

## Analysis of Loss of Ultimate Heat Sink considering Cliff-Edge Effect in APR1000

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### 1. Introduction

The Loss of Ultimate Heat Sink (LOUHS) is a representative Design Extension Condition (DEC) without significant fuel degradation event in the Pressurized Water Reactors (PWRs), in which the capability to access the normal heat sink is lost during normal operation. In this evaluation, the LOUHS is assumed as a simultaneous loss of the Essential Service Water System (ESWS) and the Circulating Water System (CWS), reflecting the actual configuration of heat removal paths on the primary and secondary sides.

International nuclear safety standards, including IAEA SSG-2 [1] and WENRA [2], emphasize the necessity of identifying “cliff-edge effects,” that is, instances of severely abnormal conditions caused by an abrupt transition from one status of a facility to another following a small deviation in a parameter or variation in an input value.

Conventionally, in Accident Management Program (AMP), the LOUHS event has been evaluated using a realistic approach with nominal initial and boundary conditions. In this evaluation, however, a sensitivity analysis is conducted using the combined approach, which applies conservative initial and boundary conditions to a Best Estimate (BE) computer code, in order to explicitly identify cliff-edge effects on primary overpressurization. Then, the results of the combined approach are compared with those of the realistic approach to demonstrate that the plant maintains an adequate safety margin in terms of overpressure, which is the main safety concern in LOUHS.

### 2. Analysis Methodology for Cliff-Edge Effect

The LOUHS event, which causes a decrease in heat removal by the secondary system, has been evaluated using a realistic approach. Nominal values are assumed for initial and boundary conditions without the quantification of uncertainties.

This study aims to evaluate LOUHS by conducting a sensitivity analysis with a combined approach that comprehensively addresses adequate margins to safety against cliff-edge effects through the additional consideration of uncertainties.

The combined approach utilizes a BE computer code in conjunction with conservative assumptions of plant state, as well as conservative initial and boundary conditions as shown in Table 1. The combined approach

is adopted in this study because sufficient criteria and basis data for quantifying uncertainties are currently scarce. Consequently, the combined approach provides a robust alternative by ensuring safety margins.

Table 1. Options for Performing Deterministic Safety Analysis [1]

Option	Computer code	Assumptions	Initial and boundary conditions
Conservative	Conservative	Conservative	Conservative
Combined	BE	Conservative	Conservative
Best estimate plus uncertainty	BE	Conservative	BE Partly most unfavorable conditions
Realistic	BE	BE	BE

### 3. Sensitivity Analysis for Cliff-Edge Effect

In the absence of quantification of comprehensive uncertainty analysis, the sensitivity analysis is performed to identify the limiting initial conditions that maximize the Reactor Coolant System (RCS) pressure. To ensure a fundamentally conservative evaluation of the margin in terms of overpressure, a base case is defined with conservative assumptions.

The base case assumes maximum core power of 102 %, maximum pressurizer water level of 60 %, minimum steam generator water level of 35 % and maximum core mass flow rate of 116 % in terms of overpressure for limiting conditions. These values represent qualitatively conservative initial conditions that impose a more demanding thermal-hydraulic response on the primary system. Based on this conservative base case, the sensitivity analysis is conducted with key parameters: Steam Generator Tube Plugging (SG TP), core inlet temperature and RCS pressure. The results of the sensitivity study for key parameters in terms of RCS peak pressure are shown in Table 2.

The sensitivity study is conducted in three steps to identify the most limiting set of initial conditions for the LOUHS analysis.

In the first step, The SG TP is selected as the sensitivity parameter, while all other initial conditions are maintained at the base case values. 10 %TP condition is selected as the limiting case for sensitivity analysis. This is because 10 %TP case reduces the effective heat

transfer area of the steam generator, which limits the rate of heat removal from the primary to the secondary side and consequently results in a higher RCS peak pressure.

In the second step, the 10 %TP condition identified in the first step is applied to the base cases. Then, a sensitivity analysis of the core inlet temperature is performed. The results indicated that the minimum core inlet temperature demonstrates the limiting RCS peak pressure. This is because lower coolant temperatures increase the coolant density, resulting in a higher primary mass inventory, which in turn results in a higher RCS peak pressure.

In the third step, with the 10 %TP and the minimum core inlet temperature condition fixed as limiting parameters, a sensitivity analysis on the key parameter for RCS pressure is carried out. The results demonstrate that the nominal RCS pressure shows the most conservative RCS peak pressure among the sensitivity cases considered.

Table 2. Result of the Sensitivity Analysis

Step	Key Parameter	Value	Result of RCS Pressure
1 <sup>st</sup>	SG TP	0 %TP	18.198 MPa (2,640 psia)
		10 %TP	18.243 MPa (2,646 psia)
2 <sup>nd</sup>	Core Inlet Temperature	292.22 °C (558.0 °F)	18.299 MPa (2,654 psia)
		295.83 °C (564.5 °F)	18.243 MPa (2,646 psia)
		300.0 °C (572.0 °F)	18.144 MPa (2,632 psia)
3 <sup>rd</sup>	RCS pressure	14.686 MPa (2,130 psia)	18.286 MPa (2,652 psia)
		15.513 MPa (2,250 psia)	18.299 MPa (2,654 psia)
		16.030 MPa (2,325 psia)	18.251 MPa (2,647 psia)

Based on the three-step sensitivity study described above, the combination of 10 %TP, minimum core inlet temperature (292.22 °C, 558 °F), and nominal RCS pressure (15.513 MPa, 2,250 psia) is identified as the limiting case.

For additional conservative assumptions in boundary conditions, the Reactor Protection System (RPS) High Pressurizer Pressure (HPP) setpoint is adjusted to its maximum value, and the Passive Auxiliary Feedwater Actuation Signal (PAFAS) setpoint is also determined as the minimum value.

#### 4. Analysis Results Using the Limiting Case

The limiting case, derived from the sensitivity analysis described in the Section 3, is used to evaluate the cliff-edge effect with respect to the RCS overpressurization. The result of the limiting case is compared with the result from the conventional realistic approach to confirm that the plant maintains sufficient safety margin to overpressurization even under the conservative assumptions.

The major sequence of events is summarized in Table 3. As shown in Table 3, the RCS peak pressure of 18.30 MPa (2,654 psia) occurs at approximately 6.2 seconds, driven by the rapid pressurization resulting from the loss of secondary heat removal. The Passive Auxiliary Feedwater System (PAFS) is automatically actuated to remove decay heat through the secondary system. Subsequently, an operator action is assumed to open Reactor Coolant Gas Vent System (RCGVS) valves for further depressurization. Through the automatic action of PAFS and appropriate operator action, the primary system successfully reaches the entry condition for the shutdown cooling system. As a result, the core water level is adequately maintained, ensuring that fuel integrity is maintained without any significant fuel degradation throughout the event.

Table 3. Sequence of Event for LOUHS Considering Cliff-Edge Effect

Time (s)	Event
0.0	LOUHS occurs (Main feedwater pump trip and turbine trip)
5.4	Reactor trip (HPP)
5.7	Main steam safety valve open
5.8	Pilot operated safety relief valve open
6.2	RCS peak pressure occurrence (18.30 MPa, 2,654 psia)
31.7	PAFS actuation

Figure 1 compares RCS pressure transients between the realistic BE approach and limiting case from the sensitivity analysis. As shown in Figure 1, the sensitivity analysis exhibits a rapid pressure rise and a higher RCS peak pressure compared to the realistic BE approach.

For DEC without significant fuel degradation, the acceptance criteria for reactor coolant pressure boundary integrity are defined as 125 % of the design pressure, considering the low probability of the event occurrence. The analysis results show that the RCS peak pressure reaches 18.30 MPa (2,654 psia), demonstrating that the plant maintains a substantial safety margin against overpressure failure, even under the conservative assumptions of the sensitivity analysis case. Consequently, despite the more severe pressurization observed in the sensitivity analysis, the RCS peak pressure remains within the acceptance criteria. Specifically, the result demonstrates a sufficient safety

margin against the overpressurization limit.

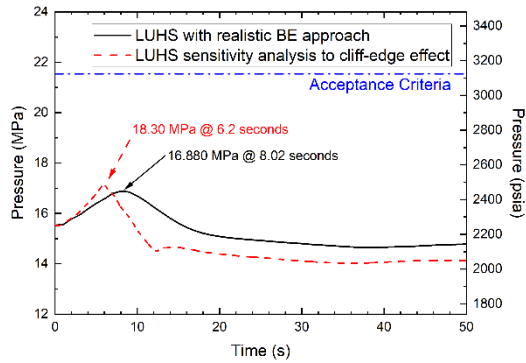


Figure 1. Comparison of RCS Peak Pressures Considering Cliff-Edge Effect for LOUHS

## 5. Conclusions

In accordance with international nuclear safety standards, the evaluation of cliff-edge effects is essential to ensure that small deviations in plant parameters do not cause abrupt and severe consequences when using the best estimate analysis without quantification of uncertainties. To fulfill this safety objective, this study evaluated the potential cliff-edge effects during a LOUHS event in the APR1000. By applying a combined approach, a limiting case is derived through sensitivity analysis to maximize RCS pressure under conservative assumptions within the range of limiting conditions for operation. The results confirmed that the RCS peak pressure remained well below 125 % of the design pressure, and the plant maintained adequate safety margin against overpressure, avoiding the cliff-edge effects. Also, by automatic actuations of safety systems and appropriate operator actions, the plant can be maintained in a safe state.

In the future safety evaluations, the application of this sensitivity analysis using the combined approach can be improved to support the safety analysis for DEC-A. Specifically, the cliff-edge effects for DEC-A will be evaluated to further assure the overall safety robustness of the nuclear power plant.

## ACKNOWLEDGEMENTS

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## REFERENCES

- [1] "Deterministic Safety Analysis for Nuclear Power Plant," Specific Safety Guide No. SSG-2, Rev.1, 2019, IAEA.
- [2] "WENRA Safety Reference Levels for Existing Reactors 2020," 2021, WENRA.