JOYO SAFETY ANALYSIS FOR POWER INCREASE OF JOYO

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Table of Contents

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ABSTRACT

Some problems in recently conducted safety evaluation for future power increase from presently authorized 50 MWt of the experimental fast reactor "JOYO" are presented in this paper.

1. INTRODUCTION

The design of "JOYO" was completed in 1969, and approval for construction was issued in the beginning of 1970. The reactor is now under commissioning tests with an object of attaining criticality in 1976.

The entire plant equipment have been designed for a target thermal output of 100 MW. However, taking into account the facts that the reactor is the first sodium cooled reactor to be built in Japan and various problems are to be solved through R & D work, the first phase reactor power has been proposed as 50 MW for which safety evaluátion approval by the authorities concerned was granted in February, 1970. Since then, in parallel with the construction of "JOYO", various research and development works have been conducted in order to confirm the core performance characteristics.

These research and development works involve; (i) Nuclear mock-up tests by use of Fast Critical Assembly, FCA, (ii) Hydraulic tests on full core assembly and also on single subassembly by use of water and sodium. (iii) Irradiation tests, post irradiation analysis and mechanical strength tests on fuel elements, (iv) Calibration tests on various instruments and (v) Collection and evaluation of various material properties, etc. While applying the results of these research and development works into design and analysis, improvement and reevaluation were made on methods of design and analysis of the reactor core and on determination of safety margin in thermal design.

With the results of core analysis performed in order to determine the maximum attainable thermal power of the present "JOYO" core configuration (Mark-I core), the second phase reactor power and the maximum burn-up of the core fuel have been set as 75 MW and 42,000 MWD/T respectively. For this power level, a detailed evaluation relating to safety as well as nuclear, thermohydraulic, mechanical performance at normal operation has been conducted from Oct. 1973 to April 1975.

Future program for "JOYO" includes also a plan to modify the present core configuration into a more suitable core for irradiation of fuels and materials, after successful operation with the present core. This is called "JOYO" Mark-II core. The objective of the Mark-II core is to obtain the maximum attainable neutron flux, 5×10^{15} n/cm^2-sec, and as wide an irradiation zone as possible.

The designed Mark-II core consists of two zones; core zone and reflector zone, having fuel pellets of smaller diameter and a higher plutonium enrichment than those of Mark-I. Steel reflector has been chosen from a standpoint of efficient heat removal and neutron economy. "JOYO" Mark-II core will be operated at 100 MWt. This will be the third phase reactor power. Safety analysis on the "JOYO" Mark-II core has also been carried out based on the results of a series of conceptual designs of Mark-II core since Oct. 1973.

Preparatory work for safety evaluation on the second phase

reactor power (Mark-I, 75 MW) to be given by the authorities is now undergoing.

Safety evaluation on the third phase reactor power (Mark-II, 100 MW) by the authorities is currently scheduled in the year 1976.

The fundamental principles adopted in the safety analysis concerning the second and third phase reactor power are as follows.

(a) Adoption of the fundamental philosophy approved in safety evaluation for the first phase reactor power. (Mark-I, 50 MW).

(b) Full utilization of the results of the R & D which have been and being done at the O-arai Engineering Center (OEC) of PNC and elsewhere.

(c) Revision of accident analysis codes as needed.

(d) Adoption of more detailed and elaborate method for the Hypothetical Core Disruptive Accident.

Hazard evaluation of gaseous radioactive waste to be released from "JOYO" plant during normal operation into environment is one of the most serious matters to be thoroughly evaluated. Therefore, it must be carried out using more realistic and elaborated calculational methods, various meteorological data obtained at O-arai site in the recent years (1971 ~ 1973), and results of re-evaluation on the radioactive wastes to be released from the reactor.

2. ACCIDENT ANALYSES

2.1 Classification of Accidents

Various accidents are divided into three categories as shown below. The first is called "reactivity accidents" which arise from abnormal reactivity insertion to the core by some troubles, malfunctions or miss operations of reactor control instruments. The second is called "mechanical accidents" which are caused by some mechanical troubles or damages of reactor equipment. The third is called "remaining miscellaneous accidents".

Major items of accident analyses are as follows.

(1) Reactivity Accidents

a. Start up accident

b. Control rod withdrawal accident at full power operation

c. Fuel subassembly loading accident

d. Sudden sodium flow increasing accident in primary coolant circuit

e. Sudden sodium flow increasing accident in secondary coolant circuit

f. Sudden air flow increasing accident in main air cooler

g. Fuel slumping accident

h. Fuel subassembly bowing in core

(2) Mechanical Accidents

a. Loss of electric power supply accident

b. Loss of pumping power accident of primary coolant circuit

c. Loss of pumping power accident of secondary coolant circuit

d. Loss of blower power accident of main air cooler

e. Local flow blockage accident

f. Primary piping failure accident at operating condition

g. Primary piping failure accident at reactor shut down condition

h. Secondary piping failure accident

i. Primary piping failure accident of auxiliary coolant system

j. Refueling accident

k. Radioactive waste release accident from waste disposal equipment

l. Radioactive waste release accident from reactor service system

(3) Remaining Miscellaneous Accidents

a. Trouble or failure of control rod drive systems

b. Trouble or malfunction of valves

c. Failure of sodium handling equipment

d. Earthquake, fire, flood, typhoon

2.2 Treatments of Reactivity Accident

In the safety analysis of start-up accident, following assumptions have been made in compliance with the request from the Safety Committee at safety evaluation on the first phase reactor power. One of control rods (regulating rods) is to be withdrawn continuously with maximum reactivity insertion rate, without any shut down action at the scram level both of start-up range (10^{-5} full power) and intermediate power range instrumentation (10^{-1} full power). In such accident, the reactor power increases rapidly until it reaches the scram level of power range instrumentation. Then safety rods are inserted into core zone and power decreases.

In the safety analysis of sudden sodium flow increasing accident of the primary or secondary circuit, it is assumed that the driving motor on the pump is to be short-circuited and the flow increases in stepwise. The maximum flow increment at the accident is obtained from the relation between pressure drop of the coolant circuit and the Q-H curve of the mechanical pump. In the analysis of fuel slumping accident, the fuel pellets consist of PuO_2-UO_2 ceramics, are assumed to densify up to the theoretical density and slump instantaneously inside the cladding tube. Thus, a stepwise reactivity, 22 ¢ for Mark-I core and 27 ¢ for Mark-II core, is introduced into the reactor which is operating at normal full power condition.

The analysis of fuel subassembly bowing in the core has also been carried out for the reason that it is worth while to analyse performance characteristics of the core from the standpoint of safety of the reactor, though it does not come under the accidents.

2.3 Treatments of Mechanical Accidents

It was approved by the Safety Committee for the first phase reactor power that local flow blockage accident is unlikely because of the reasons shown below.

(i) Careful control of impurities in the sodium coolant makes it possible to prevent from dissolving a large quantity of impurities in sodium.

(ii) There are many orifices on the entrance nozzle of a fuel subassembly in six radial directions so that all the orifices can not be blocked instantaneously.

(iii) Even if sodium coolant flow in a fuel subassembly decrease as a result of partial flow blockage, it may be able to maintain the minimum flow rate required.

(iv) The temperature sensors are installed in the instrument grid just above the outlet of each individual fuel subassembly of "JOYO".

The coolant temperature at the sensor position may deviate from true temperature of exist coolant of a subassembly due to cross flow mixing with coolant of adjacent subassemblies. Therefore, study was made on hydraulic profile of sodium flow in the core of "JOYO", using a three dimensional model of seven subassemblies and a two dimentional model of seven flow zones. It revealed that the minimum detectable blockage rate lies between 10 % and 15 % of the flow area, corresponding to the maximum remaining flow rate of about 95 %, and the maximum detectable blockage rate lies between 67 % and 77 % of the flow area, corresponding to the minimum remaining flow rate of about 40 %.

For the second phase reactor power (Mark-I, 75 MWt), we have also analyzed local channel blockage which can not be detected by the thermocouples installed above the outlet of the fuel subassemblies as shown in Fig. 1. These analysis involve (a) Local blockage caused by pieces of cladding tubes, structure materials or sodium compounds. In those cases, blockage was assumed to occur at two representative locations, one at the center of the core and the other at the upper end of the active core. (b) Pin contact caused by fuel pin bowing. For case (a), calculation was carried out for full blockage of unit subchannel of coolant flow and complete blockage of coolant channel surrounding a fuel pin as the most likely case.

For case (b), two bowing patterns of fuel pins were considered. One is bowing of an adjacent fuel pin toward the centerline and the other toward the centerline of an adjacent coolant cell as shown in Fig.1.

The maximum internal pressure of the cladding tube of "JOYO" fuel is 70 kg/cm^2 and the maximum neutron dose is about 10^{21} to 10^{23} nvt.

Therefore threshold temperature for burst rupture of cladding must be over 800 °C. This has been verified by an out of pile experiments.

As for the examination of the possibility of burst rupture, hot spot temperature of the hottest pin was considered. In consequence, following conclusion could be drawn that it is hardly possible to cause burst rupture of cladding, because the maximum temperature of fuel is far below melting point of fuel and also the maximum cladding temperature far below 800 °C. As for the examination of the possibility of creep rupture, nominal temperature of the hottest pin was considered because of the low occurrence probability of local blockage. As the maximum cladding temperature does not exceed 670 °C for this case, it seems that it is hardly possible to become a problem for safe operation of "JOYO".

It was also approved by the Safety Committee for the first phase reactor power that large scale instantaneous breakdown accident of primary piping at operating condition is unlikely because of the reasons shown below.

(i) All primary piping of "JOYO" is made of double wall stainless steel type 304.

(ii) Careful control of impurities in the sodium coolant makes it possible to prevent from significant corrosion.

(iii) Only a small scale failure of inner piping wall may occur, if any, which may be caused by a defect of wall such as crack.

Even if any such a crack may exist, it cannot be developed to an unstable ductile fracture which leads to a large scale instantaneous breakdown accident of primary piping before sodium leakage from the crack of the piping wall can be detected by sodium leak detectors.

The aims of having the outer guard wall of primary piping is;

(a) to form the flow passage of nitrogen gas which is used to preheat the primary coolant circuits.

(b) to detect sodium leakage from a crack of the primary piping.

(c) to hold up sodium coolant leaked from the primary piping.

In the safety evaluation of "JOYO", primary piping failure accident at reactor shutdown condition is also considered as well as at operating condition. The accident is assumed to take place as follows:
Reactor is shutdown at a failure of primary inner piping wall. The nitrogen gas atmosphere inside reactor containment is substituted to air for repairing the defected piping wall. Then, an outer guard wall failure accident of the primary piping is assumed to take place at that time, which leads to sodium spill on the floor. The sodium spilled on the floor reacts with air, generating a large quantity of heat, and eventually it leads to a release of radioactive materials from the containment into environment.

Some problems associated with the analysis of secondary piping failure accident are as follows.

As secondary circuit of "JOYO" is made of single wall piping whose material is a low alloy steel, following two things were requested by the Safety Committee for the first phase reactor power. (Mark-I, 50 MW).
First, problems such as deterioration in mechanical strength, thermal stress and corrosion must be carefully examined, by carrying out out-of-pile experiments. Second, in the analysis of the secondary piping failure accident, an instantaneous rapture of the secondary piping which leads to a loss of cooling function should be assumed. It revealed that the reactor soundness is assured by a slow scram caused by a signal of high inlet coolant temperature of reactor vessel.

2.4 Buck-up Analyses

There are some other problems which have been examined. These involve (a) Entrainment of gas bubbles in the primary coolant flowing into core zone, (b) Gaseous fission product ejection from fuel pin hole, (c) Accidents of fuel handling machines, (d) Double faults of active devices, (e) Anticipated transient without scram, (f) Post accident heat removal. (examination of the possibility of melt through accident).

For the problem (d), three cases were chosen.

(i) Primary pump sticking (loop A) + start-up failure of ponney motor. (loop B)

(ii) Primary pump tripping (loop A) + start-up failure of ponney motors. (loop A & loop B)

(iii) Primary pump sticking (loop A) + secondary piping failure (loop B)

For the problem (f), we also have examined the possibility of post accident heat removal from molten fuel on the core supporting plate or on the safety vessel as well as on the reactor vessel. The case of holding model on the bottom of the reactor vessel was briefly discussed at safety evaluation of the first phase reactor power.

3. HAZARD EVALUATION

3.1 Outline of the Guidelines for Reactor Site Evaluation

The Guidelines for Reactor Site Evaluation are to be applied to the evaluation of possibility of radiation injury of public that should suffer in both major accident and hypothetical accident.

These Guidelines were adopted by the Japan Atomic Energy Commission on May 27, 1964 for the use of the Safety Committee J.A.E.C., in evaluating safety of large stationary land based reactors in relation to the site conditions. The outline of the Guidelines is as follows:

(i) The basic aims to be attained by the Guidelines are,

(a) No radiation injury should be suffered by the public in case of the major accident foreseeable from a technical point of view.

(b) No remarkable radiation injury should be suffered by the public in case of the hypothetical accident which is inconceivable from a technical point of view. (such hypothetical accident is meant, for example, one in which some of the safety protection devices, whose effect had been counted upon in the assumption of the major foreseeable accident, would fail to function).

(c) In case of the hypothetical accident, the effect on the genetic dose of the national population should be sufficiently small.

(ii) The site conditions to fulfill the above basic aims are,

(a) All the area within a certain distance from the reactor should be non-residential. By "certain distance" is meant, in this case, that distance within which, were individuals to remain, they would be judged liable to receive a radiation injury in case of the worst foreseeable accident. To determine this distance the following doses are tentatively to be taken as constituting a radiation injury; 150 rems for the thyroid (infant), 25 rems for the whole body.

(b) The non-residential area should be surrounded within a certain distance from the reactor by a low population area. By "certain distance" is meant, in this case, that distance within which, were individuals to remain, they would be judged liable to receive a radiation injury in case of the hypothetical accident. To determine this distance, the following doses are tentatively to be taken as a rough measure of constituting a radiation injury; 300 rems for the thyroid (adult), 25 rems for the whole body.

(c) There should be a certain distance between the reactor site and an area of dense population so that the total population dose will not exceed a certain level in case of the hypothetical accident. To determine this level, examples adopted in other countries (for instance, 2 million man-rems) are to be taken as reference.

The Guidelines are to be applied to the site evaluation of all stationary land reactors with a thermal output of 10 MW or more, and also shall serve a references in judging sites for reactors smaller than 10 MW.

The outline of hazard evaluation considered for the site evaluation of the experimental fast reactor "JOYO" both for the second phase power (Mark-I, 75 MW) and for the third phase power (Mark-II, 100

MW) is as follows.

3.2 Major Accident

Some of the fuel pins with the maximum burn-up of fuel, 42,000 MWD/T in case of the second phase reactor power (Mark-I, 75 MW), 50,000 MWD/T in case of third phase reactor power (Mark-II, 100 MW), are assumed failed with holes on cladding, and are releasing fission product gases into the sodium coolant, which corresponds to 2 % of the maximum core inventory of the fission product gases.

The transfer rate of fission product gases from the liquid sodium coolant to the argon covergas is assumed to be 100 % for the rare gases and 10^{-5} for the iodine. The reactor is assumed to be still operated at full power. The argon gas, which contains fission product gases transferred to the covergas, is led to a storage tank of waste gases. When the gas pressure in the storage tank rises up to the maximum design pressure of 9 kg/cm^2, the tank is assumed to break out suddenly and to release the gases into the surrounding cellar.

The hazard evaluation of this major accident is therefore treated as follows. Ten percent of the stored waste gas in the tank is assumed to leak out directly to the atmosphere on the ground level which corresponds to an increment of the pressure increase in the cellar, and the remaining is assumed to be released from the exit of the stuck, 80 m high.

3.3 Hypothetical Accident

The experimental fast reactor "JOYO" has been designed to accommodate a maximum hypothetical accident which predicates gross melting and consequential compaction of the core into a smaller volume that produces a large and rapid increase in reactivity. As a result of the reactivity introduced by the core compaction, the reactor becomes prompt critical and the nuclear excursion is finally terminated by explosive disassembly of the core. The forces causing the disassembly are high pressure produced in the fuel by the power burst. The outline of the hypothetical accident assumed for "JOYO" is as follows.

In the calculation, a flattened radial power shape in the possible largest core is assumed for a pessimistic evaluation. Furthermore, such hypothetical assumptions as that sodium does not exists in the core, the reactor keeps operating at the rated power level despite negative reactivity to be added by loss of sodium in the core, and no scrams, are made. Then core starts melting and falling by gravity.

As the negative feedback reactivity to be considered in the calculation, only two feedbacks are considered. One is due to core disassembly effect and the other is due to Doppler effect. As for the relation between pressure and temperature for oxide fuel, a threshold type equation-of-state has been used.

The relation between pressure and temperature for UO_2 is obtained by using the compressibility factor "z", reduced pressure "Pr", reduced temperature "Tr" and excess internal energy "$(U^* - U)/(RTc)$" tabulated by Hougen and Watson.

The results of calculations have shown that the effective working energy, produced by the recriticality accident, is less than 200 MW-sec. Therefore, it was applied to the plant design of "JOYO" and the working energy equivalent to 50 Kg weight of TNT explosives was adopted as a design base.

Against the explosion, the safety vessel can withstand and hold

the sodium level for the decay heat removal of the core.

Following assumptions are made for the hazard evaluation.

(i) All of the fuels in the core have already reached its maximum burn-up when the core meltdown accident occur. Of fission product inventory in the core, 100 % of the rare gases, 10 % of the iodine and 1 % of the solid are assumed to be ejected in the reactor containment instantaneously.

(ii) When the recriticality accident occurs, sodium is assumed to be ejected through the mechanical gaps of the rotating plugs, carrying plutonium and uranium, etc., and reacting instantaneously with air in a containment.

(iii) Inorganic iodine floating in air within containment decays by gravitational sedimentation and deposition of particles by diffusion. The equivalent half-life time of the inorganic iodine is assumed to be 1 hour.

(iv) The plutonium oxide aerosol may behave together with the sodium oxide, uranium oxide and structural steel aerosols. The mass concentration of the aerosol decays by gravitational sedimentation and particle deposition due to diffusion. The relationship between the initial mass concentration and the half-life time is analytically modeled.

(v) 90 % of iodine and plutonium which leak from containment to the annulus space of containment are assumed to be selectively collected in charcoal filters installed.

(iv) For the hazard evaluation of radioactive effluents, a calculation has been made of gamma ray exposure not only due to a radioactive cloud formed by the effluent from the reactor but also due to skyshine of radioactive materials flooded in the reactor containment.

(vii) The weather conditions used in the calculation for the distribution of nuclide concentration in the radioactive cloud are as follows. The inversion layer lies on the exit of the stuck, 80 m high, and the nuclide concentration in the cloud is fumigating below the layer only for two days after the hypothetical accidents. For this period of time, calculation is performed by the so-called submersion model. Thereafter we assume the Gaussian type distribution of concentration, and one of the stability type is chosen out of the six stabilities of Pasquill type which gives the maximum dose rate.

The lateral spread is assumed to be 30 degrees and the wind speed is assumed to be 2 m/sec.

Besides the above analysis, more realistic and more elaborate accident analyses have been made in order to describe various phases of the progression from initiation of accident to disassembly of core in the hypothetical accidents.

Those include analyses of the fuel - sodium interaction and the pressure propagation through the primary coolant systems which involve not only main coolant piping systems but also auxiliary coolant piping system. The results of the analyses have shown that reactor vessel, primary pipings, intermediate heat exchangers, primary pumps will not fail in such accident.

In the safety evaluation of the first phase reactor power, the decay heat associated with irradiated fuel was calculated by using the curve given by J.G. Yevick in Fast Reactor Technology (the M.I.T. Press).

An attempt has been made to improve the values of decay heat of fission products by accumulating relevant information. Namely, decay heat due to known short-lived fission products has been added and theoretically estimated decay heat of some fission products has been included. Feasibility of the present method of calculating the beta and gamma energy release rates of the fission products has been evaluated by comparing with several experiments, mainly those of thermal neutron fission of ^{235}U. It can be said that the theoretically estimated nuclear data for short-lived nuclides are useful and reasonable for calculation of the decay heat of fission products.

The decay heat associated with driver fuel irradiated to target exposure in the experimental fast reactor "JOYO" was calculated by using the improved computor program "FP-S" with a special nuclear data library containing 506 radioactive and 125 stable nuclides. This program was developed by Tasaka and Sasamoto in JAERI.

The results for the second phase reactor power (Mark-I, 75 MW) are shown in Fig. 2 and for the third phase reactor power (Mark-II, 100 MW) in Fig. 3. An estimation for uncertainty was made in order to identify range of applicability of the computor program by changing input parameters such as the neutron energy spectrum, the fissile nuclides, the neutron capture reactions of fission products and the number of fission products. It has been concluded that addition of 10 % of calculated decay heat as a safety margin should be sufficient to be on safe side in the safety analyses of "JOYO".

The contribution to the decay heat from the heavy-element activation products, ^{239}U and ^{239}Np, can be determined by calculating the concentrations of these isotopes, considering the neutron spectrum in the lattices of interest (conversion ratio, cross sections), together with experimentally measured half-life and emmitted radiation.

Thus, the heavy-element decay energy release can be expressed as fractions of initial power, P/P_0, as shown below.

For ^{239}U

$$\left(\frac{P^{239}U}{P_0}\right)=2.28\times10^{-3}\left(\frac{\Sigma c}{\Sigma f}\right)\cdot\left[1-e^{-4.91\times10^{-4}T_0}\right]\cdot e^{-4.91\times10^{-4}T_s} \quad \cdots\cdots\cdots (1)$$

For ^{239}Np

$$\left(\frac{P^{239}Np}{P_0}\right)=2.17\times10^{-3}\left(\frac{\Sigma c}{\Sigma f}\right)\{7.0\times10^{-3}(1-e^{-4.91\times10^{-4}T_0})(e^{-3.41\times10^{-6}T_s}-e^{-4.91\times10^{-4}T_s})$$
$$+(1-e^{-3.41\times10^{-6}T_0})\,e^{-3.41\times10^{-6}T_s}\} \quad \cdots\cdots\cdots\cdots\cdots\cdots (2)$$

where

$$\left(\frac{\Sigma c}{\Sigma f}\right)=\left(\frac{^{238}U\text{ capture reaction rate}}{\text{total fission reaction rate}}\right)$$

T_0 = irradiation time, sec

T_s = cooling time, sec

Here also, 10 % of calculated value has been added as a safety margin of the heavy-element decay heat in the safety analysis of "JOYO".

4.4 Transport of Halogen

The primary coolant system of the experimental fast reactor "JOYO" may be contaminated by radioactive nuclides released from failed fuel. Of those, iodine is one of the important element in safety evaluation. Iodine may be transferred to the argon covergas which is released into the environment through the gaseous waste processing systems, such as the storage tank and the charcoal filters, etc.

Knowledge of behavior of iodine in sodium and covergas is therefore important for the design, operation, maintenance and safety of "JOYO" plants. In particular, the radiation injury for the thyroid, suffered by the public during the normal reactor operation, has to be thoroughly evaluated for the intake of contaminated vegetables, cow's milk and human milk as well as the inhalation of the released iodine.

For convenience, we define a transfer rate of the iodine from liquid sodium to argon covergas, P, which is the ratio of atoms of iodine in the covergas devided by atoms of iodine in the sodium coolant. Under the normal operational condition of "JOYO", we have evaluated the transfer rate of iodine from liquid sodium to argon covergas as follows:

(i) If we use the data of the vapor pressure of NaI, then P becomes 3.8×10^{-7}.

(ii) If we use the data of the vapor pressure of sodium and the partition coefficient, then P becomes 1.4×10^{-6}.

(iii) If we extrapolate the experimental data of the PIRANA and of the KEWB pulse reactor to the conditions of "JOYO", then P becomes 1.5×10^{-6}.

(iv) If we extrapolate the data of the experiments with the sodium in-pile-loop (SIL) in JAERI and assure the adhesion coefficient of iodine to be 0.99 sec^{-1}, then P becomes 1.7×10^{-7}.

Therefore, it is concluded that an assumption of the iodine transfer coefficient of 1.0×10^{-5} should be reasonable for the safety analysis of "JOYO".

4.5 Evaluation of Tritium and Carbon-14

Tritium and carbon-14 represent a significant part of the radioactive materials released to the environment from "JOYO". As the attention paid to environmental affairs increases, further knowledge of the behavior of tritium and carbon-14 released from a reactor becomes increasingly important. The formation of tritium occurs by fission in nuclear fuel and also by neutron activation reactions, mainly with lithium and boron isotopes.

(i) Fission

If the yield of tritium in fast fission is assumed to be 1.6×10^{-4} atoms per fission, fast reactors should produce about 2.5×10^{-7} Ci/MW-sec.

(ii) Boron in Shim Rods

The activation of ^{10}B by two mechanisms are considered:

(a) The fast neutron reaction : $^{10}B(n, 2\alpha)^{3}H$.

(b) The two-step reaction : $^{10}B(n, \alpha)^{7}Li(n, n\alpha)^{3}H$, which is estimated to produce only about 4 % of tritium compared with that produced in the first reaction.

(iii) Other Sources

Other sources of tritium in "JOYO" involve fission in natural uranium ($\leq$ 3.5ppm), activation of lithium ($\leq$ 20ppm) and boron ($\leq$ 4ppm), which are impurities existing in the sodium coolant.

Main production of tritium in the sodium coolant is due to the activation reactions of lithium, which produce about half as many of tritium atoms compared with that produced by fast fission.

For "JOYO", following assumptions are made in the evaluation of radiation injury due to tritiuum release.

(a) Only 1 % of the tritium, produced in the core, is released from the stuck of "JOYO" into the environment as gaseous radioactive wastes.

(b) Only 10^{-2} % of the tritium, produced in the core, is discharged into the sea as liquid wastes.

(c) Concerning the hazard evaluation of gaseous tritium released from the stuck, the external dose as well as the internal dose, due to inhalation, absorption through skin and intake from foods have been considered.

One of the most important problems concerning design, operation and maintenance for the radiation shielding of "JOYO" is the source evaluation of ^{14}C, ^{41}Ar, ^{22}Na, ^{24}Na, etc.

Carbon-14 is produced by neutron activation of nitrogen gas, due to a reaction of $^{14}N(n, p)^{14}C$ which takes place both in the void space inside the safety vessel and in the double wall annulus gap space of the reactor vessel, which is used for preheating of the reactor vessel.

Multi-group and multi-dimensional calculation must be performed in order to evaluate quantity of carbon-14 generated at various places of the reactor, far away from the core. However, such calculation is not simple, because "JOYO" has many layers of materials and some of them are of complicated geometry.

As for the accuracy in estimating neutron flux distribution of various part of the reactor, in particular at such a large distance from the core, there exist still considerably large uncertainties in its analytical results.

Therefore the more reliable analytical method as well as the improved nuclear data for the reaction of $^{14}N(n, p)^{14}C$ are highly desirable. We are hoping that it will be possible to collect meaningful information on the generation of carbon-14 in "JOYO" by the

operational experience of the reactor.

4.6 Diffusion of Radioactive Cloud

Various method have been reported for estimating the concentration of radioactive materials released from the point source to the atmosphere. Of those, the so-called "English Method" is widely used in hazard evaluation in Japan. This method is based on the assumption that both the lateral and vertical concentration distributions due to the atmospheric turbulence are Gaussian, and uses the diffusion parameters proposed by F. Pasquill.

The ground-level exposure-rate by the gamma-rays from a radioactive cloud is basically given by the next equation.

$$D(x_0, y_0, 0) = \frac{K_0}{4\pi} \mu_a E \int_0^{\infty}\int_0^{\infty}\int_{-\infty}^{\infty} \frac{B(E, r)}{r^2} e^{-\mu r} \cdot \chi \cdot dy \cdot dz \cdot dx \quad \cdots\cdots (3)$$

where

$D(x_0, y_0, 0)$: exposure-rate (μR/hr)

K_0 : constant converting activity to exposure-rate, and

$$K_0 = 2.03 \times 10^9 \left(\frac{\text{dis.m}^3.\mu\text{R}}{\text{Mev.Ci.hr}}\right)$$

E : primary energy of a photon (Mev/dis)

μ : total linear absorption coefficient (m^{-1})

μ_a : true linear absorption coefficient (m^{-1})

$B(E, r)$: dose-build-up factor

X_0 and Y_0 : coordinates of receptor (m)

χ : concentration of a gamma-emitter in air (Ci/m^3) and it is assumed to be of the general Gaussian type

r : distance between the receptor and a volume element of the cloud (m)

Above equation was integrated numerically, using Salzer-Zucker's, Gauss' and Neuton-Cotes' formulae.

In order to calculate the radiation injury to the public during the normal reactor operation, various meteorological data have been prepared; such as the mean inverse wind speed, the occurrence frequency of the six stability conditions (from A to F), and the occurrence frequency of the sixteen wind directions, at O-arai site based on the measured data obtained in the recent three years (from 1971 to 1973).

In case of the continuous release of gaseous radioactive wastes for the evaluation of the gamma dose, we have considered not only the effect of the wind fluctuation in the sector at the given direction but also the contribution from the sector at the neighbouring direction.

In case of the intermittent release, stability type D is assumed and the maximum released frequency at given direction is determined so that the confidence level of the binominal distribution for the occurence probability of the wind direction may exceed 97 %.

For the evaluation of the beta dose, the submersion model is used.

As for the weather conditions used for the hazard evaluation of accidents, we have already described it in section 3.3.

4.7 Fuel-Sodium Interaction

As described above, we have performed more refined and elaborate accident analysis that predict various phases of the progression from accident initiation to hypothetical core disassembling. Those include the fuel-sodium interaction and the pressure propagation in the primary-coolant systems. It has been concluded that the recriticality accident, considered as a hypothetical accident at the safety evaluation

for the first phase reactor power of "JOYO", is a more pessimistic evaluation compared with that resulted from the current elaborate model. Continuous withdrawal of a control rod without scram was considered as an accident initiation and the reactivity insertion rate at the prompt critical was determined.

The analysis of the excursion accident was performed using "VENUS" code and the effective working energy, the amount of the molten fuel and the mean temperature of the molten fuel, which are to be used as input data in the analysis of the fuel-sodium interaction were obtained.

A computor code for analysing fuel-sodium interaction has two subroutines. One calculates the pressure caused by the fuel-sodium interaction and the other calculates the response of the core structure, reactor vessel and rotating plug. The former program was named "SUGAR" code which was developed based on the modified Cho-Wright model, by revising it for evaluating the temperature distribution of the molten spherical fuel more precisely. The latter program was named "CAMEL" code which was developed base on the pseudo-two-dimensional analysis code of the structure response, named "ASPRIN", by revising it for calculating the pressure propagation from the mixing zone to the core structure zone taking into account the compressibility of sodium coolant.

The fundamental assumption adopted in the analysis are as follows.

(a) Spherical molten fuels are mixed with sodium uniformly in space.

(b) Change of fuel volume is ignored.

(c) Liquid sodium begins boiling when the shock wave pressure due to the thermal expansion of sodium decreases below the saturated vapor pressure of sodium.

(d) Heat transfer and mass transfer between the mixing region, where fuel-sodium interaction occurs, and the surrounding region are ignored.

(e) Cylindrical geometry of mixing region where fuel-sodium interaction occurs, is assumed.

(f) The surrounding sodium in the radial direction is compressible and the sodium slug in the axial direction is incompressible.

(g) Heat transfer between fuel and sodium is expressed as follows.

$$h = h_0 [1-\exp(t/\tau)] \quad \cdots\cdots\cdots \quad (4)$$

where

h_0 : heat transfer coefficient in case of perfect mixing of fuel particles with sodium.

τ : fragmentation delay time for fuel particles τ is assumed 5 m sec.

h : infinite (if sodium is liquid phase)

: $1(w/cm^2\ {}^{o}k)$ (if sodium is two phase)

: $0.2(w/cm^2\ {}^{o}k)$ (if sodium is gas phase)

(h) Mass ratio of fuel to sodium is equal to that of normal composition of the core.

(i) Spherical radius of molten fuel is assumed to be 0.01 cm.

(j) The mechanical strength of the neutron shields and of the core barrel are taken into consideration for the analysis of the pressure propergation.

As the results, the maximum residual plastic strain is about 0.9 % (Mark-I, 75 MW), 0.4 % (Mark-II, 100 MW) at the trunk of the reactor vessel and about 8.7 % (Mark-I, 75 MW), 3.7 % (Mark-II, 100 MW), at just below the flange of the reactor vessel. These values

are well within the design limits of those structures. The analytical models of "SUGAR-CAMEL" seem to have following two weak points.

(i) As the treatment of expansion of the region where fuel interacts with sodium is inaccurate, calculated pressure propagation has some uncertainties.

(ii) The shape of the deformation of the structures should be given a priori.

In order to evaluate adequacy of the analytical models of "SUGAR-CAMEL", an accurate deformation pattern of the vessel has been calculated using "SUGAR-PISCES" code, which is a combined code of the program for calculating fuel-sodium interaction, named "SUGAR", with the program for calculating two-dimensional structure response, named "PISCES".

In consequence, it was found that the maximum residual plastic strains obtained by "SUGAR-CAMEL" code were about 1/8 at the trunk of the vessel and about 1/3 at just below the flange of the vessel, compared with those obtained by "SUGAR-CAMEL" code calculation. We have also performed the calculation of the pressure propagation through the primary coolant pipings, the intermediate heat exchangers and primary pumps, which involve not only main coolant systems but also anxiliary coolant system. It was concluded finally that reactor vessel, primary pipings, intermediate heat exchangers, primary pumps cannot be failed as described previously.

4.8 Liquid Waste Disposal

Following assumption are made for the evaluation of the radiation injury due to the release of liquid radioactive wastes.

(i) The amount of the liquid radioactive wastes released from "JOYO" into the seawater is 10 mCi/yr

(ii) The nuclide composition of the wastes are:

^{60}Co : 40%
^{90}Sr : 20%
^{137}Cs : 40%

(iii) The radiation injury to be caused by the wastes, due to intake of the marine products, is evaluated for total body, bone, liver and gastrointestinal tract.

(iv) The amount of intake of the marine products is as follows.

fish	-----------	200 g/day
mollusca	------	10 g/day
crustacean	----	10 g/day
seaweeds (raw	-	40 g/day (3 months)
dry	-	10 g/day (9 months)

5. CONCLUSION

As described above, the detailed evaluation relating to the safety analysis both for the second phase reactor power (Mark-I, 75 MW) and for the third phase reactor power (Mark-II, 100 MW) has been conducted from Oct. 1973.

In this paper, major technical topics resulted from the recently conducted safety evaluation for these power increases have been discussed, comparing with the safety evaluation made for the first phase reactor power.

By the series of works recently conducted for the safety evaluation of "JOYO", it was concluded that favourable results were obtained. However, works for such safety evaluation will be continued further more in order to assure safe operation of "JOYO".

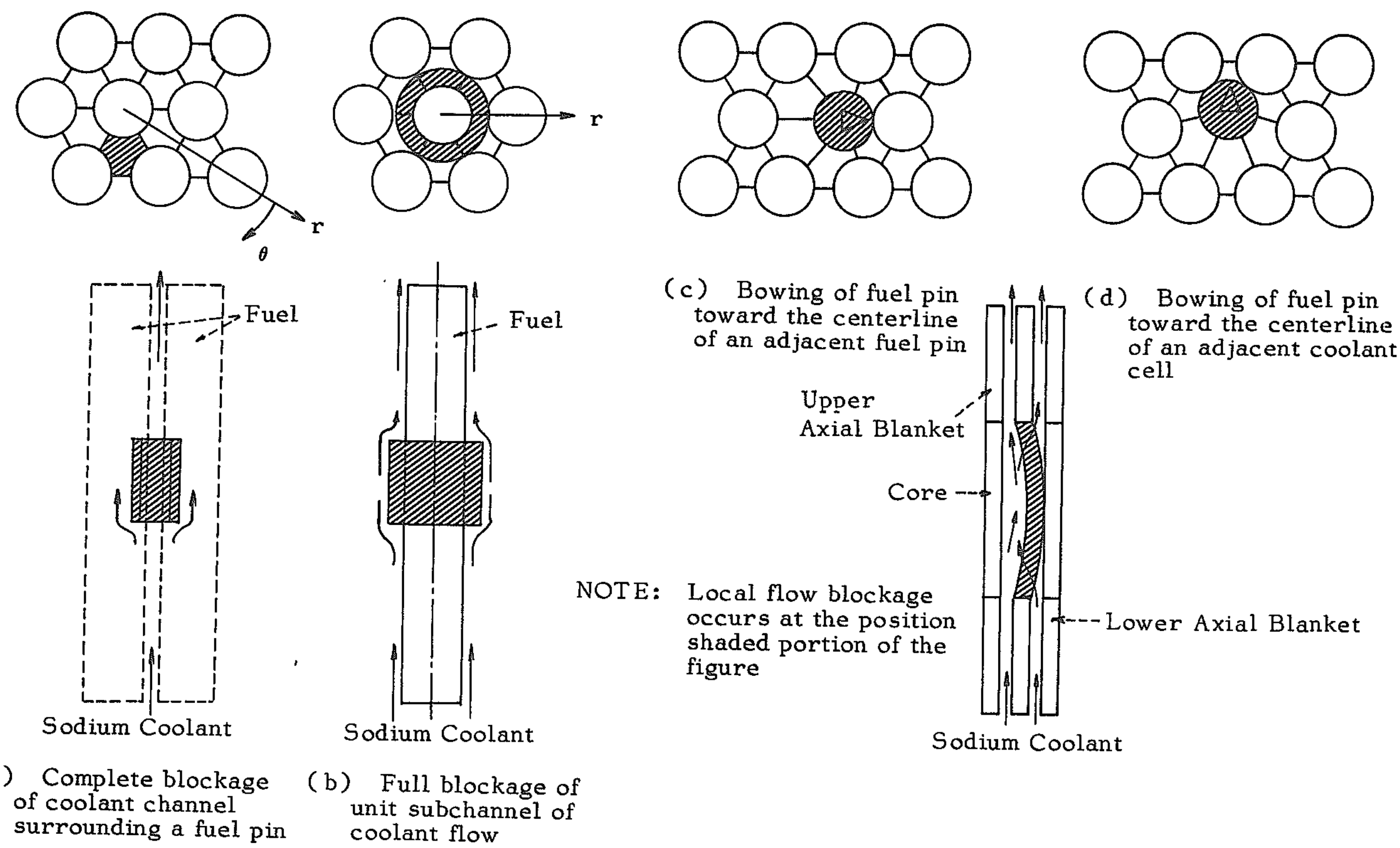

Fig. 1 ANALYTICAL MODEL OF LOCAL CHANNEL BLOCKAGE

Table 1 Reactivity Feedback Coefficients of "JOYO"

Reactivity Feedbacks \ Reactor power	Mark-I		Mark-II
	50 MWt (*1)	75 MWt (*2)	100 MWt (*3)
Doppler Coefficient ($T\frac{dk}{dT}$)	-1.0×10^{-3} ~ -3.0×10^{-3} $\Delta k/k$	-0.6×10^{-3} ~ -2.5×10^{-3} $\Delta k/k$	-0.37×10^{-3} ~ -2.22×10^{-3} $\Delta k/k$
Linear axial expansion coefficient of fuel pin	-1.5×10^{-6} ~ -4.0×10^{-6} $\Delta k/k/°C$	-2.0×10^{-6} ~ -4.6×10^{-6} $\Delta k/k/°C$	-2.5×10^{-6} ~ -5.6×10^{-6} $\Delta k/k/°C$
Expansion coefficient of cladding	-0.1×10^{-6} ~ -0.2×10^{-6} $\Delta k/k/°C$	-0.59×10^{-6} ~ -1.2×10^{-6} $\Delta k/k/°C$	-0.63×10^{-6} ~ -1.6×10^{-6} $\Delta k/k/°C$
Expansion coefficient of coolant	-5.0×10^{-6} ~ -10.0×10^{-6} $\Delta k/k/°C$	-5.9×10^{-6} ~ -12.3×10^{-6} $\Delta k/k/°C$	-5.7×10^{-6} ~ -14.0×10^{-6} $\Delta k/k/°C$
Expansion coefficient of lower core supporting plate	-7.0×10^{-6} ~ -15.0×10^{-6} $\Delta k/k/°C$	-6.2×10^{-6} ~ -12.8×10^{-6} $\Delta k/k/°C$	-11.3×10^{-6} ~ -18.9×10^{-6} $\Delta k/k/°C$

(Note)

(*1) These values were submitted to the Safety Committee for the first phase reactor power (50 MW).

(*2) These values were used for the safety evaluation of the second phase reactor power (75 MW).

(*3) These values were used for the safety evaluation of the third phase reactor power (100 MW).

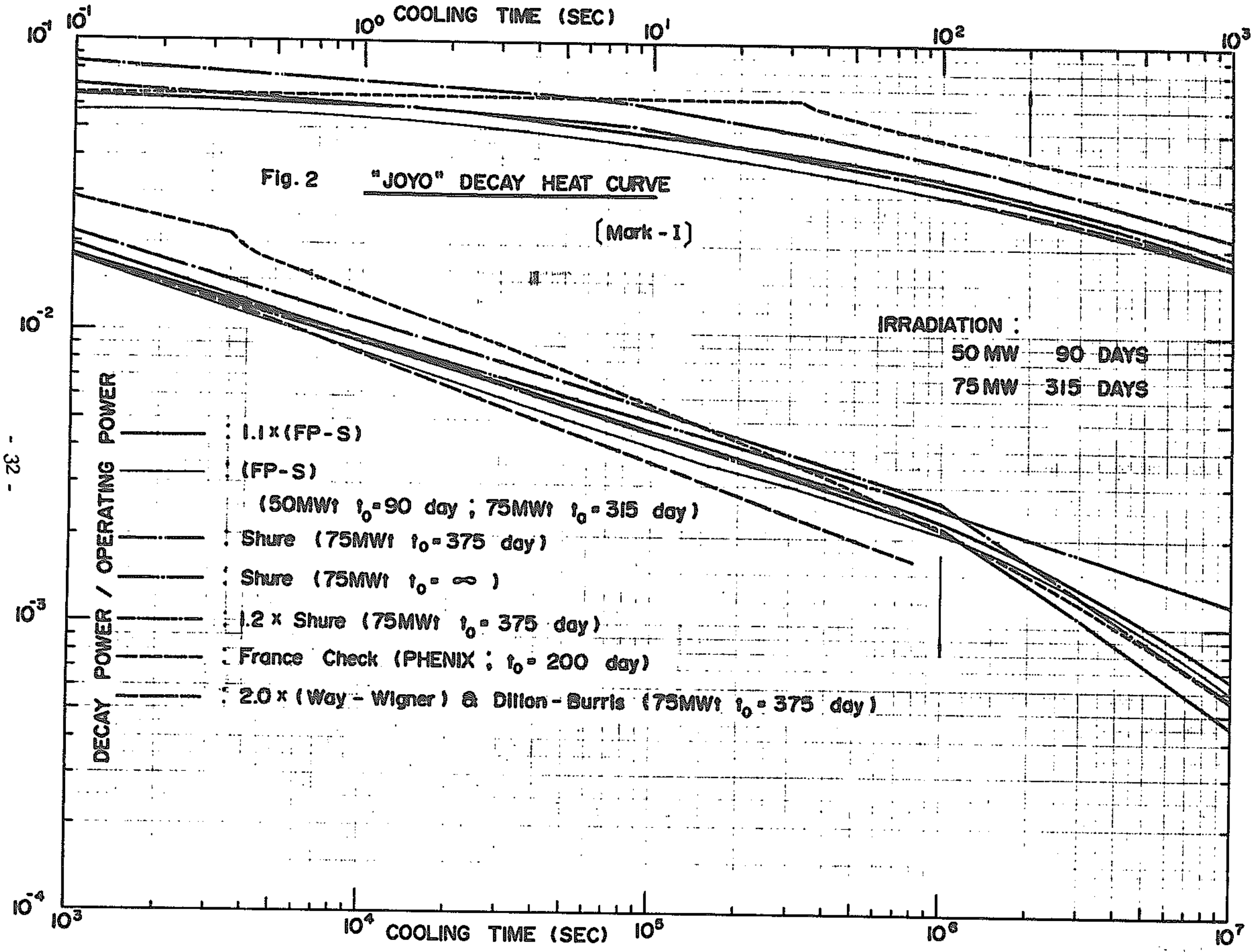

Fig. 2 "JOYO" DECAY HEAT CURVE

(Mark - I)

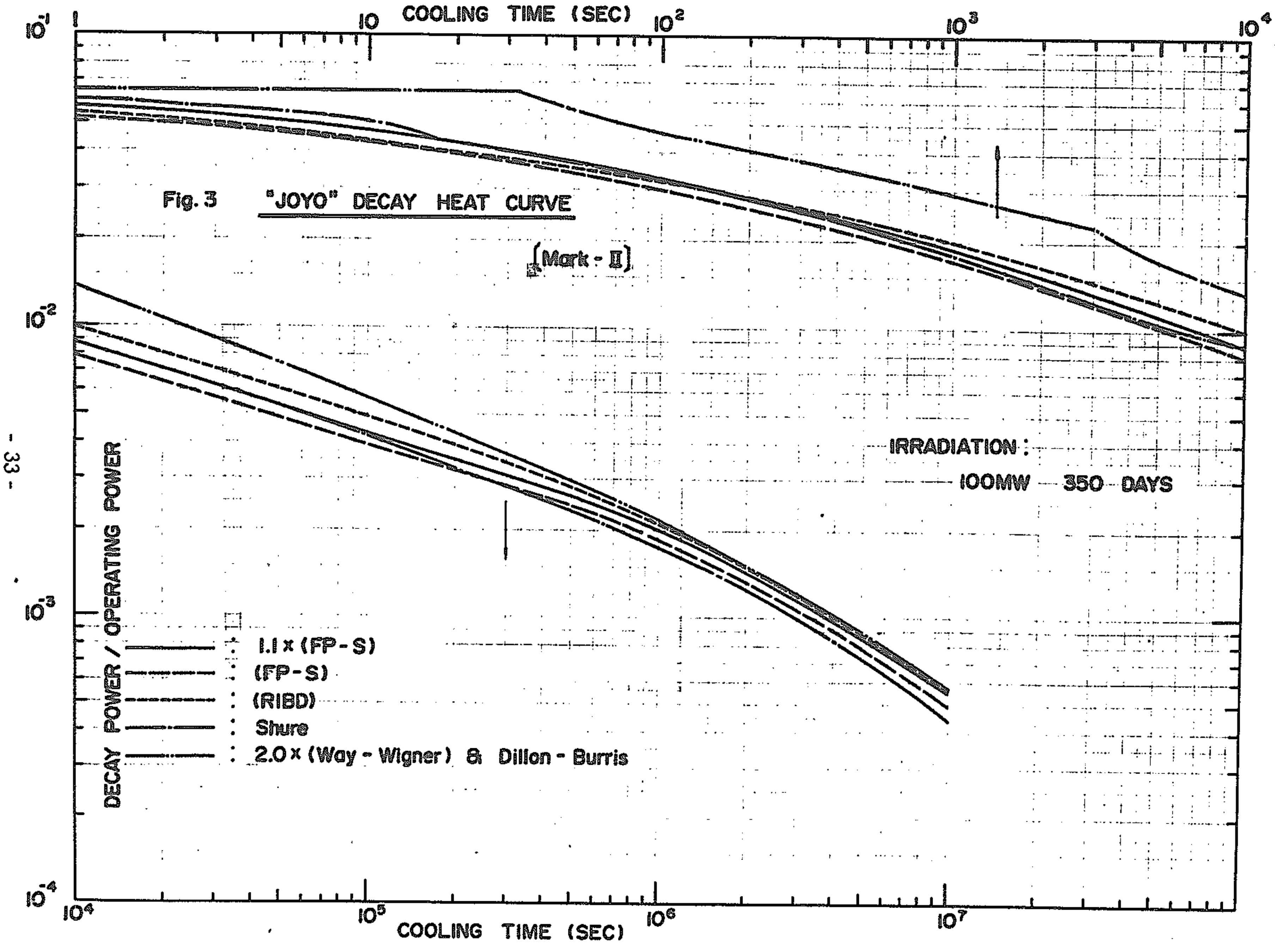

Fig. 3 "JOYO" DECAY HEAT CURVE

(Mark - II)