Development of a Testing Loop System for Simulating CRUD Behavior Under Flexible Reactor Operation Condition

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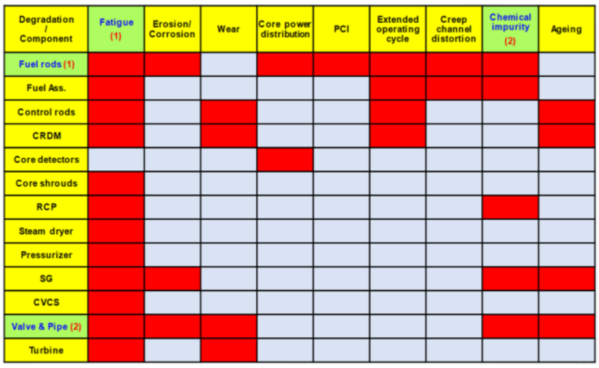
1. Introduction

As global energy policies evolve toward carbon neutrality, the integration of conventional and renewable energy sources has become a strategic priority. While renewable energy sources offer clean alternatives, they inherently suffer from intermittency, which is vulnerable to environmental variability, which poses challenges to grid stability and reliability. To compensate for these fluctuations, conventional energy sources are now expected to operate with increased flexibility, adjusting their output dynamically in response to real-time demand.

Nuclear power, as a low-carbon and reliable baseload energy source, is gathering renewed attention in this context. One of the key innovations enabling its role in flexible energy systems is “flexible operation” of nuclear reactors. This includes two primary modes: frequency control, which responds to real-time grid imbalances, and load-following, which adjusts output based on pre-planned power demands. Several European countries have already adopted such operational schemes in their nuclear fields, underscoring the need for Korea to develop and validate corresponding technologies.



Degradation / Component	Fatigue (1)	Erosion/ Corrosion	Wear	Core power distribution	PCI	Extended operating cycle	Creep channel distortion	Chemical impurity (2)	Ageing
Fuel rods (1)	🟥	🟥		🟥	🟥	🟥	🟥	🟥	
Fuel Ass.	🟥					🟥	🟥	🟥	
Control rods	🟥		🟥			🟥			🟥
CRDM	🟥		🟥			🟥			🟥
Core detectors				🟥					
Core shrouds	🟥								
RCP	🟥							🟥	
Steam dryer	🟥								
Pressurizer	🟥								
SG	🟥	🟥						🟥	🟥
CVCS	🟥								
Valve & Pipe (2)	🟥	🟥	🟥					🟥	🟥
Turbine	🟥		🟥						

Fig. 1. Key degradation mechanisms affecting reactor components under flexible operation condition

According to the International Atomic Energy Agency (IAEA) report [1], flexible operation must be accompanied by rigorous evaluations of structural integrity across reactor systems (Fig. 1). Of particular importance is the core management domain, where the behavior of CRUD (Chalk River Unidentified Deposits) on fuel cladding surfaces must be characterized. CRUD, formed by the deposition of corrosion products within the reactor core, can induce phenomena such as CRUD-Induced Localized Corrosion (CILC) and CRUD-Induced Power Shift (CIPS), which can significantly impact reactor safety and thermal performance [2,3].

In this study, we aim to investigate how CRUD formation characteristics change on fuel cladding surfaces during daily load-following operations. To achieve this, we are developing a high- temperature & pressure experimental loop designed to simulate flexible reactor operation and replicate the thermal and hydrodynamic conditions. This paper presents the conceptual design of this loop and shares preliminary results from early-stage testing to validate its feasibility.

2. Methods and Results

This section outlines the potential impact of flexible operation on CRUD formation, and introduces the design approach of a test loop developed to simulate relevant conditions. Preliminary results are presented to demonstrate the loop’s feasibility for evaluating CRUD behavior under realistic operating scenarios.

2.1 Effect of “Flexible Operation” on CRUD

Adopting flexible operation in nuclear power plants introduces several factors that may influence CRUD formation, including variations in thermal output, changes in water chemistry (such as pH and boron concentration), and shifts in hydrodynamic conditions like flow rate and pressure. Among these, changes in pH during power maneuvering are typically minor and are not expected to significantly affect CRUD solubility. In contrast, boron concentration can fluctuate considerably during power reduction and recovery, as it is directly used to control reactivity [3]. While these fluctuations may not directly influence CRUD formation, they could lead to boron hide-out within the CRUD layer, potentially contributing to phenomena such as CILC and CIPS, warranting close monitoring [4]. Flow rate and pressure may also vary slightly during flexible operation, but their impact on CRUD behavior is generally limited. The most critical factor, however, is the change in thermal output [5]. In Korea, current plans anticipate reducing reactor power down to 50% of nominal capacity, with a ramp rate of approximately 0.5 %/min for both power reduction and recovery.

As power decreases, fuel and cladding surface temperatures drop accordingly. This thermal change affects nucleate boiling behavior on the cladding surface. Repeated on/off cycles of subcooled nucleate boiling, along with localized boiling zones, may lead to significant changes in the axial distribution of CRUD. Moreover, a reduction in the temperature difference (ΔT) between the cladding surface and the bulk coolant can suppress boiling altogether. This enables rewetting of porous CRUD by coolant, weakening particle cohesion and adhesion, and potentially leading to partial CRUD dissolution. Even though coolant flow remains relatively constant during flexible operation, CRUD layers that have weakened cohesion are more susceptible to shear forces induced by core turbulence. As a result, delamination and redistribution of CRUD may occur. During power ramp-up, re-intensified local boiling could cause floating CRUD particles to redeposit onto the cladding surface. These repeated CRUD mobilization and redeposition cycles may produce a more heterogeneous axial CRUD profile, with varying density, featuring compact layers near the base and loosely bound particulate structures at the top.

Other operational parameters, such as ramp rate, frequency of power cycling, and the extent of power reduction, can further influence CRUD behavior. Considering these effects, we are developing an experimental loop focused on simulating thermal output variation as a key parameter for evaluating CRUD formation under flexible operation conditions.

2.2 Testing Loop Development

The loop system is primarily composed of four key sections: a coolant supply unit, a pressurization unit, a high-temperature and high-pressure test section, and a metal ion source injection unit. The base circulation flow rate is maintained at 18 L/hr, with the capability to increase up to 55 L/hr. Dissolved oxygen (DO) levels are kept below 5 ppb, while dissolved hydrogen (DH) is controlled within the range of 25–50 cc/kg-H_2O, simulating primary coolant chemistry under PWR conditions. The test section autoclave is designed to accommodate Zr-alloy cladding tubes (outer diameter: 9.5 mm, wall thickness: 0.6 mm), inserted into a sample holder with a 15 mm inner diameter. The test section is engineered to sustain operational conditions up to 360°C and 200 bar. To simulate nuclear fuel heat output, an internal cartridge heater is embedded within the cladding tube, with a maximum power output of 5 kW. To accelerate CRUD formation, iron (Fe) and nickel (Ni) ions are injected through a dedicated source injection system, enabling controlled chemical dosing into the circulating coolant.

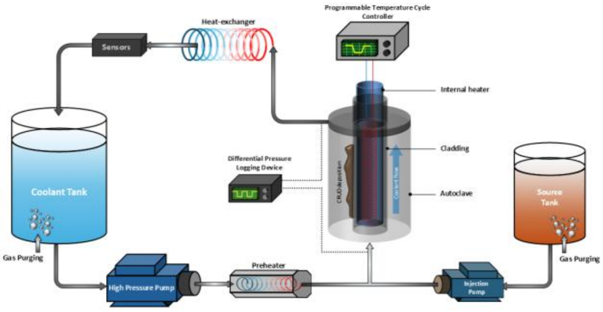


Fig. 2. Simplified schematic of CRUD experimental loop system simulating flexible operation

A distinctive feature of this loop lies in its ability to simulate thermal cycling and CRUD detachment, reflecting realistic load-following operation. The internal heater's output is controlled via a programmable system that applies cyclic temperature profiles. This setup facilitates periodic nucleate boiling on the cladding surface by turning boiling "on" and "off" according to the set thermal cycle. To ensure that the coolant's cooling response time is sufficient relative to the heating dynamics, a series of preliminary tests were conducted and will be discussed separately. Given the practical challenge of fully replicating the core flow velocity found in nuclear reactors, the loop employs a strategy of high-flow pulsing to simulate CRUD detachment events. Specifically, short-duration (3–5 min) high-velocity flow pulses are introduced periodically during power ramping cycles. These pulses apply strong shear stress impacts to the CRUD layer, promoting mechanical detachment, particularly in weakened deposits formed during power-down phases. Additionally, a differential pressure monitoring system is implemented across the test section. By logging real-time pressure differences, this system enables continuous tracking of pressure drops associated with CRUD growth and subsequent detachment, offering a non-intrusive means of observing CRUD evolution.

2.3 Preliminary Results

To evaluate the effectiveness of the proposed loop system, it is essential to verify whether the coolant temperature responds appropriately to changes in thermal output, i.e., internal heater power, in a way that reflects realistic load-following operation in nuclear reactors. A preliminary test was conducted using a similar loop system under a flow rate of 18 L/hr, in which the internal heater output was manually controlled to simulate power ramp-down scenarios. The objective was to assess whether the coolant temperature could decrease at a sufficiently realistic rate in response to heater power reduction.

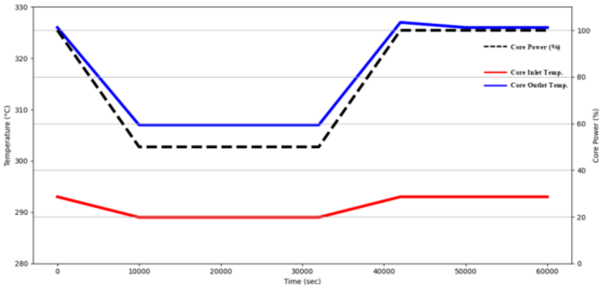


Fig. 3. Simulated core coolant temperatures under flexible power operation (100-50-100 %) conditions based on Panciak et al., Energies 2024, 17, 6373 [6].

To replicate actual reactor coolant conditions, the preheater was set to 555 °C, achieving a coolant inlet temperature of approximately 290°C and an outlet temperature near 320°C, as shown in Fig. 3. The internal heater temperature representing full power was set at 530 °C, and then manually decreased to 480°C to simulate a 50% power reduction, following a ramp-down rate of 0.5%/min. Thermocouples were installed at the bottom, middle, and top sections of the test section to monitor the local coolant temperatures and assess their transient response. For validation, these results were compared with a simulated daily load-following scenario (100–50–100 %) for the APR1400 reactor [6], in which the core outlet and inlet temperatures dropped by approximately 20 °C and 5 °C, respectively, upon a 50 % power reduction (Fig. 4). In comparison, the experimental loop showed a similar trend, with the outlet temperature decreasing by ~20 °C and the inlet temperature by ~8 °C. This indicates that the loop is capable of following thermal behaviors similar to those in real reactor conditions during flexible operation.

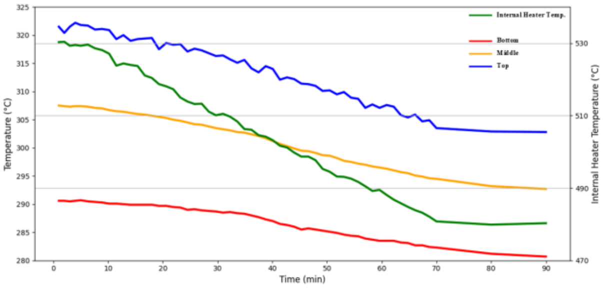


Fig. 4. Coolant temperature response test with manually controlled internal heater output

Going forward, the implementation of an automated heater control system and a differential pressure logging system will further enhance the loop's utility. By enabling real-time monitoring of CRUD growth and detachment under thermal cycling, the loop is expected to serve as a reliable device for studying CRUD behavior under flexible power operations.

3. Conclusions

This study aims to develop a high-temperature & pressure loop system to evaluate the effects of flexible operation on CRUD formation on nuclear fuel cladding surfaces. During load-following operations, repeated thermal cycling may lead to CRUD detachment and redeposition, potentially altering its structure. To simulate these phenomena, a CRUD simulation loop is currently being designed with automated temperature cycling control and a differential pressure logging system. The loop is expected to contribute to the development of flexible operation technologies by enabling structural integrity assessments of fuel cladding under representative thermal and hydraulic conditions.

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REFERENCES

[1] IAEA (International Atomic Energy Agency). "Non-baseload operation in nuclear power plants: load following and frequency control modes of flexible operation." (2018): 1-190.
[2] Deshon, J., Hussey, D., Kendrick, B., McGurk, J., Secker, J., & Short, M. Pressurized water reactor fuel crud and corrosion modeling. JoM, 63(8), 64-72. 2011
[3] Zou, L., Zhang, H., Gehin, J., & Kochunas, B. Coupled thermal-hydraulic/neutronics/crud framework in prediction of crud-induced power shift phenomenon. Nuclear Technology, 183(3), 535-542. 2013
[4] Seo, Seungjin, Nakkyu Chae, and Sungyeol Choi. "Sensitivity evaluation of boron hideout index in the various operation conditions of pressurized water-cooled reactor." Nuclear Engineering and Technology (2025): 104036.
[5] Seungjin Seo. Multi-physics model for CRUD growth and boron hideout during long-term PWR operations. Ph D Thesis. SNU, 2024.
[6] Panciak, Ivan, and Aya Diab. "Dynamic Multiphysics Simulation of the Load-Following Behavior in a Typical Pressurized Water Reactor Power Plant." Energies 17.24 (2024): 6373.