SAFETY DESIGN CONCEPT AND ANALYSIS FOR THE UPGRADED JRR-3

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ABSTRACT

The Research Reactor No.3 (JRR-3) is under reconstruction for upgrading. This paper describes the safety design concepts of the architectural and engineering design, anticipated operational transients and accident conditions which are the postulated initiating events for the safety evaluation, and the safety criteria of the upgraded JRR-3. The safety criteria to confirm the safety of the reactor, are defined taking account of those of the Light Water Reactors and the characteristics of the research reactor. As for the example of the safety analysis, this paper describes analytical results of a reactivity insertion by removal of in-core irradiation samples, a pipeline break at the primary coolant loop and flow blockage to a coolant channel, which are the severest postulated initiating events of the JRR-3.

1. INTRODUCTION

JRR-3, the first domestically built research reactor in Japan, had been operating since 1962. After 21 years of operation, an upgrading program was planned to respond to the demand for better neutron beam and irradiation conditions. The whole of the existing old core with the biological shield was removed by the one-piece-removal method, and the new core, which is quite different from old one, will be constructed at the site of the old one. The JRR-3 is to be upgraded to a 20 MW(th), light water moderated and cooled, beryllium and heavy water reflected, pool type reactor using 20% low enriched UAl_x-Al(LEU) plate type fuel. Besides, nine horizontal beam tubes (arranged tangentially to the reactor core), one cold neutron source (vertical thermo-siphon circulation type), five neutron guide tubes (for thermal and cold neutrons) and irradiation facilities will be installed for beam experiments in basic research in solid state physics, production of radioisotopes, irradiation testing of reactor materials or fuels and activation analysis.

The major features of the neutronics design are as follows.

The maximum fast neutron flux will be 3.0×10^{14} n/cm^{2}s in the fuel region, and the maximum thermal neutron flux will be 2.3×10^{14} n/cm^{2}s in the heavy water reflector region at the beginning of the equilibrium core. It has been confirmed that the maximum flux of this reactor will amount to more than the required value, 2.0×10^{14} n/cm^{2}s. The power peaking factors, which are a barometer of the flatness of the power distribution in the core, are expected to be 1.23 as the radial peaking factor, 1.42 as the axial peaking factor and 1.51 as the local peaking factor at the beginning of the equilibrium core without irradiation samples. The excess reactivity will be 16 %Δk/k in the cold clean core. It will decrease with the progress of burnup and become 9 %Δk/k at the end of the equilibrium core.

The major features of the thermal-hydraulic design are as follows.[1]

Two modes will be adapted for core cooling under normal operation, one is a natural circulation cooling for "Low power range" up to 200KW (th) and the other is a forced-convection cooling for "High power range" up to 20MW(th). A flow direction in the core for the forced-convection cooling mode will be downward. With downflow in the core at the normal operation, a core flow reversal should occur after reactor shutdown or scram. The primary coolant flow rate in normal operation will be 2400 m^3/h and the coolant velocity in each subchannel of the fuel elements will be about 6.2 m/s. The minimum DNBR in the rated operation is calculated to be 2.1 and the maximum fuel temperature is calculated to be 107°C.

2. OUTLINE OF THE UPGRADED JRR-3

2.1 Core components

Figure 1 shows the core components of the upgraded JRR-3 and surrounding installations. The upgraded JRR-3 is a pool type reactor.

The depth of the pool will be about 8 m. The core components consist of the core, heavy water tank, plenum and structural components. The core will be submerged in the pool. Experimental facilities such as a cold neutron source and horizontal experiment holes, a primary cooling system and a control rod driving mechanism will be installed around the core components.

2.2 Core

The core is cylindrical in shape, about 0.6 m in diameter and about 0.75 m in height. It is composed of fuel elements, control rods, irradiation elements and beryllium reflectors. Its specific power is rated at 156 kW/l. A configuration of the core is shown in Fig. 2.

The fuel elements will be MTR type (UAl_x-Al dispersion fuel), with an enrichment of 20wt%. There are two kinds of element, a standard fuel element and a fuel follower element. There is 300 g ^{235}U and 190 g ^{235}U in each element, respectively. There will be 26 standard fuel elements and 6 fuel follower elements.

The neutron absorbing control rods are made of hafnium, and connected to the fuel follower elements mentioned above. They are driven by the control rod driving mechanism installed beneath the core.

An irradiation element is similar to a standard fuel element in its outer dimensions and has an irradiation hole of 60 mm diameter at the center. There are five elements in the core.

Beryllium reflectors will be installed between the fuel region and the inner wall of the heavy water tank. They will comprise twelve pieces. Eight pieces will have irradiation holes. Four holes will be used for irradiation experiments and the other holes for surveillance tests.

2.3 Heavy water tank

The heavy water tank will be a double cylindrical type aluminum vessel, with a height of about 1.6 m and an outer diameter of about 2 m. The heavy water filling it will act as a reflector. Irradiation thimbles, horizontal beam tubes and cold neutron source facilities will be installed in it. It will have a cooling system in order to remove about 0.7 MW of heat generated in it.

2.4 Cooling system

A schematic diagram of the cooling system is shown in Fig. 3. Two main pumps and two auxiliary pumps will operate in normal operation of the reactor. The auxiliary pumps will also be used in order to remove the decay heat after the shutdown of the reactor. The coolant flow in the core is planned to be downward for the purpose of reduction of the radiation in the pool. A ^{16}N decay tank will be installed in the primary cooling system.

The total coolant flow rate in normal operation will be 2400 m^3/h and the coolant velocity in each subchannel of the standard fuel elements in the core will be about 6.2 m/s.

When the thermal power is less than 200 kW(th), the core can be cooled by natural circulation between the core and the reactor pool, induced by opening a natural circulation valve installed on the plenum.

The secondary cooling system will discharge heat transmitted from the primary cooling system and heavy water cooling system to the atmosphere via a cooling tower.

3. SAFETY DESIGN CONCEPTS

3.1 Safety design concepts

The safety design concepts of the JRR-3 are as follows.

(1) To shutdown the reactor

Six control rods will be designed independent with one another and the detection system of scram will have two networks. Moreover, the JRR-3 will have a heavy water dump system which will shutdown the reactor by dumping heavy water from the heavy water tank.

(2) To cool the reactor core

Decay heat removal after the reactor scram can be attained by only putting the fuels in water except for a short time just after the reactor shutdown, because the power of the JRR-3 is rather small compared with that of LWR. The JRR-3 will have, therefore, two syphon break valves which are installed to keep pool water above a fixed level even in an accident condition such as LOCA (Loss of Coolant Accident). The JRR-3 will have two auxiliary pumps, and also an emergency electric power supply system which is composed of batteries and generators. When off-site power is lost, the reactor is shutdown. Then, electric power is supplied without a break to auxiliary pumps from an emergency power supply system in order to remove decay heat.

(3) To limit or prevent the release of fission products

The JRR-3 will have three barriers as other reactors that prevent or limit the transport of fission products to the environment, which are fuel cladding, reactor pool water and the reactor containment. First, the reactor protection system and the engineered safety system will be therefore installed to prevent the fuel melting at accident conditions as well as anticipated operational transients. Second, the reactor pool is expected to have a function to limit the transport of fission products such as Iodine-131 to the reactor room at accident conditions by keeping the pool water level. Last, the emergency exhaust system will discharge fission products from the reactor room through charcoal filters. Thus, the system will limit the release of fission products to the environment.

3.2 Postulated initiating events for the safety evaluation

The behavior of the reactor to postulated initiating events are analyzed taking the action of the reactor protection system into account, to confirm the safety of the reactor. The postulated initiating event is generally called the design basis event (DBE). The DBEs include anticipated operational transients which occur once or more in the reactor life, and accident conditions of lower probability which give severer consequences to the reactor and the environment. When more than two similar events exist, the event giving severest consequences is selected. The DBEs in the JRR-3 are listed in Table 1.

3.3 Safety criteria

The criteria to comfirm the safety of the reactor, are defined as Table 2, taking account of the criteria of the LWR and the characteristics of the research reactor.

The anticipated operational transients must be terminated before damaging the reactor core, and the normal operation condition must be continued soon after the anticipated operational occurrence. The criterion (a) in Table 2 is defined to prevent the burnout of fuel by the power-cooling mismatch. The criterion (b) limits the fuel temperature increase in case of a big reactivity insertion, where the criterion (a) is satisfied in some cases. The temperature limit (400°C) is defined as the non-blistering temperature of the U-Al_x dispersion type fuel. The criterion (c) prevents the fuel rupture or the decrease in coolability due to the deformation of the fuel plate by thermal stress, etc.. It is defined that the deformation would take place when the stress in fuel plate overcome the yield strength. The criterion (d) prevents the pressure increase in the primary cooling system which leads to the damage of the components.

The safety criteria in accident conditions are given to show that the fuel would not melt if the postulated accident took place, and that the barriers against the release of fission products are designed adequately. The criterion (d) is a final goal at the safety control of reactors. Other criteria (a), (b) and (c) are the important check points in the accident process to guarantee the criterion (d). The plate type fuel has characteristics that the melting temperature is low and the thermal time constant is small compared with the fuel of LWR (i.e. the thermal time constant is only 0.01 s in plate type fuel, 7 s in LWR fuel). The plate type fuel therefore must not be dried out for a moment, so the fuel must always be put in water in any accident conditions.

4. SAFETY ANALYSIS

4.1 A brief of codes used in the analysis

The analyses were carried out using the computational codes, EUREKA-2 and THYDE-P developed at JAERI, and HEATING5 developed at ORNL.

(1) EUREKA-2

The EUREKA-2 code can predict the course and consequence of a reactivity insertion accident.[2] This code provides a coupled thermal-hydraulic and point kinetics capability. The feedback reactivities are evaluated by a importance weighted sum of the contribution in each region of the core.

(2) THYDE-P

The THYDE-P code can analyze a thermal-hydraulic behavior of anticipated operational transients and accident conditions in LWR.[3] This code applies an one-dimensional node and junction method. The code modification was made for a package of the heat transfer correlations and the DNB heat flux correlations, which are applicable to both downflow and upflow in plate type fuels under the conditions of low pressure and low temperature. The heat transfer package was especially developed for the core thermal-hydraulic design and the safety analysis of the JRR-3, based on the heat transfer experiments in which thermal-hydraulic features of the JRR-3 core were properly reflected.[4]

(4) HEATING5

HEATING5 code[5] is designed to solve steady-state and or transient heat conduction problems in one-, two-, or three-dimensional Cartesian or cylindrical coordinates or one-dimensional spherical coordinates. The thermal conductivity may be anisotropic. Materials may undergo a change of phase. Heat generation rates may be dependent on time, temperature and position, and boundary temperatures may be time-dependent.

4.2 Analytical results and discussions[6-9]

4.2.1 A reactivity insertion by removal of in-core irradiation samples

This transient would occur when irradiation samples would be removed quickly from the irradiation holes, under full power operation.

Figure 4 shows the transients of the reactor power, and the fuel temperature at the hot spot. 0.01 s later after the transient begins, the reactor power (neutron flux) reaches scram point of 22MW, but due to the scram delay time of 0.1 s, the reactor power continuously increases up to about 24.8MW. After that, the reactor power is suppressed by the control rod insertion. The minimum DNBR is calculated to be about 1.8 and the maximum fuel temperature is 119°C. Figure 5 shows the inserted, the total and the scram reactivity until 1.6 s after the transient begins. Feedback reactivities of moderator temperature effect and doppler effect are about only 10% of the inserted reactivity, when the reactor power is maximum (24.8MW) in this transient. And they don't have a significant effect on the transient. The reactor is suppressed by the scram reactivity. The analytical results show that the transient can safely come to an end by monitoring a neutron flux.

4.2.2 A pipeline break at the primary coolant loop (Effluent of primary coolant due to pipe rupture)

The large pipe break won't be expected to occur in a research reactor which is operated under the condition of low pressure and low temperature. We made an assumption in the analyses of postulated piping failures that maximum break area was Dt/4 (D : Pipe diameter, t : Pipe thickness).

We'll show the analytical results of the accident under the pessimistic assumption that the break would have the maximum area, and occur at the suction line of the main pumps. The main pumps would therefore fail to work due to pump degradation because of the suction of air through the break into the primary coolant loop. This assumption would make the accident severer.

Figure 6 shows the changes of the pool water level and the core flow after the accident begins until the reactor pool isolation. While the core flow decreases due to the pump degradation, the primary coolant discharges through the break. The pool water level starts to go down. The reactor scram takes place when the primary coolant flow decreases below a specified value (85% of the normal flow rate). After the coastdown of the main coolant pumps is completed, the forced-convection core cooling by the auxiliary pump continues until the reactor pool isolation. When the natural circulation valve is opened, a part of the primary coolant flow bypasses the core through the natural circulation valve, and then the core flow decreases slightly. After the pool water level decreases to the level of the syphon break line, it is kept constant and the decay heat is removed by natural circulation cooling.

The accident sequence has two important thermal-hydraulic behaviors according to Fig. 6 from the safety point of view.

(1) Just after the accident initiation

Figure 7 shows the transients of the power, the core flow and the fuel surface heat flux at the hot spot until 4 s after the accident begins. Figure 8 shows the fuel temperature at the hot spot and the DNBR. The accident is dominated by the competitive process of the decrease in the core flow against the decrease of the decay heat after the reactor scram. The minimum DNBR 1.7, and the highest fuel temperature 120°C of the whole transient were calculated here at the beginning of the control rod insertion. These values meet the criteria of anticipated operational transients, even in an accident condition.

(2) After the reactor pool isolation

Figure 9 shows the core flow, the fuel surface heat flux and the heat transfer coefficient at the hot spot at the flow reversal. Figure 10 shows the DNBR and the fuel temperature at the hot spot. Before the pool isolation occurs, the fuel temperature reaches a certain value because all the decay heat is removed by the primary cooling system, using the auxiliary pump. One auxiliary pump is assumed to fail in this accident. The fuel temperature starts to increase just after the reactor pool isolation because of the accumulated energy in the fuel, and has the peak value, 80°C. After the core flow reversal, the fuel temperature reaches another steady value in the natural circulation. The DNB heat flux decreases with the flow coastdown to the minimum value, and stays at the value during the short flow reversal time. The surface heat flux suddenly decreases with the heat transfer coefficient decreasing due to the flow coastdown. The heat transfer coefficient reaches the lowest level calculated by Nu=4 and stays at the value during the flow reversal. The minimum DNBR was calculated to be 3.1. This shows that the decay heat level is sufficiently low at the flow reversal.

4.2.3 Flow blockage to a coolant channel

Flow blockage to coolant channels would occur by some extraneous things which come from outside of the reactor pool, may block the coolant flow channels of the core. If flow blockage to coolant channels would occur, fuel temperature will increase due to flow rate decrease of coolant channels. One standard fuel element was supposed as flow blockage channel in the credible accident and the whole core was supposed as flow blockage channels in the hypothetical accident of the JRR-3 safety assessment. In the safety assessment, fission products were released from inside of fuel plates of flow blockage channels to the primary cooling system due to failure of fuel plates.

In the accident analysis of flow blockage to coolant channels, we'll make a focus on the event that the flow blockage channel is only one sub-channel in the fuel element which is the hottest-channel, because one standard fuel element or the whole core flow blockage is postulated in the safety assessment. Transient analysis of the flow brockage to coolant channels is difficult, so steady-state calculation is selected. This selection would make the accident severer. Heat generation in fuel plates both side of the flow blockage channel are removed from normal channel sides.

Figure 11 shows the steady-state calculation results of fuel plate surface temperature distribution both of flow blockage channel side and normal channel side, and coolant temperatute distribution of normal channel side. The minimum DNBR was calculated to be 1.1, and the highest fuel temperature was calculated to be 150°C. In this case, fuel plates do not fail, because of the highest fuel temperature is rather low compare with the melting temperature.

5. CONCLUDING REMARKS

This paper presented the safety design concepts and the analyses carried out for three of the design basis events of the JRR-3. The analyses showed the following results.

(1) The operational transient initiated by reactivity insertion due to removal of in-core irradiation samples can be suppressed safely by the reactor scram from the monitor of the neutron flux.

(2) In the accident initiated by a pipeline break at the primary coolant loop, although a sudden increase of the fuel temperature and a steep decrease of the DNBR would occur at the flow reversal, the peak temperature and the minimum DNBR can meet the design basis criteria of anticipated operational transients, even in an accident condition.

(3) The steady-state calculation results of flow blockage to a coolant channel show that fuel plates do not fail in case of one sub-channel flow blockage.

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REFERENCES

[1] Y. Sudo, H. Ando, H. Ikawa and N. Onishi, Core Thermohydraulic Design with 20% LEU Fuel for Upgraded Research Reactor JRR-3, Nucl. Sci. Tecnol. Vol.22, No.7, pp.551 564, July 1985.

[2] N. Onishi, T. Harami, H. Hirose and M. Uemura, A Computer Code for the Reactivity Accident Analysis in a Water Cooled Reactor, (in Japanese), Rep. JAERI-M 84-074, 1984.

[3] Y. Asahi, Description of THYDE-P (Preliminary Report of Methods and Models), Rep. JAERI-M 7751, 1978.

[4] Y. Sudo, H. Ikawa and M.Kaminaga, Development of Heat Transfer Package for Core Thermal-hydraulic Design and Analysis of Upgraded JRR-3, in Proceeding of the International Meeting of Reduced Enrichment for Research and Test Reactors, Petten, The Netherlands, October 14-16,1985.

[5] W. D. Turner, D. C. Elrod, I. I. Siman-Tov, HEATING5-An IBM 360 Heat Conduction Program, Rep. ORNL/CSD/TM-15

[6] T. Harami, M. Hirano, Y. Asahi, A. Kosaka and N. Onishi, A Safety Analysis of the Research Reactor JRR-3, in Proceeding of the International Meeting of Reduced Enrichment for Research and Test Reactors, Petten, The Netherlands, October 14-16,1985.

[7] T. Harami, M. Uemura and N. Onishi, Reactivity Initiated Accident Analyses for the Safety Assesment of Upgraded JRR-3, (in Japanese) Rep. JAERI-M 84-142, 1984.

[8] M. Hirano and Y. Sudo, Analytical Study on Thermal-Hydraulic Behavior of Transient from Forced Circulation to Natural Circulation in JRR-3, Nucl. Sci. Tecnol. Vol.23, No.4, pp.352 368, April 1986.

[9] H. Ando, H. Ikawa, K. Yoshimura and N. Onishi, Sefety Analysis of the Upgraded JRR-3 with HEATING5 Code, (in Japanese), Rep. JAERI-M 85-019, 1985

Table 1 Design Basis Events in the JRR-3

1. Anticipated operational transients

(1) Reactivity Insertions

(a) Withdrawal of control rod st startup.

(b) Withdrawal of control rod at high power operation.

(c) Reactivity insertion by removal of in-core irradiation samples.

(d) Reactivity insertion by increase of primary coolant flow rate.

(2) Decrease in coolability

(a) Primary coolant pump failure and flow coastdown

(b) Secondary coolant pump failure and flow coastdown

(c) Loss of commercial electric power.

(3) Other occurrence

(a) Reactor power increase due to failure of heavy water tank.

2. Accident conditions

(1) Decrease in coolability

(a) Flow blockage to coolant channels.

(b) Effluent of primary coolant due to pipe rupture.

(c) Primary coolant pump abrupt failure without coastdown.

(d) Secondary coolant pump abrupt failure without coastdown.

(2) Release of radioactive materials.

(a) Release of heavy water due to failure of heavy water coolant loop.

Table 2 The safety criteria in the JRR-3

1. Anticipated operational transients

(a) The minimum DNBR shall be more over 1.5.

(b) The maximum fuel meat temperature shall not be over the blisterring temperature (400°C).

(c) Fuel plates shall not be deformed significantly.

(d) The pressure in the primary cooling system shall not be over 1.1 times the design pressure.

2. Accident conditions

(a) The core shall be put under water in any case.

(b) The core shall not be damaged severly, and be coolable.

(c) The pressure in the primary cooling system shall not be over 1.2 times the design pressure.

(d) The radiological risk to the public shall be as low as reasonably achievable.

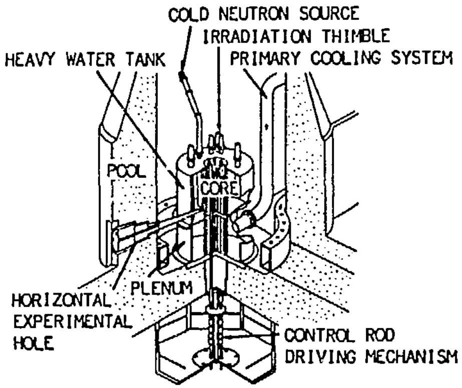

Fig. 1 Core components

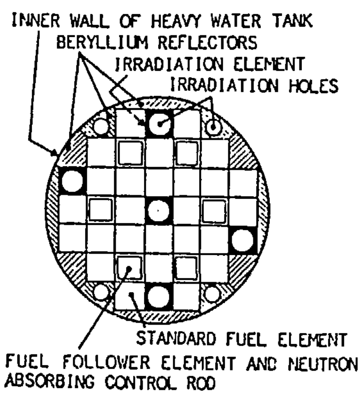

Fig. 2 Configuration of the core

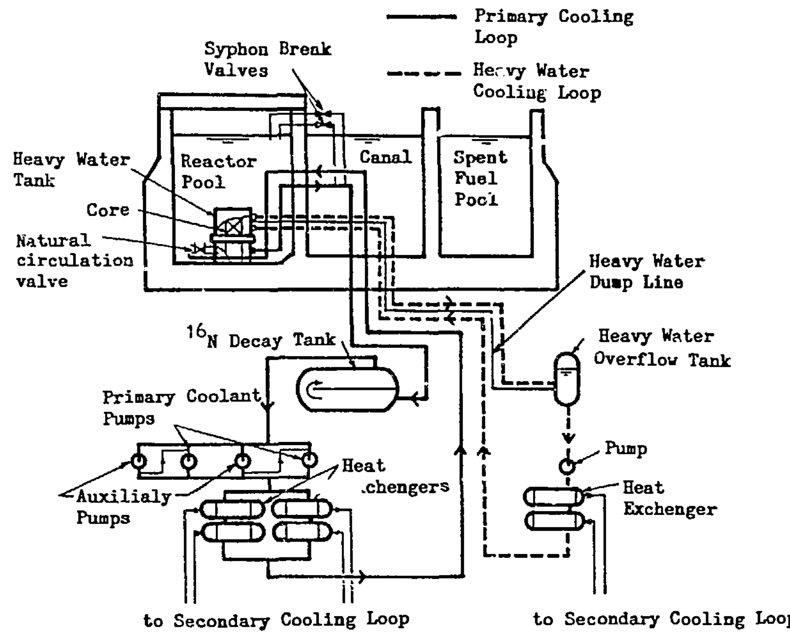

Fig.3 Schematic Diagram of Primary Cooling Loop and Heavy Water Cooling Loop

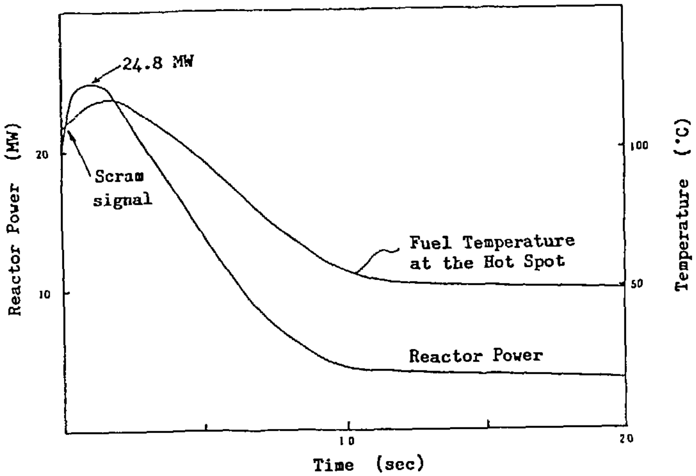

Fig.4 Reactor Power and Fuel temperature at the Hot Spot (Reactivity Insertion by Removal of In-core Irradiation Samples)

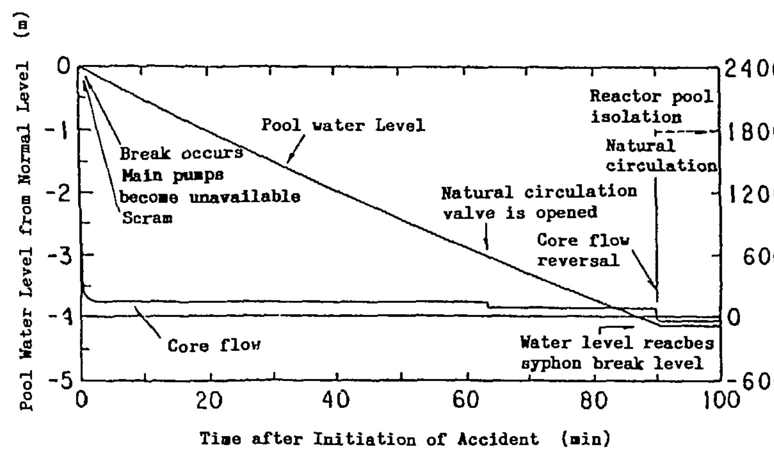

Fig.6 Transient of the Pool Water Level and the Core Flow (Primary Coolant Pipeline Break)

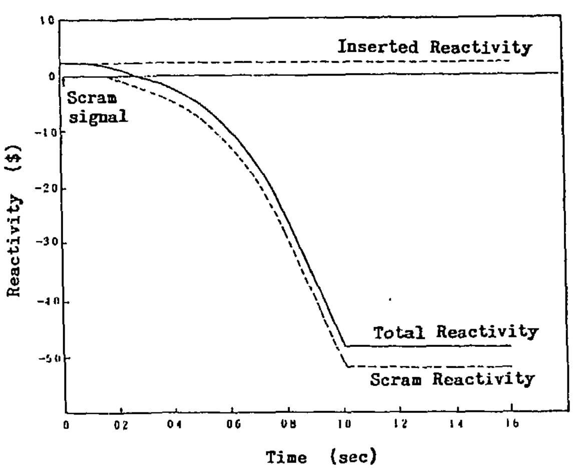

Fig.5 Inserted, Total and Scram Reactivity (Reactivity Insertion by Removal of In-core Irradiation Samples)

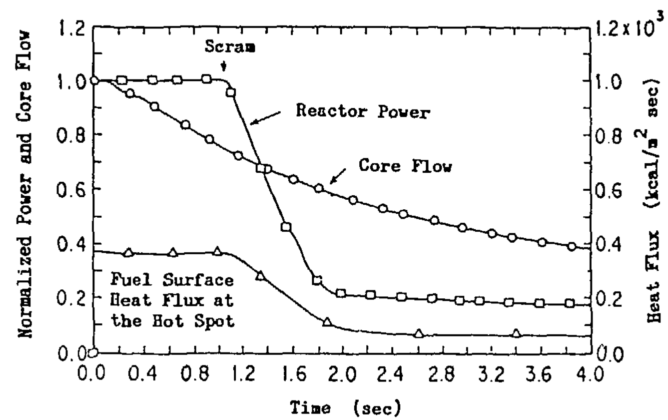

Fig.7 Normalized Power and Core Flow, and Fuel Surface Heat Flux at the Hot Spot (Primary Coolat Pipeline Break)

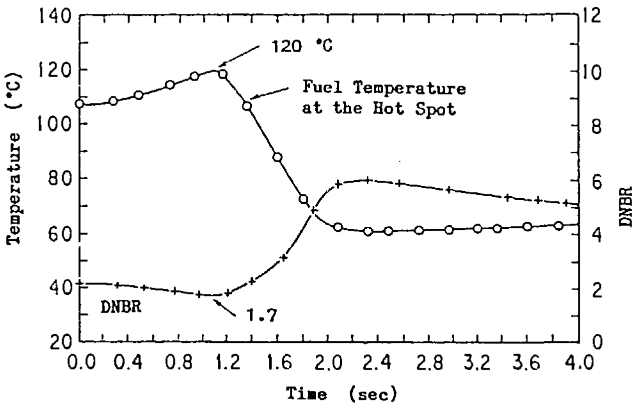

Fig.8 Fuel Temperature at the Hot Spot and DNBR (Primary Coolant Pipeline Break)

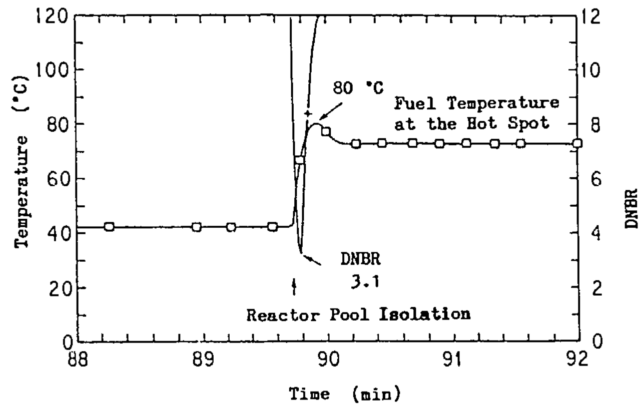

Fig.10 DNBR and Fuel Temperature at the Hot Spot at the Flow Reversal (Primary Coolant Pipeline Break)

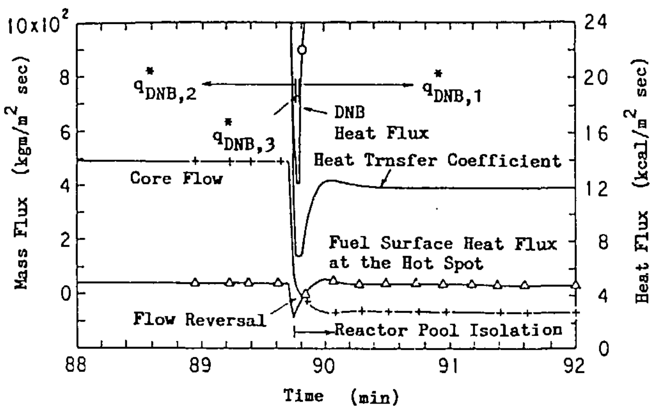

Fig.9 Core Flow, Fuel Surface Heat Flux at the Hot Spot, Heat Transfer Coefficient and DNB Heat Flux at the Flow Reversal (Primary Coolant Pipeline Break)

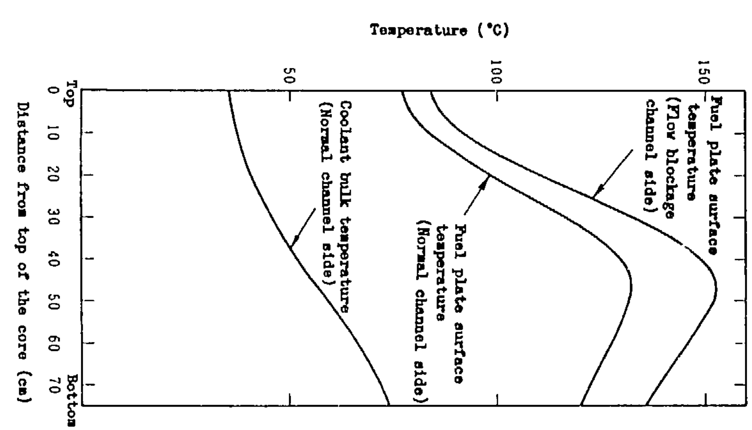

Fig. 11 Steady-state calculation results of Flow blockage to a coolant channel