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Operating Experience of Shimane Nuclear Power Station and Core Management Engineering Systems

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ABSTRACT: The Shimane Nuclear Power Station of Chugoku Electric Power Company went into commercial operation on March 29, 1974 as the first Japanese-made commercial nuclear power plant. After scheduled operation, the plant was shut down on February 16, 1975 for the first annual maintenance and inspection outage. The plant capacity factor of 1974 fiscal year has been as high as 75.6% and fuel has maintained integrity for more than a year and a half since start-up. Described is the core management engineering system applied to the operating strategy for mitigating thermal duty of the fuel and to core management engineering including control rod pattern required for operation of the nuclear reactor.

INTRODUCTION

CONSTRUCTION of the Shimane Nuclear Power Station Unit 1 was started on February 11, 1970, and the unit went into commercial operation on March 29, 1974 as the first Japanese-made commercial nuclear power plant. The plant has been operating smoothly according to schedule without any unexpected accidental outage.

Fig. 1 shows the actual operating power, and Table 1 the operating results. In Fig. 1, the power changes during continuous operation indicate control rod sequence exchange and control rod pattern adjustment. The plant has been maintained smooth operation of a high plant capacity factor and maximum exposure of the fuel bundle has reached about 9,780MWD/T. From the fuel bundle inspection at the scheduled annual maintenance outage and from various operating data, it has been shown that the fuel has been operated with integrity.

These good operating results owe much to carefully planned operation based on core management engineering as well as to nuclear plant design, manufacturing and installation technologies. In constructing and operating this

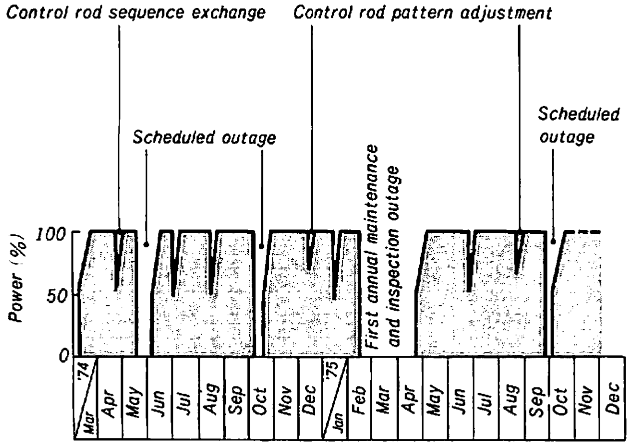

Fig. 1–Actual Operating Power.
Shown is the change in actual operating power since the commencement of commercial operation in March 1974.

TABLE 1. OPERATING RESULTS

Item	Unit	Fiscal year 1974 April 1, 1974 –March 31, 1975	First half fiscal year 1975 April 1, 1975 –Sept. 30, 1975
Generating time	Hours	6,941	3,765
Accumulated electric power	kWh	3.048×10^9	1.674×10^9
Average core increment exposure in specified period	MWD/T	4,750	2,635
Average core exposure at the end of specified period	MWD/T	5,608	7,358
Maximum fuel bundle exposure	MWD/T	6,876	9,776
Load factor (*1)	%	95.5	96.6
Plant capacity factor (*2)	%	75.6	82.8

*1 Load factor $= \frac{\text{Total power generated}}{\text{Operating hours} \times \text{Rated power}} \times 100(\%)$

*2 Plant capacity factor $= \frac{\text{Total power generated}}{\text{Calendar term} \times \text{Rated power}} \times 100(\%)$

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TABLE 2. SPECIFICATIONS OF NUCLEAR CORE AND FUEL

Basic specifications of the nuclear core and fuel are shown for a BWR plant with 460MWe rated electric power and 400 fuel bundles.

Rated electric power	**460MWe**
Rated thermal power	**1,380MWth**
Reactor pressure	**71.7kg/cm^2·a**
Total core flow	**21,770T/h**
Equivalent core diameter	**3,440mm**
Effective core height	**3,660mm**
Cell pitch	**152mm**
Number of fuel bundles	**400**
Thickness of clad	**0.9mm**
Outer diameter of clad	**14.5mm**
Pellet diameter	**12.4mm**
U weight	**78.2TU**
Number of control rods	**97**
Number of poison curtains (temporary use)	**172**
Maximum linear heat rate	**17.5kW/ft**
Minimum critical heat flux ratio	**1.9**

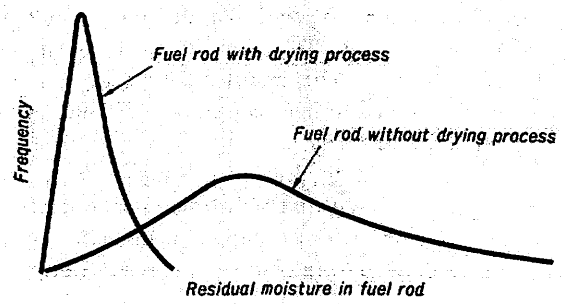

Fig. 2–Frequency Distribution of Residual Moisture in Fuel Rods. Residual moistures are compared between fuel rods which have and have not undergone the drying process.

power plant, we gave full play to the results of our basic research, to secure fuel integrity. Also, by applying core management engineering centering on a program system based on the nuclear thermal hydraulic design technology that has been developed for many years, we have been able to achieve sophisticated operation in a way to minimize the thermal duty of the fuel.

Described below are some of the technologies that have provided the foundation for successful operation of this plant, specifically fuel integrity which is closely related to core performance; operating strategy; and core management engineering which is indispensable to the operation.

NUCLEAR CORE AND FUEL SPECIFICATIONS

This nuclear power station is equipped with a boiling water reactor (BWR). Specifications of the nuclear core and fuel are presented in Table 2.

Reactor power is controlled through adjustment of core flow by means of a recirculation pump and control rod withdrawal operation.

FUEL INTEGRITY AND OPERATION STRATEGY

More than 20 years of research and development efforts have gone into BWR fuel on a worldwide scale. Thus there has been long operating experience with BWR fuel, which is now at a fully practical stage. Yet instances of clad damage, although at a low probability, have been reported. Cited as major causes may be:

(1) Local hydride formation on the inside surface of the clad.
(2) Pellet clad interaction.

Prevention of hydriding

Local hydride on the clad surface results from chemical reaction between the zircaloy of the cladding tube and the

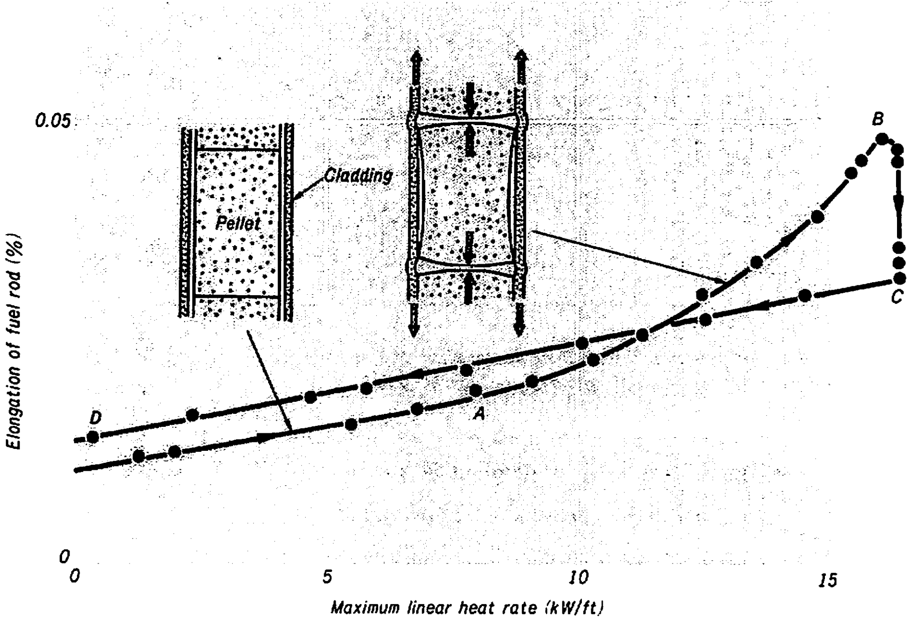

Fig. 3–Elongation Behavior of Fuel Rod. This is a result of a fuel rod irradiation experiment conducted in the Halden Boiling Water Reactor of Norway.

trace amount of moisture absorbed on the UO_2 pellet surface during fuel fabrication. The zirconium hydride thus produced is a brittle material and has a smaller density than zircaloy, so that volume increase is caused and this may lead to local failure of the clad.

After a thorough study of this failure mechanism and testing various countermeasures, we have introduced a hot vacuum drying process for the fuel rod, to eliminate residual moisture during fuel fabrication. Distribution of residual moisture in fuel rods with and without this drying process is shown in Fig. 2. With the introduction of this hot vacuum drying process, the possibility of local hydriding has been almost completely eliminated.

Prevention of pellet clad interaction

Pellet clad interaction as used here is a generic term covering all phenomena accompanying the expansion from inside the clad caused by the expansion of the UO_2 pellet contained.

Fig. 3 shows some of the results of an irradiation experiment—conducted in the Halden Boiling Water Reactor of Norway—on fuel rods of the same design as those used in the Shimane reactor. In Fig. 3, the 0-A-B part of the curve represents the period of increase of the maximum linear heat rate. Section B-C is where the heat rate is maintained constant, and C-D where it decreases. When the heat rate rises to point A, mechanical interaction between fuel pellet and clad is initiated. And up to point B, elongation of fuel rod progresses along with an increase in maximum linear heat rate. At B-C, while the heat rate is held constant, the fuel rod length shrinks in what is known as relaxation. When the heat rate is reduced, the fuel rod length decreases, with a slight plastic deformation remaining.

Fig. 4 shows the relaxation of fuel rod elongation, viewing the phenomenon in Fig. 3 time-wise. When the maximum linear heat rate increases, fuel rod elongation also increases, but when the heat rate is kept constant, fuel rod elongation decreases. When the heat rate is raised slowly as shown by the dotted line, elongation is relaxed during the rise, so that elongation is reduced as shown by the dotted line. These results indicate that the operating strategy described in a later section in this paper is effective in reducing fuel rod deformation, viz., in mitigating the mechanical interaction between fuel pellet and clad.

CORE MANAGEMENT ENGINEERING

In order to realize a method of operation that ensures fuel integrity, one needs detailed information on operational management engineering in addition to fuel manufacturing technology. The engineering for preparation of information required for nuclear reactor operation is referred to as core management engineering, which will be explained in the following.

Fig. 5 presents a core management engineering system required for the operation of a nuclear power station.

Core management program systems

Core management program systems to compute core management engineering information required for the operation of a nuclear power station are shown in Fig. 6. They consist of performance evaluation program systems and performance prediction program systems.

The performance evaluation program systems accumulate information on fuel types, installation location, basic computing constants as nuclear constants, exposure, isotopic composition, and operating data. Operating data are evaluated and stored, so that past operating conditions can be simulated.

The performance prediction program systems are made

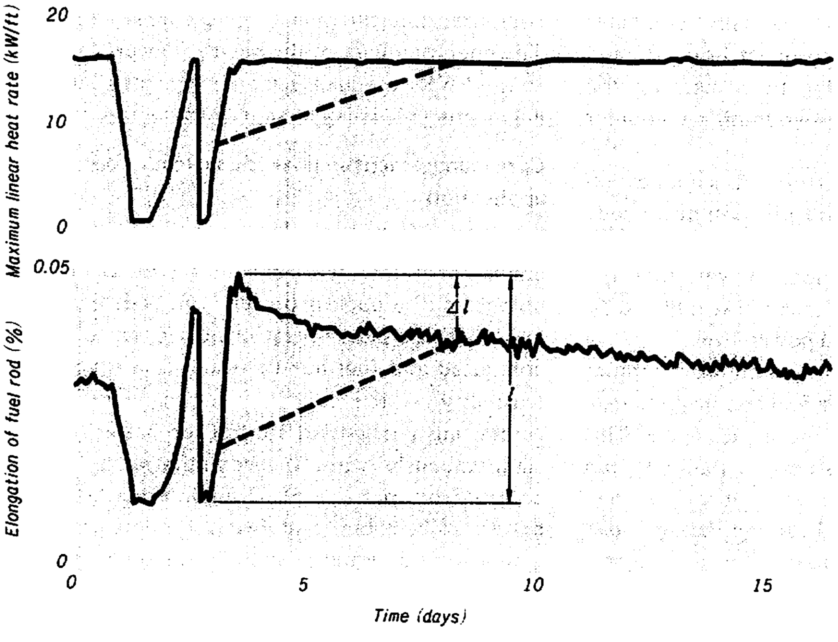

Fig. 4—Relaxation of Fuel Rod Elongation.
The result shown in Fig. 3 is presented in terms of time-wise change in maximum linear heat rate, to show the relaxation of fuel rod elongation.

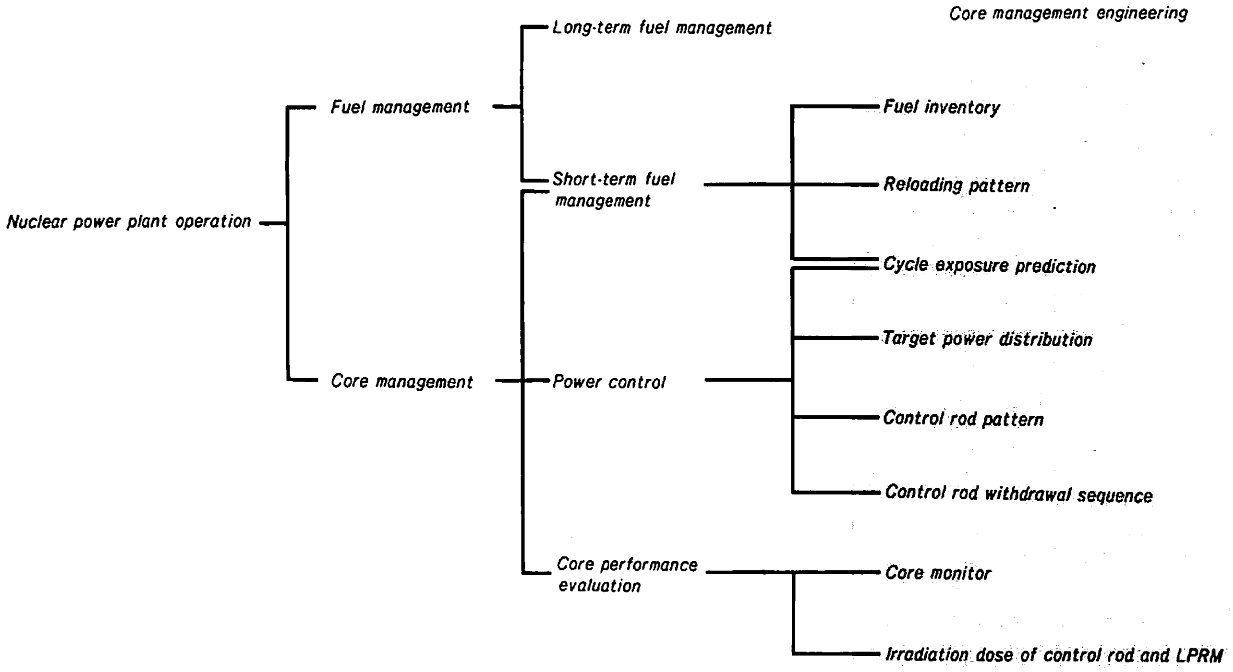

Fig. 5–Core Management Engineering for Nuclear Power Plant.
A system of core management engineering required for operating a nuclear power plant is shown.

up of a control rod pattern programming computer code, a fuel cycle reload pattern programming computer code, and a cold critical evaluation programming computer code. These systems compute engineering plans necessary for core management on the basis of the operating data evaluated and stored by the performance evaluation program system.

The steady state control rod pattern programming computer code must provide high computational accuracy and quick and timely response. With two optimizing algorisms this computer code automatically calculates control rod patterns with respect to linear heat rate and achieved exposure. Fig. 7 shows the flow chart for the steady state control rod pattern programming computer code.

The start-up sequence and control rod sequence exchange programming computer code is a computer code that includes xenon dynamic performance evaluation. It is used for computation of start-up sequence by withdrawal of control rod under a certain limited linear heat rate and by adjustment of core flow up to desired power level.

The fuel cycle reload pattern programming computer code is used for computing plans for fuel bundle discharge, shuffling, and loading at the end of a fuel cycle. This computer code, used for reload pattern programming, can simultaneously determine working procedures for fuel movement. Moreover, it can be used for computing reload pattern programming for multiple fuel cycles as well as a single fuel cycle, and can determine long-term balance of discharge-charge fuel.

The cold critical evaluation programming computer code is for cold critical evaluation of the cold condition of nuclear core.

The engineering plan computed by the BWR core management program systems is reviewed from the viewpoint of operational procedure, then applied to the nuclear power plant. The nuclear power plant is operated on the basis of the engineering plan determined, while being monitored and controlled by various instrumentation devices. Also, the plant is monitored by an on-line process computer, and the operating data are stored and analyzed. The operating data of the nuclear power plant are fed back to the BWR core management programs systems so that the engineering plan may be adjusted as required.

Core management engineering system's operational application

From the power station system operation plan, an annual operating plan including start-up and scheduled outage of the nuclear power plant is determined. On the basis of this annual operating plan, a reload pattern plan is computed and fuel bundle installation is determined. The feasibility of the annual operating plan is reviewed by computing a long-term control rod pattern plan. After, or simultaneously with, the computation of the long-term control rod pattern plan, an intermediate control rod pattern to be actually applied is computed. Then, a start-up and power-up sequence and control rod sequence exchange program are incorporated, to determine a control rod pattern plan. In determining a finally applied control rod

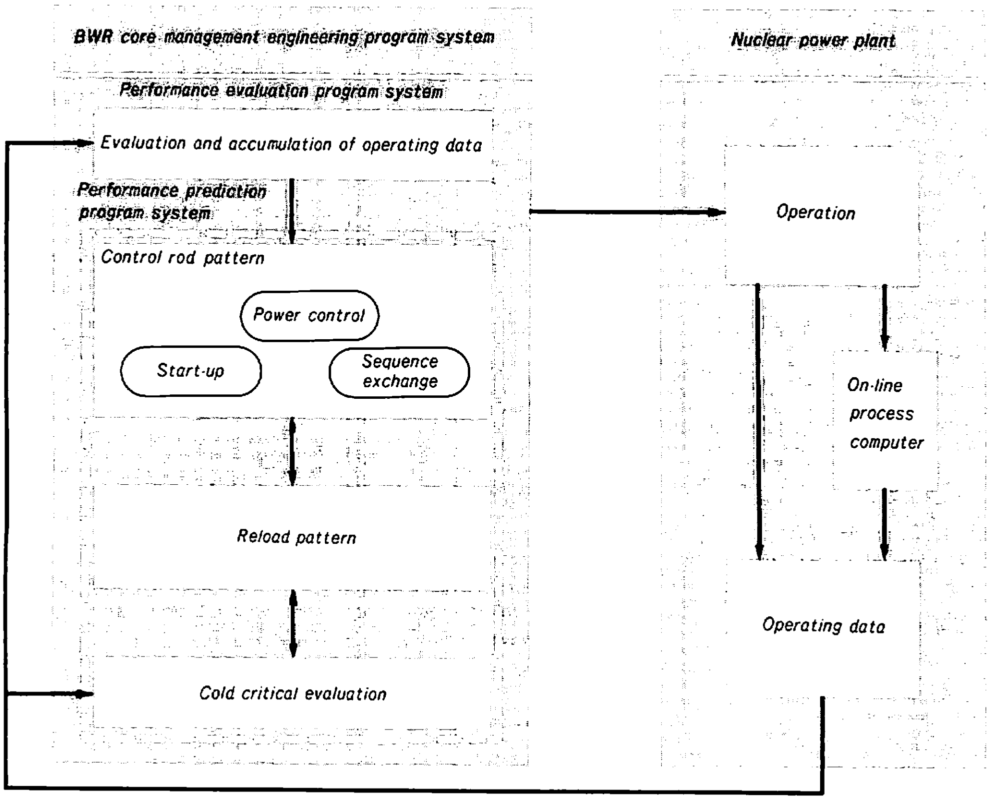

Fig. 6–Core Management Engineering System.
A core management engineering system required for nuclear power plant is illustrated, together with the flow of data in the system.

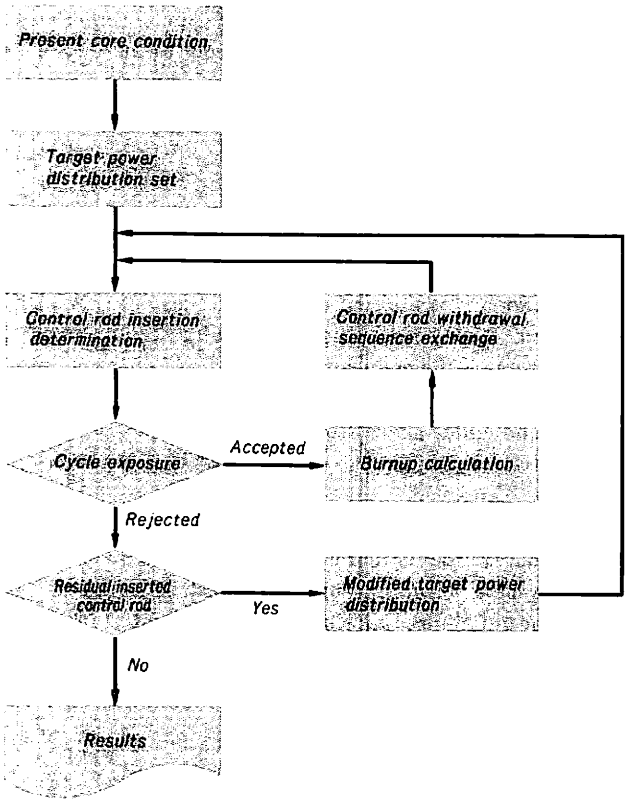

Fig. 7–Automatic Control Rod Pattern Determination Program Flow Chart.
An optimized control rod insertion determination flow chart for an automatic control rod pattern determination program is shown.

pattern plan, discussion of details is repeated between customer and manufacturer.

For the present nuclear power plant, the following operating strategy which was determined from the results described previously in this paper was applied:

(1) Flatten power distribution and decrease maximum linear heat rate.
(2) Perform control rod withdrawal at a low linear heat rate, and control power by core flow adjustment at a higher linear heat rate.

A detailed control rod pattern matched with operating conditions was computed and operation was carried out according to the strategy listed above. For the appropriate application of control rod patterns, much depends upon cautious and flexible operation technology for a nuclear power plant.

Fig. 8 shows an instance of power-up from cold condition after fuel loading. The control rod is withdrawn at low linear heat rates. At higher linear heat rates, power is increased by core flow adjustment. In this way, operation can be performed at a low linear heat increase rate for the fuel rod. Fig. 9 shows some actual results of maximum linear heat rate. Operation can be performed at a low maximum linear heat rate, and an adequate thermal margin can be secured against the design limit. Flattening of power distribution throughout the operating term will increase the thermal margin.

Some operating data are shown in Table 1, which shows that fuel integrity has been maintained and a high plant capacity factor achieved through the operating term.

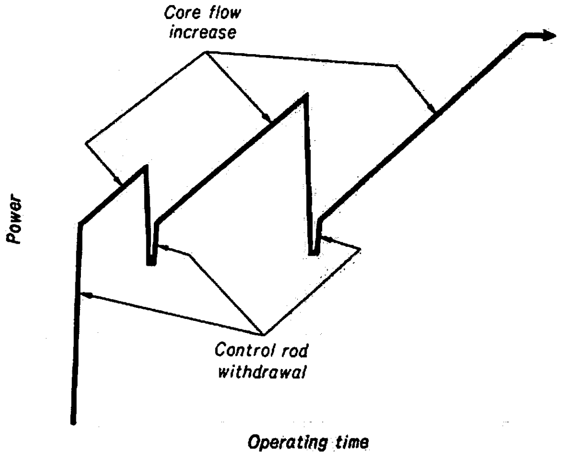

Fig. 8–Start-up from Cold Condition After Reloading.
At low linear heat rates, the control rod is withdrawn, and at high linear heat rates, power is increased by core flow adjustment. In this power-up pattern, the thermal duty of the fuel can be reduced.

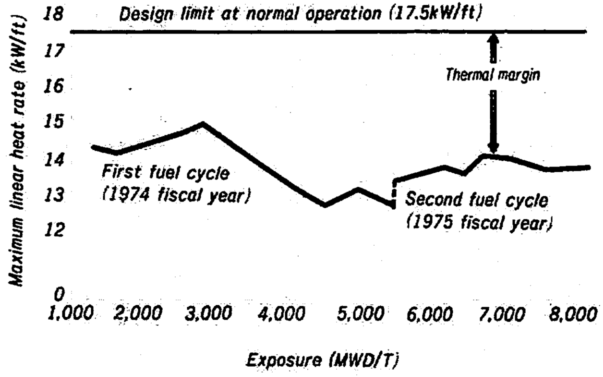

Fig. 9–Maximum Linear Heat Rate.
The maximum linear heat rates in the first and second fuel cycles are shown, together with their thermal margin against the design limit at normal operation.

CONCLUSIONS

This nuclear power station is the first Japanese-made commercial BWR power plant. Attention was focused on core management following the start of commercial operation. In February 1975, the plant underwent its first annual scheduled maintenance and inspection outage. The first reloading was performed then, and the plant went into its second year of operation.

As for core management engineering, an appropriate operating plan was carried out by applying core management engineering systems through the cooperation of customer and manufacturer. Concerning operational strategy, operating procedures that mitigate thermal duty of the fuel rod were established, and operation was carried out on the basis of detailed analytical prediction.

The valuable engineering experience gained through this work is thought to contribute immensely to the technical improvement of boiling water reactors.

The authors wish to express their sincere thanks to those who have provided their generous cooperation in this project.

ABSTRACTS FROM HITACHI HYORON

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Evaluation of Transient Test Results of Shimane Nuclear Power Station Unit 1, Chugoku Electric Power Company

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The Shimane Nuclear Power Station of Chugoku Electric Power Company is the first boiling water reactor (BWR) plant that has been constructed with a Japanese manufacturer as the main contractor.

Start-up tests for the new power station began on May 1, 1973 and ended successfully on March 29, 1974. The start-up tests highlighted abnormal operational transient tests.

This paper first defines and classifies the abnormal operational transients that are reasonably anticipated to occur, by considering the principal construction of the plant. The guidelines used for the selection of test items are also presented.

Secondly, typical test results are discussed. They include the following:
(1) Closure of all MSIV's.
(2) Turbine trip.
(3) Load rejection.
(4) Trip of one feedwater pump.
(5) Trip of two recirculation MG sets.

These tests proved the inherent safety and continuous