DESIGN OF SMALL REDUCED-MODERATION WATER REACTOR (RMWR) WITH NATURAL CIRCULATION COOLING

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ABSTRACT

A small scale around 300 MWe reduced-moderation water reactor (RMWR) concept has been developed at JAERI. In order to overcome the scale demerit for a small reactor, the passive safety features has been intended to be introduced to simplify the system, and hence, to reduce the plant cost.

For the core design, a tight-lattice fuel rod arrangement under the high core average void fraction BWR type concept is introduced to attain a high conversion ratio over 1.0. The negative void reactivity coefficients are also required for the design, and the very flat short core concept is adopted. This short core design is also essential to make the natural circulation cooling possible. The core average discharge burn-up of 60 GWd/t and the operation cycle length of 24 months are also attained in the present core design.

For the system design, simplification of the system introducing the passive safety features is the basic approach to reduce the plant cost, in addition to the natural circulation core cooling concept. For example, the high pressure coolant injection system using pumps is replaced with the passive accumulator system and the emergency diesel generators can be removed, resulting in the effective cost reduction. The cost evaluation of the nuclear steam supply system gives about 20 % reduction.

The MOX fuel in the RMWR contains Pu around 30 wt% and is irradiated to a high burn-up. Therefore, the fuel safety evaluation has been performed and the acceptable results have been obtained from the thermal feasibility point of view.

1. INTRODUCTION

An advanced water-cooled reactor concept named Reduced-Moderation Water Reactor (RMWR) has been developed[1],[2] at JAERI in cooperation with JAPC and with the technical support from the Japanese LWR vendors. The reactor aims at achievement of a high conversion ratio over 1.0 with plutonium (Pu) mixed oxide (MOX) fuel, based on the well-experienced water-cooled reactor technology. Such a high conversion ratio can be attained by reducing the moderation of neutrons, *i.e.* reducing the water fraction in the core, and is favorable to realize the long-term energy supply with the uranium resources, the high burn-up / long operation cycle achievement due to the small burn-up reactivity loss, or the multiple recycling of Pu.

Although the previous work was mainly focused to realize 1,000 MWe class large reactors, the present study has been performed to establish a small reactor concept around 300 MWe with the passive safety features, under a collaboration among JAERI, JAPC, Hitachi Ltd. and Tokyo Institute of Technology (TIT) with the governmental funding from the innovative and viable nuclear energy technology (IVNET) development project. For small reactors, what is called the scale demerit should be overcome. In the present study, this is intended to be overcome by simplifying the plant system, introducing the passive safety features. One of the major passive safety features is the natural circulation primary cooling system. Additionally, several passive safety components are intended to be utilized from the plant cost reduction point of view.

In the present paper, presented are the three investigated points, that is, the design of a 300 MWe class reduced-moderation core for the high burn-up, the long operation cycle and the natural circulation cooling, the design of a small plant system with the passive safety features and the results from the MOX fuel irradiation behavior evaluation for the high burn-up range.

2. REDUCED-MODEATION CORE DESIGN FOR NATURAL CIRCULATION COOLING

The design targets for the RMWR core were set as in the following:

- Conversion ratio : more than 1.0
- Void reactivity coefficient : negative value
- Core average burn-up : 60 GWd/t
- Operation cycle length : 24 months

The first two targets are the basic major points required to every case in our RMWR design study. The last two targets are, however, added in the present study from the economical point of view.

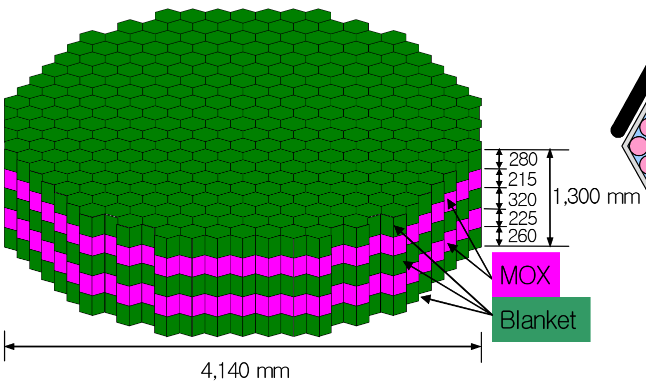
According to the previous experiences on the RMWR core design[1], the pressure loss across the core tends to be large in the RMWR core. This is basically due to the characteristic tight-lattice core configuration with the rod gap width around 1.0 mm, required for the high conversion ratio. Only the exception is the double-flat-core design for the BWR-type core. In this design, the core water mass flow rate is significantly reduced to realize a high core average void fraction as well as the fuel rod is very short around 1 m. By these reasons, the pressure drop along the fuel assembly is evaluated to be around 30 kPa, which is about one fifth of that in the current ABWR design, and hence, the natural circulation cooling of the core is expected to be possible in this design. Therefore, the double-flat-core design was adopted in the present study.

Table 1 Major dimensions and characteristics of the core

Item	Unit	Design value
Electric power output	MWe	330
Core circumscribed radius	m	2.07
Core average burn-up	GWd/t	60
Core effective height	m	0.76[1]
Core exit quality	%	52
Core void fraction	%	69
Core pressure drop	MPa	0.04
Average Puf enrichment	%	10.4
Conversion ratio	–	1.01
Max. power density	kW/m	42
MCPR	–	1.3
Void reactivity coefficient	$10^{-4}\Delta$ k/k / %void	-0.5
Fuel cycle length	month	24

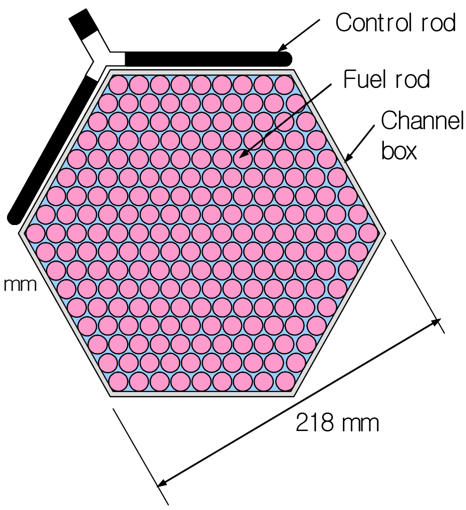
1: In addition, upper and lower blankets of 0.28 and 0.26 m

The core design obtained, satisfying the four design targets listed above, is summarized in Table 1 and is schematically shown in Fig. 1. The cross section of an assembly with the channel box is shown in Fig. 2 along with the Y-shaped control rod. The core consists of 282 hexagonal fuel assemblies, each of which consists of 217 fuel rods with the outer diameter of 13.0mm arranged in the triangular lattice in gap width of 1.3mm. The core part is very short, *i.e.* 0.76 m high, and consists of two MOX regions and an internal blanket region between them. Adding the upper and lower blanket regions of 0.28 and 0.26 m high, the total axial length is 1.21 m. This very short double-flat-core design makes the void reactivity coefficient to be negative. The average fissile plutonium (Puf) content in the MOX regions is 18 % in this design.



No. of fuel assembly 282
No. of control rod 85

Figure 1 Overview of whole core



No. of fuel rod 217
Fuel rod diameter 13.0mm
Fuel rod gap 1.3mm

Figure 2 Cross sectional view of fuel assembly

The control rods in the Y-shaped design contain B_4C with highly enriched ^{10}B, and the follower structure made of graphite is introduced to remove the water, *i.e.* the moderator, to attain a high conversion ratio. The control rods with the follower are inserted from the bottom of the core as in current BWRs. The total number of them is 85 and they are distributed at the ratio of approximately one for three fuel assemblies. The cold shut-down margin is more than 1 % as in current BWRs, even at the beginning of the cycle. Detailed core analyses for the control rod operation plan have been performed and 18 control rods out of 85 are assigned to for the operation during the cycle. Through this operation scheme, the maximum linear power density is evaluated to be 42 kW/m and the minimum critical power ratio (MCPR) evaluated with the modified CISE correlation[3] is 1.3, as listed in Table 1.

As given in Table 1, the core average void fraction is designed to be very high and 69 % in this core to achieve the high conversion ratio more than 1.0. In order to realize the high core void fraction, the core water flow rate is reduced to be 4,300 t/h, which is about one third of the corresponding scaled down value from the ABWR. Even under the smaller core flow rate, the value for MCPR is evaluated to be 1.3, suggesting the enough core cooling. Due to this smaller core water flow rate, the evaluated pressure drop across the fuel assembly is very small and 0.04 MPa, in comparison with the value of 0.17 MPa for the current ABWR. Therefore, the natural circulation cooling is possible in this core design.

3. DESIGN OF SMALL PLANT SYSTEM WITH PASSIVE SAFETY FEATURES

It is widely known that the construction cost per the power output of the plant becomes higher for the smaller power output scaled-down design. That is, the small power reactor design tends to have a disadvantage in the economical point of view, although it has other attractive points to require the low initial capital cost and to have the flexibility in the plant installation corresponding to the power demand. Therefore, the passive safety features are intended to be utilized in the present study to improve the economy of the plant, through the simplification of the system. They are, of course, also favorable to eliminate the human factors from the safety systems as much as possible and to realize the transparent safety. Based on this direction, the natural circulation cooling of the core was adopted in the present design, and also, the appropriate passive safety components were decided to be introduced. Although a few different design concepts were investigated, a *hybrid* design under the combination of the passive and the active components seems to be promising and are described in the following.

By introducing the passive components into the system, simplification of the system is expected and this results in the improvement of the plant economy. However, the passive components tend to be weaker in the working force and also tend to be larger in size. Therefore, in the present hybrid system design, the passive components are limited to be introduced to effectively improve the plant economy. The schematic overview of the plant system concept is shown in Fig. 3. This has the characteristic points as described in the following.

1) *Natural Circulation Core Cooling*

In this design, the natural circulation core cooling is introduced, and hence, the circulation pumps and related power supply can be eliminated. This results in the simplified and economical core cooling system. Also, in this system, the steam/water separator and the steam dryer can be eliminated due to the low steam velocity from the core, and hence, the gravitational steam/water separation is expected to be possible.

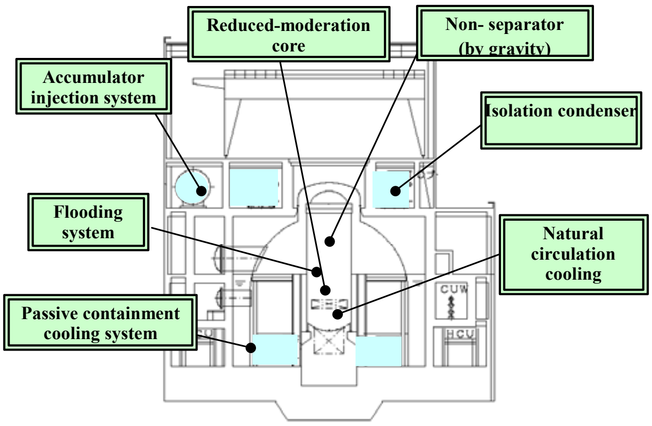


Figure 3 Plant system concept for hybrid design

2) *Simplification of Safety System*

Table 2 summarizes the safety system components and their features. They are also illustrated in Fig. 4. For cooling of the core, the passive accumulator system (Acc) is introduced as the high pressure ECCS. This has enough capacity to prevent core uncovery for all LOCA events and to keep core flooding for one day after the accidents. The capacity is also designed to be more than the volume of the dry well, and hence, the "*in vessel retention*" (IVR) is possible in the severe accidents. After the one day core cooling by Acc, the active core flooding system (FLS) continue to cool the core for a long period by injecting the water from the suppression pool to the core. Introduction of Acc can eliminate the active high pressure ECCS components as well as the related emergency diesel generator (DG) capacity.

For the reactor isolation situation, the passive isolation condenser (IC) is introduced for the core cooling and the RCIC system driven by the steam turbine is eliminated.

Table 2 Safety system component and features

	ABWR	RMWR	Features
Electrical power	1,356 MWe	330 MWe	Small power reactor
Core cooling	・HPCF : 2 ・RCIC : 1 ・LPFL : 3	・Accumulator injection system : 1 ・FLS : 2	Using the active system can keep core covery for a long term. Adopting passive Acc system can reduce emergency power resource capacity.
PCV cooling	RHR : 3	WW	PCV cooling can be continued more than 7 days by passive system.
Shutdown cooling	RHR : 3	・RHR/CUW : 1 ・IC	Adopting passive system can reduce active system.
Emergency power resource	Emergency DG : 3 (5,000kW by water cooling)	・Emergency DG : 2 (450kW by air cooling) ・Maintenance DG : 4	Adopting passive system can reduce emergency power resource capacity.

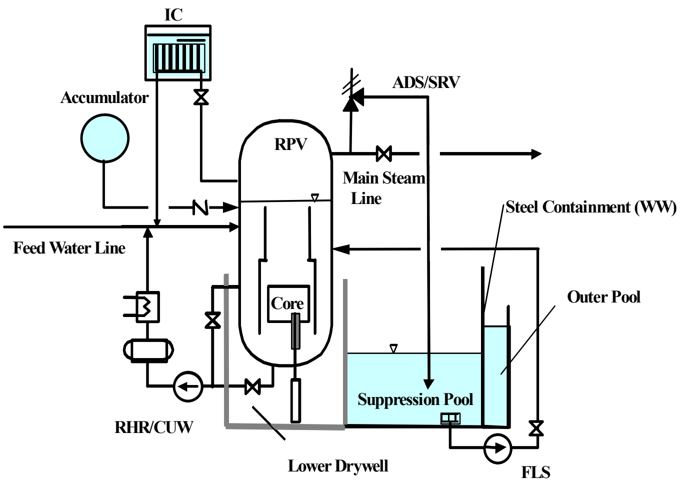


Figure 4 Concept of safety system

Figure 5 shows the analyzed results of the reactor water level during the drain piping break LOCA. Since the drain piping is located at the bottom of the pressure vessel, this accident is the severest case from the view point of core flooding. The results show that the Acc injection can keep the water level above the top of the core for one day after the LOCA initiation. Although the active component of FLS is not activated in the present analysis in order to show the performance of Acc, FLS is supposed to be activated to keep long-term core cooling by injecting the water from the suppression pool even after one day from the LOCA initiation.

3) *Passive Heat Release Capability from Containment*

The released heat in the containment vessel during the accident is absorbed in the suppression pool for one day and further heat removal is possible for a long period by the evaporation of the water in the outside pool of the containment. This system gives the enough time margin to cool the core even for the severe accidents as well as the design base accidents.

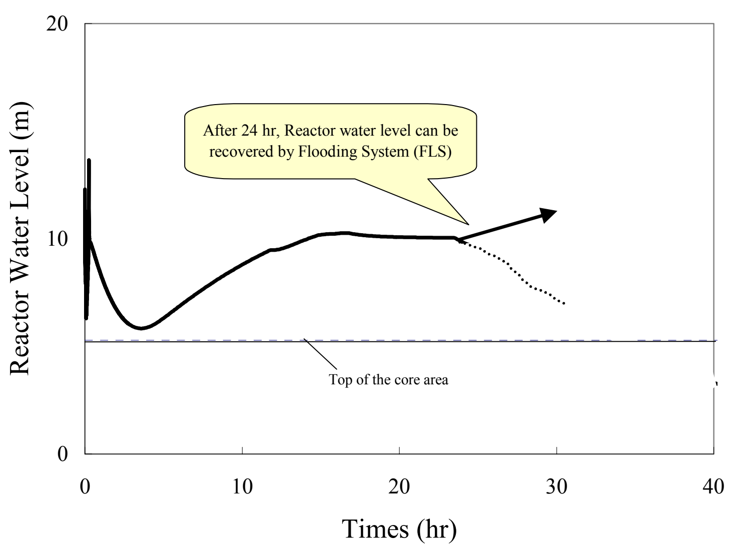


Figure 5 Analyzed water level during drain line break LOCA

4) *Other Improvement in Design*

In the present design, all the pipes in large diameter, such as for the main steam line and the feed water line, are intentionally located above the top level of the core. Therefore, small capacity of ECCS is enough to keep the core covered with the water during LOCA.

The reactor building design is improved based on the simplified system composition based on introduction of the passive features. The reactor pressure vessel is located in the lower level, and hence, the design is also improved from the seismic point of view.

In this design, the construction period is reduced due to the steel containment vessel, the improved design of the building and the structure utilizing the steel frames and the compactness in the present design.

The construction cost is estimated for the present hybrid system design. It is widely known that the construction cost per the power output of the plant becomes higher for the smaller output scaled-down design. That is, the cost obeys the law of "*0.6 – 0.75th power of the output*". Therefore, for example, when a scaled-down ABWR with 300 MWe is assumed, the construction cost per the power output is expected to become 1.45 – 1.8 times of ABWR. As described above, the active high pressure ECCS is replaced to the passive Acc in the present design. Although this change reduces the cost to some extent, the most important point, *i.e.* the most effective in economy, is the significant reduction in the emergency DG capacity. In addition, the decay heat removal system is also passive.

The cost evaluation of the nuclear steam supply system (NSSS) is presented in Fig. 6. Based on the 0.7th power law, the construction cost per power output for 300 MWe NSSS becomes about 1.6 times of ABWR. However, introducing the passive high pressure ECCS in the present design, about 20 %

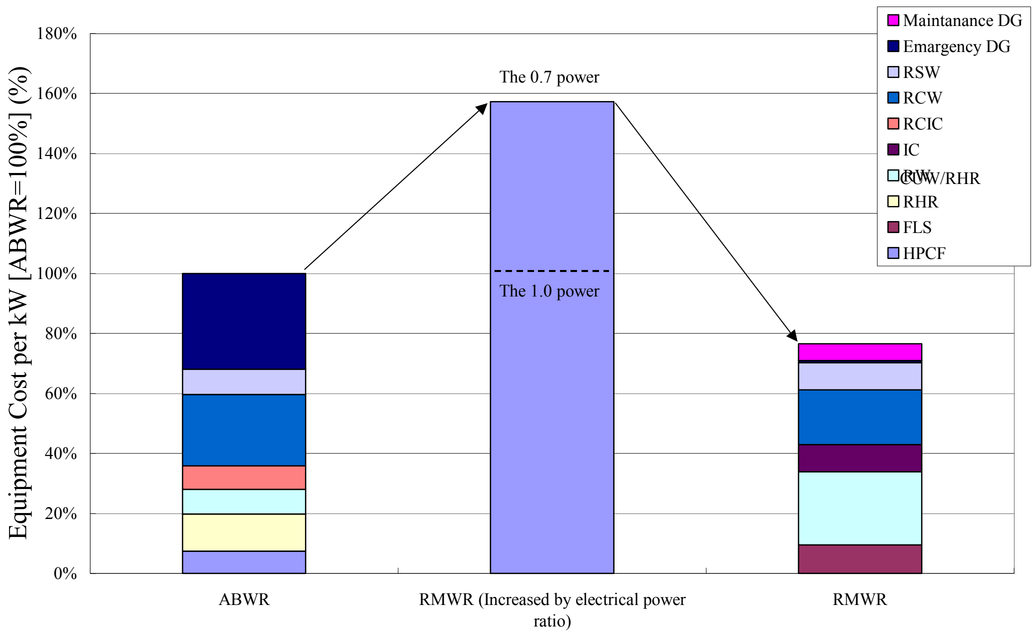


Figure 6 Cost evaluation of NSSS

reduction in const is estimated to be possible. This mainly comes from the significant reduction in the emergency DG capacity. In the present design, only the small capacity of the emergency DG is required, and hence, the air-cooled system can be adopted instead of the expensive water-cooled system for the larger capacity.

4. MOX FUEL IRRADIATION BEHAVIOR EVALUATION

The MOX fuel for the RMWR has a high Pu content about 30 wt% and is irradiated to a high burn-up about 100 GWd/t. These tough irradiation conditions make it an essential task to evaluate the thermal and mechanical feasibilities of the fuel rods. Therefore, the safety evaluation analysis of the fuel behavior has been conducted using a fuel performance code FEMAXI-RM. The code FEMAXI-RM, which is an advanced version of FEMAXI-V[4],[5] has been developed to cope with the analysis of the RMWR fuel rods with such features as combined structure of MOX and blanket parts.

The present analyses were conducted for a singe rod which is assumed to have the highest burn-up in the RMWR core. The models or materials properties applied to the analysis, such as fuel thermal conductivity, FP gas diffusion and release, creep rate, are derived or extrapolated from those used in the usual analysis of LWR fuel rods. For the first analysis, particular focus was imposed on the thermal behaviors, such as FP gas release and internal pressure increase. They are induced by the fuel temperature rise.

Figure 7 shows one example from the calculated results for the internal pressure rise along with the burn-up increase, which is essentially caused by the FP gas release from fuel pellets. In the analysis, two different models for the thermal conductivity are used to investigate their effects on the results. One is Baron model[6], which has some dependency or reduction in the thermal conductivity on the burn-up progress. The other is MATPRO-11 model[7], which has no dependency on the burn-up. The calculated internal pressures increase gradually, but the Baron model gives much higher value. This is because the model gives lower thermal conductivities of the fuel, and hence, gives higher fuel temperatures, resulting in a higher FP gas release from fuel pellets. Although the internal pressure at the end of the life is about 6 MPa, it does not exceed the coolant pressure of 7.2 MPa. This predicts that the cladding will never cause “Lift-Off” even at the very high burn-up.

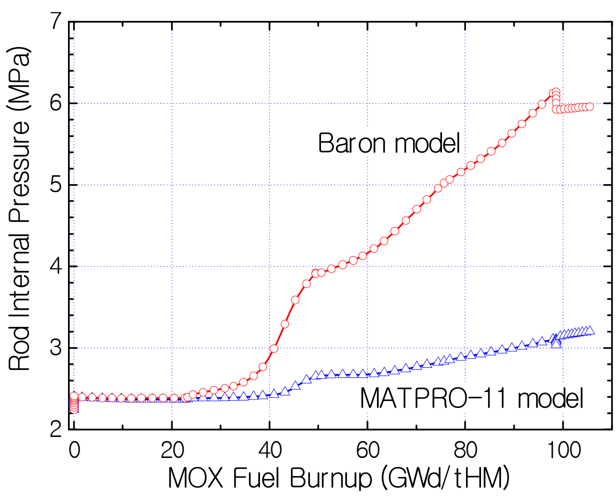


Figure 7 Rod internal pressure change with burn-up increase

The above analytical results suggest that the MOX fuel rod has no particular thermal behaviors that will raise safety and reliability concerns. However, the MOX fuel behavior with such a high Pu content has been neither fully understood nor foreseen in very high burn-up region. Therefore, a precise characterization of input data and material properties models is inevitable to evaluate the fuel safety and reliability on the basis of code predictions. In addition, modeling of the thermal conductivity degradation with burn-up extension of pellet, swelling by the FP gas pores which are generated around Pu-rich spots, and fuel rod deformation behavior are main issues to be considered in the code analysis hereafter. For these issues, irradiation experiments for the MOX fuel is of vital importance.

CONCLUSIONS

The conceptual design study for the small 330 MWe RMWR operated under the natural circulation core cooling has been performed. The core design was accomplished under the BWR-type double-flat-core design, satisfying the design targets of a high conversion ratio more than 1.0, the negative void reactivity coefficient, a high discharge burn-up of 60 GWd/t and a long operational cycle of 24 months as well as the very low pressure drop of 0.04 MPa along the fuel assembly.

In order to increase the economical competitiveness, the passive safety features are intended to be utilized and a plant system with the *hybrid* design, under the combination of the passive and the active components, are proposed. This introduces the passive safety system for the high pressure ECCS and it is estimated to be significantly effective to reduce the NSSS construction cost per the power output by about 20 % lower than that for the corresponding scaled-down ABWR plant.

The fuel safety evaluation has also been performed for the highly enriched MOX fuel of about 30 wt% Pu at the high burn-up around 100 GWd/t. Up to the present, the analytical results for the MOX fuel rod thermal behaviors give the acceptable results.

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ACRONYMS

CUW : Reactor Water Clean-up System
DG : Diesel Generator
FLS : Flooding System
HPCF : High Pressure Core Flooder
IC : Isolation Condenser
IVR : In-Vessel Retention
LPFL : Low Pressure Flooding System
PCV : Primary Containment vessel
RCIC : Reactor Core Isolation Cooling System
RCW : Reactor Auxiliary Cooling Water System
RHR : Residual Heat Removal System
RSW : Reactor Auxiliary Sea Water System
WW : Water Wall