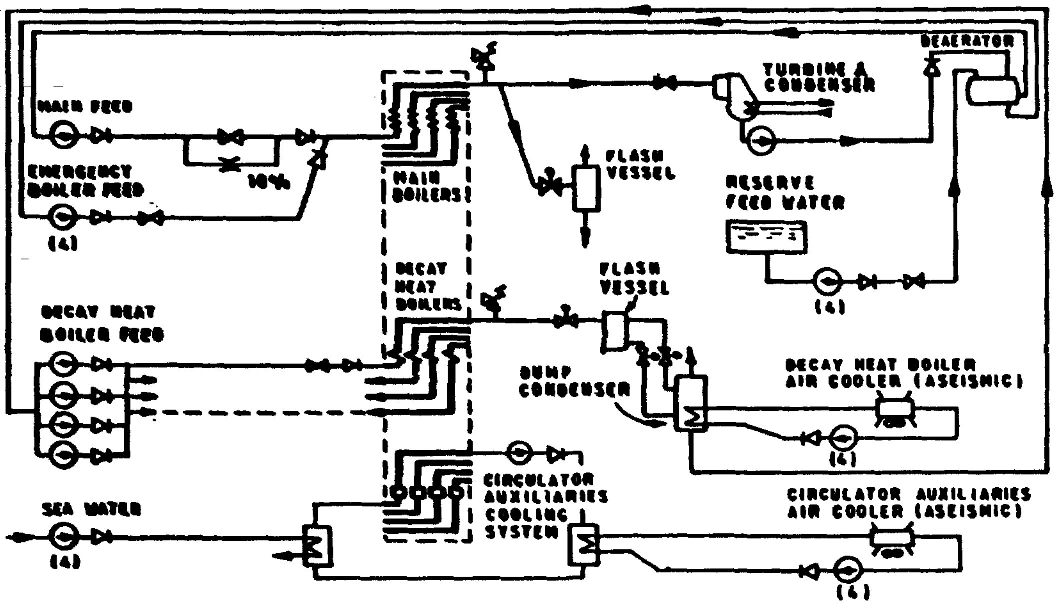

COOLING SYSTEMS (SCHEMATIC) FIG 4

(DECAY HEAT BOILER SYSTEM EXTENDED IN DETAIL. NUMBERS INDICATE PUMP REDUNDANCY)

THE REACTOR SAFETY STUDY OF EXPERIMENTAL MULTI-PURPOSE VHTR DESIGN

T. YASUNO, S. MITAKE, M. EZAKI, K. SUZUKI
Japan Atomic Energy Research Institute
Tokai Research Establishment
Tokai-Mura
Naka-Gun
Ibaraki-Ken
Japan

1. Introduction

Over the past years, the design works of the Experimental Very High Temperature Reactor (VHTR) plant have been conducted at Japan Atomic Energy Research Institute. The conceptual design has been completed and the more detailed design works and the safety analysis of the experimental VHTR plant are continued.

The purposes of design studies are to show the feasibility of the experimental VHTR program, to specify the characteristics and functions of the plant components, to point out the R & D intems necessary for the experimental VHTR plant construction, and to analyze the feature of the plant safety.

The experimental VHTR must be provided with the function as follows.

- Demonstration test for future nuclear process heat applications.
- Irradiation test for development of fuels and materials for high-temperature use.
- Confirmation test for large VHTR plant safety.

The design conditions of experimental VHTR were determined due to considerations of the three functions mentioned above. These conditions are summarized in Table 1. Thermal power of 50 MWT is considered combinient for doing the demonstration test, and the coolant temperature 1000°C at the outlet of the reactor is required from a nuclear heat utilization system such as a direct steel making.

In this paper the summary of system design and safety features of the experimental reactor are indicated. Main issues are the safety philosophy for the design basis accident, the accidents assumed and the engineered safety systems adopted in the design works.

2. Outline of the experimental VHTR design

The system diagram of the experimental VHTR is shown in Fig.1. The reactor cooling system consists of two cooling circuits in symmetric arrangement on the right and left sides. Each primary circuit are connected to the secondary circuits with intermediate heat exchanger. The heat generated by the reactor is removed in the steam generators installed in the secondary circuits or supplied to the heat utilization component such as a steam reformer arranged in parallel to the steam generator. The helium gas temperature at the outlet on the secondary side of the intermediate heat exchanger was decided 930°C from the requirement of the available heat utilization systems. Auxiliary cooler installed in parallel to the intermediate heat exchanger is used when the cooling capacity of the main cooling system is lowered, in addition to the cases of refueling and shutdown of the reactor. Table 2 summarizes the main parameters of the experimental VHTR plant.

The core structure of reactor is illustrated in Fig.2. The core consists of 73 columns and the fuel column consists of seven elements piled up. Reflectors are arranged outside the core and also graphite block reflectors are placed at the top and bottom of core. Figure 3 indicates the fuel element. The burnable poison made of B_4C particles and graphite powder is loaded at the corners of the standard fuel elements.

The helium coolant flows in the core from the coolant inlet tube of coaxial structure at the lower part of the reactor vessel and arrives at the reactor upper plenum. And the helium flows through the cooling channel of the fuel column. The average temperature of helium mixed in the lower plenum is 1000°C.

A pair of control rods is installed in every seven columns. There are 14 rods (7 pairs) in the core and 24 rods (12 pairs) in the reflector part. The control rod drive mechanism is placed in the stand pipe at the pressure vessel head.

3. Safety design of the experimental VHTR

In the current study of the experimental VHTR design, the spectrum of accidents and transients is analysed. The examples of accidents for the experimental VHTR are listed in Table 3. The safety fearturs of the experimental VHTR are as follows.

- The integrity of pressure boundary of the primary coolant is postulated for fission product containment.
- The pressure difference betweenthe primary and the secondary cooling systems in the normal operation is maintained in order to prevent fission product leakage from the primary to the secondary cooling system.
- The high reliability of reactor shut-down system such as the control rod and the reserve shut-down mechanism are postulated.
- The reactor plant is secured against possible accidents of heat utilization system combined to the reactor.

The selected design basis accidents for the engineered safety system of the experimental VHTR are representative of three fundamental events; fission product release, loss of cooling and air ingress. Table 4 summarizes this classification and also identifies the engineered safety system provided for the design basis accidents.

As shown in Table 4, the posibility of reactivity insertion resulting from control rod ejection or the core dropping is precluded by the design. The orifice set at the bottom of the control rod channel prevents to generate enough withdrawal force to float control rod in the rapid upflow of coolant through the core following the primary system depressurization.

And also the anchor device of the control rod drive mechanism in the stand pipe is designed to prevent the control rod drive mechanism from launching through the failed closure.

In our design, however, a hypothetical withdrawal of a single control rod pair is assumed to restrict the maximum worth of control rod.

Core drop is precluded by the integrity of core support with the protection system described below.

(1) Isolation

At the rapid depressurization accident of the primary coolant system following secondary pipe rupture outside the reactor containment building the isolation of primary system is importants to prevent a release of the primary coolant contaminated by fission products. Because the heat transfer tubes of the intermediate heat exchanger are designed with the pressure difference between the primary and the secondary cooling circuits, the tube breaks must be assumed in the case of secondary coolant depressurization. The amount of the primary coolant release is mainly depend on the number of IHX failed tube and the shutoff time of isolation valve. To determine the shutoff time of the isolation valve, resulting doses are calculated. Fig.4 shows the dose at the site boundary in the case of the accident. From the comparision of this results and the guideline of plant site selection the shutoff time required for the isolation valve is specified as less than 10 Sec. Although the guideline of plant site selection was developed for safety assessment of light water reactors, it is applicable to all type of nuclear plants to assure the health and safety of the public. The design conditions and the structure of the isolation valve are shown in Table 5 and Fig.5 respectively.

(2) Cooling

The auxiliary cooling system consists of two cooling circuits in parallel to main cooling circuits respectively.

Each auxiliary cooling circuit is capable of cooling the core in the case of loss of forced circulation due to primary or secondary circulator stick. The specification of the auxiliary cooling system are summerized in Table 6, and the typical result of transient analysis is indicated in Fig.6. In this analysis the trip of a primary circulator was assumed, and the emergency reactor shutdown with the signal of abnormal coolant flow rate delayed about 9 sec. Fig.6 shows that the decay heat is removed by the auxiliary cooling system starting up at 5 minutes after the circulator trip.

In order to assess the capability of reserve cooling system, primary cooling pipe rupture is analyzed. Fig.7 shows the result of transient analysis of the maximum temperature in the reactor following the failure of the primary coolant boundary.

In this analysis the pressure in the reactor vessel is assumed to be about 1 kg/cm^2a and the natural circulation of coolant is assumed in the theoretical model for the calculations of heat transfer in the reactor. The temperature of reactor vessel during accident depends on the natural circulation, of which the flow directions are upward in the core and downward at the core barrel. Due to natural circulation the temperature of the vessel top becomes high gradually. The flow rate of natural circulation calculated under the low pressure condition is less than about 10^{-2} % of that at full power operation. The maximum temperatures are suppressed under the limited value by the operation of the reserve cooling system with the water tubes surrounding the reactor vessel. The concept of the reserve cooling system is indicated in Fig.8. The decay heat is removed from the reactor vessel surface by radiation and air natural convection.

(3) Air ingress

In a hypothetical accident such as the primary cooling pipe rupture, the protection against the core graphite oxidation by air ingress must be considered besides the core cooling. The inert gas system prevents the core support graphite from being oxidized by the air ingressed after the primary pipe rupture accident. Fig.9 indicates the result of analysis on oxidation of the core support post graphite. The air is inhaled into the core with the natural circulation through the ruptured pipe.

The nitrogen gas is injected into the lower plenum from the nozzle inserted near the support post to produce an inert gas atmosphere. The main specifications of the inert gas system adopted as one of the engineered safety system in the experimental VHTR design are summarized in Table 7. All of the engineered safety system adopted in the design are illustrated in Fig.10.

4. Discussions

The engineered safety system adopted in the experimental VHTR are described.

Nevertheless the results of the preliminary analysis have indicated that the design of these system is accepted by the guideline, the more detailed studies must be pursued to verify the analyses.

In order to complete the safety analysis report of the experimental VHTR, the various kinds of research and development works are needed to obtain the informations and data for the assessment of the engineered safety system.

The main activities for these systems are as follows.

- Demonstration experiments showing the anchor device is effective to prevent the control rod ejection.
- Intermediate heat exchanger tube test showing the integrity under accident condition.
- High temperature tests of the isolation valve.
- Performance test of the reserve cooling system.
- Verification of the computer program used in the analyses.

In conformity with the reactor design, the research and development program is in progress at Japan Atomic Energy Research Institute.

TABLE 1. BASIC DESIGN CONDITIONS OF THE EXP. VHTR

REACTOR THERMAL OUTPUT	50 MW
REACTOR OUTLET COOLANT TEMPERATURE	1000 °C
REACTOR INLET COOLANT TEMPERATURE	THE TEMPERATURE MUST BE DETERMINED WITH REGARD TO SPECIFICATIONS OF THE REACTOR VESSEL AND HELIUM CIRCULATOR AS WELL AS TO THERMAL CHARACTERISTICS OF THE CORE
FUEL	UO^2 KERNEL, COATED FUEL PARTICLE GRAPHITE MATRIX DISPERSION-TYPE
FUEL ELEMENT TYPE	PIN-IN-BLOCK TYPE
DIRECTION OF COOLANT FLOW	DOWNWARD-FLOW THROUTH THE CORE
PRESSURE VESSEL	STEEL
NUMBER OF PRIMARY COOLING CIRCUITS	2
HEAT TRANSMISSION	INDIRECT (ADOPTION OF IHX)
PRIMARY COOLANT PRESSURE	40 KG/CM^2G
SECONDARY COOLANT PRESSURE	HIGHER THAN PRIMARY COOLANT PRESSURE
COMPONENTS IN THE SECONDARY CIRCUIT	HEAT REMOVAL COMPONENTS HEAT UTILIZING COMPONENTS

TABLE 2. MAIN PARAMETERS OF VHTR PLANT

REACTOR THERMAL OUTPUT	50 MW
REACTOR VESSEL OUTLET / REACTOR VESSEL INLET TEMPERATURE	1000/395 °C
COOLANT PRESSURE	40 KG/CM^2G
PRIMARY COOLING CIRCUITS	2 LOOPS
FUEL TYPE	LOW ENRICHED UO_2, COATED PARTICL PRISMATIC BLOCK-PIN TYPE
PRESSURE VESSEL	STEEL
CORE DIMENSIONS (DIAMETER/HEIGHT)	2.7/4.0 M
NUMBER OF FUEL ELEMENTS (COLUMNS/BLOCKS)	511 (73/7)
NUMBER OF CONTROL RODS	19 PAIR
PRESSURE VESSEL INNER DIAMETER	5.95 M
POWER DENSITY (AVEG./MAX.)	2.2/5.7 W/CM^3
FUEL INVENTORY (AVERAGE ENRICHMENT)	1.74 ON U (3.2 W/O)
BURN UP (AVEG.)	10,000 MWD/T
INTERMEDIATE HEAT EXCHANGER	HELICAL COIL TYPE
NUMBER/LOOP, THERMAL OUTPUT	1, 25 MW
AUXILIARY COOLER	SHELL AND TUBE TYPE
NUMBER/LOOP,	1
STEAM GENERATOR	SHELL AND TUBE TYPE
NUMBER/LOOP, THERMAL OUTPUT	1, 25 MW
HELIUM CIRCULATOR NUMBER/LOOP	CENTRIFUGAL TYPE, 1
AUXILIARY HELIUM CIRCULATOR NUMBER/LOOP	CENTRIFUGAL TYPE, 1

TABLE 3. EXAMPLES OF ACCIDENTS FOR EXP. VHTR

- TRIP OF A PRIMARY OR A SECONDARY CIRCULATOR
- TRIP OF A FEEDWATER PUMP
- LOSS OF OFFSITE ELECTRIC POWER
- A CONTROL ROD WITHDRAWAL BY AN OPERATOR ERROR
- CORE COOLANT CHANNEL PARTIAL BLOCKABE BY FAILURE
- NON-CONTROLLED WITHDRAWAL OF A CONTROL ROD
- ROD DROP FROM HIGH WORTH TO LOW WORTH REGION OF CORE
- STICK OF A PRIMARY OR A SECONDARY CIRCULATOR
- STICK OF A FEEDWATER PUMP
- CORE BYPASS OF COOLANT FLOW
- IHX TUBE BREAK
- PRIMARY COOLANT SYSTEM LEAK WHICH REQUIRES REACTOR SHUT DOWN WITH NORMAL PROCEDURE
- SECONDARY COOLANT SYSTEM LEAK WHICH REQUIRES REACTOR SHUT DOWN WITH NORMAL PROCEDURE
- AUXILIARY COOLER TUBE LEAK
- PRIMARY COOLING PIPE RUPTURE IN THE CONTAINMENT
- SECONDARY COOLING PIPE RUPTURE
- FAILURE OF FP ADSORPTION BED IN THE COOLANT PURIFICATION SYSTEM

TABLE 4. CLASSIFICATION OF ACCIDENTS FOR EXP. VHTR

CLASSIFICATION	ENGINEERED SAFETY SYSTEM	DESIGN BASIS ACCIDENT
CONTAINMENT SYSTEM	REACTOR CONTAINMENT	PRIMARY COOLING PIPE RUPTURE
	ANNULUS PURGE SYSTEM	PRIMARY COOLING PIPE RUPTURE
	AIR PURIFICATION SYSTEM IN CONTAINMENT	PRIMARY COOLING PIPE RUPTURE
	ISOLATION VALVE	SECONDARY COOLING PIPE RUPTURE
EMERGENCY CORE COOLING SYSTEM	AUXILIARY COOLING SYSTEM	LOSS OF PRIMARY OR SECONDARY FORCED CIRCULATION
	RESERVE COOLING SYSTEM	PRIMARY COOLING PIPE RUPTURE CORE BYPASS OF COOLANT
INERT GAS SYSTEM	N_2-GAS INJECTION SYSTEM	PRIMARY COOLING PIPE RUPTURE

TABLE 5. DESIGN CONDITIONS OF ISOLATION VALVE

LOCATION	INSIDE AND OUTSIDE OF CONTAINMENT
FLUID	HE
TEMPERATURE	960 °C
PRESSURE	52 KG/CM^2G
FLOW RATE	30,300 KG/H
PRESSURE LOSS	0.1 KG/CM^2
DRIVE MECHANISM	AIR CYLINDER
SHUTOFF TIME	7 SEC

TABLE 6. AUXILIARY COOLING SYSTEM

- FUNCTION	REMOVAL OF DECAY HEAT DURING NOMAL SHUTDOWM PERIOD AND DURING ABNORMAL CONDITION SUCH AS THE MAIN CIRCULATOR STICK
- HE FLOW RATE	10% OF THE MAIN LOOP FLOW RATE
- TYPE OF COOLER	SHELL AND U-TUBE TYPE HE-WATER HEAT EXCHANGER

TABLE 7. INNERT GAS SYSTEM

- FUNCTION	PROTECTION AGAINST THE CORE GRAPHITE OXIDATION FOLLOWED AIR INGRESS
- DBA	PRIMARY COOLANT PIPE RUPTURE
- SYSTEM	N_2-GAS HOLDER, INJECTION NOZZLE
- PERFORMANCE	FLOW RATE 25 NM^3/HR OPERATION TIME 30 DAY

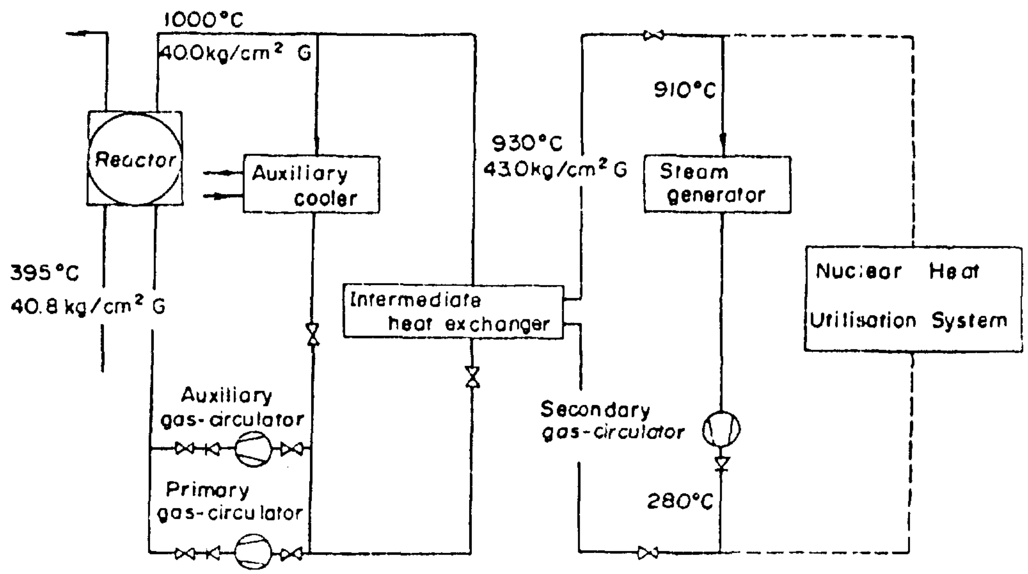

FIG. I SYSTEM DIAGRAM OF EXP. VHTR

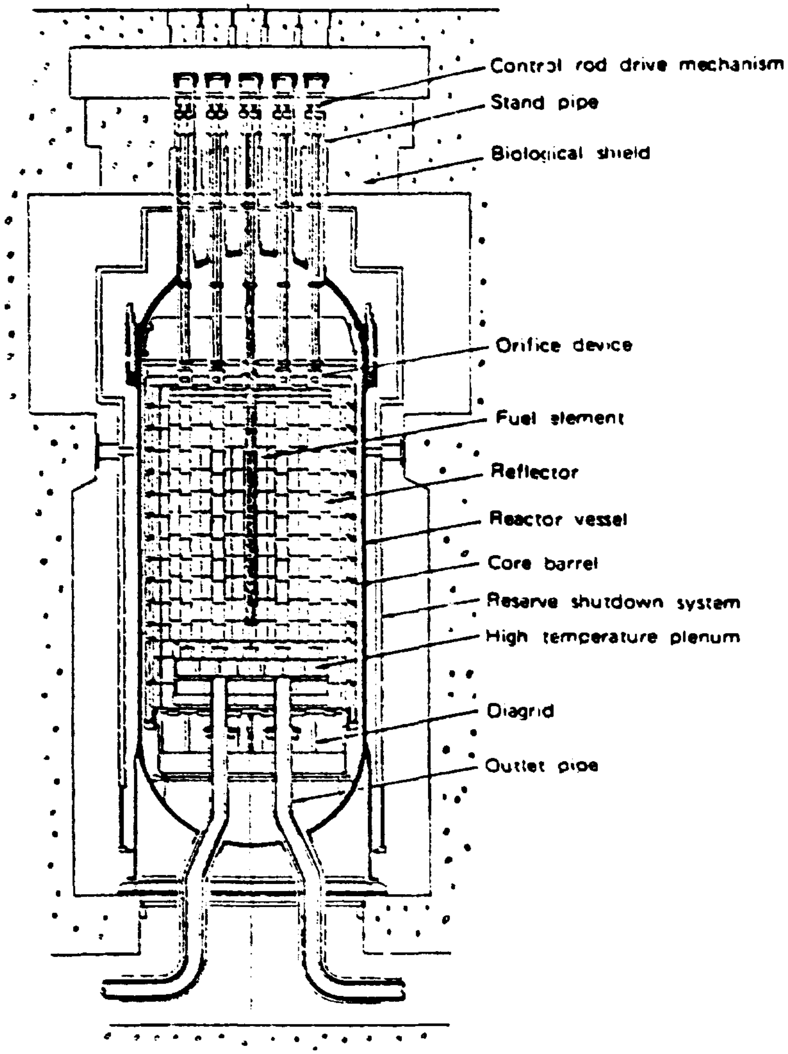

FIG. 2 EXPERIMENTAL VHTR

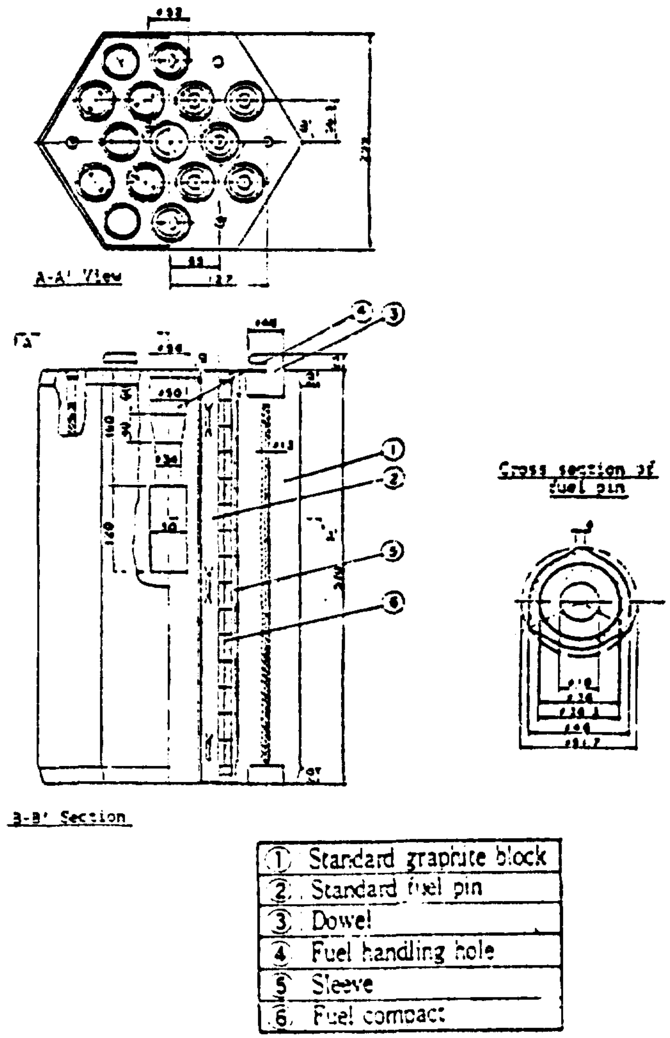

FIG.3 STANDARD FUEL ELEMENT

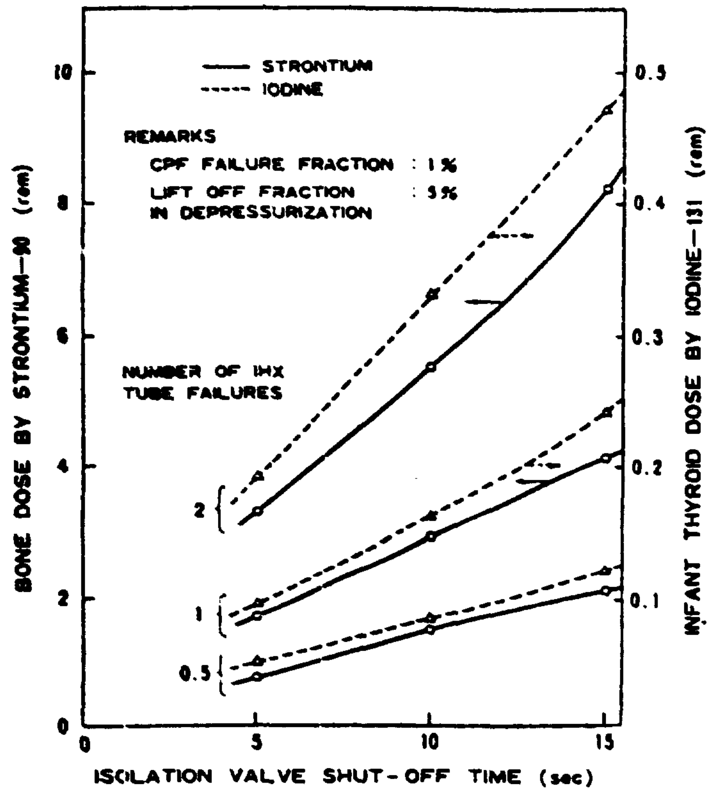

Fig. 4 Dose at the site boundary following secondary pipe rupture

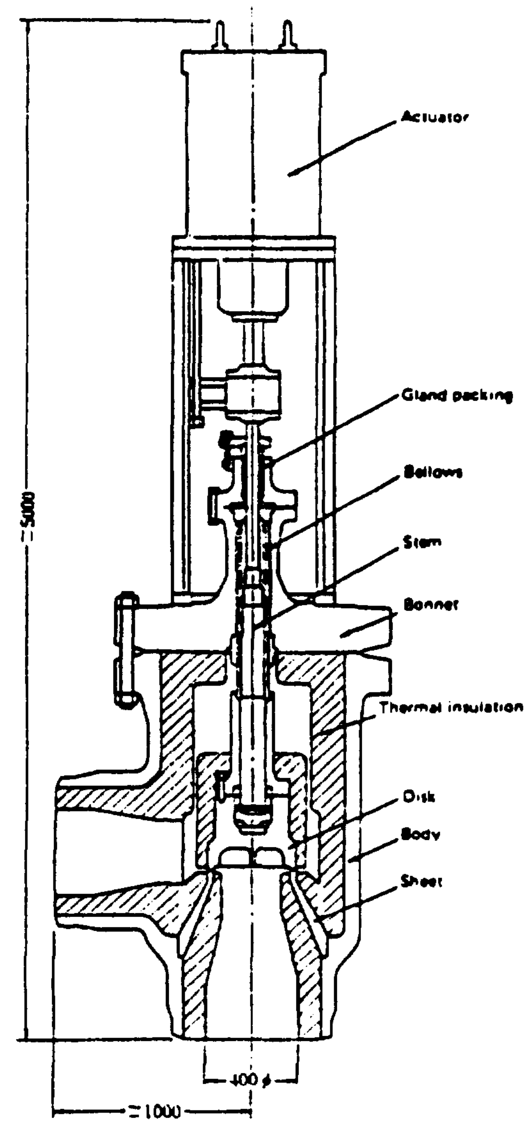

Fig. 5 High temperature isolation valve

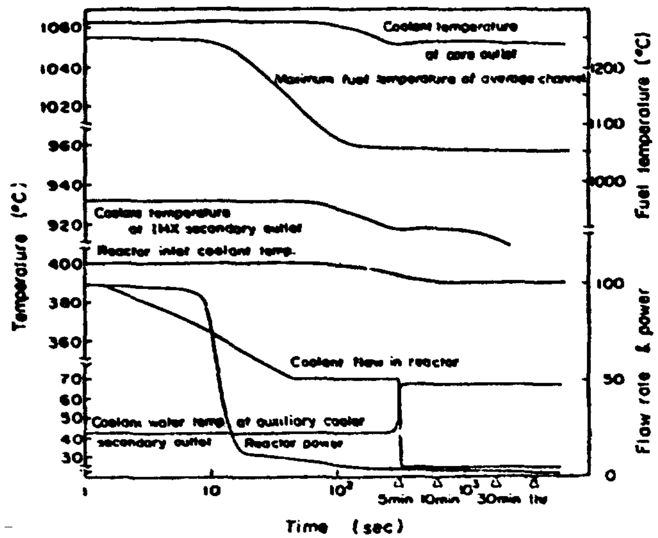

FIG. 6 TRANSIENTS OF POWER AND TEMPERATURE FOLLOWING THE LOSS OF FORCED CIRCULATION OF PRIMARY LOOP

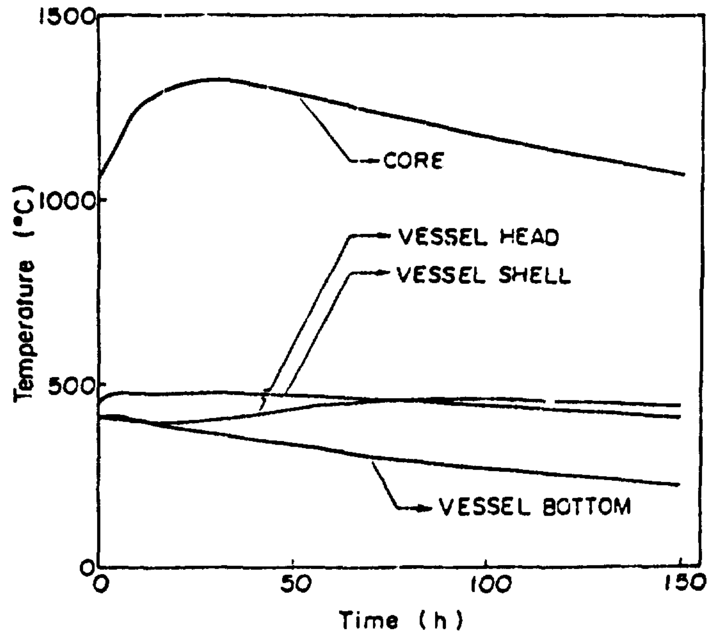

FIG. 7 TRANSIENTS OF MAXIMUM TEMPERATURE IN THE REACTOR FOLLOWING THE FAILURE OF PRIMARY COOLANT BOUNDARY

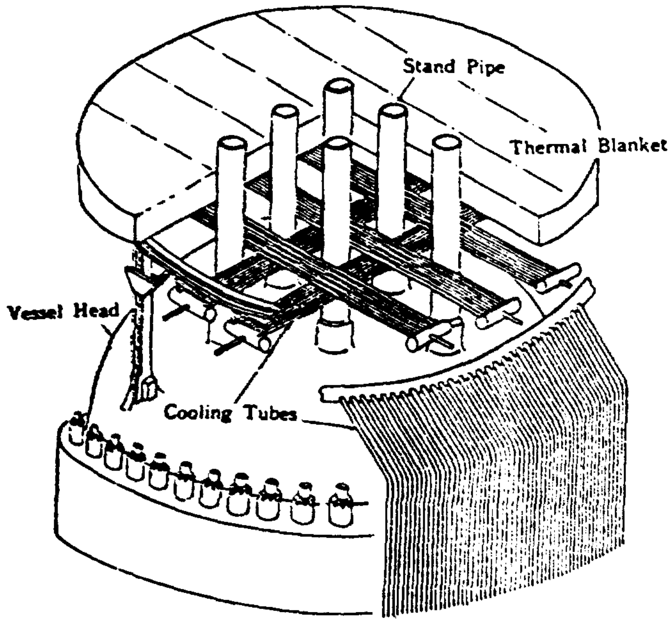

FIG. 8 A CONCEPT OF RESERVE COOLING SYSTEM

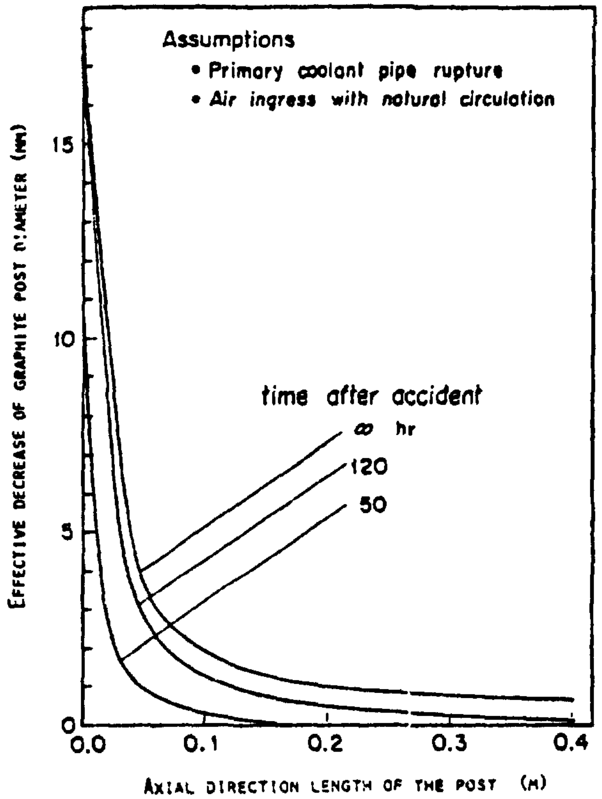

FIG. 9 AIR OXIDATION OF THE CORE SUPPORT POST GRAPHITE FOLLOWING A PRIMARY COOLANT PIPE RUPTURE

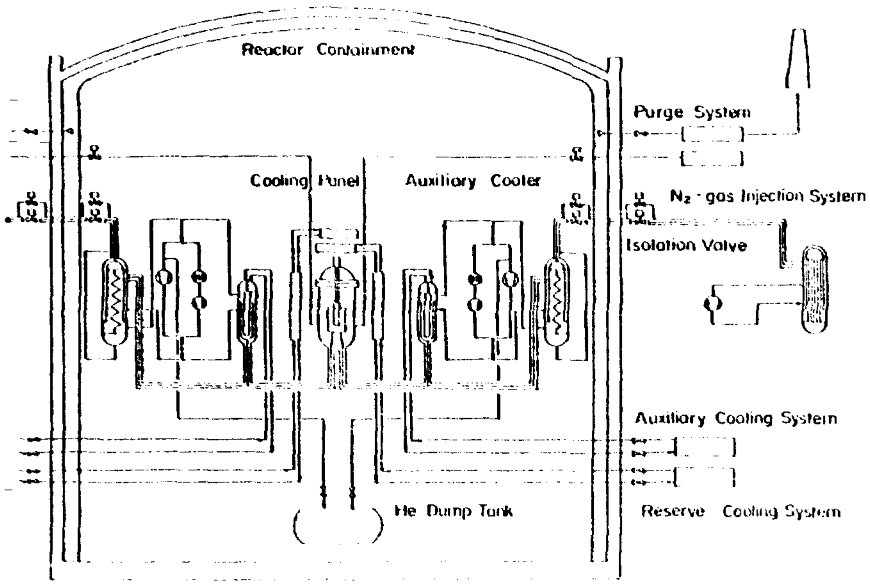

FIG. 10 ENGINEERED SAFETY SYSTEM OF EXP. VHTR

HTGR SAFETY PHILOSOPHY

V. JOKSIMOVIC, C.R. FISHER
General Atomic Company
San Diego, California
United States of America

ABSTRACT

The accident at the Three Mile Island has focused public attention on reactor safety. Many public figures advocate a safer method of generating nuclear electricity for the second nuclear era in the U.S. The paper discusses the safety philosophy of a concept deemed suitable for this second nuclear era.

The HTGR, in the course of its evolution, included safety as a significant determinant in design philosophy. This is particularly evident in the design features which provide inherent safety. Inherent features cause releases from a wide spectrum of accident conditions to be low. Engineered features supplement inherent features. The significance of HTGR safety features is quantified and order-of-magnitude type of comparisons are made with alternative ways of generating electricity.

NUCLEAR SAFETY IN AFTERMATH OF TMI

The accident at the Three Mile Island nuclear plant has focused public attention on nuclear safety. The Kemeny Commission Report (Ref. 1) gave the light water reactor generally good marks for protecting the public. The regulatory apparatus, including how the Nuclear Regulatory Commission and the industry have performed, was sharply criticized but reform and not suspension of the nuclear option was the Commission's final recommendation. More attention to man-machine interfaces was clearly mandated. Before the Kemeny Report became public, the NRC and the utility industry had undertaken their own steps to assure that the lessons of the Three Mile Island accident would be fed back promptly to all nuclear utilities so that they could make design modifications and improve plant operating procedures. In June 1979, the TVA Board approved a declaration on atomic safety which can be viewed as a new national yardstick for public policy in this field. As a result of all this, the safety operating nuclear plants will certainly be improved, as will the safety of the 100,000 MW(e) of nuclear capacity now under construction.

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