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Analysis of the Chernobyl accident from 1:19:00 to the first power excursion

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Abstract

Many researchers have reported that the root cause of the Chernobyl accident has not been clarified still now. Since many of them discussed the accident without a precise thermal-hydraulic investigation, thermal-hydraulic calculations coupled with neutronic calculations have been done on the basis of the recorded result at the Chernobyl Unit-4. Plant configurations and operational conditions were given to the code on the basis of reported result and published papers. Calculation could trace plant parameters from 1:19:00 to the first power excursion without any discrepancies measured at the Chernobyl Unit-4. Reactivity slightly smaller than 1β by the positive scram is concluded as a possible direct cause of the accident, which acts as a trigger to increase the reactor power. Other possibilities as a trigger of the accident such as cavitation in pumps and pump coast-down were investigated. The importance of the calculation from the stable condition is also described in this paper in order not to bring unnecessary assumptions into the calculation.

1. Introduction

It is almost two decades since the Chernobyl accident on 26 April 1986. However, the root cause of the Chernobyl accident is not clear enough still now according to the papers published after the first official meeting held by IAEA (1986). The summary of the accident was also investigated and reported by INSAG-A (1986) and USNRC (1986). So far, the accident was mainly investigated from the following three scenarios: (1) the positive reactivity insertion by the flaw of the scram rods, (2) loss of pumping power by cavitation, (3) the positive reactivity insertion only by pump coast-down. The mechanism of the positive reactivity insertion by the scram rods has been experimented and calculated, and many researchers think that this is one of the causes of the accident. However, some said that only the voiding by the coast-down of re-circulation pumps was the root cause of the accident.

Toffer and Twitchell (1987) pointed out that insertion of fully withdrawn control rods introduced positive reactivity in the bottom of the core. Hall (1988) pointed out that the consensus on the positive scram was growing in USA and Canada. Chao et al. (1988) calculated the accident with the assumption that scram added positive reactivity to the core at the first 6 s of their insertion. They explained that this phenomenon was caused by water being displaced from the absorber rod. They assumed the positive reactivity as large as 1.25β and 5.6β reactivity from void in order to trace the Chernobyl accident using the RETRAN code. Adamov (1988) tried to calculate the power excursion using a tree-dimensional computer code, and concluded that the accident started in the lower half of the core. Abagyan (1992) mentioned the same thing. Chan and Dastur (1989) also investigated that the negative scram reactivity was achieved only in the case where the distribution of the neutron flux was convex shape. Velikhov et al. (1991) explained the plant operation conditions just before the accident in their paper, and clarified the necessity of the experiment using the turbo-generators at the Chernobyl Unit-4. They also mentioned that the positive reactivity might be introduced by the safety rods when the axial power distribution of the core was double humped shape taking into account of Xe build-up. They mentioned that this phenomenon was caused by the replacement of a water column of 1.25 m between a control rod and a graphite displacer. Andriushchenko et al. (1991) measured reactivity of a scram rod using a graphite assembly with 7 m in height and 7.5 m in diameter, and observed that the

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positive reactivity was inserted into the core when the safety rod was entered into the core. They calculated this phenomenon using a neutronic code. Since the prediction overestimated the measured result, they adjusted the prediction. Finally they concluded that approximately 0.75β reactivity was inserted into the core at the Chernobyl accident. Landeyro and Buccafurni (1991) calculated the amount of reactivity by the positive scram for various conditions. In many cases, the values of reactivity are larger than 1β. Nevertheless, they concluded that the positive reactivity due to insertion of the emergency control rods played only a minor role in the reactivity balance of the accident. D'Angelo (1997) investigated the positive scram effect using the French CRONOS three-dimensional code under various hypotheses on the axial shape of the initial power distribution. In the case where the power distribution was perturbed by Xe build-up, 1β at 1.75–2.25 s after the initiation of scram by the AZ-5 scram button and approximately 2β at 3–4 s after the AZ-5 activation.

Martinez-Val et al. (1990) assumed the cause of the accident that void reactivity due to a loss of pumping power had overcome the Doppler-effect. They investigated the positive scram by the removal of water columns from the core but they did not confirm any contribution of the scram rods on the insertion of positive reactivity. They also insisted that cavitations in the pumps could have induced two-phase flow that resulted in a loss of pumping power. Fletcher et al. (1988) also assumed the pump cavitation as a trigger of the accident.

On the other hand, Wakabayashi et al. (1987) traced the accident by various codes in the area of neutronics and thermal-hydraulics. They analyzed the similar magnitude of the power excursion with the coupling of two codes that calculated the plant thermal-hydraulics with the FATRAC code and power excursion with the EUREKA-2 code. In their case, the analysis was not completely consistent due to the data transfer at 1:23:00 from one code to the other. The power excursion was calculated without any specific trigger to add the positive reactivity instantaneously. Therefore, only the void reactivity due to the coast-down of re-circulation pumps was assumed as the cause of the power excursion. In their analysis, calculated reactor power at 1:23:40 was approximately 600 MW, which is not consistent with the reported result.

Most investigations were conducted under the assumption that above-mentioned two effects, i.e., the positive scram and the positive reactivity by voids, are calculated separately with the neutronic codes or plant transient calculation from the point in time just before the power excursion. It is not clear that the codes, which they used could trace the plant parameters from 1:19:00 measured at the Chernobyl Unit-4 without any discrepancies. The necessity of the calculation of the reactivity by the positive scram can be understandable, however, a separate void effect calculation is meaningless because a trigger event influences very much the generation of voids in the core. Since, calculation method and codes are developed and modified during two decades, the thermal-hydraulic and neutronic analyses have been conducted in this study to investigate the root cause of the accident on the basis of the findings since the first meeting in Vienna.

2. Analytical model

2.1. Computer code

The plant behavior was calculated using the NETFLOW code that was originally developed as the ATRECS code for a heavy-water-moderated light-water-cooled pressure-tube-type reactor by Mochizuki and Hayamizu (1988) and modified by the Mochizuki (1994) to calculate flow instabilities. Mochizuki and Kishida (1998) modified the NETFLOW code to calculate sodium flow system and verified the capability. At present this code can calculate thermal-hydraulics of various reactors such as light-water reactors, heavy-water reactors and liquid-metal reactors. This code is one-dimensional network code developed for a piping network analysis in order to design the optimum residual heat removal system and an auxiliary cooling system together with behaviors of the primary system. Equations relating thermal-hydraulics are transformed into finite differentials with the fully implicit method. One point kinetic equations with six delayed neutron families are used to calculate neutron flux. Since this code is one-dimensional, detailed thermal-hydraulic correlations are necessary in order to predict accurately the plant behavior. Thermal-hydraulic correlations, control system performances and the total prediction capability were verified by using full-scale experimental facilities of the advanced thermal reactor (ATR) which is a light-water-cooled heavy-water-moderated pressure-tube-type reactor developed in Japan and the prototype reactor Fugen.

2.2. Analytical model

In the present study, 1/4 loop of the Chernobyl reactor is modeled as shown in Fig. 1. In this figure, L, OD, ID and N stand for length, outer diameter, inner diameter and number of pipes, respectively. Size of components and length of piping are decided referring to several papers, for example Aleksakov et al. (1980), Emel'yanov et al. (1977), Dollezhal and Emel'yanov (1980), Dollezhal (1981), Aden et al. (1977), Almenas et al. (1998) and a brochure of RBMK-1000 which Japanese delegation obtained at the occasion of visiting of the Leningrad NPP before the accident. Bends are assumed at the same location as indicated in the figure. Since total fuel channels were 1661, fuel channels of 415 connected to a drum separator are considered in the present model. These channels of 415 are divided into seven groups in order to investigate the flow oscillation during the transient. Although two re-circulation pumps were operated for 1/4 loops of the Chernobyl, one pump model was adopted in the present model. Piping length, diameter, bends and location of valves were modeled as possible as they were. Pump heat input is given by a heater provided just downstream of the re-circulation pump. Heating power from the graphite stacks to the coolant is calculated by giving heat conductance and temperature difference between the graphite stack and the pressure tube. Temperatures of piping and pressure tubes are calculated by solving the equation of heat conduction together with heat transfer from piping to air through insulator.

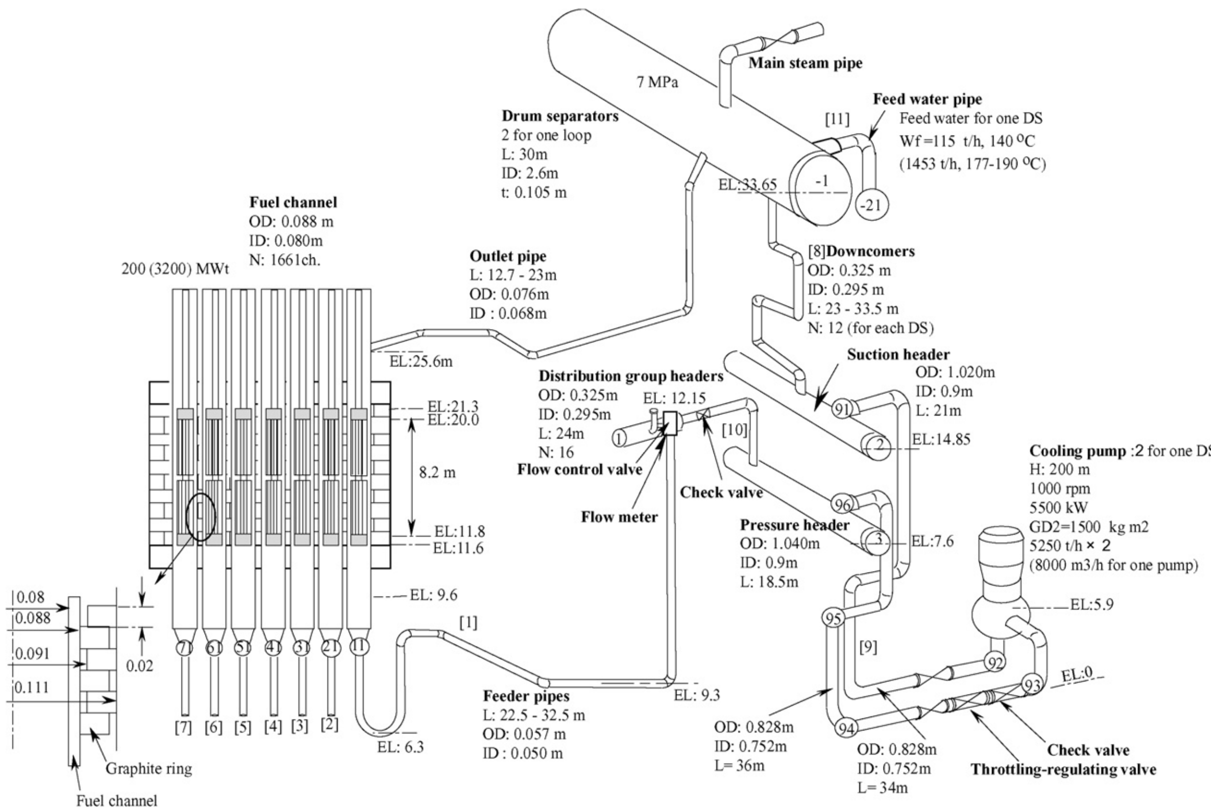

Fig. 1. Analytical model of the Chernobyl reactor.

2.3. Analytical conditions

So far, most analyses were conducted from the time just before the accident. However, operators in the Chernobyl reactor added perturbations to the plant before the pump coast-down. In the first meeting held in Vienna, plant transient from 1:19:00 was shown together with calculated result using an analog computer from this point in time. In order to escape from the unnecessary perturbations added by the analysis, starting time of the calculation is decided at 1:19:00. The overall sequence of the accident is explained by Kress et al. (1987).

The initial conditions at 1:19:00 for 1/4 core are as follows:

Nuclear thermal power	50 MW (reported)
Heating power by pumps	7 MW (investigated)
Heating power by graphite	Approximately 20 MW (investigated)
Total core flow rate	10,500 t/h (reported)
Feed water flow rate	115 t/h (reported)
Main steam flow rate	115 t/h (reported)
Temperature of feed water	140 °C (assumption)
Pressure of steam drum	6.9 MPa (reported)
Water level of drum separator (DS)	NWL-0.57 m (reported)
NWL (normal water level)	1.8 m from the bottom of DS (investigated)
Pump revolution speed	1000 rpm (investigated)
Pump head	200 m (investigated)
Flow rate of one pump	8000 m^3/h (investigated)

As for the neutronic characteristics, Doppler reactivity and void reactivity are assumed as shown in Fig. 2. Kinetic parameters are assumed as shown in Table 1. These values are the same values, which were used in the analysis reported by Wakabayashi et al. (1987). These values were calculated by WIMS-D and CITATION codes with the assumption of burn-up of 10.3 GWd/t. The reason why the same curve was adopted in the present study is to show that no accident does occur naturally under the situation of pump rundown. Graphite ring temperature at 400 °C reported in the same reference is adopted and is assumed to be constant for the analytical interval.

As for the transient calculation from the time at 1:19:00, feed water flow rate recorded during the accident is given to the code

Table 1
Kinetic parameters

Prompt neutron life time (s)	6.4×10^{-4}
Ratio of delayed neutron (β)	
β_1	4.08×10^{-4}
β_2	1.18×10^{-3}
β_3	1.04×10^{-3}
β_4	2.15×10^{-3}
β_5	7.66×10^{-4}
β_6	1.83×10^{-4}
β_{eff}	5.73×10^{-3}
Decay constant, λ (1 s^{-1})	
λ_1	1.22×10^{-3}
λ_2	3.15×10^{-2}
λ_3	1.20×10^{-1}
λ_4	3.20×10^{-1}
λ_5	1.39
λ_6	3.78

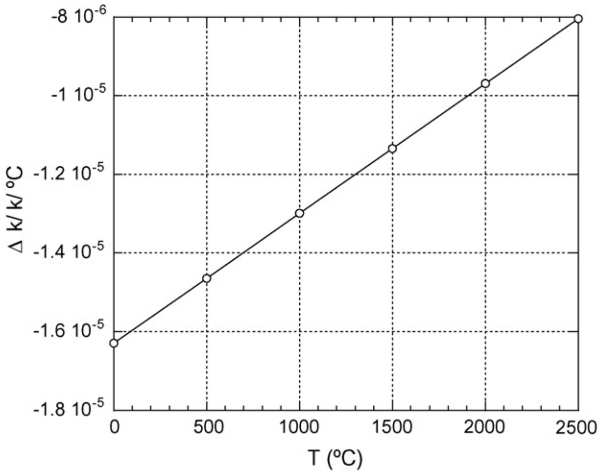

Fig. 2. Doppler reactivity coefficient and void reactivity coefficient.

as a boundary condition. Steam flow rate from the drum separator is decreased at 1:23:08 and zero at 1:23:14. At this point of time, there is no exit of steam generated in the core. Furthermore, the positive reactivity by safety rods is assumed during 6 s from the time at 1:22:40 when an operator pushed the AZ-5 button. Peak reactivity is 0.93β and the time dependency curve of the reactivity is estimated on the basis of a figure reported by Andriushchenko et al. (1991). The assumed reactivity history by the positive scram is shown later together with the other reactivities that govern the accident. Another assumption of the calculation is that the reactor power is automatically controlled by absorber rods with small reactivity during the calculation, which enables to maintain the power at constant just before the power excursion. Pump coast-down is not simulated properly unless the design value is given to the code. By this reason the large inertia of the pump is assumed to trace the pump coast-down curve.

2.4. Calculation result

Equilibrium conditions at 1:19:00 were taken by 2000 s transient calculation without any perturbation. These initial conditions were transferred to the transient calculation. The time step for thermal-hydraulics was 1 s and for neutronics 0.01 s from 1:19:00 to 1:23:10 and time step from 1:23:10 is 0.2 and 0.002 s, respectively. If any strange assumptions were introduced into the calculation, the trends for 4 min and 40 s shifted from the recorded result by the SKALA system.

The calculation results by NETFLOW trace the recorded results without any discrepancies as shown in Fig. 3a and b. These figures show plant parameters for one half of the reactor. Closed symbols show the data measured by the SKALA system. When feed water flow rate is increased by a factor of 4 from the level of the plant requirement, system pressure is lowered and water level in the drum separator is increased. Re-circulation flow rate is increased slightly along with the decreasing system pressure. This is caused by the decreasing of pressure loss in the two-phase region due to the decreasing of void fraction. When the stop valve of turbo-generators is closed, the system pressure increases. When the large reactivity by the positive scram is added to the reactor, the reactor power excursion has occurred.

Fig. 4 shows the magnified power transient at around 1:23:44. The square closed symbol shows the reported trend by USSR and the broken line shows the calculated result multiplied by 4. As shown in two figures, reactor power is controlled at the initial power level until the insertion of the positive reactivity. This is the same result as the reported one at the IAEA meeting. The calculated power converted to the full core escalates suddenly up to approximately 2.6×10^5 MW that is approximately 80 times of the rated thermal power of 3200 MW. At this moment, calculated

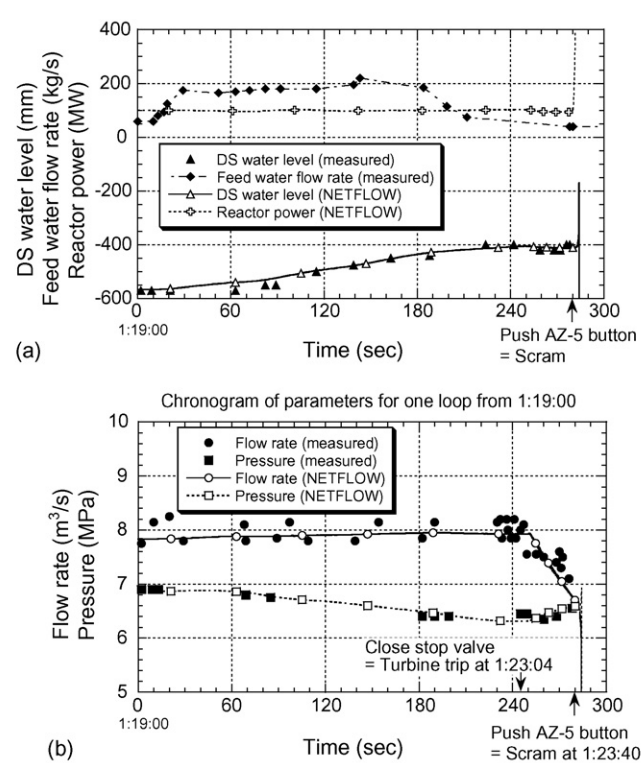

Fig. 3. (a) Comparison between recorded plant parameters and calculation by NETFLOW; (b) comparison between recorded plant parameters and calculation by NETFLOW.

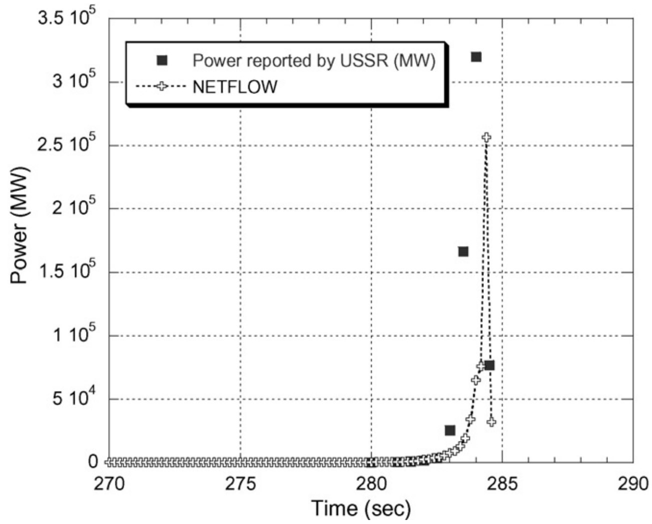

Fig. 4. Comparison of power transient.

pellet temperature is beyond the melting temperature. The calculation using the NETFLOW code is terminated after the first peak power is calculated. This is because the pellet temperature is out of the range. This is beyond the capability of the code, and much powerful computer code is preferable. Fig. 5 shows the history of reactivity regarding void, Doppler and total together with the assumed reactivity by the positive scram resulting in this accident. Since the reactivity by absorber rods is almost zero, the curve is omitted from this figure. The value of void reactivity was slightly negative because the voids collapsed before the initiation of AZ-5 button. The behavior of voids will be discussed later.

Density wave oscillation is also investigated during the calculation period. But no oscillation is calculated. This is because the ratio of pressure drop in the two- to single-phase region became smaller than the rated condition.

3. Discussion

3.1. Cause of the accident

According to the thermal-hydraulic calculation coupled with simple neutronic model of the Chernobyl reactor, the accident is seemed to be simulated. The present calculation shows a different story about void fraction behavior in the core compared to reported scenarios. One should keep in mind that even in the rated condition, the RBMK reactor has a characteristic that the void fraction in the lower part of the core is very small because the thermal equilibrium steam quality becomes positive nearly top of the lower half fuel bundle. Before the accident at the Chernobyl Unit-4, since re-circulation flow rate was extraordinarily high, it is estimated that there was no void in the lower half of the core on the basis of the present calculation. This situation is shown in Fig. 6. These trends indicate thermal equilibrium steam qualities at the top of the core and the center. The quality is calculated by the following equations:

$$x = \frac{h - h_{\mathrm{sat}}}{h_{\mathrm{g}} - h_{\mathrm{sat}}} \tag{1}$$

$$h = h_{\mathrm{in}} + \frac{Q}{W} \tag{2}$$

where x stands for thermal equilibrium quality, Q power (kW), W flow rate (kg/s), h_{in} inlet enthalpy (kJ/kg), and h_{sat} for saturation enthalpy of liquid (kJ/kg). If the value is negative, it means subcooled single-phase flow. If the quality is positive less than 1, it means two-phase flow.

When the stop valve is closed in order to trip turbines, the system pressure increases and this suppresses the void generation by the pump coast-down because the flow reducing is approximately 15% by the point in time of the scram action. The top of the core was almost single-phase just before the AZ-5 button was activated. Heck et al. (1987) mentioned the same thing. In the reported result at the (IAEA, 1986) meeting, steam quality at the exit of the core decreased and leveled

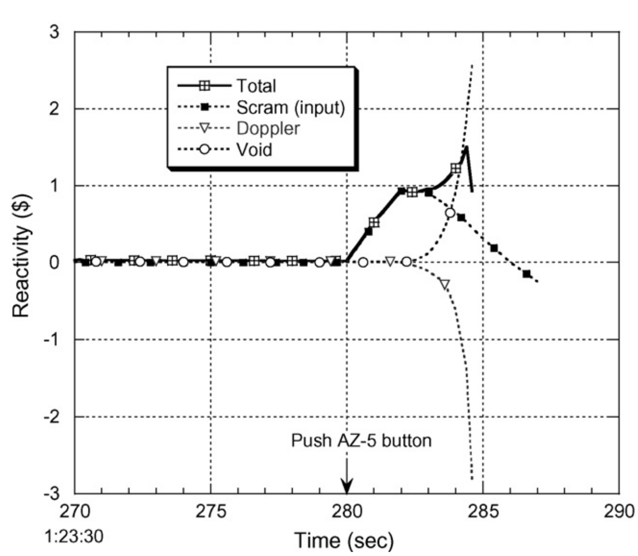

Fig. 5. History of reactivity during the accident.

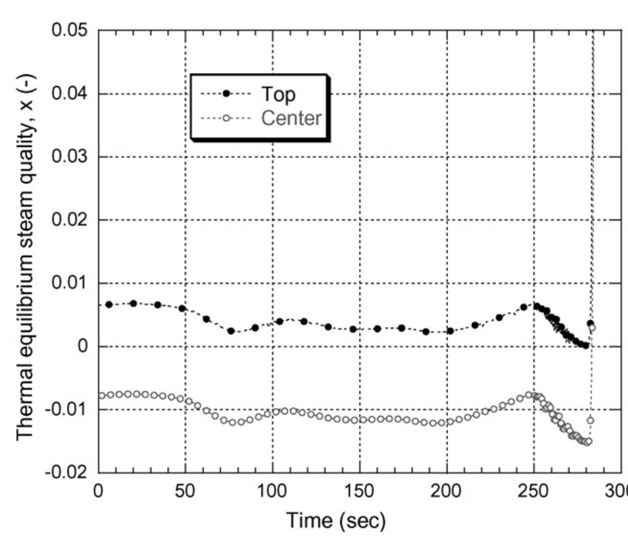

Fig. 6. History of steam quality in the core.

off almost constant value of 0.005 before the turbine trip. This is the similar tendency as the present calculation although the value is slightly high. This means that the void fraction in the core was not increased during the pump coast-down although many researchers believed the enhancement of void generation during the pump coast-down. The measured behavior of water level in the separator drum supports this fact. If void fraction in the core increased, water level in DS must be increased by water from the core. Since the thermal-hydraulic behavior before the power excursion is traced by the code without any discrepancies, it is concluded that the voiding was suppressed during pump coast-down by the system pressure increase. However, it goes without saying that the pump coast-down was important to aid the voiding in the core when another perturbation was added. Only the positive reactivity insertion is supposed to be the direct cause of the accident. This event becomes a trigger of the accident and increases the void fraction in the core, which introduces the enormous reactivity into the core. A void fraction in the low steam quality region has a characteristic that the fraction increases sharply as the steam quality increase as shown in Fig. 7. Therefore, the amount of increase of the void fraction under the transient was large especially in the lower half of the core. Since the prediction accuracy of void fraction is very important in the present analysis, an empirical void–quality correlation proposed by Mochizuki and Ishii (1992) verified with data measured in a pressure tube for various system pressures is adopted.

In the present analysis, the peak reactivity by the positive scram is estimated approximately 0.93β that is not as large as the values estimated by Chao et al. (1988) and Landeyro and Buccafurni (1991). If this reactor was operated at much higher power condition or core flow rate was small, the void fraction in the core was high and an increment due to the trigger of the positive reactivity was small. If core flow rate was half of the experiment, the power increase remained by a factor of 5 of the initial power 200 MW under the assumption of the same magnitude of positive scram. Therefore, eight pumps operation

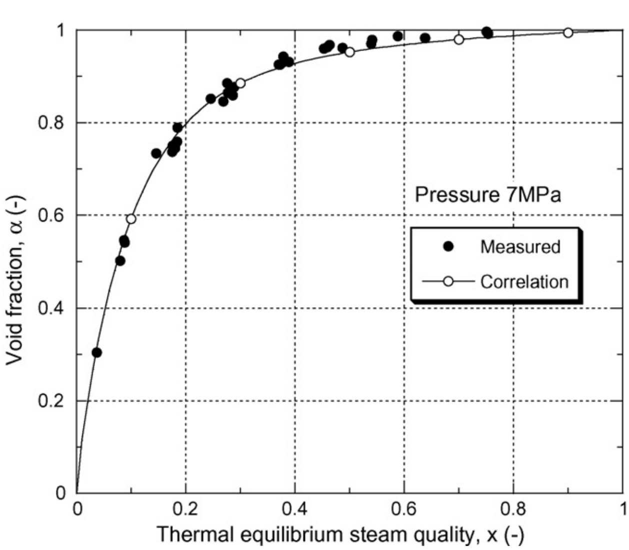

Fig. 7. Relationship between steam quality and void fraction.

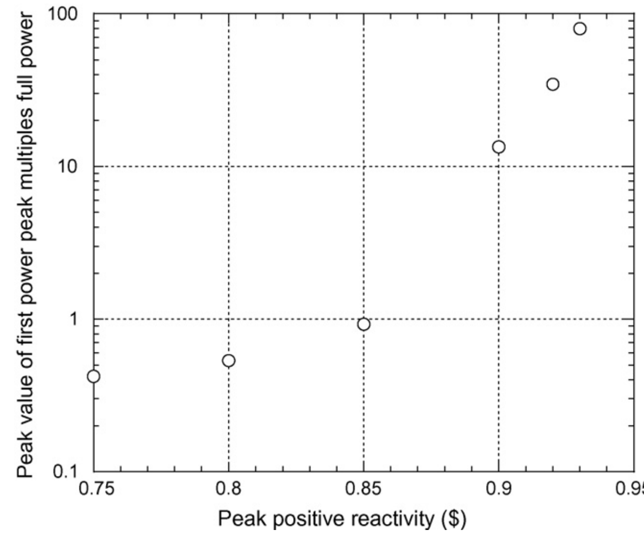

Fig. 8. Relationship between peak positive reactivity and peak reactor power.

that supplies a large amount of coolant into the core is one of the cause of the accident.

Fig. 8 shows a relationship between peak reactivity for positive scram during 6 s and peak reactor power at first peak. As shown in this figure, peak power increases sharply as increasing the peak reactivity by the positive scram. When positive scram is introduced into the core, it triggers to generate voids in the fuel channels especially in the region of lower half of the core. It results in a power increase by void reactivity.

Many researchers calculated this event with various assumptions just before the pump coast-down. Some uncertainties can be introduced in the calculation by this reason. Calculation from the equilibrium conditions at 1:19:00 is preferable. When they calculate this event at the time point close to the accident, they have to take a steady state to start the transient calculation. Since several perturbations were added to the reactor, the initial steady state might be different from the real condition. This could be one of reasons why many researchers could not calculate the Chernobyl accident although they used sophisticated 3D codes.

Although the NETFLOW code cannot calculate the second power peak, this event is discussed here. As reported by Ishikawa et al. (1987), pellets were fragmented and could release mechanical energy. It is obvious that many pressure tubes were damaged by the first power peak, and the second peak was generated by the pipe break caused by the energetic force and/or pressure tube ablation by melted fuel. According to the ablation experiment by Shimozaki et al. (1999) using melted alumina and a Zr–2.5% Nb alloy pressure tube with 4.3 mm in thickness submerged into water, penetration time was in the range of 0.83–2.3 s for several cases. When heat capacity of UO_2 and thickness of pressure tube of 4 mm is considered, the pressure tubes of the Chernobyl reactor might be broken within 2 s. Since the surroundings are high temperature graphite stacks, large amount of vapor might be generated. In some case, generated vapor had lifted up the upper shielding structure that all pressure tubes and control rod guide tubes were penetrated. Generated vapor in the core and withdrawal of control rods might cause the crucial explosion that is second peak.

3.2. Possibility of cavitation in pumps

Martinez-Val et al. (1990) and Nordström (1988) insisted that cavitation in the pump might occur. Fletcher et al. (1988) also calculated the cavitation using RELAP5/mod2. However, calculated important plant parameters such as pressure and water level of the drum separator shifted from the measured points with significant differences, something might affect the NPSH of re-circulation pumps. Chan et al. (1988) explained the unlikeness of the pump cavitation in their letter. In the present calculation, degradation of pumping power is not considered. If flow rate through the reactor is not as large as reported one, voiding may play much more important role to increase the reactor power.

When one verifies the trend of re-circulation flow rate reported by USSR, there is a stepwise decrease at around 1:23:10. It seemed us that the cavitation occurs at around this period. Therefore, the operating conditions at this moment should be checked carefully.

Non-dimensional specific speed n_s of the pump is calculated by the following equation:

$$n_s = NQ^{1/2}(gH)^{-3/4} \tag{3}$$

where N is rotation speed (1/s), Q the volumetric flow rate (m^3/s), g the gravitational acceleration (m/s^2), and H is the head (m).

The value n_s is calculated as 8.43×10^{-2} when characteristic data of pump is given. Cavitation coefficient σ is defined as follows as a function of NPSH, H_{sv} (net positive suction head):

$$\sigma = \frac{H_{sv}}{H} \tag{4}$$

$$\rho g H_{sv} = P_{in} - P_s \tag{5}$$

If cavitation coefficient is in the following condition:

$$\sigma \geq 2.78 n_s^{4/3} \tag{6}$$

cavitation will not occur in the pump according to the empirical experiences.

Pressure at the inlet of the pump, P_{in}, is calculated by NET-FLOW as 6.542 MPa. Since coolant temperature at the inlet of the pump is calculated as 277.9 °C, saturation pressure P_s is 6.219 MPa. H_{sv} is calculated as 57.285. Therefore, cavitation coefficient becomes 0.286. This value is large enough compared to the right hand side of Eq. (5). Therefore, cavitations will not be generated in the pump at this moment. After this moment, the system pressure is increased due to closure of the stop valve. Hence, there is an enough margin for cavitation. The NPSH requirement of the pump is 23 m according to the survey of papers. Simple comparison between suction pressure and saturation pressure can conclude that there is a margin for cavitation. It can be concluded that the pump cavitation might not occur when the pump trip is initiated. Since the net positive suction head is increasing during the pump coast-down as indicated indirectly in Fig. 5, there are enough margins to cavitation. This result coincides with the explanation of Hall (1988).

4. Conclusion

Plant parameters before the accident were traced by the NET-FLOW code on the basis of reported results and information on the plant. A good agreement was obtained between the operational trends and prediction. The calculation from 1:19:00 is preferable in order to escape from uncertainties that are possibly introduced in the initial steady state calculation by various assumptions. The present analysis shows that the reactivity trigger more than 1.0β during 3 s is not always necessary to cause this grade of accident. Most possible candidate of the trigger is the reactivity insertion by the positive scram which increases the void fraction in the core and the void feedback causes the enormous power increase. The importance of the void–quality correlation should be taken into account especially in the present situation. If the run-down test of turbo-generator was conduced under the condition where the void fraction was much higher than the value before the accident or the number of re-circulation pumps was half, the crucial accident would not occur.

Pump coast-down is not a direct cause of the accident because void generation in the core during the pump coast-down was suppressed by the system pressure increase due to the stop valve closure but is very important event to help the void generation after the reactivity insertion by the positive scram. There is no symptom that the cavitation may occur in the pump.

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