table, the fuel temperature reactivity, which is the most important safety parameter in the Zr-ROX fueled cores, is sufficiently negative in the Th-ROX cores as low as that of the UO₂ core. The Th-ROX fueled cores also have a larger negative void reactivity in comparison with the Zr-ROX core. The power peaking factor of Th-ROX core becomes the comparable value to that of UO₂ core.

By using these reactivity coefficients, analyses of reactivity initiated accident (RIA) and loss of coolant accident (LOCA) were performed for W-Pu and R-Pu Th-ROX fueled PWR.

<table>
<thead>
<tr>
<th>Table 9.5.1 Fuel temperature and void reactivities (%Δk/k) and power peaking factor of Th-ROX and Zr-ROX fueled cores in comparison with UO₂ core (BOC)</th>
</tr>
</thead>
<tbody>
<tr>
<td>weapons-Pu</td>
</tr>
<tr>
<td>Zr-ROX</td>
</tr>
<tr>
<td>Fuel temp. reac. 600→900K</td>
</tr>
<tr>
<td>Void reac. 0→40% void</td>
</tr>
<tr>
<td>Peaking</td>
</tr>
</tbody>
</table>

* The Japan Research Institute, Limited
The fuel properties used in the analyses are shown in Table 9.5.2. These properties are theoretical values estimated based on the properties of the components in Th-ROX for the molecular composition of $\text{ThO}_2: \text{PuO}_2: \text{MgAl}_2\text{O}_4 = 0.45:0.05:0.5$. Figure 9.5.1 shows the temperature dependencies of the thermal conductivity and the specific heat.

<table>
<thead>
<tr>
<th>Property</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Specific heat (kJ/kg/K)</td>
<td>0.62 (at 800K)</td>
</tr>
<tr>
<td>Thermal conductivity (W/m/K)</td>
<td>5.5 (at 800K)</td>
</tr>
<tr>
<td>Density ($10^3$ kg/m$^3$)</td>
<td>5.9 (95% TD)</td>
</tr>
<tr>
<td>Melting temperature (K)</td>
<td>2300</td>
</tr>
</tbody>
</table>

![Graph showing thermal conductivity and specific heat](image)

Fig. 9.5.1 Thermal conductivity and specific heat of T-ROX fuel

The RIA event under hot zero power and hot full power conditions, and a cold-leg large break LOCA in Th-ROX fueled PWR were analyzed by using EUREKA-2$^4$ and RETRAN2$^5$ codes, respectively, both for BOC and EOC conditions. The calculation models for the accident analyses are based on the conventional 4-loop 1100 MWe class 17×17 type PWR plant. The results are summarized in Table 9.5.3, with those of the Zr-ROX fuels with and without UO$_2$ additive and the UO$_2$ fuel estimated with the same calculation method.

Under RIA condition in the core of Zr-ROX fuel without additives, the maximum fuel enthalpy becomes far larger than the limiting value in the conventional UO$_2$ PWR of about 1MJ/kg, and the maximum fuel temperature also rises much higher than the melting temperature of about 2200K. In the Th-ROX fuel, the fuel enthalpy and temperature are both
well improved to the similar value to the UO\(_2\) added Zr-ROX fuel. The enthalpy value per unit mass in the ROX fuels are still 2 times larger than the value in UO\(_2\) fuel. In the NSRR pulse experiments of ROX fuels, it is reported that the ROX fuel pin failure condition is comparable to that of UO\(_2\) fuel pin, in terms of the fuel enthalpy per unit volume\(^6\). By taking into account the density difference of Th-ROX from UO\(_2\), the volumetric enthalpy in the Th-ROX fuel approaches to the value in the UO\(_2\) fuel.

In the LOCA analysis of Zr-ROX fuel core without additives, the cladding temperature becomes higher than the limit of 1470K (1200°C). The cladding temperature is sufficiently decreased to less than 1470K in the Th-ROX fueled core.

<p>| Table 9.5.3 Maximum fuel enthalpy and fuel temperature in RIA and peak cladding temperature in LOCA of Th-ROX and Zr-ROX fueled cores in comparison with UO(_2) core |
|---------------------------------|-----------------|-----------------|-----------------|-----------------|</p>
<table>
<thead>
<tr>
<th>Enthalpy (MJ/kg)</th>
<th>Enthalpy (GJ/m(^3))</th>
<th>Fuel temp. (K)</th>
<th>Cladding temp. (K)</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>W-Pu</strong></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Zr-ROX</td>
<td>&gt;&gt;1.0</td>
<td>&gt;&gt;2200</td>
<td>&gt;1470</td>
</tr>
<tr>
<td>Zr-ROX-15UO(_2)</td>
<td>0.81</td>
<td>4.5</td>
<td>1700</td>
</tr>
<tr>
<td>Th-ROX</td>
<td>0.80</td>
<td>4.7</td>
<td>1580</td>
</tr>
<tr>
<td><strong>R-Pu</strong></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Zr-ROX-8UO(_2)</td>
<td>0.80</td>
<td>4.5</td>
<td>1700</td>
</tr>
<tr>
<td>Th-ROX</td>
<td>0.76</td>
<td>4.5</td>
<td>1500</td>
</tr>
<tr>
<td><strong>UO(_2)</strong></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>0.39</td>
<td>4.3</td>
<td>2080</td>
</tr>
</tbody>
</table>

Reference:
10. Nitride Fuel and Related Pyrochemical Technology

This study aims at developing nitride fuel cycle for transmutation of long-lived minor actinides (MAs) and advanced fast reactors. The activity of R&D includes the fabrication and property measurements of nitride fuel, irradiation tests and molten salt electrorefining in relation to pyrochemical reprocessing. In addition, some basic properties of actinide compounds are investigated in order to contribute to preparation of actinide thermodynamic database.

Two (U,Pu)N and one (U,Pu)C fuel pins irradiated at fast test reactor JOYO were subjected post irradiation examinations under joint research with JNC. Non-destructive examinations were completed and fuel performance under the irradiation condition was demonstrated. Furthermore, information on fuel stack elongation, radial and axial distribution of fission products, radial profile and bowing of fuel pins, fission gas release from fuel pellets and so on was obtained. Following non-destructive examinations, the fuel pins were transported to JAERI Tokai for destructive examinations.

Thermochemical measurements have been carried out for neptunium bearing nitrides. At present, most of fundamental experimental data are lacking for them. Heat capacity of NpN was measured by DSC (differential scanning calorimeter) from 323 to 1023K. Compared with drop or adiabatic calorimetry, differential scanning calorimetry has advantage of compactness of the apparatus and small amount of samples suitable for highly radiotoxic test materials. On the other hand, it was considered to have problems in accuracy at elevated temperatures. However, by use of a sophisticated manner, heat capacity of NpN could be measured with enough reproducibility. The results show that heat capacities of UN, NpN and PuN have almost the same value over the temperature range investigated.

In pyrochemical reprocessing of nitride and metal fuels, liquid metal such as Cd or Bi is usually used as cathode for recovering Pu and MAs together. Here the electro-deposition and dissolution behavior of Np at liquid Cd and Bi electrodes in LiCl-KCl eutectic melts was investigated by cyclic voltammetry compared with that at solid cathode. The results were thermodynamically analyzed by a lowering of activity of Np at liquid Cd or Bi due to the formation of intermetallic compounds. Furthermore, following the demonstration of recovery of Pu into liquid Cd cathode with high concentration, co-deposition behavior of U and Pu into liquid Cd cathode at different U/Pu ratios (1/12.2 and 1/7.4) in the salt phase was examined. It was proved from the mass balance that U and Pu were successfully recovered at high current efficiency in both cases. Microprobe analysis after co-deposition suggests the formation of intermetallic compound of U, Pu and Cd. In addition, aluminum nitride, AlN seems to be a hopeful material for cathode crucible.
10.1 Post Irradiation Examinations of (U,Pu)N and (U,Pu)C Fuel Irradiated at Fast Test Reactor JOYO (1)

Research Group for Advanced Fuel
(E-mail: arai@popsvr.tokai.jaeri.go.jp)

The irradiation of He-bonded two (U₀₈Pu₀₂)N fuel pins and one (U₀₈Pu₀₂)C fuel pin at fast test reactor JOYO was completed in September 1999 under the joint research of JAERI and JNC named as "basic irradiation tests of carbide and nitride fuels for fast reactors". It was the first irradiation test of "advanced ceramic fuels" in Japan under fast neutron circumstances. In this campaign JAERI was mainly in charge of fabrication of fuel pellets and fuel pins, while JNC licensing of irradiation, assembling of irradiation rig and irradiation at JOYO. Design of the fuel pins was performed by both institutions together.

After cooling period for a few months, the fuel pins were transferred to FMF (Fuel Monitoring Facility) of JNC Oarai Engineering Center and subjected to non-destructive PIEs (Post Irradiation Examinations). The maximum linear power and burnup were calculated at ~780W/cm and ~40GWD/t, while the maximum temperature of cladding tube (PNC 1520) during the irradiation was evaluated at 912K. Any failure of the fuel pins has not been observed. Items of non-destructive PIEs of the fuel pins include inspection of external appearance, X-ray radiography, gamma-ray scanning, measurements of weight, length, bowing and diameter besides puncture test. These results are summarized below.

Inspection of external appearance of the fuel pins revealed slight discoloration and traces of contact with inner surface of the compartment. It is considered, however, that these phenomena did not affect the fuel performance itself. The situation inside the pins could be clarified by X-ray radiography. Although some cracking of the pellets was observed, neither central hole of the pellets nor gap between each pellet was found for in all cases. Fuel stack length of (U,Pu)N and (U,Pu)C before and after irradiation is shown in Table 10.1.1. An elongation percentage observed lay between 1.4-1.8%, which is comparable with MOX pellets with ~93%T.D. but higher than those with ~85%T.D. at the similar burnup. These results were derived from higher swelling rate of nitride and carbide than MOX pellets. Weight of the fuel pins before and after irradiation is also shown in Table 10.1.1. Significant change of the weight was not found as anticipated. Gamma-ray scanning was carried out to obtain information on the distribution of typical fission products such as Rh-106, Zr-95 and Cs-137. It was observed that part of Cs-137 migrated to plenum region along temperature gradient. Other fission products such as Rh-106, Zr-95, Pr-144 and Ru-103 remained within fuel stack and the distribution was almost was almost flat corresponding
to axial burnup distribution (the maximum peaking was ~1.05). Distribution of the fission products within pellets was revealed by gamma-ray tomography. The results also showed almost uniform distribution of Rh-106 and Zr-95 and radial shift of Cs-137 to outer region. The results of measurement for the fuel pin length and bowing are shown in Table 10.1.2. No significant change of the fuel pin length before and after measurements was found and the bowing seemed similar with MOX pin in the same irradiation rig. Radial profiles of diameter of two (U,Pu)N pins are shown in Fig. 10.1.1. The difference of characteristics of the fuel pins exist in the initial He-gap between fuel pellet and cladding tube, namely 0.32mm for B9N01 and 0.17mm for B9N02 fuel pins. Nitride fuel pin with smaller gap width (B9N02) shows uniform increase of diameter, while that with larger gap width (B9N01) shows typical ovality. Carbide fuel pin with gap width of 0.18mm shows a similar radial profile with B9N02 pin. These phenomena seem to have an important relation with FCMI (Fuel and Clad Mechanical Interaction) that might become severe at higher burnups. Two (U,Pu)N fuel pins and one (U,Pu)C pin were subjected to puncture test to obtain information on FP gas release from fuel pellets. The results are shown in Table 10.1.3. Fission gas release for the fuel pins lay between 2.3-5.2%, which was much lower than MOX fuel irradiated under the similar irradiation condition. Isotopic ratio of Xe and Kr released was almost similar with the case of MOX fuel.

After completion of non-destructive PIEs mentioned above, destructive PIEs for two (U,Pu)N fuel pins have been started at FMF. In addition, part of two (U,Pu)N fuel pins and integral (U,Pu)C fuel pin were transported from JNC Oarai Engineering Center to JAERI Tokai for the detail destructive PIEs at RFEF (Reactor Fuel Examination Facility). The destructive PIEs are underway at both JAERI and JNC's hot cells. It is expected that basic irradiation behavior of advanced ceramic fuels under fast neutron circumstances is clarified by analyzing the results of non-destructive PIEs mentioned above and destructive ones obtained hereafter.

Acknowledgement

Authors wish to express their thanks to the staff of JNC Oarai Engineering Center and RFEF of JAERI Tokai relating to this irradiation campaign for valuable information and comments.
Table 10.1.1  Fuel stack length and fuel pin weight before and after irradiation

<table>
<thead>
<tr>
<th>Pin No.</th>
<th>Fuel</th>
<th>Fuel stack length (mm) before</th>
<th>Fuel stack length (mm) after</th>
<th>Difference (mm) (%)</th>
<th>Fuel pin weight (g) before</th>
<th>Fuel pin weight (g) after</th>
<th>Difference (g)</th>
</tr>
</thead>
<tbody>
<tr>
<td>B9N01</td>
<td>(U,Pu)N</td>
<td>200.1</td>
<td>203.7</td>
<td>+3.6 (1.8)</td>
<td>630.2</td>
<td>630.0</td>
<td>-0.2</td>
</tr>
<tr>
<td>B9N02</td>
<td>(U,Pu)N</td>
<td>198.8</td>
<td>201.5</td>
<td>+2.7 (1.4)</td>
<td>637.6</td>
<td>637.3</td>
<td>-0.3</td>
</tr>
<tr>
<td>B9C01</td>
<td>(U,Pu)C</td>
<td>199.5</td>
<td>202.5</td>
<td>+3.0 (1.5)</td>
<td>625.7</td>
<td>625.3</td>
<td>-0.4</td>
</tr>
</tbody>
</table>

Table 10.1.2  Fuel pin length before and after irradiation, and the maximum fuel pin bowing

<table>
<thead>
<tr>
<th>Pin No.</th>
<th>Fuel</th>
<th>Fuel pin length (mm) before</th>
<th>Fuel pin length (mm) after</th>
<th>Difference (mm)</th>
<th>Maximum fuel pin bowing</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>before</td>
<td>after</td>
<td>position(mm)</td>
<td>angle(°) bowing(mm)</td>
</tr>
<tr>
<td>B9N01</td>
<td>(U,Pu)N</td>
<td>1532.5</td>
<td>1532.0</td>
<td>-0.5</td>
<td>626.4</td>
</tr>
<tr>
<td>B9N02</td>
<td>(U,Pu)N</td>
<td>1532.5</td>
<td>1532.3</td>
<td>-0.2</td>
<td>646.0</td>
</tr>
<tr>
<td>B9C01</td>
<td>(U,Pu)C</td>
<td>1532.5</td>
<td>1532.6</td>
<td>+0.1</td>
<td>634.1</td>
</tr>
</tbody>
</table>

Table 10.1.3  Results of puncture test for fuel pins

<table>
<thead>
<tr>
<th>Pin No.</th>
<th>Fuel</th>
<th>Burnup (at%)*1</th>
<th>Pressure (Torr) *2</th>
<th>Gas volume (cc) *2</th>
<th>Gas composition (%)</th>
<th>FGR (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td>Kr</td>
<td>Xe</td>
</tr>
<tr>
<td>B9N01</td>
<td>(U,Pu)N</td>
<td>4.20</td>
<td>1165.6</td>
<td>14.66</td>
<td>2.2</td>
<td>19.3</td>
</tr>
<tr>
<td>B9N02</td>
<td>(U,Pu)N</td>
<td>4.19</td>
<td>1311.8</td>
<td>15.58</td>
<td>3.5</td>
<td>29.8</td>
</tr>
<tr>
<td>B9C01</td>
<td>(U,Pu)C</td>
<td>4.27</td>
<td>881.17</td>
<td>11.14</td>
<td>2.0</td>
<td>17.4</td>
</tr>
</tbody>
</table>

*1 Averaged value
*2 At STP

Fig.10.1.1  Radial diameter profile of irradiated (U,Pu)N fuel pins with He-gap width of 0.32mm (left, B9N01) and 0.17mm (right, B9N02).
10.2 Heat capacity of neptunium mononitride

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Nitride fuels are considered as candidate fuels for transmuting minor actinides by fast reactors and accelerator driven systems\textsuperscript{1).} Neptunium is one of the key elements among minor actinides because of its large accumulated amount in spent fuels and long half-life compared to those of other elements such as americium and curium. Therefore, knowing the heat capacity of neptunium mononitride, NpN, is essential for thermochemical studies of nitride fuels for transmutation. At present, experimental data of thermodynamic properties of NpN are almost lacking. Estimates of the heat capacity of NpN\textsuperscript{2)} and the Gibbs energy of formation of NpN evaluated from the vaporization behavior of NpN\textsuperscript{3,4)} have been reported by the authors. However, measurements of the heat capacity have not been reported. Drop calorimetry and adiabatic calorimetry have been mainly used for high temperature heat capacity measurements. On the other hand, differential scanning calorimetry is suitable for heat capacity measurements of radiotoxic materials because it has a compact and simple mechanism, which leads to ease of installing into gloveboxes or hot cells, and smaller samples are required. So heat capacity of neptunium mononitride was investigated with a differential scanning calorimeter(DSC). Furthermore, the DSC measurements of uranium mononitride, UN, plutonium mononitride, PuN and uranium dioxide, UO\textsubscript{2}, were carried out in order to make sure of the accuracy of our instrument.

The samples of NpN was prepared by carbothermic reduction of neptunium dioxide, NpO\textsubscript{2}. The synthesized NpN was then pulverized and pressed into a green disk under a pressure of about 400MPa and heated again for sintering at 2003K in a Ar-8%H\textsubscript{2} mixed gas stream. The dimension of this disk was about 4 mm in diameter and about 1 mm in height. The sintered UN, PuN and UO\textsubscript{2} sample disks were also prepared in a similar manner. The differential scanning calorimeter(DSC-8270H) manufactured by Rigaku Co. Ltd is modified by separating the furnace into a glovebox with a high-purity argon gas atmosphere where oxygen and moisture contents are kept less than 1 ppm. This calorimeter is a heat flux type DSC and its sensors are constructed of Pt/13Rh alloy thermocouples. So the temperature differences between reference holder and sample holder, which build up due to different heat capacities, are measured by the thermocouples. The DSC measurements were made in the temperature range from 323 to 1023K in a high-purity argon atmosphere with a flow rate of 100 ml/min. Specific heats were determined using the standard ratio method. A measurement of heat capacity consists of baseline scan, standard scan and sample scan as shown in Fig. 10. 2. 1. The baseline scan is obtained by scanning empty pans put in the reference and sample holder, respectively. The standard scan is obtained by
scanning sapphire enclosed with the lid of the pan. These three successive scans should be carried out under the same experimental conditions by taking care of the sample geometry. But the samples are not always placed at the same positions of sample holder as in the preceding scan and these different positions among three successive scans may lead to the deviation of isotherm baselines as shown in Fig.10. 2. 1, where isotherm baselines are obtained by holding the furnace temperature for 30min and the heating portion is programmed at the rate of 10K/min. So the temperature range of interest is divided into intervals of 100K and separate heat capacity runs are made for each interval checking coincidences of these isotherm baselines. Heat capacities measured over these intervals are then combined to give a continuous function of heat capacity vs. temperature.

More than six heat capacity runs were made for UN, PuN and UO₂ in order to check the reproducibilities of experimental data obtained from our instrument. These results of specific heat of UN, PuN and UO₂ are plotted in Fig. 10. 2-10. 2. 4 together with the reference data given by Oetting⁵,⁶ and Cordfunke⁷. Fitting of the experimental data were carried out by the least-square method, which gave the following equations:

\[
\begin{align*}
\text{UN: } & \text{C}_\text{p}(\text{J/mol·K})=50.5+1.06\times10^{-2}T+1.77\times10^{-6}T^2-7.45\times10^5/T^2 \\
\text{PuN: } & \text{C}_\text{p}(\text{J/mol·K})=53.3-1.28\times10^{-3}T+1.24\times10^{-5}T^2-6.39\times10^5/T^2 \\
\text{UO}_2: & \text{C}_\text{p}(\text{J/mol·K})=55.3+4.43\times10^{-2}T-1.54\times10^{-5}T^2-4.03\times10^5/T^2
\end{align*}
\]

The deviations of the present data of UN, PuN and UO₂ from the calculated ones are 4.5%, 5.6% and 5.9%, respectively. These fitting curves are also indicated with bold lines in these figures and in good agreement of the reference data. The calculated results for UN, PuN and UO₂ agree with the reference data with an accuracy of 2.3%, 2.7% and 2.7%, respectively. On the other hand, five heat capacity runs for NpN were made and these results are plotted in Fig. 10. 2. 5. The heat capacity of NpN obtained in this study is expressed by the following equation:

\[
\text{NpN: } \text{C}_\text{p}(\text{J/mol·K})=43.4+2.31\times10^{-3}T-7.62\times10^{-6}T^2-9.08\times10^4/T^2
\]

The deviation from the calculated one is 3.6% and the fitting curve is in good agreement with the reference data of UN and PuN as shown in this figure.

References:
7) Cordfunke, E.H.P., Koning, R.J.M. (Eds.) : "Thermodynamical Data for Materials and Fission

Fig. 10.2.1 Typical DSC curves of base line, standard and UN scans

Fig. 10.2.2 Specific heat of UN

Fig. 10.2.3 Specific heat of PuN

Fig.10.2.4 Specific heat of UO₂

Fig. 10.2.5 Specific heat of NpN
10. 3 Electrode Reactions of Np at Liquid Cd and Bi Cathodes in LiCl-KCl Eutectic Melts

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In the electrorefining process that is a key step in the pyrochemical reprocessing, spent metallic and nitride fuels are anodically dissolved in LiCl-KCl eutectic melt, and actinides are recovered at a cathode.1, 2) It has been proposed that Pu and minor actinides (MA) would be recovered as alloys at the liquid metal cathodes such as Cd, Bi, etc.3) In the present work, the electrode reactions of the Np³⁺/Np couple at the surface of liquid Cd and Bi were investigated, compared with the case of solid Mo electrode. The difference between the Np³⁺/Np redox potentials at the solid electrode and at liquid Cd or liquid Bi electrode was thermodynamically interpreted by the formation energy of Np-Cd alloy or Np-Bi alloy.

Cyclic voltammograms for the redox reaction of the Np³⁺/Np couple were shown in Fig. 10. 3. 1. Curve 1 shows the voltammogram for the redox reaction at the interface between the LiCl-KCl eutectic melt containing 0.333 wt % NP₃ and liquid Cd at a scan rate of 0.01 Vs⁻¹ at 723 K. Curve 2 shows the voltammogram obtained at the interface between the LiCl-KCl melt in the absence of NP₃ and liquid Cd. The initial rise observed near -0.6 V corresponded to the oxidation of Cd (Cd = Cd²⁺ + 2e⁻); this was confirmed by changing the concentration of CdCl₂ in the salt phase. The final descents observed around -1.4 V in curves 1 and 2 corresponded to the reduction of LiCl (Li = Li⁺ + e⁻). There are a cathodic and an anodic peaks in curve 1. In the region of the potential scanning rate between 0.01 and 0.1 Vs⁻¹, the anodic and the cathodic peak potentials, Eₜp and Eₚₚ, remained almost constant. These peaks must be attributed to the redox reaction of the Np³⁺/Np couple, since the number of the transferred electron was calculated to be 2.8 - 2.9 from the relation between Eₜp and the half-peak potential, Eₚₚ, of the cathodic peak in the cyclic voltammogram. The cathodic peak current increases with an increase in temperature. This temperature dependence can be understood by the variation of diffusion coefficient of Np³⁺ in the salt,4) since the cathodic peak current increases in proportion to the square root of the diffusion coefficient of Np³⁺ in the salt. The cathodic peak current increased linearly with the concentration of NP₃. The above characteristics indicate that the redox reaction of the Np³⁺/Np couple at liquid Cd electrode is almost irreversible.

On the other hand, there was no significant current peak within the potential range from -0.6 to -1.5 V in the case of Mo working electrode in NP₃-LiCl-KCl system as shown in curve 3 in Fig. 10.3.1. The cathodic and the anodic peaks observed around -1.6 V are ascribed to be the redox reaction of the

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Np\(^{3+}\)/Np couple at Mo electrode.\(^4\) The redox potentials of the Np\(^{3+}\)/Np couple at liquid Cd electrode at 723, 773 and 823 K appeared more positively by about 0.204, 0.191 and 0.169 V, respectively, than those at Mo electrode, irrespective of the concentration of NpCl\(_3\). It was supposed that the difference between the redox potential of the Np\(^{3+}\)/Np couple at liquid Cd electrode and that at Mo electrode was attributed to a lowering of the activity of Np in Cd phase. The potential difference can be explained by assuming the formation of NpCd\(_n\) or NpCd\(_n\).\(^5\) The most stable intermetallic compound among the Np-Cd system, NpCd\(_n\) (n=11 at 723 K and n = 6 at 773 and 823 K), will be formed in liquid Cd phase as Eq. (1).

\[ \text{Np} + n\text{Cd} = \text{NpCd}_n \]  

(1)

The thermodynamic equilibrium constant, K, in the above reaction is written as, \( K = [\text{NpCd}_n]/[\text{Np}][\text{Cd}]^n \), where square brackets, [ ], represent activities of NpCd\(_n\), Np and Cd, respectively. On the other hand, the redox potential of the Np\(^{3+}\)/Np couple at the solid electrode vs. Ag/AgCl reference electrode, \( E_{\text{Np}^{3+}/\text{Np}} \), is expressed as follows:

\[ E_{\text{Np}^{3+}/\text{Np}} = E_{\text{Np}^{3+}/\text{Np}}^0 + \frac{RT}{3F} \ln[\text{NpCl}_3]/[\text{Np}]. \]  

(2)

Accordingly, the redox potential of the Np\(^{3+}\)/Np couple at the liquid Cd electrode, \( E_{\text{Np}^{3+}/\text{Np} - \text{Cd}} \), is described as:

\[
E_{\text{Np}^{3+}/\text{Np} - \text{Cd}} = E_{\text{Np}^{3+}/\text{Np}}^0 + \frac{RT}{3F} \ln[K][\text{NpCl}_3][\text{Cd}]^n/[\text{NpCd}_n]
= E_{\text{Np}^{3+}/\text{Np}}^0 - \frac{\Delta G_{\text{NpCd}_n}^0}{3F}
+ \frac{RT}{3F} \ln[\text{NpCl}_3] + n\frac{RT}{3F} \ln[\text{Cd}] - \frac{RT}{3F} \ln[\text{NpCd}_n] \]  

(3)

where \( \Delta G_{\text{NpCd}_n}^0 \) is the standard Gibbs energy for formation of NpCd\(_n\) (\( \Delta G_{\text{NpCd}_n}^0 = -RT \ln K \)). Therefore, the potential difference, \( \Delta E \), between \( E_{\text{Np}^{3+}/\text{Np} - \text{Cd}} \) and \( E_{\text{Np}^{3+}/\text{Np}} \) is represented as Eq. (4):

\[ \Delta E = \frac{\Delta G_{\text{NpCd}_n}^0}{3F} + n\frac{RT}{3F} \ln[\text{Cd}] - \frac{RT}{3F} \ln[\text{NpCd}_n] \]  

(4)

Most of the deposited Np is expected to exist as NpCd\(_{11}\) at 723 K and NpCd\(_6\) at 773 and 823 K in the vicinity of the surface of liquid Cd phase. It seems reasonable to suppose that [NpCd\(_n\)] are close to unity locally. Since Cd is the principal constituent of the liquid metal phase, [Cd] must also be close to unity. If one takes into account the above assumptions, Eq. (4) can be simplified to Eq. (5):
\[ \Delta E \equiv - \Delta G_{\text{NpCld}}^0/3F. \]

From the obtained \( \Delta E \) values, \( \Delta G_{\text{NpCld}}^0 \) values at 723, 773 and 823 K were evaluated as 59.0, 55.3 and 48.9 kJmol\(^{-1}\), respectively.

As for Np-Bi system, the mechanism of electrode reaction of the Np\(^{3+}/\text{Np}\) couple at liquid Bi electrode can be the same as that at Cd electrode. Curve 1 in Fig. 10.3.2 shows the voltammogram for the redox reaction at the interface between the LiCl-KCl eutectic melt containing 0.333 wt % NpCl\(_3\) and liquid Bi at a scan rate of 0.01 Vs\(^{-1}\) at 723 K. Curve 2 in Fig. 10.3.2 shows the voltammogram obtained at the interface between the LiCl-KCl melt in the absence of NpCl\(_3\) and liquid Bi. In the case of Bi electrode, \( \Delta E \) values at 723, 773 and 823 K were obtained as 0.340, 0.315 and 0.290 V, respectively. The difference between the redox potential of the Np\(^{3+}/\text{Np}\) couple at liquid Bi electrode and that at Mo electrode was attributed to a lowering of the activity of Np in Bi phase, due to NpBi\(_2\) formation in Bi phase.\(^9\) Accordingly, \( \Delta G_{\text{NpBi}}^0 \) values at 723, 773 and 823 K were evaluated as 98.4, 91.2 and 83.9 kJmol\(^{-1}\).

References:
10.4 Plutonium and Uranium Recovery Experiments into Liquid Cadmium Cathodes

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In order to study the influence of U in the molten salt phase on the electrorecovery of Pu into a liquid cadmium cathode (LCC), we have conducted two experiments, Run-1 and Run-2, to recover U and Pu at different U/Pu ratios in the salt phase.

Fig. 10.4.1 shows the experimental apparatus. 1200 g of molten LiCl-KCl eutectic salt was contained in a 124 mm inner diameter iron vessel. The initial salt of Run-1 contained 0.36 wt% of U and 4.4 wt% of Pu, accordingly the Pu/U ratio was 1/12.2. In the case of Run-2, the salt contained 0.67 wt% of U and 5.0 wt% of Pu resulting the U/Pu ratio to be 1/7.4 initially. 1400 g of Cd was placed at the bottom of the vessel and used as an anode to supply U and Pu into the salt phase during the electrolysis. The initial concentrations of U and Pu in the Cd anode were 0.29 wt% of U and 1.46 wt% of Pu for run-1 and 0.30 wt% of U and 1.11 wt% of Pu for Run-2.

The cathode used in the present study is shown in Fig. 10.4.2. In each experiment, the cathode Cd was contained in an aluminum nitride (AlN) crucible, which was fabricated by Cercom. The AlN crucible contains 0.15 wt% of yttrium and less than 100 ppm of other metallic impurities. In order to remove the solidified cathode ingot after each experiment easily and to use the crucible repeatedly, the inside wall of the crucible had a taper of 4 degrees. An AlN stirrer was used for mixing around the boundary between the salt and the cathode Cd. The stirrer was attached to an iron shaft that ran at 40 rpm. This iron shaft worked both as an electrical contact to the cathode and as the rotating drive of the stirrer. In order to avoid deposition of U and Pu on the iron shaft, an AlN tube was placed on the stirrer as an electrical sheath. The diameter of the LCC was 42 mm at the surface.

The experiments were carried out in a high purity argon glove box. The experimental apparatus was lowered into an electric furnace that was placed in the glove box. The cell temperature was kept at 773K during the electrolysis.

The electrorecoveries were conducted at the cathode current density of 23.4 mA/cm²; this value was determined to avoid deposition of Li that is the salt component into the LCC. The LCC potentials versus an Ag/AgCl reference electrode in the salt phase during each electrolysis are

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shown in Fig. 10.4.3. The LCC potentials became more negative gradually until the LCCs were saturated with U and Pu. The quantities of electricity passed during the electrolysis corresponded to 15.6g of Pu for Run-1 and 15.8g of Pu for Run-2.

After each electrorecovery, the AlN crucible was pulled up from the salt phase and cooled for solidification. The cathode ingot of Run-1 formed a projection, whose maximum height was 19mm, as shown in Fig. 10.4.4. The cathode weight gain during Run-1, which corresponds to the amount of U and Pu recovered into the cathode, was 14.4g corresponding to the weight ratio of U and Pu in the cathode to be 10.7 wt% and the cathode current efficiency to be 92.0%.

In the case of Run-2, the weight gain of the cathode ingot was 20.8g, exceeding about 5g the quantity of electricity passed at Run-2. By cutting the Run-2 ingot, it was also shown that the ingot contained some amount of salt. Then, the existence of U might have affected the condition of the cathode Cd phase such as viscosity.

The weight ratio profiles of U and Pu in the Run-1 cathode are plotted in Fig. 10.4.5. After the cathode Cd was saturated with U and Pu, the samples taken from the Cd showed almost stable values, meaning that over saturated U and Pu, which exist as solid phases in the LCC, can not be taken by drawing. On the contrary, the final weight ratio of U and Pu calculated from the mass valances of U and Pu in the system showed close value with the weight gain of the cathode. Based on the mass balances, the amount of U and Pu recovered in the Run-1 cathode were 1.65g of U and 11.9g of Pu, corresponding to the current efficiency to be 87%. The separation factor of U versus Pu, i.e. the U/Pu ratio in the cathode product divided by the U/Pu ratio in the salt, to be 1.6. In the case of Run-2, the amount of U and Pu recovered in the cathode based on the mass balances were 2.34g of U and 12.4g of Pu, corresponding to the current efficiency to be 95% and the separation factor of U versus Pu to be 1.3. Accordingly, U and Pu were recovered into the LCC at a high current efficiency in each experiment.

A microscopic observation of the cross section of the Run-1 cathode Cd showed that the Cd phase consisted of two phases, a phase looked like almost pure Cd and a phase seemed to contain U, Pu, and Cd. An EPMA observation of the Run-1 ingot, as shown in Fig. 10.4.6, also showed that U and Pu exist in the same phase uniformly, and almost no U or Pu exists in another phase. This means that U, Pu, and Cd made a ternary compound in the Cd phase.

At Run-2, the cathode ingot containing U and Pu was recovered from the AlN crucible without breaking the crucible. Though the cathode crucible of Run-1 was broken, the crucible had been used three times for Pu recovery experiments before Run-1. Any change of color was not found on the surface of the AlN materials. Since no deposit was found on the outside of the crucible, the electric conductivity of the AlN at 773K is negligible. Then the AlN seems to be a hopeful material for the cathode crucible.
Fig. 10.4.1 Apparatus for electorecovery experiments

![Apparatus for electorecovery experiments](image1)

**Fig. 10.4.2 Cross section of LCC**

![Cross section of LCC](image2)

Fig. 10.4.3 LCC potential-time curves

![LCC potential-time curves](image3)

Fig. 10.4.4 Cathode ingot after Run-1

![Cathode ingot after Run-1](image4)

Fig. 10.4.5 weight ratios of elements in LCC

![Weight ratios of elements in LCC](image5)

Fig. 10.4.6 EPMA images of Run-1 cathode ingot

![EPMA images of Run-1 cathode ingot](image6)
10.5 Mass-spectrometric determination of oxygen potential of urania-yttria solid solution

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Many works on oxygen potential of fluorite type solid solution between urania and trivalent rare earth oxides have been reported. Experimental results show that the solid solutions containing trivalent ions, M$^{3+}$, generally have higher oxygen potentials than the pure urania. As Lindemer's review\(^1\) points out, however, the data of the hypostoichiometric solid solutions illustrate considerable discrepancy among several investigators. The present work concerns mass-spectrometric measurements on the evaporation of two type urania-yttria solid solutions: one is the solid solution expressed as nearly $U_{1-x}Y_xO_{2-x}(x=y/2)$ and the other is a nearly stoichiometric solid solution.

The reduced specimens R9 and R12 expressed as $U_{0.94}Y_{0.09}O_{1.96}$ and $U_{0.88}Y_{0.12}O_{1.95}$ contain a large amount of oxygen vacancies, while specimens S6 and S12 expressed as $U_{0.91}Y_{0.06}O_{1.99}$ and $U_{0.88}Y_{0.12}O_{1.99}$ have nearly stoichiometric composition. The former specimens were prepared by pressing $UO_2$ and $YO_{1.5}$ mixed powders and sintering at 2073K in hydrogen. For the latter, powders coprecipitated from a nitrate solution with ammonia were pressed and sintered at 1973K in argon/8% hydrogen. Preparation and characterization of the solid solution specimens were described in previous papers\(^2,3\). The details of a quadrupole mass spectrometer equipped with a Knudsen cell used in this study were described elsewhere\(^4\).

In the nearly stoichiometric specimens, uranium-bearing species found in the vapor phase were $UO_3^+$ and $UO_2^+$. On the other hand, $UO^+$ was observed instead of $UO_3^+$ over the reduced solid solutions. Elementary uranium ion, $U^+$, was also identified over the solid solution containing the most amount of oxygen vacancies(R12). When determining oxygen potential, $\Delta\overline{G}(O_2)$, of the specimens dealt in this work, it is required to calculate an equilibrium oxygen partial pressure, $P(O_2)$. In the present work, $P(O_2)$ is deduced as follows. Considering the following equilibrium in gas phase:

\[
\begin{align*}
UO(g) &= U(g) + 1/2O_2(g) \quad (1) \\
UO_2(g) &= UO(g) + 1/2O_2(g) \quad (2) \\
UO_3(g) &= UO_2(g) + 1/2O_2(g) \quad (3)
\end{align*}
\]

$P(O_2)$ associated with eq.(1) can be obtained by using the detected ion intensities of I(U) and I(UO) and the relation of $P-K-I-T$ from the following relation:

\[
\begin{align*}
\Delta^*H(eq.2, 298K) &= -RT \ln[P(O_2)] - 2P(U)/P(UO)] + T[1/2\text{feff}(O_2,g) + \text{feff}(U,g) - \text{feff}(UO,g)] \\
&= -RT \ln[P(O_2) + 1/2I(U)/I(UO)] + T[1/2\text{feff}(O_2,g) + \text{feff}(U,g) - \text{feff}(UO,g)]
\end{align*}
\]

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where \( f_e = -[G'(T) - H'(298)]/T \), which are available from the reference data. \( P(O_2) \) values for eqs. (2) and (3) can be also obtained in similar manners. The temperature dependence of \( \Delta \overline{G}(O_2) \) is directly deduced from the relation \( \Delta \overline{G}(O_2) = RT \ln P(O_2) \), for four specimens and the numerical expression of \( \Delta \overline{G}(O_2) \) is as follows:

\[
\begin{align*}
S12: & \quad \Delta \overline{G}(O_2) (J/mol) = 393,700 \pm 30,900 (-13.75 \pm 16.30)T, \\
S6: & \quad \Delta \overline{G}(O_2) (J/mol) = 520,200 \pm 15,700 (-70.47 \pm 8.046)T, \\
R9: & \quad \Delta \overline{G}(O_2) (J/mol) = 1,262,000 \pm 42,300 (-287.3 \pm 11.13)T, \\
R12 \text{ from eq. (2): } & \quad \Delta \overline{G}(O_2) (J/mol) = 1,224,000 \pm 21,200 (-264.0 \pm 10.70)T, \\
R12 \text{ from eq. (3): } & \quad \Delta \overline{G}(O_2) (J/mol) = 1,310,000 \pm 16,400 (-299.4 \pm 8.390)T.
\end{align*}
\]

The \( \Delta \overline{G}(O_2) \) values for R12 deduced by eqs. (1) and (2) almost agree each other by considering the uncertainty of the thermodynamic quantities used in calculation. \( \Delta \overline{G}(O_2) \) values for nearly stoichiometric specimens are much higher than those for pure urania at the same \( x \). This fact has been already indicated by several workers investigating \( UO_2 \)-trivalent rare earth oxide system. On the other hand, \( \Delta \overline{G}(O_2) \) values for reduced ones almost agree with those for pure urania at the same \( x \). From the temperature dependence of \( \Delta \overline{G}(O_2) \), the partial molar entropy \( \Delta S(O_2) \) and the partial molar enthalpy \( \Delta \overline{H}(O_2) \) of oxygen are calculated by a least-squares fit using the following thermodynamic relationships:

\[
\Delta S(O_2) = -\frac{\partial \Delta \overline{G}(O_2)}{\partial T}
\]

and

\[
\Delta \overline{H}(O_2) = \Delta \overline{G}(O_2) + T\Delta S(O_2).
\]

These calculated values are shown in Figs. 10. 5. 1 and 10. 5. 2, respectively. For comparison, these figures also include the experimental and theoretical values for pure urania and the \( M^{3+} \) solid solutions containing about 10% foreign cation. As shown in these figures it appears that the nearly stoichiometric specimens already have the thermodynamical properties on the hyperstoichiometric side. This fact suggests that the steepest change of \( \Delta \overline{G}(O_2) \) arises in the hypostoichiometric region as in the case of \( M^{2+} \) solid solutions. On the other hand, thermodynamic quantities for the reduced specimens agree well with those for pure urania. These results are much different from those deduced from some models reported in the past which indicate that both the \( \Delta S(O_2) \) and \( \Delta \overline{H}(O_2) \) values for the hypostoichiometric \( M^{3+} \) solid solutions are much higher than those for the pure urania as shown in the figures. The reduced specimens, however, can have almost the same concentration of the trivalent and tetravalent ions as pure urania at the same hypostoichiometric composition since the reduced specimens can have few \( U^{3+} \) and \( U^{5+} \). If the distribution of these cations and oxygen vacancies in the cation and anion sublattices is identical to that for pure urania, the configurational entropy for the reduced specimen should be in accord with that for pure urania. So their \( \Delta S(O_2) \) values can agree each other since the configurational entropy is considered to be the principal cause of variation of \( \Delta S(O_2) \) with \( x \). While we do not know why little difference of the \( \Delta \overline{H}(O_2) \) values between the reduced specimens
and pure urania arises. But the fluorite type pure urania and its solid solutions are regarded as ionic crystallines and Coulomb interaction principally contributes to cohesive energy. If the reduced specimens have the same distribution of cations and anions and the same distances among these ions as pure urania, their cohesive energies should agree each other. So the $\Delta \overline{H}(O_2)$ values for the reduced specimens may also approach to that for pure urania. On the other hand, a large difference of $\Delta \overline{H}(O_2)$ between the $M^{3+}$ solid solutions and pure urania indicated by several authors is still not quantitatively elucidated. Then, further experiments and theories for hypostoichiometric $M^{3+}$ solid solutions are required for knowing the characteristics in detail.

References:

Fig. 10.5.1 Variation of $\Delta S(O_2)$ for urania-yttria solid solutions with O/M ratio together with those for pure urania and other solid solutions with about 10% foreign trivalent cations.

Fig. 10.5.2 Variation of $\Delta \overline{H}(O_2)$ for urania-yttria solid solutions with O/M ratio together with those for pure urania and other solid solutions with about 10% foreign trivalent cations.
11. Nuclear Transmutation System Studies

Research items were classified into three main themes: 1) a conceptual design of an accelerator driven transmutation system (ADS), 2) the study on materials for ADS development and 3) the activity on partitioning and transmutation technology.

As to a conceptual design of an ADS, the transmutation capabilities were confirmed for the minor actinides in the core region and the long-lived fission products such as I-129 in the blanket region. In order to estimate effects of plant design parameters on the transient response to beam trips, an analysis of beam trip transients was made for a proposed lead-bismuth cooled ADS. The lead-bismuth inventory has a primary effect on the plant response to the beam trip transient.

The MEGAPIE (MEGAwatt PIlot Experiment) international project is closely related to the above three themes. Its activity is to design, build, operate and explore a liquid lead-bismuth spallation target for 1MW of beam power at the existing spallation neutron facility, SINQ of Paul Scherrer Institute, PSI. JAERI will contributes to MEGAPIE tasks such as fluid dynamics and structural mechanics, solid-liquid interface, neutronics and nuclear assessment, radiation effects and post-irradiation experiments. The first results of neutronic benchmark concerning neutron leakage, spallation products and heat deposition were obtained.

Static corrosion tests in liquid lead-bismuth were performed in a basic research program of the materials study. From the results in oxygen saturated Pb-Bi at 550°C for 500h, it was shown that the thickness of corrosion films decreases with increasing Cr content in steels. A new promising technique to study the properties of small irradiated-specimens was developed to determine the constitutive equation of elastic-plastic materials by the indentation technique using plural indenters with different apex angles.

Cooperation with different research groups, research institutes and universities has been pursued for development of partitioning and transmutation technology. Recent information and results were also exchanged in the JAERI-CEA(Centre National de la Recherche Scientifique) meeting on partitioning and transmutation.
11.1 Conceptual Design Study for Iodine Transmutation Using Accelerator-driven System

K. Nishihara and H. Takano

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A conceptual blanket design for $^{129}$I transmutation is proposed for an accelerator-driven system (ADS) that is designed to transmute minor actinides (MAs). In this ADS, 250kg/yr of MA and 56 kg/yr of iodine are simultaneously transmuted, and they correspond to the quantities generated from 10 units of existing light water reactors. Furthermore, an introduction scenario and the benefit of iodine transmutation are studied for future introduction of fast breeder reactors.¹)

The ADS core design is almost the same with an ADS core described in Ref. 2 without existence of the iodine blankets attached at top, bottom and side of the MA core as shown in Figure 11.1.1. The core characteristics for MA core region are shown in table 11.1.1.

In the present calculations, we consider NaI as the chemical form of iodine.² Isotope separation of $^{127}$I and $^{129}$I is not done. Both of them capture a neutron and become stable xenon gas, $^{130}$Xe and $^{128}$Xe. The isotropic ratio is 81% of $^{129}$I and 19% of $^{127}$I. For the purpose of thermalising neutrons, a part of pins in the blanket consists of zirconium hydride (ZrH$_{1.7}$). The thicknesses and ZrH$_{1.7}$ volume ratios of each blanket are optimized to achieve the minimum inventory.

Table 11.1.1 also shows the blankets parameters obtained by the optimization. The inventory of iodine in the ADS is 1600 kg, and the transmutation ratio is 3.5%/yr. The cycle lengths of the axial and radial blankets are 2 and 4 years, respectively. The total system
inventory including iodine in the reformation plant is 2200 kg.

This system inventory is very large so that the operations of 45 power plants during 9 years are necessary to prepare the iodine for an ADS system. Figure 11.1.2 shows the scenario of introducing ADSs in the case that the ADS transmute MA and iodine from the LWRs and FBRs in 1st stratum. FBRs are supposed to be introduced at 2050 and substitute the LWRs at 2120. To quantify the benefit of iodine transmutation, the end of nuclear times is supposed to be 2200.

The benefit is estimated from the view point of intake dose according to the river water scenario. Figure 11.1.3 shows the intake dose ratio of $^{129}$I from the transmutation systems and disposal sites of TRU waste. The loss in the reformation of iodine is supposed to be emitted the biosphere immediately, and the remains of iodine at 2300 are disposed in the same manner as the TRU waste. The toxicity in the long term is reduced to that of shorter term.

For the estimation of the shorter term, the loss ratio in reformation of iodine is significant. The ratio of 0.1 % used in this scenario is groundless, and it will be required from such analysis.

References
Table 11.1.1 Core characteristics of the ADS

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
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<tr>
<td>Initial k-effective</td>
<td>0.95</td>
</tr>
<tr>
<td>Coolant material</td>
<td>Lead-bismuth</td>
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<tr>
<td>Inlet temp.</td>
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<tr>
<td>Outlet temp.</td>
<td>445 °C</td>
</tr>
<tr>
<td>Velocity</td>
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<tr>
<td>Core height</td>
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</tr>
<tr>
<td>Diameter</td>
<td>1150 mm</td>
</tr>
<tr>
<td>Pin diameter</td>
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</tr>
<tr>
<td>Pin pitch / diameter</td>
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</tr>
<tr>
<td>Fuel</td>
<td>Nitride (^{14}N enriched)</td>
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<tr>
<td>(MA/Pu in initial)</td>
<td>MA60%+Pu40%</td>
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<tr>
<td>Inert matrix</td>
<td>ZrN (volume ratio: 68%)</td>
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<tr>
<td>Initial heavy metal loading</td>
<td>4000 kg</td>
</tr>
<tr>
<td>Transmutation of heavy metal</td>
<td>250 kg/300days</td>
</tr>
</tbody>
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Blanket | Axis | Side |
<table>
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<th></th>
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<th></th>
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</thead>
<tbody>
<tr>
<td>Blanket thickness</td>
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<tr>
<td>Pin diameter</td>
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<td>20.0</td>
</tr>
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<td>Pin pitch / diameter</td>
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<td>1.2</td>
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<tr>
<td>Volume ratio: NaI/(NaI+ZrH$_2$)</td>
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<td>40%</td>
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<tr>
<td>Gas plenum</td>
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<tr>
<td>Initial iodine loading</td>
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<td>839 kg</td>
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<tr>
<td>Transmutation per 300 days</td>
<td>34.5 kg</td>
<td>22 kg</td>
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<tr>
<td>Transmutation ratio per 300 days</td>
<td>4.5%</td>
<td>2.6%</td>
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</table>

Fig. 11.1.2 The number of ADSs and the amount of iodine in the case that nuclear times end at 2200 year, and, then the ADSs are operated until 2300 year.

Fig. 11.1.3 Intake dose rate of $^{129}$I from the transmutation systems and disposal sites.
11.2 Analysis of Effects of ADS Design Parameters on Response to Beam Trip Transient

T. Takizuka, K. Tsujimoto, K. Nishihara and M. Hishida*  

(E-mail: takizuka@omega.tokai.jaeri.go.jp)

Negative impact of frequent accelerator beam trips on operation of an accelerator-driven system (ADS) has attracted much attention recently1). An analysis of beam trip transients was made for a proposed lead-bismuth cooled ADS to examine the effects of plant design parameters on the transient response. Transients of the primary coolant temperature, the water/steam temperature, the water/steam pressure, the turbine flow rate, and the electric output were calculated using a simple flow network model.

The ADS plant2) is designed to fission minor actinides from about 10 units of 1000 GWe LWRs, producing a thermal power of 800 MWe. Thermal energy generated in the subcritical core and the spallation is removed by forced flow of lead-bismuth primary coolant and transferred via steam generators (SGs) to the secondary water/steam system. A conventional saturated steam turbine is used for power generation. A simplified flow diagram of the reference ADS plant is depicted in Fig. 11.2.1. The parameters varied around the reference plant design point were the core inlet temperature, core exit temperature, primary lead-bismuth inventory, and the steam drum volume for saturated steam turbine cycle. A superheated steam cycle plant was investigated for various core inlet temperatures. The cases analyzed are listed in Table 11.2.1.

Before the beam trip, the plant is assumed to operate steadily under the normal condition at the rated core power. On beam trip, the core thermal power immediately falls down to 0, decay heat being neglected. The operating procedure after a beam trip in an ADS is different from that after a scram in a critical reactor, because the accelerator beam recovers automatically within a short period of time in many cases. The primary lead-bismuth pumps and the recirculation water pumps are not controlled until the turbine/plant trips, running at 100% flow rate. The plant trips to avoid overcooling when the SG primary outlet temperature drops below 200°C. The turbine trips when the steam drum pressure falls below 1.55 MPa for the saturated steam cycle, and when the SG steam outlet temperature falls below 328°C for the superheated steam cycle.

Figure 11.2.2 presents the temperature transient at the SG lead-bismuth inlet for the reference case and cases 1 and 2, and the SG steam outlet for the case 10. The case 1 with 30% larger lead-bismuth inventory than the reference case shows a slower transient, and beam-off

* Mitsubishi Heavy Industries, Ltd.
time shorter than 410 s is allowed without turbine/plant trips. The case 2 with 30% smaller lead-bismuth inventory shows a faster transient, and the allowable beam-off time is 250 s. The case 10 for the superheated steam cycle shows very rapid drop of the SG outlet steam temperature. Superheat can not maintain beyond 120 s, leading to a much shorter allowable beam-off time around 80 s. The allowable beam-off time for each case is summarized in Table 11.2.1.

In conclusion, the lead-bismuth inventory has a primary effect on the plant response to the beam trip transient. In contrast, the steam drum volume, operating temperature level, and core temperature rise have little effect on the allowable beam-off time. The allowable beam-off time of the superheated steam turbine plant is much shorter than that of the saturated steam turbine plant.

References

<table>
<thead>
<tr>
<th>Case</th>
<th>Core inlet temperature (°C)</th>
<th>Core outlet temperature (°C)</th>
<th>Pb-Bi inventory (%)</th>
<th>Steam drum volume (%)</th>
<th>Steam turbine cycle</th>
<th>Allowable beam-off time (s)</th>
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<tr>
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<td>330</td>
<td>100</td>
<td>100</td>
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<td>130</td>
<td>100</td>
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<td>125</td>
<td>saturated</td>
<td>~250</td>
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<td>100</td>
<td>150</td>
<td>saturated</td>
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<td>100</td>
<td>100</td>
<td>saturated</td>
<td>~320</td>
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<tr>
<td>5</td>
<td>480</td>
<td>280</td>
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<td>100</td>
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<tr>
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<td>9</td>
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<td>380</td>
<td>100</td>
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<td>superheated</td>
<td>~80</td>
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<tr>
<td>10</td>
<td>530</td>
<td>430</td>
<td>100</td>
<td>-</td>
<td>superheated</td>
<td>~80</td>
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</table>
Fig. 11.2.1 Simplified flow diagram of the reference ADS plant

Fig. 11.2.2 Temperature transient initiated by a beam trip
11.3 Preparation for MEGAPIE International Experiment

Y. Kurata, M. Futakawa, T. Osugi, T. Takizuka, M. Kureta and K. Tsujimoto
(E-mail: ykurata@popsvr.tokai.jaeri.go.jp)

The development of a liquid metal spallation target is a key issue for realization of an "Accelerator Driven Subcritical System" (ADS) to reduce the amount and radio-toxicity of long-lived radioactive waste. Therefore, MEGAPIE (MEGAwatt PLot Experiment)\textsuperscript{1}) was projected to design, build, operate and explore a liquid lead-bismuth spallation target for 1MW of beam power at the existing spallation neutron facility, SINQ of Paul Scherrer Institute, PSI. The objective of MEGAPIE project is to demonstrate, in an international collaboration, the feasibility of a liquid lead bismuth target for spallation facilities at a beam power level of 1 MW.

Original members of MEGAPIE cooperative agreement are COMMISSARIAT A L’ENERGIE ATOMIQUE, CEA(France), the Centre National de la Recherche Scientifique, CNRS(France), L’Ente per le Nueve Tecnologie, l’Energia e l’Ambiente, ENEA(Italy), the Forschungszentrum Karlsruhe GmbH, FZK(Germany), the Paul Scherrer Institut, PSI(Switzerland) and the Studies Centrum voor Kernenergie/Centre d’Etudes de l’Energie Nucléaire, SCK.CEN(Belgium). JAERI will also participate the MEGAPIE international experiment.

Fig.11.3.1 shows phases and major milestone of the MEGAPIE project. After baselining and feasibility study of phases 1 and 2, conceptual design and engineering design of the liquid metal target are performed. Detailed design and manufacturing in phase 5, and system integration and testing in phase 6 will be performed. Operation of MEGAPIE target will be carried out in 2004. Post irradiation experiment (PIE) and decommissioning of irradiated MEGAPIE target are planned in phase 8.

Fig.11.3.2 shows the basic concept (left) and conceptual design (right) of the MEGAPIE target. A main flow pump and a bypass flow pump are put into the target container. The concept of a pumped bypass flow is adopted to cool sufficiently the
window during transients. The main flow guide tube serving to separate the upward flow in the center of the target from the downward flow in the outer annulus is planned. The heat exchange system is designed such that freezing of the liquid metal can be safely avoided in the system.

There are many tasks such as project management, scientific design support, target system and ancillary system in order to realize the MEGAPIE experiment. JAERI proposes a work package which should be possible contribution to the MEGAPIE project. In the task area defined by MEGAPIE, JAERI will contribute to the following task of the scientific design support : tasks 4, 5) fluid dynamics, structural mechanics, task 7) solid-liquid interface, task 9) neutronics and nuclear assessment, task 10) radiation effects and task 11) PIE.

Reference

Fig. 11.3.1 the MEGAPIE project phases and major mile stones
Fig. 11.3.2 The basic concept (left) and conceptual design (right) of the MEGAPIE target.
11.4 MEGAPIE Neutronics Benchmark and JAERIs Results

K.Tsujimoto
(E-mail: ktsujomega.tokai.jaeri.go.jp)

MEGAPIE is a joint initiative from PSI, CEA, CNRS, FZK and ENEA to design and safely operate a liquid lead-bismuth (Pb-Bi) spallation target for the 1 MW proton beam available in the existing spallation neutron facility SINQ at PSI.

The MEGAPIE project requires a precise neutronic and nuclear assessment of the Pb-Bi spallation target (neutron leakage, heat power deposition, material damages, spallation product yield) which is of first importance for the design of the target, and which offers also to associations the opportunity to compare/qualify neutronic models and codes on a full-scale 1 MW spallation experiment. The goal of this benchmark is to evaluate neutronic performances of the MEGAPIE target in a simplified cylindrical-beam model. It may constitutes a first step to evaluate source terms like heat deposition, neutron leakage, damages in materials, $^{210}$Po yield, etc. for the project and compare results obtained with different neutronic models and codes. It can be noted in particular that the $^{210}$Po yield has to be evaluated as precisely as possible for safety concerns.

A part of geometry in this benchmark problem is shown in Fig. 11.4.1. This is a magnification around the beam window. All input data required for the benchmark are given, for example, energy of 575 MeV and current of 1.74 mA (beam power is 1 MW) are assumed for the incident proton beam. The calculations were carried out using NMTC/JAM, $^{1}$ MCNP-4A $^{2}$ and DCHAIN-SP $^{3}$ codes. The NMTC/JAM code was used for the energy region above 20 MeV while the MCNP-4A code was used for energy region below 20 MeV with JENDL-3.2 library. The DCHAIN-SP code was used to analyze decay and build-up characteristics of spallation products. The results for heat deposition in Pb-Bi target and structural material including beam window are shown in Table 11.4.1. These results were compared with the results by other participants, such as CEA, PSI, CNRS, FZK and ENEA.

References


Fig. 11.4.1 Simplified cylindrical geometry of the MEGAPIE target (magnification around the beam window)
Table 11.4.1 Heat deposition in the structural materials

<table>
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<tr>
<th>Pb-Bi</th>
<th>Heat deposition (W/cm²)</th>
<th>316L steel tube</th>
<th>Heat deposition (W/cm²)</th>
<th>T91 window</th>
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<th>AlMg₃ window</th>
<th>Heat deposition (W/cm²)</th>
<th>Moderator AlMg₃ vessel</th>
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11.5 Static Corrosion Test in Liquid Pb-Bi Alloy

Y. Kurata and M. Futakawa

(E-mail: ykurata@popsvr.tokai.jaeri.go.jp)

Research and development for an accelerator driven system (ADS) have been carried out widely because it has possibility to transmute minor actinides and long-lived fission products in nuclear spent fuel\textsuperscript{1-3}. Liquid Pb-Bi alloy has attracted special interest as spallation targets and coolants for ADS. In this study, the initial result of the static corrosion test at JAERI is reported.

Corrosion behavior in liquid Pb-Bi alloy is not fully understood because many parameters such as oxygen concentration, temperature and various alloying elements influence it. It is essential to carry out the well-designed experiment using the apparatus with care of controlling oxygen concentration in order to clarify effects of these parameters on corrosion behavior.

A schematic diagram of the static corrosion equipment in Pb-Bi is shown in Fig.11.5.1. The retort contacting molten Pb-Bi is made of quartz. Initially the static corrosion test was performed in oxygen saturated Pb-Bi at 550°C for 500h. In this experiment, Ar gas containing several % O\textsubscript{2} was used as cover gas over liquid Pb-Bi. PbO with about 1mm thickness formed on the surface of the molten Pb-Bi during the corrosion test. Test specimens were pulled out from the molten Pb-Bi and cooled in the cover gas. Effects of Cr and Ni contents in steels on corrosion in liquid Pb-Bi were studied using 2.25Cr-1Mo steel, F82H, Mod.9Cr-1Mo steel, austenitic stainless steel, JPCA(14Cr-16Ni-2Mo), 410ss 430ss and pure iron.

The cross-section of the tested specimen was observed using optical microscope, laser microscope and EPMA after cutting corrosion specimens. Fig.11.5.2 shows the relationship between the thickness of the corrosion film and Cr content in steel. As shown in this figure, there seem to be four groups of steels which have different corrosion properties in oxygen saturated Pb-Bi at 550°C: pure iron with the thickest corrosion film, steels containing 2.25~9%Cr with about 20 μm thickness corrosion film, JPCA and martensitic/ferritic stainless steel containing 14 ~16%Cr with very thin corrosion film. Fig.11.5.3 shows optical micrographs of the cross section of materials after corrosion in oxygen saturated Pb-Bi at 550°C for 500h. Nickel was plated on the specimen surface to protect corrosion films. The lead-bismuth stuck to the specimen surface during pulling out from the molten Pb-Bi was found in several samples (Fig.11.5.3(b), (d), (f) etc.). Thickness of the corrosion film changes depending on materials. The corrosion film of pure iron is the thickest as shown in Fig.11.5.3(g). The corrosion film of F82H, Mod9.Cr-1Mo steel and 2.25Cr-1Mo steel is thicker than that of austenitic stainless steel, JPCA. Type 410ss, type 430ss and Mo have thin corrosion films. Pores are observed beneath the corrosion
film in Fig.11.5.3(a) and (b). It is also observed that corrosion proceeds along grain boundaries in JPCA as shown in Fig.11.5.3(c). From the results of the initial static corrosion test in oxygen saturated Pb-Bi at 550°C for 500h, it was shown that the thickness of corrosion films decreases with increasing Cr content in steels.

References

Fig.11.5.1 Schematic diagram of the static corrosion equipment in liquid Pb-Bi.

Fig.11.5.2 Relationship between thickness of the corrosion film and Cr content in steels.
Fig. 11. 5.3 Optical micrographs of the cross-section of (a) F82H, (b) Mod.9Cr-1Mo steel, (c) JPCA(14Cr-16Ni-2Mo), (d) 410ss, (e) 430ss, (f) 2.25Cr-1Mo steel, (g) Fe and (h) Mo. The corrosion test was carried out in oxygen saturated Pb-Bi at 550°C for 500h.
11.6 Identification of Constitutive Equation by Indentation Technique using Plural Indenters with Different Apex Angles

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A novel technique, which is very promising as small specimen tests to study the properties of irradiated materials, was developed to determine the constitutive equation of elastic-plastic materials by the indentation technique using plural indenters with different apex angles\(^1\). The characterized curves describing the relationship among the yield stress, work hardening coefficient and the work hardening exponent were established through FEM analyses. Identification for the constants of the constitutive equation was made based on the relationship between the characterized curves and the hardness given by the indentation load-depth, L-D, curve.

The FEM analyses were carried out to systematically examine the relationship between the L-D curve and the constants of the constitutive equation, using an explicit FEM code, LS-DYNA\(^2\), which enables us to robustly analyze a large deformation accompanying with contacting behavior. In this analysis, the indenter and specimen were treated as axisymmetric two-dimensional bodies to take calculative efficiency into account. The indenter used in the experiment was modeled to be conical. The apex angle of indenter was determined as taking account of geometrical similitude between indenters in the experiment and analysis. That is, because of the geometrical similitude, the projected area \(A_p\) at the penetrated depth \(D_p\) is related to \(D_p\) by

\[
A_p = gD_p^2
\]

where \(g\) is the geometrical factor of the indenter. The apex angle of the conical indenter with the same g-value, 24.5, as a Vickers indenter gets to be 70.30°. The modeled indenter was perfectly rigid. The mesh size was given to be sufficiently fine to keep accuracy: the minimum element size around an apex contact zone was 0.05 \(\mu\)m. The total number of the elements used in the model was 1509. The loading rate in the calculation was small enough to neglect an inertia effect as a static condition.

The constitutive equation of the material was assumed to be a simple power law that is generally believed to be applicable to normal metallic materials as follows:

\(*\) Niigata University
\[ \sigma = E \varepsilon \quad \sigma \leq \sigma_y \quad (2) \]
\[ \sigma = A(\varepsilon_0 + \varepsilon)^n \quad (3) \]
\[ \varepsilon_0 = (\sigma_y / A)^{1/n} - (\sigma_y / E) \quad \sigma > \sigma_y \quad (4) \]

where, \( \sigma \) is true stress, \( \varepsilon \) true strain, \( E \) Young’s modulus, \( \sigma_y \) yield stress, \( A \) work hardening coefficient and \( n \) work hardening exponent. The \( E \)-value can be experimentally obtained from unloading curves\(^1\). In order to determine the three values of unknown constants in Eqs. (2)-(4), the three different indenters were prepared in experiments, i.e. trigonal pyramid indenters with the \( g \)-values of 12.5, 24.5 and 49.0, which have the apex angles of 63.15°, 70.30° and 75.80°. Figure 11.6.1 shows a schematic drawing of the L/D-D curves. During loading the L/D-D curves increase almost linearly with increasing depth, and the \( S \)-values are dependent on the apex angle. The \( S \)-value is associated to the hardness that is believed to be an inherent value of the material and reflect the mechanical properties. The constants of the pre-assumed constitutive model are determined based on the relationship between \( \sigma_y \), \( A \), \( n \) and the \( S \)-value.

The technique was applied to evaluate two kinds metallic materials: nickel based alloy, Inconel 600, and aluminum alloy, A5056. Figure 11.6.2 shows the L/D-D curves measured for Inconel 600 using the trigonal pyramid indenters; \( I_1 \), \( I_2 \) and \( I_3 \). During loading, except for early stages of small loads, the slopes of \( S_1 \), \( S_2 \) and \( S_3 \) were almost constant, independently of the indentation load. We determined the values of \( A \), \( n \) and \( \sigma_y \) from the relationship between \( A_i \), \( n_i \), \( \sigma_y \), and \( S_i \). In order to validate the identified constitutive equations for Inconel 600 and A5056, the uniaxial tensile tests and the FEM analyses using the identified constitutive equations were carried out. Figure 11.6.3 shows the comparison between experimental and analytical results on the elongation-load curves. Figure 11.6.4 shows the failure deformation of the specimen compared with the analytical results. Despite the kinds of materials, good agreements are given between them on both the elongation-load curves and the failure deformation. It was found from Figs. 11.6.3 and 11.6.4 that the analytical results are representative well to the actual necking behavior, and that the presented technique based on the indentation tests could be applied to find out optimum parameter values in given constitutive equation.

References
1) M. Futakawa, T. Wakui, Y. Tanabe and I. Ioka, Identification of constitutive equation by


Fig. 11.6.1 Schematic drawing of L/D-D curves

Fig. 11.6.2 L/D-D curves of Inconel 600

Fig. 11.6.3 Comparison between experimental and analytical results on elongation-load curves

Fig. 11.6.4 Comparison between experimental and analytical results on failure deformation
12. **Nuclear Ship Research and Development**

Design studies of two very small-scale and highly compact reactors adopting a passive safety system — **Submersible Compact Reactor**, SCR and **Marine Reactor for Ground use**, MR-1G— completed in this year. The SCR with thermal output of 1.25 MW aiming usage for a submersible scientific research vessel especially in under-sea of the Arctic Ocean has been studied from 1998. Design study of the MR-1G with thermal output of 1 MW for heat supply at an office building has been conducted from 1999. Furthermore, design study on a heat supply system by using 100 MWt reactor, MR-100G has commenced in 2000.

Regarding the SCR, the design conditions such as the power size or operation time per a research cruising were set up on the base of a needs survey. The reactor core is optimized to be capable of operating for a long-term without refueling for 10 years with the load factor of 50%. The passive safety system can reject the decay heat to seawater at the condition of any ship posture even in case of a founder. Estimation of activated corrosion products in the primary loop has revealed the operation without the purification system to be feasible. Under ship motion such as heaving, the core can be cooled by natural circulation, confirmed analytically.

The reactor concept of MR-1G is based on the SCR, which is designed to operate for a long-term core life, with another passive safety system, and has not the purification system either. The MR-1G, an exclusive reactor for heat supply, will be set in a basement of an office building. The MR-100G, which is also an exclusive reactor for heat supply, will be sited underground deeply at a big city.

Key technology adopted in these reactors is an in-vessel type control rod driving mechanism (CRDM), which is installed inside the reactor vessel for safety improvement and compactness of the system. Development of this CRDM has completed.

An advanced automatic operation system has been developed by using operator's knowledge for an integral-type marine reactor. Hybrid automatic operation system for the advanced marine reactor capable of smooth transfer from a normal operation to an abnormal operation has completed. The nuclear engineering simulation system of the advanced marine reactor, which has been used for designing of the automatic operation system was improved on hardware such as simulator's control system, graphical tool, CRT and so on.
12.1 Simple Evaluations of Fluid-induced Vibrations for Steam Generator Tube Arrays in Advanced Marine Reactors (MRX,DRX)

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Both of advanced marine reactor (MRX) and deep sea research reactor (DRX) are the integral-type PWR, and the steam generators are installed in the reactor vessel. Steam generators are the once-through helical-coil tube types. Heat transfer tubes, in which the secondary fluid flows, are arranged in the inner annular space between the inner shroud and the reactor vessel.

Flow-induced vibrations are calculated by simple methods, and the arrangements of tube support structures are evaluated.

The vibration by the Karman's vortices

It was checked analytically that the vortex shedding frequency of a Karman's vortices is separated enough from the natural frequency of the heat transfer tube.

Since there are no formula available to a multi-supported system, the following equation (1) for bending vibration of a straight rod with a two-supported system was applied by introducing a correction coefficient C.

\[ f = C \frac{\lambda^2}{2a d^2} \sqrt{\frac{1000 EI}{w}} \]  \hspace{1cm} (1)

The C was set to 0.918 for MRX and 0.993 for DRX, which were obtained as ratio of frequencies by two equations for two-supported systems – the one for the straight rod and the other for the ring rod -. That is, the ratios of frequency by the equation (2) to the frequency by equation (1) putting C=1 were used.

\[ f_1 = \frac{1}{2\pi R^2} F \left( \frac{EI}{GJ} \right) \sqrt{\frac{1000 EI}{w}} \]  \hspace{1cm} (2)

The following formula was used to estimate the vortex shedding frequency by Karman's vortices.

\[ f_K = S_i \times \frac{V}{d_o/1000} \]
Flow velocity of the primary cooling water $V$ was computed by the following formula.

$$V = \frac{Q_i}{A \times 10^{-6} \times \frac{1}{3600}}$$

Where $A$ is channel area outside the tube by the following formula for MRX, which are square arrays.

$$A = \frac{\pi}{4} \left( D_o^2 - D_i^2 \right) - \pi d_o \sum_{k=1}^{n} d_k - A_{sp}$$

For DRX, which is a staggered arrangement, it was computed by the following formula.

$$A = \frac{\pi}{4} \left( D_o^2 - D_i^2 \right) - \pi d_o \sum_{k=0}^{n} d_{2k+1} - A_{sp}$$

On the base of the experiment\(^2\) of Chen, the Strouhal number of the tube was calculated by the following formula for MRX.

$$S_r = 0.1 \times \ln\left[A_n \left( X L - 1.0 \right) + 1.0 \right]$$

For DRX, the Strouhal number was obtained by the following formula.

$$S_r = 1.15 \times \left[ 0.2 - (0.2 + B)e^{xM * x} + B \times e^{(2.0 \times \times)} \right]$$

The random vibration by turbulence

The random vibration due to turbulence flow is out of the question if the frequency of it is separated enough from the natural frequency. A following formula is used to estimate the frequency of random vibrations.

$$f_{rb} = \frac{1000 \times V \times d_o}{L_T \times T} \left[ 3.05 \left( 1 - \frac{d_o}{T} \right)^2 + 0.28 \right]$$

Compared with the random vibration $f_{rb}$ by turbulence, it was found to be separated from the natural frequency three or more times, and it can be said that a significant vibration due to turbulence is not produced.

The flow-induced elastic vibration

A marginal non-dimension flow velocity producing a flow-induced elastic vibration
was evaluated using the formula of Connors for a straight-pipe group.

\[ u_e = C \left\{ \frac{\rho \delta}{\left( \pi d_o^2 \right)} \right\}^{0.5} \]

The non-dimension flow velocity \( u_n \) of the fluid outside the tubes can be expressed with the following formula.

\[ u_n = \frac{V}{(f_2 d_o)} \]

As a result of evaluation, the non-dimension flow velocities outside the tubes are smaller than the marginal non-dimension flow velocity of the flow-induced elastic vibration, and it can be said that a flow-induced elastic vibration is not produced.

The vibration by the flow inside the tube

It was also evaluated that the non-dimension flow velocity inside the tubes are smaller than the marginal non-dimension flow velocity producing the unstable vibration and the vibration can not happen.

The non-dimension flow velocity on the out-of-plane vibration of a bent tube is expressed with the following formula.

\[ V_N = \sqrt{\frac{W_f}{1000EI}} \times V_f R \]

Conclusion

It was revealed that the steam-generator tubes of MRX and DRX do not have possibility of generating the Karman's-vortices excitation vibration, the random vibration by turbulence, a flow-induced elastic vibration, and the unstable vibration by the flow inside the tube.

References
1) The Japan Society of Mechanical Engineers, a mechanical-engineering handbook
12.2 Effect of Ship Motions in a Small Marine Reactor Driven by Natural Circulation

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In JAERI, very small reactors, DRX\textsuperscript{1} and SCR\textsuperscript{2} for submersible research vessel has been designed on the basis of needs investigation of sea research. Although the reactor is a PWR type, self-pressurization and natural circulation system are adopted in a primary system for small size and lightweightness. The fluid flow condition of the reactor core outlet is designed to be the two-phase with a low quality. Although the flow of a primary system is the two-phase flow with a low quality, the density wave oscillation may occur according to operating conditions. Moreover, since there are ship motions such as heaving (the vertical direction acceleration), when a submersible research vessel navigates on the sea surface, the circulation flow of the primary system is directly influenced by this external force. In order to maintain stable operations of the reactor, it is necessary to clarify effects of this external force due to ship motion.

The reactor coolant flow mentioned in this report is the two-phase natural circulation flow coupled with the neutron kinetics, and the flow is considered to close to the characteristics of a BWR. However, there are different features from that of BWR, because a reactor core size is very small, the flow is in a low quality, and the external force by ship motions are added. These characteristics were analytically investigated using the transient thermal-hydraulic analysis code RETRAN-02/GRAV\textsuperscript{3} improved to simulate the effect of ship motions.

The noding model for DRX is shown in Fig. 12.2.1. The mode for SCR is almost the same as this. The core is divided into two regions: an average and

Fig.12.2.1 Noding model of DRX
hot channels. Feed water comes into the steam generator from junction J50 and steam goes out from J65. Boundary condition of the analysis is that the feed water flow rate and the pressure of the steam pipe outlet are constants.

As a typical example, response of the reactor system of the DRX is shown in Fig.12.2.2 to heaving (as shown in (a)) of the oscillation period, 5 sec. and the magnitude, 0.1g. The reactor power (b) oscillates with the same period of heaving. The reactor power oscillation is caused by the reactivity associated to the moderator density. The core flow rates (c) also oscillate with almost the same period of heaving. The amplification of the core outlet flow rate having a distorted sinusoid is smaller than that of the core inlet flow rate. This is considered to be due to steam void in the core. It is one of the features of the two-phase flow. The steam void fraction in the core (d), which is about 30% and 15% at the upper and the middle of the hot channel, oscillates with the same period. The analysis shows that greater amplitude of heaving induces to greater oscillations of the core flow rate and the reactor power.

By changing heaving period, response characteristics of the reactor power and the core flow rate were surveyed. Amplitudes of both oscillations have peaks at the heaving period of 5 sec. as shown in Fig.12.2.3. This peak is considered to stem from resonance of the natural circulation flow and the heaving.

Since resonance of the heaving acceleration and the natural oscillation of loop flow causes the flow and reactor power oscillate largely, it should be avoided. In order to suppress the amplitude of this oscillation, some measures are tried by setting a large flow resistance at the core inlet, reducing the reactivity of moderator density, and gas pressurization, i.e., by

![Fig.12.2.2 Heaving: Period=5sec, Amplitude=0.1g](image-url)
adding a non-condensable gas such as nitrogen in the steam dome at the upper part of the reactor vessel. The most effective measure is found to be gas pressurization.

In the case of gas pressurization from 8.4 MPa to 10 MPa, filling nitrogen into the steam dome (N20), the amplitude of reactor power oscillation becomes a third of the original case, and that of the core inlet flow rate becomes a half of it. The core flow rates of the inlet and the outlet are almost the same. As shown in Fig.12.2.3, amplitudes of this case have no peaks and the resonance can be avoided. The reason is that there is not steam void in the core and the single-phase natural circulation flow is established. The analysis on SCR gave almost the same behavior as that of DRX.

The study on dynamics of DRX and SCR under the ship motions by RETRAN-02/GRAV can conclude as follows: (1) Heaving gives an in-phase oscillations of the core flow and the reactor power. Oscillations depend on the amplitude and the period of heaving. Especially with the heaving period of 5 sec., the core flow and the reactor power oscillate with large amplitude, which is caused by resonance of the system flow and heaving. (2) Effective measure to suppress the oscillation is to pressurize the primary loop by filling a non-condensable gas into the steam dome, because the flow state changes to a single-phase flow by pressurization. (3) Effect of the reactor power variation on the natural circulation flow behavior under heaving is very small. Resonance of the reactor power and the system flow are not appeared in the present analysis.

References
12.3 Radiation Shielding Design of Very Small Reactors, SCR and MR-1G

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Advanced Marine Reactor Laboratory has completed conceptual design of very small reactors, Submersible Compact Reactor, SCR, for undersea scientific observation vessel ¹), and MR-1G for heat supply for an office building in a city ²). This paper summarizes radiation shielding design for the very small reactors. Thermal output of the SCR is 1.25 MW and that of MR-1G is 1.0 MW.

Shielding Design for SCR

As for shielding design of the SCR, evaluation of dose equivalent rate in an accommodation area of a scientific research vessel was carried out. Since crew are boarded for the all day time during navigation in the accommodation area, this area should be located in outside the controlled area boundary and the dose equivalent rate in this area should be less than 1.7 μSv/h. The one dimensional radiation transport code, ANISN ³), was used for dose equivalent rate evaluation considering the major structural materials for use to reduce radiation from the reactor core, such as iron-made reactor vessel of 9.5 cm thickness, iron shield inside the containment of 17 cm thickness, iron containment vessel of 7.0 cm thickness, and iron pressure wall of 11.5 cm thickness.

Dose equivalent rate in the machine room, which is the nearest room to the reactor core, is 14 μSv/h during reactor operation at full power. The machine room can be classified as the controlled area, where is relatively low dose area, and radiation workers are allowed to enter this room without any restriction for 48 hours in a week. Dose equivalent rate in the accommodation area varies from 4 to 8 μSv/h. These result indicate that additional shield is necessary to reduce the dose equivalent rate to design criteria, so that the accommodation area can be classified as outside the controlled area. By assuming that dose equivalent rate in the accommodation area is 1.7 μSv/h and 60 days cruise for one mission of the scientific observation, accumulated dose equivalent during one cruise is 2.4 mSv. Since the accumulated dose equivalent exceeds the annual dose limit for public, 1 mSv, all crews including scientists are required to be assigned as radiation workers from radiation protection aspect. Shielding performance analysis reveals that it is impossible for crew to enter the
reactor compartment during the reactor operation because dose equivalent rate outside the reactor containment vessel is more than $10^3$ Sv/h.

Shielding analysis for a case of immediately after reactor shutdown was also carried out using gamma-ray source from the fission products calculated by ORIGEN2 \(^4\) with a condition that the reactor was operated at full power for 2200 days, which corresponded to the core life of the SCR. Dose equivalent rate in the accommodation area in this case is less than $10^{-3}$ μSv/h, and that outer surface of the research vessel beside the reactor is 77 mSv/h. The result indicates that appropriate radiation control is necessary if a ship such as a support ship approached to this research vessel.

As for the hypothetical accident, it is assumed that the fission products are released into the containment vessel immediately after reactor scram following to continuous reactor operation at full power for 2200 days. The ratios of the amount of fission products released into the containment vessel to those which existed in the core before the release were assumed to be 100 % for noble gas, 5 % for iodine, 1 % for halogen except for iodine and 0.01 % for the other nuclei in the containment vessel. The source term was calculated by ORIGEN2 and the shielding calculation was carried out using QAD-CGEP2 \(^5\) to obtain dose equivalent in the accommodation area. The analysis indicates that the calculated accumulated dose equivalent for 30 days for crew in a case of hypothetical accident is lower than annual dose equivalent limit to public, 1 mSv.

**Shielding Design for MR-1G**

The MR-1G was designed to be constructed in a factory, installed in basement of an office building after being transported by a trailer, and transported again to a factory after 10 years operation. Since the containment vessel of the MR-1G has a function of the spent fuel transport cask, it is required that shielding design of the MR-1G satisfies the design criteria for the spent fuel transport cask. Shielding design criteria for the spent fuel transport cask, in which dose equivalent rate at 1 m apart from the surface of the cask should not exceed 100 μSv/h was considered. In the present study, analysis of shielding performance of the MR-1G for a case after reactor shutdown was carried out.

Conceptual study of the MR-1G assumes that MR-1G would be removed from the office building within several days after 10 years reactor operation. Radiation shielding optimization was carried out aiming reduction of dose equivalent rate after 24 hours from reactor shut down when most of the short lived nuclei would be decayed.
Table 12.3.1 Radial direction shield thickness of MR-1G

<table>
<thead>
<tr>
<th>Shield</th>
<th>Thickness (cm)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Core barrel</td>
<td>4.0</td>
</tr>
<tr>
<td>Water between core barrel and reactor vessel</td>
<td>22.4</td>
</tr>
<tr>
<td>Reactor vessel</td>
<td>5.0</td>
</tr>
<tr>
<td>Containment vessel</td>
<td>5.0</td>
</tr>
<tr>
<td>Lead shield</td>
<td>15.0</td>
</tr>
<tr>
<td>Resin shield</td>
<td>10.0</td>
</tr>
<tr>
<td>Copper fin and air</td>
<td>10.0</td>
</tr>
</tbody>
</table>

Shield structure of the containment vessel of the MR-1G is based on design of the spent fuel transport cask, TN-12A. As for the inner shield structure, additional shields were added inside the core barrel and the reactor vessel aiming reduction of radiation. In Table 12.3.1, shield thicknesses of the MR-1G in radial direction are shown. Shielding performance was evaluated by the ANISN code using gamma-ray source from the fission products after 24 hours from the reactor shutdown calculated by ORIGEN2 with a condition that the reactor was operated at full power for 1600 days corresponding to the core life of the MR-1G. The calculated results are shown in Table 12.3.2. The results indicate that the present shielding design satisfies the design criteria.

Table 12.3.2 Calculate dose rate equivalent at 1 m apart from surface of containment

<table>
<thead>
<tr>
<th>Dose rate equivalent (μSv/h)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Radial direction</td>
</tr>
<tr>
<td>Upward direction</td>
</tr>
<tr>
<td>Downward direction</td>
</tr>
<tr>
<td>The design criteria</td>
</tr>
</tbody>
</table>

References

12.4 Design Study on Very Small Reactor for Heat Supply

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Statistics on energy consumption of Japan shows that amount of energy for public use is growing constantly, while that for industrial use remains almost the same level from 1975 as shown in Fig.12.4.1. This statistics indicates the public energy consumption will have an important share in Japan. About 60% of the public energy consumption is for heating, hot-water supply, and air conditioning at present, which can be supplied by a heat supply system. The heat supply system can be also advantageous in case of a big disaster, because the system does not make cause of fire. It is said that people especially living in big cities are worried about a large-scale fire following to a natural disaster, such as a big earthquake. Furthermore, the heat supply system is favorable for a less-children society predicted as a Japanese society in the near future.

Then, Japan Atomic Energy Research Institute (JAERI) is studying the heat supply system with nuclear reactors.

Although the conventional nuclear power plants have been built in remote places, the reactors for heat supply must be built near the energy demand area that consume energy. In the case of the existing big cities in Japan, it is difficult to build the

Figure 12.4.1 Change and the items of energy consumption
nuclear reactors on the ground due to the weak geology. Then JAERI has studied installing the nuclear reactor of heat supply underground in a large depth. The underground structures consist of tunnels, auxiliary cavities and main cavities containing a nuclear reactor, a spent-fuel place, etc. (see Fig.12.4.2).

The requirements for the heat supply reactor sited near a big city can be considered as follows. (1) enhanced safety, (2) compactness and lightweightness, (3) short time installation, (4) long-term operation and easy maintainability, (5) safety characteristics against disasters such as seismic or fire, (6) effective isolation, (7) a high social acceptability taking account of convenience, economy, etc.

Conceptual study of the heat supply reactor MR-100G has been performed on the base of the design study of the MRX which is an advanced marine reactor for a practical nuclear powered ship. The major parameters and, the conceptual drawings of MR-100G are shown in Table 12.4.1 and Figure 12.4.3. The MR-100G is so called “integral type self-pressurized water reactor” containing steam generators inside the reactor vessel. A core is cooled by natural convection of the coolant. The reactor does not have a pressurizer; operating pressure inside the vessel is determined by coolant temperature. Reactivity control was made by the only control rods without soluble boron. The control rods are driven by in-vessel type control rod drive mechanisms (CRDMs). A containment vessel is 8.5m in diameter, 22m in height. A large water tank for emergency core cooling in case of accidents is arranged above the reactor vessel and is pre-pressurized by nitrogen gas. The containment vessel is evacuated to insulate thermally the pressure vessel during normal operation. The battery driven valves are

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**Fig.12.4.2 Passively Safe small Reactor for Distributed energy system**
opened automatically to inject the water into the containment vessel. The underground main cavity is made from reinforced concrete with a steel liner, and has the shape of vertical half-sphere cylinder with a height of 35m, and a diameter of 24m.

Since cavities set underground in the large depth near a big city will be the constructed on the base of a quaternary deposit, it is important to assess the stability of the cavities. It is shown by analyses with a finite element method assuming the physical-properties of the foundations that the static stability of the main cavity was securable with temporary structures.

The future assessment items are influence of ground water, the last heat sink of a reactor and a strategy for obtaining understanding of technical subjects, such as the dynamic stability at the time of the earthquake of an underground cavity etc.

Table 12.4.1 Specification of MR-100G

<table>
<thead>
<tr>
<th>Primary Coolant System</th>
<th>Steam generator</th>
</tr>
</thead>
<tbody>
<tr>
<td>Core Outlet Temp. (°C)</td>
<td>Type</td>
</tr>
<tr>
<td>233</td>
<td>Once-through straight-tube</td>
</tr>
<tr>
<td>Pressure (MPa)</td>
<td>Steam Temp. (°C)</td>
</tr>
<tr>
<td>3.0</td>
<td>180</td>
</tr>
<tr>
<td>Thermal Output (MW)</td>
<td>Containment vessel</td>
</tr>
<tr>
<td>100</td>
<td>Diameter (m)</td>
</tr>
<tr>
<td></td>
<td>8.5</td>
</tr>
<tr>
<td>Core</td>
<td>Height (m)</td>
</tr>
<tr>
<td>No. of fuel assemblies</td>
<td>22</td>
</tr>
<tr>
<td>37</td>
<td></td>
</tr>
<tr>
<td>Fuel life time (years)</td>
<td></td>
</tr>
<tr>
<td>8</td>
<td></td>
</tr>
<tr>
<td>Others</td>
<td>Integrated, Natural circulation, Self-Pressurization</td>
</tr>
</tbody>
</table>

Reference
1) Handbook of energy and economic statistics in Japan (1999)
12.5 Improvement of the Integral-Type Reactor Simulator

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JAERI had designed a light-weight and compact integral-type reactor with thermal output of 100MWh for use of next generation nuclear ship. A real-time simulator for this integral-type reactor was developed to study its operation performance and to help automatic operation system design. This simulator is a part of the Nuclear Ship Engineering Simulation SYstem (NESSY) already competed and is installed on the same hardware as that of the MUTSU simulator\textsuperscript{1, 2}. The simulator can be used as the MUTSU simulator or the integral-type reactor simulator. The integral-type reactor simulator will be also used for simulation of a passively safe small reactor exclusive of heat supply.

Since the hardware of NESSY become old now, it is necessary to improve a performance of hardware and Human Machine Interface (HMI) of software of the simulator for designing of automatic operation system of the integral-type reactor. The supervisory and control part of the integrate-type reactor simulator was improved in 1999.

In 2000, a simulator control part of the integrate-type reactor simulator was improved. This part is mainly used by off-line, for example, start/stop or setup of an initial value of simulation. Furthermore, the computer load by simulation was reduced as much as possible by programming using Visual Basic (VB: Microsoft). So, it is shown that there is no trouble in simulation, even if this gives a simulation control capability to a data server, which connects with the simulation calculating machine in network. For this reason, the simulation control part was realized on the data server, and the hardware configuration became compact as shown in Fig.12.5.1.

As the result, the following points were improved on hardware and software of the integral-type reactor simulator.

(1) Operability: Operation of the simulator becomes easy and simple by enhancement in HMI.
(2) Versatility: Creation and modification of application programs can be done easily by using the versatile software for supervisory/operation screen creation (In-Touch) and the versatile program-languages (for example, VB).
(3) Maintainability: By using a conventional personal computer (PC), a kind of instrument to be maintained was reduced.
(4) Extendibility: A PC can connect comparatively easily to the CRTs, memories, printers and so on. Therefore, exchange of instruments in case of an instrument break down and on the occasion of change to a new model is comparatively easy.

(5) Portability: Windows OS can provide many PC’s operation and many application programs can be operated on Windows. Moreover, since the application programs adopted in this improvement are conventional, it is easy and simple work to transplant to other PCs.

In addition, PC’s under supervisory/operation can be extended to 20 sets. Remote supervisory/operation can be performed using the Ethernet. The MUTSU simulator is scheduled to be improved as same as this improvement.

Fig. 12.5.1 Improvement of a simulator control system

References
13. Facility Operation and Techniques Development

There are four reactor-engineering facilities such as Heat Transfer Fluid Flow Test Facility, Fast Critical Assembly (FCA), Tank-type Critical Assembly (TCA) and Very High Temperature Reactor Critical Assembly (VHTRC). Maintenance work for VHTRC and operations of other facilities were carried out as scheduled. Major activities of each facility of this fiscal year are summarized briefly below.

(1) The Heat Transfer Fluid Flow Test Facility was operated for various experiments such as Departure from Nucleate Boiling Test for Advanced Reactor, Measurement Test of Void Fraction Distribution by Neutron Radiography etc. And Thermal Fluid Safety Test (Ingress of Coolant Event/Loss of Vacuum event) for Fusion Reactor was carried out.

(2) The FCA was operated according to various purposes of experiments and recorded the operation time of 452.6 hours. Maintenance activities, fuel management and physical protection were done, i.e., renewal of the exhaust equipment for the reactor room, repair for defects of coating on fuels etc. And the Physical Inventory Verification (PIV) of nuclear fuel materials was carried out.

(3) The TCA was operated for the nuclear constant experiments and the training courses of the Nuclear Technology and Education Center (NuTEC). And it was recorded 200 times operation in 84 days. Fuel management, Physical Protection and PIV were also carried out.

(4) Maintenance works were done for VHTRC such as the monthly and annual inspections according to the safety regulation. The PIT and PIV for VHTRC were also carried out. The decommissioning of VHTRC reactor assembly and control panel was performed from Sep.2000 to Mar.2001.

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In FY-2000, operation and maintenance of Heat Transfer and Fluid Flow Test Facilities were carried out as scheduled.

As for the maintenance of test facilities, annual official inspections of the pressure vessels and the steam generator located at both of Chemical/Mechanical Engineering Building and Thermal Engineering Test Facilities were carried out.

As for the operation of test facilities, the following three tests were performed.

1) DNB Test for Advanced Reactor
2) Measurement Test of Void Fraction Distribution by NRG\(^1,2\)
3) Thermal Fluid Safety Test(ICE/LOVA) for Fusion Reactor

As for the reconstruction of test facilities, the following two test facilities were reconstructed.

1) Suppression Line was added to Integrated ICE/LOVA Test Facility
2) Steam/Water Separator Vessel was added to DNB Test Facility.

As for the representative R&D, void fraction measurement system(II) was developed.

References:

   Autumn Meeting of the Atomic Energy Society of Japan E29 (2000)
   JAERI-Research 2000-043
13.2 Operation Report of FCA

M. Saitoh, K. Satoh, K. Hayasaka, K. Kurosawa and H. Sodeyama

(E-mail: saito@fca001.tokai.jaeri.go.jp)

Operation of Fast Critical Assembly (FCA) was carried out in accordance with the experimental schedule on the FCA XXI-1 assembly. Operation of 91 times was carried out in 70 days. No scram was recorded during the operation. The total operation time was 452.6 hours and the integrated power was 0.24 kWh. A total number of 5485 criticality operations have been recorded at the end of this fiscal year since the first achievement of criticality on the 29 of April 1967. For the safety regulation of operation, two days were devoted to the monthly inspection and about 25 weeks from December 2000 to the annual inspection. Routine maintenance activities were done during the inspection to provide maximum operation days for the experiments. The exhaust equipment for reactor room was renewed for overage.

As for fuel management, the defects of coating on Enrich uranium, Natural uranium and Depleted uranium metallic fuel were repaired for about 18,000 plates and about 2,900 blocks by spraying the surface with colloidal solution of fluorocarbon in order to prevent the fuel from oxidation. Weights of the coating on the fuel plates were obtained from the difference of the weight between before and after coating.

As for the physical protection (P/P), security control of the gate was done restrictively and the system was maintained properly.

In connection with safeguard, IAEA and NSB* carried out monthly inspection under the international treaty. They made the PIV (Physical Inventory Taking) from the 27 to the 29 of June. Prior to the PIV we performed item counting, weighing and non-destructive assay of the fuel from the 19 to the 26 to of June.

* NSB: Nuclear Safety Bureau
13.3 Operation Report of TCA

K. Murakami and Y. Hoshi

Operation of Tank-type Critical Assembly (TCA) was carried out in accordance with the schedule on the experiments (the verification experiment on the nuclear constant, the measurement of reactivity worth of the ThO₂ fuel rod and the safety sheet) for the Research Group for Reactor Physics and on the experiments for the training courses of NuTEC (Nuclear Technology and Education Center) and on the experiments (Evaluation for sub-criticality measurement) for the Critical Safety Laboratory. The total operation time was 252 hours and integrated power was 18.6Wh during 200 times operation in 84 days. A total number of 11,023 criticality operations had been recorded at the end of this fiscal year since the first achievement of criticality on the 23rd of August 1962.

As for the nuclear material physical protection (P/P), the management of the entrance and exit was done restrictively and the system was maintained properly.

In connection with safeguard the inspection of nuclear material, stored at TCA, was carried out by IAEA and NSB* under the international treaty. The Physical Inventory Taking (PIT) of the fuels was performed on the 22nd of June and IAEA and NSB made the Physical Inventory Verification (PIV) on the 23rd of June by means of item counting, and non-destructive assay

* NSB: Nuclear Safety Bureau
13.4 Maintenance Work Report of VHTRC

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(E-mail: ono@vhtrc01.tokai.jaeri.go.jp)

The decommissioning of reactor assembly and control panel which is the first step of
the decommission construction of the VHTRC started from September 2000 and it was
finished in March 2001.

The other reactor equipment facilities such as reactor building and fuel storage room
of the VHTRC continue from now on maintenance and management in future.

According to the safety regulation for maintenance, the 16 weeks were devoted to the
annual inspection from July to October in 2000.

As for management, Physical Inventory Taking (PIT) was carried out from May
11 to 12 by means of item counting for fuel compact and fuel disk. IAEA and STA made the
Physical Inventory Verification (PIV) under the international treaty on May 12. No anomaly
was confirmed. The maintenance activity was also taken on the physical protection (P/P)
system.

The sensitivity and function of the system were examined and calibrated.
14. Activities of the Research Committee

The department of Nuclear Energy System serves as a secretariat of the following four research committees organized by JAERI: Japanese Nuclear Data Committee; Research Committee on Reactor Physics; Research Committee on Marine Reactors; and Research Committee on Partitioning and Transmutation Technologies. The purpose and the expected task of each research committee are summarized here. The detailed activities of each committee are presented in the following sections.

Japanese Nuclear Data Committee

The committee is organized to promote the evaluation of nuclear data and the production of group constants for application fields. There are three subcommittees, six standing groups and a steering committee under the Committees. The Committee also takes a task of compiling the activities of the International Nuclear Data Committee and the Work Party on International Evaluation Cooperation under OECD/NEA/NSC.

Research Committee on Reactor Physics

The committee reviews the research activities related to reactor physics in Japan and supports the activities of Nuclear Science Committee (NSC) of OECD/NEA. The committee consists of three working parties: reactor physics of accelerator driven system; reactor physics of LWRs loaded with next generation fuels; and conceptual study of advanced reactors.

Research Committee on marine Reactor

The committee is organized to review and discuss the research and development activities on marine reactor design and related technologies. The committee include two subcommittees on the conceptual design of a highly compact marine reactor and the nuclear ship MUTSU. The committee meeting was held once in the fiscal year to discuss about the present research activities.

Research Committee on Partitioning and Transmutation Technologies

The committee is organized to review and discuss the research and development (R&D) activities on partitioning and transmutation at JAERI. In the committee meeting the R&D plan and the activities were reviewed and discussed. Furthermore, future R&D plans were introduced and collaborated research plans were presented.
14.1 Activities of Japanese Nuclear Data Committee

A. Hasegawa

The Japanese Nuclear Data Committee (JNDC) consists of three subcommittees, six standing groups and a steering committee. Each subcommittee consists of several working groups (WG). The Committee Meeting of JNDC was held at July 2000 to discuss the nuclear data activity in the previous fiscal year and plans for the fiscal year 2000. Discussions were made on several topics including the planning of the 2000 Symposium on Nuclear Data as well as domestic and international collaboration on nuclear data.

The 2000 Symposium on Nuclear Data, which is held every year in November, was held. And a specialists’ meeting on Reactor Group Constants, which is a topical meeting held every year on the selected hottest topics in that period, was also held on February 2001.

The activities of subcommittees and standing groups are briefly summarized below.

Subcommittee on Nuclear Data

1) High Energy Nuclear Data Evaluation WG:

The evaluation is progressing in two phases. In the phase-I, the data up to 50 MeV for IFMIF (International Fusion Material Irradiation Facility) project is evaluated for neutron and proton induced reactions. In the phase-II, the data evaluation for high-energy neutron/proton induced reactions up to 3 GeV will be made. Data requests are very keen by the joint projects for High Intensity Proton Accelerator of JAERI Neutron Research Center and KEK. Following is the status of each SWG.

- IFMIF Neutron File Compilation SWG: Neutron file compilation is the main task of this SWG. Up to now, evaluation of 43 nuclides has been finished. The files are in the reviewing stage including FORMAT check.
- MeV and GeV File compilation SWG: Compilation and evaluation of phase-II data is a main mission. Evaluations for the priority 2 nuclides (about 40 nuclides) are performed together with the code preparation inevitable for the fundamental evaluation tools in this energy range (Quick-Gnash, QMD, JAM).
- For other Sub-Groups like Photonuclear Data, PKA/KERMA and High Energy Activation Cross-section, their activities has also been progressed.

2) Covariance Data Evaluation WG: All missions anticipated at first including establishment of covariance evaluation methodology and their tool developments have been completed successfully. This WG has been dissolved. Some persons are moved to Heavy nuclides evaluation WG to continue the evaluation of covariance data for heavy nuclides.

3) Evaluation and Calculation System WG:

Recommended parameters required in the nuclear model calculations such as OMP, level
density, gamma strength functions, as well as advanced methodologies like multi-modal fission, and essence from the latest frontiers of theoretical calculations are discussed. The results will be reflected to RIPL-2 (Reference Input Parameter Library Version 2).

4) Charged Particle Nuclear Data WG: This WG is responsible for the JENDL (alpha,n) Reaction File. Evaluation has been progressed.

5) Delayed Neutron Data Evaluation WG: This group was set up due to the follow up work of OECD/NEA/WPEC (Working Party on Evaluation Cooperation) subgroup 13 to investigate the delayed neutron data related problems. Evaluations of delayed neutron yields and spectra for main actinides of JENDL 3.3(U-235,-238,Pu-239) are also the mission of this group. Data evaluation in 6 group time dependent scheme have been made.

6) Evaluation WG on Intermediate Mass Nuclides: This WG was set up due to the revision work for JENDL-3.3. Re-evaluation work and relevant checking has been finished for Na-23, V-51, Co-59, Cr, Ti, Ni, W, Nb and Er (total of 32 nuclides). Some follow up from the results of benchmark tests were made.

7) Evaluation WG on Heavy Mass Nuclides: This WG was set up due to the revision work for JENDL-3.3. Re-evaluation work has been made for U-233, -235,-236,-238, and Pu-236,-239,-240,-241,-242. Working group has been watching the feedback information from benchmark tests made by subcommittee of reactor constants.

**Subcommittee of Reactor Constants**

1) Reactor Integral Test WG: Benchmark test of JENDL-3.3 tentative version for fast and thermal reactors has been made. Although over all agreements (C/E values) in Keff for thermal system as well as fast system revealed superior to JENDL 3.2 data, some problems for SUS reflected cores are found.

2) Shielding Integral Test WG:

   For the tentative version of JENDL-3.3, benchmark test were made for main shielding materials such as Al, Si, Na, Ti, V, Cr, Fe, Co, Ni, Cu, Nb, W. All the results were fed back to the evaluation WG on intermediate mass nuclides

3) Dosimetry Integral Test WG: All of the missions were completed and the group was dissolved.

4) Standard Group Constants WG: Revision work for JSSTDL library has been progressed. Report of the JSSTDL-300 are being prepared. Future directions of this WG has been discussed.

**Subcommittee on Nuclear Fuel Cycle**

The subcommittee on nuclear fuel cycle consists of three WGs, i.e., Decay Heat Evaluation WG, WG on Evaluation of Nuclide Generation and Depletion, and FP Mass Yield Evaluation WG.
For the First WG, JNDC FP Decay Data File has been made and opened as JNDC FP Decay Data File 2000. A new measurement of decay heat of U-235 has been made at YAYOI U-Tokyo reactor in cooperation with JNC. For the second WG, new entry for PWR and BWR-MOX has been added to ORIGEN-2 Library and published a report. Methodology for the evaluation of sensitivity in one group cross section is discussed. Investigation on the produced nuclides originating from different data source (JEF-2.2, ENDF/B-VI) has been made. For the last WG, this group was organized so as to work with IAEA CRP. To give precise mass yields data for minor actinides as well as major actinides covering the wide range of incident neutron/proton energies is the main mission. Collection of mass yields data evaluated/measured in Japan was made. The systematic made by Moriyama and Ohnishi was applied and analyzed. A possibility for updating of recent Benlinure’s semi-empirical model was investigated. Measurements of mass yields for Am-241 and 243 by proton bombardment up to 30 Mev were made.

Standing Groups
1) CINDA Group : Papers on neutron induced reaction data published in Japanese journals and reports are surveyed. Total of 212 entries were sent to the NEA Data Bank in the last one year to update the CINDA master database.
2) ENSDF Group : The evaluation of nuclear structure data is the duty of this group for nuclei with mass numbers from 118 to 129. Re-evaluation has been nearly finishing for A=120, 122, 126 and 128, and some efforts for A=122, 123 and 129 were made.
3) Group on Atomic, Molecular and Nuclear Data for Medical Use : Survey work has been made for the radiopharmaceutical data needed in the field of nuclear medicine. A new home page for this group was opened to promote and encourage WG missions.
4) JENDL Compilation Group : File compilation and editing were made for the JENDL-3.3.
5) Editorial Group of "Nuclear Data News" : Three issues of "Nuclear Data News" (No.66-68) which is a periodic informal journal circulated in nuclear data communities of Japan (written in Japanese) were published in a yearly base. Total of about 450/issue were distributed in the nuclear data communities without fee.
6) High Priority Request List Group: A revision was made for the latest version of HPRL, this is mainly from ADS (Accelerator Driven System) application’s requests. This group is a coordination group to set up a Japanese Requests Lists from domestic data users and a world-wide request list HPRL that is maintained at OECD/NEA/WPEC (Working Party on Evaluation Cooperation).

The 2000 Symposium on Nuclear Data
The Symposium was held at Tokai Research Establishment of JAERI on 16-17th November 2000. Japanese Nuclear Data Committee and Nuclear Data Center of JAERI organized this symposium. In the oral sessions, presented were 18 papers on the topics for
recent experiments including heavy elements synthesis and a new N=16 magic number, JENDL-3.3 developments status and their benchmark tests including integral test of JENDL-3.2, and international session. In the poster session, presented were 40 papers concerning experiments, evaluations, benchmark tests and on-line database on nuclear data. Total of 155 attendees including 16 foreigners, about 80 persons outside JAERI were gathered. The percentage of foreigners in the attendees exceeds 10 %, therefore the symposium is no longer the domestic one.

Specialists’ Meeting on Reactor Group Constants

The meeting was held on 22-23rd February 2001 at JAERI Tokai site. This meeting was held to exchange latest information on group constants development and to coordinate the group constants processing frame for upcoming JENDL-3.3 release. Total of about 60 specialists including 25 outside JAERI were attended. Topics presented and discussed are status and future of group constants in each field. From free discussion about future reactor group constants preparation, there is a mutual consensus for the necessity of Common Group Constants generation for JENDL-3.3, even though the format or funds are not clear yet.
14.2 Activities of the Research Committee on Reactor Physics

M. Nakagawa and T. Osugi

The committee reviews research activities related to reactor physics in Japan and supports the activities of Nuclear Science Committee (NSC) of OECD/NEA. The committee consists of three working parties and steering community.

The 70th meeting of the Research Committee on Reactor Physics (RCRP) was held on July 7, 2000. In the meeting, the following topic was discussed: Reactor physics experiments for improvements of conventional BWRs and for design study of reduced-moderated tight-lattice core in NCA (Toshiba Nuclear Critical Assembly). The meeting also devoted to the distribution of documents, which were discussed at the 11th meeting of NEA/NSC held at OECD Headquarters, Paris, June 7 - 9 2000.

Working Parties

Three working parties, as shown below, were established in FY-1999. The interval of their activities was limited in two years: FY-2000 was the second year. The meeting in each working party was held twice in FY-2000. The following topics were focused on and were discussed in the meetings.

1) Working Party on Reactor Physics of Accelerator Driven Systems,
   - Status of ADS project in Japan,
     JAERI-KEK high-intensity proton accelerator joint project,
     Neutron factory program in KUCA,
   - Influence of nucleon-nucleon scattering cross section model on ADS core characteristics,
   - Review of sub-criticality measurement techniques,
   - Investigation of reactivity control in ADS,

2) Working Party on Reactor Physics of LWRs Loaded with Next Generation Fuels,
   - Proposal of Benchmark Problems for Calculation Accuracy Evaluation on LWRs Loaded with Next Generation Fuels,

3) Working Party on Conceptual Study of Advanced Reactors,
   - Review of Advanced Reactor Concepts,
     Ultra small prompt critical reactor,
     Reactor used for space development

The activities for two years in each party were introduced in 2001 annual meeting of AESJ, March, 2001. The summary reports will be published by each party.
14.3 Activities of Research Committee on Marine Reactors

T. Ishida

(E-mail ishida@koala.tokai.jaeri.go.jp).

The committee is organized to review and discuss research and development (R&D) activities on marine reactor design and related technologies. The committee consists of two subcommittees: One is the subcommittee on concept design of a highly compact marine reactor (SCR) for usage of submersible research vessel. The second, which is at rest now, is the subcommittee for review of Mutsu irradiated fuel and its forthcoming testing. The former subcommittee has a working group for detailed discussion on specific issues. The committee's members are from Japan Atomic Energy Research Institute (JAERI), universities, other research institutions and industries including reactor or ship manufactures and utilities.

The committee meeting is normally held once a year. The R&D plan and activities of the subcommittees for the FY 2000 were reviewed and discussed in March 2001.

Subcommittee on concept design of a highly compact marine reactor

The subcommittee meeting was held once in March 2001 jointly with the committee meeting. Activity of the working group consisting of the experts such as physical oceanographers, submersible designers, a navigation system instrument designer, etc., in which the operation condition, the operational design, and the safety of the submersible research vessel with the SCR for cruising at undersea of the Arctic Ocean were discussed and filed as a report, was introduced and reviewed in the subcommittee meeting.
14.4 Activities of Research Committee on Partitioning and Transmutation Technologies

K. Tsujimoto and Y. Kurata

(E-mail: ktsujii@omega.tokai.jaeri.go.jp)

The committee is organized to review and discuss the research and development (R&D) activities on partitioning and transmutation at Japan Atomic Energy Research Institute (JAERI). The committee’s members are from the universities, research institutes, industries and JAERI.

The committee meeting is normally held once a year. The R&D plan and the activities for the FY 2000 were reviewed and discussed on 1st February 2001. After an introduction of recent activities on partitioning and transmutation studies in foreign countries, the R&D activities and results in JAERI were reviewed and discussed. The main topics of the meeting were as follows:

- Transmutation experiment facilities planned in joint intense accelerator project of JAERI-KEK,
- Development of new extractant and partitioning process,
- Fuel fabrication, irradiation experiment and dry reprocessing for the nitride fuel.

Future R&D plans for the partitioning and transmutation system were introduced: The present status and the next five-year program were presented and discussed for the partitioning, fuel fabrication, nuclear transmutation and reprocessing processes. The international MEGAPIE (Megawatt Pilot Experiment) program for Pb-Bi spallation target experiments at PSI, Switzerland was also introduced with the possible contribution from JAERI in the field of neutronics, thermo-dynamics, material tests for the Pb-Bi target.

Collaborated research plans for the FY2001 between JAERI and universities, and other institutes were presented and approved by the committee: The development of oxygen sensor in liquid lead-bismuth with the Tohoku University, The research about partitioning-transmutation technique of long-lived nuclides with CRIEPI(Central Research Institute of Electric Power Industry), and The research of thermal-hydraulic feature in a liquid target region of accelerator-driven system with the Tokyo Institute of Technology.
Publication List

1. Nuclear Data, and Atomic and Molecular Data


2. Reactor Physics


2) Nagaya, Y.: "Reference Solutions for 3-D Radiation Transport Benchmarks by a Monte


3. Advanced Reactor System Studies


4. Heat Transfer and Fluid Flow


5. Energy System Analysis and Assessment

6. Reactor Structural Materials


7. Advanced Materials for Nuclear Applications

8. Compatible Materials Development for Advanced Nuclear Systems

9. Rock-like Oxide Fuel for Plutonium Burning in LWRs
burning in LWRs”, Proc. 8th Int. Conf. on Nuclear Engineering (ICONE 8), April 2-6, 2000, Baltimore, USA (2000) (CD-ROM).

10. Nitride Fuel and Related Pyrochemical Technology


11. Nuclear Transmutation System Studies


12. Nuclear Ship Research and Development


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Appendix II Engineering Facilities Related to the Department

FCA : Fast Critical Assembly

The FCA is a split-table type facility of horizontal matrix structure designed for studying nuclear characteristics of fast reactor. The construction of the FCA was started in 1965 and the first core went critical on 29th April, 1967. The main features of facility are summarized as follows:

Type: Split-table type of horizontal matrix structure
Size: $2.8 \times 2.8 \times 1.3$ m (each half assembly)
Fuel: Enriched uranium and plutonium (Plate type)
Other material: Sodium, stainless steel, aluminum oxide, polystyrene etc.
(Plate type)
Maximum power: 2kW
Assembly name: FCA-I~FCA-XXI

Critical experiment using enriched uranium cores were made in 1960s investing basic characteristics of fast reactor cores. Mock-up experiments were extensively made in 1970s for the Fast Experimental Reactor JOYO and the Prototype Fast Breeder Reactor MONJU. In 1980s, the main subjects of experiments were the investigation of the core characteristics of an axially heterogeneous large fast breeder reactor and the core physics study on a high conversion light water reactor. In early 1990s, the reactor physics experiment of metallic-fueled LMFBR was carried out. Since 1995, international benchmark experiments for $\beta_{eff}$ was carried out. Since 2000, the reactor physics experiments for ADS have been conducted by using uranium fueled core.

VHTRC : Very High Temperature Reactor Critical Assembly

The VHTRC is a low-enriched uranium fueled and graphite moderated / reflected critical assembly. At VHTRC, reactor physics experiments have been carried out mainly for the verification of the neutronics design of the HTTR.
Main features of VHTRC

Type: Split table of hexagonal prism (prismatic block structure)
Size: 2.4m across the flats and 2.4m long
Fuel: 2.4 and 6wt% enriched UO₂
        Coated particle fuel compact, Pin-in-block type
Moderator/reflecter: Graphite
Core temperature: Room temperature to 210°C by electric heaters
Maximum power: 10W
Auxiliary equipments:
        ① Sample heating device (up to 800°C)
        ② Pulsed neutron source

TCA: Tank-type Critical Assembly

The TCA is a light-water-moderated critical facility to provide the experimental data on light water reactor physics. The construction of the TCA was started in 1961 and the first criticality was attained on 23rd August, 1962.

Main features of TCA

Type: Light water moderated Tank-type
Size: Typically 0.5m × 0.5m × 1m
        (Core Tank 1.8m diam. × 2.1m height)
Fuel: Low-enriched UO₂ and PuO₂- UO₂ fuel rod
Moderator: Light Water
Maximum Power: 200W
Auxiliary equipments: Pulsed neutron source
        Neutron absorbing materials (soluble or solid state)
国際単位系（SI）と換算表

表1 SI基本単位および補助単位

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表2 SI単位と併用される単位

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1 eV=1.60289×10⁻¹⁹J
1 u=166054×10⁻²³kg

表3 固有の名称をもつSI単位

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<td>セルシウス度</td>
<td>°C</td>
</tr>
<tr>
<td>光度</td>
<td>ルーメン</td>
<td>lm</td>
</tr>
<tr>
<td>放射能</td>
<td>ベクル</td>
<td>Bq</td>
</tr>
<tr>
<td>吸収線量</td>
<td>グレイ</td>
<td>Gy</td>
</tr>
<tr>
<td>線量等</td>
<td>シーベルト</td>
<td>Sv</td>
</tr>
</tbody>
</table>

表4 SIと共に暫定的に維持される単位

<table>
<thead>
<tr>
<th>項</th>
<th>名称</th>
<th>記号</th>
</tr>
</thead>
<tbody>
<tr>
<td>オングストローム</td>
<td>A</td>
<td></td>
</tr>
<tr>
<td>クーガー</td>
<td>b</td>
<td></td>
</tr>
<tr>
<td>ガル</td>
<td>gal</td>
<td></td>
</tr>
<tr>
<td>クリ</td>
<td>c</td>
<td></td>
</tr>
<tr>
<td>ラド</td>
<td>rad</td>
<td></td>
</tr>
<tr>
<td>レム</td>
<td>rem</td>
<td></td>
</tr>
</tbody>
</table>

1 A=0.1nA=10⁻⁹m
1 b=10²nA=10⁻⁶m³
1 bar=0.1MPa=10²Pa
1 Ci=3.7×10⁸Bq
1 R=2.58×10⁻⁴°C/kg
1 rad=1cGy=10⁻²Gy
1 rem=1cSV=10⁻²SV

換算表

<table>
<thead>
<tr>
<th>単位</th>
<th>記号</th>
<th>変換</th>
</tr>
</thead>
<tbody>
<tr>
<td>N(10⁻¹dyn)</td>
<td>kgf</td>
<td>lbf</td>
</tr>
<tr>
<td>1</td>
<td>0.010972</td>
<td>0.224809</td>
</tr>
<tr>
<td>9.80665</td>
<td>1</td>
<td>2.20462</td>
</tr>
<tr>
<td>4.44822</td>
<td>0.453592</td>
<td>1</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>粘度</th>
<th>η(1Pa·s)=10²N·s/m²=10²μPas(μポアス)</th>
<th>(g/cm·s)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>10²η</td>
<td>10²μ</td>
</tr>
<tr>
<td>0.01</td>
<td>10²η</td>
<td>10²μ</td>
</tr>
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</table>

表5 接頭語

<table>
<thead>
<tr>
<th>接頭語</th>
<th>記号</th>
<th>倍数</th>
</tr>
</thead>
<tbody>
<tr>
<td>E</td>
<td>10⁻⁸</td>
<td>エクサ</td>
</tr>
<tr>
<td>P</td>
<td>10⁻¹⁵</td>
<td>ペタ</td>
</tr>
<tr>
<td>T</td>
<td>10⁻¹²</td>
<td>テラ</td>
</tr>
<tr>
<td>G</td>
<td>10⁻⁹</td>
<td>ギガ</td>
</tr>
<tr>
<td>M</td>
<td>10⁻⁶</td>
<td>ミリ</td>
</tr>
<tr>
<td>k</td>
<td>10⁻³</td>
<td>キロ</td>
</tr>
<tr>
<td>h</td>
<td>10⁻²</td>
<td>ヘクト</td>
</tr>
<tr>
<td>da</td>
<td>10⁻¹</td>
<td>デカ</td>
</tr>
<tr>
<td>d</td>
<td>10⁻²</td>
<td>デシ</td>
</tr>
<tr>
<td>c</td>
<td>10⁻³</td>
<td>センチ</td>
</tr>
<tr>
<td>m</td>
<td>10⁻⁶</td>
<td>ミリ</td>
</tr>
<tr>
<td>μ</td>
<td>10⁻⁶</td>
<td>ミクロ</td>
</tr>
<tr>
<td>n</td>
<td>10⁻⁹</td>
<td>ナノ</td>
</tr>
<tr>
<td>p</td>
<td>10⁻¹²</td>
<td>ピコ</td>
</tr>
<tr>
<td>f</td>
<td>10⁻¹⁵</td>
<td>フェット</td>
</tr>
<tr>
<td>a</td>
<td>10⁻¹⁸</td>
<td>アト</td>
</tr>
</tbody>
</table>

（注）
1. 表1-5は国際単位系、第5版。国際対照局1985年採用による。ただし、1 eV および1 ιの値はCODATA1998年再値値によった。
2. 表4には海里、ノット、アール、ベクト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のカトリオンに分類されている。
4. E C関係理事会指示ではbar、barnおよび「血圧の単位」mmHgを表2のカトリオンに入れる。

(86年12月26日現在)