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THE EVALUATION APPROACH OF SPENT NUCLEAR FUEL INTERGRITY BY USING FINITE ELEMENT METHOD

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Abstract. *In Korea, the storage capability for Spent Nuclear Fuel(SNF) in the nuclear power plants will be expected to be saturated in the near future. SNF have to transfer from spent pool in the plants to another area by using the safety way. Handling and transportation of SNF are common key processes for the SNF management. During this activity, SNF is required to evaluate structural integrity by using the proven evidences of the tests or other reliable methodologies. The SNF test, however, is accompanied by very high cost and safety issues because of handling irradiation object. Alternatively, the computational methodologies(ex: Finite Element Method, Numerical Method, etc.) are usually employed as a technical solution based on the information of other effects. In this paper, the simulations for analyzing the fuel integrity is conducted by using verified finite element model during the regulated normal and postulated conditions. At first, the model is designed as the many beam elements considering full size. Second, the verification process was performed for having the mechanical properties and dynamic characteristic at the model. At the end, SNF drop was simulated at the several conditions. In the result of this paper, SNF has stress within elastic zone at 0.3 m drop but SNF is expected to be damaged result from free drop simulation of 9.0 m.*

1 INTRODUCTION

In 1978, Kori 1 began commercial operation as first nuclear power plant in Korea. Since then, the nuclear power industry has developed. In today, a total of 25 nuclear power plants are currently in operation. Nuclear power generation uses principle of converting from heat energy of nuclear fuels to electricity. At this time, fission products are generated at the fuels loaded in the reactor inevitably. For that reason, evaluating integrity of nuclear fuel assemblies is an essential process to be ensured confining uranium and irradiated products to fuel rods during handling and transportation, including operation of the power plants. To evaluat-

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ing the integrity of nuclear fuels, it is important to identify the material/mechanical properties and to estimate the fuel behavior under those conditions.

Nuclear fuel assemblies consist of fuel rods containing uranium pellets, grids with complex shapes for supporting the rods, top / bottom nozzle connected to reactor structures, and guide tubes which control rods for controlling the heat output are inserted. The manufactured nuclear fuel assemblies are installed in new fuel transport containers for transporting to nuclear power plants without any damage. Generally, the transported fuel assemblies are loaded into the reactor by a crane and produce heat in harsh environmental conditions of high temperature and high pressure for five years. During the loading period, the metal materials constitute oxidized, hydrogenated, and deteriorated through the chemical reaction with the cooling water. After the power generation, they wait in the spent pool at the power plants until they are post-treated or disassembled.

Due to the spent fuel in standby, the storage capacity of the spent pool will soon be saturated at the nuclear power plant in Korea. SNFs have to be transported by using the safety way to the other area. Therefore, the handling and transportation of SNF for post-treatment is a rising issue faced by the Korea's nuclear industry [1, 2].

In order to confirm the safety of the handling and transportation of the un-irradiated fresh nuclear fuel, it can be explained by reflecting the material and mechanical characteristics and analyzing each accident condition for predicting nuclear fuel behavior in advance. For this purpose, various mechanical tests about unirrdiated fuel assemblies are conducted by mechanical test facility when developing nuclear fuel [3]. Investigating irradiated nuclear fuel assemblies are also a very good way to determine mechanical properties includes all effects. But it is difficult in reality due to important issue connected with radiation shielding of the facility and tester safety. Alternatively, the classical mechanics method and finite element analysis are widely used to predict the behavior of nuclear fuel assemblies in nuclear reactors based on test result and logical deduction [4, 5].

For safety of SNF transportation, radiation shielded containers are required. In order to prove the integrity of the container, 10CFR71 [6] of US Federal Regulations define normal accidents and postulated conditions of various cases. Several studies have been conducted to determine that these requirements are met. The US Department of Energy's research institutes assessed their integrity in consideration of the characteristics of the SNF and selected representative SNF to carry out integrity analysis together with the transport container. The one research regarding rail way transportation of SNF conducted by US nuclear fuel companies and research institutes[7, 8].

Prior to this study, the two studies were performed by KEPCO NF to selecting the representative nuclear fuel, which is expected to be the most damaged among many type of nuclear fuels supplied to Korea. The one is the hand calculation considering of geometry data and the other is the finite element analysis through the simplified single beam reflected dynamic characteristics [9, 10]. Both models are designed from several hundred fuel rods with another components to one beam. These models are useful to come to a rough conclusion in the short time. However, the hand calculation model is difficult to describe the behavior of the nuclear fuel according to the time and estimate impact force under regulated normal and postulated conditions. In case of simplified single beam model, dynamic behavior of the fuel can be obtained by solving numerical equations. The stress of fuel rod estimated by reflecting displacement of the beam model results under assumption that is rod motion is equal to the fuel model. Therefore, simplified single model is limit to analyzing stress at the each part for evaluation of integrity.

In this paper, the model was designed as many beam elements considering full size to describe the motion of the nuclear fuel and evaluate the integrity of the fuel rods under handling and transportation. At first, the representative nuclear fuel selected previous studies was modeled by using geometry data of un-irradiated condition. To the next, the model was verified by comparing with result of mechanical test. The some parameters was changed in the model to make the SNF conditions. And drop simulations were performed under selected conditions for finding maximum stress and damage area of the model.

2 MODELING AND MODEL VERIFICATION

2.1 Nuclear fuel model

In previous studies, impact simulations were performed on a single beam with multiple lumped masses by using finite element model verified dynamic characteristics compare with the test results. The beam model can describe the dynamic and impact behavior of the un-irradiated fresh nuclear fuel from a macroscopic point of view. However, if one characteristic (e.g., contact condition with fuel rods and grids, temperature effect of material, etc.) is changed in the nuclear fuel, the model is difficult to predict the exactly fuel motions during impact simulations since the changed effect is not to directly reflected in the model. For estimating the behaviors of spent nuclear fuel, it is necessary to full scale model that is modeling by several components. In this study, the FE model as shown in figure 1 (a) was designed by the components and the FE model of the nuclear fuel consists mainly of beam elements and analyzed by ANSYS 16.0.

2.2 Verification process

To verify the actual assemblies with the dynamic characteristics of the fuel model considered in Section 2.1, the following sequence was performed. Firstly, mass was checked comparing with the actual nuclear fuel and analytical model weight. And the model analysis was performed to confirm natural frequencies and mode shapes of the FE model. Lateral free vibration test result was used for energy dissipation of the model in time domain. Finally, some parameters of the model were controlled for matching with the lateral and the axial impact test.

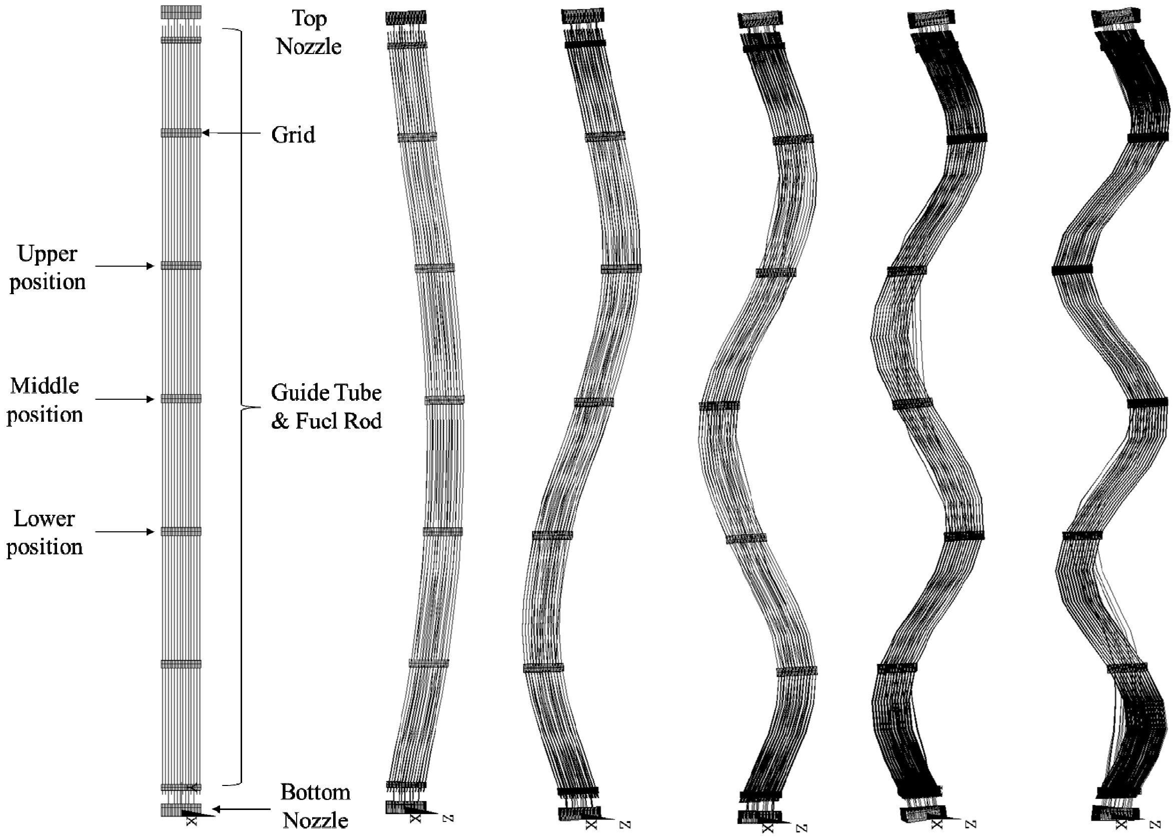
2.3 Vibration characteristics

The mass of the fuel modeled as shown in Figure 1 (a) was matched to within 0.35 %. The natural frequencies and mode shapes were extracted as shown in Table 1 and Figure 1 (b) to (f) by the modal analysis results. It can be confirmed that the natural frequency difference between the test and analysis result within about 10 %, but Mac value is decreased at the 3rd mode due to difference of joint connection mechanism of grids and guide tubes.

Mode number	1	2	3	4	5
Frequency difference (%)	0.30	10.02	0.20	3.57	2.76
Mac value	0.92	0.81	0.59	0.70	0.83

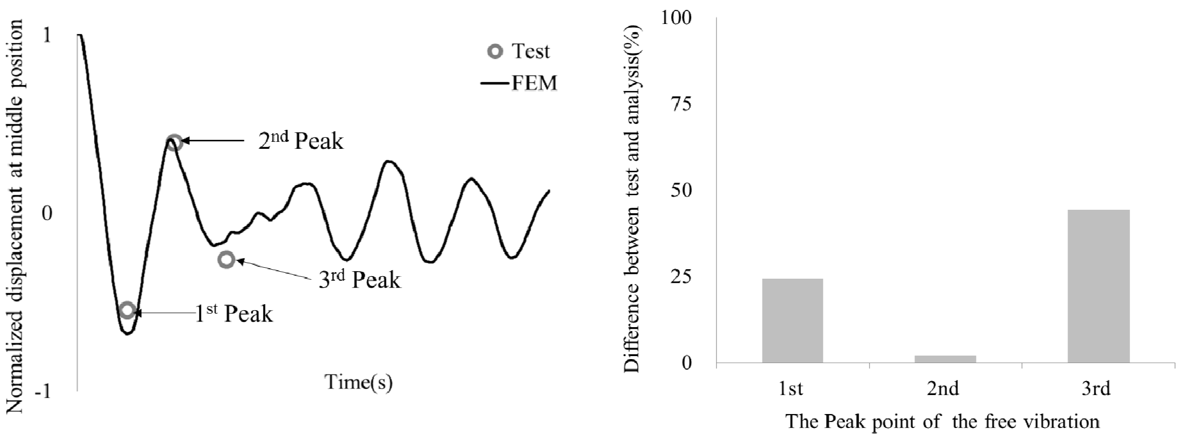
Table 1: Mode frequency between the test and the analysis

Figure 2 (a) shows the data obtained by free oscillation result from initial displacement of mid grid in the lateral direction. Nuclear fuel has decrease amplitude according to increasing time due to energy dissipation mechanism such as frictional force between grid and rod contact, material damping, impact of fuel tube and uranium pellet, etc. The results of the finite element analysis and the test results agree well in the first cycle. But after second peak, there was a large error due to the difference of some nonlinear characteristics between test and model result. However, since most of the impact behavior occurs before the first peak approach, it is enough for estimating fuel behavior in the time domain. And the displacement of the model at the first peak is larger than test so that the model is conservative in terms of impact simulations.



(a) FE model (b) 1st (c) 2nd (d) 3rd (e) 4th (f) 5th

Figure 1: Finite Element(FE) model and Mode shape



(a) Free vibration (b) Difference

Figure 2: Lateral free vibration of the nuclear fuel

2.4 Impact characteristics

Nuclear fuel is exposed to various environments during operating, handling and transportation. To prove integrity of nuclear fuel from the damage cause by external force, impact tests were carried out at the time of nuclear fuel development. Figure 3 (a), (b) and (c) show the impact force characteristics when three grids are collided with the surface of a wall according to initial displacement of the mid gird. The impact force in the graph was normalized by maximum force generated in the lateral impact test. The force was measured caused by free drop from constant distances and shown in Figure 3 (d). The value was normalized by maximum axial impact force obtained during the axial drop test. In the case of the finite element model, the lateral and axial characteristics were large difference to compare with test data at the small displacements but the difference is decreased within 20 % as the displacement increased.

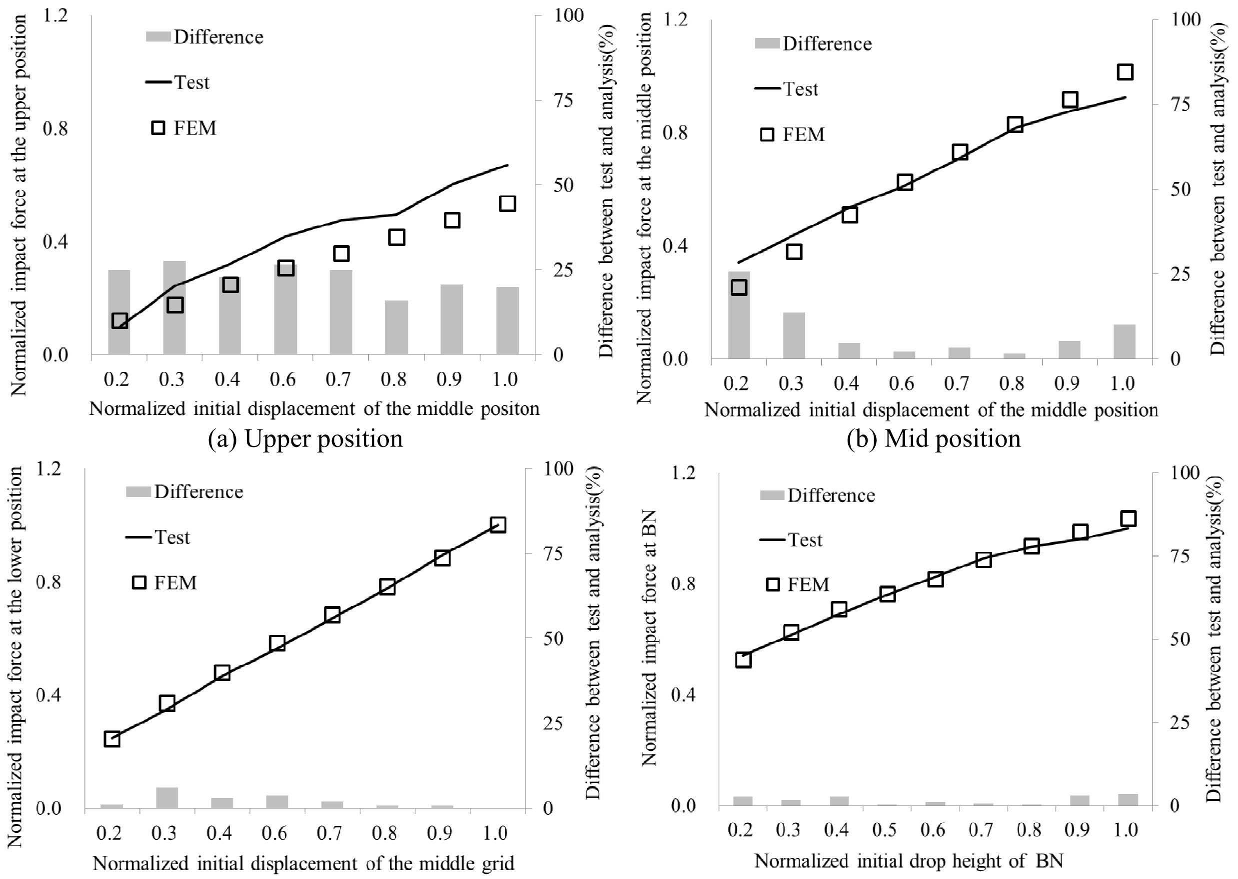
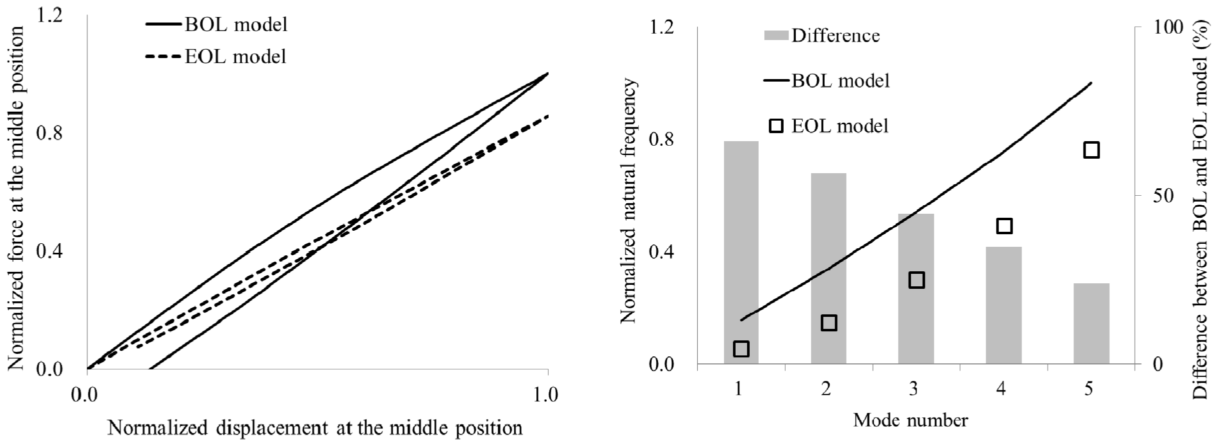


Figure 3: Impact results of the tests and analyses

2.5 Consideration of EOL condition

Nuclear fuel assemblies are loaded into the reactor and exposed to cooling water for a long time under high temperature and pressure conditions, resulting in creep, hydrogenation of materials, and irradiation effect. Especially, the lateral stiffness of the fuel is decreased cause by the relaxation of the spring and the dimple supporting the fuel rod. At the Beginning Of Life(BOL), the loads of spring and the dimple are acting on the fuel rods for keeping the distance between fuel rods and then the loads are removed at a high temperature and growth of the grids by irradiation or thermal relaxations and the diameter reduction of the fuel rod by high pressure. The gaps between the grid and fuel rods occur, and the stiffness of the spring and the dimple is also changed by irradiation effect. Therefore, some of the parameters were adjusted in the finite element model to describe the End Of Life(EOL) condition, which was verified by considering only the contact conditions except the material properties. As a result, the lateral stiffness and energy dissipation(e.g., hysteresis loss, etc.) were reduced because the frictional loads are removed from the EOL model in the ideal state as shown in Fig. 4, and the mode frequency of the EOL model was also reduced when the mode shape of the fuel rod is only considered.



(a) Hysteresis of lateral axis (b) Mode frequency

Figure 4: analytical characteristic of the EOL model

3 DROP SIMUALTION DURING HANDLING AND TRANSFORTATION

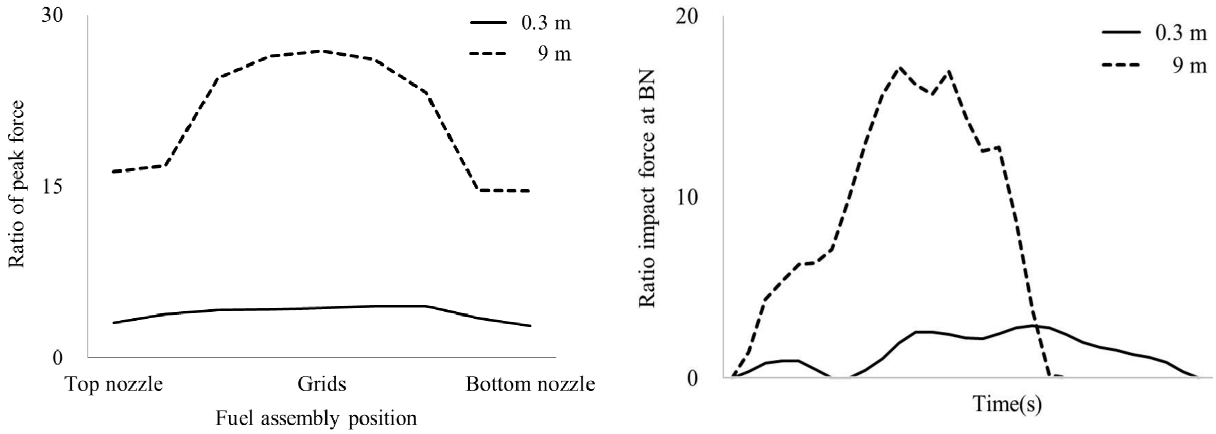
3.1 Drop condition

When the cask containing SNF assemblies is dropped according to normal / hypothetical scenarios during handling and transport, and impact force is acted on SNF. This can cause large stresses on the fuel rods supported by the fuel skeleton and expose the radioactive material due to fuel rod damage. Therefore, the 10CFR 71 of US federal regulations contains various terms for the prevention of serious fuel rod damage and release of radioactive material from the packaging under normal and postulated accident conditions. The height of free drop is 0.3 m considering total weight including cask at the normal condition and 9 m drop height is required at the postulated accident. In these analyses, the drop simulations were performed at 0.3 m and 9.0 m for one fuel, assuming that the cask has rigid body motion without any deformation.

3.2 Drop result

Figure 5 (a) shows the deformed fuel shape when the impact force reaches the maximum value during the drop simulation. The grids and top/bottom nozzle as well as fuel rods are crashed with the ground surface. So, impact forces acted on the fuel rod directly. Figure 5 (b) shows the impact force on the top/bottom nozzle and the grids under each drop condition, based on the maximum force generated in the lateral impact test in section 2.4. In the case of 0.3 m drop, impact force was generated almost uniformly on the fuel, but in 9.0 m drop condition, the impact force on the fuel rod portion was largely affected. Figure 5 (c) shows the load in the time domain during the axial drop. When the nuclear fuel was dropped from 9.0 m, the force was generated about 17 times higher than the maximum force occurred in the axial impact test.

(a) Deformed shape after impact(Displacement scale factor: 5)



(b) Maximum impact force of lateral drop (c) Impact force of axial drop

Figure 5: Deformation and impact force of the drop simulation

4 EVALUATION OF STRUCTUAL INTEGRITY

In the previous study, the integrity of the fuel rod was evaluated by reflecting the deformation of simplified beam model to the rod model. In this paper, Table 2 shows the maximum stresses form fuel scale model acting on the guide tube and the fuel rod based on the yield stress of the material. In the case of 0.3 m drop, the stresses of the guide tubes and the fuel rods did not exceed the yield stress. However, when falling at 9.0 m, the maximum stress of the fuel rods and the guide tubes exceed the yield stress. It is possible to cause breakage or damage. Also, the guide tube showed that the maximum stress near the impact surface in the axial drop simulation exceeded the yield stress, and the fuel rod showed maximum stress in the section where the maximum moment occurred in the lateral drop simulation.

Drop height	Component	Lateral drop	Axial drop
0.3 m	Fuel rod	0.49	0.08
	Guide tube	0.06	0.04
9.0 m	Fuel rod	**3.20**	0.45
	Guide tube	0.52	**1.15**

Table 2: Ratio of the maximum stress to the yield stress

5 CONCLUSIONS

- Verification process was performed to describe unique mechanical characteristics of the nuclear fuel.
- The impact forces and stresses of each component were estimated under drop conditions.
- By analyzing the drop simulation of this model, it is possible to select the section where the maximum stress is expected to occur.
- A lateral drop can produce the greatest stress on the fuel rod, and maximum stress occurs at guide tubes in the axial drop.

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