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Reactor Physics Activities in Japan (June 1991－July 1992)

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This report reviews the research activity in reactor physics field in Japan during June, 1991－July, 1992. The review was performed in the following fields : nuclear data evaluation, calculational method development, fast reactor physics, thermal and intermediate reactor physics, advanced core design, fusion reactor neutronics, nuclear criticality safety, shielding, incineration of radioactive nuclear wastes and national programs.

The main references were taken from journals 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

JAERI－M 92－209

日本における炉物理活動
（1991年6月～1992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

（1992年12月7日受理）

本報告は，1991年6月から1992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，熱及び中間中性子炉物理，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，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 report reviews activities in the field of the reactor physics in Japan from June, 1991 to July, 1992. Hitherto, the review activities had been performed in order to present the summary on the reactor physics activities in Japan to OECD/NEA Reactor Physics Committee(CRP). Although the activities of CRP is absorbed into the newly born NEA Nuclear Science Committee, the Reactor Physics Committee of JAERI/Atomic Energy Society of Japan argued the way of succession of the review and decided to publish as JAERI M report in the almost same form as before independently from NEA activities because this report might be a single document to review the reactor physics activities in Japan.

The contributors to this document are

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1. NUCLEAR DATA EVALUATION

As to the JENDL-3 projects, publications for JENDL-3 Fission Product Nuclear Data Library, JENDL Dosimetry File and JENDL Gas-production Cross Section File were made.

Neutron cross sections of fission product(FP) nuclides are important to predict burnup performance of fission reactors. Total of 172 nuclides from ^{75}As to ^{159}Tb were evaluated[1] to provide data for the JENDL-3 fission product nuclear data library. Evaluation was made on the basis of recent experimental data reported up to 1988 and the nuclear model calculations. The paper gives overviews of the evaluation process employed.

The JENDL Dosimetry File based on JENDL-3 was compiled[2] and integral tests of cross section data were performed by the Dosimetry Integral Test Working Group of JNDC. In the file, total of 61 nuclides of cross sections mainly taken from JENDL-3 and their covariance data taken from IRDF-85 are given. Several integral tests were made to confirm the applicability comparing the average cross sections with measured ones in various neutron fields such as fission neutron fields, fast reactor spectra, DT neutron fields and Li(d,n) fields. It is found that the file gives better results than IRDF-85.

The JENDL gas-production cross section file was compiled[3] by taking cross-section data from JENDL-3. The data are given to 23 nuclei or elements in light nuclei and structural materials.

To contribute to the activation cross section library, a model calculation of neutron activation cross section was carried out[4] with SINCROS-II for all stable isotopes of molybdenum. The general agreement between the calculated cross sections and measured ones were obtained. The calculation method is confirmed to be applicable to the estimation of production cross sections for the long-lived radioactive nuclides.

Sixteen neutron activation cross sections were measured[5] for (n,2n), (n,p), (n,n'p) and (n,α) reactions producing short-lived nuclei with half-lives between 20 s and 7 min in the energy range of 13.4 to 14.9 MeV for F, Mg, Si, Ti, Cr, Ni, Rb, Sr and Ag.

To remove the gaps among the available data for ^{197}Au(n,γ) cross sections, which are one of the standard cross sections in the kev capture work, absolute measurements are made[6] at neutron energies of 23 and 967keV using reaction rate measurements.

To give an experimental background for self-shielding factors, which are very important in neutronic calculations in fission reactors, self-shielding factors for the neutron capture reactions of ^{238}U and ^{232}Th were measured[7] in the resonance energy region of 1-

35keV, using a neutron time-of-flight method with an electron accelerator. The self-shielding factors were obtained from sets of neutron transmission ratios and self-indication ratios. For ^{238}U, an energy dependent structure was observed which is not reproduced by JENDL-2 or ENDF/B-IV. JENDL-3 gives better results but discrepancy is still remained. For ^{232}Th no remarkable discrepancy was observed in the unresolved range, but JENDL-2 and JENDL-3 tended to give smaller values in the resolved resonance region.

To give precise prediction of decay heat generated from fission product, three papers were published in this period.

The first one is a simplified method to evaluate uncertainty of calculated decay heat[8)] by the numerical analyses for the sensitivity coefficients only to the uniform changes of the parameters used for the estimation of fractional yields. The sensitivity for the charge distribution width was less than 1/10 of that for the most probable charge Z_p. Evaluated uncertainties of decay heat for thermal neutron induced fission of ^{235}U after burst irradiation were ranging from 2.98% at the cooling time of 2.7s to 0.74% at 12,000s.

The second one is also a paper for the uncertainty of decay heat[9)], this paper treats for the summation calculations for the thermal fission of ^{235}U and ^{239}Pu and the fast fission of ^{238}U through sensitivity analyses using data given in the JNDC FP nuclear data library. The uncertainties analyzed are those relevant to decay energies, fission yields and decay constants. Thus obtained maximum uncertainties of burst fission were 2.8% for ^{235}U, 3.2% for ^{239}Pu and 4.2% for ^{238}U in the range of cooling time between 1 and 10^9s, and in the case of infinite irradiation, the corresponding level of maximum uncertainty was below 1.6% for all fissioning nuclides.

The third one is the calculation for the β-ray spectra of individual fission products[10)] by using the β-decay data assuming every β-decay allowed transition. For the nuclides without measured decay data the β-feeding function was evaluated with the gross theory of β-decay and the β-ray spectrum was calculated by the function. The β-ray spectra from aggregate fission products after a burst fission were calculated and they are compared with the measured ones for thermal fission of ^{235}U, ^{239}Pu and ^{241}Pu obtained at Oak Ridge National Laboratory. Excellent agreement was obtained for the data at shorter cooling time than 10s.

Neutron yields from (α,n) reaction and spontaneous fission, which are very important in analyzing radiation shielding of spent fuel storage, transport, and safe handling, were collected and evaluated in one book[11)] to determine the recommended values. The book includes thick target yields from (α,n) reactions, neutron yields from spontaneous fission, neutron energy spectra of (α,n) reactions and spontaneous fissions. Neutron production data of compounds can be obtained by using the neutron yield and stopping power of each single element.

Several data books for the nuclear data users are published, they are:

•Curves and Tables of Neutron Cross Sections of Fission Product Nuclei in JENDL-3[12]: containing cross section curves and tables for average cross sections of 38 energy intervals, thermal values, resonance integrals, 14MeV values, fission averages, Maxwellian average and one group cross sections for typical reactor spectrum,

•Comparison of Double-differential Neutron Emission Cross Sections Calculated from Evaluated Nuclear Data Libraries with Experimental Data[13]: containing the DDX data for 32 elements from Li to ^{235}U calculated from JENDL-3 comparing with experimental data and calculated values from ENDF/B-VI and JEF2,

•List of Strong Gamma-rays Emitted from Radionuclides[14]: This is a compilation of intense gamma-rays, with energy values greater than 1 keV, emitted from the decay of radioactive nuclides. The three strongest gamma-rays originating from each nuclides are listed from the ENSDF file as of Feb. 1991. This list is also available in floppy disk.

In this period following three topical meetings were held in Japan at Tokai Research Establishment of JAERI in connection with the Nuclear Data Research.

The Specialists' Meeting on High Energy Nuclear Data[15] was held on Oct. 3-4, 1991, with the participation of forty specialists. The needs of the nuclear data in the high energy region up to a few GeV was appealed for the various applications; from the spallation neutron source for the waste managements to the dosimetry for the space astronauts. The consensus, that the wide collaboration is necessary to produce the evaluated file and should be established, has been obtained.

The 1991 Symposium on Nuclear Data[16] was held on Nov. 28-29, 1991. Fifteen papers in oral, 38 in posters were presented, including nuclear data activities in Thailand, status reviews of JENDL-3 projects, nuclear data needs in OMEGA and ESNIT program, and topics on knowledge technology and reactor physics.

The NEANSC Specialists' meeting on Fission Product Nuclear Data was held on May 25-27, 1992. Total of 50 specialists were participated including 21 from abroad. This meeting was held in the frame work of NEANSC Working Group on International Evaluation Cooperation.

The Fourth Meeting of the NEANSC Working Group on International Evaluation

Cooperation was held on 28-29, 1992 at Tokai establishment of JAERI. In this meeting working arrangements were discussed and revised so as to make official participation of IAEA Nuclear Data Section to the working group members.

Entries from Japan for the World Request List for Nuclear Data WRENDA 91/92 were summarized[17] and they are sent to the NEA Data Bank. In the list total of 182 new requests are registered, most of the new items are related to the OMEGA project undergoing in Japan.

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2. Calculational Method Development

Theoretical works were made in such wide area as coupled reactor kinetics, noise analysis, application of higher harmonics and response matrix method. These contribute to accurate prediction and understanding of reactor core behavior.

Nodal kinetic equations for coupled reactors, namely multi-point reactor kinetics equations whose dependent variables are the fission sources of each reactor were derived rigorously, using kinetics parameters with the explicit dependence on a perturbation, from the time dependent multi-group diffusion equation. Exact expressions for the coupling coefficients, neutron life time and the change of the coupling coefficients due to the perturbation were obtained. These equations can be applied for any reactor by dividing a core into appropriate subregions.[1)]

A new method for multivariate autoregressive modeling of a vector time series was proposed, which has an advantage in guaranteeing less-biased estimation in a least-squares sense by ensuring orthogonality between the components of a residual vector. An expression for a multivariate autoregressive model with a zero-lag coefficient matrix was derived theoretically and utilized for the modeling based on a weighted least-squares procedure. It was shown that the present modeling differs from the ordinary one in estimating open-loop properties[2)] and that the new modeling yields satisfactory results even when there is a significant correlation between the components of residual noise. Two different sets of multidimensional noise were analyzed: the 1989 benchmark test and the PWR noise[3)].

The improved power method in the higher-harmonic eigenvalue calculation of nuclear plant thermal-hydraulic dynamics was applied to the λ- and ω_P-eigenvalue problems of thermal and fast reactor systems to obtain the two-dimensional, few-energy-group higher-harmonic eigenvalues and eigenfunctions. The efficiency of the method was confirmed in the large-scale neutron diffusion problems and the interesting spatial patterns of the eigenfunction were obtained. The discrepancy between λ- and ω_P-eigenfunctions, is significant in a thermal system with low-absorbing region (*e.g.* reflector).[4)]

The finite Fourier transformation method was applied to solve the one and two dimensional diffusion equations. It was shown that the equation by the nodal Green's function method in Cartesian coordinate can be derived as a special case of the finite Fourier transformation method. Numerical examples were given for a hexagonal system and shown to be superior to the conventional finite difference method.[5)]

It is well known that response matrixes are easily calculated by using the symmetries of a node. However, they produce small negative fluxes in the node. To reduce these negative fluxes, a calculational method was developed using the incoming currents averaged on a half side of a node. Each incoming current on the half side is expressed in a vector. Applying symmetrical operators to the vectors, one obtains the various incoming current vectors. From these vectors, a complete orthogonal system is obtained. As the correspondences between the incoming current vectors and the symmetrized neutron fluxes are known, the flux can be calculated, which corresponds to an arbitrary incoming currents on half sides of the node. Refs. 6 and 7 describe the calculational methods of square and hexagonal nodes, respectively. The methods were applied in Ref.8 to the nodes in which strong absorbers are present.

A theoretical expression of reactivity interaction between control rods was developed on the basis of the explicit higher–order perturbation method. The expression shows that the magnitude of the interaction is inversely proportional to the λ–mode eigenvalue separation and dependence on the rod patterns is governed by the higher–harmonic eigenfunctions and the adjoint ones. Application of the formula was illustrated by numerical calculations carried out for a 1–D model of a large fast reactor core.[9)]

The multiband method was applied to analyses of critical experiments related to the high–conversion core at the Kyoto University Critical Assembly in order to accurately treat the resonance self–shielding in heterogeneous cells. Three–band parameters were generated using the self–shielding table installed in the SRAC code. The k_∞ values calculated by this method have been compared to those by the VIM Monte–Carlo calculations. Both results agree within 0.3%Δk for all the cases considered.[10)]

Speedup of Monte Carlo codes is greatly expected in the fields of reactor core, shielding, criticality safety and fusion neutronics calculations. Effort to develop vectorized codes continued for the use on supercomputers. Evaluation of efficiencies on massive parallel processors was made as an alternative attractive approach to speedup.

A multigroup general purpose Monte Carlo code GMVP has been developed. The vectorization algorithm is based on a stack–driven zone selection method. GMVP can treat repeated rectangular and hexagonal lattices together with combinatorial geometry. The performance of the code was evaluated by solving various types of problems. The code was installed on other four different supercomputers to investigate portability and computer dependence of code performance.[11)]

The vectorization method was studied to achieve a high efficiency for the precise physics model used in the continuous energy Monte Carlo method. The collision analysis task was reconstructed on the basis of the event based algorithm, and the stack-driven zone-selection method was applied to the vectorization of random walk simulation. These methods were installed into the vectorized continuous energy MVP code. A computation speed of this code is faster by a factor of 8~22 on the FACOM VP-2600 vector supercomputer compared with the conventional scalar codes.[12)]

A method was proposed to treat a hexagonal lattice for a Monte Carlo calculation. The present method has a flexibility in geometry description and is particularly suitable for a vectorized code using the combinatorial geometry. The calculational algorithm is simple and reduces the number of conditional IF statements in a program. This method has been implemented in a vectorized Monte Carlo codes GMVP and MVP. The speedup by the vectorization is about a factor of 24 on the FACOM VP-2600 computer.[13)]

The applicability of Monte Carlo codes to a parallel computer was studied. Two Monte Carlo codes - a radiation shielding analysis code MCACE and a criticality analysis code KENO-IV - were examined and modified in order to execute on an highly parallel computer, AP-1000. The MCACE code has achieved a speedup of 52 times on the AP-1000 equipped with 64 cell processors. In the case of the KENO-IV code, a speedup of 13 times has been achieved by using 32 cell processors.[14),15)]

Three-dimensional (3-D) neutron transport benchmark problems proposed from Osaka University to NEACRP in 1988 have been calculated by many participants and the results have been summarized. The results of k_{eff}, control rod worth, and region-averaged group fluxes for proposed four core models calculated by various 3-D transport codes have been compared. The solutions of the four core models are quite useful as benchmarks for checking the validity of 3-D neutron transport codes.[16)]

In development of AI-based design support systems, it is desirable to choose a comprehensive framework based on the scientific theory of design. A framework for AI-based design support systems for nuclear reactor design was proposed based on an explorative abduction model of design. The fundamental architectures of this framework were described on knowledge representation, context management and design planning.[17)]

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3. FAST REACTOR PHYSICS

An improved coarse-mesh 3-D (hexagonal-Z) discrete ordinates transport calculation method has been developed in order to eliminate calculational error due to coarse meshes. This method employs an weighted diamond difference approximation. As the result of test calculation compared with Monte-Carlo method, it was found that the errors of k_{eff} and power distribution are remarkably reduced by the present method.[1]

Difference between core performance parameters of small fast reactor cores calculated by JENDL-2 and JENDL-3 has been investigated based on sensitivity analysis. In order to compare the k_{eff} difference between small fast reactors and large fast reactors, fast reactor cores with core volume of 12 to 4000 l were treated. For small Pu cores the k_{eff}s calculated by JENDL-3 were smaller than those by JENDL-2 by about 0.8%. On the contrary, the k_{eff} s of large Pu cores calculated by JENDL-3 were larger than those by JENDL-2 by about 0.5%. For U cores the k_{eff} difference was independent of core volumes.[2]

Utilizing C/Es obtained by the JUPITER experiment analyses, prediction accuracies by the neutronic calculation method were evaluated for a two-region homogeneous core design of 1000MW-size LMFBR and were compared with the target accuracies which were evaluated from the standpoint of reactor physics R&Ds, i.e., expectations of attainments by foreseeable future developments of calculation method. In the case of criticality, as an example of one of the evaluated results, the calculation method has a systematic error of -0.7%Δk. The random errors of $\pm$0.9 and $\pm$1.0%Δk of the method for zero and rated powers are a little larger than the target accuracies of $\pm$0.5 and $\pm$0.7%Δk. The large random errors come primarily from lack of critical experiments with Pu higher isotopes.[3]

The cross-section adjustment work based on the JFS-3-J2 set and utilizing the results of the JUPITER program has been completed the first stage. The C/E values after the adjustment reach unity very closely, and the radial dependencies of C/E values for reaction rate and control rod worth were significantly reduced. The adjusted library was applied to a demonstration FBR core design of 600MWe and the prediction accuracies were compared with those of the E/C bias factor method. The results of prediction accuracy for the k_{eff} and other core characteristics such as reaction rate ratio and control rod worth show the superiority of the adjustment method. The burnup-related parameters like burnup reactivity loss and breeding ratio are impossible to be corrected by the bias factor method, while the adjustment method improve the prediction accuracy by a factor of 2 from the original library.[4]

The uncertainties in predicting the burnup properties of 1000MWe and 300MWe fast breeder reactors have been evaluated based on the cross-section adjustment method. The 70 group cross-section set JFS3-J2 was adjusted using integral data for the fast critical assemblies FCA and ZPPR. The burnup reactivity loss was increased by 16.8% for a one-year operation of a 1000MWe reactor, and by 4.3% for a half-year operation of a 300MWe reactor by cross-section adjustment. The excess reactivities necessary at the beginning of an equilibrium cycle were reduced from 6.4% and 6.3%, on the 2 sigma confidence level, for the 1000MWe and 300MWe reactors, respectively, to 4.7% for both reactors after adjustment.[5)]

The general expression of noise coherence function including the energy dependence of neutron detection cross section and the higher harmonic contributions in multi-dimensional model is developed on the basis of the modal expansion technique. Applicability of the method is demonstrated by numerical calculations carried out for a two-dimensional model of large fast reactor assemblies ZPPR-9 and -13C. The agreement between the theory and the measurement is satisfactory, which indicates the validity of the theory and the calculational model employed. In the assemblies, the coherence for a detector pair in a specific location can be approximately described by including the fundamental and the first harmonic modes.[6)]

Critical experiments on metal fueled FBR cores which had been continued at FCA since October 1989 have been completed. Physics parameters were measured on a ~150MWe-size clean benchmark core, composed of plutonium and enriched uranium mixture, and also on a zone-type core, having a plutonium loaded test region. The standard calculational method which was employed for the JUPITER analyses and is based on JENDL-2 library predicted well the experimental results except the following two items. By original JENDL-2 library the C/E of ^{238}U Doppler effect was 0.83, however it was improved to 0.93 by taking into account the resonance parameters above 50 kev which is not contained in the original library.[7)] C/E value for sodium void worth was 1.3 and the sample reactivity worth of ^{10}B was overestimated by about 10%.[8)]

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4. Thermal Reactor Physics

Theoritical work

The effect of a reactivity insertion in a point reactor is formulated[1)] as a linear feedback process, and a simple exponential mode representation is proposed as an analytical solution that is valid at small items. With a ramp reactivity insertion as an example, it is shown that this analytical solution is valid for a time interval that is much longer than the time-step size used in the conventional numerical integration, and it is thus useful in reducing computing time.

Critical experiment and analyses

The effect on reactivity of changes in the coolant levels in the pressure tubes of a pressure tube type heavy water reactor is experimentally studied[2)] to clarify the effect of an axial coolant void fraction distribution. The coolant void fraction distribution is simulated by stepwise changes in the coolant levels in the DCA. The reactivity and axial distribution of the thermal neutron flux are measured for a 25.0-cm-pitch square-lattice core with a positive coolant void reactivity. By using a simplified model to take note of typical reactor physics parameters, it is clarified that this anomalous phenomenon is caused by the combined effect of the flattened S curve change in the thermal neutron absorption and the even flatter S curve change in the neutron leakage caused by the changes in the coolant levels.

The accuracy of the nuclear design code system for the High-Temperature Engineering Test Reactor (HTTR) is evaluated[3)] for the neutronic characteristics that depend on core temperature by analyzing the overall temperature coefficients of reactivity and the effective multiplication factors obtained by an experiment in which the Very High Temperature Reactor Critical Assembly (VHTRC) is heated from ambient temperature to 200°C. The calculated coefficients agree well with the measured coefficients, and the calculated effective multiplication factors for different temperatures agree with measured factors within an uncertainty of 0.6%.

The validity to estimate the subcriticality of a test region in a coupled reactor system using only measurable quantities on the basis of Avery's coupled reactor theory was numerically examined[4)]. A coupled reactor experiments of two region system performed at the Tank-type Critical Assembly (TCA) in JAERI was analyzed. With the coupling coefficients obtained by the numerical calculation, the multiplication factor of the test region was evaluated by two formulas; one for the evaluation using only the measurable quantities and the other for the accurate evaluation. It was found that the estimation using only the measurable quantities is valid only for the coupled reactor system where the subcriticality of the test region was very small within a few dollars in reactivity, and that it is not applicable to a general coupled reactor system.

Neutron flux distributions in the Kyoto University Critical assembly were measured with a position-sensitive ^{3}He proportional counter[5]. The counter had a sensitive length of 118 cm and an outer diameter of 2.5 cm. The position resolution was 2.7 cm in the experimental conduction. The measurable neutron flux (at peak position) was limited to the order of $10^4 n \cdot cm^{-2} \cdot s^{-1}$ due to the pileup of signal pulses.

An efficient calculation method of an axial diffusion coefficient has been developed[6]. This is quickly calculated by using a Good Lattice Point Method. Study on the geometry dependence shows that the axial diffusion coefficient of a rectangular cross section of low density channel is larger than that of a square one, under the constraint that the channel volume is conserved. This method was verified throgh an analysis of reactivity worth measurement of a square void channel installed in the TCA.

BWR technologies

A prototype knowledge-based system that assists plant site engineers in power maneuver planning for boiling water reactor (BWR) startup and load-following has been developed[7]. The new system makes an initial plan and modifies it based on both expertise stored in a knowledge base and simplified programs using a one-point BWR core model. The time required for this system is <30 min. The quality of the plans generated by the system is almost the same as that directly generated by experts.

The core characteristics during load following operations are investigated using a reactor core simulator[8]. It has been shown that the core flow change required to compensate the Xe reactivity change produces much greater change of the void reactivity than that required for power level changes, and that the resulting local power change in the lower part of the core is greater than that in the upper part, because the Xe concentration in the lower part is hardly compensated by the core flow control.

An on-line algorithm has been developed for predicting the critical control rod pattern, reduce the mental strain on operators while withdrawing control rods in the BWR plant startup operation[9]. The proposal algorithm estimates a target eigen-value (eigenvalue bias) for a three-dimensional neutron kinetics model with a neutron source incorporating actual neutron detector readings. The critical control rod pattern is then predicted based on the estimated eigenvalue bias. The algorithm has been verified through an actual startup operation on a BWR model-5 plant.

A non-linear programming method has been developed for optimizing the axial enrichment and gadolinia distributions for the reload BWR fuel under control rod programming[10]. The problem was to minimize the enrichment requirement subject to the criticality and axial power peaking constraints. A rapid and practically accurate core neutronics model was developed to describe the batch-averaged burnup behavior of the reload fuel. A core burnup simulation algorithm, employing a burnup-power-void iteration, was also developed to calculate the rigorous equilibrium cycle performance. Optimization of the 24-region fuel, for demonstrative purpose, has shown a potential improvement in BWR fuel cycle economics, which will guide future advancement in BWR fuel designs.

Research reactors

Calculational studies are conducted of neutron and gamma-ray transport in the beam tubes of a proposed high-flux reactor for the Advanced Neutron Source[11]. The calculations were carried out by coupling two, two-dimensional discrete calculations. Calculated results show that the spectral characteristics of the particle fluxes at the exits of the beam tubes are similar to those at the entrances.

The conversion of the JMTR core from MEU fuel to LEU fuel is scheduled to be made in November 1993[12]. The safety review was finished and the installation license on the use of the silicide fuel was issued on February 28, 1992. The LEU fuel for the JMTR is silicide (U_3Si_2) fuel with 4.8 gU/cm^3, and burnable absorbers of cadmium wire are placed in each side plate. The use of the silicide fuel allows the JMTR to operate 26 consecutive days without refueling which is presently carried out after 12 day operation. Operating characteristics of the JMTR remain unchanged.

To study cold neutron intensity from liquid hydrogen moderator and from solid methane moderator, a time-of-flight experiment was conducted[13] at the electron linac at Hokkaido Univ. The cold neutron intensity from the 15 cm thick hydrogen moderator is about 1.6 times as high as that from the 5 cm one, in the case of the graphite reflected moderator assembly with a Cd decoupler. However, even in the case of the reflected assembly, the cold neutron intensities from the liquid hydrogen moderators are much less than that from the solid methane moderator; intensity ratios of the cold neutrons emitted from the hydrogen moderator to the solid methane one are about 36% for the this hydrogen moderator and about 56% for the thick one.

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5. ADVANCED CORE DESIGN

A prototype knowledge-based system that assists plant site engineers in power maneuver planning for boiling water reactor (BWR) startup and load-following has been developed.[1] In the conventional method, engineers formulate these plans based on their own expertise and on core simulation programs that require long running times. To more quickly provide a suitable plan, the new system makes an initial plan and modifies it based on both expertise stored in a knowledge base and simplified programs using a one-point BWR core model. The time required for this system is < 30min. The system was tested for 1100-MW (electric) BWRs. The quality of the plans generated by the system is almost the same as that of plans directly generated by experts. The test results show that the system is useful in preparing a suitable plan efficiently and quickly.

A high-conversion boiling water reactor (BWR) core design was proposed that conserved natural uranium through a high conversion ratio that was achieved through efficient utilization of the vapor void in the BWR core.[2] The proposed reactor concept employs fuel bundles with a square channel box and cruciform control rods, which are commonly used in conventional BWRs. Thus, it is possible to use current BWR core internals and vessel designs with minimal modifications, which makes the entire reactor system design more feasible.

A concept of the direct cycle light water reactor operating at supercritical pressure was studied.[3] This reactor does not need the recirculation lines, steam separators and dryers. The number of coolant loops of the 1,145MWe reactor is only two. The turbines will be small compared with LWR's by adopting the supercritical water as the coolant. These features greatly simplify the reactor plant system. The thermal efficiency is also improved 19% relatively from PWR's. It should be pointed out that much experience of the technology has been accumulated in the fossil-fired power plants operating at supercritical pressure more than 30yr.

A neutronic feasibility of the steam-cooled fast breeder reactor (SCFBR) was assessed by adopting the supercritical steam as a coolant.[4] The reactivity of the core changes negative against both voiding and flooding. Use of the fixed moderator, zirconium hydride, is effective for decreasing the positive void reactivity. Both the direct and indirect cycle reactor will be feasible. The indirect cycle system was designed. Thermal efficiency is 9% higher than current PWRs. The LOF and LOCA were analyzed. The coastdown time of the blowers should be larger than 30s. The blowdown after the large-break LOCA showed that the core should be cooled in 10s. The main advantage of SCFBR is the potential of realizing low-cost FBRs based on the existing experience with steam coolant technology. The technical disadvantages of SCFBR are the low breeding ratio and the high coolant pressure leading to thick walls of the pressure vessel.

A new core concept with a negative sodium void reactivity coefficient has evolved.[5] The core is composed of two core layers in the axial direction. The core layers are separated by an internal blanket, the central region of which comprises a neutron-absorbing material such as boron carbide or tantalum. Consequently, the two core layers are completely decoupled are regards neutronics, leading to an effective increase in neutron leakage from the core region when sodium is voided. This design is expected to be free from the disadvantages of a large core radius, as seen in a conventional spoiled core such as a pancake core. The design is described in detail, and its application to a 300-MW (electric) metal fuel core and to a 450-MW (electric) minor actinide burner core is given as an example.

Number of design studies have been carried out focusing on the reduction of sodium void reactivity which is becoming one of the most important issues in fast breeder reactor development. The following new core concepts have been introduced.

A design study was carried out to evaluate the performance of oxide fueled axially heterogeneous cores (AHCs), focusing on the reduction of sodium void reactivity to further enhance the inherent safety characteristics of a 1000MWe liquid metal fast breeder reactor.[6] It was found that the effects of the above-core axial sodium gap on the sodium void reactivity reduction were enhanced a great deal when combined with the AHC concept. The proposed AHC was characterized by the displacement of the upper axial blanket by the axial sodium gap ; the height of the sodium gap above the inner core having the internal blanket was slightly larger than that of the outer core sodium gap ; and the internal blanket was located just below the outer core midplane. It was shown that the proposed AHC gave a reasonably small sodium void reactivity without imposing significant penalties on the burnup reactivity swing and breeding, leading to economic advantages, especially through an extended refueling interval and higher burnup.

A design study has been made on high burnup and large sized FBR cores which mitigate the consequences of unprotected accidents.[7] The reactivity feedbacks due to the core radial expansion, the sodium density change and the Doppler effect have significant influence on the maximum coolant temperature in the unprotected loss of flow event. The excess reactivity is also an important parameter for the safety in the unprotected transient over-power event. As the results of sensitivity analysis, a pancake-shaped and low power density core concept is proposed which offers effectively enhanced safety.

A new concept of an axially double-layered core has been studied for reduction of sodium void worth to zero.[8] In this concept, the core is divided into two axial regions by placement of an internal blanket region and a thick neutron absorber region in order to decouple the upper and lower cores neutronically. It has been shown that the axially double-layered core concept has potential to achieve both passive shutdown capability and zero sodium worth while keeping a breeding ratio greater than 1.0 and a relatively small diameter.

A reduction method for positive sodium void reactivity which is one of the most important safety issues in fast breeder reactor development has been studied.[9] The DU-Pu-Zr metallic assembly with graphite moderator where the ratio of moderator to fuel is about 3 show better nuclear characteristics such as breeding ratio, burnup, void and Doppler reactivities than those obtained with the use of oxide, carbide or nitride fuels. A concept of high safety and breeding fast reactor core in which reactor power is instantaneously reduced by the void and Doppler reactivities for transient accidents is proposed.

A compact and robust reactor system using natural circulation designated by "transportable reactor" is being studied.[10] This reactor is controlled by inserting or pulling out a fuel assembly, driven by a core driving mechanism, into or from a core region that is surrounded by a radial reflector. Arrays of heat pipes in the upper plenum are cooled by liquid metal coolant which then flows downwards between the reactor vessel wall and the reflector ring. The primary heat pipies transfer the heat to the secondary heat pipes. Thermoelectric cells between both heat pipe arrays convert the heat to electricity (-10MW). A ratio of height and diameter to the cylindrical fuel region was determind 1:1 to minimize the critical core volume. The core diameter was set to be 40cm as an approriate size from considering the easiness of extrapolation or interpolation of the core size.

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6. Fusion Reactor Neutronics

A variety of investigations have been carried out to meet the strong requirement from the fusion reactor development, in terms of blanket neutronics, radiation shielding, induced radioactivity, nuclear heating, and also D-D plasma diagnostics. In the framework of JAERI/USDOE collaboration on the fusion blanket neutronics, the last experiment has been carried out, namely Phase-IIIC. The experimental configuration was based on the former Phase-IIIB, making large opening. The results demonstrated that the opening decreases appreciably the tritium production rate in the region of opposite side of the opening.

The extensive effort was devoted to the experiments of bulk-shielding with large SS316 assemblies[1] using FNS facility under the aid of ITER/EDA R&D program. The first experimental results along with their analysis were presented at the ANS 10th Topical Meeting on Technology of Fusion Energy.[2,3] Meanwhile, the activities related to the fusion shielding experiments conducted at FNS facility was reviewed to give a present status of art in this particular field[4] and the paper was given in the ANS Topical Meeting on New Horizons in Radiation Protection and Shielding, held at Pasco in USA. The continuous energy neutron and photon cross section library based on JENDL-3 for the MCNP calculation has been made.[5,6] The outline of the library, FSXLIB-J3, was also presented in the same Meeting. Shielding characteristics concerning heterogeneous effects as well as gap streaming were investigated in relation to ITER/EDA.[7,8] They tried to find out more general information for optimization of shielding configuration by applying a parametric search.

Current issues relevant to induced radioactivity in the fusion reactor components requested a versatile data base of activation cross sections. A useful report on the graphical presentation for data associated with induced activities appeared as the second version of it for THIDA-2 code system.[9] The activation cross section data treated were updated by using a comprehensive evaluation. An experimental effort was reported on the integral examination of long-lived radioactivities induced in the D-T neutrons at the 10th Topical Meeting on Technology of Fusion Energy.[10] This experiments was conducted under the JAERI/USDOE collaborative program on fusion neutronics. It demonstrated that there is considerably large discrepancies in the activities between experiment and calculations with various cross section data libraries, THIDA-2, REAC*2, DKR-ICF, RACC and so on. A need to continue the experimental endeavor was clearly shown.

Along with the induced radioactivity experiment, a nuclear heat deposition measurement has been performed under the JAERI/USDOE collaboration program. The experimental analysis for nuclear heat on ten different potential materials was presented at the same 10th Topical

Meeting.[11] The uncertainty rage for the KERMA factors of neutron and γ-ray could be assured to be within ± 50 % as long as the D-T neutron dominant field was concerned.

A code was developed at Kyushu University to calculate the neutron energy deposition in compressed D-T spheres for the inertial confinement fusion.[12] The transports of fusion neutrons and their recoils are simultaneously solved based on the Boltzmann-Fokker-Plank equation. It is found that calculations neglecting the recoil-ion transport overestimate the deposition to plasma ions.

The D-D plasma discharge experiments started in JT-60U from July in 1991. The D-D neutron measurement was conducted during the experiments by means of several techniques.[13] The foil activation measurements gave the total D-D neutron yields as well as flux distribution in the torus hall of JT-60U.[13.1-13.3] The dose rate and induced activities after the discharge were measured with TLD and γ-ray spectormeter.[13.4,13.5] It revealed that the neutron field was dominated by low energy neutrons. A ^{3}He proportional counter was applied to measure the D-D neutron yield and spectrum.[13.6] It was pointed out that the further effort is needed to establish this technique. A new technique for D-T neutron energy spectrum measurement based on counter telescope was proposed by the group of Nagoya university to measure incorporating with a scintillator as the radiator.[14] The development is still underway.

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7. NUCLEAR CRITICALITY SAFETY

There was an international conference on nuclear criticality safety, ICNC'91, at Oxford in U. K. from September 9 through 13, and 18 papers were presented from Japan. The followings were their abstracts inclusive of some papers appeared recently in other proceedings and a journal.

A series of critical experiments on a single core and two coupled cores with slab-type concrete geometry were made at the Tank-type Critical Assembly (TCA) of the Japan Atomic Energy Research Institute (JAERI) to make clear the reactivity effect of concrete used in the design of nuclear fuel storage and handling facilities[1]. Each unit of the core is composed of 2.6 w/o UO_2 fuel rods moderated with light water. Both the reflector effect in the single core and the neutron interaction effect in two cores were determined by the water level worth method.

Critical experiments on a neutron trap cores composed of two rectangular cores separated by a water gap and borated stainless steel (B-SUS) plates were performed at TCA[2]. The thickness, boron content and position of B-SUS plates in the water region between two cores were changed parametrically, and reactivity effects were determined from the change in measured critical water levels.

Aiming at improving the accuracy of criticality safety analysis, the following studies were accomplished at the Japan Institute of Nuclear Safety (JINS)[3]. The higher order Legendre expansion term (up to P_3) was added to the MGCL cross-section library; The collision probability method was applied to evaluate the resonance self-shielding effect in the systems which contain both solid and liquid nuclear fuels. The improvement of analytical accuracy brought about by the P_3 scattering term was clearly shown for the systems composed of mutually interacting array. Applicability of the collision probability method was examined with the benchmark critical test problem proposed by NEACRP in 1986.

A new MGCL (Multi-Group Constants Library) with P_3-scattering data, named MGCL-J3, was produced from the JENDL-3 library, the newest evaluated nuclear data library at JAERI. MULTI-KENO was updated and named MULTI-KENO-Ⅱ to treat triangular array system, and to calculate neutron scattering by hydrogen nuclide exactly. For a series of UO_2F_2 solution critical experiment, calculated k_{eff}'s by MULTI-KENO-Ⅱ with MGCL-J3 revealed to be closer to the measured value 1.0 than those by KENO-Ⅳ with MGCL-B4[4].

Kadotani and Hariyama proposed[5] to use eigen-vectors from the matrix K calculation of the KENO code to accelerate the convergence of the fission distribution. The proposed method was successfully applied for criticality safety problems of complicated lattices composed several different cells.

To theoretically estimate the reactivity effect due to a spatial variation of nuclear fuel concentration, there are the Goertzel's necessary condition, and the "fuel importance" theory proposed by M. M. Shapiro and M. Otsuka. In order to verify these theories, Yamane et al. performed[6] systematic measurements of reactivity effect due to the nonuniformity in the fuel distribution at the Kyoto University Critical Assembly (KUCA). The neutron flux distribution and fuel importance distribution were also determined. A nonuniform assembly whose fuel concentration in the center region was 40% higher than the uniform one was found to have an excess reactivity of 0.3% Δ k/k, with the same total uranium mass for which the uniform assembly was just critical. Moreover, the spatial distributions of thermal neutron flux and of fuel importance were flatter than the uniform assembly, as expected by the Goertzel's condition and the fuel importance theory.

A numerical algorithm to determine the optimal fuel distribution for minimum critical mass, or maximum k-effective, was developed by Hirano et al.[7] using the Maximum Principle in order to evaluate the effect of non-uniformly distributed fuel on reactivity. This algorithm maximized the Hamiltonian directly using an iterative method under a certain constraint, such as maintenance of criticality or total fuel mass. It ultimately reached the same optimal flattened fuel-importance distribution as Dam's algorithm, which was based on perturbation theory. This method was applied to plutonium nitrate solution systems.

Effect of matrix solvent on k_{eff} was examined[8] using the Monte Carlo code MULTI-KENO for systems of plutonium solution of the aqueous nitric acid and the organic TBP-dodecane. A comparison of k_{eff} values was made for a mixer settler model having a constant Pu inventory, but varying profiles of liquid-liquid interfacial level or concentration of Pu in the bank. An effect of formation of the $Pu(NO_3)_4$-TBP third phase between two phases on the k_{eff} of the system was examined. A specific non- uniform profile of Pu in which Pu accumulated in a central part of the system showed a higher reactivity of the system.

From spatial- and time-decay constants measured by the exponential and the pulsed source experiments, the static reactivities or the effective neutron multiplication factors were determined for subcritical light-water -moderated and -reflected cores of various sizes[9]. Some C/E comparisons on those quantities revealed that the exponential experiment was advantageous compared to the pulsed source experiment on the point that the difference in neutron flux between the experimental and the calculational was much smaller even in a highly subcritical system affected by the reflector.

A compact formula was developed[10] which converts a measured noise quantity, namely the spectral ratio of three-detector noise outputs, to subcriticality. It was a generalization of the one-point kinetics conversion formula which Mihalczo originally used in his Cf-driven neutron noise method. The formulation included modal effect of measurements, and reduced the spurious detector-position dependence of the resulting reactivity. The frequency-dependent detector importance function, which H. V. Dam used in BWR noise analysis, was employed. The function gave the field of view of a detector, taking account of time-delay and spatial attenuation that took place between the neutron noise source and its detection. The proposed method was more compact and adequate for systems with deep subcriticality than the ω_p modal expansion method. The measured ratios were analyzed and compared with the Monte Carlo results.

An active neutron multiplication method was studied[11] in laboratory experiments aiming at establishing and improving nondestructive measurement methods for a spent light water reactor fuel bundle in water. The fuel rods used in the experiments consisted of 1.0-cm-diam UO_2 pellets enriched to 1 to 3 wt% ^{235}U, clad in 1.18-cm-o.d. aluminum tubes. The rods were arranged in square arrays spaced to form a 1.52-cm lattice. The analysis was carried out by two- or three-dimensional, three-energy-group diffusion calculation. A simple empirical expression for the correlation between the neutron flux and the effective neutron multiplication factor k_{eff} was proposed in which a constant term was added to the well-known formula for one-point subcritical flux. The new expression was found to be applicable to high-precision measurements. Through an experimental study of the correlation between the neutron flux profile and the neutron source response to a detector and studies of the new correlation expression, an improved measurement system was presented that was suitable for more precise measurements of k_{eff}.

For the system with spent fuels, the criticality analysis needs to focus on the both top and bottom parts of fuel rods since these parts are less irradiated than the remaining part due to axial power shape during operation. The effect of burnup on criticality was analyzed[12] by KENO Monte Carlo code. The neutron multiplication factor of a spent fuel cell was analyzed to evaluate the reactivity change when the burnup distribution was taken into consideration.

Fuel assemblies have transverse and also axial non-uniform burnup distributions. Therefore, Itahara and Shimada[13] evaluated the criticality safety effect of actual burnup distribution for PWR fuel assembly on a design based on uniform burnup distribution. The evaluation was carried out for a spent fuel transport cask, a spent fuel storage pool and the continuous dissolver in Rokkasyo Reprocessing Plant.

Japan Nuclear Fuel Service Co. Ltd. and Toshiba Corp. developed a burnup monitor for a commercial reprocessing plant to verify the burnup of LWR fuel assemblies for criticality control[14]. The measurement method used by the monitor, which will be installed in the spent fuel storage facility at the Rokkasyo Reprocessing Plant, consisted of gamma-ray spectrometry, passive neutron counting, gross gamma-ray profile and passive neutron self-interrogation. A proving test on the method was carried out for fifty three irradiated LWR fuel assemblies.

Monte Carlo code KENO-IV was applied to the calculation of the kinetic parameter β_{eff}/ℓ [15]. In the calculation, the β_{eff} value was derived from using the averaged β value over those of fissioning nuclides in a fuel system and the fraction of k_{eff} due to delayed neutrons. For the ℓ value, the generation time calculated by the KENO-IV code was used. The calculated results were compared with measured data for UO_2 and PuO_2-UO_2 lattices, and UF_4-Paraffine compound assemblies. The comparisons showed that the calculated results were within ± 10 % of the measured data.

Three different void-models used in criticality accident analysis of aqueous fissile solutions were compared and the effect of initial neutron density on the criticality accidents was investigated[16]. Different effects were found in calculated peak power due to differences in void-models for large inverse stable periods. In addition, evaluation of the initial neutron production density in slightly enriched uranyl nitrate solutions showed that initiation of the first persistent fission chain would be delayed with fast reactivity insertions. These effects, however, were not significant from a practical standpoint of view, since fast reactivity insertions were unlikely to occur in practice.

JAERI is revising and supplementing the first criticality safety handbook of Japan published by the Science and Technology Agency (STA) in 1988, reflecting the opinions of the Working Group chaired by Prof. K. Nishina[17]. The revised version will cover the reactivity effects more precisely and incorporate chemical process data and accident evaluation data.

JAERI is constructing a Nuclear Fuel Cycle Safety Engineering Research Facility[18], NUCEF, where the following research themes will be studied which are essential for evaluating safety problems relating to back-end technology in nuclear fuel cycle facilities:

a. nuclear criticality safety research,

b. research on advanced reprocessing process and partitioning, and

c. research on transuranic waste treatment and disposal.

Two criticality experiment facilities, STACY and TRACY, to perform nuclear criticality safety research related to the reprocessing of light water reactor spent fuels, are under construction. STACY, Static Criticality Facility, will be used for the study of criticality conditions of solution fuels, uranium, plutonium and their mixtures. TRACY, Transient Criticality Facility, will be used to investigate criticality accident phenomena with uranium solutions.

A nuclear fuel treatment system was installed to supply needed solution fuel to STACY and TRACY. The system consists of several processes such as dissolution, concentration, purification etc. in reprocessing plants. From the viewpoint of nuclear criticality safety, such special attention, as criticality safety control with single and multiple units, should be taken into account. Besides, interlock systems are applied to some important equipments in order to prevent criticality accidents[19].

Investigation of criticality safety was performed[20] for cylindrical annular tanks of NUCEF. Basic criticality safety limits were identified for NUCEF annular thickness and inner diameter of fuel solution region. In addition, criticality safety characteristics of size effect for other parameters were also studied in order to make useful for fabrication of annular tanks of NUCEF.

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8. SHIELDING

The Research Committee on Radiation Behavior of the Atomic Energy Society of Japan has completed its work to make up shielding benchmark problems to be utilized for accuracy estimation of shielding calculation codes for high energy accelerators and also has finished preparing manuscripts of a handbook of neutron shielding design. The handbook which is to be published in near future from the Atomic Energy Society of Japan, includes the latest status of neutron shielding designs and the description of neutron shielding data and the way of its shielding designs.

The Topical Meeting on New Horizons in Radiation Protection and Shielding was held at Pasco, Washington on April 26 to May 1, 1992. The fields of radiation protection, health physics and shielding are intertwined with every aspects of the nuclear industry. Thus, the meeting covered sessions devoted to accelerators, aircraft, fusion and fission reactors, medical facilities and space. There are a lot of papers presented from Japan, the largest in number next to USA. About neutron shielding materials, the KRAFTON and Mannan were studied for their shielding ability[1)] and neutron shielding materials for transport packaging were developed and their shielding test was made at YAYOI.[2)] As for fast reactor shielding problems, neutron streaming experiment and its analysis were performed as a part of JASPER program,[3)] and a shielding design study of an In-Vessel Storage system using B_4C as a removable radial shield was performed.[4)] As a topics about shielding code development, a Monte Carlo code MORSE-AZA was developed, which can treat azimuthally dependent albedo event.[5)] About JENDL-3 library, two papers were presented: one is a development of FSXLIB-J3, a MCNP continuous energy cross section library,[6)] and another is the integral test of cross sections in JENDL-3 based on Monte Carlo analyses.[7)] About the works in connection with the simple methods for a shielding analysis, a simple analytical expression was developed for the quick estimation of streaming radiation fluxes in ducts,[8)] and the behavior of gamma-ray buildup factors in two stratified shields were investigated.[9)] In the field of accelerator shielding, a study was made of neutron streaming problem through a labyrinth from a 40 MeV proton cycrotron room[10, 11)] and the systematics of the inclusive neutron yield by light and heavy ions was studied to derive a simple expression for the systematics of equilibrium neutron yield.[12)] As part of the evaluation of the radiation shielding safety in the fuel cycle, development was made of an adjustment method to calculate the neutron energy spectra due to (α,n) reactions from light elements with actinide α-particle emitters.[13)] In connection with the problem of dosimetry, a real time personal dosimenter was developed by using two types of silicon p-n junction detectors,[14] and extensive shielding data and methods for evaluating 1-cm dose equivalent was developed for photons, β-rays and neutrons.[15] As for fusion nucleonics and shielding, shielding experiments for fusion reactors ever performed at the FNS/JAERI were reviewed,[16] and a series of fusion neutron streaming benchmark experiments and their analyses performed were summarized.[17]

In the field of radiation transport calculations, a Monte Carlo code MORSE was revised by introducing a bremsstrahlung calculation function, in which multiple scattering of electrons are taken into account.[18] Preparing input data for DOT3.5 is a tedious work. Thus, in order to avoid this difficulty, a DOG-II, an input generator program for DOT3.5, was developed.[19] In the field of shielding, recently emphasis is shifting to higher energy regions above 20 MeV where scarce cross section data are available. Transmission of intermediate-energy neutrons and gamma rays through various shield materials was studied[20]. Further, buildup factors for medium energy neutrons up to 400 MeV were estimated.[21] As for the practical shielding design study, a preliminary radiation protection design of Large Helical Device(LHD) was performed and the overview was summarized.[22]

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9. INCINERATION OF RADIOACTIVE NUCLEAR WASTES

A feasibility of fast fission system confining long-lived nuclides without other supporting system as synergetics for fuel sustainment and waste incineration was studied from the aspects of nuclear material balance and neutron economy.[1) The continuous utilization of fast fission system which confines all actinides in the reactor but discharges all FP will lead to huge accumulation of radioactive wastes such as ^{129}I, ^{135}Cs, ^{107}Pd, ^{93}Zr, ^{99}Tc, ^{126}Sn and ^{79}Se in the far future. Then the feasibility of the system that these long-lived seven FP are also confined in the reactor with actinides was studied. In this scheme, all the long-lived nuclides to be disposed of were exposed with neutrons in the reactor and removed as different nuclides after nuclear transmutation. As the wastes stored in the repository was composed of only shorter-lived nuclides, total amount of radioactive wastes in the repository was suppressed to be less than a few tons per 3GWt reactor.

Number of design studies for super long-life core have been performed by utilizing the phoenix properties of TRU.[2),3),4)]

A feasibility of a 1000MWe super long-life LMFBR core (SLLC) loading TRU was studied which can operate without fuel exchange during plant life.[2)] The reactivity change of the TRU-loaded SLLC during 30 years (EFPY) is about 5% and it is less than half that of the SLLC with no TRU loaded. The values of the maximum linear heat rate for both of the SLLC are about 270w/cm. They would increase to some extent if control rods are inserted in order to suppress the excess reactivity. The average fuel burnup is about 19GWd/t and the maximum fast neutron fluence is about 7×10^{23}nvt. This irradiation condition is almost the same as the target of a commercial level FBR core. The maximum cladding temperature was evaluated on condition that the reactor outlet and inlet temperatures are 530 and 375 ℃, respectively. Although it is preliminary since no control rod insertion effects is considered in the calculation of the power distribution, the maximum cladding temperature is low enough compared with its limit value (typically, 700℃). Total loading mass of Np, Am, and Cm is year 17ton and 10ton of it (about 60% of loading mass) is transmuted during the 30year reactor operation. Transmuted mass of Np, Am, and Cm is about 340kg per year, which is nearly equal to the mass of those from 13LWRs of the same power output. It was found that the SLLC generating electricity as well as transmuting TRU is feasible from the neutronic point of view by optimum zoning of TRU loading to sustain reactivity and suppress power variation during plant life.

Conceptual design study of minor actinide burner reactors was performed to obtain a reactor model with very hard neutron spectrum and very high neutron flux in which minor actinides can be fissioned efficiently.[5] Two models of burner reactors were obtained, one with minor actinide alloy fuel core and Na-cooling, and the other with minor actinide nitride particle fuel core and He-cooling. Minor actinide transmutation in the burner reactor was compared with that in power reactors of LWR and FBR from the reactor physics and fuel cycle facilities view points. The minor actinide burnup rate per year is significantly higher than those in power reactors. For the minor actinide transmutation using power reactors, the design changes of the radiation shielding of the whole fuel cycle facilities because of the increase of strong neutron emitting nuclides.

A computer code system "ABC-SC" was developed to analyze actinide transmutation in a fast reactor. The code system consists of the collision probability calculation code SLAROM for effective cross section calculation, the diffusion code CITATION and the nuclide buildup and decay calculation code ORIGEN-2. The thermal-hydraulic calculation codes were attached to calculate the axial temperature distribution of fuel elements for actinide burner reactors, one for metal fuel with Na-cooling and the other for coated particle fuel with He-cooling.[6] The core transient behavior calculation code "EXCURS" for a Na-cooled oxide fuel reactor was modified for the application to a Na-cooled actinide metal fuel burner reactor.[7]

A transmutation of fission products was studied at PNC.[8] Four transmutation methods by accelerator were compared in terms of transmutation energy and effective half-life for ^{137}Cs. It was found from this comparison that the transmutation energy was the least in the μCF method and the largest in the electron method among the four methods. But the difficulty in constructing the transmutation devices for these methods may be the opposite, namely, the device in the μCF method may be the most difficult to be constructed. As for effective half-lives of ^{137}Cs in the four methods, all of them were calculated to be in the order of several years. A problem with transmutation by accelerator is that it is difficult to transmute a large quantity of fission products by the present accelerators because of their small beam currents. To transmute a great deal of fission products, high power accelerator is needed. As one of such accelerators. The super-powered electron linear accelerator is developed at PNC. As a first step, PNC is developing 10MeV-100mA test electron linac, especially, high power klystron and accelerating tube for high beam current. The klystron achieved its maximum output power of 330 kW. The accelerating tube with resonant ring is under construction.

Two concepts of an accelerator-driven actinide transmutation system were studied.[9] One of the concepts is based on a subcritical core of metallic minor actinide fuel and the other on a subcritical molten salt core. In the former concept, the system consists of a proton linear accelerator and a Na-cooled subcritical core of minor actinide alloy fuel which are loaded around a tungsten target. By introducing 1.5GeV and 39mA proton beam into the system, about 250kg of minor actinides are transmuted annually by fast fission and 820MW thermal powers are produced.

The expected advantage of a molten salt core concept is the continuous in-line processing of irradiated molten salt fuel.[10] In this study, the minor actinide chloride molten salt was used in stead of the fluoride molten salt since the fluoride is less soluble to actinides even though it is the most stable as the molten salt so far proposed. Also a fluoride fuel core is less attractive for fast fission of minor actinides because of soft neutron spectrum in it. The system consists of a proton liner accelerator, a subcritical molten salt core and a ex-core processing system of the molten salt fuel.

For designing an accelerator-driven transmutation system, spallation neutron production rate and transport process in a target is the fundamental data. An integral experiments were carried out to study the neutron transport process using the 50MeV booster proton synchrotron facility of KEK(National Laboratory of High Energy Physics).[11] In the experiments, the induced reaction rates were measured using the cylindrical activation samples inserted in the lead bulk target. For analyzing the measured data, the Monte Carlo spallation code NMTC/JAERI was used. The calculation generally agrees with the experiments except for the reaction rates at the deeper locations on the beam incident axis.

The benchmark study on the computational models of spallation reactions was carried out to examine the causes of discrepancies observed in the calculations of spallation products yield between the spallation cascade code NMTC/JAERI and HETC/KFA2.[12] It was resulted that the main cause of discrepancies in the difference of the width of the post-fission yield curve between the codes.

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10. NATIONAL PROGRAMS

1 JOYO

The 9th periodic inspection, two duty cycle operations and a special cycle operation have completed (or continued) in this period.

During the periodic inspection, one of 6 control rods has been relocated from the 3rd row to the 5th row as a first step of the transformation to the MK-Ⅲ core. the 24th duty cycle operation was completed and the 25th duty cycle is now under going. In these duty cycles, some special type fuel subassemblies and many material irradiation rigs have been installed in order to irradiate the LMFBR materials and fusion reactor materials. A type B special fuel "B8", which was conducted for the verification of ferritic steel cladding, has been loaded and started the irradiation test at the 25th duty cycle operation. The durations of the 24th and 25th duty cycle are 42 and 66 days, respectively.

In the 24th special cycle operation, the second power to melt experiment was performed successfully. All of 24 test fuel pins with various design parameters such as pellet-cladding gap, pellet theoretical density and O/M ratio were irradiated at 100MWt power for 10 minutes. The post irradiation examination in under way.

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 continued in preparation for the licensing.

2 MONJU

The function tests, which is the first phase of the pre-operational test program of MONJU, were started in May 1991 to confirm integrated system functions without nuclear heating. Next phase, called the start-up tests, are criticality test (fuel loading), physics test, nuclear heating test and power operation test. It is expected to get the first criticality in 1993 as the criticality test. Measurements of reaction rate distribution, control rods worth, reactivity coefficients, coolant flow rate in the core, etc. are planned in the physics tests.

3 Demonstration Fast Breeder Reactor

The conceptual design of the demonstration reactor was carried out to confirm technical feasibility of top entry loop type reactor. The general plant design, analytical evaluations and confirmation tests were completed by the end of March of 1992, with an emphasis on major sodium systems such as core, fuel, reactor structure, intermediate heat exchanger, primary pump, steam generator, decay heat removal system, reactor building and so on.

Trough technical analysis and evaluations, design concepts of major systems and components were established, and the technical feasibility in terms of structural integrity, safety, operability and maintainability was confirmed.

Confirmation tests were also carried out to investigate the phenomena important to technical feasibility and which may not be fully ascertained by technical analysis and evaluation. These tests included the natural circulation test, the gas entrainment test and sloshing test with multiple surfaces.

Based on these design study and confirmation tests, design concepts of the top entry loop type reactor plant was established and technical feasibility was confirmed.

4 FUGEN

Fugen has continued stable full power operation from September 1991 to August 1992. The scheduled shut-down for the 17th refueling was carried out on October to December 1991, and for the 18th refueling and the 10th annual inspection was conducted on April to July 1992.

At the 17th refueling, 10 mixed oxide (MOX) and 18 UO_2 fuel assemblies were charged. As a result, core configuration was 140 MOX (including expermental fuel) and 84 UO_2 (including special fuel) fuel assemblies. And on the 18th refueling, 18 MOX and 16 UO_2 fuel assemblies were charged, and then reactor core configutation was 132 MOX (including experimental fuel) and 92 UO_2 (including special fuel) fuel assemblies.

At Fugen, up to date, 354 UO_2, 371 MOX and 16 special 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 3168 effective full power days of operation up to the end of August 1992.

Thirty four UO_2 fuel assemblies were transported to PNC Tokai reprocessing plant, and one experimental segmented fuel assembly was conveyed to JAERI (Tokai village) for post irradiation examination. Two 36-pin experimental fuel assemblies for the demonstration plant were taken out from the reactor, and then six Gd_2O_3 poisoned MOX fuel assemblies and one experimental segmented fuel assembly were loaded continuously.

At the 10th annual inspection, a new control system applied Fuzzy logic which PNC has developed since 1986 was introduced to a lower flow rate feed water valve of reactor feed water control system. This system improved the controllability of the water level in the steam drum when the power of the reactor was changed during start up or shut down.

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 1995, and the commercial operation in 2001.

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 project, 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. The Committee for Technical and Economic Studies on Nuclear Energy Development and the Fuel Cycle (NDC) of NEA convened the expert committee for an international collaboration on the problem "Making Extra Gains from Actinides", where it was confirmed that studies on actinides partitioning and transmutation should not be positioned in opposition to current policies on the geological disposal of the radioactive wastes. The specialists' meeting on accelerator-based transmutation was held at Paul-Scherrer Institute, Switzerland 24-24 March 1992.

Concepts of accelerator-based transmutation systems and nuclear design problems of the systems with emphasis of target facilities and their interfaces with accelerators were discussed at the meeting.

8 Light Water Reactor

Two light water power plants (Tomari Unit 2 PWR 579 MWe and Kashiwazaki-

Kariha Unit 2 BWR 1100 MWe) were recently brought into commercial operation. Total capacity of the nuclear power plants is presently 32059 MWe. The Japanese first nuclear ship "Mutsu" equipped with 36 MWt PWR reached critical again in March 1990 and its performance tests were successfully conducted. A series of experimental voyages are now under way.