Invited Article

A PRELIMINARY EVALUATION OF UNPROTECTED LOSS-OF-FLOW ACCIDENT FOR A PROTOTYPE FAST-BREEDER REACTOR

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ABSTRACT

In the original licensing application for the prototype fast-breeder reactor, MONJU, the event progression during an unprotected loss of flow (ULOF), which is one of the technically inconceivable events postulated beyond design basis, was evaluated. Through this evaluation, it was confirmed that radiological consequences could be suitably limited even if mechanical energy was released. Following the Fukushima-Daiichi accident, a new nuclear safety regulation has become effective in Japan. The conformity of MONJU to this new regulation should hence be investigated. The objectives of the present study are to conduct a preliminary evaluation of ULOF for MONJU, reflecting the knowledge obtained after the original licensing application through CABRI experiments and EAGLE projects, and to gain the prospect of in-vessel retention for the conformity of MONJU to the new regulation. The preliminary evaluation in the present study showed that no significant mechanical energy release would take place, and that thermal failure of the reactor vessel could be avoided by the stable cooling of disrupted-core materials. This result suggests that the prospect of in-vessel retention against ULOF, which lies within the bounds of the original licensing evaluation and conforms to the new nuclear safety regulation, will be gained.

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1. Introduction

In the original licensing application for the prototype fast-breeder reactor, MONJU, the event progression during an unprotected loss of flow (ULOF), which is one of the technically inconceivable events postulated beyond design basis, was evaluated [1]. Through this evaluation, it was confirmed that radiological consequences could be suitably limited...
even if mechanical boundary energy was released. Following the Fukushima-Daiichi accident, a new nuclear safety regulation [2] has become effective in Japan, in which the significant accident sequences to be evaluated for light water reactors are prescribed. In order to investigate the conformity of MONJU to the new regulation, the significant accident sequences required for light water reactor should also be evaluated for sodium-cooled fast reactors. The Japan Atomic Energy Agency has been involved in the safety evaluation of MONJU from the viewpoint mentioned above.

The objectives of the present study are to conduct a preliminary evaluation of ULOF for MONJU, reflecting the knowledge newly obtained after the original licensing application, and to gain the prospect of in-vessel retention (IVR) for the conformation of MONJU to the new regulation.

Because the Core Damage Frequency of MONJU, considering the severe-accident measures for the prevention of core disruptions, is remarkably low \( (\sim 10^{-7}/\text{reactor}/\text{yr}) \) [3], the event progression during hypothetical core disruptive accidents should be evaluated along the most probable scenario for deterministic safety assessment. Although the conservative evaluations are conducted so as to assess the effect of phenomenological or analytical uncertainties, conservative conditions due to such uncertainties are not superposed because of this lower Core Damage Frequency. In the present study, the event progression derived by ULOF is evaluated based on the concept described above, with the knowledge newly obtained through CABRI experiments [4,5] and EAGLE projects [6,7], which are efficiently reflected in the evaluation methodologies and computational analytical tools.

2. Event progression in ULOF

The ULOF case will be caused by the unsuccessful operation of reactor shutdown systems under the loss-of-flow condition due to a coastdown of the primary cooling system. MONJU has two independent shutdown systems consisting of primary control rods and backup control rods, where the backup control rod system is prepared as a severe-accident measure for the prevention of core disruptions. Therefore, the core disruption derived from ULOF means the complete failure of redundant/diverse shutdown systems under the loss-of-flow condition.

In the evaluation of event progressions during ULOF, the whole sequence was categorized into the following three phases according to the core disruption status: (1) initiating phase, (2) transition phase, and (3) postaccident material relocation/postaccident heat removal (PAMR/PAHR) phase. The outline and progression of these phases are illustrated in Fig. 1.

In the initiating phase, fuel pin disruption caused by coolant boiling due to loss of flow would result in axial fuel dispersion in a subassembly (SA). In the original licensing application, the initiating phase was evaluated using the SAS3D code [8], and the calculation result under conservative conditions showed 380 MJ in mechanical energy.

In the transition phase, a molten-core pool would be formed due to the failure of SA walls, and the molten fuel would be discharged through the control rod guide tubes (CRGTs). In a past calculation related to the licensing application, the transition phase was evaluated using the two-dimensional SIMMER-III code [9,10], and the calculation result under conservative conditions showed 150 MJ in mechanical energy.

In the PAMR/PAHR phase, molten fuels discharged through CRGTs would be relocated toward the low-pressure plenum (LPP) and fragmented/quenched by sodium coolant in the LPP. The fragmented/quenched fuel particles would form a so-called debris bed, and the decay heat generated in the debris bed would be stably removed by natural convection or forced convection with the restarted pony motor as an accident.
management measure. In the past evaluation for the PAMR/PAHR phase, however, uncertainties were involved in molten fuel fragmentation and debris bed formation, and the stable cooling of discharged core materials were discussed supposing the ideal debris bed condition.

Against the three phases described above, the Japan Atomic Energy Agency has newly conducted a series of preliminary evaluations reflecting the newest knowledge and computational analytical tools. The methodology of the present study and its results are described in the next section.

3. Evaluation of event progression reflecting newest knowledge

The evaluation methodology of the event progression for the initiating and transition phases basically follows the calculation procedures related to the past licensing applications. In the present evaluations, however, the computational analytical tools were appropriately revised, reflecting the experimental knowledge newly obtained after the original licensing application.

The evaluation methodology for the PAMR/PAHR phase, on the other hand, should be established based on theoretical considerations, experimental database, and state-of-the-art analyses, because the past evaluation was semiquantitatively conducted under several assumptions regarding molten fuel fragmentation and debris bed formation.

The methodology of the present study and its results are described in the following subsections.

3.1. Initiating phase

In the present evaluation of the initiating phase, the newest version of the SAS4A code was applied to simulate the event progression instead of the SAS3D code that was used in the original licensing application of MONJU.
The main contributor of positive reactivity feedback in the initiating phase is coolant void reactivity. By contrast, fuel Doppler, fuel axial expansion, and axial dispersion of disrupted fuel driven by fission gas would provide negative reactivity. Through the CABRI [4,5] and TREAT [11] experimental programs, an effective experimental database has been established for the initiating phase. Based on this database, (i) coolant boiling, (ii) fuel axial expansion, (iii) fuel disruption and axial dispersion in the boiling coolant-channel region, (iv) cladding rupture in the nonboiling coolant-channel region, and (v) FCI (fuel–cooler interaction) void development have been well understood. The SAS4A code [12] applied to the present evaluation for the initiating phase, which has mechanistic models corresponding to each of the important elements, has been effectively validated with this database. Through this established and reliable evaluation method, the relationship between the key design parameters and the severity of the consequence of the initiating phase has been analyzed with a theoretical approach [13,14].

In the present SAS4A evaluations, the calculation geometry, as shown in Fig. 2, was used, in which the third part of axisymmetric core in End of High-burned Equilibrium Cycle condition was modeled. In order to evaluate the most probable scenario and assess the effects of uncertainties, the following conditions were calculated:

(a) Reference condition: The void and Doppler reactivities were set in the designed values. The fuel pin disruption and axial fuel dispersion were simulated by the standard model predicted from the CABRI experiments. Concerning the fuel pin disruption, in particular, the pin failure position and the failure propagation were evaluated by taking into account the cladding intensity and cavity pressure loading, contrary to the past evaluation in which the pin failure was conservatively determined based on the molten fuel ratio only. The negative reactivity feedback due to fuel axial expansion and the fuel dispersion driven by fission gas were also most probably simulated based on the CABRI experiments. Thus, the reference condition was calculated under the best estimate assumption eliminating excessive conservativeness.

(b) Conservative condition: In order to assess the uncertainty in the void reactivity and Doppler coefficient, these values were varied within twice the standard deviation, which would correspond to ±20% of void reactivity and ±14% of Doppler reactivity. The uncertainty range above was estimated using the newest nuclear design method, ADJ2000R. Other calculation conditions were similar to the reference condition.

The transient of reactivity and normalized power in the reference condition evaluated by SAS4A is shown in Fig. 3, in which the contents of reactivities are displayed. As shown in Fig. 3, the negative reactivity feedback due to fuel dispersion should become effective prior to the prominence of void reactivity due to coolant boiling, and should be superior to the positive reactivity caused by cladding dispersion. Fig. 3 suggests that the prompt criticality should be avoided in the initiating phase under the reference condition. Because the net reactivity would reach a quasi-static state at 26.98 seconds after the onset of ULOF and the core region would consist almost entirely of coolant void as shown in Fig. 4, the calculation was connected to the SIMMER code at this time, so as to evaluate the transition phase described in the next subsection.

Concerning the conservative condition, by contrast, the maximum values of net reactivity for various cases are displayed in Fig. 5, where the maximum values of net reactivity were obtained from the reactivity transient calculations in the same way as in the reference case. The calculation results are summarized in Table 1, in which the maximum power and core fuel temperature are also displayed, as well as the maximum net reactivity. As shown in Fig. 5 and Table 1, the prompt criticality should not take place within any combinations of conservative void and Doppler reactivities.

The present evaluation for the initiating phase can be summarized as follows: (I) the uncertainties to be considered for the initiating phase could be much reduced by introducing the newest nuclear design methods and by applying the newest SAS4A code in which the experimental database obtained in the CABRI programs, etc., were efficiently reflected. (II) Contrary to the past evaluation, the present SAS4A calculations showed that the reduction of excessive uncertainty could bring the elimination of mechanical energy release due to prompt criticality, even under the conservative conditions.

The material distribution and core status at the end of the initiating phase, evaluated by SAS4A under the reference condition, would be connected to the subsequent evaluation for the transition phase.

### 3.2. Transition phase

In the present evaluation of the transition phase, the three-dimensional SIMMER-IV code was applied to simulate the event progression instead of the two-dimensional SIMMER-III code that was used in past evaluations related to the licensing application of MONJU.
The main contributor of positive reactivity feedback in the transition phase is the molten fuel compaction in the molten core pool that will be formed due to the failure of SA walls after the initiating phase. By contrast, the discharge of molten fuel through CRGTs would provide negative reactivity feedback. Through the EAGLE [6,7] experimental programs, an effective database has been established for the fuel discharge behaviors in the transition phase. Based on this database, the following aspects have become well understood: (i) thermal loading from disrupted core materials to CRGT structure, (ii) failure of CRGT structure and formation of discharge path, (iii) reasonable treatment of FCI pressure, and (iv) blockage possibility inside CRGTs. The three-dimensional SIMMER-IV code applied to the present evaluation, which was developed by extending two-dimensional SIMMER-III [9,10], has been effectively validated with this database. SIMMER-III/IV are multivelocity field, multiphase, multicomponent, Eulerian, fluid dynamics codes coupled with a space-dependent neutron kinetics model. The conceptual overall framework of SIMMER-III/IV is shown in Fig. 6. The entire code consists of three elements: the fluid dynamics portion, the structure (fuel pin) portion, and the neutronics portion. The fluid dynamics portion is interfaced with the structure portion through heat and mass transfers at structure surfaces. The neutronics portion provides a nuclear heat source based on the mass and energy distributions calculated by the other portions. The experimental knowledge and physical models obtained in the EAGLE programs were efficiently reflected and appropriately validated in the current SIMMER-III/IV [15,16].

In the present SIMMER-IV evaluations, the calculation geometry as shown in Fig. 7 was constructed by connecting the material distribution and core status evaluated by SAS4A for the initiating phase. The position of CRGTs in the horizontal cross section could be suitably represented in the three-dimensional SIMMER-IV, contrary to the past two-dimensional SIMMER-III in which CRGTs were modeled in annular shape in the core region. Thus, in two-dimensional geometry, the radial motion of molten core materials would be inhibited by the annular-shaped CRGTs, and nonphysical axisymmetric/coherent fuel motion would overestimate the fuel compaction behavior leading to mechanical energy release. The concerns peculiar to two-dimensional geometry cited above could be suitably mitigated by introducing three-
dimensional calculations. Here, in order to evaluate the most probable scenario and assess the effects of uncertainties, the following conditions were calculated.

(a) Reference condition: The initial condition of the transition phase was connected from the SAS4A calculation under the reference condition of the initiating phase. The discharge behavior of molten fuel through CRGTs was simulated by the standard model based on the EAGLE experiments, and the penetration behavior of disrupted fuel into the pin bundles of the lower/upper axial blankets was also simulated using the standard model based on some freezing experiments [17]. In addition, the effect of FCI due to CRGT failure, which may enhance the molten fuel compaction leading to recriticality and mechanical energy release, was treated in the standard parameters calibrated through the validation analyses of EAGLE experiments. Thus, the reference condition was calculated under the best estimate assumption eliminating excessive conservativeness.

(b) Conservative condition: In order to assess the uncertainty in the discharge and penetration behaviors on the discharge and penetration behaviors.

Table 1 — SAS4A results under various conservative conditions.

<table>
<thead>
<tr>
<th>Void Doppler</th>
<th>-2σ</th>
<th>-1σ</th>
<th>Conservative (Nominal)</th>
<th>+1σ</th>
<th>+2σ</th>
</tr>
</thead>
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<tr>
<td>-2σ</td>
<td>1.14</td>
<td>0.741</td>
<td>0.822</td>
<td>0.857</td>
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<td></td>
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<td>6</td>
<td>9</td>
<td>10</td>
<td>15</td>
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<td></td>
<td>2463</td>
<td>2379</td>
<td>2461</td>
<td>2462</td>
</tr>
<tr>
<td>-1σ</td>
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<td>0.743</td>
<td>0.800</td>
<td>0.845</td>
<td>0.863</td>
</tr>
<tr>
<td></td>
<td></td>
<td>6</td>
<td>9</td>
<td>12</td>
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<td></td>
<td>2559</td>
<td>2385</td>
<td>2464</td>
<td>2470</td>
</tr>
<tr>
<td>Conservative</td>
<td>1.00</td>
<td>0.792</td>
<td>0.812</td>
<td>0.853</td>
<td>0.887</td>
</tr>
<tr>
<td>(Nominal)</td>
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<td>8</td>
<td>10</td>
<td>14</td>
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<td>2644</td>
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<td>2504</td>
</tr>
<tr>
<td>+1σ</td>
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<td>0.783</td>
<td>0.822</td>
<td>0.846</td>
<td>0.891</td>
</tr>
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<td></td>
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<td>8</td>
<td>10</td>
<td>13</td>
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<td></td>
<td>2405</td>
<td>2407</td>
<td>2502</td>
<td>2495</td>
</tr>
<tr>
<td>+2σ</td>
<td>0.86</td>
<td>0.787</td>
<td>0.826</td>
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</table>

Values in upper row indicate net reactivity, in the middle row indicate maximum power (\(P_0\)) and in the lower row indicate maximum core fuel temperature (K).

Fig. 6 — Framework of SIMMER-III/IV code.
providing negative reactivity feedback, these behaviors were artificially suppressed. Concerning the discharge behavior, the start of molten fuel discharge due to CRGT failure was delayed until core pressure reached 3 MPa. Concerning the penetration behavior, by contrast, the penetration length was suppressed to half of the standard model by adjusting the related parameters. In addition, the effect of FCI on the molten fuel compaction was assumed to be the allowable overestimated value by adjusting the sodium amount contributing to FCI phenomena. Other calculation conditions were similar to the reference condition.

The transient of reactivity and normalized power in the reference condition evaluated by SIMMER-IV is shown in Fig. 8, and the transient of fuel distribution in each region is displayed in Fig. 9. As shown in Fig. 8, the reactivity would not reach recriticality in the transition phase under the reference condition. In the early stage of the transition phase up to about 28.0 seconds, a power transient with a maximum reactivity of 0.94$ would be caused due to the falling of upper dispersed fuels, which would have migrated upward in the initiating phase. The failure of CRGT structure and the formation of discharge path would take place by 30 seconds for all CRGTs, and the molten fuels in the core region would be discharged through CRGTs. The fuel discharge through CRGTs would result in a remarkable subcritical state with about $-50$ in reactivity. The fuel inventory remaining in the core region would ultimately (after 35 seconds) reach about 50%, and that discharged below core bottom (including LPP), above core top, and into core periphery (including radial blanket) would ultimately be about 20%, 15%, and 15%, respectively.

Concerning the conservative condition, by contrast, the transient of reactivity and normalized power are displayed in Fig. 10, where recriticality would take place around 29.5
seconds due to fuel compaction because the discharge of molten fuel through CRGTs was artificially suppressed under the conservative condition. The mechanical energy released by this recriticality would be about 30 MJ, as shown in Fig. 11. Contrary to the past two-dimensional conservative evaluation showing 150 MJ in mechanical energy, the present evaluation under the conservative condition showed that the released mechanical energy would be significantly reduced because the nonphysical axisymmetric/coherent fuel compaction peculiar to two-dimensional geometry was appropriately mitigated.

The present evaluation for the transition phase can be summarized as follows: (I) the three-dimensional evaluation methodology using the SIMMER-IV code was developed and validated, in which the experimental database obtained in the EAGLE program was efficiently reflected so as to appropriately simulate the fuel discharge behavior through CRGTs and (II) contrary to the past two-dimensional evaluation, the present three-dimensional calculation using SIMMER-IV showed that recriticality would not take place under the reference condition, and that the mechanical energy release under the conservative condition would be significantly suppressed due to the mitigation of nonphysical axisymmetric/coherent fuel compactions peculiar to two-dimensional geometry.

The amount of fuel ultimately discharged from the core region under the reference condition would be about 50% in the present evaluation. In the subsequent evaluation of the PAMR/PAHR phase, it was assumed that all of the discharged fuel would be relocated into the LPP. This assumption could envelope the uncertainties in the amount of discharged fuel and the condition of the debris bed from the viewpoint of stable cooling in LPP and achievement of IVR.

3.3. PAMR/PAHR phase

In the past evaluation of the PAMR/PAHR phase, as shown in Fig. 12, the discharged molten fuel through CRGTs should be fragmented/quenched in LPP, and the decay heat generated in the debris bed consisting of the fragmented/quenched fuels should be stably cooled, where the coolability of debris bed was discussed based on the Lipinski model [18]. The past evaluation, however, involved uncertainties in the fragmentation of molten fuels and the formation of the
debris bed. In the present evaluation of the PAMR/PAHR phase, the coolability of discharged fuels in LPP was demonstrated using the super-COPD code [19], FLUENT code [20], and heat balance calculations without supposing the fragmentation and debris bed formation.

As the uncertainties in the fragmentation of molten fuels and the formation of the debris bed would be derived by the coexistence of molten fuel injection and sodium vapor development in the limited space of LPP, it was assumed in the present study that the molten fuel discharged through CRGTs would not be fragmented at all and would be accumulated as an ingot in LPP. This assumption could envelope the uncertainty in debris bed conditions from the viewpoint of its coolability, because the surface area of the ingot consisting of the discharged materials would be significantly limited compared to the fragmented particles. Based on the assumptions above, the coolability of discharged fuel through CRGTs was evaluated using the following procedure. (i) Evaluation of sodium flow rate by super-COPD: Reflecting the blockage condition around the core region at the end of the transition phase, the sodium flow rate in LPP was evaluated with super-COPD code under the condition illustrated in Fig. 13. (ii) Evaluation of velocity distribution in LPP by FLUENT: Reflecting the sodium flow rate evaluated by super-COPD, the velocity distribution around LPP was simulated by FLUENT code using the calculation geometry as shown in Fig. 14. (iii)
Evaluation of coolability in LPP by heat balance calculation: Reflecting the velocity distribution around LPP evaluated by FLUENT, the coolability of discharged fuel in LPP was estimated using heat balance calculations with a geometric model as shown in Fig. 15.

The sodium flow rate, evaluated by super-COPD under the condition shown in Fig. 13, is displayed in Fig. 16. Considering the blockage condition around the core region, the sodium flow rate in LPP would be equivalent to that in the radial blanket region of 76 kg/s shown in Fig. 16.

The velocity distribution around LPP, evaluated by FLUENT using the calculation geometry shown in Fig. 14, is displayed in Fig. 17. The coolant velocity at the upper surface of discharged materials in LPP and that at the lower surface of the core catcher, which would have the greatest effect on the coolability of discharged materials, would be \( u_1 = 18 \) cm/s and \( u_2 = 0.4 \) cm/s, respectively.

The coolability of discharged fuel in LPP, estimated by the heat balance calculations using the geometric model in Fig. 15, is summarized in Fig. 18, in which the evaluation results of several conservative cases for assessing the sensitivity of \( u_1 \) and \( u_2 \) are also displayed. As shown in Fig. 18, the case using the evaluated velocities (Case 1; \( u_1 = 18 \) cm/s and \( u_2 = 0.4 \) cm/s) suggests that the decay heat generated by discharged fuel could be stably cooled in LPP. In the case with half of \( u_1 \) (Case 4; \( u_1 = 9 \) cm/s and \( u_2 = 0.4 \) cm/s) or half of \( u_2 \) (Case 2; \( u_1 = 18 \) cm/s and \( u_2 = 0.2 \) cm/s), the decay heat could also be successfully removed in LPP. In the case using half of \( u_1 \) and half of \( u_2 \) (Case 5; \( u_1 = 9 \) cm/s and \( u_2 = 0.2 \) cm/s), partial failure of the LPP structure might take place, but the discharged fuels could be cooled/retained on the core catcher, where the integrity/failure of the LPP structure was determined based on a structure intensity model, as shown in Fig. 19, in which the effect of gravitational/thermal load on the creep rupture are suitably considered. Therefore, the

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**Fig. 15** – Calculation model to evaluate coolability in low-pressure plenum (LPP).

**Fig. 16** – Sodium flow rate evaluated by super-COPD.

**Fig. 17** – Sodium velocity around low-pressure plenum (LPP) evaluated by FLUENT.
discharged fuel through CRGTs would be stably cooled in LPP by decay heat removal under the condition of the evaluated velocities without assuming molten fuel fragmentation.

The present evaluation for the PAMR/PAHR phase can be summarized as follows: (I) the decay heat removal of discharged fuel through CRGTs was evaluated without assuming the molten fuel fragmentation and debris bed formation in LPP and (II) contrary to the past evaluation involving uncertainties in molten fuel fragmentation and debris bed formation, the present evaluation showed that stable cooling of discharged core materials could be achieved even if fragmentation was incomplete.

4. Discussion for the prospect of IVR

Based on the present and related past evaluations, the whole spectrum of event progressions initiated by ULOF can be displayed in the event tree as shown in Fig. 20. The straightforward sequence (I) in this event tree, without any branching to lower level sequences, corresponds to the event progression evaluated under the reference conditions in the present study.

Because the mechanical energy release due to prompt criticality in the initiating phase should not take place even under the conservative conditions as described in the section “Initiating phase,” the deviation at the branch heading should not be expressed here.

For the transition phase, by contrast, the deviation caused by the recriticality should be expressed because it would take place under the conservative condition as discussed in the section “Transition phase.” Contrary to the past two-dimensional evaluation showing 150 MJ in mechanical energy under the conservative conditions, the present conservative...
evaluation, taking into account the three-dimensional motion of core materials, showed that the released mechanical energy would be significantly reduced to about 30 MJ. Because the postdisassembly expansion analyses—even assuming the 150 MJ in mechanical energy release—suggests that no impulse/impact pressure and no leakage of sodium from the primary coolant boundary would take place, the IVR expressed by sequence (7) in Fig. 20 could also be achieved in the present evaluation.

Concerning the PAMR/PAHR phase, the blockage and fragmentation of discharged core materials are not assumed in the present study, which should be rather conservative assumptions from the viewpoint of stable cooling in LPP. As discussed in the section “PAMR/PAHR phase” for Fig. 18, the excessively conservative condition, with a coolant velocity at 1 order of magnitude smaller than the evaluated velocity, would result in the failure of the core catcher. The possibility of sequence (3), therefore, is significantly low, and its contribution to the risk of IVR failure would also be negligible. In the related past evaluations under comparatively optimistic conditions, in which the blockage and fragmentation of discharged core materials are assumed, the IVR could also be achieved as shown in sequences (4), (5), and (6) in the event tree.

In order to further consolidate the prospect of IVR, the authors have investigated the coolability of the remaining core materials in the disrupted core region based on the theoretical considerations and numerical evaluations using computational codes such as SIMMER [9,10], FLUENT [20], and ASFRE [21]. A clear vision for the stable cooling of the remaining core materials has been obtained. The present and related evaluations suggest that, in MONJU, IVR could be achieved with an extremely high possibility even if the hypothetical core disruptions are assumed.

5. Conclusions

A preliminary evaluation of ULOF for the prototype fast-breeder reactor, MONJU, was conducted reflecting the knowledge newly obtained after the original licensing application. The present study showed that no significant mechanical energy release would take place, and that thermal failure of the reactor vessel could be avoided by the stable cooling of disrupted core materials. This result suggests that the prospect of IVR against ULOF, which lies within the bounds of the original licensing evaluation and conforms to the new nuclear safety regulation, will be gained.

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References


