#562: Study on the classification and optimization of non isolated small LOCA accidents of EPR nuclear power units under shutdown conditions

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Abstract: This paper focuses on the classification of non-isolated small break loss-of-coolant accidents (LOCA) for the EPR nuclear power unit at Taishan Nuclear Power Plant under shutdown conditions. Through the calculation of primary circuit pipe break frequencies and the use of event tree analysis to determine the frequency of stuck-open safety valve events in the pressurizer, the study determines the frequency of non-isolated small breaks under shutdown conditions. Probabilistic safety assessment methods are employed to determine the risk importance of non-isolated small breaks under shutdown conditions. Based on the event frequency and risk importance analysis results, non-isolated small LOCA accidents under operational states C, D, and E are categorized as design extension accidents. This analysis provides technical support for the modification of the Final Safety Analysis Report (FSAR) and operating technical specifications for the EPR nuclear power unit, thereby achieving the goal of simplifying requirements in the operating technical specifications, shortening major maintenance durations, and enhancing the capacity factor of the unit.

Keywords: EPR; FSAR; Initiating Event; LOCA

1. INTRODUCTION

According to the principles for determining the occurrence frequencies of design-basis accidents in the Final Safety Analysis Report (FSAR) for the EPR nuclear power unit, the determination of occurrence frequencies for some design- basis accidents in the current FSAR is relatively conservative. This leads to a large number of initiating events in Chapter 15 of the FSAR and numerous safety management requirements with complex logical relationships in the EPR unit's operating technical specifications. These factors limit the implementation conditions for online maintenance and fail to fully utilize the redundancy design advantages of the EPR nuclear power unit's four-train safety systems. Therefore, it is necessary to optimize the accident classification for the EPR unit.

In the FSAR for the EPR unit, the frequency of small break LOCA accidents under shutdown conditions is primarily based on the frequency of small breaks under power conditions, which is considered conservative. Existing research on LOCA frequencies mainly focuses on power conditions and all conditions, with limited detailed studies on initiating events under shutdown conditions. The frequency of such events is typically given based on the proportion of time spent in shutdown conditions without precise calculation of the initiating event frequency. This paper presents a precise analysis of the initiating event frequency and safety importance for non-isolated small break LOCA accidents under shutdown conditions. By optimizing the classification of non-isolated small break LOCA accidents under shutdown conditions, we aim to refine the operating technical specifications and enhance the operational performance of the power plant.

2. SCOPE AND CLASSIFICATION OF EVENTS FOR EPR NUCLEAR POWER UNIT

In 2016, the National Nuclear Safety Administration issued the "Regulations for the Design Safety of Nuclear Power Plants," which categorizes the operating states of a nuclear power plant into: normal operation, anticipated operational occurrences (i.e., events expected to occur during the lifetime of the nuclear power plant), design-based accidents, and design extension conditions.

Taishan Nuclear Power Plant, in accordance with regulations, categorizes accident conditions in the FSAR into design basis accidents (DBAs) and design extension conditions (DECs). Design basis accidents refer to hypothetical accidents that lead to accident conditions in a nuclear power plant, where the release of radioactive materials remains within acceptable limits. Design extension conditions refer to accident conditions not considered within the scope of DBAs; they include conditions that do not cause significant damage to the core and core melt (severe accident) conditions.

The EPR nuclear power unit categorizes design basis accidents into four classes based on the annual occurrence frequency. The criteria for dividing the four classes of DBC (Design Basis Accident) accidents are shown in Table 1.

Table 1: DBC accident division criteria of taishan nuclear power plant

Category	Describe	Frequency
DBC-1	Normal operation-related transient	
DBC-2	Expected running event	10-2/reactor year<f
DBC-3	Rare accident	10-4/reactor year <f<10-2/reactor year
DBC-4	Extreme accident	10-6/reactor year <f<10-4/reactor year

Note: f is the generation event/accident frequency

DEC accidents are classified based on the state of the core caused by the accident into DEC-A: design extension conditions without core melt/supra-design-basis accidents; and DEC-B: specifically severe accidents leading to core melt.

Events with a lower frequency than design basis accidents and not considered as initiating events for design basis accidents can be classified as DEC-A. Typically, if the event frequency is less than 10^{-6} per reactor-year, the event can be categorized as a DEC-A event.

3. EPR NUCLEAR POWER UNIT EVENTS CLASSIFICATION OPTIMIZATION PROCESS

Determining the assumed initiating events for a nuclear power plant is often a process that involves an iterative interplay between design and analysis, engineering judgment, and operational experience, without strict guidelines to follow. In the United States, a typical list of assumed initiating events was established in the 1970s, and subsequent plants have generally referred to this list with minor adjustments. For new reactor designs, there is a requirement to evaluate whetheradditional initiating events not included in the list need to be added.

This paper optimizes the accident classification for the EPR nuclear power unit through a comparative analysis of initiating event lists to identify targets for classification optimization. The importance of initiating events is determined through an initiating event importance analysis, and the accident category is determined by combining the initiating event frequency analysis. The specific process is illustrated in Figure 1.

3.1. Comparative analysis of initiating events

At present, the third generation PWR nuclear power units at home and abroad mainly include AP1000, EPR and Hualong No.1, etc. There are differences in the initiating event lists for these types of nuclear power units. The optimization of initiating event classification for the EPR nuclear power unit primarily involves analyzing the upstream design sources and conducting a horizontal comparison with typical PWR nuclear power unit initiating events to identify differences in the initiating event lists and analyzing initiating events that can be optimized.

3.2. Frequency analysis of initiating events

Based on industry practice and the classification requirements for initiating events in EPR nuclear power units, the frequency of initiating events is commonly used as a criterion for accident classification. This paper determines the frequency of initiating events through frequency analysis. Based on the results of the freq uency analysis and the EPR accident classification criteria, it is judged whether the classification of initiating events can be changed.

3.3. Importance analysis of initiating event

Probabilistic safety assessment technology is a method for evaluating and understanding risks in nuclear power plants. It is an important tool for managing and reducing risks and is also the best quantitative tool for assessing the state of nuclear safety. For the analysis of the risk importance of initiating events, PSA techniques are used to evaluate their risk importance.

An industrially significant initiating event refers to an event with a conditional core damage frequency (CCDF) of ≥1.0×10^{-6}, and a contribution of ≥1% to the core damage frequency (CDF) of the industry or nuclear power plant group. Using probabilistic safety assessment (PSA) techniques, the risk importance is determined through initiating event importance analysis. If the initiating event is not risk-significant, further classification can be conducted.

3.4. The expert panel

To make full use of the perspectives and experience of experts from various specialties, an EPR nuclear power unit initiating event optimization expert panel is formed, consisting of probabilistic safety analysts, deterministic safety analysts, and personnel from relevant departments of the power plant. The expert panel is responsible for reviewing the process of initiating event classification and making comprehensive decisions regarding the classification results of initiating events for the EPR nuclear power unit. The main content of expert decision-making includes:

- The PSA personnel shall review the technical sufficiency, analysis process integrity and analysis results of the frequency analysis and importance analysis of the initiating event;
- Deterministic safety analysts review the classification process and results of initiating events;
- Relevant professionals of the power plant shall review the impact of the analysis results on the power plant.

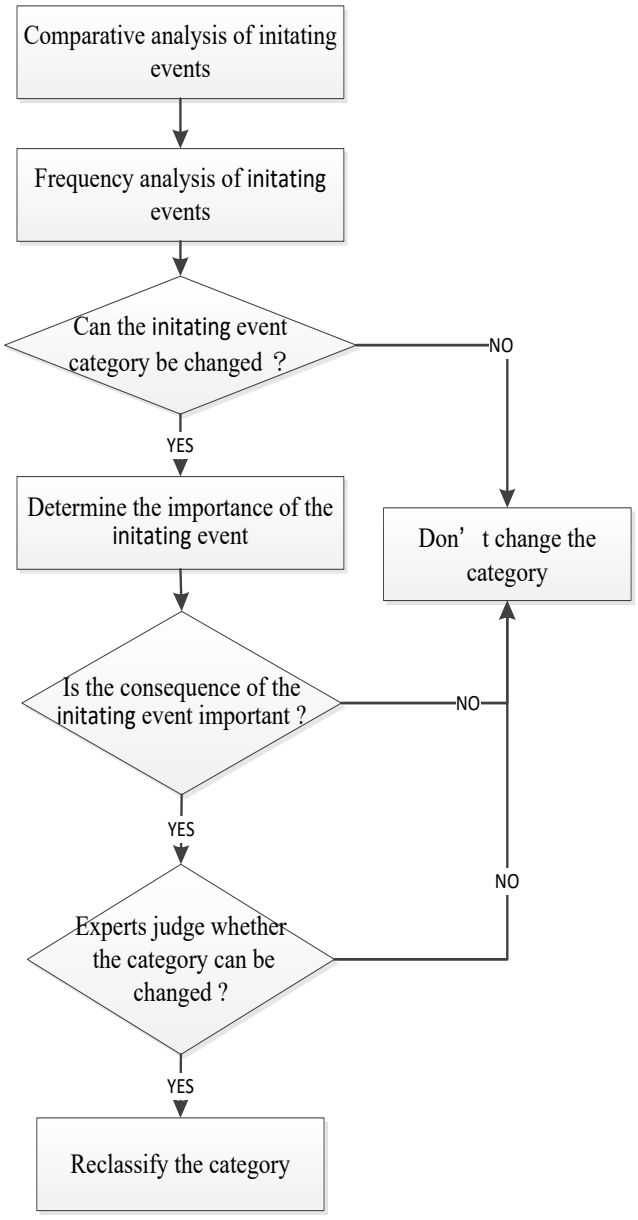

Figure 1: EPR nuclear power unit incident optimization process

4. CLASSIFICATION AND OPTIMIZATION OF NON-ISOLATED SMALL LOCA ACCIDENT UNDER SHUTDOWN CONDITION OF EPR NUCLEAR POWER UNIT

4.1. Comparative analysis of initiating events

By analyzing the upstream design sources of initiating events for the EPR nuclear power unit and conducting a horizontal comparison with initiating events for other nuclear power units such as the AP1000 and Hualong No.1, differences in the initiating event lists are identified, and initiating events that can be optimized are analyzed. Compared to other pressurized water reactor (PWR) nuclear power units such as Hualong No.1 and AP1000, the EPR nuclear power unit's FSAR considers additional initiating events for design basis accidents (DBAs), including the following accidents:

a. Small LOCA (non-isolated) under operational states C, D, and E, including breaks in the emergency boron injection system lines.
b. Breaks in the residual heat removal system lines inside (outside) the containment under operational states C, D, and E (isolatable).
c. Cooling-related accidents in the spent fuel pool.

The division of operational states C, D, and E under shutdown conditions for the EPR nuclear power unit is shown in Table 2.

Table 2: Status C, D, E division of EPR nuclear power unit

Status		Depaint
C	Ca	RIS/RRA low-pressure safety(2/4), the first circuit is closed, four steam generators are available, two main pumps are running (Column 2 and column 3), the temperature of the First Circuit (100-120℃); The pressure of the First Circuit is less than 30 bar
		RIS/RRA low pressure safety(4/4), the first circuit is closed, four steam generators are available, one main pump is running (Column 3) the first circuit temperature (55-100℃); The First Circuit pressure is less than 30 bar
		RIS/RRA low pressure safety(4/4), the first circuit is closed, two steam generators are available, one main pump is running (Column 3) the first circuit is single-phase, the temperature of the First Circuit is about 55℃; The pressure of the First Circuit is less than 30 bar
		RIS/RRA low pressure safety(4/4), the first circuit is closed, 2 steam generators are available, 0 main pumps
	Cb	RIS/RRA low pressure safety(3/4), the first circuit can be pressurized and the water level of the first circuit is more than 3/4 of the loop, two steam generators are available; The temperature of the First Circuit is 15 -55℃; The pressure of the First Circuit is less than 1 Bar
		RIS/RRA low pressure safety(3/4), the first circuit can be pressurized and not closed, the water level of the First Circuit is greater than or equal to 3/4 of the loop, two steam generators are available; The temperature of the First Circuit is 15-55℃; The pressure of the First Circuit = 1 Bar
D	D1	RIS/RRA low voltage safety(2/4), the first loop is open, the water level of the first loop is greater than or equal to 3/4 of the loop,Primary loop temperature 15-55℃; Primary loop pressure = 1 bar, before material change
	D2	RIS/RRA low voltage safety(2/4), the first loop is open, the water level of the first loop is greater than or equal to 3/4 of the loop, Primary loop temperature 15-55℃; Primary loop pressure = 1 bar, after material change
E		Cold stop reactor for material exchange; The reactor pool is full of water
F		Complete unloading of core

These accidents can be considered as objects for initiating event classification optimization for the EPR nuclear power unit. This paper primarily details the optimization process for non-isolated small LOCA initiating events under operational states C, D, and E as an example of initiating event optimization for the EPR nuclear power unit.

4.2. Do not isolate small loca under shutdown conditions

Under operational states C, D, and E for the EPR nuclear power unit, when the Reactor Injection System (RIS) and Residual Heat Removal System (RHRS) are connected in residual heat removal mode during shutdown conditions, a non-isolated small break loss-of-coolant accident (LOCA) in the Reactor Coolant System (RCP) or Emergency Boronization System will cause a decrease in RCP pressure. The equivalent diameter of the break is less than or equal to DN50 (equivalent area less than or equal to 20 cm²). This does not include leaks in the Reactor Coolant System that are compensated by the Chemical and Volume Control System (CVCS).

Non-isolated small LOCA includes breaks in the primary circuit piping up to the second valve of the connected system, and small breaks caused by stuck-open safety valves in the pressurizer that cannot return to the closed position.

4.3. Frequency analysis of non isolated small LOCA initiating events under shutdown conditions

(1) Source of initiating event

The frequency of small break accidents for the EPR nuclear power unit under operational states C, D, and E is based on NUREG-5750 data, combined with pipe dimensions and operating times through calculations. However, temperature and pressure conditions, which are key factors in pipe breaks, are not considered as variables affecting the frequency of accident occurrence.

In NUREG-5750, the median value for the initiating event frequency of small LOCA under power conditions is 5×10^{-4}/year, while the estimated initiating event frequency for small LOCA in the reference plant is less than 5×10^{-4}/year. The initiating event frequency for small LOCA accidents under full power conditions is 5×10^{-4}/year, and these pipelines operate for approximately 1% of each year. Therefore, the frequency of small LOCA under operational states C, D, and E for the EPR nuclear power unit is less than or equal to 5×10^{-6}/year.

In fact, compared to full power conditions, the pressure boundary of the primary circuit is significantly reduced in states C, D, and E, which would result in a substantially lower frequency of small LOCA occurrences.

(2) Analysis of primary pipeline rupture

In fatigue crack growth analysis, the Paris equation is the most widely used crack growth model, and its specific form is expressed as:

Equation 82: Fatigue crack growth.

$$da/dN = C \cdot \Delta K^{n}$$

Where:

- da/dN is the crack growth rate,
- a is the crack growth length, and N is the number of cycles;
- C and N are empirical constants determined by the material;
- ΔK is the range of stress intensity factors.

From the reference literature "A critical analysis of crack propagation laws," the parameters C and n for the Z3CN20.09M austenitic stainless-steel pipes used in the nuclear power unit were obtained. A finite element model of the primary circuit pipe containing a crack was established using the commercial finite element analysis software ANSYS. Stress intensity factors were separately analyzed for the pipe under conditions of 15.5 MPa and 310°C, and 3 MPa and 180°C, with the results shown in Figures 2 and 3. According to Equation (1), the crack growth rate under power conditions is approximately 250 times that under residual heat removal (RHR) connection conditions. Under otherwise similar conditions, the frequency of pipe breaks under shutdown conditions will be significantly lower than under power conditions.

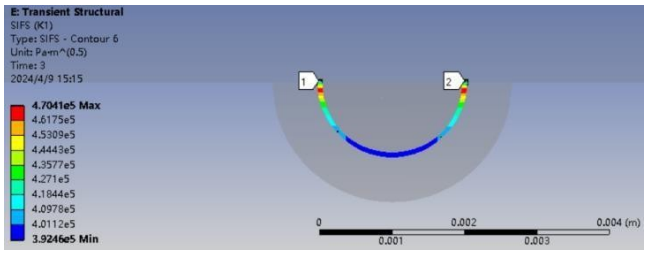

Figure 2: Stress intensity factor analysis of primary pipe stress intensity factor under power condition

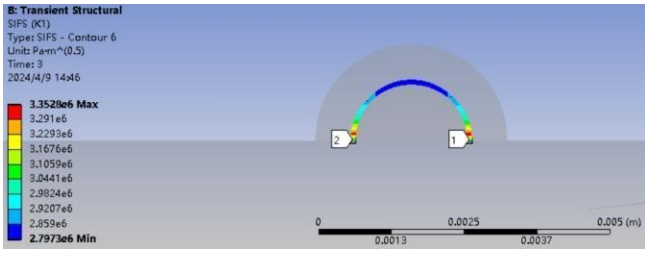

Figure 3: Analysis of stress intensity factor in primary pipe stress intensity factor under rhr connection condition

According to international generic data sources, the typical frequency value for small break accidents in power conditions for PWR is $5E\times10^{-04}$/reactor-year. Due to lower stress, the frequency of primary circuit pipe breaks under shutdown conditions is lower than under power conditions. In operational state Ca, the frequency of a LOCA accident in the primary circuit is 1/28 of that under power conditions according to the study. In operational states Cb, D, and E, the primary circuit is already at or near atmospheric pressure, and it is unlikely for LOCA accidents to occur in the primary circuit piping in these states; therefore, the frequency of small LOCA accidents in these operational states is not calculated.

Under the operation state Ca, the frequency of a non-isolated small LOCA in the primary circuit is calculated according to the reference document "Experimental study on fatigue crack growth characteristics of main pipe of domestic PWR nuclear power plant." The calculation process is shown in Equation (2):

Equation 2: Calculate the frequency of small non isolation LOCA.

$$F = \frac{1}{28} * F_a * \frac{t_{Ca}}{t_a}$$

Where:

- Fa is the frequency of LOCA of primary pipeline under power condition;
- TCa is the operation time of EPR unit under operation Ca;
- ta is the operation time of EPR unit under power condition.

According to Equation (2), the frequency of a small break in the primary circuit under operational state Ca is calculated to be less than 1.00E-06/reactor-year. For operational states Cb, D, and E, stress analysis indicates that the primary circuit is already at or near atmospheric pressure, and it is unlikely for LOCA accidents to occur in the primary circuit piping under these states.

(3) Pressurizer safety valve stuck-open

The EPR nuclear power unit is equipped with three depressurization lines at the top of the pressurizer, each fitted with a safety valve. When the Residual Heat Removal (RHR) system is not in operation, the pressurizer safety valves provide cold state over pressure protection for the primary circuit, preventing brittle failure of the pressure vessel shell before reaching the design pressure of the coolant system at low temperatures. When the primary circuit pressure exceeds the set-point of the pressurizer safety valve, the pressurizer safety valve opens. If a pressurizer safety valve sticks in the open position, a small LOCA event is initiated through the safety valve. Primary circuit coolant will flow through the depressurization line into the pressurizer depressurization tank. Indicators of temperature, pressure, and level in the depressurization tank will alert the operators to the stuck valve. If the operator closes the shutoff valve, the LOCA event will be terminated. Under operational states C, D, and E, after the RHR system is in operation, over pressure protection for the primary circuit is provided by the RHR safety valves. Therefore, only the LOCA event caused by a stuck -open pressurizer safety valve in state Ca needs to be considered. The frequency of LOCA due to a stuck-open pressurizer safety valve can be quantified using the event tree shown in Figure 4.

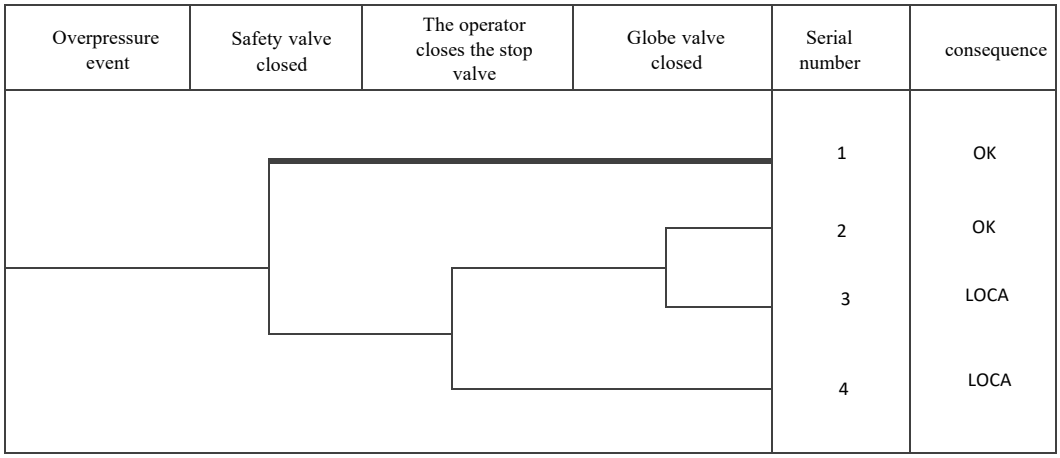

Figure 4: Calculation of opening frequency of stabilized safety valve

Through the analysis, the frequency of a LOCA event caused by a stuck-open pressurizer safety valve under operational state Ca is much less than 1×10^{-6}/reactor-year.

(4) Small LOCA frequency can not be isolated under operation conditions C, D and e

Based on the analyses in sections (2) and (3), the initiating frequency of non-isolated small LOCA accidents for the EPR nuclear power unit under operational state C is less than 1×10^{-6}/reactor-year. Under states D and E, the primary circuit is already at or near atmospheric pressure, and it is unlikely for LOCA accidents to occur in the primary circuit piping under these states.

4.4. Risk importance analysis of non isolation small loca under shutdown conditions

Through the probabilistic safety analysis model of Taishan Nuclear Power Plant, the risk importance of small LOCA under operational states C, D, and E is analyzed, with the results shown in Table 3. The total contribution of non-isolated small LOCA under shutdown conditions to the core damage frequency is 9.6×10^{-9}/year, accounting for approximately 0.5% of the internal event CDF and 0.3% of the total CDF. From the analysis results, it can be concluded that non-isolated small LOCA under operational states C, D, and E are not risk-significant initiating events.

Table 3: analysis results of importance of non-isolated LOCA in states C, D, and E

Incident	Depaint	Incident	CDF(1/reactor year)	CCDP
IE PBS1_Ca12	Status Ca12 small break	6.48E-06	7.0E-10	1.08E-04
IE PBS1_Ca 34	Status Ca34 small break	8.64E-06	1.1E-09	1.27E-04
IE PBS1_Cb	Status Cb small break	1.68E-05	1.2E-09	7.14E-05
IE PBS1_D	Status D small break	1.34E-05	6.4E-09	4.78E-04
IE PBS1_E	Status E small break	3.84E-05	1.2E-10	3.13E-06

4.5. The expert decision and reclassification

In the EPR nuclear power unit's FSAR, non-isolated small LOCA under operational states C, D, and E are classified as DBC-4 in the accident classification.

The EPR nuclear power unit initiating event optimization expert panel reviews the results of initiating event frequency and initiating event importance analysis, and comprehensively determines the initiating event classification based on deterministic analysis and power plant operational experience. According to the EPR nuclear power unit's accident classification standards, small LOCA under operational states C, D, and E can be reclassified as DEC-A accidents.

5. CONCLUSIONS AND RECOMMENDATIONS

Through initiating event frequency analysis and initiating event importance analysis, the initiating frequency of non-isolated small LOCA accidents under operational states C, D, and E for the EPR nuclear power unit is less than 1.00E -06/reactor-year, and these are non-risk-significant initiating events. According to the EPR nuclear power unit's accident classification standards, small LOCA (≤DN50) under operational states C, D, and E can be classified as DEC-A accidents. Based on the classification results, relevant professionals at the power plant can modify Chapter 15 of the FSAR and seek approvalfrom the nuclear safety regulatory authority. After obtaining approval, according to the principles of the operating technical specifications, the requirements for small LOCA (≤DN50) under operational states C, D, and E in the operating technical specifications can be removed, thus simplifying the related management requirements in the operating technical specifications, shortening the overhaul period, and improving the unit capacity factor.

This paper discusses the method of optimizing the classification of non-isolated small LOCA accidents under operational states C, D, and E for the EPR unit, which may not be applicable to the optimization of other accident classifica tions. However, this optimization method provides ideas for the optimization of other accidents. Other initiating events can be reviewed and identified according to this optimization process, and targeted engineering analysis and statistical analysis can be carried out to explore ways to achieve a balance between the design capacity factor and conservativeness in the EPR unit.

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