**ATR-MOX Fuel Design and Development**

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Japan Nuclear Cycle Development Institute (JNC) developed plutonium and uranium mixed oxide (MOX) fuels for an advanced thermal reactor (ATR) for a flexible utilization of plutonium. JNC made endeavors to obtain well-homogenized MOX pellets by a ball mill mixing method with a variety of raw powders, including MOX powder by a microwave-heating denitration process. A total of 772 MOX fuel assemblies were utilized in the ATR prototype reactor “Fugen” without failure through its whole operation period. The adequacy of fuel fabrication system, including fuel design and quality control was demonstrated by the remarkable irradiation result. JNC developed an advanced MOX fuel assembly for an ATR demonstration reactor. The integrity of the advanced fuel assembly under in-core conditions was proven by various irradiation experiments. The ATR demonstration project was unfortunately canceled in an economic aspect, but extensive irradiation data of MOX fuels has contributed to develop LWR-MOX fuels. The technique established through the ATR-MOX fuel fabrication was also applied to the fabrication of FBR-MOX fuels.

**KEYWORDS:** ATR, MOX fuel, fuel performance, irradiation data, high burnup, ball mill mixing

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**I. Introduction**

Japan Nuclear Cycle Development Institute (JNC) developed an advanced thermal reactor (ATR), which was a heavy-water-moderated, boiling-light-water-cooled, pressure-tube-type thermal reactor in a policy of using domestically-initiated technology in Japan. JNC also made endeavors to develop the plutonium and uranium mixed oxide (MOX) fuels for the ATR in the same strategy. In 1970s, JNC developed base technology on MOX fuel fabrication and researched the property of MOX pellets at the plutonium fuel development facility (PFDF) in Tokai works. JNC commenced to fabricate the first-load ATR-MOX fuel assemblies for the ATR prototype reactor “Fugen” at the plutonium fuel fabrication facility (PFFF) in 1974. A total of 772 MOX fuel assemblies were utilized in Fugen through the operation period from 1979 to 2003. None of them showed an indication of fuel failure. Most of them were standard fuel assemblies, which were composed of 28 fuel rods. After the establishment of the 28-rod-type fuel assembly, a main concern was transferred to the development of an advanced MOX fuel assembly composed of 36 fuel rods for an ATR demonstration reactor. A total of 11 lead fuel assemblies were irradiated in Fugen to develop the advanced fuel assembly. The integrity and the irradiation performance of the 36-rod-type assembly were confirmed by these irradiation experiments accompanied with post-irradiation examinations (PIEs).

The ATR demonstration reactor project was unfortunately cancelled mainly on an economic aspect in 1995, owing to materialization of MOX fuel utilization in light water reactors (LWRs) in Japan. The MOX fuel fabrication technology established through numerous ATR-MOX fuel fabrication experiences was also applied to the fabrication of MOX fuels for fast breeder reactors (FBRs) in JNC. ATR-MOX fuels filled a significant role to establish plutonium utilization system in Japan.

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**II. ATR-MOX Fuel Development Program**

Principal specifications and structures of several types of ATR fuel assemblies are shown in Table 1 and Fig. 1. These fuel assemblies were of cylindrical configuration in which fuel rods were arranged in three concentric rings to fit into a pressure tube. Fuel rod structure composed of Zry-2 cladding, upper end plug and lower end plug was similar to that of boiling water reactor (BWR). The plutonium content of outer ring rods was lower than that of inner and intermediate ring rods, because neutron was fully thermalized in the heavy water region and thermal neutron flux at the outer ring was higher than that at inner and intermediate rings.

The burnup enhancement program of ATR-MOX fuels is illustrated in Fig. 2 with reference to the burnup enhancement of BWR-UO$_2$ fuels in Japan. Standard fuel assembly for Fugen was composed of 28 fuel rods and its maximum assembly burnup was 20GWd/t. The adequacy of fuel design for the standard fuel assembly was confirmed by plenty of irradiation experiences in Fugen. The 36-rod-type fuel assembly was chosen for the ATR demonstration reactor to keep a linear heat rating moderate, because the burnup enhancement caused the increase of power peaking. The fuel rod diameter of 36-rod-type fuel assembly was reduced to...
because the same size pressure tube in diameter with Fugen was adopted in the ATR demonstration reactor. The maximum assembly burnup was 35GWd/t in an early design. Axial plutonium content distribution was introduced for the advanced fuel assembly to reduce axial power peaking in design rationalization. Several UO₂-Gd₂O₃ fuel rods were installed in the fuel assembly to compromise power discrepancy between fresh fuel assemblies and those irradiated in a core. These improvements enabled the diminution of the pressure tube number and contributed to the construction cost reduction in the ATR demonstration reactor design. The maximum assembly burnup increased slightly up to 38GWd/t by these modifications. Some series of irradiation experiments were conducted in Fugen to develop the advanced fuel assembly.

The ambitious program towards 55GWd/t with the
54-rod-type assembly was planned to cope with burnup enhancement of LWR-UO
to fuels in the aspect of fuel cycle
costs. Some out-of-pile tests such as hydraulic test,
endurance test were performed. The program was
discontinued before a start of an irradiation experiment in
Fugen, owing to cancellation of the ATR demonstration
reactor project and the limited operation period of Fugen.

III. ATR-MOX Fuel Development Activities

1. Fuel Design Technique Development

Fuel design code system established for ATR-MOX fuels
is shown in Fig. 3. Design criteria of ATR-MOX fuels were
settled on the base of those of BWR-UO
to fuels. Design code
"ATFUEL" was in charge of fuel performance evaluation,
which was a main part of the design system. The adequacy
of ATFUEL code was confirmed by irradiation experiment
data. The verification result of fuel centerline temperature is

Fig.3 Fuel design system and criteria for ATR-MOX fuels
shown in Fig. 4. The fuel centerline temperature was in-situ monitored by thermocouples of instrumented irradiation rigs at the Halden boiling water reactor (HBWR) in Norway. Temperatures calculated by ATFUEL code were slightly higher than those measured. Fuel rod inner pressures predicted by ATFUEL code are plotted in comparison with the inner pressures estimated by PIEs in Fig. 5. ATFUEL code tended to overestimate the fuel rod inner pressure. It was confirmed that ATFUEL code as the design code gave the safety margin to design criteria. JNC also got a statistical design evaluation method ready for the high burnup fuel assembly up to 55GWD/t to rationalize the safety margin.\(^1, 2\)

Fig. 4 Verification of “ATFUEL” code on fuel centerline temperature

Fig. 5 Verification of “ATFUEL” code on fuel rod inner pressure

3. Out-of-pile Experiments

Out-of-pile experiment facilities were constructed at O-arai site to develop ATR fuels and to confirm the safety of ATR core. Radial power distribution in the fuel assembly was evaluated by criticality experiments at the deuterium critical assembly (DCA). The critical heat flux (CHF) data as functions of steam quality, spacer pitch and other parameters were accumulated by hydraulic tests at the 14 MW heat-transfer-loop (HTL). The hydraulic properties such as pressure drop coefficients and two-phase multiplier coefficients were measured at the component-test-loop (CTL). The integrity of the fuel assembly under in-core conditions was confirmed by the endurance test at the CTL. These out-of-pile experiments significantly contributed toward developing ATR fuels. The obtained data were also used for licensing of the leading irradiation experiments in Fugen.

4. Irradiation Experiments

A variety of irradiation experiments were carried out to grasp the irradiation performance of ATR-MOX fuels not only in the steady state but also in off-normal events. A total of 11 leading fuel assemblies of the 36-rod-type were irradiated in Fugen to investigate the fuel performance in operational conditions and to get irradiated fuel rod segments for transient tests. The maximum assembly burnup among them reached to 38GWD/t. The PIE work of the maximum burnup assembly is under way to contribute to the CANDU option for the disposition of weapon-grade plutonium.
Some topics on the fuel performance of ATR-MOX fuels are described in the next chapter.

IV. Irradiation Performance of ATR-MOX Fuels

1. Fission Gas Release Behavior

Linear heat rating of MOX fuel rods tends to be higher than that of UO$_2$ fuel rods in the late irradiation period. The burnup at plutonium enriched spots in MOX pellet matrix discriminatingly increases. These may cause the promotion of fission gas release of MOX fuels in comparison with UO$_2$ fuels. Well-homogenized MOX pellets could be obtained with the single step ball-mill mixing method established by JNC. Fission gas release behavior of ATR-MOX fuels as a function of fuel rod burnup is shown in Fig. 7. The fission gas release rate data of BWR-UO$_2$ fuel rods are also plotted in the figure. The fission gas release rate increased in keeping with burnup beyond 10GWd/t. These fission gas release rate data are plotted to the linear heat rating experienced after 10GWd/t in Fig. 8. There was not clear discrepancy in the fission gas release behavior between ATR-MOX fuels and BWR-UO$_2$ fuels.

2. He Gas Release Behavior

MOX fuel pellets show a considerable amount of He gas release to a free volume in the fuel rod, because alpha-emitters such as $^{238}$Pu, $^{242}$Cm are created and accumulated in MOX pellets. He gas release behavior of ATR-MOX fuels is shown in Fig. 9. He gas release may cause internal pressure increase of the MOX fuel rod. On the contrary, He gas release improves gap conductance between pellets and cladding. The effect of He gas release was adequately taken into account in the ATR-MOX fuel design.

3. Fuel Rod Elongation

Fuel rod elongation rates of ATR-MOX fuels as a function of fast neutron fluence are shown in Fig. 10. The fuel rod is extended by cladding irradiation growth and by the mechanical interaction between pellets and cladding. ATR-MOX fuel rods showed slightly higher elongation rate than BWR-UO$_2$ fuel rods. Stress-relief annealed claddings were used in ATR fuels instead of recrystallization annealed claddings in BWR fuels. The difference in the heat treatment affected the cladding irradiation growth. In the ATR assembly design, the assembly length was controlled mechanically in keeping with elongation of fuel rods fixed to upper and lower tie plates and difference among fuel rods was absorbed by a rod spring attached at the top of a fuel rod to avoid fuel rod bowing.

4. Behavior in Power Ramp Experiments

Power ramp experiments were performed in the HBWR to
investigate the ATR-MOX fuel behavior during power transient events. Power ramp test rig is illustrated in Fig. 11. Power ramp rate was controlled by He gas pressure in a thermal neutron shielding coil after each fuel rod was moved from the upper waiting position to the lower test position. 11 fuel rod segments of 14.5mm in diameter were tested in two types of power ramp sequences after base irradiation in Fugen up to 15 – 22GWd/t. 6 rods were tested in a multi-step ramp sequence to estimate the approximate failure threshold and to evaluate the effect of Zr-liner attached Zry-2 cladding. 5 rods were tested in a single step ramp sequence to confirm the failure threshold on power transient events. The data obtained from these power ramp experiments are shown in Fig. 12 and Fig. 13 in comparison with data of BWR-UO fuels in almost same power ramp sequences. All 11 rods were irradiated to terminal linear heat ratings ranged from 58.3kW/m to 68.4kW/m without failure. ATR-MOX fuel rods kept their integrity beyond the failure threshold of BWR-UO fuels. It was confirmed that ATR-MOX fuels had the same level of safety margin for power transient events with BWR-UO fuels at least.

![Fig.11 Power ramp test rig in the HBWR](image)

![Fig.12 Power ramp experiments in multi-step sequence for ATR-MOX fuel rods](image)

(a) Power ramp sequence

(b) Power ramp experiment results for ATR-MOX fuels in comparison with BWR-UO fuels

Fig. 12

![Fig.13 Power ramp experiments in single-step sequence for ATR-MOX fuel rods](image)

(a) Power ramp sequence

(b) Power ramp experiment results for ATR-MOX fuels in comparison with BWR-UO fuels

Fig. 13

5. Behavior under Reactivity-initiated Accident Condition

ATR-MOX fuel rods of 14.5mm in diameter were pulse-irradiated at the Nuclear Safety Research Reactor (NSRR) in Japan to investigate fuel behavior under reactivity-initiated accident (RIA) condition and to confirm the applicability of ATR-MOX fuels to the failure threshold determined from LWR fuel database. PuO₂ particles of 400 and 1100μm in equivalent diameter were deliberately
embedded at the surface of MOX pellets in several fuel rods to examine the effect of large PuO₂ particle to the failure threshold. The failure threshold of MOX fuels with PuO₂ particles was not different from that of reference MOX fuels without the particle. Contrary to observation of PuO₂ particle melting, significant damage at cladding inner surface was not discerned in PIE after pulse tests (see Fig. 14).

Pulse experiments with 6 fuel rods irradiated in Fugen up to 20-30GWd/t were also conducted in the NSRR. The peak fuel enthalpies in the pulse irradiations with 6 fuel rods reached to 335 J/g – 586J/g, resulted in no failure of fuel rods. The experiment data are plotted in Fig. 15 with LWR-UO₂ fuel database. These ATR-MOX fuels kept their integrity beyond the failure threshold line settled by LWR-UO₂ fuel database. It was confirmed that the failure threshold under RIA conditions could also apply to the irradiated MOX fuels. These data were also taken into account in the field of licensing of LWR-MOX fuels in Japan.

6. Irradiation Database of MOX Fuels for the Water Reactor

JNC conducted a variety of irradiation experiments in the steady state condition and in off-normal conditions to develop the ATR-MOX fuel. The obtained data were accumulated as an integrated database in which fabrication data (pellet density, plutonium content, cladding diameter etc.), irradiation history data (linear heat rating, burnup etc.) and PIE data (fuel rod elongation, cladding diametric change, fission gas release, pellet density change etc.) were consistently recorded. The database can be available for the verification of fuel design code for LWR-MOX fuels.

V. Conclusions

1. A total of 772 MOX fuel assemblies were irradiated in Fugen without failure. The adequacy of the fuel fabrication system, including fuel design and quality control was proven by plenty of experiences.
2. The development of the advanced fuel assembly according to burnup enhancement program was achieved by a series of development activities. The integrity of the 36-rod-type assembly was confirmed up to the design burnup: 38GWd/t by irradiation experiments conducted in Fugen.
3. JNC established the fuel fabrication technique to obtain well-homogenized pellets through a variety of ATR-MOX fuel fabrication experiences. The MOX fuel fabrication technique was also applied to the fabrication of FBR-MOX fuels.
4. The fuel performance not only in the steady state but also in off-normal events was investigated to develop the ATR-MOX fuel. It was revealed that there was no essential difference on the irradiation performance in the water reactor between well-homogenized MOX fuels and UO₂ fuels. The pulse irradiation data of ATR-MOX fuels were taken into account in licensing LWR-MOX fuels in Japan with regard to applicability of MOX fuels to the failure threshold under RIA conditions.
References