TECHNICAL REPORT

Transient Analysis for Design of Primary Coolant Pump Adopted to JAERI Passive Safety Reactor JPSR

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In a course of a design study of the JAERI passive safety pressurized water reactor (JPSR), a complete loss-of-flow transient caused by a trip of all pumps was analyzed with the RETRAN code to determine an inertia of canned-motor pump utilized as the primary coolant pump and to confirm feasibility of the design condition. This transient was selected because the pump had a low inertia rotor inducing fast flow coastdown, and among the transients in which the pump had dominant effect on the departure from nucleate boiling (DNB), the analyzed transient was severest in view of the DNB occurrence. The DNB threshold was related, based on sensitivity calculations, with the coolant density reactivity coefficient and the pump inertia. From the calculations, it was concluded that the pump inertia higher than 250 kg·m² (8% of the ordinary PWRs) was necessary for preventing the DNB occurrence for the present design of JPSR, regardless of the actuation of the reactor scram. The DNB occurrence could be prevented only by the inherent nature of the reactor core which reduced the power by insertion of negative coolant density reactivity during the transient and this was one of major features of JPSR. It was shown by a rough estimation that the necessary condition could be practically realized by incorporation of a cylindrical-type flywheel.

KEYWORDS: JPSR reactor, passive safety, design, loss of coolant flow, canned pump, moment of inertia, density reactivity, departure from nucleate boiling, reactor accidents, transient analysis, RETRAN code

I. INTRODUCTION

In Westinghouse-type pressurized water reactors (ordinary PWRs), the safety under emergency conditions is attained not only by inherent safety nature of the reactor core but also by engineered safety features and highly technical actions by operators. In order to realize a higher level of the safety in a future nuclear power plant, various passive safety reactor concepts, such as PIUS(1), AP-600(2)(3), MS-600(4) and the system integrated pressurized water reactor (SPWR)(5), have been proposed.

Also at the Japan Atomic Energy Research Institute (JAERI), a concept of a JAERI passive safety reactor (JPSR) is being developed for realizing a new PWR concept which requires less operationnal and maintenance work than ordinary PWRs(6). In the JPSR, canned-motor pumps were selected for the main reactor coolant pumps as in the AP-600, in consideration of their high reliability(2). The canned motor in this pump eliminates the shaft seal and the sealant supply system. The former enhances safety by eliminating the possibility of a shaft seal loss-of-coolant accident (LOCA), and the latter contributes to simplify the chemical and volume control system (CVCS) and the auxiliary coolant supply system. However, it is difficult to attach a large flywheel to an ordinary canned-motor pump, which hence has low moment of inertia of the rotor. The influence of fast flow coastdown following trip of all pumps can be severe in view of DNB occurrence. Therefore, a rotor with high inertia, actuation of reactor

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scram or other countermeasures may be required to prevent the DNB occurrence.

Considering the above situation, the present analysis has been carried out to evaluate required inertial moment for the canned-motor pump and necessity of reactor scram to prevent the DNB occurrence. Based on the calculated results, incorporation of a flywheel to the pump is discussed to confirm feasibility of achieving the necessary inertial moment.

II. ANALYTICAL METHOD

1. Features of JPSR Concept

As presented in the reference (6), one of the most important concept of the primary system design of JPSR is adoption of an enhanced inherent matching nature of core power generation and heat removal from the primary coolant system, in other words, a highly inherent load-following capability. If the core power inherently follows change in heat removal rate from the primary coolant system to the secondary system with small thermal expansion of the primary coolant to be absorbed by a practical size of the pressurizer, such a reactor system may have more safety and load-following capability. Because a possibility of occurrence of a loss-of-coolant accident caused by a stuck open of relief valves of a pressurizer can be reduced due that a closed primary coolant system can be realized due to small expansion of the coolant. In order to realize such a reactor system, JPSR adopts following ideas: (a) elimination of the boron from the primary coolant for realizing large coolant density reactivity coefficient, (b) adoption of a low linear heat generation rate in the core for reducing Doppler effects, and (c) adoption of a large pressurizer for dumping pressure changes and for absorbing thermal expansion of the primary coolant. Based on these basic ideas, the following concept has been constructed for the 600 MWe-class, “JP-600”. The major design parameters of JP-600 are listed in Table 1.

The gross thermal and electric outputs are set to be 1,853 MWt and 630 MWe, respectively. As shown in Fig. 1, the pressure vessel installs a typical reactor core of an ordinary 3-loop PWR with 17×17 type fuel bundles by modifying the thermal output and the number of fuel bundles so as to satisfy necessary conditions for realizing the highly inherent load-following capability derived in a previous work(7). The geometry data of fuel rod and fuel bundle are the same as those of the 17×17 fuel bundle in the ordinary PWRs. However, the average linear heat generation rate is set to be about 75% of the ordinary PWRs, because the Doppler effects on power change are reduced due to small fuel temperature change during transient. The reactivity coefficients due to changes in fuel temperature, coolant temperature and density, listed in Table 1, have been specified by the nuclear design calculations(8) under the conditions of the 5% enriched uranium and no chemical shim.
JPSR has two primary coolant loops. Each loop consists of a hot leg, two cold legs and an once-through steam generator (OTSG) which is analogous to the OTSG designed by the B&W company as shown in Fig. 2. In each loop, two primary coolant pumps are installed in the hot leg because change in the primary coolant temperature in the hot leg is smaller than that in the cold leg during a load-following operation. Due to the advantages as described before, the canned-motor pumps were adopted as the main reactor coolant pumps. However, during flow reduction accidents, the canned-motor pump has a shortcoming of low inertial moment in view of the criterion of the DNB ratio (DNBR) which is defined as a ratio of the DNB heat flux to the local heat flux. The design value of the inertial moment is determined in the present analysis.

2. Evaluation Method

Three transients are normally treated in the safety analysis for the ordinary PWRs: the partial loss-of-flow, the pump seizure and the complete loss-of-flow events. The latter transient is considered to be the severest in view of the DNB occurrence because the trip of all pumps are assumed, while the failure of only one pump is assumed in the former two events. Accordingly the complete loss-of-flow event is selected to determine the inertia(9).

The occurrence of DNB is evaluated by the criterion of DNBR=1 for the transients treated in the present study. DNBR is estimated by simulating the hottest bundle by a single channel separately from the average core without consideration of cross flow using the RETRAN-02/MOD 3 code(10). The DNB heat flux is calculated by the B&W-2 correlation which is incorporated in RETRAN. The heat transfer correlations used before the DNB occurrence are of Dittus-Boelter and Thom for single phase liquid and nucleate boiling heat transfer regions, respectively.

3. Input Model

The input data model the loss-of-flow transient due to a trip of all pumps in JPSR. From the objective of the analysis, only an early phase of the transient is analyzed during
about 10 seconds after the trip of all primary coolant pumps.

The input data represent only the primary coolant system and the reactor core as shown in Fig. 3. The secondary coolant system has no significant effects on DNBR during the first 10 seconds because the transport time of coolant from the OTSG to the core is about 8 seconds in JPSR under the rated condition. Therefore, the secondary systems, such as the main steam line and the feedwater system, are not modeled in the present calculation. Instead, heat removal from the primary coolant is modeled by a simple heat exchanger model incorporated in the RETRAN code. In this model, the history of heat removal rate is given explicitly by an input table. In the present calculation, the heat removal rate is set to be constant during the transient.

The reactor core is modeled by two channels: the average channel representing 144 fuel bundles with three vertical nodes and the hottest channel with 12 vertical nodes with peaking factors of 1.62 and 1.24 in horizontal and vertical directions, respectively. The hottest channel is attached by two heat slabs in which the one models the 263 average rods and the other the hottest rod. Since the local peaking factor of the hottest rod is not given by the nuclear design calculation, it is assumed to be 1.2 based on the typical value for the ordinary four-loop PWR(11). In order to ensure conservatism, the engineering hot-channel factor of 1.03 and the uncertainty in the nuclear hot-channel factor of 1.05 are considered in a hot channel based on the safety analysis of the ordinary PWRs. The cross flow between two channels is not modeled in the present analysis for conservative evaluation. The pressure loss through the core is set to that of the ordinary PWR.

The reactor power is calculated by the point reactor kinetics model. The reactivity components considered are the changes in coolant density, coolant temperature and fuel temperature (Doppler effect). The reactivity coefficients used in the calculations are set to be those of the BOEC because the coolant density reactivity coefficient is smaller and more conservative than that of the end of equilibrium cycle (EOEC). The six delayed neutron groups and eleven fission products are considered for the fission power and the decay power, respectively. Analytical condi-
tions of the reactor scram are set as follows based on the safety analysis of the ordinary PWRs. The reactor scram is assumed to be actuated by a low coolant flow at 87% of the rated value with a delay time of 1.0 s. The total scram reactivity is set to be that of the BOEC given by the nuclear design calculation and the stroke time is assumed to be 3 s. The insertion curve of the scram reactivity is assumed to be analogous of that of the ordinary PWRs.

The transient characteristics of the canned-motor pumps are modeled by the pump component model incorporated in RETRAN which uses the homologous law for representing the pump head and torque characteristics and solves the moment balance equation of the rotor for representing the rotational speed transient. Since the pump is assumed to have the same characteristics as the Westinghouse type pump with a specific speed of 780 (m, m³/s, rpm) (5,200 (ft, gpm, rpm)) installed in the ordinary four-loop PWRs, the homologous curves for that pump incorporated in RETRAN are used. The pump head is set to be 110 m which is the required head to maintain the steady state at the rated flow condition listed in Table 1 and is the same as that of the B&W plant. The rated values used in the present analysis satisfy the similarity law of the centrifugal pump and are listed in the Table 2. The inertial moment is treated as a variable parameter in the analysis. Since the value of 3,110 kg·m² is used in the safety analysis of the ordinary PWR, the fraction to this value is used in the discussion. The base case is assumed to be 8% of the ordinary PWRs based on the preliminary analysis.

The steady state is established for the rated flow rate, pressure and core inlet temperature conditions with 102% reactor power for conservative estimation. The complete loss-of-

### Table 2 Pump parameter used in calculation

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Rated pump head</td>
<td>110 m</td>
</tr>
<tr>
<td>Rated volumetric flow</td>
<td>6.45 m³/s</td>
</tr>
<tr>
<td>Rated torque</td>
<td>8,821 kg·m</td>
</tr>
<tr>
<td>Rated fluid density</td>
<td>667 kg/m³</td>
</tr>
<tr>
<td>Rated speed</td>
<td>1,906 RPM</td>
</tr>
</tbody>
</table>

flow accident is initiated by tripping the electric power supplied to all coolant pumps at the initiation of transient calculation.

### III. RESULTS AND DISCUSSION

#### 1. Base Case Results

Figure 4 shows the core inlet flows in the average channel and the hot channel. As shown in the figure, slightly steeper flow reduction is calculated in the hot channel. This is caused by higher void fraction due to higher heat flux in the hot channel. The flow rate in the hot channel becomes minimum at about 2 seconds after the initiation of the transient. Due to combined effect of the flow reduction and the core power reduction, the maximum void fraction is experienced at about 3 seconds. Due to a large volume of the pressurizer, the pressure increase is less than 0.2 MPa during the transient and small enough for maintaining integrity of the pressure boundary.

The core power history is shown in Fig. 5. The core power decreases due to insertion of negative density reactivity immediately after the initiation of the transient. At the time when the reactor scram is initiated at 1.5 s which is determined by the scram condition described in the subsection II.3, the core power already decreased by about 50% of the initial core power. As shown in Fig. 6, the minimum DNBR is calculated to be 1.37 at 3.05 s in the second node upstream the hot channel exit. The reactor scram is not
still completed at that time.

2. Dependency of DNBR on Inertial Moment

In order to know the effect of the inertia on the DNBR, sensitivity calculations are performed with changing the inertia. Also the effect of the reactor scram is analyzed by performing a case without scram.

The calculated result is show in Fig. 7. This figure shows that the trend of DNBR dependency is almost the same in both the cases with and without scram. If the inertia is higher than 20% of the ordinary PWR, the minimum DNBR (MDNBR) is kept to be 2.46 which is the minimum value of DNBR during the steady state. This is because DNBR increases from the initial value during the transient since the rate of decrease in the heat flux at the fuel rod surface is larger than that of the DNB heat flux due to slow flow coast-down. In the region of the inertia less than 20% of the ordinary PWRs, DNBR decreases with decrease of the inertia. This is because the rate of decrease in DNB heat flux becomes higher than that of the surface heat flux with decrease in the inertia. In the region of inertia less than 7% of the ordinary PWR, DNB occurs regardless of actuation of the reactor scram since the decreasing rate of the DNB heat flux is too fast. Only in the region from 7% to 20%, the effect of the scram appears. In the cases with scram, DNB does not occur when the inertia is higher than 7% of ordinary PWRs. On the other hand, the cases without scram with inertia higher than 8% do not experience DNB. The effect of the scram corresponds to difference in inertia by 1%. Based on the above results, it is shown that, if the inertia is larger than 8% of the ordinary PWR (250 kg·m²), DNB does not occur regardless of the scram actuation.

3. Effect of Coolant Density Reactivity Coefficient

In order to investigate the DNB threshold, the sensitivity calculations are carried out. In the calculations, only the density reactivity coefficient and inertia are changed, because the magnitude of the Doppler reactivity insertion is small as shown in Fig. 5. The density
reactivity coefficient of the present design of the JPSR core changes by 5.2% of the BOEC value during the equilibrium fuel cycles. However, since the reactivity coefficient may change in a future improved design of JPSR and should be determined by the nuclear calculation, it is treated as a sensitivity parameter in the present study.

The calculated results are shown in Fig. 8. This figure shows that DNB does not occur in the region above the threshold line that corresponds to a boundary between the regions in which DNB occurs and does not occur. In the region of the higher inertia, the reactor scram shifts below the threshold. However in the region of lower inertia, the effect of the reactor scram becomes small because the core power should decrease quickly before the scram becomes effective to prevent the DNB occurrence.

Based on the results described above, it is concluded that the inertia of the canned pump adopted for the present design of JPSR must be higher than 8% of the ordinary PWR, 250 kg·m², for inhibiting DNB occurrence regardless of actuation of reactor scram.

4. Discussions
The B&W-2 correlation used in the calculation is valid in the ranges of the mass flux from 1,017 to 5,425 kg/m²·s and the flow quality from −0.03 to 0.20. However, in the present calculation, the calculated flow conditions are slightly out of the ranges. At the time of MDNBR in the base case, the mass flux and the quality were 676 kg/m²·s and 0.258, respectively. Based on the above situation, in order to confirm adequacy of the RETRAN results, subchannel analyses were performed with the COBRA-IV-I code.(14)(15). The EPRI-Columbia correlation(16) was used in the COBRA-IV-I calculation. The calculation was performed with the same conditions as the base case of the RETRAN calculations but without the reactor scram. The MDNBR was estimated to be 1.825 which was larger than the RETRAN results of 1.092. Based on the calculated results, it was found that the major contributor to the difference of the MDNBR was the flow model. The subchannel analysis shows the RETRAN model can estimate the MDNBR with large safety margin. Similar result was given in the other calculation with a different inertia.

According to the information described in the reference (3), a base model of the canned-motor pump used for the design study of AP-600 has low rotating inertia of 42.14 kg·m² which is considered to have no enhancement of inertia such as flywheel. In JPSR, since the inertia of 250 kg·m² is needed, the inertia of 208 kg·m² should be added to the model. This additional inertia is realized by installing a cylindrical-type flywheel made by a heavy metal such as uranium alloy with a size of 0.3 m I.D., 1.0 m O.D. and 0.5 m length. When we use the casing with a diameter of 1.2 m, which is practi-
cally reasonable size, the flywheel can be installed inside the casing. Since the configuring of existing canned-motor pumps is not published, more rigorous discussion should be made after data will be published.

Under the complete loss-of-flow accident in the ordinary PWRs, the reactor scram with control rods is needed for preventing the DNB occurrence even adopting usual coolant pumps which have larger inertial moment than the canned-motor pumps. One of the reason is that the coolant density reactivity coefficient is set to be zero in the safety analysis and the reactor scram is considered as the only contributor to reduce the reactor power, since the boron concentration in the primary coolant is high and the reactivity effect is small in BOEC. The same situation can be considered to occur in some of the new passive safety reactor concepts. In JPSR, since the chemical shim is eliminated to realize an inherent matching nature of the core power and the heat removal from the primary coolant system, the coolant density reactivity has a large value even in BOEC. As seen in the results, the inherent nature of the JPSR core contributes to prevent the DNB occurrence even in the failure of the reactor scram, and enhances safety in JPSR during the complete loss-of-flow accident.

IV. CONCLUSION

In order to identify the necessary specification of the inertial moment of the canned-motor primary coolant pump to be adopted in JPSR, the complete loss-of-flow accident caused by trip of all pumps, which is considered to be severest in view of the DNB occurrence, was analyzed with the RETRAN code. The threshold of DNB occurrence has been related with the coolant density reactivity coefficient and the pump inertia. Based on the analysis, it is concluded that the pump inertia higher than 250 kg·m² (8% of the ordinary PWRs) is the necessary condition for inhibiting the DNB occurrence regardless of actuation of the reactor scram for the present design of JPSR having a high density reactivity coefficient. The inherent nature of the reactor core contributes to prevent the DNB occurrence in the transients with failure of the reactor scram. It was shown by a rough estimation that the necessary condition could be practically realized by incorporation of a cylindrical-type flywheel.

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REFERENCES


