Status of the Kansai Electric Power Company's implementation to the Proactive Safety Assessment for Continuous Improvement and Recommendation on how it should be

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1. The status of the regulatory initiatives in the Safety Assessment Reporting System for Continuous Improvement

1.1 Overview of the Safety Assessment Reporting System

In order to ensure the safe and stable operation of nuclear power plants, it is necessary to work constantly on improving nuclear safety. Under the regulatory oversighting, the operational organization should take further improvement measures voluntarily and continuously, without staying within the regulatory framework. For this, it is important to continue efforts to widely collect and confirm the latest scientific and technical findings (hereinafter referred to as "the latest findings"), and to reflect it in the design and operation of nuclear power plants.

The accident at the Fukushima Daiichi Nuclear Power Plant has taught us that we, Japanese operators, are not effectively utilising the latest findings. The follow-ups on new technology and uncertain hazard information have been insufficient.

The June 2012 revision of the Japanese Nuclear Reactor Regulation Law (enacted in December 2013) established a system and processes of ‘Safety Assessment Reporting for Continuous Improvement’ (hereinafter simply referred to as ‘SAR’) aimed at requiring nuclear operators take voluntary safety improvement measures on a continuous basis. This assessment is carried out within six months of the end of the periodic fuel outage and is then submitted to the Nuclear Regulatory Authority (hereinafter referred to as 'NRA') without delay. In the SAR process, the safety of each reactor at the end of its periodic fuel outage is supposed to assessed, several improvement measures (‘additional measures’) are identified, and a plan for future safety improvements is developed, to implement the measures autonomously.

The International Atomic Energy Agency (hereinafter referred to as 'IAEA') Safety Standards SSG-48[1] identifies the process of becoming obsolete due to updates in relevant standards, codes, regulations and knowledge as "non-physical aging." This may affect the long-term safe and stable operation of nuclear power plants. Now in Japan, there is consideration of utilizing the SAR to address such obsolescence as part of efforts to ensure the long-term safety of nuclear power plant operation.

In this paper, we introduce an example of Kansai Electric Power Company ('KEPCO')'s efforts to utilize the SAR process and propose ways to improve the nuclear safety.

1.2 Contents and description of SAR

The contents and description of the SAR to be submitted is shown in Table 1.

Table 1: Contents and description of the SAR

Chapter 1 Measures for ensuring the required safety performance and demonstration of compliance with the requirements	1.1 Introduction and General Considerations	Frequency
	1.2 Site Characteristics	Every fuel outage
	1.3 Safety Objectives and Design Rules for Structures, Systems and Components	
	1.4 Operational Limits and Conditions, and Management for Safety	
	1.5 Safety Analysis to Demonstrate Compliance with the Requirements	
Chapter 2 Proactive measures for Safety Improvement	2.1 Policy on Continuous Efforts to Improve Safety	
	2.2.1 Status of Activities to achieve Operational Safety 2.2.2 Latest National and International Scientific and Technical Findings	
	2.3 Safety Improvement Plan	
	2.4 Description of Additional Measures	
	2.5 External Review	
Chapter 3 Investigation and Analysis of Proactive Measures for Safety Improvement	3.1.1 Assessment for Internal and External Events 3.1.2 Deterministic Safety Assessment	
	3.1.3 Probabilistic Risk Assessment for Internal and External Events 3.1.4 Assessment of Safety Margin (Stress Test)	Every 5 years or after major plant mod.
	3.2 Medium- and Long-Term Assessment of the Activities for Safety Improvement (Proactive Safety Review: “ PSR^+ ”)	Every 10 years or after major plant mod.
Chapter 4 Global Assessment	4.1 Global Assessment	Every fuel outage
	4.2 Safety Improvement Plan	

1.3 Actual examples of KEPCO's SAR.

As of November 2023, KEPCO had submitted SAR a total of 15 times for five reactors (Takahama Units 3 and 4, Ohi Units 3 and 4 and Mihama Unit 3).These plants have restarted after the accident in Fukushima, and now operational, in compliant with the new regulatory standards. With the restart of Takahama Units 1 and 2 in 2023, there will be a total of seven plants compliant with the new regulatory standards. SARs will be carried out up to seven times in about one year. The actual records and forecasts of SARs by KEPCO are shown in Fig. 1.

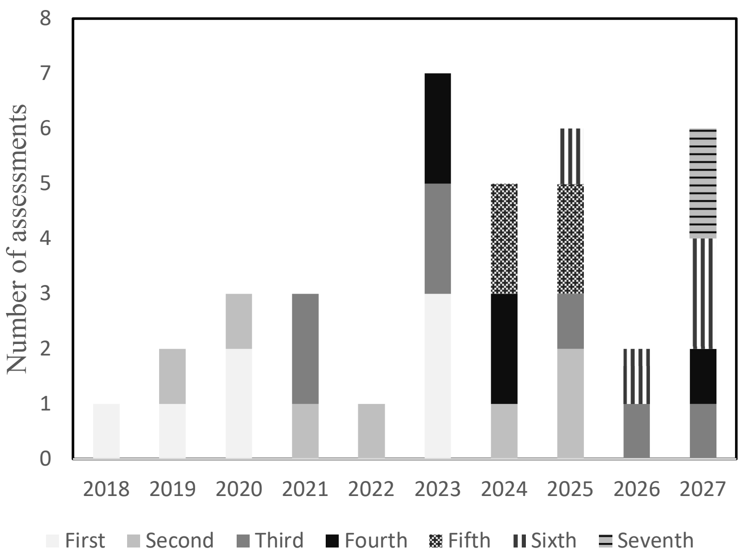

Figure 1. The track record and forecasts of KEPCO's SARs (including the periodic safety review carried out by the conventional rule for Takahama Units 1&2 in 2023)

Kyushu Electric Power Company and Shikoku Electric Power Company have also submitted similar reports, with 17 and 2 reports respectively. The total number of SARs for operational plants complying with the new regulatory standards is expected to be increased in the future in entire Japan.

In the SARs of KEPCO to date, many extracted additional measures have been taken to improve safety. Examples include measures such as (i) installation of the circuit to auto-activate the Emergency Core Cooling System ('ECCS') Recirculation Mode, (ii) Reactor Coolant Pump ('RCP') Shut-Down-Seal ('SDS'), (iii) installation of an Open Phase Condition detection system, (iv) establishment of the performance review conference, and (v) enhancement of Severe Accident training by Probabilistic Risk Assessment ('PRA') results, etc.

1.3.1 Installation of the circuit to auto-activate the ECCS Recirculation Mode

An overview of the improvement of the operation system of the ECCS Recirculation Mode is shown in Figure 2. This measure adds an additional circuit to activate automatically the ECCS Recirculation Mode in the event of a loss-of-reactor-coolant accident by signal of the tank water level.

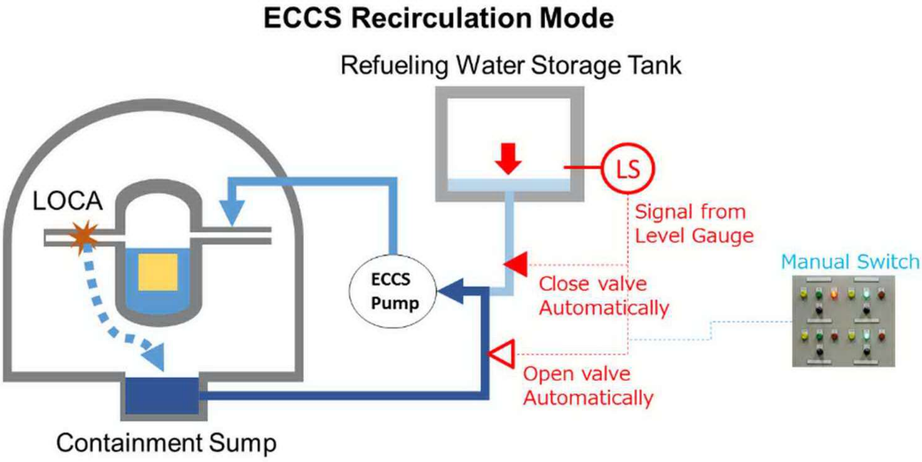

Figure 2. Summary of the ECCS Recirculation Mode

1.3.2 Installation of RCP-SDS

An overview of the RCP-SDS is shown in Figure 3. This measure is aimed at taking actions to protect against the small leakage of coolant accident from the RCP seal in the case of a loss of all AC power supply.

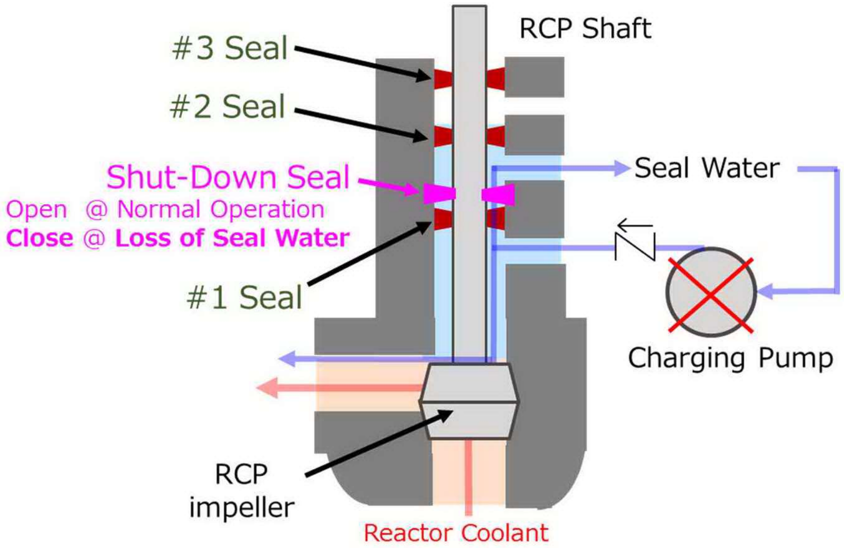

Figure 3. Summary of the RCP-SDS

These two measures above (1.3.1 and 1.3.2) were decided to be implemented, considering that the results of the PRA showed that they significantly improved the core damage frequency ('CDF'), and that they were judged to be effective even after considering several qualitative key elements referred to the Integrated Risk Informed Decision Making ('IRIDM') concept. Furthermore, these two measures were identified as plant designs that improved safety more when compared to domestic PWR plant designs. This is also positioned as a response to Obsolescence as referred to in IAEA SSG-48.

1.3.3 Installation of the Open Phase Condition detection system

The Open Phase Condition detection system has been installed on the station electric bus bar. This measure has been assessed for implementation based on new findings related to the U.S. Operating Experience. This measure is a typical example of new national and international findings being reflected in practice.

1.3.4 Establishment of a Performance Review Conference

The purpose of the conference is to facilitate power plant performance improvement activities. To bring together plant executives, comprehensive review is provided against plant performance based on a variety of indicators, to pursue excellence in operating nuclear power plants around the world. This measure is derived from the results of a self-assessment of the power plant's operational activities.

1.3.5 Enhancement of Severe Accident training by PRA

Enhancement of Severe Accident training refers to education and training using risk information (critical scenarios and operations of risk importance obtained from the PRA. It raises the awareness of operators and emergency safety personnel, and improves their ability to respond to plant transient and accidents.

Measures such as 1.3.4 and 1.3.5 are "software measures" that review procedures, knowledge, and organization, not plant modifications of structures, systems, and components ("SSCs"). It is rather desired that more such software measures will be extracted in the future during the operational phase of the plant.

In addition to these examples, KEPCO is voluntarily and continuously improving the safety of its nuclear power plants by evaluating various latest findings and selecting and implementing measures to reduce risks related to the nuclear safety.

1.4 The collection of latest findings

As mentioned in 1.1, although all operational organization in Japan have collected and evaluated the latest findings even before the accident at Fukushima, it was desirable to improve the extraction (screening) of information relating to new technologies and science, especially of external hazards. The NRA's 'Operational Guidelines on the SAR for Commercial Power Reactors' [2] provides the content of the assessment and the items to be included in the report. It requires the collection and description of the latest domestic and foreign scientific and technical findings, including the following. KEPCO's SAR covers all the information required by the NRA Guidelines, (1) through (5). Item (6) was added voluntarily.
(1) Research and other safety-related activities to further improve the safety of equipment essential for ensuring the safety of nuclear power plants.
(2) Lessons learned from the operating experience of nuclear power plants in Japan and abroad
(3) Data necessary for carrying out PRA
(4) Domestic and foreign standards, etc.
(5) Information from international organisations and domestic and foreign academic societies, etc.
(6) Manufacturers' proposals for improving equipment safety.
For the latest findings information obtained from (1) to (6), the following information is excluded (screen-outed) to select useful latest findings expected to lead to improved safety.

- It is a combination of previous findings and does not propose any new methods, etc.
- Already reflected.
- Need to keep an eye on future research trends.
- To apply in actual work, further research is needed, etc.

Details of the screening example are shown in Figure 4.

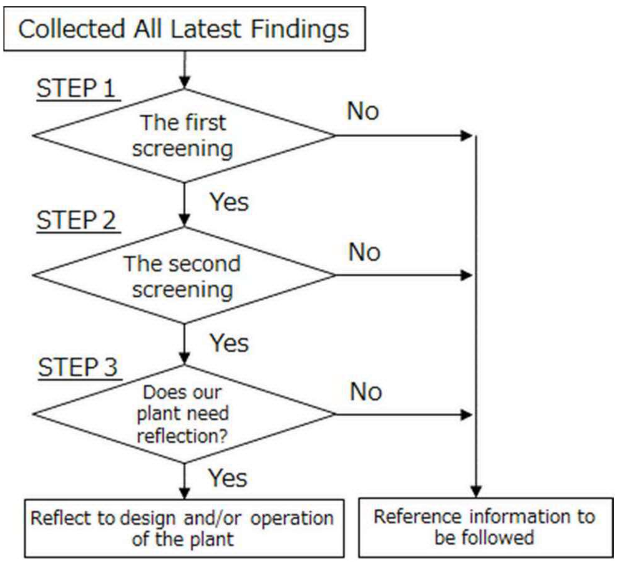

Figure 4. Example of the screening method of the Latest findings

The findings (1) through (6) presented above are collected, cases directly related to SSCs important to safety and operational management aspects of the power plant are extracted and the necessary improvement measures are examined in Chapter 2 and 3 (See table 1) of a SAR. As a result of the examination, plans for improvement measures are developed in Chapter 4 .

1.5 Other results and utilisation of SAR

In the chapter 3.1 (see table 1), the significant risk scenarios are analysed to select the risks that need to be reduced and the measures to reduce them. The PRA assesses the risk of core damage and containment failure for external events caused by earthquakes and tsunamis, in addition to the risk of internal events caused by plant equipment failure, etc. In addition, in the stress test the plant's capability is evaluated based on the overall safety margin (safety margin) of the plant as a whole in the case of an earthquake or tsunami exceeding assumptions. As a result of the stress test implemented until 2023, it was confirmed that the all 5 reactors of KEPCO have a sufficiently large capacity to withstand earthquakes, tsunamis, and the combination of earthquakes and tsunamis after the introduction of safety improvement measures based on the accident at the Fukushima Daiichi Nuclear Power Plant. (Takahama Unit 1 & 2 will be examined in the first SAR.)

In the chapter 3.2, the 14 safety factors based on the IAEA safety standard SSG-25[3] are compared to the current plant status with the latest national and international findings from a medium- to long-term perspective and find to be comparable. In this way, additional measures to be taken in the future can be extracted.

By the mean of SAR, various safety improvement efforts have been possible.

1.6 Current issues in the SAR System recognized by the operators

Although there have been various positive results, the following issues were experienced in the process of SAR as an operator. We believe that some chapters should be streamlined so that more resources can be invested, especially in reviewing the latest findings in Chapter 2 and in the Proactive Safety Review (PSR$^+$) activities in Chapter 3.2, such as enhancing the consideration on the obsolescence of codes, standards regulations and knowledge.

First, for Chapter 1 and first two sub-chapters of Chapter 3, we compile and prepare the permitted licensing applications and Technical Specifications on which the following assessments are based, but we feel that the amount of resource investment and the effect for safety improvement are not at all balanced compared to the other chapters. Second, with regard to the investigation on the activities for operational safety in Chapters 2, we feel that there is no longer a need to necessarily compile a summary of each inspection, considering the daily activities by operators and the introduction of the regulatory inspection system that was introduced in 2020.

The nuclear regulatory inspection system has provided the concept of risk-informed and performance-based regulation, and operators have implemented corrective action programmes and configuration management. Past studies on PSR have also indicated the need to conduct periodic safety reviews, taking into account the domestic situation and newly introduced inspection systems [4].

Therefore, based on these regulatory dynamism, in order to optimize the SAR system, the results of the effectiveness of assessment and issues identified for each part of the SAR are presented in the next chapter of this paper.

2. Investigations on the relationship between SAR System and other safety systems

2.1 Activities and documentations that are highly relevant to the SAR System

Table 2 shows the results of organizing the descriptions in the chapters and the activities and documents that are considered to be highly relevant to them.

As the results of organizing, for Chapter 1, there is some overlap with previously licensed or approved documents, and the initial goal of always sharing the current status (As-Is) of facilities with NRAs in consideration of emergency response is generally achieved in the inspections. In particular, design information on facilities, which is an important update, is shared with the NRA as a Design Basis Document,

along with changes in design requirements due to plant modifications, and configuration management is performed. Some parts, such as the site characteristics in Chapter 1.2, which have a time lag in updates, need to be included in the SAR on an ongoing basis.
In addition, regarding Investigations for Operational Safety activities in Chapter 2, it was confirmed that even if such a subchapter does not exist, continuous improvement activities are being carried out by operators, and the status of such activities can be monitored by NRAs. NRAs can also monitor the status of such activities.

Table 2: Systems that are highly relevant to the SAR System.

Contents and Description of SAR	Highly relevant Activities and/or Documentation	Remarks
1.1 Introduction and General Considerations	Equivalent to the Permitted Licensing Application	All are equivalent to previously permitted or approved documents except 1.2 and 1.5 (these parts should be considered to be updated in the transparent manner.) The regulator may access these all documents at any time in a Nuclear Regulatory Inspection System.
1.2 Site Characteristics	Equivalent to the Permitted Licensing Application (excluding portions that were changed after the permit was issued)	
1.3 Safety Objectives and Design Rules for Structures, Systems and Components	Equivalent to the Permitted Licensing Application and to the Design Basis Documents controlled in the Configuration Management by the Licensee	
1.4 Operational Limits and Conditions, and Management for Safety	Equivalent to the Approved Tech. Spec. of the Site	
1.5 Safety Analysis to Demonstrate Compliance with the Requirements	Equivalent to the Permitted Licensing Application (excluding portions that were changed after the permit was issued)	
2.1 Policy on Continuous Efforts to Improve Safety	–	(general statements)
2.2.1 Status of Activities for Operational Safety	Improvements are routinely planned, done and validated through the Corrective Action Programme and several Internal/External Oversights	Improvements are observed under the Nuclear Regulatory Inspection System.
2.2.2 Latest National and International Scientific and Technical Findings	The latest knowledge of the entire electric power industry will be collected and shared on a regular basis, and reflected on a daily basis by the operators.	Ensuring the transparency of this mechanism is an issue.
2.3 Safety Improvement Plan	–	(depending on the results of above assessments)
2.4 Description of Additional Measures	–	
2.5 External Review	–	
3.1.1 Assessment for Internal and External Events 3.1.2 Deterministic Safety Assessment	Equivalent to the Permitted Licensing Application and periodic annual assessments complied with the approved Tech. Spec. (excluding portions that were changed after the permit was issued)	Same as Chapter 1
3.1.3 Probabilistic Risk Assessment for Internal and External Events 3.1.4 Assessment of Safety Margin (Stress Test)	N/A (SAR-specific activities)	To be enhanced (assess every 5 or 10 years or after major plant mod.)
3.2 Medium- and Long-Term Assessment of the Activities for Safety Improvement		
4.1 Global Assessment	–	(depending on the results of above assessments)
4.2 Safety Improvement Plan	–	

2.2 Discussion on the desired role of the SAR System, based on the three regulatory systems (Regulatory Standards, Inspection System and SAR System) and recommendations for further improvement of the operator's safety activities

When designing the SAR System, it is assumed that the system is a joint system based on the FSAR (Final Safety Analysis Report) of the United States, and European PSR (Periodic Safety Review) systems as overseas knowledge. The U.S. FSAR allows the regulatory body to obtain license-based updates from operators annually, as UFSAR (Updated Final Safety Analysis Report). Although this role is taken by Chapter 1, it can also be accomplished by relevant document and inspection system, as organized in 2.1.

The IAEA specifies the content of the FSAR in IAEA SSG-61 [5], which is generally covered by Chapter 1. On the other hand, the IAEA defines the PSR framework in SSG-25. SSG-25 recommends an overarching assessment by 14 safety factors and an integrated examinations by Global Assessment, which is covered by Subchapter 3.2. This Chapter 3.2 should continue to be maintained with continued better content to strengthen the response to Obsolescence and other issues. Therefore, we believe that substituting Chapter 1 with the activities of regulators and operators and redirecting resources to strengthening Chapter 3 will contribute to making the SAR system more effective while maintaining its important role with reference to the original U.S. FSAR and European PSR.

The proactive assessment in Chapter 3.2 are based on the domestic standard [6], and the areas for improvement, which is the input for the Global Assessment of Chapter 3.2 (this Global Assessment is not equivalent with the Global Assessment of Chapter 4 which is done for entire SAR for a reactor), are not sufficiently collected because they are mostly addressed and eliminated by the assessment by Subsection 2.2.1, which is conducted at each outage during 10 years. On the other hand, Subsection 2.2.1 also overlaps with the constant or periodic self-assessment of operators. Although activities for operational safety are investigated and additional measures are extracted in small pieces, after only a short evaluation period for each periodic fuel outage, this is no different from the improvement activities carried out within the Quality Management System, and oversight process for each safety activity conducted by external organizations and regulatory body. From a certain evaluation period, such as 10 years, SSG-25 requires the identification of areas for improvement from a medium- to long-term perspective. Therefore, Subsection 2.2.1 should also be implemented at the same time as the assessment of Chapter 3.2 once every 10 years, to collect and discuss many areas for improvement, thereby enhancing the evaluation activities of Chapter 3. We have reviewed a number of backfitting requests from operators, including those to reflect lessons learned from the Fukushima accident, and found that all of them are related to any Safety Factors of SSG-25, and that they respond to any of obsolescence by reflecting latest findings. In order to respond obsolescence in subchapter 3.2, especially in the Global Assessment, it is necessary to secure as much time as possible for the assessment and allocate limited resources to it. As indicated in previous studies [4], an extended evaluation period and communication with NRA inspectors stationed at the power plant would also be helpful.

As an issue extracted for the subchapter 2.2.2, the mechanism to collect and share latest scientific and technological findings among nuclear industry should be open in a transparent manner. One of the lessons learned from the accident at Fukushima is considered undesirable to extend the period for reporting the status of reflecting the latest findings. Therefore, it is desirable to have a scheme to constantly update the status of reflecting latest findings and to share the status to regulatory body and relevant stakeholders such as a specific group in domestic academia. This scheme may be accomplished by any shared service linked with relevant players, and not be limited to the SAR System itself. Now the operators’ working group for the SAR aims to develop a method for sharing latest findings across all nuclear operators and is considering a mechanism for sharing results of the first and second screening. The benefits of this would improve transparency of each operator's decision on findings, and ensure that the public (including NRA) is constantly notified with latest findings. If such scheme can be established, it may be possible to make the frequency of the reporting of this subchapter 5 years, or to make it unnecessary to include the information in the SAR.

In addition, the SAR system needs to be considered from the perspective of communicating to society through public announcements. It is necessary to consider the degree of description in the report, and also the method of response not limited to the report.

4. Summary

As discussed in Chapter 3 of this paper, all of the contents of SAR assessed every outage can be replaced by existing or new activities. In order to voluntarily and continuously improve the safety of nuclear power plants, it is important to expand the scope of voluntary efforts by operators, and these additional proposals to make the SAR system a more effective system also contributes for it. First, the frequency of SAR reports should be improved and compiled every five years in accordance with the PRA and Stress Test period; second, the unit of assessment should be changed to the site (or twin-plant) from reactor. Based on the situation reported in the past, no significant differences have occurred in the evaluation results for power plants that are twin plants, and we would like to make effective improvements by allowing submissions on a site-by-site basis. We would like to propose the ideas described in this paper in the regulatory committees as well as conference bodies such as academic societies, so that SAR System can be implemented in a more effective system.

Bearing in mind that voluntary and continuous safety improvement is of the essence, we will not overconfidently continue our nuclear safety efforts and will continue our efforts to continuously identify and assess the risks of nuclear power generation and eliminate or reduce these risks.

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