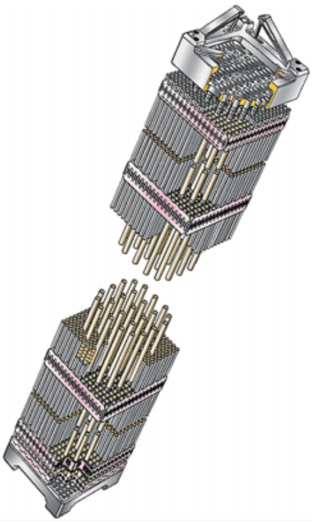

Development of Advanced Fuel and Core for High Reliability and High Performance

AKIHIRO WAKAMATSU*1
KOICHI NUNOKAWA*1
MAKOTO NAKANO*1
MANABU HAMASAKI*1
YOSHIKAZU UNO*2
TOSHIYUKI KAWAGOE*2

Mitsubishi Heavy Industries, Ltd. (MHI) has developed the fuel and core of the MHI's PWR (pressurized water reactor) for burnup extension to increase higher economic efficiency on the basis of its high reliability, and considerable and excellent operating experience. Recently, utilities have applied the 55,000 MWd/t fuels licensed discharge burnup of 55,000MWd/t in assembly average to commercial reactors with excellent results. MHI is promoting the development of further burnup extension to improve higher economic efficiency. MHI is also undertaking the development of fuels and cores with high reliability and flexibility toward the longer cycle and plant uprating operation of commercial reactors.

1. Introduction

MHI has a considerable and excellent operating experience with PWR fuel since it first supplied PWR to the Kansai Electric Power Co. Inc., Mihama Unit 1 in 1970. As of August 2006 more than 17,000 assemblies have been produced (**Fig. 1**).

In the early days, MHI experienced many problems with the fuels (including fuel leakage) imported from Westinghouse Electric Corporation of the United States and various other problems in the 1980s. However, as a result of steady root-cause analysis and countermeasures, MHI realized the excellent operating experience that no fuel leakage was observed for approximately 13 years from 1991.

On the other hand, along with the promotion of countermeasures against fuel problems, MHI started development programs to increase the licensed discharge burnup from 39,000 MWd/t to 48,000 MWd/t in the 1980s, and the 48,000 MWd/t fuel (Step 1 fuel) was put into commercial use in the early 1990s. In 2004, the licensed discharge burnup was raised from 48,000 MWd/t to 55,000 MWd/t (Step 2 fuel), with the fuels used in commercial reactors showing excellent performance (**Fig. 2**).

The actual results of the fuel and core development and the future prospects are described below.

2. Experience and status of development of fuel and core

MHI has promoted improvements in the economic efficiency of its fuel and core mainly through burnup extension, maintaining high reliability. In the meantime, MOX fuel and reprocessed uranium fuel have been utilized for the more effective use of uranium resources.

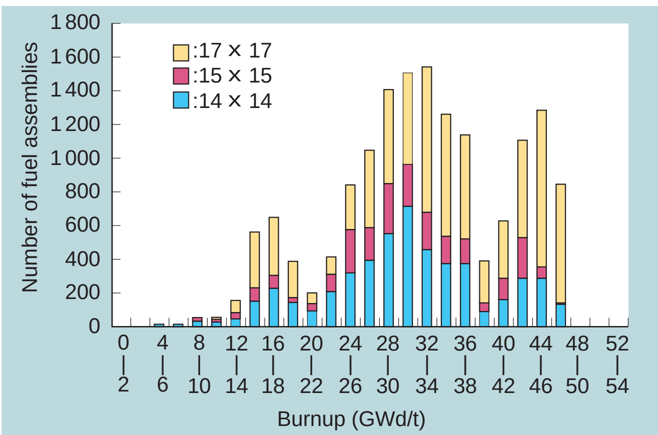

Fig. 1 Burn-up Experience of Mitsubishi Fuels (as of March 2006)

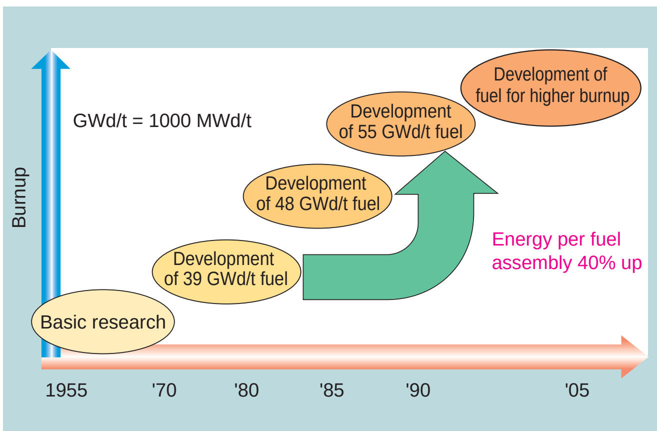

Fig. 2 Burnup extension of fuel

*1 Nuclear Energy Systems Headquarters
*2 Kobe Shipyard & Machinery Works

The burnup extension resulted in the enhanced enrichment of uranium and eventually in the increase of the licensed discharge burnup, leading to reduced fuel costs through the reduction in the number of reloaded fuel. That is, the uranium enrichment, 3.0 - 3.4%, of the early fuel (licensed discharge burnup of 39,000 MWd/t) has been improved to 3.8 - 4.1% in Step 1 fuel (48,000 MWd/t) and to 4.6 - 4.8% in Step 2 fuel (55,000 MWd/t).

By increasing the discharge burnup of fuel, the operating conditions become severer which is likely to increase the cladding corrosion and the internal pressure of the fuel rods. Therefore, by acquiring irradiation test data of fuel rods in test reactors and obtaining other data regarding its characteristics through out-of-pile mechanical and hydraulic flow tests, and material properties through material tests, and by verifying the design method and fuel behavior, the development of both fuels and cores has been promoted.

In Step 2 fuel (55,000 MWd/t), high-density pellets with a theoretical density of 97% are used to reduce the number of spent fuel assemblies by increasing the uranium weight per fuel rod. Further, pellets containing highly enriched gadolinia doped fuel up to 10% are used as burnable poison in order to suppress the excess reactivity at the beginning of the cycle and to flatten the power distribution.

As a countermeasure against cladding corrosion increased by long-term fuel use and the temperature rise at the cladding surface under a variety of fuel rod operations, MDA (Mitsubishi Developed Alloy) cladding material was developed. Its improved corrosion resistance is equivalent to ZIRLO^{TM} cladding produced by Westinghouse Electric Corporation. Both MDA and ZIRLO^{TM} are suitable for Step 2 fuel.

A FINE code, a Mitsubishi fuel rod design code, was developed for Step 1 fuel and was modified for Step 2 fuel. Therefore, the FINE code is now applicable to higher gadolinia content pellets, and MDA and ZIRLO^{TM} cladding tubes. In addition, the FINE code has models on the degradation of pellet thermal conductivity with burnup based on the actually measured fuel centerline temperature. The FINE code was verified with an extensive database both with irradiated and un-irradiated materials, which covers high burnup behavior and high gadolinia doped fuel behavior.

In order to maintain and improve design accuracy even in cores loaded with burnup extended fuel, we have introduced a three-dimensional core nuclear design method (**Figs. 3 and 4**). This is based on the PHOENIX-P/ANC code system and can eliminate the excessive margin involved in the conventional core design combining axial one-dimensional and horizontal two-dimensional calculations. We continue to improve the design method and have developed the PARAGON/improved ANC system, as an advanced and improved version of PHOENIX-P/ANC, employing the latest nuclear data and calculation methods.

When introducing these advances in fuel and core, such as burnup extension, we took a number of measures to ensure the safety of the plant equipment, e.g. an increase of number of control rods, an additional boron supply tank and a higher boron concentration, in order to keep the reactor shutdown margin required form a safety analysis against a reduction of the control rod and boron worth due to the rise in the uranium enrichment.

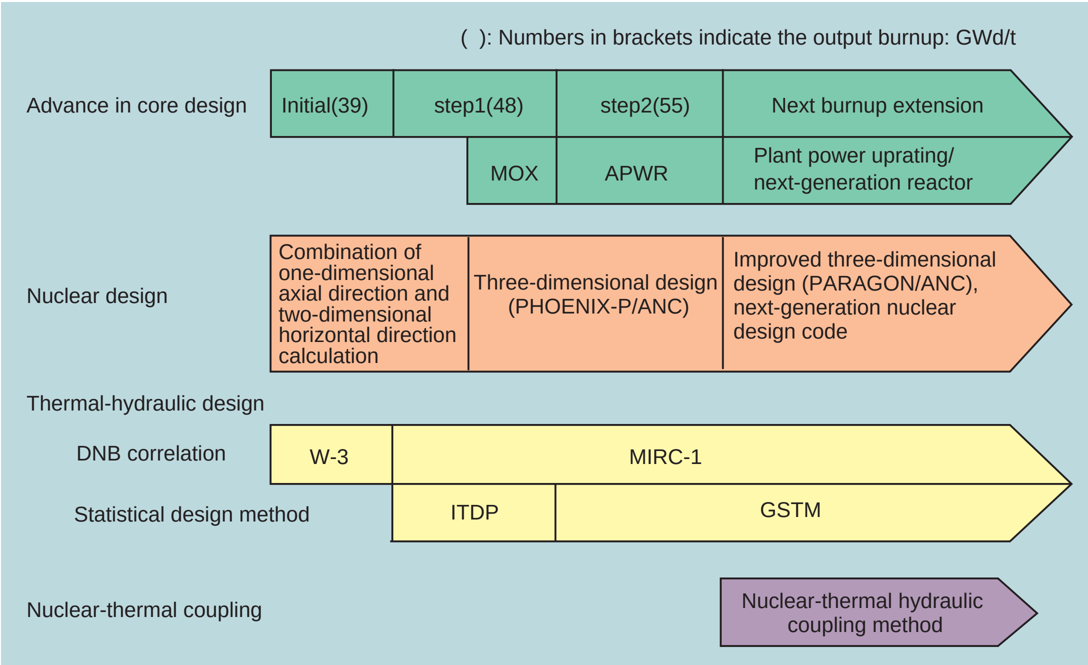

Fig. 3 Transition and prospect of core design process

Design code	Core calculation system	Core calculation method
First generation LEOPARD/ HIDRA/ PANDA	One-dimensional + two-dimensional combined calculation	Diffusion calculation
Second generation PHOENIX-P/ANC PARAGON/ Modified ANC	Three-dimensional nodal calculation	Diffusion calculation
Third generation; next-generation code (under development)	Three-dimensional fuel rod by rod calculation	Transport calculation

Fig. 4 Advanced core nuclear design code

Further, in order to enhance the flexibility of the core design by ensuring the thermal margin, the generalized statistical thermal-design method (GSTM) has been introduced for thermal hydraulic design (**Fig. 5**). The GSTM is a further advanced statistical design method over the improved thermal design procedure (ITDP) which was the first statistical design method for MHI thermal design. In this method, the uncertainty of the major input parameters such as the reactor power and reactor coolant temperature, and of the DNB(Note 1) correlation (MIRC-1) are combined statistically, and a rational assessment of DNBR(Note 2) is carried out by adopting the Monte Carlo method with higher accuracy for statistical processing.

Note 1: DNB (Departure from Nucleate Boiling)

The transition of the heat transfer mode from normal nucleate boiling (excellent state of heat transfer due to continuous generation of vapor bubbles on the fuel rod surface) to film boiling (the state of poor heat transfer due to the fuel rod surface becoming covered with a steam film) is called DNB (departure from nucleate boiling). If a DNB occurs, the fuel rod cladding is likely to be damaged due to the excessively high temperature rise. Hence, the utmost care is taken in the reactor design so that the fuel rods will not experience DNB during normal operation and anticipated transients.

Note 2: DNBR (Departure from Nucleate Boiling Ratio)

The ratio between the thermal flux on the fuel rod surface likely to cause DNB and the actual thermal flux of the fuel rod surface is called DNBR (departure from nucleate boiling ratio), and indicates the allowance before a DNB state is attained.

Next, in order to improve fuel reliability, a thorough root-cause analysis was conducted by post irradiation examination in hot laboratories in the late 1980s when fuel leakages frequently occurred. The design was improved to cope with the root-cause accordingly.

MHI experienced grid fretting fuel failure in the 1980s, which has been frequently observed in a lot of overseas plants recently. MHI developed the I-type grid with its excellent fuel rod support by impr oving the grid spring properties. The I-type grid has been employed in commercial reactors since the 1990s and no fuel leakages caused by grid fretting have been observed. The I-type grid of Zircaloy material was also developed to improve economic efficiency and to reduce dose exposure, and has been used with Step 2 fuel (55,000 MWd/t).

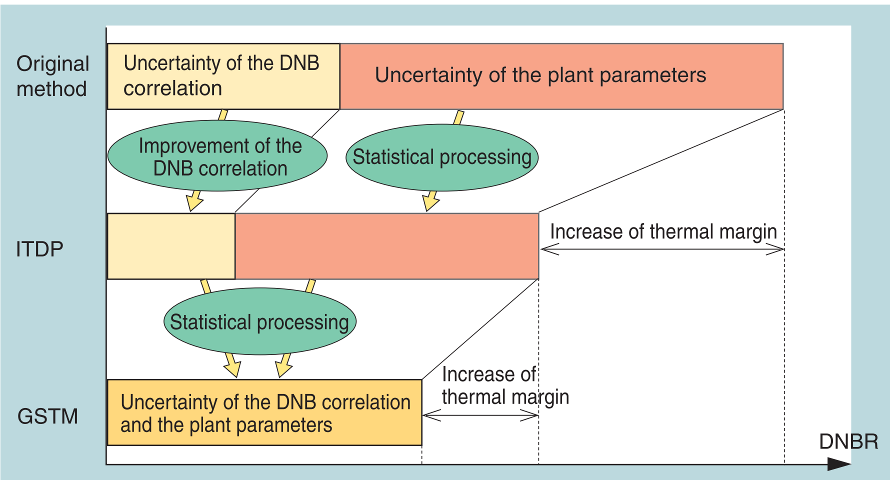

Fig. 5 Outline of expanded allowance due to improved thermal design

Further, Step 2 fuel designs ensure higher reliability as well as economic efficiency by employing a debris filter with a narrowed flow passage to the debris, and by adopting a countermeasure design to prevent incomplete rod insertion (IRI) which frequently occurred in overseas plants (**Fig. 6**).

MHI also utilizes plutonium and reprocessed uranium for more effective utilization of uranium resources.

As for the use of plutonium, MHI, as the total supplier of nuclear power plants and fuel with extensive technical expertise in fuel design, core design, plant safety analysis, plant equipment design, transportation container design, and MOX fuel processing supports the direction of the Japanese government and public utilities.

As for MOX fuel, fuel specifications with a maximum discharge burnup of 45,000 MWd/t are adopted based on the Step 1 fuel (48,000 MWd/t) design which has considerable and excellent operating experience.

MOX fuel has already been processed in the MDF (MOX Demonstration Facility) in the U.K., while MOX processing at MELOX in France has already been decided based on MHI's fuel specifications, and the preparations for processing for commercial reactors are underway.

In nuclear designs for MOX fuel, it is important to flatten the power distribution in consideration of the neutron spectrum interference effect between MOX and uranium fuels. MOX fuels are designed with an optimized arrangement of three types of fuel rod with different plutonium contents in assembly. Furthermore, MOX fuel assemblies are appropriately arranged in the core.

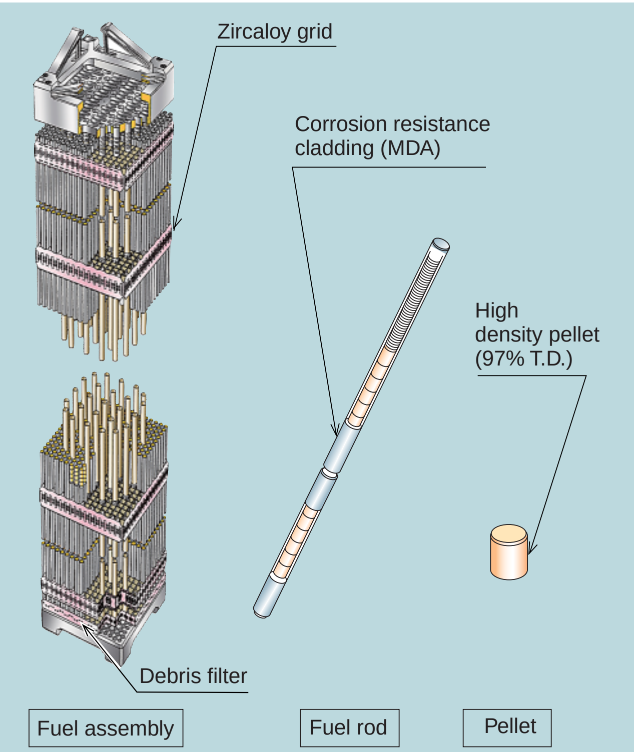

Fig. 6 Outline of structures of fuel assembly and fuel rod

Since the MOX fuel loading is suppressed to less than 1/4 to 1/3 of the core, there is no drastic change from conventional uranium fuel core characteristics. However, due attention is paid to the effect on plant equipment and facilities since the control rods and boric worth are relatively decreased. As for reprocessed uranium, uranium recovered in Japan is used according to utilities' needs. In the future uranium stored in the U.K. and France is to be gradually used.

3. Activities for improved fuel and core in the future

3.1 Fuel design improvement

Flexible cycle operation including longer cycle, plant uprating within the licensed discharge burnup of 55,000 MWd/t and the further burnup extension usage of fuel are expected as future demands. MHI addresses the fuel development to the demand.

To provide a further thermal margin for the uprated core, MHI developed a new grid spacer design with high DNB performance. The newly developed, high-performance grid enhances DNB performance by larger crossover vanes and has pressure drop equivalent to that of a conventional grid by the streamlined I-type spring.

Further burnup extensions will require further improvements in cladding corrosion resistance, and therefore development is underway of M-MDA (modified MDA), the new version of MDA with an improved chemical composition. The M-MDA has been irradiated in a commercial PWR reactor in Spain and is now under the fourth cycle of irradiation. The maximum oxide thickness up to the end of the third-cycle was obtained by on-site eddy-current measurement and is given in **Fig. 7**.

As shown in Fig. 7, the SR material of M-MDA used for fuel cladding shows higher corrosion resistance than MDA and ZIRLO^{TM} (SR: stress-relieved annealing material, RX: recrystallized annealing material). Currently, irradiation is taking place in the fourth cycle, the final cycle, and is scheduled to be completed before long.

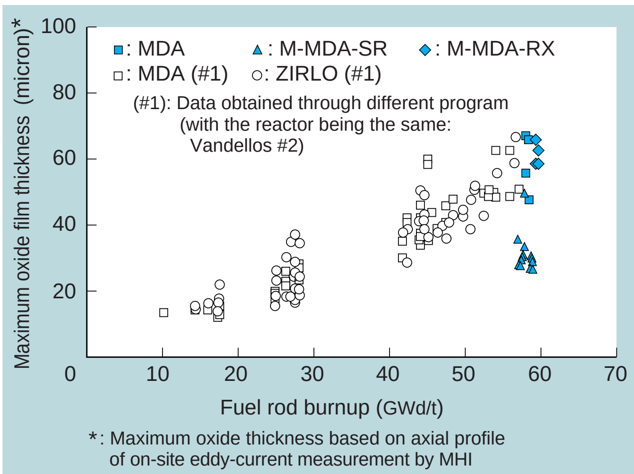

Fig. 7 M-MDA corrosion data in reactor

Furthermore, the development of J-Alloy™, a new Zr-Nb alloy, with even higher expected corrosion resistance as a cladding material is being conducting jointly with domestic industrial partners. The out-of-pile tests are almost finished and an irradiation program commenced in a Spanish commercial PWR in April, 2006. These cladding materials will be applied in commercial reactors depending on the operating conditions such as time to market, burnup, and power density.

3.2 Advanced core design

The nuclear-thermal hydraulic coupling method has been developed for existing plant uprating as an advanced core design technology.

In the conventional design method, the nuclear and thermal hydraulic characteristics were evaluated separately from the PWR point of view. In that case the nuclear characteristics, such as the power distribution, were conservatively treated as input for the thermal hydraulic design.

The nuclear-thermal hydraulic coupling method, however, can take into account the feedback effect on the power distribution in the core and the core power level due to the local changes in the coolant density inside the core and the local void generation. By appropriately reducing the excessive margin involved in the conventional method, it is possible to minimize the influence on the plant safety evaluation due to plant uprating and to maintain the flexibility of fuel operation at the present level.

For the nuclear-thermal hydraulic coupling method, we have developed a three-dimensional nuclear kinetic code, ANCK, and a three-dimensional drift flux type thermal hydraulic code, MIDAC. The ANCK code is an enhanced version of the nuclear design code ANC with a kinetic calculation function added, and it can evaluate the nuclear behavior of the transient core with complete consistency with the static core nuclear design. The MIDAC code can treat three-dimensional two-phase transient flows, and evaluate the coolant behavior in the core during normal operation and the anticipated transient, so that compared with the conventional code, it can be reliably applied to three-dimensional flows caused by a void generation specially under low pressure and/or low flow-rate conditions. The simultaneous calculation of ANCK and MIDAC with time dependence is carried out for a highly accurate assessment of the feedback effect of the nuclear and thermal hydraulic characteristics in a core. The validity of these codes has been confirmed by experimental analysis and benchmarking with the standard codes.

For future developments, MHI is making assiduous efforts to develop the next-generation code (Fig. 3), that can directly evaluate the core nuclear characteristics using a three-dimensional neutron transport calculation with a fuel rod as the calculation unit. The next-generation code will take in the latest nuclear data and calculation methods. The purpose of the next-generation code is to maintain or decrease the calculation uncertainty by the direct treatment of the fine structure inside the core using transport calculations even if the core components become more complicated due to the adoption of a core with a further burnup extension.

The next-generation code requires a vast number of calculations compared with the present design codes, we are therefore developing techniques which shorten the calculation time while maintaining accuracy, such as the latest high-speed calculation, and parallel computation.

4. Conclusion

MHI has developed burnup extension fuels and an advanced cores and continues to develop advanced fuels that can realize longer cycle operation and plant uprating, as well as higher burnup fuel while keeping a high level of reliability and economic efficiency of nuclear power plants, using integrated nuclear power plant technology.


Akihiro Wakamatsu


Koichi Nunokawa


Makoto Nakano


Manabu Hamasaki


Yoshikazu Uno


Toshiyuki Kawagoe