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STATIC CORE PERFORMANCE SIMULATOR SCOPERS-2 FOR LIGHT WATER REACTORS AND ITS APPLICATION

(Paper submitted to the NEACRP 21th Meeting, Tokai, November 1978)

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Takanori SHIMOOKE, Masafumi ITAGAKI* and Masao OSANAI*

日本原子力研究所
Japan Atomic Energy Research Institute

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Takanori SHIMOOKE, Masafumi ITAGAKI*
and Masao OSANAI*

Reactor Safety Evaluation Laboratory,
Tokai Research Establishment, JAERI

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SCOPERS-2 is a generalized FLARE-type computer program simulating both PWR and BWR. Features and the calculation model (generalized FLARE-type nodal equation, migration kernel, etc.) are first described.

A calculation is then given of the core of nuclear ship MUTSU (PWR) for an example of the code application. The power distribution calculated by SCOPERS-2 and by CITATION (3-dimensional diffusion code) are in good agreement.

Keywords: One-Group, Coarse-Mesh, Three-Dimensional Calculation, Neutron Balance Theory, Response Matrix Theory, Nuclear-Thermal Coupling, LWR, Burnup, FLARE Code, SCOPERS-2 Code

* Japan Nuclear Ship Development Agency

軽水炉核特性シミュレーション計算コードSCOPERS－2とその応用

日本原子力研究所東海研究所安全解析部
下桶敬則・板垣正文 * ・小山内正夫 *

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SCOPERS－2は一般化されたFLAREモデルに基づく炉心核特性計算コードで，PWRおよびBWRの両原子炉を扱うことができる。

この報告書では，SCOPERS－2コードの特徴および計算モデルを簡単に紹介した後，その応用の一例として，原子力船「むつ」炉心（PWR）に適用した計算の方法とその結果を示す。SCOPERS－2で計算された炉心内出力分布は三次元拡散計算コードCITATIONの結果と良く一致した。

* 日本原子力船開発事業団

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1. Introduction

SCOPERS-1 (an earlier version of SCOPERS-2) program was written by one of the authors (T. Shimooke) for Japan Nuclear Ship Development Agency in March, 1972. This program was referred to in the report, JAERI-M 5805.[1] Later, its modification was made by the same author at JAERI which is called SCOPERS-2.

SCOPERS-2 (and SCOPERS-1) was developed to be one of the successors of the FLARE, a BWR simulation program given by D.L. Delp et al.[2]

A generalized FLARE-type nodal equation is used in SCOPERS-2 together with the nodal-form neutron migration kernel which is derived by a relatively rigorous way and replaces the empirical kernel used in the FLARE. Moreover, an effort was made to expand the flexibility of the original FLARE program. The resultants are reliable theoretical basis for the program model and the application to both types of Light Water Reactors, each being in wider options than before.

Present report describes the code performance, the physical model used in SCOPERS-2, and the result of an application to the Nuclear Ship MUTSU reactor (one of pressurized water reactors).

2. General Description of the Code

2.1 Code Performance

SCOPERS-2 is a computer program that can calculate a nodal power distribution in a steady state for a three-dimensional, xyz, core geometry, and core reactivity both of a boiling water reactor and a pressurized water reactor. SCOPERS-2 also can describe an axial void distribution (for BWR) or a coolant temperature distribution (for PWR) at any fuel positions in the core, which are compliance with the power distribution. SCOPERS-2 can be used, moreover, to perform a burnup calculation and to predict a three-dimensional exposure distribution and the core-total burnup obtained.

SCOPERS-2 is designed to perform these calculations by a rather simple physical model. A neutronic balance is solved by a "generalized FLARE-type nodal equation." In the framework of a coarse mesh representation of a xyz core geometry, the equation describes the one-group representative Ψ that is essentially proportional to a fission rate (called hereafter neutron source), in terms of an infinite multiplication constant k_∞ and a "migration kernel" $K_{i\to j}$. An additional simplification involves the replacement of the reflector by an albedo at the core surface so that only mesh points within the active fueled region are considered. In SCOPERS-2, a reactor core can be described by e.g. one mesh point horizontally at the center of each fuel element, and about a dozen of points vertically. This small number of mesh points and the one-group representation of neutrons allow us to have a fast throughput for a three-dimensional core calculation.

The neutron source Ψ at each node is calculated as a function of k_∞ at the node, the sources at the eighteen neighboring nodes, and a "migration kernel" $K_{i\to j}$ which is a measure of the probability that a neutron born at node i is absorbed at node j. This migration kernel is a given function of the noding space L_x, L_y, L_z, the Fermi age τ of thermal neutron and the thermal neutron diffusion area $1/\kappa^2$. These age and area are calculated at each node based on a fit to the moderator void ratio or to the moderator temperature. The infinite multiplication constant is also given by a simple fitting equation which includes the following effects:

(a) Presence (or absence) of a control rod or a rod-cluster.

(b) Local moderator void ratio or temperature.

(c) Power-dependent equilibrium xenon and Doppler reactivity.

(d) Local fuel exposure.

The void (or coolant temperature) calculation consists of a determination of the average steam quality (or coolant enthalpy) at each node based on the inlet mass flow rate, inlet enthalpy, and the power integrated from the bottom of the channel to the node of interest. The void (or coolant temperature) is calculated by a numerical fit to a void-quality (or temperature-enthalpy) correlation curve. The inlet mass flow rate is calculated as a simple function of the channel power.

The effects of the loss in reactivity resulting from fuel burnup are included by a simple fit of k_∞ to exposure, exposure-weighted voids, and exposure-weighted local control. The data from which these fits are generated must come from independent, complex calculations, or experimental data.

SCOPERS-2 program keeps many portions of FLARE program which are still valid to use. Therefore, SCOPERS-2 owes FLARE for program resources like basic subroutines and the logic structure between them. On the otherhand, many modifications and improvements are also included in SCOPERS-2.

Main improvements on the physical model are:

(1) One-group, coarse mesh, neutron balance equation is completely revised from the semi-empirical one to the generalized FLARE-type nodal equation that has a firm foundation.

(2) The nodal interaction is considered between a node and the neighbouring 18 ones. In the old program only 6 neighbouring nodes were considered.

(3) The migration kernel used in the program is given beforehand by analytically integrating the point migration kernel by six-multiple space variables so that there is no need to input any adjusting parameter for a kernel as requested by FLARE.

(4) Albedo for the reflector effect is given by the program.
A user can input only nuclear constants λ, D, κ and τ_e for top, bottom and peripheral reflectors.
The effect on albedo of geometries of reflector is also considered automatically by built-in logic in the program, so that the user is free from adjusting input of albedo as was at FLARE.

Main improvements on the code flexibility are:

(5) A PWR as well as a BWR can be calculated now under a consistent

model and logic.

(6) The relative allocation of control rods to the core nodal assignment is enlarged from four patterns in original to seven for input modeling.

(7) The cross sectional core symmetry condition in combination with the boundary condition at an axis of symmetry are of eleven varieties for code users (It was six at FLARE).

(8) A critical search option has been completed in the code; a rod-withdrawal search for BWR and a boron-concentration search for PWR.

SCOPERS-2 can make any of the following calculations for BWR and PWR. Each of these are one calculation which can be performed by one submission of the job if a combination of options is properly selected by the user.

<u>For BWR, SCOPERS-2</u>

1) calculates the three-dimensional power distribution within the core under any of the control rod pattern at an arbitrary reactor power (Power Distribution Calculation).

2) calculates the three-dimensional void distribution in moderator under any of the control rod pattern at an arbitrary reactor power (Void Calculation).

3) calculates the critical withdrawal patterns and lengths of control rods for a given reactor power (Critical Search).

4) calculates the (thermal) power output to be performed by a reactor in compliance with the control rods patterns given (Power Search).

5) calculates the final burnup attained by each fuel element loaded in the core, taking into account the changes of power distribution with the various control patterns encountered during the course of reactor operation (Burnup Trace).

6) takes into consideration the fuel shuffling and the removal of poison curtains during the calculation described in 5) (Burnup Trace with Fuel-Shuffling).

7) predicts the final core burnup to be attained by a reactor, calculating the power and exposure distributions at each burnup step

for which the control rods positioning is searched automatically in accordance with a given priority for the withdrawal, to maintain the power level against the exposure (Critical-Exposure Iteration).

8) performs the above-stated calculation 7) with the predetermined plan on the fuel shuffling (Critical-Exposure Iteration with Fuel-Shuffling).

For PWR, SCOPERS-2

1) calculates the three-dimensional power distribution within the core at any reactor power under any control program including the part length rod-cluster and boron dilution in the moderator (Power Distribution Calculation).

2) calculates the coolant temperature distributions along the core height at any location of the core under any control program and reactor power (Coolant Temperature Calculation).

3) determines the position of the regulating rod-clusters for a given reactor power and boron concentration (Critical Search).

4) determines the concentration of boron for a given reactor power and position of rod-clusters (Boron Search).

5) calculates the (thermal) power output to be performed by a reactor in compliance with the rod-clusters position and the boron concentration given (Power Search).

6) calculates the final burnup attained by each fuel element loaded in the core, through changes of power distribution affected by the rod-clusters position and the boron dilution during the course of reactor operation (Burnup Trace).

7) takes into consideration the fuel shuffling during the calculation described in 6) for PWR (Burnup Trace with Fuel-Shuffling).

8) predicts the final core burnup to be attained by a reactor, calculating the power and exposure distributions at each burnup step for which the rod-clusters position or the boron concentration is searched automatically to maintain the power level against the exposure (Critical-Exposure Iteration).

The present version of the code will handle a maximum array of 15 × 15 × 20 nodes with mirror, or diagonal, or 90°/180° rotational symmetry. A full-core, half-core, or quarter-core representation may be run with either symmetry and for cores with or without a central fuel bundle. It needs a core memory of 127 kW and File 5 for input, 6 for output and 7 for card punch output. By utilizing a mirror symmetry with a quarter core, a 32-bundle Nuclear Ship Mutsu reactor core can be represented by 432 mesh points; 4 mesh points in each bundle with 12 points along height. Typical CPU time on FACOM 230-75 is 14 second for a solution of this problem with converged power and moderator temperatures but fixed control rod position and burnup.

2.2 Physical Model

(1) Nodal Equation for Source

Presently starting from the general nodal equation derived by Z. Weiss[4)], which was based on the response matrix theory, it will be shown that the degeneration of thermal and epithermal neutrons into one-energy-group under appropriate assumptions in coarse node system results in a generalized FLARE-type nodal equation that includes the original FLARE equation as an extreme example of the application.

Being with the two neutron-energy-groups in the Weiss theory in one-dimension, assumptions are made on the partial incident-neutron current components $j_i^{(1)}$ and $j_i^{(2)}$ of node i for group 1 and 2. It is supposed, for example, that epithermal neutrons incident by $j_i^{(1)}$ into node i complete slowing down within the node and will be all absorbed during diffusion in the same node. Once fission occurs after absorption, the multiplied neutrons will partially leak from node i to the adjacent nodes i-1 and i+1 as an epithermal group. On the contrary, thermal neutrons incident are not transmitted directly, or are reflected in a small amount which can be neglected because the reflected neutrons cannot transmit the internal informations of the reflecting node. The epithermal and thermal neutrons are thus treated in an asymmetric way, which is a feature of the present model.

The final form of the equation derived for the fission rate multiplied by η_{th}, i.e. for $\bar{\Psi}_i \equiv \eta_{th}^i \bar{k}_\infty^i J_i^{(1)}$ of node i is,

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$$[1 - k_\infty^i(1 - \sum_m \eta^i K_{i\to m} - \frac{\eta_t^i}{\eta_{th}^i} K_S^i)]\bar{\Psi}_i = k_\infty^i \sum_m \eta^m K_{m\to i} \bar{\Psi}_m \ , \tag{1}$$

where $\eta^i \equiv \eta_t^i/\eta_{th}^i + \eta_e^i/(\bar{k}_\infty^i \eta_{th}^i)$.

$\eta_e^i K_{i\to m}$ and $\eta_t^i K_{i\to m}$ are fractions of the migration kernel, $K_{i\to m}$, for epithermal and thermal neutron, respectively. The factors η_e^i and η_t^i are always normalized so that $\eta_e^i + \eta_t^i = 1$. η_{th}^i is the probability that a fission neutron is thermalized in the native node where it is born. k_∞^i and $\bar{k}_\infty^i$ are the infinite multiplication factors for the node i with and without resonance escape probability correction, respectively. K_S^i is given as $\sum_m K_{i\to m}$. Strictly, the equation is derived in one-dimension because the original Weiss equation is not proved rigorously for more than one-dimension. In the application hereafter, however, it is supposed valid also for two- and three-dimension.

The equation (1) is reduced to the original FLARE equation when $\eta_t \simeq 0$ and $\eta \simeq 1$. This implies physically that

(a) the unit node in consideration is relatively small ($\to \eta_t \simeq 0$), and

(b) there is a condition that $(\nu\Sigma_f/\Sigma_a)\cdot\eta_{th} \simeq \eta_e \simeq 1$ ($\to$ the node itself is just or nearly critical in neutron multiplication).

It must be noticed that these two conditions (a) and (b) are never satisfied simultaneously in a slightly enriched uranium-fueled thermal reactor.

(2) Migration Kernel from Node to Node

Migration kernels are shown in Table 3.1 in the form of equation which are used for calculation in the code. These are the probabilities that a neutron born in a node be absorbed in one of the neighbouring nodes during it's migration. They are derived by integrating a point migration kernel analytically by using the method described in the report, JAERI-M 5805[1].

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Table 2.1 A Summary of the Kernels Derived.

Direction	Nodal Migration Kernel
for the x- or y-direction	$K_{xx} \equiv \frac{q}{V}\sqrt{\frac{\tau}{\pi}}\left[\left\{L_y L_z - \frac{8}{3\pi} L_x (L_y + L_z) + \left(\frac{67}{12\pi} - 1\right) L_x^2\right\} \cdot A_1(L_x) + \left\{\frac{4\alpha}{3\pi}\frac{L_y + L_z}{L_x} - \left(\frac{67}{12\pi} - 1\right)\right\}\tau \cdot A_2(L_x)\right]$
for the z-direction	$K_{zz} \equiv \frac{q}{V}\sqrt{\frac{\tau}{\pi}}\left[\left\{L_x L_y - \frac{8}{3\pi}(L_x + L_y) L_z + \left(\frac{67}{12\pi} - 1\right) L_z^2\right\} \cdot A_1(L_z) + \left\{\frac{4\alpha}{3\pi}\frac{L_x + L_y}{L_z} - \left(\frac{67}{12\pi} - 1\right)\right\}\tau \cdot A_2(L_z)\right]$
for the diagonal direction in x-y plane	$K_{xy} \equiv \frac{1}{2}\frac{q}{V}\sqrt{\frac{\tau}{\pi}}\left[\left\{\frac{8}{3\pi} L_x L_z - \left(\frac{67}{12\pi} - 1\right) L_x^2\right\} \cdot A_1(L_x) + \left\{-\frac{4\alpha}{3\pi}\frac{L_z}{L_x} + \left(\frac{67}{12\pi} - 1\right)\right\}\tau \cdot A_2(L_x)\right]$
for the diagonal direction in x-z plane	$K_{xz} \equiv \frac{1}{2}\frac{q}{V}\sqrt{\frac{\tau}{\pi}}\left[\left\{\frac{8}{3\pi} L_x L_z - \left(\frac{67}{12\pi} - 1\right) L_z^2\right\} \cdot A_1(L_z) + \left\{-\frac{4\alpha}{3\pi}\frac{L_x}{L_z} + \left(\frac{67}{12\pi} - 1\right)\right\}\tau \cdot A_2(L_z)\right]$
Notes	$V = L_x L_y L_z$, $\alpha = 1.1681$, $A_1(L_x) = $ Eq. (46), $A_2(L_x) = $ Eq. (48) $K_{11} = 1 - 4K_{xx} - 2K_{zz} - 4K_{xy} - 8K_{xz}$ $q \equiv \dfrac{\left\{\frac{2}{R_0} f\left(\frac{R_0}{2\sqrt{\tau}}\right) + \frac{R_0(2\kappa^2\tau + 1)}{(2\kappa\tau)^2 - R_0^2}\right\} e^{-R_0^2/4\tau}}{\left\{\frac{2}{R_0} f\left(\frac{R_0}{2\sqrt{\tau}}\right) + \frac{R_0(2\kappa^2\tau + 1)}{(2\kappa\tau)^2 - R_0^2}\right\} e^{-R_0^2/4\tau} - \left\{\frac{2}{R_1} f\left(\frac{R_1}{2\sqrt{\tau}}\right) + \frac{R_1(2\kappa^2\tau + 1)}{(2\kappa\tau)^2 - R_1^2}\right\} e^{-R_1^2/4\tau}}$ $R_0 = 0.3582\sqrt{L_x^2 + L_y^2 + L_z^2}$, $R_1 = R_0 + L_x$

3. Application to MUTSU Reactor

A PWR as well as a BWR can be taken into consideration by SCOPERS-2. Now we have made some calculations for the Nuclear Ship MUTSU reactor as an example of the code application to PWRs.

The MUTSU reactor is a pressurized water reactor in a small size with the net output of 36 MWt for use of nuclear-propulsion of 8200 gross tonage Nuclear Ship MUTSU. The MUTSU reactor core is composed of 32 fuel assemblies as illustrated in Fig. 3.1. To have a flat power distribution, the 12 inner regions are of 3.24 w/o enrichment in U-235 and the outer 20 in the surrounding of 4.44 w/o. Each fuel assembly is composed of a 11 × 11 square lattice of 112 fuel rods and 9 burnable poison rods (BPR) as shown in Fig. 3.2. The fuel rod is made by inserting UO_2 pellets in the stainless-steel cladding tube of outer diameter 10.53 mm and active length 1040 mm. The burnable poison rod is also supplied by inserting boron-silicate glass into the same-size tube as for a fuel rod.

Vertical configuration of the core is illustrated in Fig. 3.3. The poison part of a BPR is only 625 mm long, six-tenth of the full core length. Therefore one can consider three axial regions in the core; the upper region in which there is no poison, the BPR region in which there is poison effect and the lower region of no poison. The fuel assembly is bound by a clip (38 mm axial width) at three axial locations between a top and a bottom tie-plate. The control rods are cruciform and made of the Ag-In-Cd alloy absorber and the zircaloy-2 follower. Reactivity of the reactor can be controlled only by the control rods through the reactor life.

The following is a summary of our procedure for the MUTSU power distribution calculation: Fermi age, diffusion constant and diffusion length were obtained for the migration kernels in the SCOPERS-2 model, by using LEOPARD, a cell calculation code. The k_∞ was determined for assembly-cells by a diffusion calculation. With these inputs, the SCOPERS-2 was applied to a quarter core of MUTSU reactor. It is modeled by nodes which are 8.98 cm(X) × 8.98 cm(Y) × 10.4 cm(Z) in size. Finally, the albedo-parameters in the model were adjusted to make the power distribution and core reactivity produced by the SCOPERS-2 consistent with that by the 3-dimensional CITATION calculation in the state of hot zero power (HZP).

The SCOPERS-2 calculation was then carried out for the hot full power condition (HFP) by using the albedo-parameters determined in the above-mentioned HZP calculation. The hot full power state of a reactor has the intrinsic coupling between nuclear and thermo-hydraulic phenomena, and can not be treated usually by a simple diffusion code.

3.1 Method

We here explain the calculational procedures in detail.

(1) Hot Zero Power Calculation

Fermi age, diffusion constant, and diffusion length were obtained by LEOPARD (a cell calculation code) in order to have the migration kernels. For example, Fermi ages and diffusion areas in the fuels are shown in Table 3.1 and 3.2. These parameters are calculated for both clip part and no-clip part, and averaging these two with their weights, the desired constants were obtained for the axially devided nodes in SCOPERS-2 calculation.

The k_∞ can be obtained also by the LEOPARD calculation, but the LEOPARD results are not used for our purpose, as the values obtained from 2-dimensional diffusion calculations happen to be available for the k_∞. Moreover, it is difficult to estimate the effects of the "extra" region in a cell model in LEOPARD calculation. Thus, the k_∞ used was one determined for an assembly-cell by the diffusion calculation code CITATION. These k_∞s are listed in Table 3.3 and 3.4. The k_∞s were also averaged for the nodes in the SCOPERS-2 calculation with weighting for the clip part and no-clip part.

By using these input, the SCOPERS-2 was applied to a quarter core of MUTSU reactor which is modeled by nodes, each of 8.98 cm × 8.98 cm × 10.4 cm in size (see Fig. 3.1 - 3.3 for the core size).

The specific parameters of this code, η_e, η_t, η_{th}, and K_s are input as follows.

$$\eta_e = 1.0$$
$$\eta_t = 0.0$$
$$\eta_{th} = 0.25$$
$$K_s = 1.0$$

The albedo-parameters; λ, D, κ, and τ_e were adjusted to make the power distribution produced by the SCOPERS-2 consistent with that of 3-dimensional CITATION calculation in hot zero power (HZP). The albedo-parameters were finally adjusted as follows.

Top Reflector

$\lambda^t = 3.0$ cm $D^t = 0.325$ cm $\kappa^t = 0.3068$ cm $\tau_e{}^t = 36.0$ cm^2

Bottom Reflector

$\lambda^b = 3.0$ cm $D^b = 0.275$ cm $\kappa^b = 0.2596$ cm $\tau_e{}^b = 36.0$ cm^2

Peripheral Reflector

$\lambda^p = 3.0$ cm $D^p = 0.585$ cm $\kappa^p = 0.3068$ cm $\tau_e{}^p = 36.0$ cm^2

(2) Hot Full Power Calculation

In the hot full power condition, the albedo-parameters and the zero-order coefficient of the k_∞ fitting formula are equal to those of the hot zero power condition. The coolant flow rate was assumed to be uniform in channels according to the experimental results obtained for MUTSU reactor. Xenon, Doppler, and moderator temperature reactivity coefficients were obtained by the LEOPARD cell calculations, and these are applied to the fitting functions of the SCOPERS-2. With these input, hot full power calculation was then carried out by using the thermo-hydraulic fitting function of this code. The procedure of the SCOPERS-2 calculations is shown in Fig. 3.4 for quick reference.

3.2 Results

The following presents the results of the SCOPERS-2 calculations for the HZP and HFP conditions of the MUTSU core.

(1) Hot Zero Power Calculation

In Fig. 3.5 is given a comparison of the coarse-mesh axial power distribution by the SCOPERS-2 with a fine-mesh solution by 3-dimensional CITATION for HZP critical condition. In this figure and the following, the axial power distribution is given by averaging individual axial distribution for all fuel-assemblies in the core. Both the power distributions are normalized so that the core average should be 1.0. The power distribution and the effective multiplication factor by the SCOPERS-2 are shown to be in good agreement with those by the CITATION. It is also clear that SCOPERS-2 can predict the power peak at the correct location near the center and the power increase in the top of the core where the part-length BPR terminates. On the other hand, it is due to a coarse-mesh model that SCOPERS-2 fails to describe the local dip of power caused by a clip.

The assembly-wise enthalpy-rise distribution is shown in Fig. 3.6, and the assembly-wise horizontal power distribution in the four typical plane are shown in Fig. 3.7 for both the SCOPERS-2 and the CITATION. In these figures the values of power distribution are all normalized so that the core average should be 1.0. Discrepancy between two calculations are 4 % in the hottest channel, 10 % in a corner and 6 % in a controlled for channel enthalpy-rise.

(2) Hot Full Power Calculation

The SCOPERS-2 calculation was also carried out for a typical pattern of control-rod in full power operation, i.e. G4, G3, G2 control rod groups being full out and G1 group partially inserted. This G1-group control rod position was searched by using the critical search option of this code so that the neutron multiplication factor should be 1.0.

The axial power distribution is shown in Fig. 3.8. The assembly-wise enthalpy-rise distribution and the assembly-wise horizontal power distribution in the four typical plane sections are shown in Fig. 3.9 and Fig. 3.10, respectively.

3.3 Prediction Potential of SCOPERS-2

Once the output of SCOPERS-2 is calibrated by a few number of measured values or by a more detailed calculation for a reactor concerned, then it will work to give a quick answer for a wide variety of the reactor conditions and to predict what the extended state of the reactor looks like. In the case here reported, the 3-dimensional CITATION calculation for HZP condition is used to calibrate the albedo-parameters in SCOPERS-2 model. This procedure is described in Section 3.1. And the calibrated results are shown in Section 3.2 to be close enough to the reference data. This agreement is, in some sence, not an extraordinary matter and does not necessarily give a positive vote for SCOPERS-2, because it is the adjusted results. Anyway the hot full power calculation was made, considering the nuclear thermo-hydraulic interaction for HFP condition, by the SCOPERS-2 thus calibrated to HZP state, and the result described in Section 3.2 was obtained in 1976. A year and half later, a different project has been completed to analyse the MUTSU core by using another 3-dimensional diffusion code STEADY-SHIP which has the nuclear thermo-hydraulic coupling loop in it. Moreover, the STEADY-SHIP calculation uses the macroscopic nuclear constants newly obtained by using LEOPARD (for rod cell and for structure material), ANISN (for strong absorber) and 2-D CITATION (for fuel assembly). Simply speaking, this latest analytical work for MUTSU core differs in terms of the method, the code and the nuclear data used, from the previous one used for the calibration of SCOPERS-2 model. Therefore it will provide a good and real test for the SCOPERS-2 prediction ability. The followings are the results:

(1) Criticality or Control-Rod Withdrawal Length at HFP

SCOPERS-2	G1 group, 285 mm out	Difference
Independent 3-D Calc.	G1 group, 228 mm out	57 mm

The difference is only 6 cm which is equivalent to 0.35 % $\Delta k/k$ for MUTSU core.

(2) Enthalpy-Rise Ratio in Channels at HFP

	hottest channel	controlled	corner
SCOPERS-2	1.473	1.174	0.595
Independent 3-D Calc.	1.415	1.094	0.696
Difference	4 %	7 %	14 %

Difference between the two calculations is as small as that at HZP for the hottest channel and slightly larger than for other channels.

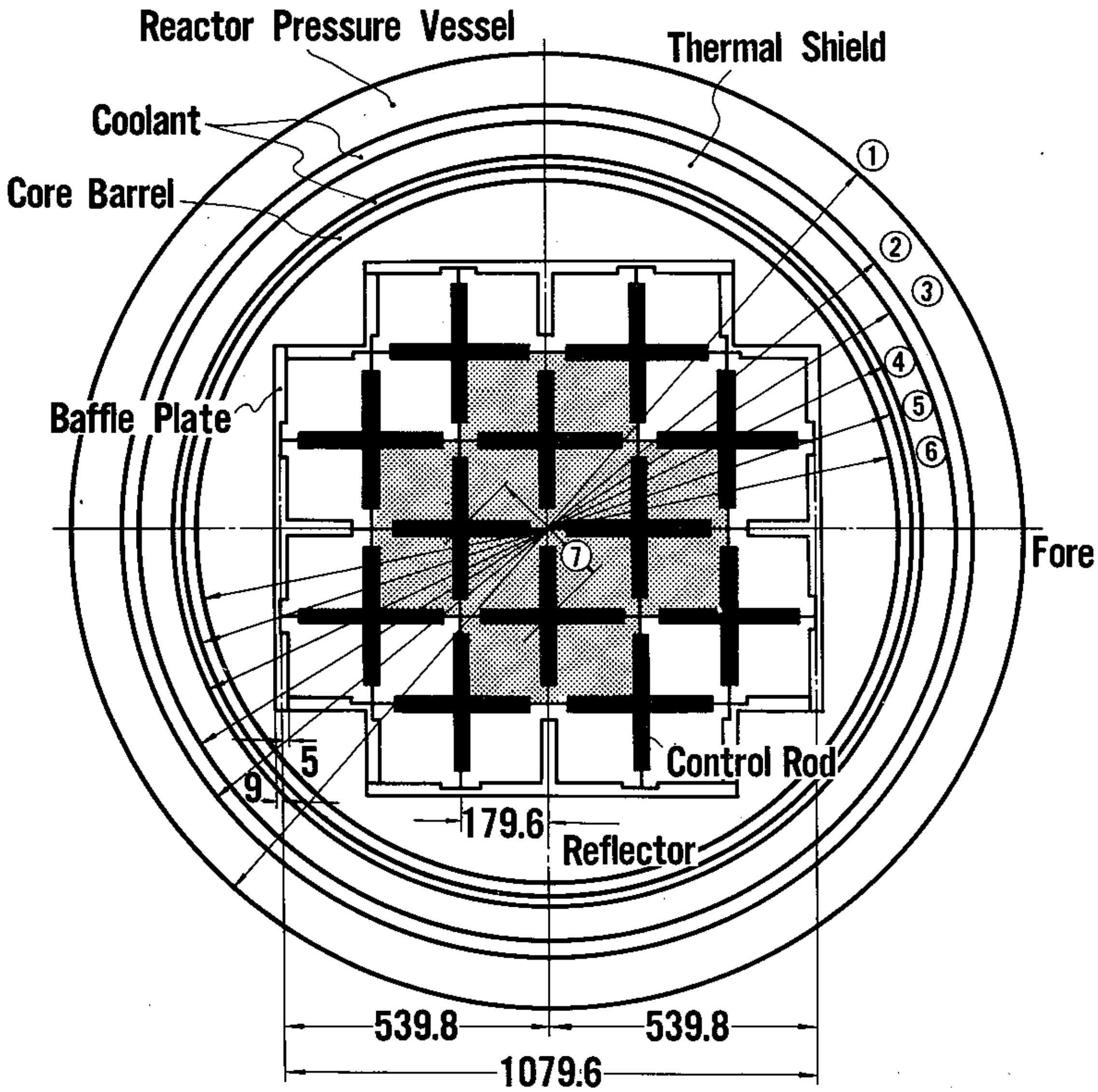
(3) Axial Power Distribution Averaged in the Core at HFP

Agreement between SCOPERS-2 and the independent 3-dimensional calculation is excellent as shown in Fig. 3.11.

(4) Axial Power Distribution Normalized for Each Fuel-Assembly at HFP

Individual axial power distribution is shown for comparison between the SCOPERS-2 (step-wise line) and the independent 3-D calculation (dot) in Fig. 3.12 and Fig. 3.13. It is clear that SCOPERS-2 can predict almost the same power profile as the detailed diffusion calculation.

It can be concluded that the present version of the SCOPERS-2 predits well the different states of a reactor when it is calibrated once to the reference data of the reactor. The case here reported is one of the example in which one can note the model's capability to predict the full power state, based on the zero power state calibration.



3.24 W/0 Fuel Assembly 4.44 W/0 Fuel Assembly

No.	Dia.(mm)	Material
1	1948	Carbon Steel (SUS304 Cladding)
2	1752	H_2O
3	1650	SUS304
4	1500	H_2O
5	1480	SUS304
6	1450	Control Rod Pitch
7	254	

Fig. 3.1 The core configuration (1): horizontal section

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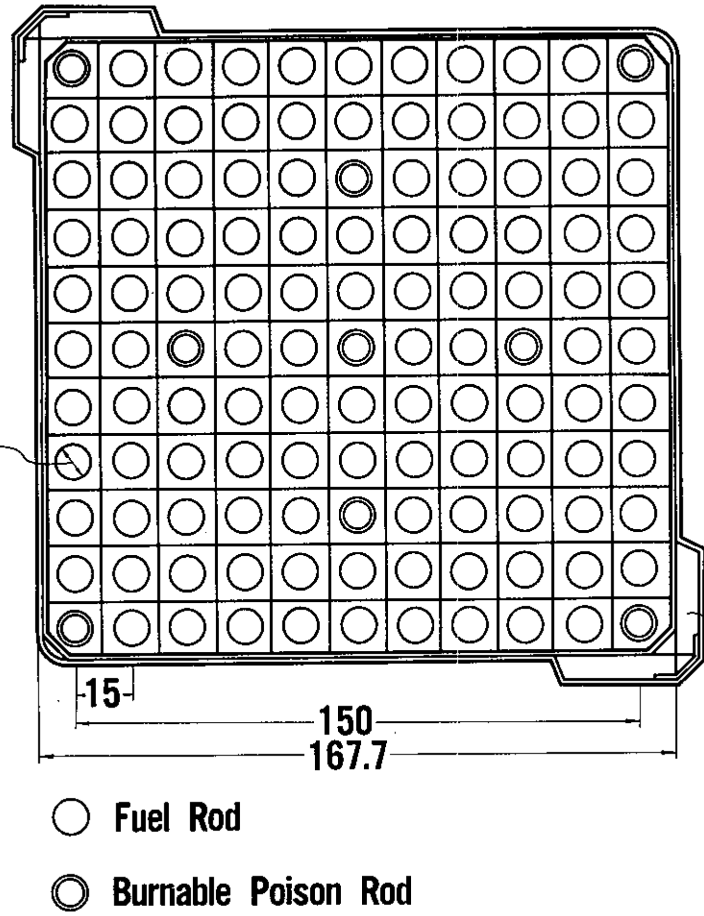

Fig. 3.2 The core configuration (2): fuel assembly

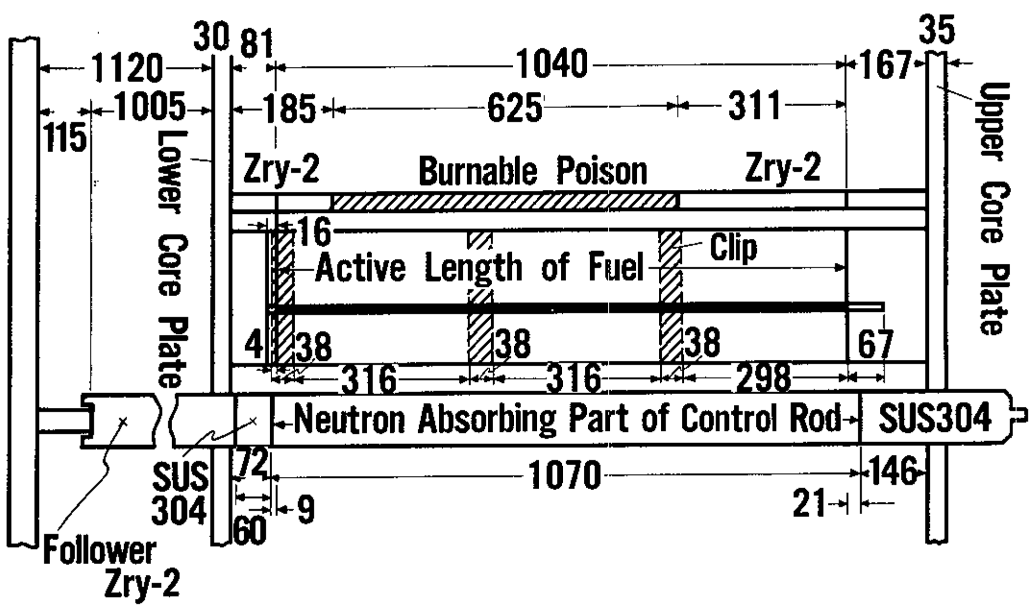

Fig. 3.3 The core configuration (3): vertical section

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Table 3.1 Fermi Ages and Diffusion Areas in 3.2 w/o Fuel

	No-Clip part	Clip part
τ	49.873	49.265
$1/\kappa^2$	4.2332	3.9117

	Upper part	BPR-part	Lower part
τ	49.848	49.812	49.674
$1/\kappa^2$	4.2198	4.2008	4.1281

Table 3.2 Fermi Ages and Diffusion Areas in 4.4 w/o Fuel

	No-Clip part	Clip part
τ	49.670	39.051
$1/\kappa^2$	3.5147	3.2989

	Upper part	BPR-part	Lower part
τ	49.644	49.608	49.468
$1/\kappa^2$	3.5057	3.4930	3.4442

Table 3.3 The k_∞ is 3.24 w/o Assembly-Cell

	No-Clip part	Clip-part	Volume average
Upper part	29.8 cm long	1.3 cm long	31,1 cm long
k_∞^{UC}	1.2536	1.1194	1.2480
k_∞^{HC}	1.0578	0.9484	1.0532
k_∞^{FC}	0.8444	0.7644	0.8411
BPR-part	56.2 cm long	6.3 cm long	62.5 cm long
k_∞^{UC}	1.1757	1.0683	1.1649
k_∞^{HC}	0.9936	0.9081	0.9850
k_∞^{FC}	0.7977	0.7363	0.7915
Lower part	7.0 cm long	3.4 cm long	10.4 cm long
k_∞^{UC}	1.2536	1.1194	1.2097
k_∞^{HC}	1.0578	0.9484	1.0220
k_∞^{FC}	0.8444	0.7644	0.8182

Note UC : Uncontrolled
HC : Half Controlled
FC : Full Controlled

Table 3.4 The k_∞ in 4.44 w/o Assembly-Cell

	No-Clip part	Clip-part	Volume average
Upper part	29.8 cm long	1.3 cm long	31.1 cm long
k_∞^{UC}	1.2977 1.3392	1.1815 1.2135	1.2928 1.3339 (1.3011)
k_∞^{HC}	1.0966 1.1430	1.0039 1.0393	1.0927 1.1387 (1.1019)
k_∞^{FC}	0.9288	0.8515	0.9256
BPR-part	56.2 cm long	6.3 cm long	62.5 cm long
k_∞^{UC}	1.2075 1.2418	1.1163 1.1439	1.1983 1.2319 (1.2050)
k_∞^{HC}	1.0248 1.0625	0.9535 0.9838	1.0176 1.0546 (1.0250)
k_∞^{FC}	0.8693	0.8118	0.8635
Lower part	7.0 cm long	3.4 cm long	10.4 cm long
k_∞^{UC}	1.2977 1.3392	1.1815 1.2135	1.2597 1.2981 (1.2674)
k_∞^{HC}	1.0966 1.1430	1.0039 1.0393	1.0663 1.1091 (1.0749)
k_∞^{FC}	0.9280	0.8515	0.9035

Note Upper : With Buffle
Lower : Without Buffle

UC : Uncontrolled
HC : Half Controlled
FC : Full Controlled

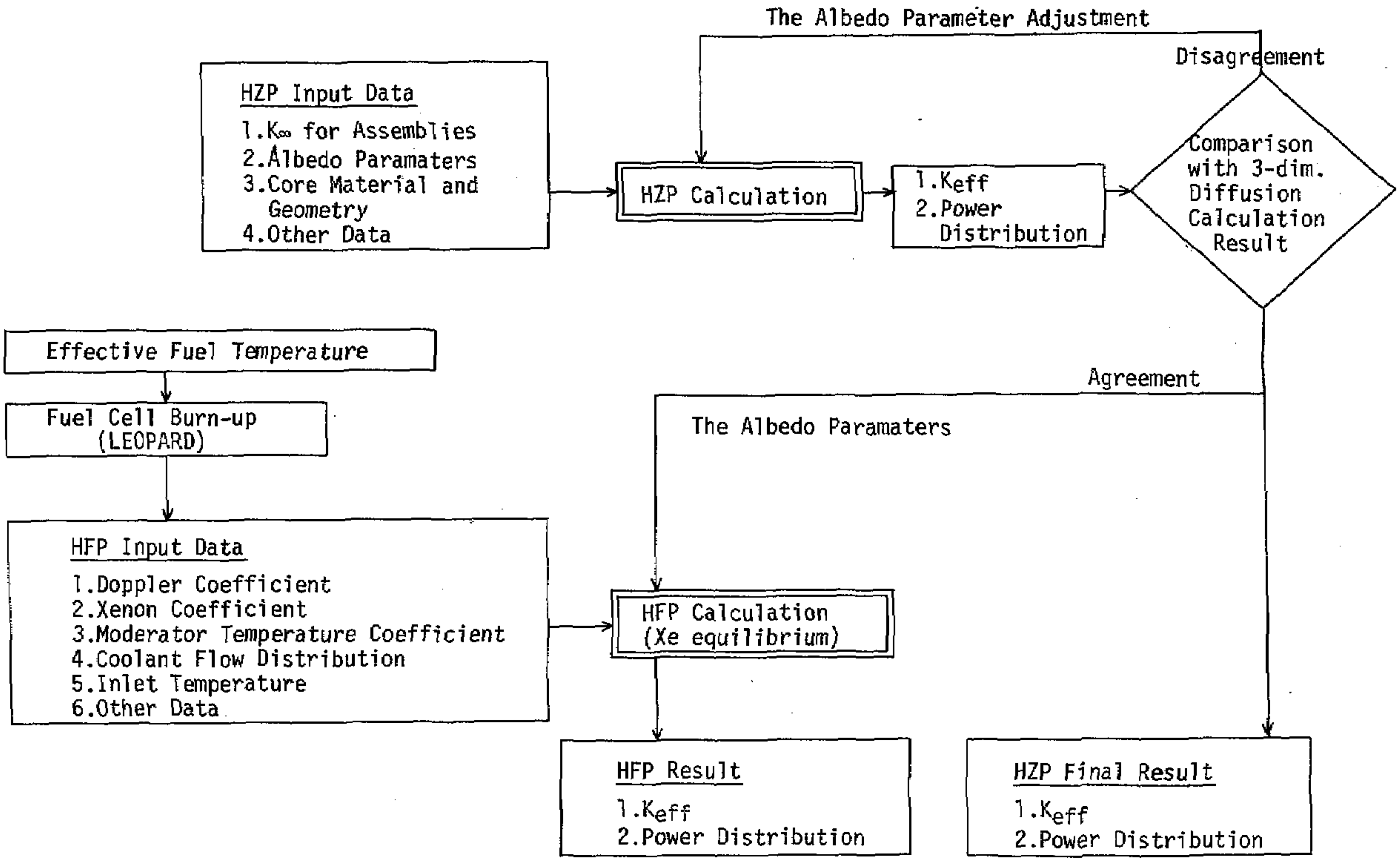

Fig. 3.4 The calculation procedure by SCOPERS-2

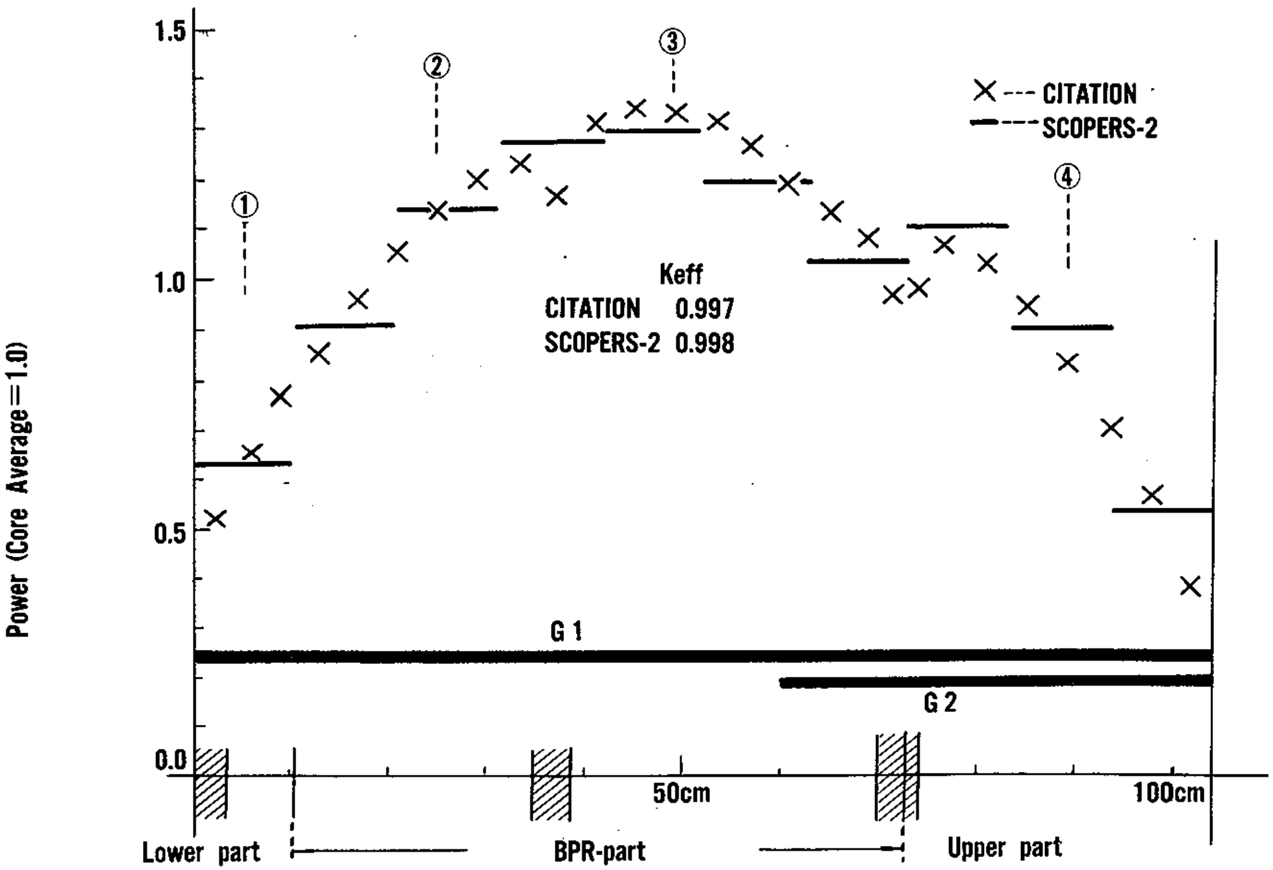

Fig. 3.5 Comparison of the axial power distribution between SCOPERS-2 and CITATION (HZP, BOL)

0.897 0.843	0.980 0.931	0.888 0.900
1.247 1.339	1.332 1.382	0.771 0.701
1.069 1.121	0.808 0.784	

Fig. 3.6 Enthalpy-rise distribution calculated by *SCOPERS-2* and *CITATION*

Upper : CITATION

Lower : SCOPERS-2

1.174	1.006	0.789
1.473	1.264	0.595
1.020	0.680	

Fig. 3.9 Enthalpy-rise distribution by SCOPERS-2 (HFP, Xe Equilibrium)

0.689 0.624	0.618 0.547	0.510 0.497
0.893 0.969	0.834 0.842	0.435 0.393
0.693 0.717	0.502 0.475	

① 5.15cm plane from the core bottom

1.215 1.178	1.096 1.027	0.905 0.888
1.577 1.767	1.460 1.492	0.772 0.694
1.218 1.257	0.886 0.831	

② 24.75cm plane from the core bottom, BPR-part

1.332 1.285	1.285 1.172	1.112 1.048
1.773 1.945	1.721 1.696	0.946 0.814
1.413 1.406	1.055 0.947	

③ 48.95cm plane from the core bottom, BPR-part

0.536 0.484	0.842 0.872	0.865 0.989
0.873 0.897	1.187 1.347	0.732 0.769
0.919 1.060	0.755 0.789	

④ 89.10cm plane from the core bottom

Fig. 3.7 Horizontal power distribution in typical planes (at HZP, BOL).

Upper : CITATION Lower : SCOPERS-2

The axial positions of the above planes ①∿④ are denoted in Fig. 3.5.

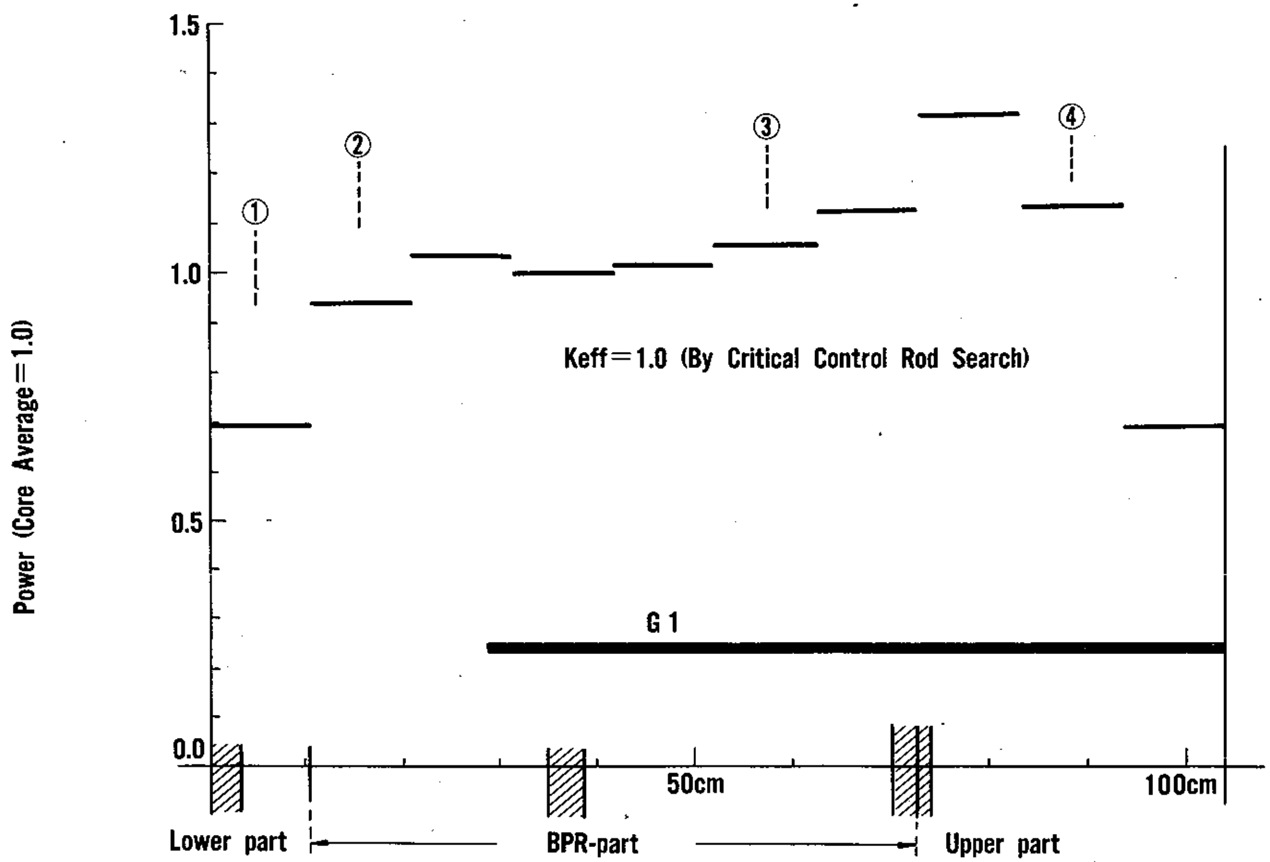

Fig. 3.8 Axial power distribution (HFP, Xe Equilibrium)

1.104	0.901	0.527
0.930	0.802	0.370
0.564	0.357	

① 0～10.4cm from the core bottom

1.476	1.206	0.703
1.283	1.072	0.480
0.770	0.523	

② 10.4～20.8cm from the core bottom, BPR-part

1.130	0.981	0.836
1.612	1.369	0.642
1.138	0.757	

③ 52.0～62.4cm from the core bottom, BPR-part

1.126	1.003	0.915
1.682	1.482	0.722
1.256	0.851	

④ 83.2～93.6cm from the core bottom

Fig. 3.10 Horizontal power distribution in each plane (HFP, Xe Equilibrium).

The axial positions of the above planes ①∿④ are denoted in Fig. 3.8.

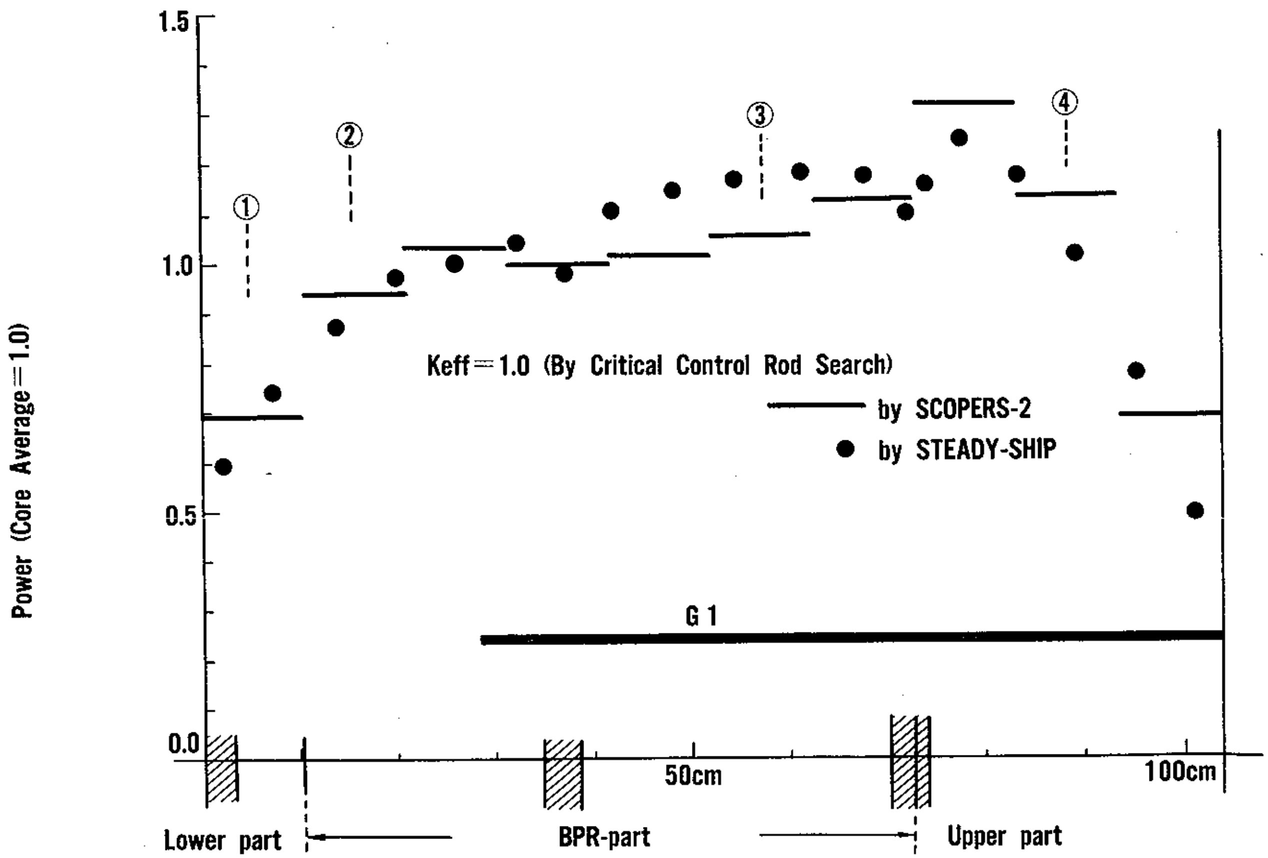

Fig. 3.11 Axial power distribution (HFP, Xe Equilibrium)

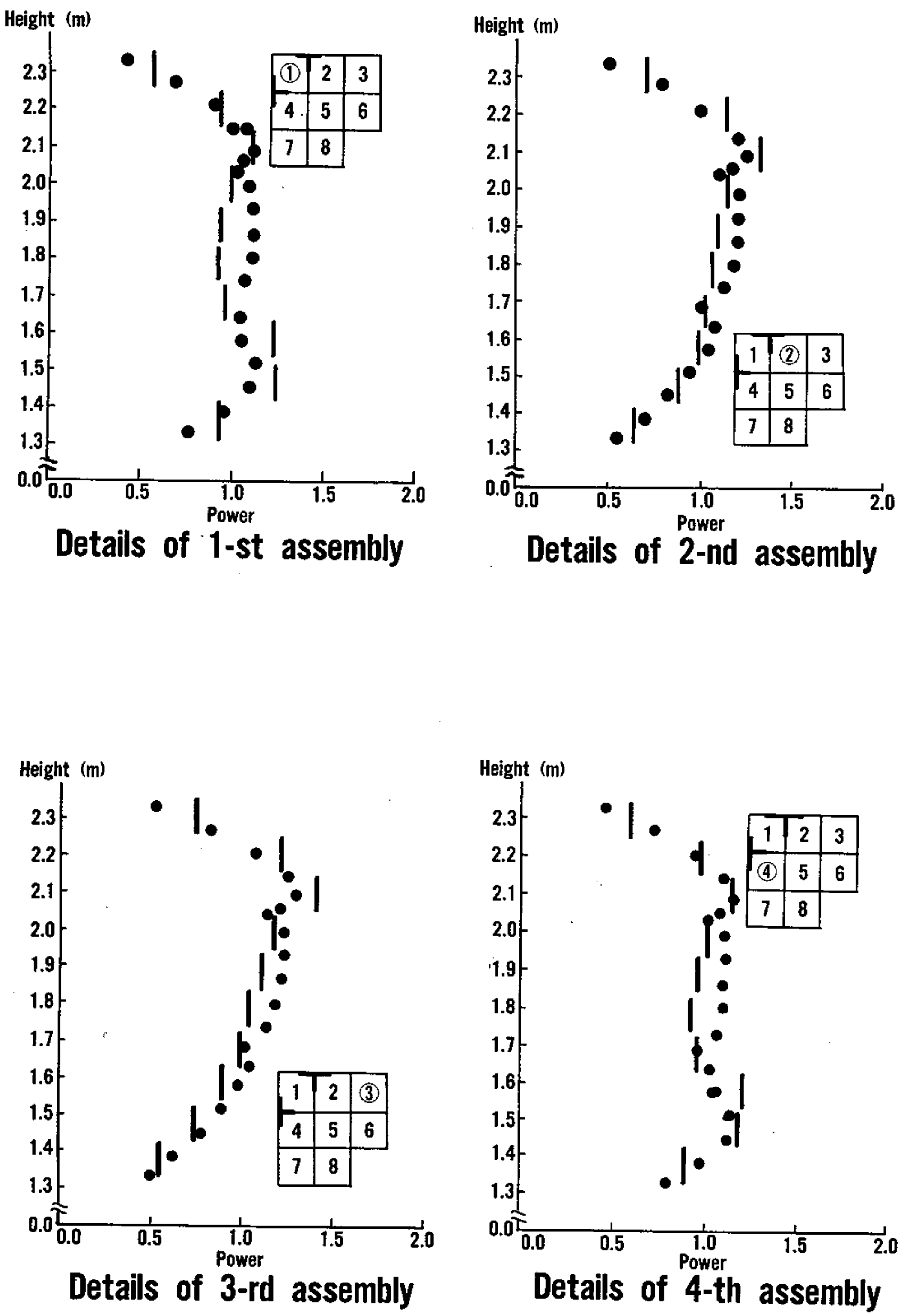

Fig. 3.12 Axial power distribution for each fuel-assembly at the hot full power condition (1).

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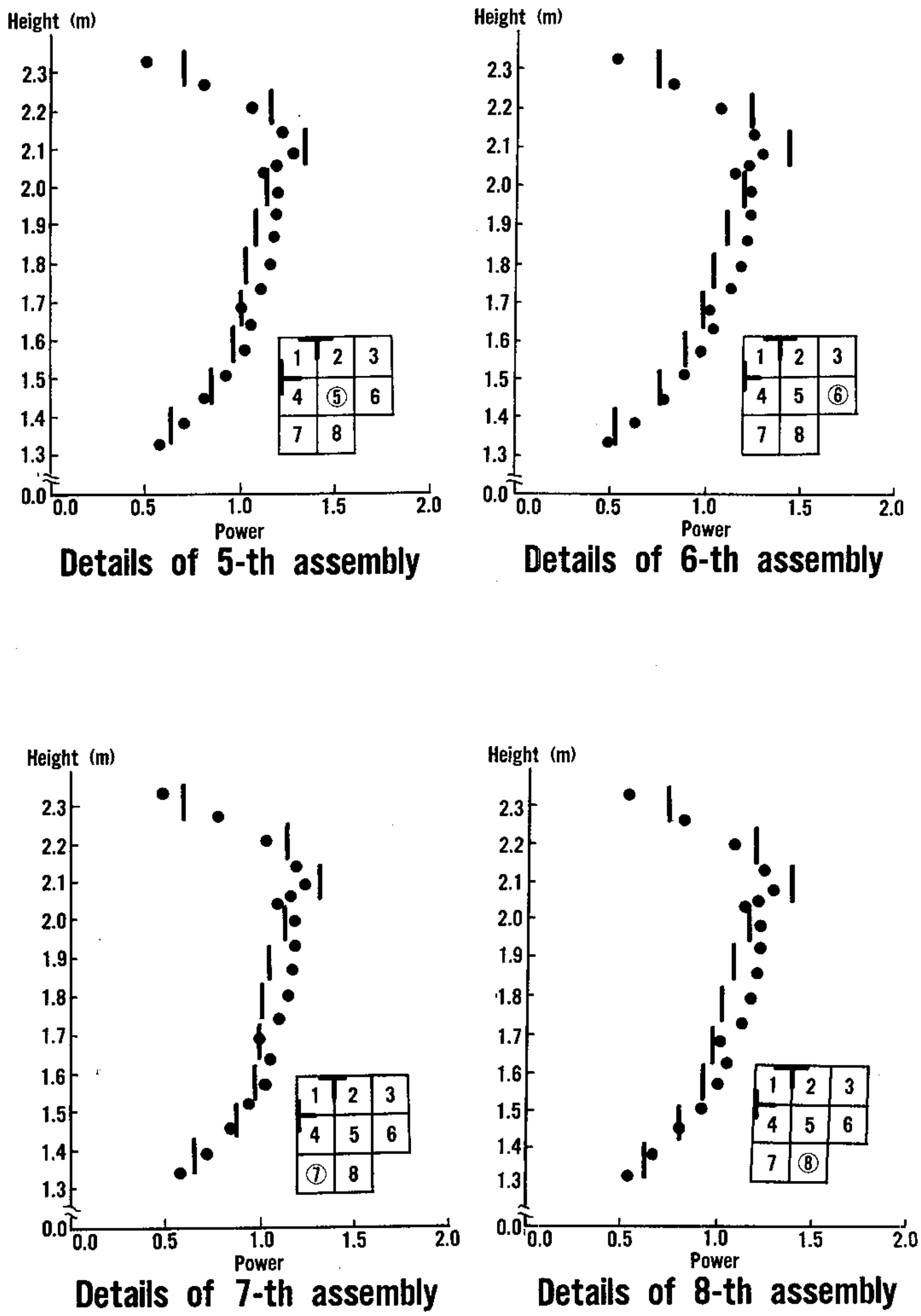

Fig. 3.13 Axial power distribution for each fuel-assembly at the hot full power condition (2).

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4. Concluding Remarks

A Static Core Performance Simulator for Light Water Reactors (SCOPERS-2) is based on a generalized FLARE-type nodal equation which includes the so-called FLARE-type equation as the extreme application. Compared with the original FLARE program, many modifications and improvements are also included in the present program of SCOPERS-2, Moreover, a PWR as well as a BWR can be calculated by the SCOPERS-2.

The usefulness of the computer program for a PWR was demonstrated in the present report through some calculations for nuclear ship MUTSU reactor. It is verified through comparison with 3-dimensional diffusion calculations that the SCOPERS-2 gives the results which are precise adequately in a practical sense. The SCOPERS-2 has been incorporated into the fuel management calculation code system at Japan Nuclear Ship Development Agency (JNSDA). Some burn-up analyses by using this program are also in preparation for MUTSU reactor at JNSDA.

This program will provide a useful tool for the reactor physics design under a wide range of parameters as well as for the fuel management of an operating reactor both for BWR and PWR, in a point of view of computer speed and accuracy.

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