JAERI 1151

JAERI 1151

Design of Plutonium Self-Sustaining Reactor

—Heavy Water Moderated Boiling Light Water Cooled—

January 1968

日本原子力研究所

Japan Atomic Energy Research Institute

日本原子力研究所は，研究成果，調査結果の報告のため，つぎの3種の研究報告書を，それぞれの通しナンバーを付して，不定期に公刊しております．

1.	研究報告	まとまった研究の成果あるいはその一部における重要な結果の報告	JAERI 1001-3999
2.	調査報告	総説，展望，紹介などを含め，研究の成果，調査の結果をまとめたもの	JAERI 4001-5999
3.	資　　料	研究成果の普及，開発状況の紹介，施設共同利用の手引など	JAERI 6001-6999

このうち既刊分については「JAERI レポート一覧」にタイトル・要旨をまとめて掲載し，また新刊レポートは「原研びぶりお」でその都度紹介しています．これらの研究報告書に関する頒布，版権，複写のお問合せは，日本原子力研究所技術情報部（茨城県那珂郡東海村）あてお申し越しください．

Japan Atomic Energy Research Jnstitute publishes the nonperiodical reports with the following classification numbers:

1. **JAERI** 1001-3999 Research reports,
2. **JAERI** 4001-5999 Survey reports and reviews,
3. **JAERI** 6001-6999 Information and Guiding Booklets.

Any inquiries concerning distribution copyright and reprint of the above reports should be derected to the Division of Technical Information, Japan Atomic Energy Research Institute, Tokai-mura, Naka-gun, Ibaraki-ken, Japan.

編集兼発行　日 本 原 子 力 研 究 所
印　　　刷　株 式 会 社 三 秀 舎

Design of Plutonium Self-sustaining Reactor
—Heavy Water Moderated Boiling Light Water cooled—

Abstract

The heavy water moderated power reactor has been studied mainly at Japan Atomic Energy Research Institute (JAERI) since 1963, and the boiling light water cooled type was selected for further development in Japan, last year.

Following this decision, the characteristics of the above reactor have been studied at JAERI, and based on the results the five nuclear power industry consortia in Japan have made preliminary designs individually. A report is made on the results obtained from the investigations and designs.

The study and investigation based on the principle of using natural uranium led to the adoption of a Pu self-sustaining cycle, from the standpoint of fuel cycle cost and power cost. This cycle has the following advantages: it is equivalent to a slightly enriched fuel cycle giving much flexibility in the core design, and it might be said about 15,000 MWD/T of burn-up would be expected with natural uranium by the catalizer of Pu produced in its own spent fuel.

The void coefficient, which influences the stability and safety of this type of reactor, would not be a major problem when loading Pu into the core. When adopting a Pu self-sustaining cycle, however, it should be noted that slightly enriched uranium would be loaded in the initial cycle, which may have a positive void coefficient. The problem will be eliminated to some extent by setting interlattice tubes in the core as in the SGHW and taking a lower ratio of moderator to fuel for the U loaded core and a higher ratio for the Pu loaded core. At the same time higher reactor exit steam quality or higher DNB margin would be expected in the Pu loaded than in the U loaded core.

On-load refueling is one of the major problems to be solved for the reactor: and two refueling methods (access from the reactor top and from the reactor bottom) were investigated and the corresponding refueling machines were designed.

An outline of the some refueling schemes, stability and safety analysis and plant design (stressed to the reactor structure) will also be given.

This report was presented at the international symposium on heavy water power reactors in Vienna which was held IAEA, from Sept. 11, 1967 to Sept. 15, 1967.

Aug. 1967

S. SAWAI
S. YASUKAWA
W. SHINODA
Division of Power Reactor Development
Tokai Research Establishment
Japan Atomic Energy Research Institute

Pu セルフサステイニング原子炉の設計
—重水減速沸騰軽水冷却炉—

要　　旨

重水減速動力炉については，1963年以来，日本原子力研究所で主に研究されてきたが，昨年，日本で開発すべき新型転換炉の炉型として沸騰軽水冷却型が選ばれた.

この決定に基づき，日本原子力研究所で上記原子炉に関する炉の特性が研究され，次いで5グループにより概念設計が行なわれたが，これは，その研究や設計結果についての報告である.

この原子炉の設計研究によると，天然ウランのみの供給を前提とした場合，燃料費や発電費の観点から，Pu セルフサステイニングサイクルが良いといえ，これを採ることに決定した．このサイクルは実質濃縮燃料サイクルなので，炉心設計に自由度があり，また自己の使用済燃料から出るPu を触媒として，天然ウランから約 15,000 MWD/T の燃焼度が得られるという特徴をもっている．さらに，この型の原子炉の安定性や安全性に大きな影響があるボイド係数は，Pu を装荷すると大きな問題ではなくなる．ただ Pu セルフサステイニングサイクルの場合，初期炉心にウランを用いると正のボイド係数の問題が起るが，これは SGHW のように炉内にインターラティス管を設けることにより相当程度解決される．すなわち，ウラン炉心の場合にインターラティス管内をボイドにして，減速材対燃料体積比 (V_M/V_F) を小にし，Pu 炉心の場合にはインターラティス管に重水を満たして V_M/V_F を大きくする方法である．一方熱的には Pu 炉心の方がウラン炉心より出口蒸気重量を大きくとれ，また DNB 安全率も大きくなる.

運転中燃料交換が行なえることは，この型の原子炉では開発すべき大きな問題であるが，これについては，上方から及び下方から燃料交換する2通りを検討し，それらの燃料交換装置の設計を行なった.

以上のほか，燃料交換計画，原子炉安全性と安定性の解析，および構造設計についてもごく概略ふれてある.

1967 年 8 月

日本原子力研究所，東海研究所
動力炉開発部
沢井　定
安川　茂
篠田　度

Contents

目　　次

1. Introduction

The heavy water moderated power reactor has been studied mainly at Japan Atomic Energy Research Institue (JAERI) since 1963, and many investigations on the reactor characteristics, works on the selection of the coolant and so on were made with five Japanese nuclear power industry consortia. On the other hand, a special committee was organized for discussing and settling the Japanese power reactor development program. Taking into consideration the above study results and report of the special committee, Japan Atomic Energy Commission selected the heavy water moderated boiling light water cooled reactor, as well as the fast breeder, for further development in Japan, last year.

Following this decision, further investigations on this type of the reactor were done at JAERI, and based on the results five Japanese nuclear power industry consortia have made preliminary

TABLE 1 Main parameters of designs

	Design 1	Design 2	Design 3
1. General			
Thermal power (MW)	570	500	512
Electrical power (MW) (Gross)	196.1	169.5	174.2
Burnup (Average) (MWD/T)	9,400	13,500	10,800
Thermal efficiency (%) (Turbine)	34.4	33.9	34.0
2. Reactor			
No. of coolant tubes	267	164	145
Lattice pitch (cm)	22.5	26	25
Fuel effective height (cm)	400	350	380
3. Pressure tube			
Pressure tube inside diameter (mm)	115.6	132	130.2
Pressure tube material	Zr-Nb	Zry-2	Zry-2
Pressure tube thickness (mm)	3.0	6.3	5.3
Calandria tube thickness (mm)	1.3	1.6	1.6
4. Coolant			
Coolant inlet pressure (kg/cm^2)	74	75	75
Coolant inlet temperature (°C)	279	274	284
Coolant outlet temperature (°C)	286	288	287
Outlet steam quality (%)	8.6	14.3	15.0
5. Fuel			
No. of rods/cluster	28	37	37
Sheath material	Zry-2	Zry-2	Zry-2
Fuel sheath diameter (outer/inner) (mm)	17.25/15.05	16.65/15.15	16.66/15.16
Sheath thickness (mm)	1.1*1	0.75	0.75
(Fuel) enrichment initial (a/o)	0.8*2 1.1 1.4	0.714*2 1.4	1.2
Equilibrium (N, U+Pu)	1.1	1.3	1.28
6. Inventory			
Fuel (ton-(U, or U+Pu))	47.2	34.4	32.8
Heavy water (Total-ton)	96.4	72.1	64.9
Structual material (Zry)	26.3	27.0	19.4

*1 designed by BWR's design criteria
*2 zone loading

designs on the prototype individually to make more clear important points in the designs and developments (TABLE 1). As many light water moderated power reactors will probably be used in Japan in the near future, the developments of heavy water moderated boiling light water cooled reactor will be done on the basis of the technology of the light water moderated power reactor; and the principal design conditions were given as follows:

(i) Plutonium self-sustaining cycle

The main object of this reactor installation is to use nuclear fuel more efficiently, of course economically, before the fast breeder becomes economical and to introduce a nuclear plant which uses a natural uranium. Therefore, one of the principal conditions of the reactor design is "Reactor should be operated with natural uranium feeding only", which was shown by Japan Atomic Energy Commission. However, the heavy water moderated boiling light water cooled reactor would be more economical by using slightly enriched uranium than by natural uranium, and plutonium self-sustaining cycle was proposed by JAERI, in order to attain the condition of using slightly enriched uranium by feeding natural uranium only.

This plutonium loading in the core makes the void coefficient rather negative; which in the uranium loaded core may be positive and present a big problem for the reactor safety and stability. It was also specified to study the method of demonstrating plutonium self-sustaining cycle in two years.

(ii) Maximum length of 4 m of fuel

The maximum fuel length was decided to be 4 m, taking into consideration the fuel manufacturing facilities and spent fuel reprocessing plants which will be constructed mainly for the light water moderated power reactors in Japan in the near future.

2. Feasibility Study and Design of Core, Stability, Control and Safety

2.1 Nuclear design

As the above condition of feeding natural uranium only gives little design margin for this type of the reactor, many feasibility studies on the lattices[1], fueling procedures, etc. were made at JAERI before the project was started. On the other hand, the conceptual design study on selecting the coolant was made for 300 MWe plants[2] using various coolants (pressurized heavy water, boiling light water, organic and carbon dioxide); and the boiling light water cooled type was mainly considered by the evaluation committee.

In the next stage, the various ways of using nuclear fuel in the heavy water moderated reactors were investigated; surveyed cases of fuel materials and their combinations are given in

TABLE 2 Surveyed cases of fuel types

Fuel	Enrichment	Note
UO_2	NU$\sim$1.6 a/o	
UO_2+PuO_2	DU (0.2 a/o)+Pu (0.4$\sim$13.6 a/o; 5 types of Pu)	Pu burner including phoenix fuel use
	NU+Pu	Pu self-sustaining
	SEU (NU$\sim$1.0 a/o)+Pu (0.2$\sim$0.8 a/o; several types of Pu)	Direct substitution of plutonium
UO_2+ThO_2	HEU+Th	^{235}U+Th ^{233}U+Th
	^{232}U+Th	Breeding potentiality

TABLE 3 Surveyed cases of lattice parameters

Parameter	Range of variation
Fuel rods in cluster	19, 28, 31, 37
Fuel pellet diameter (mm)	11~17
Fuel rod-rod gap (mm)	1~3
Gap between fuel bundle and pressure tube (mm)	1.5~7.5
Thermal insulation gap (mm)	4.0~18.5
Pressure tube material	Zry-2; Zr-Nb
Calandria tube material	Zry-2; Al-alloy
Lattice pitch (cm)	20~28

TABLE 2. The main lattice parameters and their ranges are also given in TABLE 3.

From this study, the following results were obtained.

(i) About 10,000 MWD/T of fuel burnup may be obtained from the pressurized heavy water cooled type with natural uranium, and the capital and fuel cycle costs will further be reduced by using slightly enriched uranium[1].

(ii) Positive void reactivity of the boiling light water cooled type can be much reduced by the use of plutonium bearing fuel, which not only contributes in nuclear safety but also improves thermohydraulic and dynamic characteristics[3]. Some examples are shown in Fig. 1 and Fig. 2.

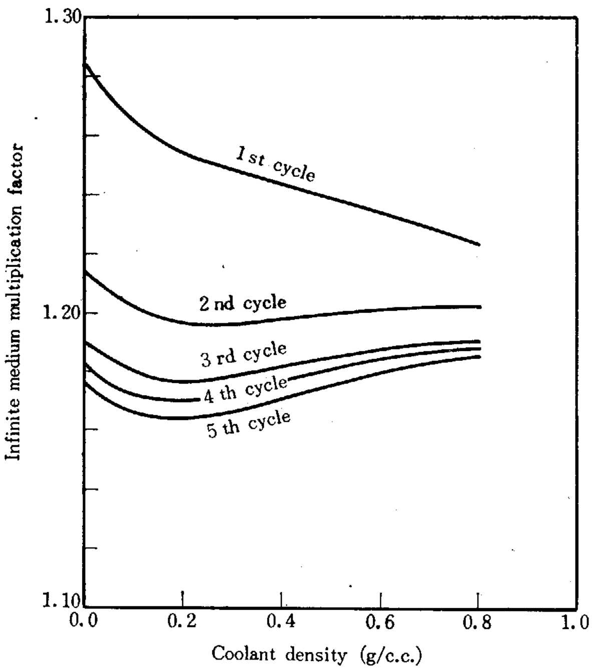

Fig. 1 The void effect of Ref. 15 E lattice

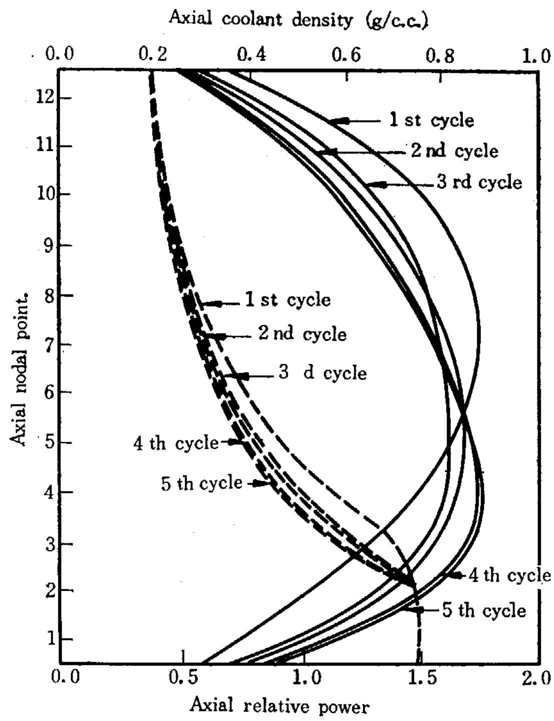

Fig. 2 The axial relative power and coolant density distributions of self-sustaininig cycles in Ref. 15 E.

(iii) When thorium, mixed with highly enriched uranium, is loaded in the pressurized heavy water cooled core, nuclear properties approach those of a reactor whose conversion ratio is 1 or slightly higher[4].

(iv) About 15,000 MWD/T of fuel burnup may be obtained with natural uranium if plutonium self-sustaining cycle is used in the boiling light water cooled lattice[3]. Moreover, from the standpoint of fuel burnup, a somewhat more moderated lattice is recommended, if only a positive void reactivity gives no serious influence on the core characteristics; or alternatively, fissile atomic concentration of base fuel material may be a little increased[5].

(v) In the uranium loaded lattice, there is little difference in burnup characteristics between a 28 rods-cluster[5)8)]. However, if the constraint on the void reactivity must be taken severely from plant control considerations, a 28 rods-cluster may be seemed to become attractive. A 19 rods-cluster is also possible to take, when plutonium fuel is employed or some suitable refueling way is used[6].

(vi) From the viewpoint of the positive void reactivity, it is desirable to reduce coolant flow area, especially the clearance between the fuel assembly and the inner wall of the pressure tube as much as the mechanical design permits.

In the third stage, several refueling ways were studied: (a) radial out-in[7)9)], (b) radial in-out[7)9)], (c) axial bidirectional[6] (there are several combinations with regard to the direction of fuel movement and of the coolant flow, both for vertical and horizontal reactors), and (d) chess board refueling with and without axial shuffling[8)9)10)13)].

From the study of the above procedures, the following results were obtained.

(i) When a radial out-in refueling is used, the fuel burnup cannot be increased without sacrificing the power flattening, and a radial in-out refueling gives a large power peaking in the core center, so that these refueling ways have no advantage[7)9)].

(ii) By the use of axial bidirectional refueling, the axial power distribution may be remarkably improved[6], even if the reactor has a relatively large positive void reactivity. If the axial power flattening is to be further improved, it is desirable to use the bidirectional refueling combined with the axial shuffling.

(iii) The chess board refueling is one of the most realistic refueling schemes, because of its simple handling. This method gives both a good power flattening and burnup characteristics. In Japan, the following schemes were studied: one-zone chess board[8)13)], two-zones out-in chess board[11] and two independent zones chess board[9]. The burnup about 1.5 to 1.8 times as much as that of the batch may be obtained if the chess board refueling is employed, and the radial gross power peaking is in the range of 1.1 to 1.5. If the axial shuffling is combined with these refueling schemes, it would be preferable to divide the core into more than two zones axially from the standpoint of power flattening, compared with two zones' result[11].

(iv) The influence of the vacant channel on the power distribution in the nearest channel was also studied by the source-sink method. The results showed nonsignificant effect might be expected when the vacant channel was filled with hot light water[9], but some distortion when it was filled with hot heavy water[11].

The typical design codes available in Japan for this type of the reactor are classified into three: (a) lattice analysis codes, (b) fuel burnup codes, and (c) miscellaneous codes, such as fine structure of nuclear-thermo-hydraulic coupled codes. Some descriptions of these codes are given in TABLE 4. These codes are now being correlated with the critical and burnup data of the light water moderated reactors and heavy water moderated ones. Some of the above results, which will be considered in the next design, are also to be examined by the critical experiments in this fiscal year.

The economic evaluation of this type of the reactor, based on the preliminary designs, has also been made by comparing with the cost of the BWR of the same power output.

TABLE 4 Process code for heavy water reactor

Name of code	Note
CLUSTER-II	Multigroup cell code for cluster lattice.
RICM	Resonance escape probability calc. code for cluster lattice.
EPSILON	Fast fission reaction calc. code for cluster lattice.
THERMOS-R	Revised THERMOS
J-THERMOPYL-1	Multigroup thermal spectrum calc. code for annular lattice.
TNS	Multigroup thermal spectrum calc. code.
MISA-Series	There are many codes in this series, such as MISA-AGC, MISA-4, 2D-MISA, 3D-MISA, which are one to three dimensional few group nuclear-thermo-hydraulic coupled codes.
WATCH-TOWER-IV	Cell burnup code
OBCD	One dimentional burnup code with several radial refueling options.
REFUEL	Axial refueling code coupled with thermohydraulic calculation.
MISA-SFL	One dimensional shuffling code.
MISA-OPR	One dimensional continuous refueling code.
SKAT-4	One dimensional scatter refueling code.
BOWRS	Three dimensional nuclear-thermo-hydraulic coupled burnup code.
OTTCUP	Three dimensional-2 group-nuclear-thermo-hydraulic coupled burnup code with several refueling options.
SATELLITE-4	Flux fine structure calc. code by source-sink method.

2.2 Thermal and hydraulic core design[8)9)10)14)]

The thermal and hydraulic design of this reactor core was made, following the design criteria of the light water moderated reactor; and investigations on the thermal and hydraulic characteristics were carried out by the use of VENUS and other codes. It may be said, however, that the burnout limit of the cluster fuel (which influences the thermal design of the core) is not yet established quantitatively, including the effects of a gap between fuel rods, wire wraps, its wrapping method, power distribution, etc., and the typical burnout or DNB limit data of the cluster fuels and BWR's design limit were used for the designs. These design limits gave rather much differences on the reactor exit steam quality in the designs with each other, and the reactor exit steam quality may also be affected by the DNB margin. These results indicated the need for the full scale heat transfer and hydraulic experiments of the cluster fuel, and preliminary full scale hydraulic experiments are to be made in this fiscal year.

Two methods were considered to reduce a amount of the light water in the core; one is to reduce the gap between fuel rods[8)] and the other is to lower the sub-cooling of the coolant at the reactor outlet (evaluating the merit of recirculating a small amount of steam)[14)] keeping the fuel rods' gap rather large with the consideration of the DNB limit. The former will affect the DNB limit and the reactor exit steam quality, and the optimum point will be studied later based on the experiments.

The coolant flow control is to be made according to the design principle: the constant coolant enthalpy rise in each channel and constant DNB margin in each channel. However, it is not so practical that each channel has its own flow control orifice, and therefore, the core was divided into several zones and the same size of the orifice was used in each zone. In the plutonium loaded core, the void coefficient is rather negative, and the flattening of the neutron flux distribution will be expected by the constant coolant enthalpy rise through each channel rather than by the constant DNB margin in each channel, which was proposed in one of the preliminary designs[9].

In the plutonium loaded core, maximum point of the power distribution approaches to the reactor inlet, as compared with the uranium loaded core, thereby giving higher DNB margin or higher reactor exit steam quality (Fig. 3).

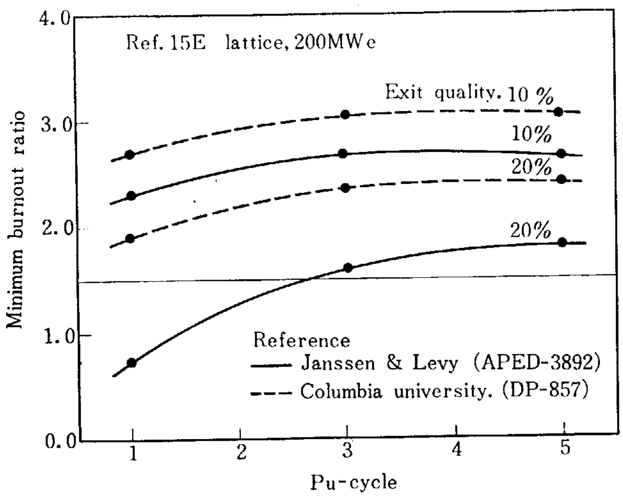

Fig. 3 Minimum burnout ratio on Pu-cycle.

2.3 Stability, Control and Safety[8)9)15)]

2.3.1 Stability

The stability is the first consideration in the dynamics of this type of the reactors. Analytical works have so far been made to investigate specifically three kinds of instabilities: (a) reactivity feedback instabilities possibly caused by a positive void coefficient of reactivity, (b) xenon-induced spatial instabilities, and (c) hydrodynamic instabilities.

In the earlier stages, efforts were concentrated on the investigation of type (a) instabilities. The results of the preliminary survey which was theoretically based on a one-point model are summarized as follows[15]: the reactor is dynamically stable for positive void coefficient, even when the coefficient takes a relatively large value, and the stability limit in terms of the void reactivity coefficient was as large as $+0.024(\delta k/k/$void fraction) in a certain reactor with $-2.4\times10^{-5}\delta k/k/^{\circ}C$ of the fuel temperature coefficient of reactivity.

Last year, the one-dimensional dynamics code, AURORA was prepared for this study[16]. The detailed analysis, using the code, confirmed most of the results of the parametric study described above[17]. Furthermore, the analysis provided informations on the dynamic characteristics of various lattices, of interest to use. Fig. 4 shows transient response of neutron flux for the stepwise disturbance of reactivity in the SGHW type of lattices, in which the lattice pitches were taken 24 cm, 26 cm and 28 cm, respectively. As shown in the Fig. 4, the neutron flux becomes divergent where the lattice pitch is 28 cm. The doubling time of this excursion, however, is of the order of 20 seconds, which is considered to be well within a tolerable range from the standpoint of control engineering.

As for type (b) instabilities, several analytical studies were involved in the preliminary designs presented by the consortia. According to the results, there may exist no unstable higher modes in the radial-azimuthal direction in a proposed prototype reactor of about 160 MWe, however, the first azimuthal mode possibly becomes unstable in a large reactor. On the other hand, circumstances become very complicated in the axial direction by the existence of voids, axial coolant flow and control absorbers. When the height of the core is larger than 5 m, the positions of axially

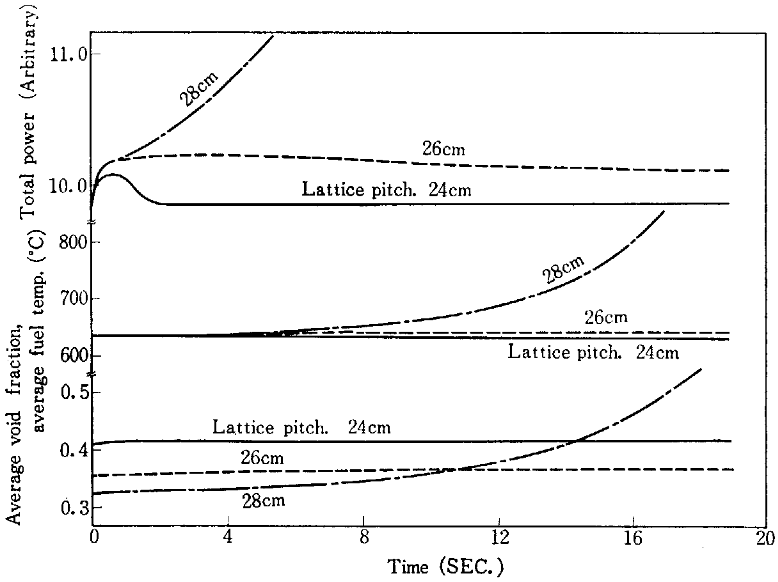

Fig. 4 Examples of transient responses. $\Delta K_D = +10^{-4}$ step (distributed doppler effect)

inserted control absorbers, if any, should be kept within a proper range so as not to excite axial higher harmonics. Now a three-dimensional dynamics code, CHAOS, is being developed for the detailed treatment of the problem[18].

The analytical study on type (c) instabilities, which was also included in one of the preliminary designs above, showed that the proposed reactor system has no tendency for the hydrodynamic instabilities. However, further investigations must be made on this problem, to prove the feasibility of this reactor concept. Work is now in progress to refine the theoretical methods in correlation with the experimental data.

2.3.2 Control

In a prototype reactor, the plant control will primarily be based on the "Turbine slaved to reactor" principle. Therefore, the reactor power will be controlled by the direct in-core flux measurements, as proposed in some of the preliminary designs, with the separate steam pressure control by an initial pressure regulator. In addition, the recirculation flow control may be incorporated into the above system to improve the load changeability of the plant. The study of such a control system showed that a load change as large as 30%/min easily can be had through the life of the proposed reactor.

The plant shows a good load following characteristics as long as the void coefficient is slightly positive ($\sim +0.015$ $\delta k/k$/void fraction); and then it can be made to respond to rapid changes in the grid frequency under the speeder governer control.

Reactor control will be effected in two ways according to the preliminary designs; (a) total power control and (b) zonal power control. In a prototype reactor, of course, zonal power control is not necessarily required from the stability considerations. However, transient power distortions, caused partly by slowly damping spatial oscillations and partly by unexpected local disturbances in the core, will effectively be controlled by this system.

Various means of reactivity control were proposed in the preliminary designs, some based on the fairly proven technology and others either still in a development stage or just a concept. Further investigations will have to be made on these methods before the final design be made. Moderator poisoning, at present, is thought to be the most promising for a wide range of reactivity control. Since the need for booster in commercial power reactors depends both on the economics and on the required plant maneuverability, it will be necessary to assess the contribution of booster rather quantitatively.

2.3.3 Safety

One of the most important points in the safety consideration is to estimate the effect of positive void coefficient of reactivity in the primary cooling system accident. An accurate prediction of the discharge rate of steam-water mixture from the broken point will be essential to the estimation of the reactivity insertion rate. The computer code, LILAC, is now being developed specifically for this study[19]. Preliminary calculations with the code showed that the inverse coolant flow in the reactor channel, which will take place in the accident of the inlet header rupture, causes a rapid increase in the voids, leading to a rapid insertion of positive reactivity.

The primary cooling system will be divided into several independent loops to minimize positive reactivity insertion in a loss-of-coolant accident, as proposed in the preliminary designs.

The reactor scram may be achieved either by gravitational drop of control absorbers or injecting poison into absorber tubes; however, a certain backup system will be required from aseismatic considerations. Fast dumping of the moderator from the calandria tank is the most promising for this purpose. The emergency coolant injection system will provide sufficient cooling of fuel assemblies in a loss-of-coolant accident. An emergency condenser will also be installed to remove thermal power when the turbine plant is accidentally isolated from the reactor system.

The containment for the reactor system will possibly be based on the concept of pressure suppression, and the container will be provided with internal sprays.

3. Structural Design[8)9)10)11)12)]

3.1 Fuel assembly

Two types of the fuel assembly, containing 28 and 37 fuel rods respectively, were designed, and for the latter assembly hexagonal and circular arrangements were considered in the preliminary designs. The wrapping wires or small spacers putting on the cladding surfaces were adopted to give a specified gap between the fuel rods, and the fuel rods were tied up with bands at intervals of about 300 to 500 mm. Five types of the fuel assemblies were designed, based on which the full scale vibration test in two phase flow will be made in this fiscal year.

Some considerations were given for using rather short fuel assemblies[12)] and refueling often, so that higher fuel burnup might be expected. In this case, however, connection mechanism, plenum spaces, etc., will be put into the core and more difficult mechanism should be considered than in the case of single long fuel assembly in each channel. It may be said that the merits derived from refueling more often with short fuel assemblies will be reduced by the demerits mentioned above.

3.2 Refueling machine

It is an important problem to decide whether the refueling machine will be accessed to the reactor top or to the reactor bottom; this influences not only the refueling machine mechanism but also the nuclear plant arrangement. The preliminary designs were made for both cases, and in the case of refueling from the reactor top, some consortia proposed the shield pool just above the reactor.

Studies were made on the design of the plug seal and the connection seal between the pressure tube and the snout of the refueling machine. It was proposed in one of the preliminary designs[11] that the top of the snout wes semispherical and the end of the pressure tube was made conical, so as to facilitate the connection of the two. An effort was also made to make simple the mechanism of the refueling machine to eliminate troubles, encountered during operation, as much as possible.

3.3 Core structure

Zry-2 and Zr-Nb alloy were considered when designing the pressure tube. The calandria tube and pressure tube are usually designed to be separate from each other. However, one-body structure of the calandria tube and pressure tube was also proposed by a certain consortium[10], so that installation of the pressure tube into the calandria tank at the reactor site would be easily made.

About 5 mm of the gap was provided between the calandria tube and pressure tube and filled with carbon dioxide or kept vacuum for thermal insulation, whose characteristics were studied taking into account gamma heating.

In the preliminary designs, the rolled joint and tandem extruded joint methods were used for connecting the Zry-2 or Zr-Nb alloy pressure tube with the stainless steel pipe; and bellows were used for absorbing a thermal expansion of the pressure tube.

4. Future Program

The preliminary design of the prototype reactor was done. Based on the results, an initial detailed design of the prototype reactor of the heavy water moderated boiling light water cooled will be made in this fiscal year, and critical experiments and hydraulic and vibration tests of the fuel assembly will also be started.

On the other hand, large scale critical experiments, full scale heat transfer and hydraulic tests, components developments, reactor safety tests, fuel and material developments, etc. will be started next year in a large scale. The development of the design code, backed up by the experiments and theory, is also to be made. With the above mentioned developments and tests, the prototype reactor is scheduled to be operated from 1974.

Acknowledgement

Authors wish to express their gratitude to Mr. H. KATO, Vice-President of Kansai Electric Power Company, Mr. N. TANAKA, Executive Director of Tokyo Electric Power Company, and other people who have been concerned with this study and design.

References

1) FURUHASHI, A. and SHINDO, R.: "Nuclear Parametric Study of the Pressure Tube-type Heavy Water Moderated Lattices", JAERI 1080 (1964)
2) "The Summary of the Conceptual Design Studies on the Coolant Selection", JAERI-memo 1574 (1964)
3) YASUKAWA, S. and SHINDO, R.: "Uses of Plutonium Fuel in Pressure Tube-type Heavy Water Moderated Thermal Reactors", JAERI-memo 2527 (1966). Paper for Presentation at IAEA Symposium on "Uses of Plutonium as a Reactor Fuel" 13 to 17 March, 1967
4) YASUKAWA, S. and SHINDO, R.: "On the Utilization of Thorium in Pressure Tube-type Heavy Water

Moderated Power Reactor, Part 1" JAERI-memo 2278 (1966)

5) "Fuel Burnup and Void Survey for Heavy Water Moderated Boiling Light Water Cooled Reactor", Internal memo
6) "Burnup Characteristics for CANDU-BLW Reactor under Bidirectional Refueling", Internal memo
7) YASUKAWA, S.: "On Burnup Characteristics of Pressure Tube-type Heavy Water Moderated Power Reactors", JAERI 1075 (1965)
8) Hitachi Ltd.: "Preliminary Design of Heavy Water Moderated Boiling Light Water Cooled Reactor (Prototype)"
9) Nippon Atomic Industry Group Company Ltd.: "Preliminary Design of Heavy Water Moderated Boiling Light Water Cooled Reactor (Prototype)"
10) Mitsubishi Atomic Power Industry Inc.: "Preliminary Design of Heavy Water Moderated Boiling Light Water Cooled Reactor (Prototype)"
11) The First Atomic Power Industry Group: "Preliminary Design of Heavy Water Moderated Boiling Light Water Cooled Reactor (Prototype)"
12) Sumitomo Atomic Energy Industries Ltd.: "Preliminary Design of Heavy Water Moderated Boiling Light Warer Cooled Reactor (Prototype)"
13) "Burnup Characteristics by SKAT-4 Code for Ref. 15E and CANDU-BLW Reactors", Internal memo
14) "Thermal and Hydraulic Characteristics of Heavy Water Moderated Boiling Light Water Cooled Reactor", Internal memo
15) SHINODA, W. *et al.*: "Stability Analysis of Heavy Water Moderated Boiling Light Water Cooled Nuclear Power Station", JAERI-memo 2158 (1965)
16) "AURORA, A One-Dimensional Dynamics Programme for Boiling Water Cooled Reactors", Internal memo
17) "Dynamic Characteristics of Heavy Water Moderated Boiling Light Water Cooled Reactors", Internal memo
18) "CHAOS, A Three-Dimensional Xenon Dynamics Programme", Internal memo
19) "LILAC, A Digital Simulation of Loss-of-Coolant Accident", Internal memo