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JPRS-JST-88-013
29 JULY 1988

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JPRS Report

Science & Technology

Japan

26TH NATIONAL SYMPOSIUM
ON ATOMIC ENERGY

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SCIENCE & TECHNOLOGY

JAPAN

26TH NATIONAL SYMPOSIUM ON ATOMIC ENERGY

Tokyo DAI 26 GENSHIRYOKU SOGO SYMPOSIUM YOKOSHU in Japanese 22-23 Feb 88

CONTENTS

Future LWR Design Improvements Discussed

43062071a Tokyo DAI 26 GENSHIRYOKU SOGO SYMPOSIUM YOKOSHU in Japanese 22-23 Feb 88 pp 11, 12

[Article by Kiyoshi Sako, Japan Atomic Energy Research Center: "The Prospects for Increasing Passive Safety and Raising Conversion Ratios in Light Water Reactors"]

[Excerpt] In recent years the growth of light water reactors has been amazing and they have a good record as highly reliable power reactors. We have seen a cumulative expansion in design, manufacture, construction, operational experience, and know-how, and its superiority has been obvious. With this as our foundation we continue to see further improvements in reliability and economy and there has been an attempt to standardize at the 1350 MWe class (ABWR, APWR).

A. Reactor Concepts and the Need for Passive Safety

However, from the standpoint of flexible response to power demand there has been a growing need for power reactors in the 600 MWe class. Although this is principally true with power companies in the United States (where there are many medium and small companies), there is also a potential need for them in Japan. The spreading out of orders is also desirable to manufacturers. Costs are reduced on existing power reactors by up-scaling and standardization, and by merely down-scaling the same designs, increased construction costs, etc., cannot be avoided. The way of thinking which has accordingly emerged involves methods which simplify reactor systems, promote lower construction costs and shortened building times, and keep costs down through measures which increase the use of passive safety in the reactor itself. From the standpoint of operation and maintenance medium and small output reactors must be simplified to keep their competitive edge and this problem will be addressed by reactor system simplification. For example, ABWRs have been rationalized by placing recirculation pumps within the reactor container and such measures will be taken even further in medium and small output reactors. The separation of safety systems into those under external control and those in the reactor itself is a problem to be faced during design selection. If passive safety features for reactors can be increased, output down-scaling will naturally follow.

1) Existing Extended Line Reactors: The SBWR which GE has proposed and the Westinghouse's AP600 are famous examples. These designs have simplified the reactors themselves and have made the emergency flooding systems, etc., passive (for instance, SBWRs have natural circulation). In Japan, the three reactor companies have each proposed their own reactor concepts. For example, Hitachi's designs have given serious consideration to earthquake-proofing by these methods and they have produced reactor concepts where reactors cannot only be built on unstable ground but construction times are also shortened due to the rationalization of both the reactor itself and its housing. These types of reactor designs fully utilize past experiences and are noteworthy for having no elements needing large-scale development and for thus being quickly achievable.

2) New Reactors: The PIUS, which Sweden's ASEA Atom Co. has proposed, has epoch-making features. For instance they have done away with the control rod and emergency cooling systems by placing the primary systems within a huge concrete reactor container filled with a large amount of low temperature boric acid solution. This design strives to keep costs down by eliminating as many active systems as possible, assuring reactor safety through the use of natural phenomena (dynamic forces, etc.), and by simplifying external system controls. Based solely on evaluations of these concepts themselves, (arguments pro and con in order to estimate not only the huge size and complexity of the reactor itself, but also the difficulty of maintenance), ASEA has received high marks for having been the first to propose these kinds of reactor concepts and design ideas on a large scale. We have also proposed a new type of reactor with similar safety theories (SPWR-system reactors with a wide range of options) and design studies are continuing. The representative design proposal 9350 MWe) in Figure 1 is an example. Instead of the control rods of integrated PWRs, its reactor vessel is placed in a tank of concentrated boric acid solution, a reactor concept midway between the PIUS and existing reactors.

B. The Requirements for High Conversion and Its Concepts

With the lengthy delay in the introduction of fast breeder reactors we have come to emphasize the efficient use of uranium resources and the prevention of the depletion of plutonium produced by light water reactors. Research was begun years ago in the European Community countries of France, West Germany, and Switzerland. In Japan, universities and JAERI have worked mainly on basic research, while the electric power companies and reactor builders have promoted practical research. All have energetically pursued international cooperation. We have thus made progress not only in our understanding of reactor physics and the special characteristics of the hot water dynamics, but also in the design of fuel assemblies and reactor structures. We have clearly defined both efficiencies and possible ranges of application.

1) The Basic Theory and Characteristics of High Conversion Light Water Reactors: High conversion is achieved mainly by first decreasing the amount of water used as moderator in the fuel lattice, strengthening the neutron spectrum and then increasing resonant absorption. Figure 2 shows a comparison of neutron spectra. Its special characteristics are that the

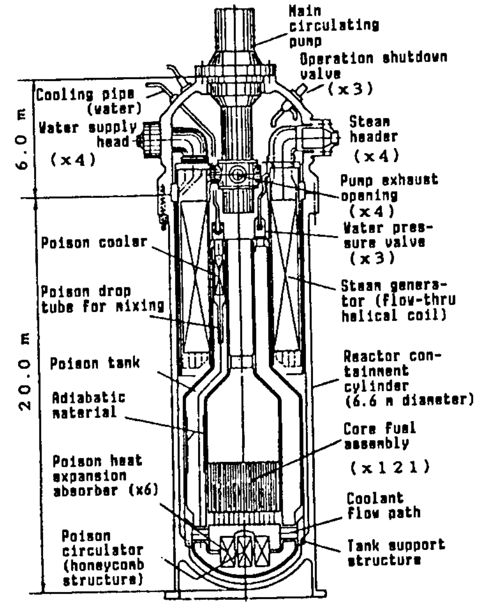

Figure 1. Representative SPWR: SPWR-H-H1100* (1100 MWt, 350 MWe)
*System integrated PWR-Hot Vessel/Hot leg 1100 MWt

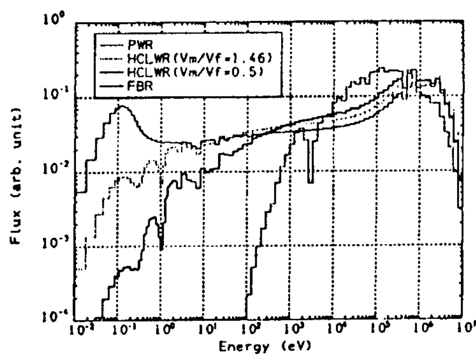

Figure 2. Comparison of Neutron Spectra (per Okumura)

amount of plutonium that it consumes is small compared to Pu thermal, and because its neutron spectrum is tight, it generates high quality plutonium.

2) Structural Features of the Reactor Core: (1) Fuel rods are densely packed and its V ratio (ratio of water to fuel volume)is thus decreased. BWR is regarded as relatively loosely arranged because of the presence of steam. (2) Core height is kept to a minimum to avoid excess pressure losses. (3) Spectrum shift is rationalized in terms of improved control and neutron economy through the use of clusters made of new materials.
(4) Because of increased neutron radiation damage, it is necessary to carefully select materials for the fuel cladding tubes. (5) Control rods (or rods made of new materials) are to be spread out as evenly as possible.

3) Applications in Existing Light Water Reactors: Hitachi and the Toshiba Group have been studying applications in BWRs. On PWRs Kansai Electric Power and the Mitsubishi Group have pushed ahead with spectrum shift reactor core designs that aim at 0.9 conversion ratios and V ratios of 1.1 to 1.4. It is thought that it will be possible to realize this level of reactor core without major modifications to existing reactors and it is hoped that in the long run a fairly large amount of uranium resources will be saved. JAERI has expanded (revised) the SRAC code to include new areas and has completed a broad survey on high conversion reactor cores. Recently, they have pushed ahead with studies emphasizing the establishment of nuclear design (computation) methods for spectrum shift reactor cores with rods made with new materials.

4) High Conversion Reactors as New Reactors: JAERI has also pursued a high performance reactor core for problem-free loading into existing light water reactors, and has generally advanced its research with experiments into criticality and hot water dynamics. An example of their design research is a reactor core under study in which by flattening the core they took advantage of the fact that the Boyd coefficient could be made negative even though the V ratio was small. It is designed to have a V ratio of 0.5 when all new materials have been incorporated, a concentration rate (rate at which Pu 239 and 241 are added to replace depleted uranium) of 10.5 percent, a core length of 0.5 m, blankets every vertical 0.3 meters, a core diameter equivalent to 4 m, and 1,100 MWt output. They expect it to perform with a burn-up rate of 60 GWD/t and a Pu survival rate of 0.96. In the lower section of the reactor in Figure 1 which is typical of the reactors into which this core will be loaded, there is one BWR-type control rod drive mechanism for every three fuel assemblies and the entire fuel assembly will have movable clusters of depleted uranium arranged nearly uniformly. For example, in the case of three-batch fuel exchange, on the third (or second) batch, it will be moved to the position where the drive equipment is installed, thus compensating for the reactivity of the burn-up. It is designed to increase core performance because the reactor itself is so safe (due to its integrated form, its built-in poison, etc.).

C. The Realization of New Reactors

There is much to be said for the introduction of new reactors to be used as power reactors along side the light water reactors which have recently expanded so greatly, but the development effort needed to bring them to maturity must not be excessive. For example, consider the SPWR diagram shown in Figure 1. Although its design is still evolving and is not at a stage which could be called definite, it is thought to be worthy of study because of attributes such as the following:

1) Features of the Reactor Itself: The size of the reactor itself is determined by the steam generator. In this design each unit weighs about 2,800 tons excluding pumps and fuel. It breaks into three parts for transport. The heaviest item (the reactor containment cylinder) weighs about 1,200 tons. The reactor body is large for the rate of output, but if it were converted to a gas reactor, it would be fairly small. Not only is there little on-site construction, but it also has the advantage of reactor system simplifications such as the lack of control rods and the obviation of planning for breaks in large diameter pipes. The core allows batch-type spectrum shift and from the standpoint of fuel economy it is superior to PWR.

2) **Possibilities of a Large-Output Plant Using Modular Reactors:** Although even one unit can stand alone, a 700 MWe plant can be made by loading two reactors into one housing container or a plant suitable for APWR by using four units. Although it retains large amounts of hot water, this is a safety factor and because simultaneous large-scale accidents in the compound reactors are not anticipated, the housing systems can be greatly simplified.

3) **Possibilities as an Export Reactor:** From another standpoint a 200 MWe output reactor would probably weigh about 1,600 tons and in this case everything inside the reactor container could probably be completed at the factory and then transported to the site. It could have export appeal.

4) **Possibilities as a High Conversion Reactor:** If it were an option to achieve the high conversion reactor mentioned previously, it would have considerable appeal. If we look further into the future, it has potential as a Th-series breeder reactor, and could be a powerful back-up.

5) **Elements To Be Developed:** The base of the atomic power industry has broadened greatly in recent years due to developments in several kinds of reactors, light water reactors in particular. For example, although atomic reactor containment vessels are fairly large, there have been no problems with construction or inspection, likewise with the reactor systems and major elements which make up the reactor. It would seem that the only elements to be developed are those which increase the economy of steam generators. Although it may be necessary to build a test reactor in order to demonstrate reactors without control rods, this should not be too burdensome.

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Reactor Decommissioning Technology

43062701b Tokyo DAI 26 GENSHIRYOKU SOGO SYMPOSIUM YOKOSHU in Japanese
22-23 Feb 88 pp 13, 14

[Article by Michio Ishikawa, Japan Atomic Energy Research Institute: "Reactor Decommissioning Technology; Planning and Experiences"]

[Excerpts] **1. Introduction**

With the expansion of atomic power generation the development of atomic reactor decommissioning measures has come to have increased importance as a back-end technology, something of great concern to the whole world. In 1982, the Japan Atomic Energy Commission set forth its basic ideology on reactor decommissioning measures in Japan in its "Long Term Plan for Atomic Energy Development and Use." The basic rule is that dismantling and removal will begin as soon as possible after research operations have ceased. They have set forth a policy to work on improving the technology and adjusting the legal system before the actual decommissioning of an atomic power generating plant becomes necessary and as a first step, by working on the Japan Atomic Energy Research Institute's (JAERI) Power Demonstration Reactor (JPDR) whose usefulness was past, to perform practical experiments and develop dismantling techniques which will be of use in the future dismantling of atomic power generating plants. Based on this policy, in 1986 with a commission from the Science and Technology Agency, JAERI began practical dismantling experiments on the JPDR, thus continuing development of the reactor dismantling technology begun in 1981.

2. The Development of Decommissioning Technology

The reason why decommissioning measures for atomic reactor facilities which have ceased operations are different from those of other facilities is that the huge, solid reactor structures contain large amounts of radioactivity produced by neutron irradiation during operation. Atomic reactors and the devices, instrumentation, and tubing of the various types of systems connected to it are tightly packed into a cramped space and their radioactive contamination results in technical difficulties during decommissioning. This is the reason why large amounts of radioactive waste are generated and why costs are so high. In decommissioning reactor facilities such as this, it is necessary to understand precisely the types, quantities, and distribution of radioactivity within the reactor facility and to develop complex engineering technologies which make use of waste disposal and management techniques, dismantling techniques corresponding to radioactivity level, etc. In 1985, JAERI had nearly finished developing these techniques and with the "Committee Investigating the Development of Atomic Reactor Dismantling Techniques," they decided that the preparations to begin dismantling the JPDR had been completed.

3. The JPDR Dismantling Plan

The JPDR was a power reactor with a heat output of 90 MWt (originally 45 MWt) used for experimental research into BWRs. Completed in October 1963, it was Japan's first atomic power reactor. It finally ceased operation in 1976. The JPDR dismantling plan was submitted to the Science and Technology Agency in 1972 and was further reviewed in 1985 based on the results of the development of the technology. All buildings and machinery on the site will be dismantled and removed, with the exception of office buildings in the nonradioactive administration areas, and the ground surface on the site will be removed to a depth of 1 meter and the area will then be landscaped. Within the facility, 99.9 percent of the radioactivity is confined within the reactor pressure vessel, the structures within the reactor, and the biological shielding. The contamination in other machinery, piping, and buildings is quantitatively only 0.1 percent. Much of the residual radioactivity is Fe 55 and Co 60 and although it measured 20,000 Ci immediately after reactor shutdown, it has since fallen to its present 4,600 Ci. Parts like the reactor pressure vessel, the internal research structures, and the biological shielding which have been highly contaminated by radiation will be dismantled from a distance using the dismantling methods which they have developed
(Figure 1).

Object	Method
Reactor pressure vessel	Arc saw cutting
Structures within the reactor	Plasma arc cutting
Piping connected to the reactor pressure vessel	Disk cutting Shaped charge cutting
Biological shielding concrete	Mechanical cutting Hydro-jet cutting Controlled blasting

Figure 1. Objects and Method of Dismantling

The exterior walls of the reactor housing container and buildings will be used effectively as walls to control any minor radioactive emissions to the environment which might occur during dismantling operations. Radioactivity monitoring equipment, ventilation equipment, and drainage disposal equipment will be maintained so that radioactive substance control will be unnecessary. They will also be used in the control and disposal of gaseous and liquid wastes discharged during the dismantling operations. About 4,000 tons of radioactive wastes are expected to be generated during the dismantling of the JPDR and although these are expected to be eventually treated and disposed of pending the establishment of laws and ordinances pertaining to the rational disposal of wastes, in the meantime, they will be appropriately separated according to type of material and level of radioactivity and temporarily stored within the JPDR facility. Dismantling is expected to require about 7 years, from 1986 to 1992 and the work is estimated to require about 73,000 man days, with radioactivity exposure of 100 rem per person.

4. The Execution of the JPDR Dismantling

Approximately 1 year has passed since the dismantling of the JPDR was begun on 4 December 1986. During that time the cover of the reactor pressure vessel and the machinery around the reactor have been dismantled and removed and the integrity of the remote reactor dismantling operations center has been maintained. In January 1988 we began the most difficult operation of all, the dismantling of the structures within the reactor using a plasma arc cutter operated by a remote-controlled robot (Figure 2).

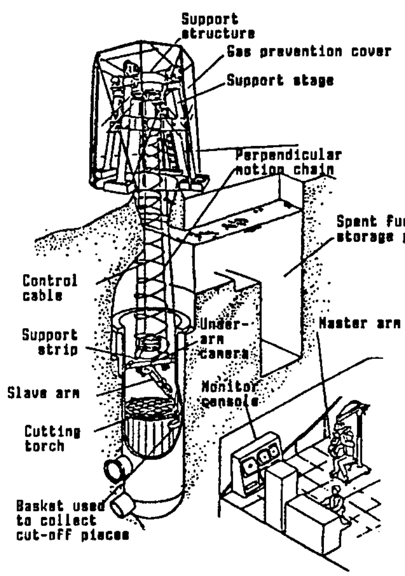

Figure 2. Conceptual Diagram of Underwater Plasma Arc Cutting Operation Using a Remote-Controlled Robot

The waste that was removed was placed in drums and 1 cubic meter containers, and stored where the machinery was removed from the dump condenser building. As to the lessons learned from experiences up to now, the filling efficiency of the drums which were loaded with waste has been low, an average of less than 1.0 in terms of specific gravity. Before dismantling and removal, experienced operations personnel who really knew the JPDR facility isolated the power sources, removed the water from the control systems, classified areas by level of radioactivity, and identified the scope of the dismantling by marking the instrumentation and machinery. This has played a big role in the smoothness of the dismantling operation. Moreover, since starting the dismantling, the frequency of ventilation filter changes within the building has increased and we realize the importance of the building ventilation equipment as a back-up to local ventilation. The work thus far has required about 13,000 man days and radiation exposure has been about 250 millirems per person.

5. Conclusion

The JPDR dismantling experiment is Japan's first experience in dismantling a power reactor and it is thought that the data and knowledge gained here will play a role in the future application of atomic power generating plant design and dismantling. Furthermore, we will gain knowledge about actual methods of treating and disposing of the waste generated by the dismantling.

A number of countries are continuing to work on these techniques, and I feel that through international cooperation we will soon see the day when a decommissioning technology acknowledged the world over is established.

13008/9365

FBR Safety, Technical Development

LWR, FBR Safety Assessments

43062071c Tokyo DAI 26 GENSHIRYOKU SOGO SYMPOSIUM YOKOSHU in Japanese 22-23 Feb 87 p 31

[Article by Yoichi Fujiie and Takeshi Morita, Tokyo Institute of Technology's Nuclear Reactor Research Laboratory: "Comparative Studies of LWR and FBR Safety"]

[Text] **1. Preface**

With the start of construction on the fast breeder reactor (prototype) which will take its place in the mainstream of the next generation of reactors following the realization of light water reactors, the development of atomic power in Japan is thought to have reached a new level. In terms of the various environmental concerns surrounding atomic power, it is clear that light water and fast breeder reactors will be following the same route for some time into the future. Accordingly, one may speak of areas in which the advance of light water reactors and the development of fast breeder reactors will push forward in mutual harmony. By comparing and contrasting safety concerns in light water and fast breeder reactors, the purpose of our basic study is 1) to summarize special safety features of both reactors, 2) to attempt to rationalize ways of thinking about safety in both reactors, and 3) to survey atomic reactor plants of the future from the standpoint of safety. To achieve these purposes, we decided to first cover present safety ideologies for both reactors, to point out important areas for study, and finally to investigate items where common safety management will be necessary.

2. Studying and Understanding the Special Safety Features of Both Reactors

The essential points of difference in safety for light water and fast breeder reactors can be resolved by looking at the differences in the energy sources which cause the anomalous generation of RI emissions. In light water reactors LOCA has been chosen as the representative hypothetical phenomenon concerning RI emissions. This means that the reactors' thermohydraulic potential governs the energy sources of the anomalous generation. On the other hand in fast breeder reactors although the actualization of chemical potential with the use of Na has a large impact on design, nuclear potential must be considered the anomalous source of their RI emissions. This is because even with the nuclear properties of fast breeder reactor cores, the pressure of the cooling materials is low and consequently LOCA rarely arise. The anomalous phenomena which are actualized by this kind of nuclear potential are represented by ATWS. However, in terms of generation frequency, it is clear that LOCA and ATWS are not necessarily comparable. In other words these representative phenomena differ as to whether contrasting phenomena are chosen on the basis of risk-orientation or ordered on the basis of generation probability. This exerts many influences on safety design, analysis, and evaluation. In choosing these representative phenomena, one must decide whether to apply clear-cut theories or to make standards for judgment based on the results of general safety evaluations called probabilistic risk analyses, in either case scrutinizing the results and future trends of safety research.

A comparison which considers safety in both reactors becomes even more important in light of accident phenomena called severe accidents where design standards have been exceeded. Accordingly, there are a great many items for investigation which are common to both reactors, like 1) events of the same type as seen in phenomena in transition, 2) policies for accident relief, which has been called disaster prevention; or accident management with policies for the design of containment vessels which are the final stage in multilevel protection, and, even more 3) the evaluation. In contrast, for instance, as seen in the five fast breeder reactor phenomena, under present circumstances it cannot be said that an entirely uniform treatment has been reached. By comparing the safety characteristics of both reactors in these areas, we have principally pointed out research topics which are mutually expandable to include severe accident evaluation, made adjustments to methods of safety evaluation, and have rethought containment system ideology.

Safety Assessments for LMFBR Plants

43062071c Tokyo DAI 26 GENSHIRYOKU SOGO SYMPOSIUM YOKOSHU in Japanese 22-23 Feb 88 pp 32, 33

[Article by Kiyoto Aizawa, Power Reactor Nuclear Fuel Development Corp.: "Probabilistic Safety Assessments for LMFBR Plants"]

[Text] **1. Introduction**

In recent years we have seen the established use of probabilistic safety assessments (PSA) which can both comprehensively and quantitatively evaluate the safety of atomic power plants and during this time the development of methods and applied research based on them have flourished even more. The application of PSA to fast breeder reactors which are still in the developmental stage and which have characteristics different from those of light water reactors is divided into three stages: conceptual design, detailed design-construction, and operation.

Of these PSA has the most application during the conceptual design stage where it achieves balances, creates effective safety ideas, strives to make planning for the arrangement of controls and machinery more suitable in areas where there are no economic or operational problems with increased operational load, and promises to lower normal risk. During detailed design and construction, it is effective in confirming whether basic designs are achievable, in composite evaluation into the attainment of safety standards, in the investigation of system interactive intervention controls between BOP and support systems on the basis of detailed design data, and also in inspection and test planning, studies into the consolidation of accident management planning, etc. At the operation stage, as with light water reactors, its goal is the on-going improvement of safety. It is expected to be effectively used in putting together an outline for operation and maintenance and in providing training programs for operations personnel. In studies such as these, comparisons with light water reactors and possibly comprehensive evaluations of uncertain influences are effective and necessary as well.

2. Conditions for Application to Fast Breeder Reactor Plants

Probabilistic assessments concerning the classification of areas for safety assessment and the reliability of principal safety systems have been performed on fast breeder plants including the "Monju" prototype now under construction, Europe's soon-to-be proposed plant, and the United States' LMR. It has thus been put to practical use in the examination of measures to improve their safety and reliability. The CRBRP in the United States, the SNR-300 in West Germany, and the CDFR in Great Britain are examples where full-scope PSA has been implemented. During these studies comparisons based on risk curves were performed between light water and fast breeder reactors, and even taking into consideration the fact that there are relatively many uncertainties in fast breeder reactor assessments, it was confirmed that even though there are not many fast breeder reactors, their safety has been maintained at the light water level. The practical use of PSA not only during the design and construction stages but also during operation is important for fast breeder reactors. Since 1982, with such practical use in mind, the Power Reactor Nuclear Fuel Development Corp. (PNC) has pushed ahead with applications research with level-3 PSA on the "Monju" fast breeder prototype. In these studies we have used independently developed safety assessment methods on new types of power reactors whose characteristics differ from those of light water reactors and our work may be said to reflect the results of our step-by-step assessment as the project advanced. In the future, as our studies progress, we expect to contribute to the characteristics of the fast breeder plant risk profile, to the analysis of measures to reduce risk, and also to the priority rankings of subjects for research and development. Through comprehensive yet sensitive analysis, we also expect to set both operational standards with conditions suitable for operation approval when one part of a multilevel safety set-up is not working and maintenance standards aimed at rationalizing efforts to reduce anomalous generation during accidents. To facilitate such application studies, I present below an outline of the conditions for fine tuning the necessary analysis coding and reliability data for machinery used in fast breeder reactors.[1]

3. The Fine Tuning of Methods for the Application of PSA to Fast Breeder Reactors

To perform system analysis it is necessary to have data on machinery failure rates, the time needed and intervals for testing, maintenance, and repair, the occurrence frequencies of phenomena causing failures, the rates for failures with common causes, etc. Because there is so little data on the machinery in fast breeder reactor plants, particularly data on sodium equipment, and what we do have is unique to fast breeder reactors, analysis studies and the consolidation of a new data base will be necessary. In June 1984, the PNC began to gather and organize data on its operational experiences with the "Joyo" experimental fast breeder reactor at its Oarai Engineering Center and with experimental sodium facilities like its 50 MW experimental steam generator facility. In January 1985, we also began the cooperation use and mutual expansion of data between Japan and the United States by concluding a special CREDO agreement with the U.S. Department of Energy for the collection, management, and organization of reliability data for machinery used in fast breeder reactors. To date we have collected data on about 21,000 pieces of equipment, about 1,500 machinery failures, and operational results for about 2.2×10^9 component hours and the search for trends using statistical analysis is underway. In the full-scale application of PSA to fast breeder reactors, in order to perform large-scale fault-tree analysis, to quantify the data, and to produce phenomena sequence models, it is necessary to handle large amounts of data quickly and effectively. Specifically, because the decay heat removal systems in fast breeder plants are special in that passive functions like natural circulation heat removal can be utilized and because delay times are thus extended, it is necessary to evaluate even more rationally human factors, unreliability, and unavailability in the repair of failed machinery and to consider models in more detail. When PSA is utilized on new types of reactors, it is necessary to alter system models throughout the various design, construction, and operational stages and to effectively use the engineering insights gained by doing so. FAUST-MODESTY-SETS-RECALL-QUEST-UNIFORM-IMPROVE are examples of analysis coding networks developed in this way. In implementing level-2 PSA in fast breeder reactors, in light of their special characteristics, one prerequisite is the fine tuning of methods which enable the quantitative evaluation of thermal and mechanical influences. Although the results of HCDA studies will be widely used for many years as just such an analysis method, when doing so, methods generally employed are those which quantitatively assess and put into integrated order phenomenon shifts in many complex areas by looking at principal phenomenon sequences from the prospective of risk and classifying them into cause processes, transition processes, reactor core expansion processes, reactor core substance rearrangements, post-accident heat removal processes, primary boundary response processes, containment structure response processes, etc. SSC, SAS, APPLOHS, SIMMER, DEBRIS-MD, PISCES, CONTAIN, etc., are examples of computer coding in which such analysis is utilized.

4. Conclusion

In fast breeder reactor plants the important safety systems are the reactor shutdown and decay heat removal systems, and because of the relative unimportance of the system which maintains the liquidity of cooling substances and because the decay heat removal systems can utilize passive functions like natural circulation heat removal, they are noted for features like the relative unimportance of its support systems and the importance of concern over common-cause failures. The PSA method is suitable for the composite evaluation of comprehensive safety problems, latent anomalies, and the importance of safety facilities. By gathering the results of future utilization research results, we will add to existing deterministic approaches and we can expect to promote the suitability of safety design and assessment, the rationalization of plant operational maintenance, the consolidation of data for decisions, and the perfection of measures concerned with operational safety and accident management.

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13008/9365

Development of High Performance FBR Fuels

43062072a Tokyo DAI 26 GENSHIRYOKU SOGO SYMPOSIUM YOKOSHU in Japanese
22-23 Feb 88 pp 36, 37

[Article by Yuji Enokido, Power Reactor and Nuclear Fuel Development Corp.: "The Development of High Performance FBR Fuels"]

[Text] **1. Introduction**

The development of FBR fuels in Japan already has a 20-year history. Through the manufacture of compound oxide fuels containing plutonium and through our experiences with irradiation experiments on the "Joyo" fast breeder test reactor and fast breeders overseas, we have come to be able to supply fuel that will be reliable enough to be used in our prototype reactor. However, with recent advances in light water reactor technology, i.e., improvements in operational economy and reliability which support high rates of operation, the technology and economic features needed to bring fast breeder reactors into actual use have reached an even higher level. At this point, with the technology from the demonstration reactor which will be built with the goal of going critical in 2003, we hope for economic features for a reactor for actual use in the 2030's. The performance of the fuels for these reactors will have to be increased and fuel cycle costs will have to be decreased. In this article I will introduce you to the form which these fuels will take in the era when fast breeders are in actual use and I will relate the results up to now of the research and development policies to achieve these.

2. Developmental Objectives for FBR Fuel Materials

Table 1 gives the principal terms of use for reactor core fuel assemblies to be used in planned Japanese fast breeder reactors. A reactor for actual use will have to have more than twice the burn-up rate and twice the amount of fast breeder neutron irradiation of the "Monju" prototype FBR, and a large-scale increase in fuel life will be necessary. These conditions are commonly recognized in many foreign countries, especially in France, and a number of countries are pushing ahead with research and development targeting radiation outputs of about 500 W/cm and a burn-up rate of 200,000 MWd/t, the results necessary for actual use.

Table 1. Terms of Use for Reactor Core Fuel Assemblies in Various Fast Breeder Reactors

	"Monju" (FBR core)	Demonstration reactor	Actual use reactor
Fuel assembly burn rate (average output) (MWd/t)	80,000	90,000-150,000	200,000
Neutron irradiation (n/cm^2) (More than 0.1 MeV)	2.3×10^{23}	$2.5 - 4 \times 10^{23}$	5×10^{23}
Pin number/fuel assembly	169	271*	331*
Radiation output (W/cm)	360	430	500
Fuel irradiation period (years)	2	3	5

*For example

3. Policies and Expectations in the Development of High Performance Fuels

In FBR fuels there is little reactive depletion during burn-up when compared to LWR fuels and when neutron waste is reduced, they characteristically can achieve 200,000 MWd/t burn-up rates without the deterioration of their amazingly high output fuel assemblies. However, to achieve this burn-up rate it will be necessary to study the physical and chemical changes in fuel plates brought about by the fissible products (f-p) that are amassed, the structural changes brought about by the swelling of the cladding tube parts and the various measures which we have against the degradation of mechanical properties. At present although irradiation results have been meager at high burn-up rates for large-scale fuel assemblies like those which will be used in the demonstration reactor, judging from a technical standpoint, from irradiation results of up to 260,000 MWd/t in fuel pins, it will be possible to achieve high burn-up rates through the development of irradiation-resistant core materials and the optimization of fuel design.

4. Results and Future Topics for Research and Development

1) The Development of Core Materials: SUS316 stainless steels are used worldwide as core materials in fast breeder reactors. In Japan up to now we have worked on improving swelling resistance and the amelioration of high temperature strength in this type of steel. Figure 1 shows trends in the amelioration of swelling resistance in the development of the improved SUS316 steel (PNC316) used in the "Monju." Because an improvement in swelling resistance can generally be expected to accompany increases in Ni content in austenite-type stainless steels, we have been performing proving tests on improved austenite steels with the goal of increasing Ni content within ranges where other characteristics are acceptable. At this point swelling latency periods are 1.9×10^{23} n/cm^2 for this type of steel

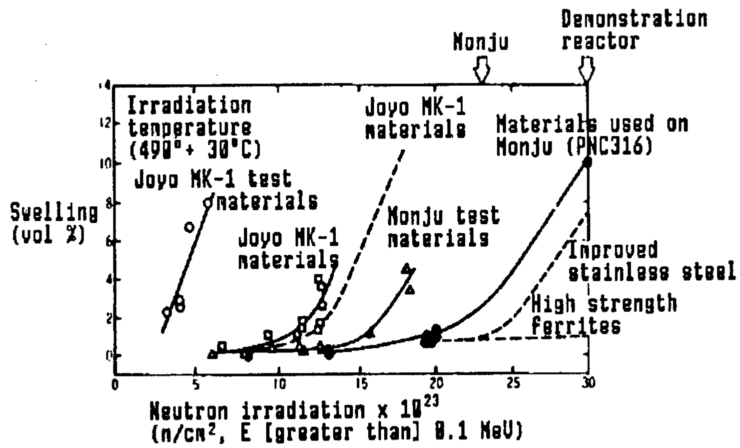


Figure 1. Swelling Resistance Amelioration in Improved SUS316 Steel and Estimates for New Materials for Use With Long-Life Fuels

(15 Cr-20 Ni, for example) and through post-irradiation testing we have confirmed that it is superior to improved SUS316 steel. Fuel design evaluations for the demonstration plant are presently being performed using these materials. However, there are limits as to how far austenite-type stainless steel can be improved, so ferrite steel with improved high-temperature strength, and specifically ferrite steel with dispersed oxide strengthening will be needed for reactors for actual use. Their development has continued apace.

2) **The Improvement of Fuel Plates:** As a result of plate volume expansion which causes increases in Cs_2 (U,Pu) O_4 and individual f-p like Mo, Zr, and Ru, in low selling cladding tubes there are amazing increases in stress due to mechanical interactions (PCMI) at high burn-up rates. It is especially important to alleviate this stress in high density plates and cladding tubes which have been broadened and flattened and there is hope for the use of soft plates which facilitate hollow plates and clips. Corrosion protection measures like coating inside surfaces with Ti will also be necessary to prevent cladding tubes from thinning out due to f-p which are corrosive at high burn-up rates.

3) **The Optimization of Fuel Design:** The outside diameter of post-demonstration reactor fuel pins will be more than 7.5 mm, and axially-nonhomogeneous fuel is one of the improved fuel candidates. With this type of fuel, not only is fuel life extended by flattening out the axis of the radiation output, but internal conversion is facilitated by the rationalization of fuel temperature at high output. Lower plenum pins are also being considered to control internal pin pressures. Although fuel assemblies in the demonstration reactor will have more than 271 fuel pins, with austenite steel cladding, tube swelling will cause interactions (BDI) between the pin bundles and lapper tubes. The permissible amount of pin bundle expansion is determined by rising cladding tube temperatures in the narrowed channels. This interdependence has been studied in detail, and

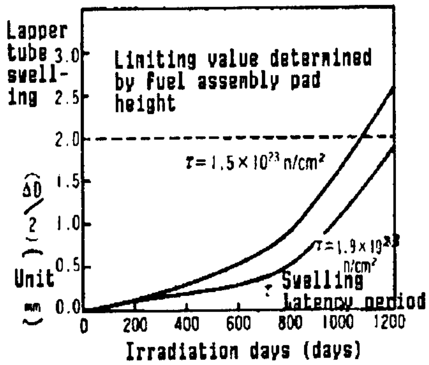


Figure 2. Analysis Examples of Interaction Between Fuel Element Bundles and Lapper Tubes

Figure 2 shows lifetime extension versus limitation values for latency period arbitrarily set at (τ) = 1.5 x 10^{23} and 1.9 x 10^{23}. On the other hand, when internal interaction between lappers (DDI) due to the expansion of lapper tubes is considered in terms of lifetime limitation, as shown in Figure 3, lifetime is once again greatly extended by improving the swelling resistance of the materials.

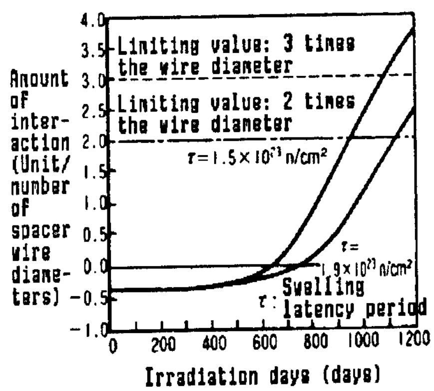


Figure 3. Analysis of Lapper Tube Swelling

4. Low Cost Fuel Assemblies

The extension of fuel lifetimes and the reduction of structural materials will both be necessary to lower costs, and we have been performing experiments on the producibility of nonlapper tube fuel assemblies, which are said to have both of these attributes. We expect these not only to have the advantage of long lifetime but also to greatly lower costs of the entire nuclear fuel cycle by simplifying the reprocessing process and reducing the amount of high-level solid waste.

5. Conclusion

Although there is much to be confirmed about fuel behavior in the future if FBR fuels are to be brought into actual use, I think we can successfully achieve economical and reliable FBR fuels with 200,000 MWd/t burn-up rates by purposefully gathering data on high performance and high burn-up rates, evaluating reactors inside and out, and making our behavior analysis coding more accurate.

13008/9365

Future Planning in Nuclear Fusion Projects

JT-60 Project

43062072b Tokyo DAI 26 GENSHIRYOKU SOGO SYMPOSIUM YOKOSHU in Japanese
22-23 Feb 88 pp 38-39

[Article by Masaji Yoshikawa, Japan Atomic Energy Research Institute: "Results and Future Plans for Development of the JT-60 Project"]

[Text] **1. Introduction**

The JT-60, the experimental critical plasma device, is a large-scale nuclear fusion device which is the mainstay of the Second Nuclear Fusion Research and Development Plan (July 1975) which was carried out under the Nuclear Power Special Research and Development Plan. Its basic goal has been to illustrate from the standpoint of plasma confinement, heating, impurity control, etc., the possibility of achieving a fusion reactor core plasma, and to clarify its characteristics (scientific proof). At this plasma performance level, one of the target ranges assigned was critical plasma condition (core temperature: several 10's to 100 million degrees, core density x confinement time ($n\tau$ value): (2-6) x 10^{19} $cm^{-3}\cdot sec$). In experiments in September and October 1987 plasma performance reached this goal at deuterium conversion values.

2. Special Characteristics of the JT-60

In conceptualizing the JT-60, we devised it so that as much of the knowledge which would be needed for future fusion reactors as possible could be obtained with it, without being limited to the goal of scientifically proving results already achieved in Japan, both within and without the Japan Atomic Energy Research Institute (JAERI). For example, diverter equipment (selective separators which are important in impurity control techniques) and studies on high frequency plasma current drivers (arrangements which stabilize and increase pulse length in tokamak reactors) are features which were not studied on similar-scale equipment like the TFTR in the United States and the JET in Europe. On the other hand, fusion fuel studies for forming DT plasmas have been performed on the TFTR and the JET, studies not pursued with the JT-60.

3. Results of the JT-60 Research

Construction on the JT-60 began in April 1978 and was completed after 7 years in April 1985. This was about 2 years after the completion of the TFTR and the JET. At this point in time heating equipment was still being built and joule experiments were begun using only plasma current joule heating. Heating equipment was completed later, in April 1987, and after testing the coupling of the heating equipment to the JT-60 itself, critical plasma experiments (heating tests) were begun (Figure 1).

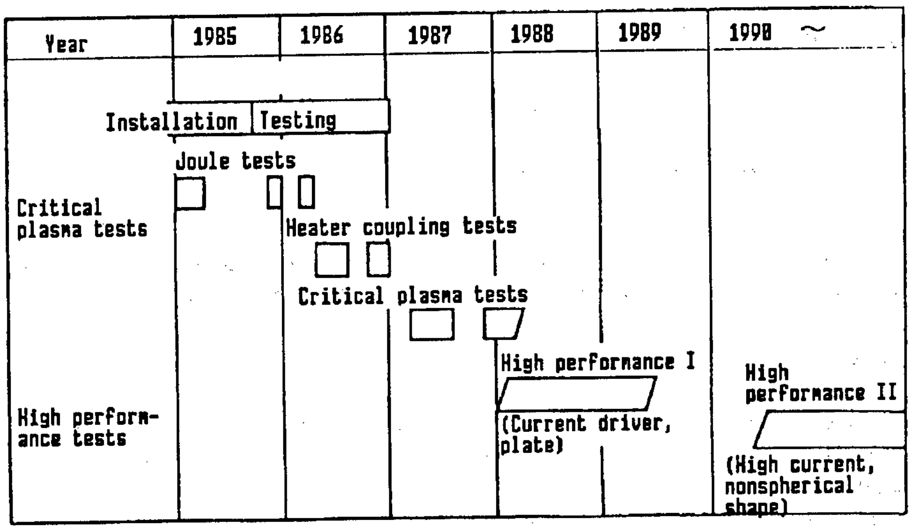


Figure 1. The JT-60 Test Program

(1) Joule Testing

The results of joule tests conducted in 1985 and the first half of 1986 are as follows:

1) We increased plasma current to 2 MA, and verified the control functions and basic condition of the equipment. The completion rate for the equipment was excellent. For example, the time required to reach 1 MA was only a fraction of that needed by the TFTR and the JET.

2) The control characteristics of the separatrix magnetic surface for the diverter were excellent. Impurities were controlled at about one-fifth through the action of the diverter.

(2) Coupling Tests and Heating Experiments

Coupling tests were begun in August 1987 by experimentally injecting heat to plasma temperatures and we were thus able to perform true heating experiments. In these experiments we emphasized the achievement of a critical plasma condition, the JT-60 target range, and we worked on operational methods and devised a number of ways to work out the bugs and increase the power of the equipment. Principal results are as follows:

1) It became clear that the most effective way to improve performance was to increase plasma current. To this end we increased plasma current to its original design value of 2.7 to 3.2 MA by increasing the power and adjusting the equipment. In order to discharge the maximum current, a part of the primary wall was replaced with graphite.

As to our operational methods, we achieved a plasma current of 3.2 MA by realizing a low q discharge (q = 2.2) through the use of NBI heating and increasing current through the use of an RF current driver. We reached the JT-60 target range in terms of deuterium equivalent conversion values (Figure 2).

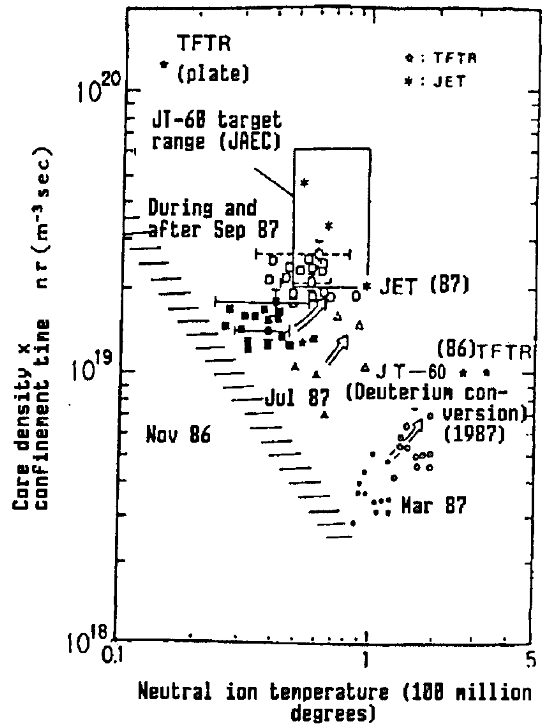


Figure 2. Performance Values Reached by the JT-60

2) We successfully maintained a 2 MA plasma current by high frequency alone using a high frequency current driver (recorded value). Through the use of NBI heating, the current driver efficiency increased to three times existing values.

3) Impurity control operations functioned as expected using the diverter.

4. Future Plans

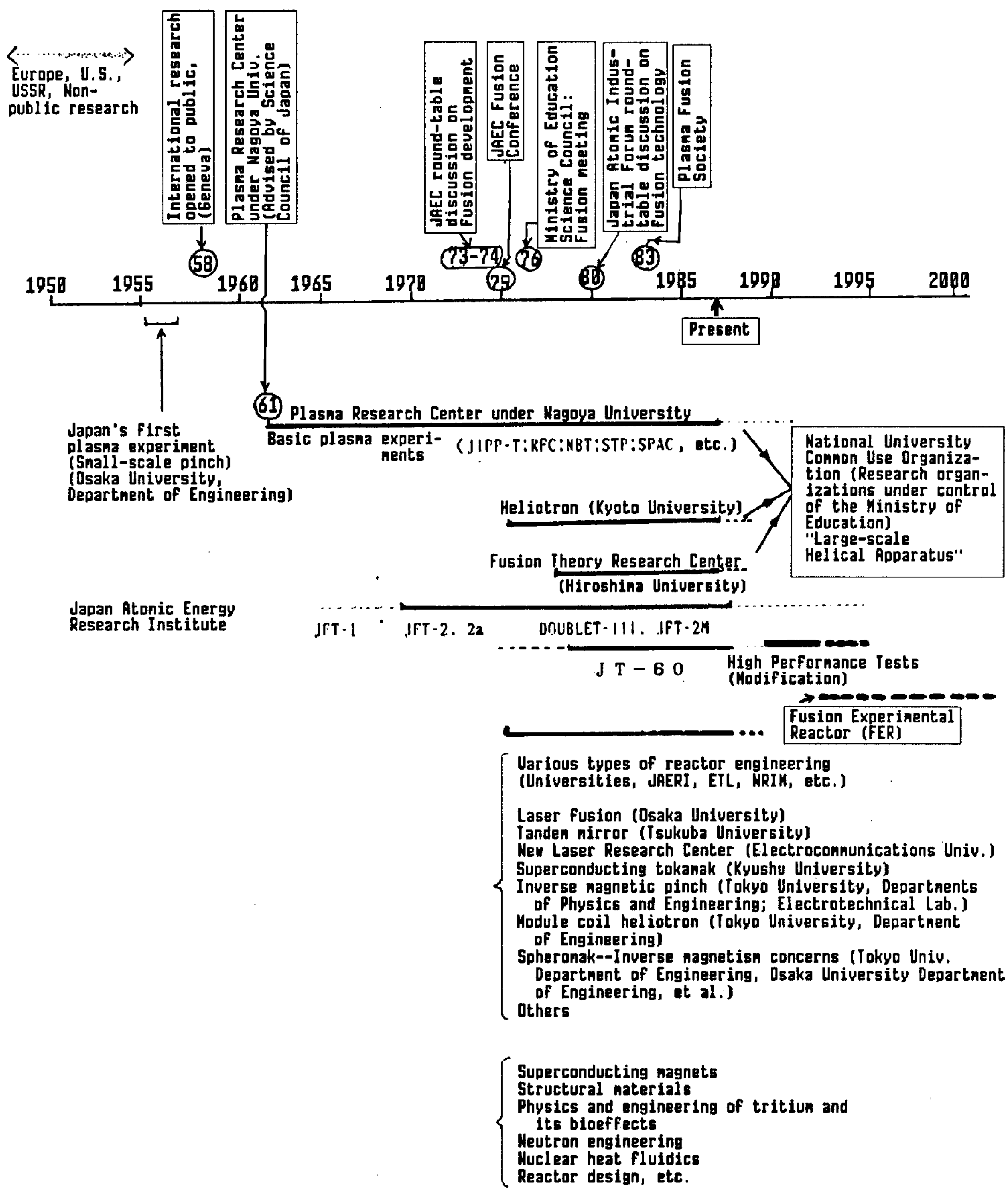
Experimentation is presently stopped on the JT-60 and we are proceeding with minor modifications so that we can perform tests with the diverter positioned on the bottom side of the plasma (it has been on the outer edge until now). Completion is expected in March 1988. We are continuing with plans to increase current to 6 MA and also to change the plasma shape to one resembling that in future large-scale devices by modifying the vacuum container and poroidal coil. By improving its performance in this way, we plan to use the JT-60 even more actively in the next stage of planning.

Planning, International Cooperation

43062072b Tokyo DAI 26 GENSHIRYOKU SOGO SYMPOSIUM YOKOSHU in Japanese
22-23 Feb 88 pp 40, 41

[Article by Tadashi Sekiguchi, Engineering Department, Yokohama National University: "Future Planning and International Cooperation in Nuclear Fusion"]

[Text] **1. Details in Advance of Nuclear Fusion Research and Development in Japan**



2. Plans for Design and Construction of Future Milestone Devices (Experimental Reactors and Devices)

Atomic Energy Commission--Fusion Conference (October 1986)
--JAEC "Long-Term Plan for Atomic Energy Development" (June 1987)

*Future Goals: Verification of basic fusion reactor engineering and technology based on the attainment of self-ignition and long-term nuclear burn (year 2000)

*Schedule Necessary:

1) After achieving the goals established (by the JAEC) for the JT-60, the start of R&D for the construction of future devices

2) It is estimated that construction of the next device will take place in the early 1990's; we will be flexible in dealing with the improvement of the tokamak performance, advances in engineering technology in related reactors, international trends, etc. (as much as possible, we will proceed under organizations developed internationally)

*Examples of design research at JAERI:

*Direction of future endeavors:

--A wide range of comparative design studies to power construction costs; the joint establishment of technology for elements which are both important and sensitive to construction cost and the implementation of the necessary R&D

--The maintenance of flexibility in design; working as much as possible at broadening the range of areas in which we can deal with unforeseen circumstances (this is the challenge of the unknown and although it seems obvious, Japan has little experience here!!)

3. Large-Scale Plasma Planning within the University Sector

Ministry of Education, Science Council, Special Fusion Meeting: active in investigation and planning since 1983

--February 1986 Report/Proposal--the establishment of a committee of investigation made up of international leaders in the technology, under the Ministry of Education

(The Essentials)

(1) When considering what is necessary for the university-related "Large-Scale Plasma Plan," keeping in mind both academic viewpoint and the latest world advances in tokamak reactors, it is appropriate to select "helical coil systems."

(2) However, future revolutionary developments are also expected in other plasma systems (tandem mirror, inverse magnetic pinch, other large-scale torus plasmas, inertial fusion, etc.) and we will push ahead with these developments as much as possible.

(3) It is appropriate to establish the new National University Common Use Organization (the so-called Research Centers Under Direct Control of the Ministry of Education, below abbreviated New Research) as recipients under the above-mentioned university-related large-scale plasma plan
(For acquisition of the land, about 40 hectare, is expected to be completed this year.)

*In May 1986 the basic design was begun for a helical-type large-scale device; in March 1987 the primary plan was proposed--At present a more detailed investigation is underway and simultaneously a study is being prepared on measures to improve and consolidate the university-related system, centering on New Research.

*Basic Design Proposal: Goals, device parameters, and construction design proposals (superconducting magnets?)

*Preparations to establish New Research will begin in April 1989

4. International Cooperation

A. Multination cooperation:
--IAEA: (Example) "INTOR"----"ITER" (International Thermonuclear Experimental Reactor)
Design studies and R&D
--Summit (See above)
--IEA: (Example) LCT, Textor, others

B. Bilateral cooperation
--Japan-United States: large-scale tokamak Doublet-III(D), others
--Japan-Europe:
--Japan-USSR: symposium sessions

13008/9365

Tritium Technology at JAERI

43062072c Tokyo DAI 26 GENSHIRYOKU SOGO SYMPOSIUM YOKOSHU in Japanese 22-23 Feb 88 pp 42, 43

[Article by Yuji Naruse, Japan Atomic Energy Research Institute: "Developments in Tritium Technology at JAERI"]

[Text] **1. Introduction**

Tritium systems in fusion reactors have an important place in the total reactor system. They consist not only of cycles which include the generation of tritium for the original loading, the supplying of fuel to the reactor core, and the recovery and circulation of tritium and deuterium from the plasma exhaust gases, but also cycles for the generation, recovery, and circulation of tritium within the blanket system. Special care must be taken in handling tritium (half-life: 12.3 years, type of emission: β rays) because it not only has a high specific emission of 10^4 Ci/g and easily replaces hydrogen in hydrogen compounds, but it also easily permeates metals at high temperatures.

At JAERI we are pushing ahead with cold testing and the consolidation of our Tritium Process Research Support (TPL) in which we intend to establish both tritium process technology based on fuel cycling in fusion reactors and safe handling techniques. We expect to begin hot testing at the end of 1987. In this article, I will present an outline of our Tritium Process Research Support and will give representative examples of the results heretofore of our research into tritium technology.

2. An Outline of Tritium Process Research Support

(1) Amount of tritium handled

--Maximum amount used per glove box test unit: 10^4 Ci/cycle
--Maximum amount used per hood test unit: 1 Ci/cycle
--Maximum amount stored: 10^5 Ci

(2) Research objectives

1) Refinement and recovery: We will perform experiments to obtain hydrogen isotope gas which has been refined by removing impurities like

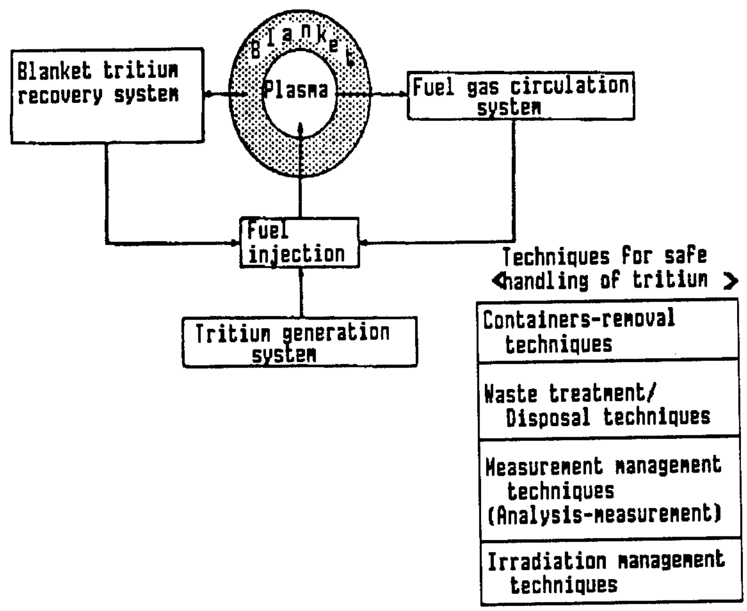


Figure 1. Conceptual Diagram of a Tritium Fuel Cycle in a Fusion Reactor

helium, moisture, nitrogen, oxygen, methane, and ammonia from hydrogen isotope gases which contain tritium (plasma exhaust gas). To develop this process, we are studying platinum membrane methods, catalytic oxidation methods, active metal bed methods, low temperature drop methods, and electrolysis methods using solid electrolytes.

2) **Hydrogen isotope separation:** We will perform experiments on refining and separating deuterium and tritium from hydrogen isotope gas (H_2, HD, HT, D_2, DT, T_2). To develop a method for separation, we will use deep cooling vapor distillation methods at varying vapor pressures and vapor diffusion methods at varying vapor diffusion coefficients.

3) **Measurement of rates at which tritium permeates and leaks into structural materials:** We will measure the rates at which tritium permeates the materials and equipment which make up the primary containment system (systems which come into direct contact with the tritium) and will thus obtain basic data for safety evaluation.

4) **Analysis and measurement:** We will develop analysis and measurement techniques regarding tritium concentration, hydrogen isotope rates, impurity concentrations, etc.

5) **Tritium containment and removal:** We plan to establish safe methods of handling tritium by operating glove boxes and tritium removal equipment.

6) **Waste concentration and solidification:** We will perform concentration and solidification tests on the tritium waste generated during the operation of the main facility.

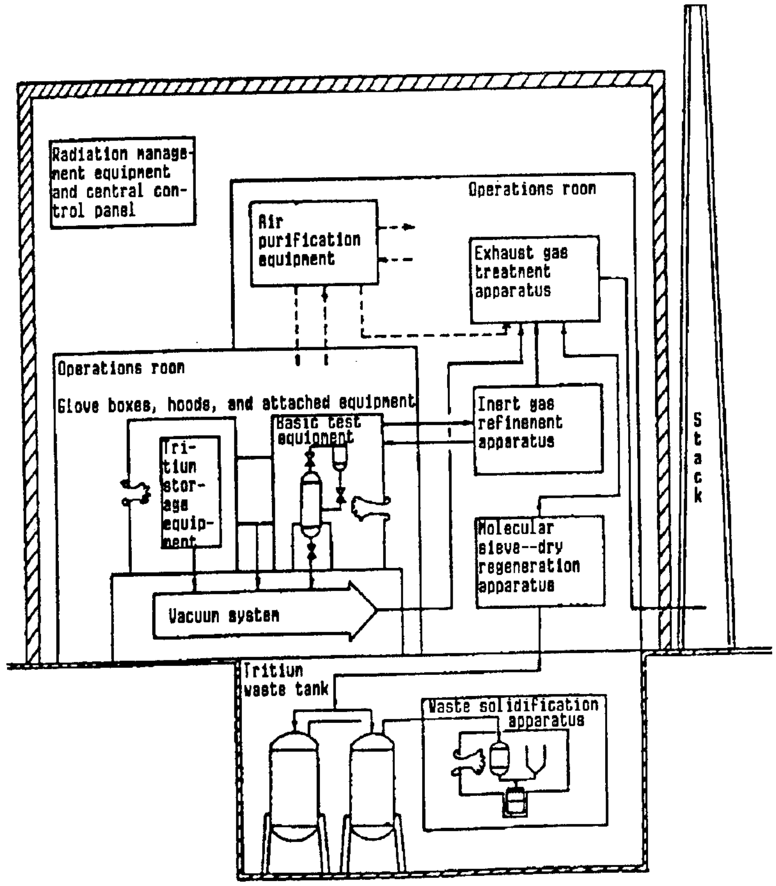


Figure 2. Tritium Confinement Concept for the JAERI Tritium Process Research Support

(3) Outline of the faculty

As shown in Figure 2, the main facility will utilize a system combining tritium removal systems with a multilevel partition containment system. When tritium is used in the chemical form of hydrogen gas, a glove box with an N_2 atmosphere will be used (anti-explosion countermeasure) and when tritium is used as water, a glove box with an air atmosphere will be used. The glove boxes will be equipped with highly airtight chambers. The tritium removal system will utilize catalytic oxidation/tritium water vapor absorption systems. The inert gas refinement apparatus will not only remove any tritium which has permeated or leaked out from the test equipment within the glove box atmosphere (N_2), but will also control the negative pressure within the glove box. The exhaust gas treatment apparatus will remove any tritium contained in the exhaust gases from the test equipment, the glove box negative pressure control systems, the waste tank vent, etc. The air purification equipment not only handles those glove boxes which have air atmospheres, but also functions as a safety system during emergencies. With the above-mentioned equipment we will be able to lower tritium emissions to the operational and surrounding environments to well below permissible levels in both everyday and emergency situations.

(4) Tritium procurement

The containers used for tritium at the main facility and during shipment to and from will be purchased frm the United States and we have completed the preparations necessary to do this.

13008/9365

Fusion Reactor Materials Research

43062073a Tokyo DAI 26 GENSHIRYOKU SOGO SYMPOSIUM YOKOSHU in Japanese
22-23 Feb 88 pp 44, 45

[Article by Shiori Ishino, Tokyo University Department of Engineering: "Advances in the Development of Fusion Reactor Materials"]

[Text] **1. Introduction**

The three large-scale tokamaks have each produced results and reactor engineering is becoming all the more important in conceptualizing future devices like the NET, FER, CIT, and ITER. As is clear from past history, the solution to new material problems has nearly always been essential to the construction of new systems. In the development of fusion reactors, there will be a wide range of problems with materials, from short-term problems with widely changing goals concerning equipment already in operation, to important medium and long-term questions concerning devices of the future, including next and next-next generation devices, prototype reactors, commercial reactors, etc. It must be emphasized that this is not a question of first solving immediate problems and then starting in on future ones. For example, in regard to radiation resistance in materials, a characteristic which will be especially important in post-demonstration reactors when fusion will be thought of as a source of energy, when one considers the number of years required for even one step in materials research and development, even assuming that research facilities are adequate, it will be essential to start on the solution of long-term problems at an appropriately early time.

In December 1986, the IEA completed a report concerning both the present state of knowledge on the materials necessary to develop fusion reactors which are economical, safe, and environmentally sound and how to promote international cooperation to solve deficiencies in that knowledge.[1] In October 1987, the Third International Conference on Fusion Reactor Materials was held at Karlsruhe.[2] I will touch upon the contents of this material and will discuss trends in the development of fusion reactor materials.

2. Principal Fusion Reactor Materials and the Present State of Their Research

Materials are considered substances or substance systems which accomplish our purposes[3] and the principal areas of materials needed for development can be divided into the five classifications listed below.[1] Table 1 illustrates the fact that problems change qualitatively according to whether their goals are established for the long or short term.

Table 1. Relationship Between Problem Phenomena and Their Object Materials

Phenomenon		Object material	Future devices	Demonstration reactor
Radiation damage		All materials	Very reactive	Severe
Fatigue-heat fatigue		Primary wall Blanket structural materials Diverter	Severe	Important
Sputtering Corrosion		Primary wall Diverter	Important	
High heat load	Normal	Diverter	Difficult	Not yet decided
	Disruption	Primary wall Diverter	Severe	To be decided
Radioactivation		All materials	Insignificant	Important
Hydrogen permeation		Primary wall Blanket structural materials Diverter	Important	Important

1) **Bundle materials for high heat flow in plasmas:** In the three large-scale tokamaks, graphite or carbon materials are the "materials inside the reactor" which directly face the plasma. Z^{eff} confinement for these has been improved to nearly one and the behavior of pure plasmas is becoming clear. Although the data base on various graphite and carbon materials is becoming more and more complete, more feedback on nuclear materials and manufacturing processes will be important. When confinement is improved, on the other hand, applications with high Z metal materials and other materials like carbide will be retested.

2) **Primary wall and blanket structural materials:** Because these are structural materials in semipermanent reactors, their behavior during use throughout their lifetime is important. Although there are many problems with these materials, like irradiation damage, heat stress, fatigue, sputtering, corrosion, disruption, radioactivation, hydrogen permeation,

etc., the greatest long-term problem is radiation damage. Radioactivation will become more severe in the future and although it will be important to strive to limit this as much as possible, I cannot help but say that at present we do not fully recognize that fusion reactors are nuclear reaction devices. We will have to investigate the possibility that the overall amount of radioactivation is decreased by improving the lifetime and performance of materials. Although heavy irradiation will be an important long-term problem, in the short run even at the FER exposure rate (3 dpa), there is no data base for large-scale structures under wet environments. I will touch upon the irradiation of structural materials later.

3) **Tritium breeding materials:** Although tritium breeding is one of the principal roles of the blanket, the combination of blanket and structural materials will be important if system breeding rates are to rise above one. Depending on the situation, it may also be necessary to use them with neutron multiplication materials. The materials being tested are liquids like Li, Pb-17Li, Pb-Li-bi, FLIBE, and Li salt solutions, and solids like LiO_2, $LiAlO_2$, Li_2SO_3, Li_4SiO_4, and Li_2ZrO_3. We have investigated or have tried to investigate many problems like inventory, permeation, compatibility, stability, soundness, heat conductivity, magnetic effects, self-shielding effects, the chemical effects of irradiation, etc.

4) **Superconducting magnet materials:** It is widely known that if superconducting magnets cannot be used in fusion reactors, then such reactors will never be economically attainable. Because magnet costs are a large proportion of reactor construction expenses, it is hoped that highly efficient magnets with strong magnetic fields will be developed.[4] In doing so, it will be important to treat large-scale magnets with an overall engineering approach, rather than with a simple material one.

5) **Special purpose materials:** Many special function materials are necessary in fusion reactors and this includes materials which are essential, although small in quantity; for example, high frequency window materials for plasma heating and gaskets, sensors, and insulators for measuring devices. Many of these are ceramic materials and it is thought that the development of materials with these functions will greatly affect other fields. However, there has been little systematic research specifically devoted to the behavior and safety of ceramic materials under intense radiation.

3. International Cooperation on Fusion Reactor Materials

Although it may be said that in many respects Japan, Europe, and the United States are all at present at about the same level of research activity, the EC has performed limited research into materials for the NET and in contrast the United States, under its Materials Research and Development Plan drawn up in 1984, has emphasized formal long-term subjects with the goal of verifying by 2005 that fusion reactors are an attractive, economical, safe, and environmentally sound energy option. Japan is special in that its universities have made a great contribution to fusion materials research and a wide range of research has been performed, from short-term to long-term problems. The distinctive character of research in each of the countries is made clear in the October 1987 ICFRM-3 report on research results.[2]

The earlier mentioned IEA report[1] called for 1) the development of a plan for joint international research on the basis of existing international cooperation, 2) the establishment of methods for developing common data bases, and 3) an immediate start on work on device selection, detail design and construction of those high energy, high speed neutron sources with the highest priorities. In regard to item 1), although the so-called Oak Ridge Matrix and BEATRIX plans were first set in motion between a number of countries under IEA protocols, the BEATRIX-II neutron irradiation tests on ^3T breeder materials have also begun and we have begun to see attempts at cooperation on the application of low radioactivation materials and superconductors. Joint TEXTOR research has forged ahead under the IEA in the same way in the PWI field. Between Japan and the United States, JAERI is continuing with HFIR/ORR irradiation research and universities have been performing irradiation research using the 1982-87 RTNS-II. In the next approximately 2 years post-irradiation testing and analysis will advance and we will gain basic information on cascade damage and fusion/fission neutron interdependence.

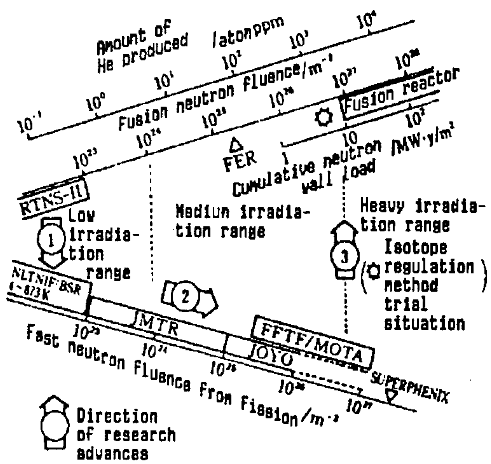


Figure 1. Conceptualization of Heavy Irradiation Research on Fusion Reactor Materials[1]

Furthermore, as shown in Figure 1,[1] FFTF/MOTA irradiation research has begun under an 8-year plan from 1987 to 1994. Under this, three programs, the U.S. Fusion Reactor Materials Irradiation Research, the above-mentioned BEATRIX-II, and the Japanese University Irradiation Research, will study irradiation under the single fusion MOTA plan. Finally, the search for a powerful neutron source should also serve as a link in IEA multinational cooperation.

4. Conclusion

Although research has made real gains under the severe economic conditions shared by many countries, in the future there will probably be more areas in which Japan, which has pushed for balanced planning, will have to lead the world. We anticipate the assistance of our friends.

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13008/9365

Long-Term Fusion Reactor Research Planning

43062073b Tokyo DAI 26 GENSHIRYOKU SOGO SYMPOSIUM YOKOSHU in Japanese
22-23 Feb 88 pp 46, 47

[Article by Kenji Sumita, Osaka University Department of Engineering: "Long-Term Plans for Fusion Reactor Engineering Research at Universities"]

[Text] **Introduction**

Recent advances in reactor core engineering have made prospects for the attainment of a fusion condition more certain and have verified the importance of those aspects of reactor engineering which are making progress in using fusion for energy production and tritium breeding. The tritium breeding blanket is one of the principal research topics in the large-scale tokamak ITER which is being built through international cooperation. Rapid progress in fusion development is expected because of the all-out effort to achieve a fusion condition. In Japan the Fusion Reactor Engineering Technology Subcommittee was founded last year in an effort to revise future planning which until then had centered on research planning connected with the Science and Technology Agency to be performed by the Japan Atomic Energy Commission's Fusion Society.

The first proposed plan for research and development relating to fusion reactor engineering at universities was the intermediate report on fusion R&D planning released in 1981 by the 11th Science Council of Japan's Fusion Research Liaison Committee. This plan proposed an emphasis on scientific research funding/special energy (fusion) research (started at that time), the advance of research by the active use of small- and medium-scale equipment in existing facilities by the researchers concerned, and the formation of a large class of researchers. Implementation has produced tremendous results and brought us to our present situation. Since this time, any number of medium-sized research facilities have been established for reactor engineering fields other than reactor core engineering (superconductivity, tritium, neutron sources, basic test devices for blanket design, etc.), and they have actively played a role at research centers. This was the beginning of the rapid progress in reactor engineering fields which has occurred in the past almost 10 years. An investigation into the fusion engineering research system by the 12th Science Council of Japan's Fusion Research Liaison Committee concentrated on the need for a central research center and on its main facilities, but did not proceed with proposals for every circumstance.

The 13th Council's study did not limit itself to established areas of learning and for the most part worked on a dispassionate reinvestigation of our future technical needs. It produced a detailed investigation regarding the urgency and importance of subjects for research in order that the requirements from various fields could be consolidated and integrated for the technical systems needed for a fusion reactor. It also seriously considered total system balance for the future goals of energy generation and tritium breeding and although at this stage there is no immediate need to fulfill actual device conditions, it verified the need for an effort to increase the prospects of doing so in the near future. It also chose areas suitable for university administration by considering the character, urgency, importance, scale, etc., of these research topics and investigated what type of equipment and policies were appropriate to dealing with the multiple classifications which followed. Furthermore, although research under various ministries and agencies both within Japan and overseas and research at universities under the Ministry of Education have both up to now progressed with suitable amounts of mutual assistance and competition between them, in the future, expenses for needed research facilities expenses will gradually increase, and we will have to make progress on more cooperation relations.

Base Research Facilities (Organizations)

Even for research plans as large as those for nuclear fusion, the formation of a national centralized research facility will not be a cure-all. On this point the role played by universities will be extremely important. There will be organizations (base research facilities) whose goal is the realization of fusion reactors and there will also be research assistance organizations (on the scale of university department lectureships in charge of related fields of learning) to assist the base research facilities through studies in related areas and it is thought that this three-tiered organization will be necessary and the most effective as well. These bases will play an especially great role in the advance of presently planned research. The base research facilities will function as leading research facilities in the various reactor engineering fields and are also expected to act as the basis for the advance of research at the central research organization. For this reason we should promote the creation of new base facilities and the transformation of existing organizations into base organizations through the strengthening, rebuilding, and establishment of adequate test equipment and facilities and through the expansion and equipping of research departments. This scale of development does not seriously exceed the capacity of our present university facilities and centers and these do not necessarily have to be common use organizations. These base research organizations will probably function as both impetus for fusion reactor engineering research and as the research system for priority areas which I will discuss later, until we see the establishment of a central research organization in reactor engineering. The things with which the base research organization will push ahead immediately are research topics like reactor materials (damage base, the behavior of surface materials), superconductivity (damage), primary wall (surface properties, structural strength), tritium (fuel management, breeding, recovery techniques), blanket (heat, fluidics, structure), fusion radiation (neutron, γ, α), and tritium's effects on living things and the environment. There will be a proposal completed as an implementable plan through discussions among researchers concerning the equipment and facilities needed for the reactor to progress (the details are expected to be presented on the appointed day, items other than those below to be in the same form).

Reactor Engineering Research at the New Research Organization at Toki

It has been decided that the immediate problem at the New Research Organization, which is expected to be established in the city of Toki, is to concentrate energy on reactor core engineering research. Whether it later meets the requirements as a central reactor engineering facility will wait for future efforts. Even so, the first stage of the Toki New Research Organization will maintain the construction of large-scale devices to be used in reactor core engineering research and it is thought that the fact that these will exist on a common site will result in the establishment of a reactor engineering department and facility. If this new group is well organized and does not seek to solve problems through a temporary system of support from other organizations and other universities, it will become the foundation for the future development of a Central Fusion Reactor Engineering Research Organization. The mission of this reactor engineering group will probably be the start of full-scale study into a systems engineering design for a fusion reactor. In regard to the time periods in which departments and facilities will be established for the more general topics for research, the principal research topics in the initial period after construction of the new organization's main equipment are as follows.

1) Superconductivity engineering (magnet engineering, cooling, insulation engineering, conductors, structural materials engineering, etc.)

2) Primary wall engineering (plasma-wall interaction, heat and structure, particle and heat control, etc.)

3) Fusion radiation engineering (particle measurement, especially the relationship between α and X-rays, radiation handling and control, etc.)

Central Fusion Reactor Engineering Research Organization

Large-scale facilities with presumed common use are considered necessary to the performance of research to answer these problems and there are some research topics whose equipment some believe cannot be realized outside of a National Central Fusion Reactor Engineering Research Organization. For these research topics, there will be no substitute for a research organization with, however, many of the necessary large-scale devices. Moreover, having the reactor core engineering research organization on the same site would be desirable but not essential from the standpoint of cooperation and it is assumed that satisfactory amounts of radiation generation and tritium use will be a possible side-effect. An investigation into the time required to establish this type of central research organization for fusion reactor engineering and into its scale will require time in the future, but such an installation will probably be necessary nonetheless. Not only would such a scale of development be difficult at university facilities, but administration would have to be spread over many ministries and agencies, beyond the research centers controlled by the Ministry of Education. A large-scale facility with equipment such as one would expect of a central research organization would be as listed, and it would exceed the level of expansion of the above-mentioned Toki New Research and Base Research organizations and would also probably concentrate on topics such as these:

1) The development of fusion materials and the development and construction of a fusion neutron source

2) The blanket for heat generation and tritium breeding

3) Fusion reactor system design and safety

Scientific Research Expenditures and Research in Priority Areas

The special fusion research begun in 1980 is now in its eighth year, and investigations into the systemization of later research in priority areas and into what it should include, in many cases have been on-going in bases within related associations. In these investigations viewpoints have shifted on two or three of the arguments presented above and investigations into measures to advance systematic research on priorities centered on research topics have intensified. In these studies, the structure of this new research system based on the extraction and ordering of research topics which exceed the framework of existing fields is being debated, and the advance of reactor engineering research whose basic theme is the unique conditions within fusion reactors is being stressed. This is close to one of the conclusions of the present stage of an investigation which has developed a world vision and discussions based on close mutual cooperation among many disciplines. Being unable to deal uniformly with the wide range of disciplines within reactor engineering, conditions would naturally vary from field to field, but I consider the systematizing of research in priority areas, together with the formation of the above-mentioned base research organizations to be necessary and very important measures in the future advance of fusion research. Moreover, although research by various ministries and agencies both within Japan and overseas and research at universities under the Ministry of Education have both up to now progressed with suitable amounts of mutual assistance and competition between them, in the future, expenses for needed research facilities will gradually increase and it goes without saying that we will have to make progress on more cooperative relations.

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