Preliminary analysis of the post-disassembly expansion phase
and structural response under unprotected loss of flow accident in prototype sodium cooled fast reactor

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Abstract
For the prototype sodium-cooled fast reactor, MONJU, the mechanical energy and structural response under energetics caused by neutronic power excursion during Unprotected Loss of Flow (ULOF) accident were preliminarily evaluated. In the first licensing of MONJU, pressure-volume relation (P-V relation) was evaluated based on the maximum theoretical work energy possible for an expanding core. It was adopted in the structural response analysis of the reactor vessel as the input. The maximum theoretical work energy is called Fuel Vapor Work Potential (FVWP) in this paper. In the successive studies of the energetics, mechanical energy was evaluated with the code in which mechanistic modelling of core expansion was implemented and this might reduce the Actual Work Potential (AWP) by an order of magnitude below FVWP. In order to evaluate the realistic structural response of the reactor vessel using AWP, method to convert the AWP to the P-V relation is necessary. Therefore, we developed the method to obtain realistic P-V relation based on the AWP by tracing the surface of the expanding core, and then we evaluated the mechanical energy and structural response under energetics during ULOF accident in MONJU using the developed method. The AWP is evaluated to 3 MJ based on the result of the latest ULOF analysis in which FVWP was evaluated to 30MJ, and sodium slug does not impact on the lower surface of the shield plug and no residual strain of the reactor vessel is evaluated. When FVWP is assumed to be 500 MJ as a hypothetical condition covering the conservative energy production, corresponding AWP is evaluated to 33 MJ. In this case, sodium slug impacts on the lower surface of the shield plug and residual strain of the reactor vessel of 0.008% at the maximum is evaluated, however the integrity of the primary boundary is still maintained.

Key words : Sodium-cooled fast reactor, Severe accident, Unprotected loss of flow, Energetics, SIMMER-IV, AUTODYN

1. Introduction

Containment of energetics, which may take place in hypothetical Core Disruptive Accidents (CDAs) of Sodium-cooled Fast Reactor (SFR), has been historically addressed in licensing applications for SFRs as one of the major issues because of the following safety characteristics of SFRs against CDAs: 1) sodium void reactivity is occasionally positive depending on the core design, and 2) a concentration of core materials due to molten fuel/steel relocation potentially results in neutronic power excursion. Energetics in SFR is such that high pressure, which is built by fuel or steel vapor generation due to neutronic power excursion during CDA or by sodium vapor generation due to Fuel Coolant Interaction (FCI), exerts mechanical load on the primary boundary, especially reactor vessel and the shield plugs on it.

In the first licensing application of prototype SFR MONJU, which was developed by Japan Atomic Energy Agency (JAEA), safety evaluation against Unprotected Loss of Flow (ULOF) was conducted as a representative severe accident with the possibility of energetics. In order to accommodate the consequences of such an accident, both energetics
containment and decay heat removal from disrupted core are necessary. MONJU equips systems to cope with them, and the effectiveness of those systems was confirmed through the safety evaluation against ULOF.

Following the accident at TEPCO’s Fukushima Dai-ichi Nuclear Power Station, the new regulatory requirements for commercial power reactors got into force in Japan in 2013. In the new regulatory requirements, measures against severe accidents and the evaluation of their effectiveness are required. In accordance with the requirements, a preliminary evaluation against ULOF was conducted for MONJU in JAEA (Suzuki et al., 2015) reflecting the latest knowledge. In this evaluation, the newly obtained knowledge and improvements of analysis tools fairly reduced the energy production during the neutronic power excursion compared to that estimated in the first licensing. However, if the uncertainty of physical phenomena during the accident progression is taken into account, a certain amount of mechanical energy release was predicted. Therefore, it is of importance to evaluate the consequence of the mechanical energy release to make sure the containment of energetics.

In order to evaluate the consequence, namely the response of the reactor structure, the thermal energy generated during the transient should be converted to the mechanical energy. In the first licensing of MONJU, pressure-volume relation (P-V relation) of the Fuel Vapor Work Potential (FVWP) was used as an input of the structural response analysis of the reactor vessel. FVWP is obtained by integrating fuel vapor pressure along P-V relation of isentropic path to one atmosphere considering the heat of condensation of the fuel vapor and enthalpy decrease of the liquid fuel (Walter and Reynolds (1981)), and was often used as an indicator to express the severity of the energetics.

On the other hand, there exist various mechanisms which can drastically reduce the mechanical energy during the transient. In order to incorporate mechanistically the effects of the physical processes occurring within the expanding core, SIMMER-II code was developed (Bell et al., 1979) at Los Alamos National Laboratory (LANL). The evaluated mechanical energy with SIMMER-II code for MONJU was more than an order of magnitude below FVWP (Kondo et al., 1985). In this paper, the mechanical energy obtained by such mechanistic code is defined as Actual Work Potential (AWP).

In order to evaluate the realistic structural response of the reactor vessel using AWP, method to convert the AWP to the P-V relation is necessary. Therefore we developed the method to obtain realistic P-V relation based on the result of the mechanistic core expansion calculation. In the following sections, background of this study, evaluation method of the realistic P-V relation, and the analysis results of the PDE phase and structural response are respectively described.

2. Background
2.1 Description of the ULOF accident

Typical severe accidents of MONJU are divided into two differently-categorized accident groups: one is called ATWS (Anticipated Transient without Scram) and the other is LOHRS (Loss of Heat Removal Systems). ATWS, unprotected accidents, are characterized by their short grace period from the initiation of the accident to core damage, whereas the LOHRS, protected accidents, has sufficiently long grace period to implement the preventive measures by operators against core damage. Energetic core disassembly is of particular concern in ATWS, and ULOF is the representative accident within ATWS of MONJU with regard to the contribution ratio of it to the core damage frequency (Onoda et al., 2016).

Typical event progression of ULOF is shown in Fig. 1 by connecting the accident phases commonly used in safety evaluations. The first accident phase is the initiating phase, which covers from normal reactor operation to core damage. The path from the initiating phase is divided into two phases: an energetic sequence leading to Post-Disassembly Expansion (PDE) phase and a non-energetic sequence leading to the transition phase. The transition phase also may lead to the PDE phase as a result of a recriticality event. In the energetic sequence, structural integrity is evaluated using mechanical energy obtained in the PDE phase analysis. After the transition phase and PDE phase, post-accident material relocation and heat removal are of concern. In order to contain the accident consequences, both decay heat removal and mechanical energy containment are indispensable.

Recent evaluation of ULOF sequences performed in JAEA (Tobita et al., 1999), (Suzuki et al., 2015) using up-to-date analytical tools revealed that the existence of many inherent mechanisms mitigating the positive reactivity insertion prevented the mechanical energy release due to power excursion in the initiating phase, even though the uncertainties of these mechanisms were considered conservatively. In contrast to the initiating phase, if the conservative assumption such that suppression of molten fuel discharge from disrupted core through control rod guide
tube was applied to the analysis of transition phase, mechanical energy release would be predicted. Therefore, mechanical energy should be evaluated for the path through transition phase energetics in PDE phase analysis.

2.2 The relation between this study and past studies of ULOF accident analysis

Table 1 compares the results of the energetics evaluation for MONJU between this study and the past studies. In the licensing in 1983, FVWP was evaluated but AWP was not evaluated. FVWP was then converted to the P-V relation and the structural response analysis was carried out using the P-V relation. In the past studies in JAEA (Kondo et al., 1985) (Tobita et al., 1999), the AWP was evaluated using SIMMER-II or SIMMER-III code which adopts mechanistic approach for the calculation of the expanding core. The maximum kinetic energy of liquid sodium slug during core expansion was evaluated in their analysis as the mechanical energy, which was represented by the liquid sodium velocity. If the sodium slug velocity was given as the input and the existence of the cover gas was ignored for the structural response analysis, it would have given overestimated pressure transient due to water hammer effect. On the other hand, we evaluated AWP in this study using SIMMER-IV code (Yamano et al., 2009), which is successor of the SIMMER-II and -III code, and the AWP was converted to the realistic P-V relation of the expanding core. This enables us to use AWP as the input to evaluate actual core expansion behavior in the structural response analysis.

Table 1  The relation between this study and past studies against the energetics of MONJU

<table>
<thead>
<tr>
<th>Energy level evaluation</th>
<th>Licensing (1983)*1</th>
<th>Kondo et al. (1985)</th>
<th>Tobita et al. (1999)</th>
<th>Licensing (2006)*2</th>
<th>This study</th>
</tr>
</thead>
<tbody>
<tr>
<td>Energetics in*4</td>
<td>IP</td>
<td>IP</td>
<td>TP</td>
<td>IP</td>
<td>TP</td>
</tr>
<tr>
<td>Tool*5</td>
<td>SAS+VENUS</td>
<td>-----</td>
<td>S-II</td>
<td>S-III</td>
<td>SAS3D</td>
</tr>
<tr>
<td>FVWP (MJ)</td>
<td>380</td>
<td>-----</td>
<td>100</td>
<td>(4100k)*6</td>
<td>330</td>
</tr>
<tr>
<td>PDE phase analysis</td>
<td>Tool*5</td>
<td>-----</td>
<td>S-II</td>
<td>S-II</td>
<td>S-III</td>
</tr>
<tr>
<td>AWP (MJ)</td>
<td>-----</td>
<td>-----</td>
<td>-----</td>
<td>-----</td>
<td>3</td>
</tr>
<tr>
<td>Structural response analysis</td>
<td>Tool*5</td>
<td>PISCES</td>
<td>-----</td>
<td>-----</td>
<td>-----</td>
</tr>
<tr>
<td>Input (MJ)</td>
<td>500*9</td>
<td>-----</td>
<td>-----</td>
<td>-----</td>
<td>3</td>
</tr>
</tbody>
</table>

*1 Japan Atomic Energy Agency, 1983
*2 Sato et al, 2007
*3 Suzuki et al, 2015
*4 IP: Initiating Phase, TP: Transition Phase
*6 FVWP was not evaluated but core average fuel temperature was evaluated
*7 Evaluated mechanical work (AWP) when energy level (FVWP) was assumed to 380MJ as input
*8 Evaluated mechanical work (AWP) when energy level (FVWP) was assumed to 500MJ as input
*9 Value decided to use, evaluated FVWP + some margin
2.3 Applicability of the physical model in SIMMER-IV code to the dominant phenomena in the PDE phase

There are three important energetics-mitigating effects inherently available for limiting mechanical energy generation (Kondo et al., 1985). They are the following: mixing of core materials and resultant heat losses from the fuel, thermal and hydraulic losses when hot core materials are discharged through the Upper Core Structure (UCS) which is composed of upper axial blanket, upper gas plenum, and upper neutron shield, and condensation of a vapor bubble when it expands into the upper sodium plenum.

At the beginning of the PDE phase, steep temperature and pressure distribution depending on the power distribution within the disrupted core region exist. The precipitous expansion of the voiding space within the disrupted core region leads to flattening of the pressure distribution as well as the temperature distribution by mixing of high temperature vapor (fuel and steel) with relatively low temperature material located around the periphery of the disrupted core. This mechanism is foreseen and modeled in the SIMMER-III code (Kondo et al., 2000).

Then the mixture of high temperature and high pressure core materials expands into the coolant channel through the UCS. The mixture is composed of molten fuel, molten steel, fuel vapor, steel vapor and fission gas, and is called CDA bubble hereafter. In this process, CDA bubble temperature decreases by heat exchange with the surrounding structures, and the motion of the CDA bubble is restrained by the narrow coolant channel. Therefore UCS gives large pressure drop between disrupted core region and upper sodium plenum, and this mechanism is the most effective in reducing mechanical energy. The effect of UCS for core expansion was experimentally studied and SIMMER-III code was validated with the experiment (Kondo et al., 2000).

When expanded CDA bubble reaches the upper sodium plenum, the pressure of the CDA bubble gives mechanical load to the reactor vessel through the inner barrel, and the thermal interaction between CDA bubble and liquid sodium takes place. Sodium vapor is consequently generated and grows up in the upper plenum by its pressure, and then accelerates sodium slug in the upper sodium plenum upward. Expansion and condensation behavior of the CDA bubble within the upper plenum was experimentally studied (Saito and Theofanous, 1979) and SIMMER-III code was validated with that experiment (Kondo et al., 2000), as well as the analytical model of sodium vapor generation due to FCI in SIMMER-III code is validated (Morita et al., 1999).

SIMMER-IV code which is utilized in this study for mechanical energy evaluation is the successor of SIMMER-III. Therefore basic physical models of SIMMER-IV to calculate mechanical energy have been validated through these anticipatory studies.

3. Evaluation method of the realistic P-V relation

The realistic P-V relation is evaluated using the result of the SIMMER-IV calculation in the PDE phase. Pressure and volume of the CDA bubble are separately evaluated. Because the pressure near the interface between liquid sodium and the CDA bubble (surface of the CDA bubble) actually acts on the sodium slug, the pressure at the surface of the CDA bubble should be used in the evaluation. The pressure of the CDA bubble surface, $P_{BB}$, is obtained by averaging the local interface pressure, $P_i$, at the whole surface of the CDA bubble, as follows.

$$P_{BB} = \frac{\sum_{i \in \text{SBB}} P_i \alpha_i V_i}{\sum_{i \in \text{SBB}} \alpha_i V_i} \quad (1)$$

The local interface pressure is obtained by finding CDA bubble surface referring to the liquid sodium volume fraction in a cell, as shown in Fig. 2. In calculating $P_{BB}$, summation in space is done only at the surface of the CDA bubble whereas all the voiding volume included in the CDA bubble is taken into account for the summation of volume of the CDA bubble, $V_{BB}$. Volume is obtained by finding CDA bubble volume referring to void fraction and summing up them.

$$V_{BB} = \sum_{i \in \text{SBBV}} \alpha_i V \quad (2)$$
What is most important in this calculation is to properly find the surface of the CDA bubble, and we found the way to confirm the validity of obtained P-V relation by using following equation.

\[ W_{CDABB} = KE_{Na} + IE_{CG} \]  

This equation is obtained by considering that the mechanical work done by the CDA bubble, \( W_{CDABB} \), should be identical to the sum of the kinetic energy increase in coolant sodium, \( KE_{Na} \), and internal energy increase in cover gas, \( IE_{CG} \), provided that the structure does not deform nor absorb the mechanical energy, as shown in Fig. 2. \( W_{CDABB} \) is calculated by summing up the work done by the CDA bubble using the pressure at the surface of the CDA bubble as follows:

\[ W_{CDABB} = \sum_{n=1}^{N} P_{BB,n}^{n} \Delta V_{BB}^{n} \]  

where

\[ P_{BB,av}^{n} = \frac{P_{BB}^{n} + P_{BB}^{n-1}}{2} \]  

\[ \Delta V_{BB}^{n} = V_{BB}^{n} - V_{BB}^{n-1} \]  

\( KE_{Na} \) is obtained by summing up the kinetic energy of the liquid sodium in each cell \( i \), as follows:

\[ KE_{Na} = \sum_{i=SN/AL} KE_{Na,i} \]  

where

\[ KE_{Na,i} = \frac{1}{2} \rho_{Na,i} \alpha_{Na,i} V_{i}^{2} + \frac{1}{2} \rho_{Na,i} \alpha_{Na,i} V_{i}^{2} + \frac{1}{2} \rho_{Na,i} \alpha_{Na,i} V_{i}^{2} \]  

\( IE_{CG} \) is obtained by summing up the work done by the cover gas at each time step as follows provided that the process is adiabatic:

\[ IE_{CG} = \sum_{n=1}^{N} P_{CG,av}^{n} \Delta V_{CG}^{n} \]  

where

\[ P_{CG,av}^{n} = \frac{P_{CG}^{n} + P_{CG}^{n-1}}{2} \]
\[ \Delta V_{CG}^n = V_{CG}^n - V_{CG}^{n-1} \quad (11) \]

\[ P_{CG}^n = \sum_{i \in SCG} P_i \alpha_i V_i \quad (12) \]

\[ V_{CG}^n = \sum_{i \in SCG} \alpha_i V_i \quad (13) \]

In calculating \( P_{CG}^n \), summation in space is done within all the cover gas volume because there is almost no pressure gradient in the cover gas region.

**4. Analysis of the PDE phase**

**4.1 Analysis method, analysis condition and assumptions**

The objective of the PDE phase analysis is to obtain the P-V relation of expanding CDA bubble that will be used in the structural response analysis. SIMMER-IV code (Yamano et al., 2009), a three-dimensional thermal-hydraulic code coupled with space dependent neutron kinetics model, was used to the PDE phase analysis.

![Diagram](image.png)

Fig. 3 Calculation geometry for PDE phase analysis using SIMMER-IV

The calculation geometry is shown in Fig. 3. The calculation geometry of the reactor core is the same as that adopted in the transition phase analysis (Suzuki et al., 2015), where subassemblies of reactor core, radial blanket and neutron shielding were modeled as independent channels (region 1, 2, 3 and 11 of Fig. 3). In the PDE phase analysis, reactor vessel, upper sodium plenum with Upper Internal Structure (UIS) and shield plug above the upper sodium plenum (region 5, 6, 7 and 9 of Fig. 3) are added on the reactor core modelling. Flow guide grid under the UIS and dip plate installed just below the coolant sodium surface are ignored. The existence of thermal shield layer under the shield plug is modelled only by adding their total thickness to the shield plug thickness. The volume of the gas gaps among the thermal shield layer is added to that of the cover gas. This treatment is based on the experimental observation (Saito et al., 1985) in which thermal shield layer was crushed and pressed on the lower surface of the shield plug when the sodium slug impacts on the thermal shield layer.

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Only thermal-hydraulic model of SIMMER-IV code was used in PDE phase analysis because the power history of the core was calculated in the preceding phase, transition phase, using neutron kinetics model of the SIMMER-IV code and the obtained power history is used in the PDE phase analysis.

Cover gas pressure and sodium temperature are set to the value at the rated power operation. Material distribution as well as temperature distribution in the core region is given by the result of transition phase analysis (Suzuki et al., 2015). The axial and radial power distribution and power history are also given by the transition phase analysis. The averaged peaking factors of the axial and radial power distributions are 1.1 and 1.2, respectively.

Following analysis conditions are assumed:

- The escape paths of the cover gas, such as pipe line of the primary argon system and clearance gaps between shield plugs, are ignored: cover gas plenum is assumed as an enclosed space,
- Downward relocation of the core materials through the lower axial blanket region are prevented,
- Fuel and steel blockages formed in the UCS region during the preceding phase are intentionally removed.

When fuel disruption and dispersion takes place during initiating or transition phase, the molten fuel and steel penetrate into the coolant channel within the UCS region and then form a blockage. Such blockage acts as flow resistance and inhibits material relocation from disrupted core region through UCS toward upper sodium plenum. However, there exist large uncertainty on the strength and duration of the blockage due to the lack of experimental information. Therefore, in the PDE phase analysis with SIMMER-IV code, fuel and steel blockages are intentionally removed from UCS region. This treatment is conservative in that the pressure increase due to ejection of CDA bubble into the upper plenum and FCI between CDA bubble and liquid sodium are overestimated.

Two analytical cases of evaluation were carried out, with FVWP as the parameter varying 30 and 500MJ.

30MJ: evaluation for MONJU with reasonable conservativeness based on up-to-date experimental knowledge and analytical method (Suzuki et al., 2015).

500MJ: a hypothetical bounding case for safety assessment.

4.2 Analytical results and discussions
4.2.1 Realistic P-V relation

Evaluated realistic P-V relation of the CDA bubble for both calculation cases are plotted on Fig. 4. The time origin of these calculations is set to the time when FVWP reached at the maximum value during the preceding phase analysis (about 20 ms after the power peak caused by neutronic power excursion).

Figure 5 shows the histories of $W_{\text{CDAB}}$ and $KE_{\text{Na}} + IE_{\text{CG}}$ for case-30MJ and 500 MJ. Good agreement of $W_{\text{CDAB}}$ with $KE_{\text{Na}} + IE_{\text{CG}}$ was obtained when volume fraction of the liquid sodium at the surface of the CDA bubble fulfills the following condition:
0.05 < \alpha_{Na,J} < 0.25.

From Fig. 5, we can confirm that the proper P-V relation was obtained.

4.2.2 Mechanisms of the CDA bubble expansion

The behavior of the CDA bubble expansion is explained for case-30MJ as follows.

At the beginning of the calculation, the surface of the CDA bubble is at around the middle of the UCS region (at the middle of the upper gas plenum region). When CDA bubble starts to expand, fission gas, fuel vapor, steel vapor and the molten core materials (molten fuel and steel) penetrate into the UCS region. The first pressure increase of the CDA bubble at around 30 ms shown in Fig. 6 (~0.6 MPa) is due mainly to fission gas flow into the UCS region, as shown in Fig 7 (~0.6 MPa at 2.1 m). Sodium vapor supports this pressure buildup. At the beginning of the calculation, gaseous phase in the UCS region is composed only of the fission gas and sodium vapor at equal rate. The partial pressure of the sodium vapor observed at 30 ms in Fig. 7 is mainly attributed to the sodium vapor which had originally existed in the UCS region at the beginning of the calculation. Small amount of the liquid sodium remained in the UCS region partly contributed to this pressure buildup. Figure 7 also shows that there is no pressure contribution of the fuel and steel vapor in the middle of the UCS region indicating that they condensed completely in this region.

The steep pressure drop seen on Fig. 7 at 30 ms at around 1.15 m, namely the pressure difference between the core and UCS region, is caused by the sudden decrease of the flow area. All the fuel pin structure below this height is destroyed by ablation and this makes large flow area, whereas they remain intact and original coolant flow area surrounded by the fuel pins is kept above this height. This abrupt geometry change provides such steep pressure drop. After 30 ms, CDA bubble expands with its interface extended into the upper plenum as shown in Fig. 8.
When the surface of the CDA bubble arrives at the lower surface of the UIS (~150 ms, see Fig. 8), the molten core materials come into mixing with the sodium left on the lower surface of the UIS and this results in sodium vapor generation due to FCI. The pressure increase of the CDA bubble from 150 to 260 ms (Fig. 6) is caused by this sodium vapor generation. The pressure drop seen on Fig. 7 at 260 ms at around 2.2 m is also caused by the sudden decrease of the flow area. All the fuel pin structure below this height is destroyed so that this provides such pressure drop.

After 260 ms the pressure of the CDA bubble gradually decreases with the increase of its volume. In this case (case-30MJ), CDA bubble expands until 390 ms when the maximum volume of it is attained, and sodium slug will not impact on the lower surface of the shield plug.

Pressure at the surface of the CDA bubble is dominated by both the pressure in the core region and friction loss between the core and the surface of the CDA bubble in the early stage of the CDA bubble expansion (0-150 ms in this case). After 150 ms, it is dominated by the sodium vapor generated by FCI. Pressure in the core region at 30 ms is dominated by the steel vapor and fission gas, and the contribution of the fuel vapor to that is very low, as shown in Fig. 7. At this energetics level, fuel vapor hardly contributes to the CDA bubble expansion and that FVWP no longer expresses the magnitude of the mechanical load working on the shield plug or reactor vessel.

The overall behavior of the CDA bubble expansion of case-500MJ is roughly the same as that of the case-30MJ except the timings of the events, the levels of the pressure and whether sodium slug impacts on the lower surface of the
shield plug or not.

Calculated FVWP for the case-30MJ is plotted on Fig. 9. It rapidly decreases from 29 MJ to 15 MJ within first 9 ms. This is due to the mixing of the core materials and flattening of the pressure in the core region. Figure 10 shows that the pressure of the core region is flattened and decreases about half of its initial value (from 7.4 MPa to 4.1 MPa) during the first 9 ms. The relocation and condensation of the fuel and steel vapor to the peripheral part of the core result in that the steel vapor dominates the pressure at the central part of the core and fission gas in the peripheral part (see Fig. 7).

![Fig. 9 History of calculated FVWP (case-30MJ)](image)

![Fig. 10 Radial pressure distribution at X/L=0.73 (X: height, L: active fuel length) during 0 - 9 ms (case-30MJ)](image)

Core materials are discharged through the UCS developing steep pressure gradient (see Fig. 7), and the pressure of the core is dominated by the pressure loss in the UCS region and condensation of the fuel and steel vapor at the peripheral part of the core.

From above mentioned analysis, three important energetics-mitigating effects are also confirmed in this study, namely, the pressure flattening at the beginning of the CDA bubble expansion, the steep pressure losses in the UCS region, and condensation of the fuel and steel vapor at the peripheral part of the core. These mechanisms reduce the mechanical work from the level of the FVWP to that at AWP as shown in Table 2.

<table>
<thead>
<tr>
<th>Core averaged fuel temperature (K)</th>
<th>FVWP (MJ)</th>
<th>AWP (MJ)</th>
<th>Reduction (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>3400</td>
<td>30</td>
<td>3</td>
<td>90</td>
</tr>
<tr>
<td>4600</td>
<td>500</td>
<td>33</td>
<td>93</td>
</tr>
</tbody>
</table>

5. Structural response analysis
5.1 Analysis method, conditions and assumptions

The objective of the structural response analysis is to evaluate the integrity of primary containment system, namely, reactor vessel, shield plug and primary heat transport system, against mechanical load resulting from energetic power excursion. In this paper, the analysis of the reactor vessel is focused.

The structural response is analyzed with AUTODYN. AUTODYN is an explicit analysis tool for modeling nonlinear dynamics of solids, fluids, gas, and their interaction (ANSIS Inc., 2013). The calculation geometry is shown in Fig. 11. Most part of the reactor structure is modeled by the Shell solver which is thin and deformable in 2-dimentional cylindrical coordinate. The fluid such as coolant sodium, cover gas and the pressure source (core) is modelled by the Euler solver. Shield plug, UIS, radial blanket subassemblies, and lower axial blanket are treated as a rigid obstacle fixed in the analytical space. The expansion behavior of the pressure source is treated by GASBUG model in which an equation of state is specified by user-supplied P-V relation. The P-V relation is obtained from the result of PDE phase analysis performed with SIMMER-IV, as shown in Fig. 4. Temperature and cover gas pressure are set to those at rated power operation: 530 ℃ and 0.155 MPa. Two calculation cases are evaluated same as those
conducted for PDE phase analysis with FVWP as the calculation parameter: 30 MJ and 500 MJ.

Radial blanket subassemblies are assumed to be rigid obstacle in this calculation, whereas they have potential to absorb mechanical energy by its deformation to some extent in realistic condition. Such deformation energy is distributed to sodium slug acceleration, cover gas compression, and reactor vessel deformation under this assumption and this assumption is somewhat conservative in evaluating reactor vessel response. Inner barrel is neglected in this analysis so that total mechanical load to the reactor vessel is conservatively evaluated. The treatment of the thermal shield layer in this analysis is the same as that treated in the PDE phase analysis.

5.2 Analytical results and discussions

The analysis results were summarized on Table 3. The analysis result of sodium movement in the vessel of case-30MJ and -500MJ are shown in Fig. 12. Coolant sodium does not impact on the lower surface of the shield plug in case-30MJ and no residual strain is calculated, whereas it impacts at 180 ms in case-500MJ. Evaluated residual strain in case-500MJ is only 0.008 % at the maximum at the conical part of the reactor vessel (Fig. 13).

Table 3 Summarized results of the structural response analysis

<table>
<thead>
<tr>
<th>FVWP (MJ)</th>
<th>slug impact</th>
<th>Residual strain (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>30</td>
<td>No</td>
<td>None</td>
</tr>
<tr>
<td>500</td>
<td>Yes</td>
<td>0.008</td>
</tr>
</tbody>
</table>

Calculated strain at 27 m in Fig. 13 is due to the pressure source expansion at an early stage, at around 25 ms into the transient. At the same time, reactor vessel got elongated to the downward direction, and the junction of the conical part and cylindrical part moved inward, and then the maximum residual strain of 0.008% was evaluated at this point (32 m). Calculated strain at 34.5 m in Fig. 13 is due to the sodium slug impact at around 180 ms into the transient. The integrity of the reactor vessel against sodium leakage, namely against boundary rupture, is secured even in case-500MJ.
because breaking strain of the reactor vessel is evaluated to 12% and the calculated residual strains are far below it.

![Material relocation in the upper plenum](image)

Fig. 12 Material relocation in the upper plenum

![Obtained residual strain in case-500MJ](image)

Fig. 13 Obtained residual strain in case-500MJ

6. Conclusions

Preliminary analyses of the PDE phase and structural response were carried out in order to evaluate the potential energetics under ULOF accident in MONJU, and following conclusions were obtained.

1) Evaluation method of the realistic pressure-volume relation which corresponds to Actual Work Potential (AWP) was developed: the method to trace the surface of the expanding core was developed.
2) Reduction of mechanical energy from Fuel Vapor Work Potential (FVWP) to AWP amounts to 90-93%.
3) AWP is evaluated to 3 MJ based on the result of the latest ULOF analysis in which FVWP was evaluated to
30MJ. With this AWP, sodium slug does not impact on the lower surface of the shield plug and no residual strain of reactor vessel is evaluated.

4) When FVWP is assumed to be 500 MJ as a hypothetical condition covering the conservative energy production, corresponding AWP is evaluated to 33 MJ. In this case, sodium slug impacts on the lower surface of the shield plug and residual strain of the reactor vessel of 0.008% at the maximum is evaluated, however the integrity of the primary boundary is still maintained.

The result obtained in the present study shows that MONJU has enough robustness against mechanical load under energetics. Various parametric analyses of PDE phase and structural response will be carried out as the future work in order to reflect those results in probabilistic risk assessment.

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Nomenclature

\( F_{\text{VWP}} \) Fuel Vapor Work Potential, mechanical energy equal to that released by an isentropic expansion of the fuel vapor to one atmosphere [J]

\( AWP \) Actual Work Potential, mechanical energy calculated by mechanistic code, mechanical work that could be done by the CDA bubble [J]

\( I_{\text{E}_{\text{CG}}} \) Internal energy increase in cover gas [J]

\( K_{\text{E}_{\text{Na}}} \) Kinetic energy increase in sodium slug [J]

\( N \) Final time step [-]

\( P-V \) relation Relationship between pressure and volume increase of expanding CDA bubble. Work done by the CDA bubble is obtained by integrating pressure along this relation.

\( P_i \) Pressure in cell \( i \) [Pa]

\( P_{BB}^n \) Pressure of the CDA bubble surface at time step \( n \) [Pa]

\( P_{BB,av}^n \) Time-averaged pressure of the CDA bubble surface between time steps \( n \) and \( n-1 \) [Pa]

\( P_{CG}^n \) Pressure of the cover gas at time step \( n \) [Pa]

\( P_{CG,av}^n \) Time-averaged pressure of the cover gas between time steps \( n \) and \( n-1 \) [Pa]

\( S_{BB} \) Set of numbers of the cell where the surface of the CDA bubble exists [-]

\( S_{BBV} \) Set of numbers of the cell where CDA bubble exists [-]

\( S_{CG} \) Set of numbers of the cell where cover gas exists [-]

\( S_{NAL} \) Set of numbers of the cell where liquid sodium exists [-]

\( V_i \) Volume of the cell \( i \) [m\(^3\)]

\( V_{BB}^n \) Volume of the CDA bubble at time step \( n \) [m\(^3\)]

\( V_{CG}^n \) Volume of the cover gas at time step \( n \) [m\(^3\)]

\( \Delta V_{BB}^n \) Volume increase of the CDA bubble between time steps \( n \) and \( n-1 \) [m\(^3\)]

\( \Delta V_{CG}^n \) Volume increase of the cover gas between time steps \( n \) and \( n-1 \) [m\(^3\)]

\( W_{CDA BB} \) Mechanical work done by the CDA bubble [J]

\( n \) Time step [-]

\( v_{Na,i,x} \) x-component of the liquid sodium velocity in cell \( i \) [m/s]

\( v_{Na,i,y} \) y-component of the liquid sodium velocity in cell \( i \) [m/s]

\( v_{Na,i,z} \) z-component of the liquid sodium velocity in cell \( i \) [m/s]

\( \alpha_i \) Void fraction in cell \( i \) [-]

\( \sigma_{Na,i} \) Volume fraction of the liquid sodium in cell \( i \) [-]

\( \rho_{Na,i} \) Liquid sodium density in cell \( i \) [kg/m\(^3\)]

References