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(JULY 1992 — JULY 1993)

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Reactor Physics Activities in Japan
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Tokai Research Establishment
Japan Atomic Energy Research Institute
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This report reviews the research activity in reactor physics field in Japan during July, 1992 - July, 1993. The review was performed in the following fields : nuclear data evaluation, calculational method development, fast reactor physics, thermal reactor physics, advanced core design, fusion reactor neutronics, nuclear criticality safety, shielding, incineration of radioactive nuclear wastes, noise analysis and control and national programs.

The main references were taken from journals and reports published during this period. The research committee of reactor physics is responsible for the review work.

Keywords: Reactor Physics, Review Work, Nuclear Data, Calculational Method, Core Design, Criticality Safety, Incineration, Reactor Control

日本における炉物理活動
(1992年7月～1993年7月)

日本原子力研究所東海研究所
(編)炉物理研究委員会

(1993年12月14日受理)

本報告は、1992年7月から1993年7月に至る期間の日本における炉物理研究活動をレビューしたものである。

レビューを行った分野は、核データ評価、計算法の開発、高速炉物理、熱中性子炉物理、新型炉の設計、核融合炉ニュートロニクス、臨界安全、遮蔽、放射性廃棄物の消滅処理、雑音解析と制御、国のプログラムである。

主たる参考文献は、この期間に発行された各種雑誌及びレポートより取った。

なお、本レビュー活動は、炉物理研究委員会の責任で行ったものである。

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Foreword

This annual report is probably a single English document to review the activities of the reactor physics in Japan and it may contribute for understanding Japanese activities in this field and for encouraging activities of the reactor physics in the world.

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11. National programs	T. Ikegami and H. Yoshida

1. Nuclear Data Evaluation

A comparison for various kinds of systematics used to calculate the double differential light particle emission cross sections from nuclear reactions induced by light particles was made for medium-heavy nuclei important in fusion neutronics applications. For 14 MeV incident neutron, outgoing neutron emission spectra were calculated and compared to the experimental data obtained by two Japanese groups. It is concluded that the systematics derived by Kumabe et al. and Kalbach has good accuracy. This results will be reflected to the special purpose file of JENDL.[1)]

As the first step of nuclear data evaluations about radioactive production cross sections of photonuclear reactions, following 11 reactions are evaluated; ^{23}Na(γ,n)^{22}Na, ^{25}Mg(γ,p)^{24}Na, ^{48}Ca(γ,n)^{47}Ca, ^{46}Ti(γ,n)^{45}Ti, ^{52}Cr(γ,n)^{51}Cr, ^{55}Mn(γ,n)^{54}Mn, ^{59}Co(γ,n)^{58}Co, ^{58}Ni(γ,n)^{57}Ni, ^{65}Cu(γ,n)^{64}Cu, ^{90}Zr(γ,n)^{89}Zr, ^{100}Mo(γ,n)^{99}Mo. Those nuclides are picked up because of the request from the anticipated maintenance work for electron accelerators. These nuclides are main constitute of the target and their reaction cross-sections are anticipated to be large. Data are given for each reactions from the threshold energy to 140MeV of photons. Evaluations are made by the evaluation for photonuclaer absorption cross sections using measured cross sections in giant E1 resonance region and theoretical calculations in quasi-deuteron absorption regions, and branching ration calculated from MCPHOTO code for neutron and protons.[2)]

A benchmark test of LWR (Light Water Reactor) assemblies was performed for JENDL-3 and ENDF/B-VI file. Total of 39 assemblies are analyzed. Twenty four assemblies come from gadolinia assembly experiments performed at TCA (Tank type Critical Assembly) JAERI and the rest comes from the experiments for the PWR (Pressurized Water Reactor) fuel assemblies with water holes and lumped burnable poison arrangements measured by a Babcock & Wilcox critical facility. Group constants for main fissile are generated by RABBLE and THERMOS codes. Criticality calculations are made by KENO Monte Carlo code. From the study, both ENDF/B-VI and JENDL-3 files are mostly equivalent from the practical point of view of LWR core analysis.[3)]

A data book, thermonuclear reaction listing, is published for the nuclear data users for fusion energy applications. In the listing, total of 55 thermonuclear reactions induced by lighter nuclides than ^{11}B are presented for the following items; the reaction products, the values of Q and threshold energy and papers citing these reactions.[4)]

The proceedings of the NEANSC Specialists' meeting on Fission Product Nuclear

Data[5] was published. This meeting was held on May 25–27, 1992 at Tokai site of JAERI, JAPAN, to review the current status on evaluations and measurements of fission product data restricted to the cross–sections and fission yields. Total of 52 specialists were participated including 21 from abroad(9 countries and 2 international organizations). This meeting was held in the frame work of NEANSC Working Group on International Evaluation Cooperation.

A specialists' meeting on evaluation and processing of covariance data are held on 7th –9th Oct. 1992, at ORNL, USA.[6] Two papers are presented from Japan. One is relating to the data fitting method in the LSM (the least squares method) to explain the origin of Peelle's Pertinent Puzzle (PPP) and to give a criterion to obtain the same LSS (the least–squares solution) where the LSS obtained before and after a transformation of data are sometimes different from each other.[7] The other is an example of covariance data generation from experimental and theoretical data for ^{54}Fe and ^{56}Fe nuclides. Only level density parameters are the sources of the theoretical uncertainties in the study. It is understood that short range correlations is strengthen through the experimental data.[8]

A symposium on Nuclear Data Evaluation Methodology was held on 12–16 Oct., 1992 at BNL, Upton, New York, USA. Total of nine papers are presented from Japan. Their subjects are benchmark test of JENDL for fusion and shielding applications, evaluation methodologies for DDX (double differential cross–section) data, ^{238}U data, photonuclear data, covariance data, and other topics including data fitting method and AI(Artificial Intelligence) applications for nuclear data evaluations.

The 1992 Symposium on Nuclear Data[9] was held on Nov. 26–27, 1992. Fifteen papers in oral, 40 in posters (25 are contributed papers and the rest are publicity for the activities of working groups of JNDC(Japanese Nuclear Data Committee)) were presented, including review of nuclear energy in Korea, present status of revision work of JENDL–3, adjustment of group constants, nuclear data in medium energy region, nuclear data evaluation method, integral data analyses and other topics relating to the theoretical study of nuclear reaction processes.

The Fifth Meeting of the NEANSC Working Group on International Evaluation Cooperation was held on June 16–17, 1993 at Aix–en–Provence, France. In this meeting activities of 11 subgroups were reviewed and discussed. Some of the subgroups are determined to be migrated to the new standing groups rather than the term fixed subgroups to coordinate wider and long–term international relating problems such as nuclear measurements activities, evaluated data formats and processing for application libraries, high priority request list.

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2. Calculational Method Development

Various methods have been studied to improve the solution method of transport equations and to develop computer codes in this fiscal year.

An improved coarse-mesh discrete ordinate method has been proposed for three dimensional hexagonal transport calculations.[1] This method employs a new weighted diamond difference approximation which is obtained by using the neutron balance equations in divided submeshes. The weight is a function of neutron direction and scalar flux. The present method was implemented in the HEXTR code and applied to hexagonal fuel assembly calculations of high conversion reactor and fast reactors. The calculated results of k_{eff} and power distributions agree with the Monte Carlo calculations within 0.5 and 3%, respectively.

To generate assembly averaged effective cross sections, a two dimensional spectrum calculation code HARPY which can treat a hexagonal assembly has been developed.[2] The code consists of a nodal code for pin-cell coupling in hexagonal geometry and a hexagonal mesh S_n transport code HEXTR and calculates detailed neutron spectra of fuel cells in the assembly. Verification calculations have been performed for a small hexagonal assembly by comparing with a collision probability and a Monte Carlo codes. The comparison of k^{∞}', rod worths and pin power distributions have shown good agreement.

A neutron multigroup transport equation in x-y-z geometry is solved by the spherical harmonics method using finite Fourier transformation.[3] Using the first term of the Fourier series for the space variables of spherical harmonics moments, three point finite difference like equations are derived for x-, y-, z-axis directions, which are more consistent and accurate than those derived using the usual finite difference approximation, and these equations are solved by the iteration method in each axis direction alternatively. A method to find an optimum acceleration factor for the inner iteration is also proposed. The numerical examples give higher accuracy with less mesh points than the conventional finite difference method.

A nodal method to solve the multi-group diffusion equation is proposed to treat the heterogeneity of fuel assemblies of reactors.[4] Nodal equations of 3-points form in each coordinate are derived for the diffusion equation in one dimensional slab and x-y geometries using Green's function whose unknowns are only fluxes at the boundary surface of assemblies. Coefficients of the nodal equations are given by the boundary values of Green's function, which are obtained by solving 3-point difference equations in each coordinate for the heterogeneous assembly regarding it as being isolated. In this method, it is not necessary to use homogenized cross sections for the assembly, and the heterogeneity in the fuel assembly and interaction effect between assemblies can be taken into account.

The vectorization method has been studied to achieve a fast Monte Carlo transport calculation with the point estimator.[5] The algorithm was installed into a general purpose multigroup Monte Carlo code GMVP developed in JAERI for vector supercomputers and the performance of the present method were evaluated. The method can achieve a vectorization ratio of more than 98% and effective speedup of a factor of about 60 compared with a scalar code MORSE-DD.

Anisotropic diffusion coefficients for a large annular cavity are derived.[6] Neutron flux and current of the diffusion equation are expressed by using the series expansion of the Bessel function and equated with those by the transport equation. Thus obtained diffusion coefficients depend only on the geometrical configuration of cavity. A numerical comparison of diffusion calculations using these diffusion coefficients and transport calculations shows good agreement for a small size fast reactor consisting four regions : core ,reflector, cavity and shield.

Neutronic modeling for the modular high temperature pebble bed reactor during reactivity accident was investigated for safety analysis purpose.[7] Three existing dynamic models, *i.e.* the point reactor, adiabatic and improved quasistatic models were compared with each other, and their accuracy was evaluated. A one-dimensional numerical experiment was performed for simulating the severest reactivity accident involving withdrawal of all absorber rods in the reflector region. The results show that the adiabatic model gives the highest estimation of the power excursion since it overestimates the input reactivity worth. Although the point reactor model requires the minimum computation time, it underestimates the input reactivity worth and gives less accuracy in predicting the power excursion. The improved quasistatic model gives the highest accuracy and its computation time is comparable to the adiabatic model.

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3. Fast Reactor Physics

The fuel pin and subassembly heterogeneity effects on burnup characteristics in large fast breeder reactors have been studied. Homogenized cross sections have been prepared with two models. One is the homogeneous model, where homogenized cross sections are obtained based on subassembly-averaged atomic number density for each nuclide. The other is the heterogeneous model, where a pair of cell calculations is performed to obtain effective cross section homogenized over a subassembly. The first cell calculation yields homogenized cross sections over a fuel pin cell model, consisting of a fuel pin and the surrounding sodium region. The second cell calculation is carried out to obtain homogenized cross sections over a subassembly. The results of applying the two models to a typical 1,000MWe fast reactor core showed that the heterogeneity made the ^{238}U capture cross section smaller by about 2% for both initial and equilibrium cores and,.consequently, made the internal conversion ratio and breeding ratio smaller by 0.01 and 0.02, respectively.[1)]

The effect of internal blanket (IB) configuration, particularly IB radius, to the neutronics characteristics of a 1,000MWe axially heterogeneous fast breeder core has been investigated. This study proposes to select a large IB radius with the ratio of IB radius to core radius of ~ 0.8 from the view points of decreasing Na void worth and flattening power distribution. This proposed core has an IB volume fraction of about 17%, while most of the past studies proposed radius less than half of core radius and an IB volume fraction of about 17% from the view point of burnup reactivity loss.[2)]

A formula for the uncertainty of core performance parameters based on a combination of the cross-section adjustment and bias factor methods has been derived. The formula is compared with those derived from the cross-section adjustment method and the bias factor method used separately. When the method error correlation is strong between the critical assemblies and a target core, the combined method is superior to the cross-section adjustment method used alone. The combined method is, in general, superior to the bias factor method used alone. Numerical results are presented for the uncertainties of keff, the control rod worth, and the power distribution of a large fast reactor. The combined method yields a smaller uncertainty for the control rod worth calculated in dollar units than the cross-section adjustment method used alone.[3)]

The neutronic performance parameter and burnup properties are optimized for an axially heterogeneous fast reactor with an upper axial Na region considering neutronic decoupling and Na void worth. The LOF (Loss of Flow accident without scram) analysis has been carried out for the core with zero Na void worth using an effective coolant temperature coefficient. The effect of nonlinear change of Na void worth as a function of sodium density on LOF is estimated. The nonlinearity increased the average outer Na temperature by about 25°C compared to the case where Na void worth is assumed to change linearly from zero void to 100% void.[4)]

Eigenvalue separation, which is used as a criterion to determine the degree of neutronic decoupling of the core, is measured by a static flux-tilt method on Zero-Power Physics Reactor assemblies. Space-dependent nuclear characteristics, such as the rapid distributions of the reaction rates and the control rod worths, are also measured for the same assemblies. The calculation/experiment (C/E) values vary with core radius depending on the assemblies. The relationship between decoupling and C/E radial dependence is investigated, and a quantitative relation is found between the eigenvalue separation of the first radial mode and the C/E radial dependence.[5)]

A theoretical expression of the C/E (ratio of calculation to experiment) distributions for reaction rates and reactivity worths is developed on the basis of the explicit first-order perturbation formulation. The expression shows that the degree of the C/E spatial dependence depends on not only the magnitude of the Boltzmann operator errors due to cross-section errors, but also the λ-mode eigenvalue separations and the adjoint eigenfunctions of higher harmonics, and that the spatial shape of the dependence is described by the higher-harmonic forward eigenfunctions. Sensitivity analyses of the C/E radial dependence to various cross-section changes, based on the present expression, are carried out for a one-dimensional model of a large fast reactor assembly, ZPPR-13A. These results indicate that the first radial-harmonics dominantly contribute to the C/E radial dependence, and the sensitivity to a local change in a macroscopic fission cross-section is very high compared to the other perturbations.[6)]

Extensive work to improve nuclear design method and its accuracy for large LMFBR cores has been performed using the cross-section adjustment instead of the conventional E/C bias method. Analytical tools and data related to the adjustment method were established as a consistent system including JUPITER experimental and analytical results and cross-section covariance data. Using the cross-section adjustment, most analytical values obtained for core parameters in critical experiments were very close to the experimental values and past disconcerting problems of FBR core physics such as space dependence of the discrepancy between analytical and experimental values were practically solved. The adjusted cross-sections were inspected in detail from various physical aspects such as nuclear data evaluation, applicability to a set of FBR benchmark problems, statistical consistency and influence of data uncertainty, and found to possess sufficient reliability for analyzing large FBR cores. An application to a 600MWe-class FBR design core demonstrated the superiority of the adjustment method to the E/C bias method in the prediction performance of overall core characteristics. The cross-section adjustment is expected to make an improved nuclear design method for large FBR cores.[7]

Experiments were performed on metallic fueled LMFBR mock-up assemblies FCA XVI-1 and -2 which were built in the Fast Critical Facility of Japan Atomic Energy Research Institute JAERI. In these assemblies such feedback reactivity coefficients as Doppler, sodium void and fuel shift/expansion, and breeding index, neutron energy spectrum index and B_4C control rod worth were measured. The calculation based JENDL-2 data agreed well with the criticality, sodium void worth and reaction rate ratio measurements.[1] The Doppler effects measured in these assemblies were analyzed using the 70 group constant set JFS-3-J2 which is processed from JENDL-2. It was pointed out that due to the lack of resonance parameters of ^{238}U above 50kev in JENDL-2, the calculation underestimates the Doppler effect by about 10%.[2]

New devices for measuring Doppler effect up to 2000°C were developed for the measurements at FCA of JAERI. Two methods were combined to extend the temperature range of Doppler measurement to higher temperatures, one method is a small sample reactivity measurement up to 1500°C and the other is a foil activation measurement with laser heating up to 2000°C. A new cell code with ultra-fine group structure (Δu=0.25 to 4×10^{-4}), PEACO-X, was developed to calculate the resonance interaction effect between ^{238}U of the Doppler sample and tungsten which is used as the high temperature structural material of the experimental devices, and also the resonance interaction between ^{238}U in the core fuel and that in the Doppler sample. The Doppler effect measurements up to 2000°C were carried out at FCA assembly XVI-2 which is a physical mock-up core of a metallic fuel FBR.[3),4)]

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4. Thermal Reactor Physics

Critical experiment and analysis

A study[1] has been performed on the temperature coefficients of reactivity in light-water moderated and reflected UO_2 cores with soluble poisons such as boron and gadolinium. Experiments were carried out using the Tank-type Critical Assembly (TCA) in JAERI. Temperature coefficients were measured from the room temperature to about 60°C. It was found that the temperature coefficients are always negative in the experimental cores containing boron as soluble poison. On the other hand, the temperature coefficients become positive in the cores with gadolinium due to the deviation of the gadolinium absorption cross section from the $1/v$ law.

A series of tight-pitch lattice core experiments using highly-enriched-uranium fuels has been performed at KUCA (the Kyoto University Critical Assembly) B-core[2]. A fuel lattice of Vm/Vf=0.65, which is the most tight-pitch lattice achievable using the present fuel inventory at KUCA, has been utilized. Various material loading zone adj acent to the core has been placed at the upper reflector region, in order to simulate the spectral shift zone for the proposed controlled irradiation facility at the upgraded KUR.

Fuel bunching effects on reactivity were measured in a polyethylene moderated core loaded with highly-enriched-uranium (HEU) fuel at KUCA[3]. The core selected as a reference in a series of experiments had a tight-pitch lattice of which moderator to fuel volume ratio was 0.97. It was found that the measured fuel bunching effects on reactivity were negative and the critical mass become larger with increasing degree of fuel bunching. The validity of the unit cell calculation by the collision probability method for a uniformly bunched cell was examined in comparison with the continuous energy Monte carlo calculation.

To examine the applicability of foil activation technique for the estimation of neutron spectrum in a thermal reactor, Cd ratios of 8 activation foils (Au, Th, Dy, In, Mn, W, D.U. and E.U.) were measured in the void at the core center of KUCA B3/8"P36 EU-NU-EU assembly[4]. The Cd ratios were analyzed with SRAC code system using 107 group cross sections based on ENDF/B4. The calculated values except for W and D.U. almost agreed with the experimental ones. For W and D.U.C/E values were ~1.1. Since Cd ratios are sensitive to the change in neutron spectrum except for D.U., this method is useful to judge the appropriateness of calculated neutron spectrum.

The accuracy of the nuclear design of the High-Temperature Engineering Test Reactor

(HTTR) was evaluated through the critical experiments at the Very High Temperature Reactor Critical Assembly (VHTRC) on the reactivity worth of burnable poison rods, and that of water ingress by inserting polyethylene into the core. Measurement of the effective delayed neutron fraction by substitution of fuel compacts by Mn pellets[5].

BWR technologies

The neutron emission-rate (NER) method for burnup measurement of a spent LWR fuel bundle submerged in water has been studied[6] by focusing on the spatial higher harmonics effects inevitable for the application of a formula based on the one-point reactor model. Verification studies of this method have been made through calculations and simulation experiments using the fresh UO_2 fuel-rod bundles of $n \times n$ arrays ($3 \leq n \leq 23$, $0.22 \leq k_{eff} \leq 0.94$), and its applicability has been shown for the spent fuel bundle. As a practical example, an actual formula has been given for a spent PWR fuel bundle of 14×14 array of 3.5 wt% initial enrichment and cooling time of 1,000 days.

The control rods in the reactor of the nuclear ship MUTSU are classified into four groups: groups G1 and G2 are located in the central part of the core, while groups G3 and G4 are in the peripheral zone of the core. Several types of mutual interference effects among these control-rod groups were observed during reactor physics experiments with this reactor. During normal hot operations, positive shadowing was dominant between the G1 and G2 groups; the degree of the shadowing effect of one rod group depended on the position of the other rod group. Both positive and negative shadowing effects occurred between an inner rod group (G1 or G2) and an outer group (G3 or G4) depending on the three-dimensional arrangement of the control rods. A three-dimensional diffusion calculation with internal control-rod boundary conditions has proved to be useful for analyzing these various interaction effects.[7]

Plutonium recycling is an important policy for saving natural uranium resources in Japan, four demonstration mixed-oxide (MOX) fuel assemblies were loaded at Mihama Unit and irradiated for three cycles. MOX fuel assemblies are compatible with the 14×14-type UO_2 fuel assemblies and have a burnup capability equivalent to 3.4 wt% enriched UO_2 fuel PuO_2 average enrichment is ~3.8 wt%. With the experience of MOX fuel irradiation in Mihama Unit 1, we verified our current nuclear design methodology for application to the design of MOX fuel.[8]

Research and Test Reactors

The Japan Materials Testing Reactor (JMTR) is a 50-MW light-water-moderated and -cooled tank-type reactor using Engineering Test Reactor-type fuel. Core conversion from medium-enrichment uranium (MEU) aluminide fuel to low-enrichment uranium (LEU) silicide fuel of the JMTR is scheduled for 1993. The LEU silicide fuel element at 4.8 gU/cm^3 with cadmium wires in the side-plates as burnable absorbers has been selected to achieve upgraded fuel cycle performance of extended cycle length and reduced control rod movement during operation. The small amount of neutron flux change during the cycle is useful for irradiation experiments. Cadmium wires are almost depleted by the end of the operating cycle.[9)]

Xenon oscillations of a generic large graphite-moderated reactor have been studied by a multi-group diffusion code with two- and three-dimensional core models to study the effects of the geometric core models and the neutron energy group structures on the evaluation of the Xe oscillation behavior. The study clarified following; It is important for accurate Xe oscillation simulations to use the neutron energy group structure that describes well the large change in the absorption cross section of Xe in the thermal energy range of 0.1~0.65eV, because the energy structure in this energy range has significant influences on the amplitude and the period of oscillations in power distributions. Two-dimensional R-Z models can be used instead of three-dimensional R-θ-Z models for evaluation of the threshold power of Xe oscillation.[10)]

In order to improve the thermal neutron dose distribution in tissue for neutron capture therapy (NCT), the following three methods were studied: 1) the adjustment of the directional components of the thermal neutron beam, 2) the making of a void in a human body, and 3) the partial replacement of body water (i.e. light water) with heavy water in a human body[11)]. Method 1 is particularly in connection with the other methods. In order to confirm the effects of the direction characteristics of the thermal neutron beam and the size of irradiation field, simulation calculations were performed using MCNP-V3A. It was found from the calculational results that the adjustment of the irradiation field size and the incidence angle of thermal neutrons was important in designing an irradiation facility and dose estimation.

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5. Advanced Core Design

A design study was carried out to evaluate performance of large oxide fueled cores, focusing on reduction of sodium void reactivity to further enhance the inherent safety characteristics of commercial size liquid metal fast breeder reactors. The axially heterogeneous core having an axial sodium-filled gap above it gave a reasonably small sodium void reactivity without imposing significant penalties on the burnup reactivity swing and breeding. This should offer certain economic advantages, especially through an extended refueling interval and higher burnup. A transient safety analysis showed that the core is quite sluggish in response to an unprotected-loss of flow event and the energetics potential is considerably reduced in the initiating phase.[1)]

MA-burning batch-refueling core concepts have been devoted for future FBRs by employing technologies to be developed prior to their commercialization, which attain high burnup, a long operation cycle length, and a high-breeding ratio. A new technique is adopted to reduce sodium void reactivity, enhancing the safety characteristics of the MA-burning core concepts.[2)]

The introduction of zirconium-hydride ($ZrH_{1.7}$) layer between the seed and blanket is very effective in reducing the coolant-void reactivity of steam-cooled FBR. The void reactivity reduction is attributed to the rapid increase of neutron absorption and to the decrease of neutron production in the blanket due to the moderation through the layer. The reason is that the neutron multiplication factor decreases, reflecting the neutron balance of the whole core. It is effective even the layer thickness is 1 or 2 cm. Compared with the conventional uniform introduction of this moderator into the seed, the fixed layer concept is more effective in reducing void reactivity and hardly deteriorate the breeding ratio. The negative void reactivity is proved for the non-flat large-sized radial heterogeneous core where the layers are placed between the seeds and blankets. The neutron absorption rate increases, the fast fission rate decreases in the central, inner and radial blankets.[3)]

Recently, a number of low sodium-void worth metal fueled core design concepts have been proposed; to provide for flexibility in transuranic nuclide management strategy, core designs which exhibit a wide range of breeding characteristics have been developed. Two core concepts, a flat annular (transuranic burning) core and an absorber-type parfait (transuranic selfsufficient) core, are selected for this study. In this paper, the excess reactivity management schemes applied in the two designs are investigated in detail. In addition, the transient effect of reactivity insertions on the parfait core design is assessed. The upper and lower core regions in the parfait design are neutronically decoupled; however, common coolant channel creates thermalhydraulic coupling. This combination of neutronic and thermalhydraulic characteristics leads to unique behavior in anticipated transient overpower events.[4]

A large FBR core concept which has low coolant temperature reactivity coefficient was studied on its passive reactor shutdown capability. Zero or negative coolant temperature coefficient was performed, using sodium plenums around the core, core height shortening and center blanket subassemblies. It was confirmed that the concept enables us to obtain passive shutdown capability for large FBRs.[5]

For creating a FBR plant concept suitable for a FBR commercialization stage, one of desirable concepts is to be a simplification, which is a very effective way to improve safety, reliability and operability, etc. In this study, desirable core concepts, which have the passive safety with no boiling, no fuel melting characteristics against ATWS events, has been studied. An axially heterogeneous ductless core with upper sodium arrangement has been selected and key performance in core and safety design have been confirmed.[6]

A feasibility study has been performed on friendly FBR to fuel cycle, which is designed to transmute minor actinides (MA) and long-lived fission products (FP), and to simultaneously ensure inherent safety of the core and plutonium (Pu) breeding capability. in this report it is shown that an FBR based on an absorber type Axially Parfait Annular Core satisfies these requirements. It is also shown that a fuel handling system for fuels containing MA is suitable for direct fuel discharge with passive heat removal for reduction of transuranic (TRU) inventory outside reactor.[7]

A conceptual design study on Ultra Large FBR Plants (ULFBRPs) with highly economical and safety features, being expected to be introduced in the mid-21st century, has been performed. Not only to replace LWRs but also to meet the increasing energy demand, the power output of ULFBRP was set to be 4000MWe as the environment in the era of ULFBRP require enhanced safety features compared to the present-day FBRs. Nitride fueled 4000MWe cores seemed to suit better than oxide one with ULFBRP objectives. An innovative 4000MWe loop-type reactor with top-entry configuration is conceptually established. Feasibility of 4000MWe FBR plants was clarified. Further an innovative fuel handling system shows capability to enhance the economical advantages of ULFBRP.[8]

The concept of using a fast reactor core for high thermal flux reactor was studied by means of a one-dimensional nuclear analysis. Reactor configurations were investigated to obtain high thermal flux under design constraints on the total reactor power and peak power density. Graphite and heavy water were compared as the reflector material, and oxide and metallic fuels were also compared. The power density necessary for a maximum thermal flux of 10^{16} n/cm^{2}s was evaluated and compared with a conventional heavy-water-cooled high-flux reactor. The results of the analysis showed that this flux level can be attained in the present reactor at a lower power density than in a conventional high-flux reactor.[9)]

Aiming at building-up the foundation for the development of the thorium fuel cycle, the fundamental study have been carried out since 1990, and the final report was published.[10)] In this report, some design studies with the Th/U-233 fueled reactors are reported: the light water reactor with harder neutron spectrum as a U-233 production reactor, inherent safe and ultra long life lead and lead-bismuth cooled fast reactors with negative coolant void coefficient, the molten salt reactor for minor actinide transmutation and the study of thorium molten-salt nuclear energy synergetic.

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6. Fusion Reactor Neutronics

As regard to the expiration of the ten yrs long JAERI/USDOE collaborativae program on Fusion Blanket Neutronics Experiments, efforts were devoted to summarizing results and their publications. Three joint reports were issued on Phase-IIC experiments and analysis, and induced radioactivity experiments.[1)-3)] The first report on Phase-IIC experiments[1)] described in detail the experimental configuration in terms of system arrangements, neutron source characteristics, experimental procedure. All experimental data were tabulated so as to be used successive experimental analysis. The experimental analysis was given in the second report.[2)] Results from both JAERI and US were combined and discussed based on C/E values. Major focus was placed on the prediction accuracy of calculation with comprehensive nuclear data on tritium production rate in the Li_2O breeder blanket. The third joint report on the induced radioactivity experiments[3)] also included all data in terms of materials concerned, irradiation filed, counting scheme and experimental data. The unique data base is expected to be used for the validation of activation cross sections and inventory codes needed for the reactor design calculations.

Concerning the importancof neutronics relevant issues addressed in the ITER-EDA R&D program, experiments have been under way to provide key data for validating the codes and data to be used in the ITER design. Through analyses of the bulk-shielding experiment on large SS-316 stainless steel assemblies performed at FNS, importance of the self-shielding effect on shielding designs of large scale fusion devices was studied.[4)] When the self-shielding effect was not considered, DOT calculations largely underestimated resonance energy neutrons up to a factor of 2.5 at a depth of 0.91 m and overestimated gamma-ray production 35 % larger against MCNP calculations. The DOT calculations with self-shielding correction and MCNP agreed with experiments within a few tens of percentage, validity of the used nuclear data library based on the JENDL-3 was conformed. It was pointed out that the self-shielding correction is required in response functions for shielding design parameters, such as activation, displacement damage and neutron dose. In conclusion, it is inevitable for shielding designs of large scale fusion device to take the self-shielding correction into fundamental consideration.

The wall effect correction has been investigated for a small cylindricalproton recoil proportional counter used as a detector for neutron spectroscopy in an energy range of 0.01 -1 MeV.[5)] The probability that a proton escapes to wall was calculated by the method based on a path length probability function. The probability is applied to calculation of the counter response for the isotropic neutron field. This distribution of recoiled protons measured by a hydrogen-argon filled proportional counter. The effect of the correction on the unfolded neutron spectrum was examined.

Responding to the termination of second internatonal comparison on measuring techniques of tritium production rates for fusion neutronics experiments, a summary report was issued.[6] The purpose of this program was to evaluate the measurement accuracy of tritium production rates in the current measurement techniques. Two 14 MeV neutron source facilities, FNS at JAERI and LOTUS at EPFL in Switzerland, were used for this purpose. Nine groups out of seven countries participated in this program. A fusion simulated blanket assembly of simple-geometry was served as the test bed at each facility, in which Li-containing samples from the participants were irradiated in an uniform neutron field. The tritium production rates were determined by the participants using their own ways by using the liquid scintillation counting method. Tritiated water sample with unknown but the same concentration was also distributed and its concentration was measured to make a common reference. The standard deviation of measured tritium production rates among participants was about 10 % for both FNS and LOTUS irradiation levels which were corresponded to 4×10^{-13} T-atoms/Li-atom and 1.6×10^{-12} T-atoms/Li-atom at a sample, respectively This standard deviation exceeds the expected deviation of ±5 % in this program. It was presumed that the deviation of ±10 % was caused mainly by the systematic and unknown errors in a process of tritium extraction from the irradiated samples depending on each organization.

A benchmark calculation was performed on deep penetration in 3 m thick iron assembly driven by 14 MeV, using MVP continuous vectorized Monte Carlo code with JENDL-3 and ENDF/B-VI nuclear data.[7] High accuracy in reference neutron spectra and averaged cross sections at several positions in the assembly was achieved. The accuracy of multi-group calculation with JSSTDL/J3 was verified by comparing results with the reference calculations. Though both calculations with JENDL-3 and ENDF/B-VI gave similar attenuation profiles for total neutron flux, there was large difference at most one oder of magnitude in the calculations for the neutron flux at the resonance region.

An integral experiment with a shell of $P/D_2O/Li$ was carried out at OKTAVIAN facility at Osaka University.[8] To study a "compact" Pb-Li blanket, heavy water zone (100 mm thick) was sandwiched between an inner Pb neutron multiplier (100 mm thick) and a outer Li sphere (40 mm thick) to measure time-dependent profile of the $^6Li(n,t)$ reaction. A reference experiment with a light water zone (100 mm thick) was done. Measured reaction rates were compared with NITRAN-TD calculations using ENDF/B-VI or JENDL-3 nuclear data; Good agreements were obtained except Pb region where about 10% underestimation of calculations were found.

haracteristics of neutrons generated from the lithium target bombarded with high energetic deuterons of 10-40 MeV have been studied to determine the specification for the neutron irradiation material test facility (ESNIT) planned at JAERI.[9] The simple nuclear reaction model was applied to estimation of neutron flux distribution and energy spectrum, and the results showed an agreement with the reported experiment within a actor of 2. The calculation gave the basic spectrum data for estimation of damage parameters in test samples to evaluate the high energy neutron effect on them.

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7. Nuclear Criticality Safety

For a safety assessment of a nuclear fuel facility such as a fuel reprocessing facility, one needs to know the scale of criticality accidents, assumed to occur in some processes of uranium and/or plutonium fuel solutions. A simplified evaluation model for the scale, i.e. the number of total fissions, was derived theoretically by using the one-point reactor dynamic equation model, and validated through the comparison with the transient experiments and accident data reported so far in foreign countries. The conventional simplified evaluation models were quoted to show the convenience of the proposed model having less constriction in application. Some provisions were also shown concerning the application of the present model to the nuclear fuel facilities processing solution fuels for its safety assessment[1)].

To store spent fuels effectively in a pond, high-density storage systems using neutron-absorbing materials are considered for light water reactor sites. Few criticality experiments are reported on neutron absorption of Boral plates, borated stainless steel plates and cadminum plates. The reactivity properties of fixed neutron absorbers depend on the neutron spectrum of the volumetric ratio of water to fuel in the spent-fuel assembly and the width of the water gap between assemblies. Nuclear criticality experiments with borated stainless steel (B-SUS) plates in single and coupled cores of low-enriched UO_2 fuel rod clusters were performed in the Tank-type Critical Assembly (TCA). The reactivity effect of B-SUS plates as a function of thickness, boron content and position of the B-SUS plates were obtained using the critical water level method[2)].

Critical experiments for an annular core with fixed neutron absorbers in the central test region were also performed using TCA. Either cylindrical concrete containing boron carbide surrounded by aluminum pipes or boric acid solution contained in an aluminum was placed in the central region for various combinations of the diameters and poison concentrations[3)].

With the advancement and proliferation of computers, it has become possible to develop or execute complicated scientific calculation programs with smaller computers. Criticality analysis codes can be executed on personal computers (PCs) and workstations instead of mainframes. Experience utilizing safety analysis codes KENO IV and KENO Va on PC and workstations at a Japanese nuclear fuel manufacturing facility was presented[4)].

In criticality safety analysis of transport casks, an infinite array of fuel rods using a homogeneous cell is often assumed and criticality calculations are performed with the homogeneous cell, since the configuration of transport casks are complicated. Because considerable fraction of the total number of the fuel rods are in contact with reflectors, the homonization is inadequate. Features and example calculations of MULTI-KENO, which had been developed to express such configurations exactly without cell calculations, were presented[5].

It is well known that the design concepts of transportation systems, storage facilities, reprocessing processes, etc., of spent nuclear fuels can be economized due to the significant decrease in the reactivity of these fuels as a result of burnup. Burnup data were collected from 13 LWRs including 9 LWRs (5 PWRs and 4 BWRs) in Europe and USA, 4 LWRs (2 PWRs and 2 BWRs) in Japan. The collected data include such information necessary for benchmark burnup calculation as the irradiation history of the fuel samples, the composition of fuel assemblies, the sampling position and the isotopic composition of the fuel samples[6].

However, the criticality and subcriticality data of spent fuels are still not sufficient to verify criticality calculation results. These data should be systematically measured as a function of the burnup ratio, burnup history, etc. Therefore an experimental program to measure subcriticality data of spent fuel assemblies in a storage pool has begun. The objectives of this program are to apply an exponential experimental technique not only to present systematic data for fissile isotopic content, burnup history, etc., that are needed for the verification of burnup calculation codes, but also to obtain benchmark data for verification of criticality calculation codes and to demonstrate this technique as a viable method for the pre-loading burnup measurement[7].

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8. Shielding

The Working Group of Accelerator Shielding in the Research Committee on Radiation Behavior in the Atomic Energy Society of Japan published a compilation of accelerator shielding benchmark problems to be utilized for accuracy estimation of shielding calculation codes for high energy accelerators.[1] Twenty-five problems were presented for evaluating the calculational algorithm, the accuracy of computer codes and the nuclear data used in the codes. Fifteen problems are for electron accelerators and 10 problems are for proton accelerators. They are also categorized as 10 transmission experiments, 10 beam dump ones, 3 streaming ones and 2 skyshine ones.

As for studies about buildup factors, two articles were issued. One of them is an histroical review and current status of buildup factor calculations and its applications, covering the Goldstein-Wilkins buildup factors to the latest data fitted by the so called G-P formula.[2] The other presents the exposure buildup factor in stratified shields including the effects of Bremsstrahlung and fluorescent radiations.[3] Exposure buildup factors, energy spectra and angular flux distributions were calculated for plane-normal incident and point isotropic source gamma-rays of 0.1, 0.5, 1, 3, 6 and 10 MeV penetrating two-layer water-lead and lead-water shields were calculated with the point Monte Carlo code EGS4.

As for the EGS4 code, it has recently been modified to include linear polarization in the simulation of photon scattering.[4] Both the Compton and Rayleigh scattering routines were modified to properly account for the electric-field vector of the photon. A simulation calculation was successfully carried out for the absorbed-dose distribution in a soft tissue equivalent phantom for a linearly polarized incident photon beam.

A continuous energy Monte Carlo code MCNP was utilized to test the iron, carbon and beryllium cross sections in JENDL-3 through analysis of the neutron shielding benchmark experiments.[5] For the calculations, revision of the subroutine TALLYD and an appropriate weight-window-parameter assignment was accomplished in the MCNP code. The Monte Carlo calculations with JENDL-3 showed a good agreement with the measurements in a wide energy range as a whole, with an exception for the iron cross section at the energy range from 0.8 to 3.0 MeV where distinct underestimation was observed.

As for an accelerator shielding, some remarkable progresses have been realized. The accelerator shielding programme using a 90 MV AVF cyclotron TIARA in JAERI has started as a cooperative scientific project between JAERI and some universities in Japan. Quasimonoenergetic neutrons of about 43 and 70 MeV were generated through ^{7}Li(p,n) reaction with monoenergetic protons of 45 and 70 MeV and they were confirmed by a proton recoil telescope.[6,7] Penetration experiments utilizing these neutrons were made for ordinary concrete and iron shields, measuring energy spectra, reaction rates of ^{238}U(n,f) and ^{232}Th(n,f) and dose rates.

A HILO86 library, a high energy neutron and photon coupled group cross section, was revised to generate a HILO86R library by taking into account the self-shielding factor for the cross section of ten materials below 19.6 MeV.[8] Development of a code system is now in progress to evaluate the fluence to dose equivalent conversion factors for photons from 10 MeV to 10 GeV and for neutrons 15 MeV to 10 GeV. Calculations for high energy photons suggested the ambient dose equivalent (1cm depth dose) in a sphere ICRU phantom underestimated the effective dose based on the ICRP recommendation 60.

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9. Incineration of Radioactive Nuclear Wastes

Transmutation of minor actinides (MAs) was investigated for two types of BWR fuel (UO_2 and MOX) and for an HCR.[1] Irradiation calculation for pure MAs shows that, while transmutation by fission is scarce, half-lives of principal daughter actinides are much shorter than those of parent MAs, indicating potential transmutation of MAs in these reactors. Concentrated recycling of MAs was evaluated for these fuels, where additional enrichment was determined to compensate for reactivity penalty due to MAs. Fuel material balance, transmutation rate, reactivity coefficient, and other fuel cycle effects were discussed. It is concluded that, though transmutation rate is the highest in UO_2 fuel, MOX fuel is most suitable as MA-carrier, because of its minimum fuel cycle effects.

A super long life core (SLLC) loaded with MA fuel was designed aiming at continuous operation without refueling during plant lifetime and efficient reduction of MA nuclides.[2] The feasibility was studied from nuclear and thermal characteristics assuming that high burnup fuel loaded with MA nuclides can be employed. A 300MWe SLLC with small reactivity change and power swing during plant lifetime was found feasible. The reactivity change during lifetime is only 2.5%Δk and it can be easily controlled by control rods of conventional design. Total weight of MA nuclides (Np,Am,and Cm) decreases by about 5.3ton during lifetime. Annual amount of transmutation is about 160kg, which is nearly equal to the amount of MA nuclides from 6 LWRs of 1000MWe. The core is expected to meet the thermal design criteria although optimization of the number of blanket assemblies is required. The reactivity coefficients and dynamics parameters are different from those of the conventional core. The influence on core safety and control is an issue to be studied further. It can be said that the low linear heat rate is advantageous from the view point of core safety.

Minor actinide transmutation capability to incinerate very long lived transuranic elements with heterogeneous MA-recycling in internal blanket fuels in a 1000 MWe-class fast breeder U-type Asymmetric Parfait Core (APC) was evaluated.[3] An MA-internal blanket loading U-APC core has a certain potential without any significant impacts to the core characteristics compared with those for a conventional homogeneous cores used high MA content core fuels, when MA-burning requirements will be increased in the course of establishing the MA-recycling technology. Fast reactors have certain feasibility for MA reduction capability maintaining fissile inventory constant, reflecting various fuel cycle development status.

Conceptual design studies of minor actinide burner reactors to obtain a reactor model with very hard neutron spectrum and very high neutron flux in which minor actinides(MA) can be fissioned efficiently. Two models of burner reactors were obtained, one with metal fuel core and the other with particle fuel core. Minor actinide transmutation by the actinide burner reactors is compared with that by power reactors from both the reactor physics and fuel cycle facilities view point. In these burner reactors, the MA burnup rate per cycle is significantly higher than those in power reactors. MA transmutation in power reactors (LWRs and FBRs) will require the design change of the radiation shielding in the whole fuel cycle facilities because of the increase of strong neutron-emitting nuclides.[1]

The original ABR design was modified to improve such safety characteristics as the small Doppler reactivity coefficient, delayed neutron fraction and the large positive sodium void coefficient. In the modified design nitride fuel of MA is used. Mixture of enriched uranium in MA fuel increases effective delayed neutron fraction and neutron generation time significantly. Sodium was replaced with liquid lead as coolant to avoid the large positive sodium void effect.[2]

The preliminary design of an accelerator-based minor actinide transmutation plant with a solid target/core was studied. The plant consists of a high intensity proton accelerator, spallation target of solid tungsten, an subcritical core loaded with actinide alloy fuel. Minor actinides are transmuted by fast fission reactions. The target and core are cooled by the forced flow of liquid sodium coolant. The core with an effective multiplication factor of about 0.9 generates the thermal power of 80 MW by using a 1.5 GeV, 39mA proton beam. The average burnup is about 8%, about 50 kg of actinides, after one year operation at an 80% of load factor.[3]

An accelerator driven transmutation system with fluid target was also studied. The system equipped with an on-line separation loop for continuous removal of stable and short-lived transmutation products. In the preliminary study, molten chloride salt of NaCl-(Pu,MA)Cl_3 system is adopted as a target salt for MA transmutation, and Bi-Pb alloy for target fluid for neutron production to transmute FP in a heavy water blanket.[4]

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10. Noise Analysis and Control

With the aim of clarifying the process of deriving a time-series model for reactor noise analysis, a systematic and constructive approach[1] has been shown to develop an autoregressive moving-average (ARMA) model of time-series data starting from basic Langevin equations representing a multivariate stochastic Markovian system without measurement noise. This approach can also be applied easily to the system with measurement noise. Under the assumption of the irreducibility of the ARMA model, properties of zeros of the model are also studied using the system matrix representation, and conditions are presented to calculate the number of zeros of the model.

In order to study the closeness of a fitted multivariate autoregressive (MAR) model to actual properties of a dynamical process, a MAR representation of a vector noise process is derived theoretically from a set of input-output relations describing a system dynamics and an observation process with additive noise. It is shown that all information about the system is fully conveyed to the MAR model, but it appears in a contracted form.[2] In view of the fact that the MAR model thus derived can be determined experimentally using an observed time series, the problem of extracting properly the characteristics of the system dynamics is discussed, especially for open-loop characteristics such as direct transfer functions and inherent noise sources. A set of necessary conditions and a desirable checking procedure have also been obtained which are useful for making a more reliable MAR model.

To study the nonlinear dynamic characteristics of nuclear reactors on the basis of observed time series data, a new method of reactor noise analysis has been developed from the view point of chaos and fractal analyses.[3] The method utilizes the information dimension, which is one of the fractal dimensions and is the simplest quantity that can be used to determine the asymptotic behavior of the time evolution of a nonlinear dynamical system. The application of this quantity to reactor noise analysis has been proposed, and the possibility of its application to power oscillation analysis has been examined. The information dimension of this regime is equal to the number of independent oscillating modes, which is an intuitive physical variable. The validity of the proposed method has been checked by using first the time series data obtained from computer simulation of BWR dynamics and then the experimental data of power oscillation observed at an actual research reactor. The results indicate that the method is useful for a detailed analysis of reactor power oscillation.

A new concept for control of xenon oscillation[4] has been applied to solve an optimal control problem. The concept is based on two additional newly defined axial offsets, AOi and AOx together with the conventional axial offset AOp of axial power distribution. The AOi and AOx are the axial offsets of power distributions which would give the current iodine and xenon distributions under an equilibrium condition, respectively. Using these three AOs the necessary condition for a xenon oscillation to be controlled or suppressed can be clearly and simply expressed. In other words, when these three AOs meet together that is the point that the xenon oscillation dies out. It was found that an optimal control search can be performed very quickly and simply using the three axial offset concept. Using this simple requirement as the target for xenon oscillation control, the solutions of some optimal problems can also be obtained quite straightforwardly. An example of solving the minimal time control problem is shown as bang-bang control.

A study has been made on the application of fuzzy logic control schemes[5)] to automatic startup operations and load-follow control in boiling water reactors, pursuing the advantages of the fuzzy logic control methods in facilitating direct use of skillful operator knowledge for plant operation and providing robust multivariable control for complex plants. Some typical linguistic control rules of IF-THEN type are chosen in implementing the operator's control strategy for automatic pump switching, while coordinative IF-THEN rules are employed in the pressure-load coordinated control method for load-follow control. Results of computer simulation confirmed the capability of the proposed fuzzy logic control schemes to improve performance and robustness.

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11. National Programs

1. JOYO

Three duty cycles and one special operation have been completed and after then the 10th periodic inspection has been started, in this period.

In these duty cycles, four special type fuel subassemblies and thirteen materia irradiation rigs have been loaded in the core for irradiation tests of the LMFBR fuels, materials and fusion reactor materials. A type-B special fuel "B9", which was conducted for the verification of ferric steel fuel cladding, has been irradiated through the 25th duty cycle to the 27th duty cycle.

The second Fuel Failure Simulation Test was performed in the 25th special operation for studying the fission gas behavior in the cover gas. The test was carried out with an irradiation subassembly containing two fuel column predefected pins and a sibling fuel pin. The performance of the Cover Gas Clean-up System was also demonstrated.

The 10th periodic inspection is now in progress and will be continued till March 1994.

The MK-III program, which aims the improvement of irradiation performance of JOYO, is now on schedule. Extensive modification design works of the core and the cooling system have been completed.

Some efforts for R&D have been done (or continued) as follows :

- Analyses of the core distortion behavior
- Analyses of the nuclear characteristics
- Evaluation of the linear heat rate for the second power-to-melt test
- Neutron fluence measurement with dosimetry using activation technique

2. MONJU

The function tests, which are the first phase of the pre-operational test program of MONJU aiming to confirm the integrated system functions, were completed in December 1992. The non-nuclear plant performance tests, composed of the plant heat-up test by pumping heat and the procedure confirmation of equipments which will be used in the physics tests, were conducted till June 1993. Then the check and review of whole the reactor system before fuel loading was performed.

It is expected to get the first criticality in the spring of 1994. Measurements of reaction rate distribution, control rod worth, reactivity coefficients, coolant flow rare in the core, etc.

are planned in the following physics tests.

3. Demonstration Fast Breeder Reactor

"Conceptual Design Study of DFBR" was conducted to determine the plant concept and specifications of DFBR plant. The design study was focused on the evaluation and improvement of design margin in the key components in the reactor system. Based on the study, the plant concept was established and the safety features and the construction cost of the plant were evaluated including the sodium supplementary system and balance of plant.

Further studies for DFBR have also been continued on the following items :

- Element technologies to improve safety, economics, operability and maintainability,
- Safety evaluation of specific events,
- Design standards for structural materials, fuel pin material, etc.,
- Design analysis methods.

4. FUGEN

Fugen has continued stable full power operation from September 1992 to August 1993 excluding an unscheduled shutdown in October 1992 and a scheduled outage for the 19th refueling in February 1993. And now, a scheduled outage for the 20th refueling and the 11th annual inspection has been conducted since August 1993.

In the 19th refueling, 22 plutonium-uranium mixed oxide (MOX) and 14 UO_2 fuel assemblies were charged into the core. As a result, core configuration was 118 MOX (including experimental fuel) and 106 UO_2 fuel assemblies. And 28 MOX and 12 UO_2 fuel assemblies will charged in the 20th refueling.

At Fugen, up to date, 529 MOX and 476 UO_2, fuel assemblies have been discharged, and 429 MOX and 388 UO_2 fuel assemblies have been discharged for refueling.

The maximum burn-up is 19,900MWd/t for UO_2 fuel and 19,600MWd/t for MOX fuel, and no leaking fuel has been found for more than 3526 effective full power days of operation up to the end of August 1993.

Sixty eight UO_2 fuel assemblies were transported to PNC-Tokai reprocessing plant during the reported period. An experimental segmented fuel assembly was taken out from the core in the 19th refueling, but 6 Gd_2O_3 poisoned MOX fuel assemblies were loaded continuously.

In October 1992, Fugen had a two-week-long unscheduled shutdown due to small steam leakage from a main steam pipe of a turbine system. On March 20, 1993, Fugen attained the 15th anniversary of the first criticality.

5. ATR Demonstration Plant

The construction program of the ATR Demonstration Plant has started with the decision given by the Japan's AEC in 1982 that EPDC (Electric Power Development Company) be responsible, in a close cooperation with the government, electric utilities and PNC, for the construction and operation of the plant.

ATR is a heavy-water-moderated, boiling-water cooled, pressure tube type reactor originally developed by PNC. EPDC took over the results of PNC's design development work for the ATR Demonstration Plant, and started the plant design work. At present, EPDC is finalizing the design for an application for construction permit.

The capacity of the plant is 606 MWe and the whole core can be fueled with MOX fuels. The plant is expected to be located at the site in Ohma-machi Simokita-gun, Aomori-ken, According to the current schedule of the project, the construction is to start in 1996, and the commercial operation in 2002.

6. High Temperature Engineering Test Reactor (HTTR)

The construction of the HTTR facility has been successfully in progress since March 1991. At present, construction of the HTTR reactor building has been almost completed and its main components, such as a reactor pressure vessel, an intermediate heat exchanger and graphite core structures, are now manufacturing at their factories. Their installation will start in 1994 and the first criticality will be attained in 1998.

The HTTR is a block type graphite moderated, helium gas cooled reactor, loaded with low enriched uranium coated particle fuel. Its thermal output is 30 MW and its outlet coolant temperature is 850/950°C.

The objectives of the HTTR are,

(1) establishment of HTGR technologies,

(2) fuel irradiation test in large scale,

(3) tests for passive safety features of HTGRs,

(4) tests for industrial usage of nuclear heat (hydrogen gas production etc.) and

(5) high temperature irradiation tests for innovative basic researches.

7. OMEGA Program

The Japanese Atomic Energy Commission concluded in 1988 that R&D efforts for incineration of radioactive nuclear wastes should be substantially strengthened as the national research projects, which is aiming at the possible use of valuable resources in the wastes and aggressive improvements of safety assurance in the wastes management processes. This project is named the OMEGA Program.

According to the proposal by Japanese government, the OECD/NEA decided to initiate an Information Exchange Program on Actinides and Fission Products Partitioning and Transmutation in 1989, as a five-years program. Under this framework, the second information exchange meeting was held at Argonne National Laboratory in Illinois, USA November 11-13, 1992, where progress of partitioning and transmutation (P-T) technologies was discussed.

The OECD/NEA Nuclear Science Committee decided to initiate new task forces relevant to the P-T technologies in June 1992, such as "Review of Physics Aspects of Different Transmutation Concepts", "Benchmarks on Spallation Neutron Source for Accelerator-driven Transmutation System", "Thermodynamic Data Base for the P-T technologies" and "International Code and Model Intercomparison for Intermediate Energy Reactions".

8. Light Water Reactor

Three light water power plants (Oi Unit 4 PWR 1180 MWe, SHIGA Unit 1 BWR 540 MWe and HAMAOKA Unit 4 BWR 1137 MWe) were brought into commercial operation in February 1993, July 1993 and September 1993, respectively. Total capacity of the nuclear power plants is presently 36261 MWe.

Experimental voyage of the nuclear ship "MUTSU" equipped with 36 MWt PWR was completed by February 1992 and its decommissioning works were initiated. She navigated by the nuclear power on the total distance of about 82,000 km. Her valuable experimental voyage data are being analyzed by using the nuclear ship engineering simulation system at JAERI.