Reactor Core Analysis at Low Flow Condition Using THALES Subchannel Code

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ABSTRACT

During the operation of PWR (Pressurized Water Reactor) nuclear power plant, the minimum DNBR (Departure from Nucleate Boiling Ratio) should be maintained within the Tech. Spec. LCO (Limiting Conditions for Operation) to satisfy the acceptance criteria during AOO (Anticipated Operational Occurrence) and event condition. For the DNBR analysis, the calculation of the flow and energy fields is required. Some events may experience the low flow condition if the LOOP (Loss of Offsite Power) is occurred. The reactor core analysis at low flow condition is classified into two ways. One is for the power generation of the reactor core and the other is for DNBR calculation of hot node. Generally, the flow analysis for core power generation is more difficult.

The representative event which experienced the low flow condition is SLB (Steam Line Break) event. During the SLB event with LOOP, the reactor core is tripped and all RCPs (Reactor Coolant Pump) are stopped. As a result, the flow rate and pressure in reactor core decrease and the peaking factor is relatively high compared to a nominal operating condition.

For the prediction of state in the PWR reactor core, the reactor core analysis at low flow condition is performed using the subchannel code THALES (Thermal Hydraulic AnaLyzer for Enhanced Simulation of core) developed by KEPCO Nuclear Fuel. Since the subchannel analysis code was developed to solve the reactor core flow field at the high axial flow rate and pressure condition, the convergence at the low flow condition such as SLB is very difficult. Therefore, most of subchannel analysis codes fail to obtain the converged flow field at very low flow and high power, which has resulted to insufficient research of 3 dimensional analysis of SLB event.

This paper will show why we have a trouble to obtain the converged result for the reactor core analysis at low flow condition. And we show the high distortion characteristics of the flow and energy fields at the low flow condition by using subchannel analysis code. Through the core flow analysis at low flow condition, it is verified that subchannel code THALES can predict reasonable results despite of the extreme condition and that the distortion of the flow and energy field is huge.

KEYWORDS
DNBR, THALES, low flow condition

1. INTRODUCTION

During the operation of PWR nuclear power plant, the minimum DNBR should be maintained higher than the design DNBR limit which is calculated from the flow and energy properties in reactor core. Therefore, it is important to predict the internal state of the reactor core during AOO and event condition accurately.

The representative event which experienced the low flow condition is SLB event. If SLB event occurs, operation of reactor core is tripped by the RPS (Reactor Protection System). RCPs are stopped if the LOOP is assumed. Even though the core is tripped by RPS, the steam is continuously released from the main steam line, then the energy is transferred from the primary system to secondary system. Coolant flow rate decreases due to the stopping of RCPs. Pressure and temperature also decrease significantly.

In order to predict the physical properties at low flow condition in the PWR reactor core, the SLB events are analyzed using subchannel code THALES (Thermal Hydraulic AnaLyzer for Enhanced Simulation of core) developed by KEPCO NF [1]. In the following section, a brief description of THALES code is provided for a general understanding of code's capability. The low flow analysis using THALES code is then described in section 2.2. After that, results of reactor core analysis at low flow condition are presented and discussed.

2. REACTOR CORE ANALYSIS AT LOW FLOW CONDITION USING THALES CODE

2.1. General Description of THALES Code

The fuel assembly is composed of many fuel rods and several support grids. The support grids have mixing vanes to enhance the mixing of the coolant. Since the mixing vanes have complex geometric shape, it is very difficult to predict the flow fields inside of the reactor core. Therefore, including these issues, the general computational fluid dynamics have the limitations to simulate the entire reactor core. However, THALES code simulates the reactor core flow based on averaged control volumes like subchannel and node-wide channel instead of many computation grids and DNBR calculation is focused on the hot assembly.

THALES calculates the 3-D flow field with allowing the cross flow between channels [1]. The calculation procedure of THALES is shown in the Fig. 1. At first, all parameters are initialized, then, the energy equation is solved. After solving momentum equation, the code checks if convergence criterion is satisfied. If it is not converged, the iterative calculation of energy and momentum equations is performed. During the iteration, the previous parameter values are updated and flow and energy fields converges.

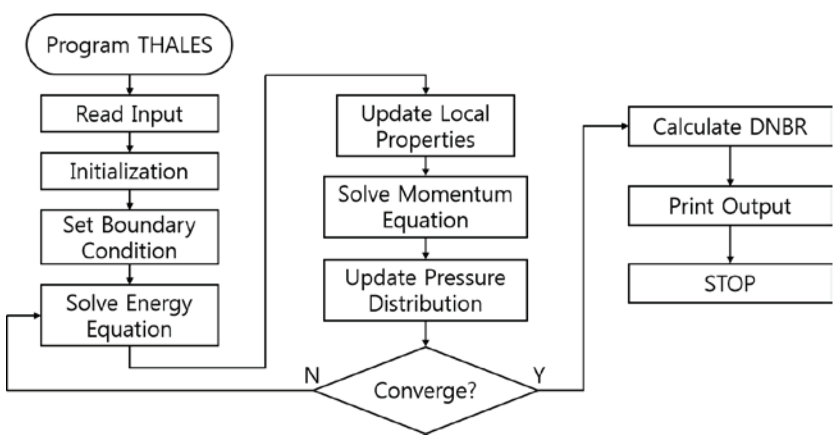

Figure 1. Calculation Procedure of THALES Code.

2.2 Simulated Condition and Core Analysis Model

The representative low flow conditions are the SLB event with LOOP and the biggest worth control rod stuck out. In this paper, the simulated low flow condition close to the SLB event with LOOP is used. The SLB with LOOP experiences low-flow rate, low-power and low-pressure conditions. The biggest worth control rod stuck out makes the high peaking factor in the core. The simulated conditions are summarized in the Table I.

Table I. Simulated Conditions for Low Flow Analysis

Core Power (MW)	225.2 (8%)
Pressure (psia)	1500.
Inlet Temp. (°F)	300.
Flow Rate (Mlbm/ft^2-hr)	0.158183 (6%)
Inlet Flow Distribution	Uniform
Outlet Pressure Distribution	Uniform

Radial power distribution of reactor core was calculated using the nuclear design code named ASTRA[2] developed by KEPCO NF. The node configuration to calculate the power distribution in the core is shown in Fig. 2. This power distribution is calculated based on the unit of node that means of 1/4 assembly. For the radial power distribution, ASTRA code needs the core enthalpy and density distribution, which is calculated from THALES code. Therefore, THALES code uses the same node configuration shown in Fig. 2. THALES code provides the enthalpy and density distributions to the ASTRA code.

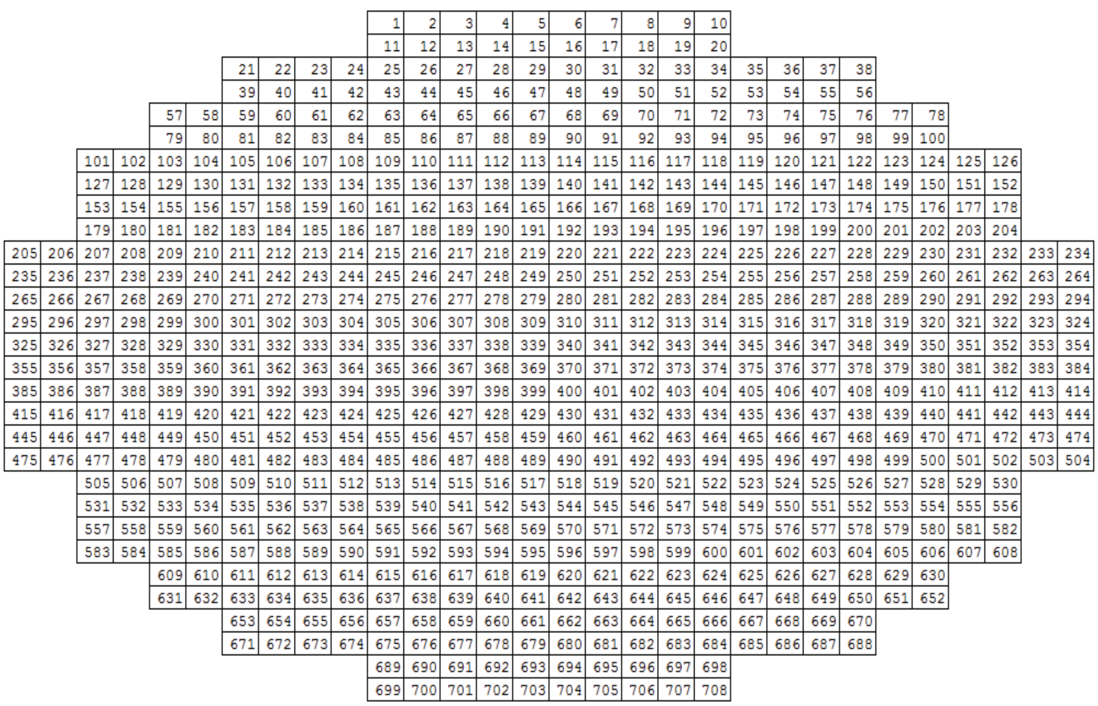

Figure 2. Node Configuration for Reactor Core Analysis

For the reactor core analysis at low flow condition, the highly distorted radial power distribution makes big trouble to simulate the core flow. Figure 3 shows the typical radial power distribution of reactor core. 4th quadrant shows the very highly distorted power distribution though the other quadrants show very low power distributions. Control rod stuck under SLB event makes very high radial peaking power. In the example power distribution of Fig. 3, maximum power is about 651 Btu/sec and the minimum power is about zero. The power deviation between quadrants is very big. This leads to the big density and enthalpy differences.

The other trouble of calculation related to the power distribution is axial power distribution. Usually, the DNBR analysis of hot node uses the model as shown in Fig. 4. In the calculation, the three dimensional power are composed of the radial power and axial power. Thus, at the same axial position, the node has the same axial power factor. But, the three dimensional calculation as shown in Fig. 2 uses different axial power distribution, which is shown in Fig. 5. The power change along the axial position is very steep. Therefore, we have more trouble to calculate the flow field based on the node unit for the DNBR analysis.

Generally, the computational fluid dynamics has a trouble for solving the problem with big changes of density in fluid. Since the analysis of SLB event is performed at high peaking factor and big density change, most of the subchannel analysis code fail to calculate the flow and energy field for the power distribution in Fig. 3. Since the original THALES code also diverges at the condition, we modified the THALES code for better convergence recently. The energy equation is solved with different iteration step compared to the momentum equation for better convergence since energy and momentum equations have different convergence speeds.

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										8	11	10	14	24	29	25	29	41	41										
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Figure 3. An Example of Radial Power Distribution

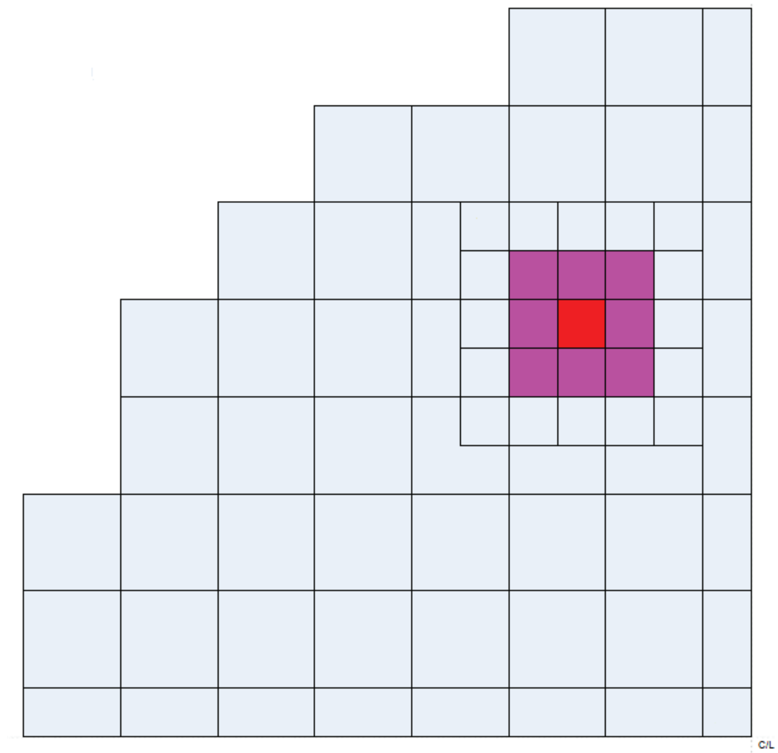

Figure 4. Core Analysis Model for Hot Node

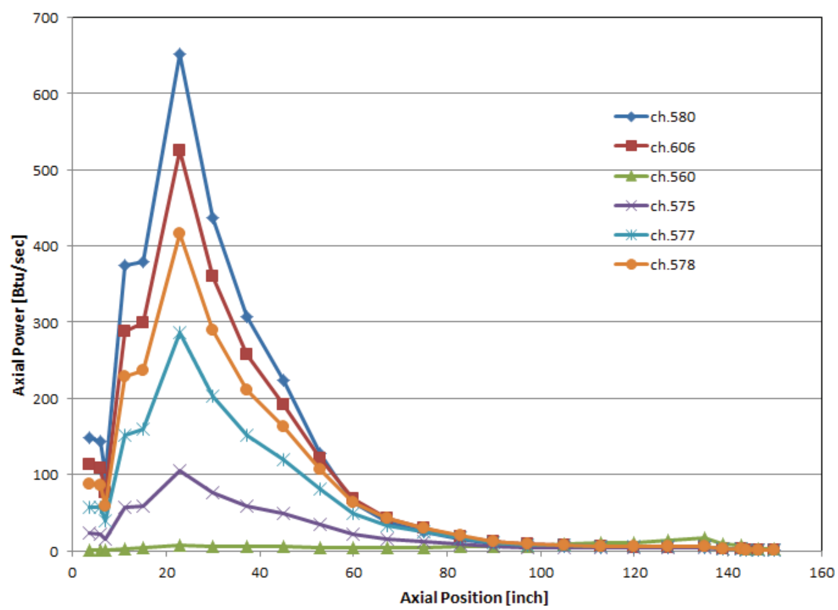

Figure 5. Examples of Axial Power Distribution

3. ANALYSIS RESULTS

Analysis with respect to the simulated condition was completed and the results are shown in this section. The big deviation of radial power distribution brings about the steep changes of density and enthalpy. Figure 6 shows the pressure change with axial position. Channel 1, 557 and 570 have low powers and channel 555 and 580 have high powers. Even if there is some deviation of pressure among channels, the difference is not so big.

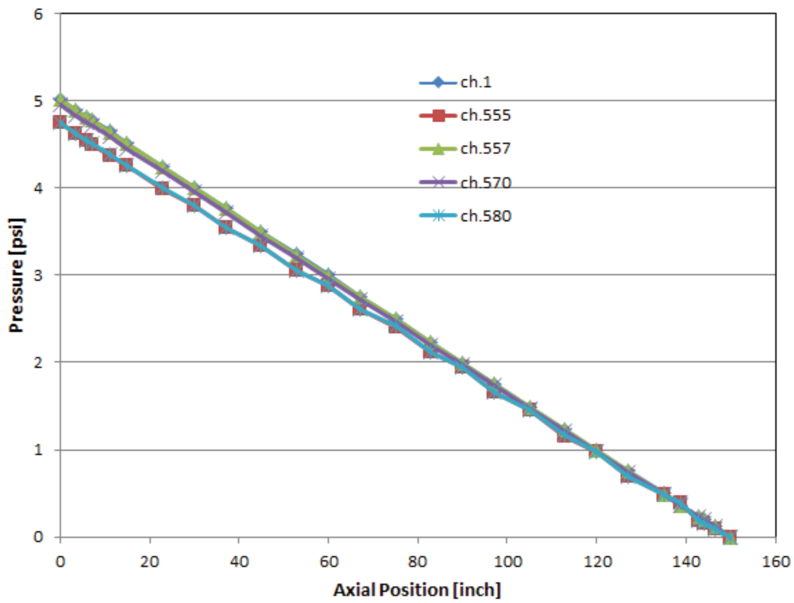

Figure 6. Pressure Distribution

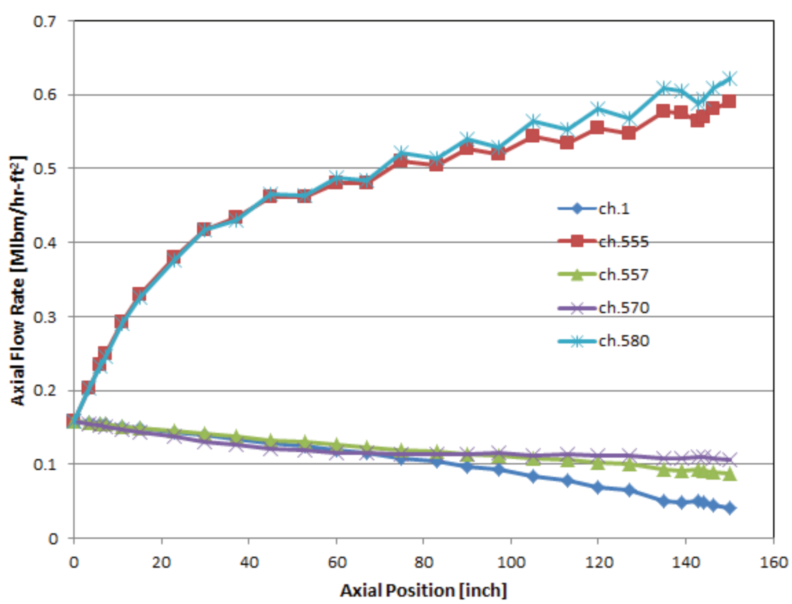

Figure 7. Axial Mass Flow Rate Distribution

Figure 7 shows the axial mass flow rate according to the axial position in the hottest channel. The initial mass flow rate is about 0.16 Mlbm/hr-ft^2. It increases so fast since there is big heat source close to the inlet. The maximum is about 0.6 Mlbm/hr-ft^2, which is about four times of inlet mass flow rate. Even if the pressure difference among channels is not big, the difference of axial flow rate among channels is big. This is caused by the high cross flow rate between channels. The steep change of mass flow rate causes the instability for analysis.

Figure 8 shows the enthalpy distribution according to the axial position. The high axial and radial powers make the big increase of enthalpy near the inlet, which makes density to be low. The big enthalpy difference among channels at the same axial position leads to a big density difference among channels.

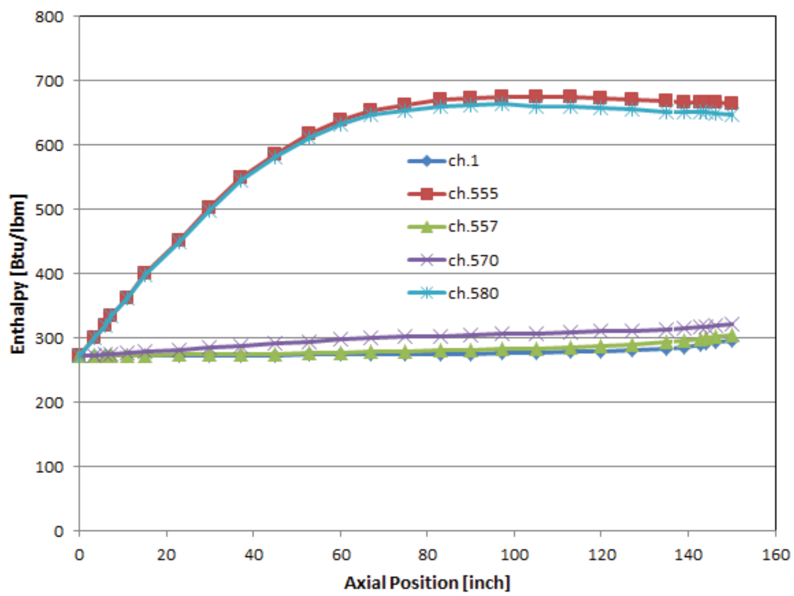

Figure 8. Enthalpy Distribution

Figure 9 shows the density distribution according to the axial position. Density decreases so fast by the high channel power. The initial density is about 58 lbm/ft^3. The exit density of high power nodes of channel 555 and 580 is about 20 lbm/ft^3. The change is so big, which makes the instability for the flow simulation. Low density makes high pressure and leads to the high velocity. This result is shown in Fig. 10.

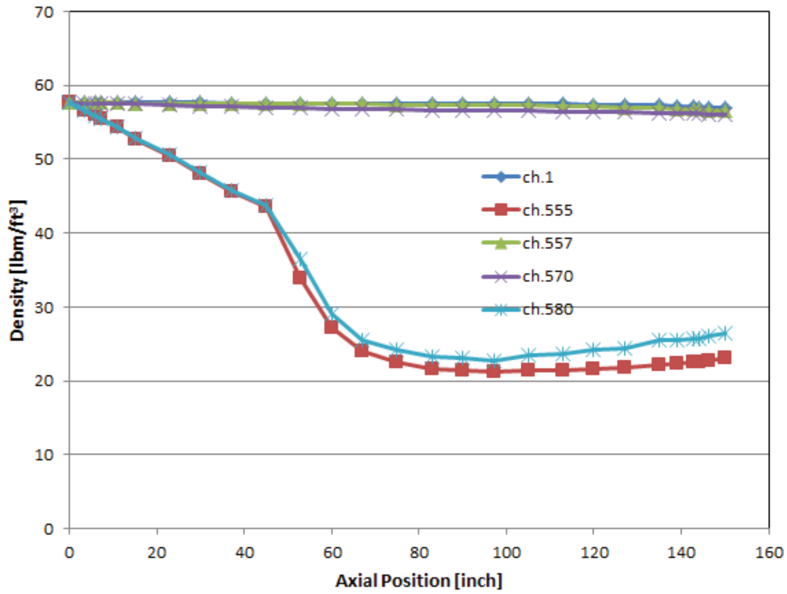

Figure 9. Density Distribution

Figure 10 shows the axial velocities of hot and cold channels. The axial velocity increases very fast near the inlet because the axial powers near the inlet increase steeply as shown in Figure 5.

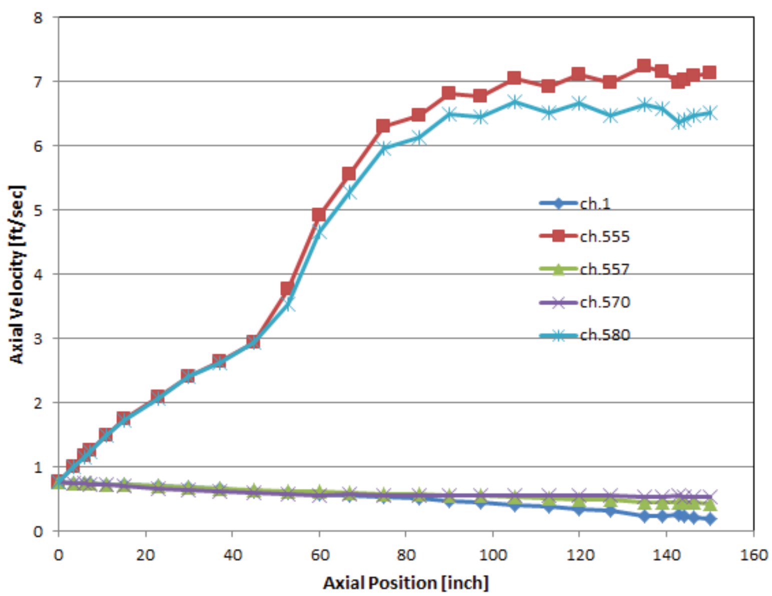

Figure 10. Axial velocity Distribution

If we use the extremely distorted radial and axial power distributions as shown in Fig. 3 and Fig. 5 respectively, the core flow analysis at low flow condition like SLB event has big trouble for the computational simulation. High power deviation makes big change of density and enthalpy, which results to the divergence of flow field solution. The big difference of flow among channels also influences to the solution of energy equations. This interaction between momentum and energy equations leads to the divergence of reactor core analysis at low flow condition. In order to improve the divergence problem, the different iteration steps between enthalpy and momentum equations was used to the modified THALES code.

4. SUMMARY AND CONCLUSIONS

Three dimensional analysis was performed using THALES code with the condition similar to the SLB event condition with LOOP. Radial power distribution was calculated using the nuclear design code ASTRA, which uses the THALES result for the enthalpy and density distributions. The deviation of radial power among nodes makes big trouble to the computational simulation. The axial power change is also steep. This brings on the big change of enthalpy and density. For more stable computation, the improved algorithm is required to the subchannel code. THALES code used the different iteration steps between enthalpy and momentum equations. The more stable treatment method will be presented in future.

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