NUREG/IA-0558

International Agreement Report

Chinshan BWR Decommissioning SBO Analysis with RELAP5/MOD 3.3

Prepared by:
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Washington, DC 20555-0001

Manuscript Completed: December 2025
Date Published: February 2026

Prepared as part of
The Agreement on Research Participation and Technical Exchange
Under the Thermal-Hydraulic Code Applications and Maintenance Program (CAMP)

Published by
U.S. Nuclear Regulatory Commission

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ABSTRACT

This project is responsible for analyzing the decommissioning planning and operational safety of Taiwan Power Company's nuclear power plants. Over the four-year project period, analysis will be conducted using the SNAP/RELAP5 program. The current project aims to establish a model for the decommissioning transitional phase of Kuosheng Nuclear Power Plant using SNAP/RELAP5, supplemented by heat flow formula calculations and TRACE analysis for validation and comparison [1]. This effort will assist Taiwan Power Company in conducting assessments related to the decommissioning transitional phase of Chinshan NPP and other associated tasks.

FOREWORD

RELAP5 is a thermal hydraulic analysis code and has been designed to perform best-estimate analysis of LOCA, operational transients, and other accident scenarios for Nuclear Power Plants. Traditionally, RELAP5 models were developed by ASCII files, which was not intelligible for the beginners of computer analysis. A graphic input interface code-SNAP is developed by Applied Programming Technology Inc. and can process the establishment of the RELAP5 models more conveniently.

Taiwan and the United States have signed an agreement on CAMP to obtain the authorization of these codes. NTHU is the organization in Taiwan responsible for the application RELAP5 and SNAP in safety analysis of nuclear power plants. Hence, the RELAP5/SNAP model of Chinshan BWR NPP has been developed. Since 2019, the Chinshan Nuclear Power Plant has officially entered the decommissioning stage. Both units have been shut down for more than six and five years, respectively. As the spent fuel pools of the two units are nearing full capacity, the fuel assemblies from the last operating cycle have been temporarily retained inside the reactors. Over time, the decay heat of this spent fuel has significantly decreased; however, a residual heat removal system must still be maintained to ensure the prevention of potential risks. This study focuses on the safety issues related to the temporary storage of spent nuclear fuel in the reactor pressure vessel during the decommissioning transition phase of Chinshan NPP A thermal-hydraulic analysis model for the decommissioning transition phase of Chinshan NPP was developed using the RELAP5 program, and a case study and safety assessment of a loss of cooling water accident were conducted.

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EXECUTIVE SUMMARY

The decommissioning plan for Chinshan Nuclear Power Plant is divided into four stages, with a duration of 25 years: the decommissioning transition phase, the decommissioning dismantling phase, the site final condition detection phase, and the site restoration phase. This study primarily simulates the scenario described in Chapter 5, Section 2 of the decommissioning plan, titled "Period with Fuel Remaining in the Core [2]." During this period, spent nuclear fuel still remains in the reactor core and due to insufficient storage space in the spent fuel pool, some spent nuclear fuel needs to be temporarily stored within the reactor core. During this period, it is necessary to continue providing power supply, maintaining cooling water replenishment measures, and other support systems to facilitate the removal of residual heat from the spent nuclear fuel.

The research method will first establish the SNAP/RELAP5 model of the decommissioning transition phase of Chinshan Nuclear Power Plant. Then, combined with the decay heat data provided by Taiwan power company for 7 days to 7 years, the SBO case is analyzed by this SNAP/RELAP5 model.

ABBREVIATIONS

CAMP	Code Applications and Maintenance Program
NRC	Nuclear Regulatory Commission
NTHU	National Tsing Hua University
SBO	Station Blackout
NPP	Nuclear Power Plant
FSAR	Final Safety Analysis Report
SNAP	Symbolic Nuclear Analysis Program
BWR	Boiling Water Reactor
TSS	Transient Simulation Steady State
EPRI	Electric Power Research Institute
SFP	Spent Fuel Pool
TAF	Top Active Fuel

1 INTRODUCTION

Chinshan NPP utilizes the General Electric's sixth-generation Boiling Water Reactor (BWR/4) paired with the Mark III containment vessel. The space above the reactor pressure vessel is referred to as the reactor cavity. During the decommissioning transitional phase at the Chinshan NPP, the reactor's upper cover will be opened to establish communication with the spent fuel pool. When modeling, considerations are given to the program's computational capabilities and the complexity of model establishment, leading to simplifications in certain area configurations.

During the initial phase of decommissioning at Chinshan NPP, known as the transition phase, the reactor core temporarily stores the most recent batch of spent nuclear fuel while awaiting the activation of dry storage. The decay heat of this spent nuclear fuel decreases with longer shutdown periods. However, the decay heat of the spent fuel temporarily stored in the reactor core remains relatively high. Therefore, this study aims to assess the impact of the decay heat of spent nuclear fuel in the reactor core during the early stage of shutdown on the development of accidents.

The study's simulation conditions are based on input from relevant sources such as the "Taiwan Power Company Nuclear Backend Operations Division, Chinshan NPP Decommissioning Plan, 2021," operational cycle assessment reports, and monitoring records during decommissioning. Initial conditions such as power, decay heat versions, and other necessary parameters are considered. Additionally, insights from case analysis reports on the decommissioning transition phases of Kuosheng NPP, conducted by NUREG/IA-0555 using the RELAP5 and TRACE analysis model, are incorporated.

Once the model is established, we will compare the results with those calculated using heat flow equations and explore four different decay heat calculation methods, such as ASB 9-2, RG 3.54, ANS 5.1, and ISO 10645. The analysis results will be compared with the thermal-hydraulic equations from an Electric Power Research Institute (EPRI) [3] report to validate the reliability of the SNAP/RELAP5 model in open-loop mode. Subsequent case simulations will be improved to assess the development of scenarios following a station blackout (SBO) and the loss of cooling water. The assessment will include determining the available time for the plant to implement recovery measures to prevent fuel exposure and ensure plant safety.

2 MODEL AND METHODOLOGY DESCRIPTION

2.1 Chinshan RELAP5/SNAP Model Description

During the decommissioning transitional phase of the Chinshan NPP, the reactor core still contains 408 bundles of ATRIUM-10 used nuclear fuel from the 28th cycle [4][5]. Accurately estimating decay heat requires considering many factors, such as the past operating power of the spent fuel, burnup, uranium concentration, time, etc. Currently, there are four main methods for calculating decay heat, including ASB 9-2 (NUREG-0800) [6] and RG 3.54 (NRC) [7] published by the U.S. Nuclear Regulatory Commission (NRC), ANSI/ANS 5.1 (American Standard) [8] published by the American Nuclear Society, and ISO 10645 (ISO 10645) [9], refer to Table 2-1 and Table 2-2.

First, there is ASB 9-2 (NUREG-0800), published by the U.S. Nuclear Regulatory Commission (U.S. NRC) in 1987. This calculation method is primarily for the design and safety analysis of light-water reactors (LWRs), particularly considering the cooling process after a reactor shutdown. It focuses on the contributions of fission products and heavy elements and is suitable for short-term decay heat calculations, thus applicable for cooling and safety needs in the short period following a reactor shutdown. Next is ANSI/ANS 5.1 (American Standards), published by the American Nuclear Society in 2014. It is currently a widely used standard in U.S. nuclear power plants. Compared to ASB 9-2, in addition to fission products, it incorporates the effects of neutron capture and actinides on decay heat, allowing for a more comprehensive decay heat calculation. It is suitable for long-term decay heat calculations and can accommodate different types of fuel core configurations. There is also ISO 10645 (European Standards), proposed by the International Organization for Standardization (ISO) in 1992. Its calculations are like ANSI/ANS 5.1, also considering the effects of fissionable materials, neutron capture, and actinides on decay heat, but its primary design is for European nuclear power plants, including their specific design and operating conditions. Finally, there is the latest decay heat calculation method, RG 3.54 (Nuclear Regulatory Commission) Revision 3, proposed by the U.S. NRC in 2022. Compared to the other methods, RG 3.54 not only considers fission products, neutron capture, and actinides but also specifically covers the effect of structural material activation on decay heat, which is crucial for long-term storage facilities. It also mentions that RG 3.54 has greater adaptability for high-burnup fuel, providing a larger safety margin for long-term fuel storage and safety management. In comparing the four decay heat calculation methods above and considering that units 1 and 2 of the Jinshan Nuclear Power Plant have been shut down for about 5 to 6 years, and the NRC regulatory guide permits its use, along with the long-term storage of fuel in the spent fuel pool and within the reactor core, RG 3.54 was chosen as the basis for this simulation. Compared to other methods, it can provide more precise data, thereby ensuring the safety of long-term storage.

The case study was designed with 12 scenarios based on the number of days since shutdown: 7, 30, 60, 90, 180, and 365 days, as well as 2, 3, 4, 5, 6, and 7 years. The boundary pressure of the overall system was maintained at one atmosphere (1.013×10^5 Pa), with the reactor core cooling water temperature set at 27.5 °C and the spent fuel pool cooling water temperature set at 26.6 °C.

The core design for both units at the Chinshan Nuclear Power Plant is identical, with a rated thermal power of 1,775 MWt (MLU to 1,840 MWt in November 2012). As RELAP5 can only accommodate one POWER setting, a proportional allocation was performed based on the decay heat generated by each fuel rod. Referencing Table 2-3, for example, on day 7, the total decay heat was 3.212 MW, with 1.536 MW from the reactor core and 1.676 MW from the spent fuel pool. The respective proportions of the total decay heat were then calculated as 0.478 and 0.522.

2.2 Analysis Methodology Description

The model was built with the bottom of the gate serving as the reference point, setting the height of the connecting pipe at +6.08 meters to link the reactor cavity and the spent fuel pool. The total volume above this reference point is 877 cubic meters. The reactor cavity is divided into 6 vertical layers and 3 horizontal layers, with the left and right layers interconnected to accurately simulate the fluid and heat transfer within each layer. The spent fuel pool is divided into 5 vertical layers and 3 horizontal layers, with heat sources placed at the bottom to simulate the decay heat, referenced in Figure 2-1 and 2-2.

To align with the current state of the decommissioning transition phase, the pressures at both the inlet and outlet are set to 1 atmosphere, and a pipe connecting to the external ventilation has been installed. Decay heat values are introduced via Heat Structure into four central Pipe components, which serve as heat sources. This ensures that the power is accurately distributed based on the specific conditions of different batches of fuel rods, with separate sections for heated and unheated components to simulate water rods, control rods, and other assemblies, referenced in Figure 2-3. The upper portion of the reactor pressure vessel, where most functional components have been removed and only water storage remains, is simulated using a single BRANCH component. Its height and volume are maintained at their original settings. The model setup is referenced in Figure 2-4.

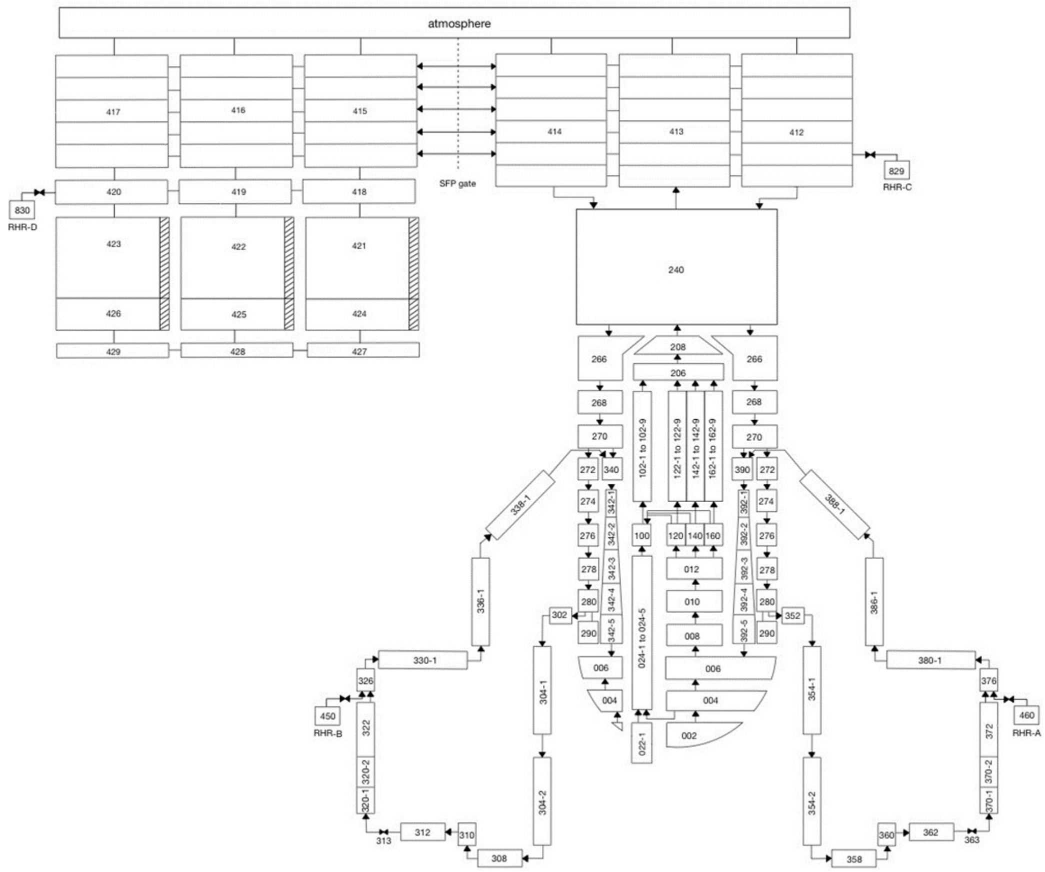

Figure 2-1 Chinshan NPP Decommissioning Model Node Graph

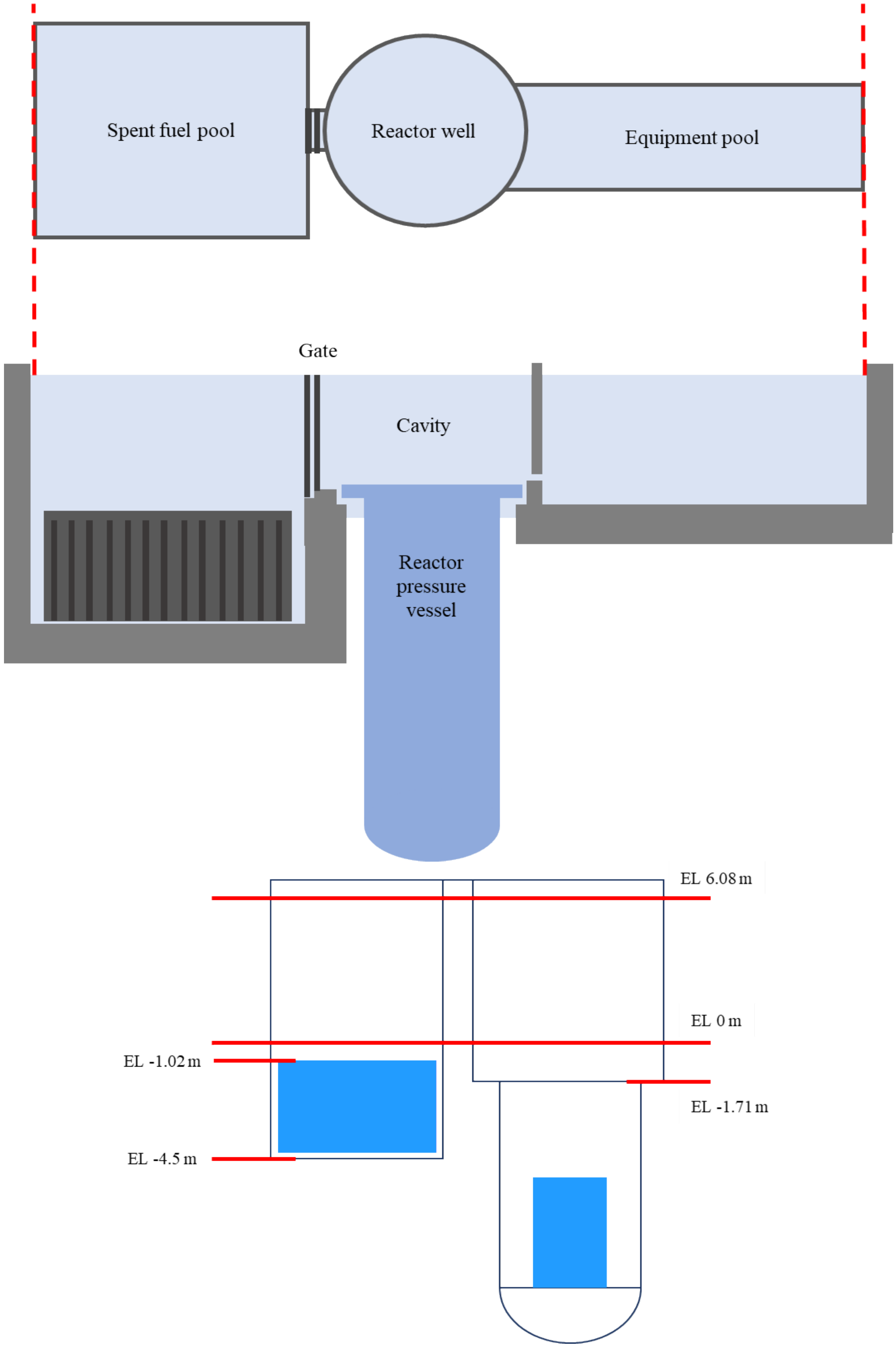

Figure 2-2 Chinshan Decommissioning Model Front and Vertical View Diagrams

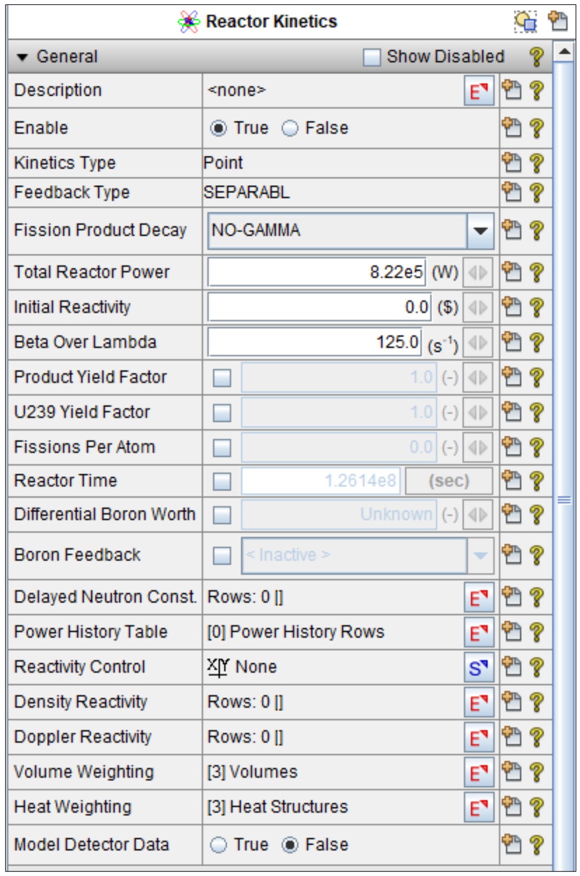

Figure 2-3 Decay Heat Power Setting

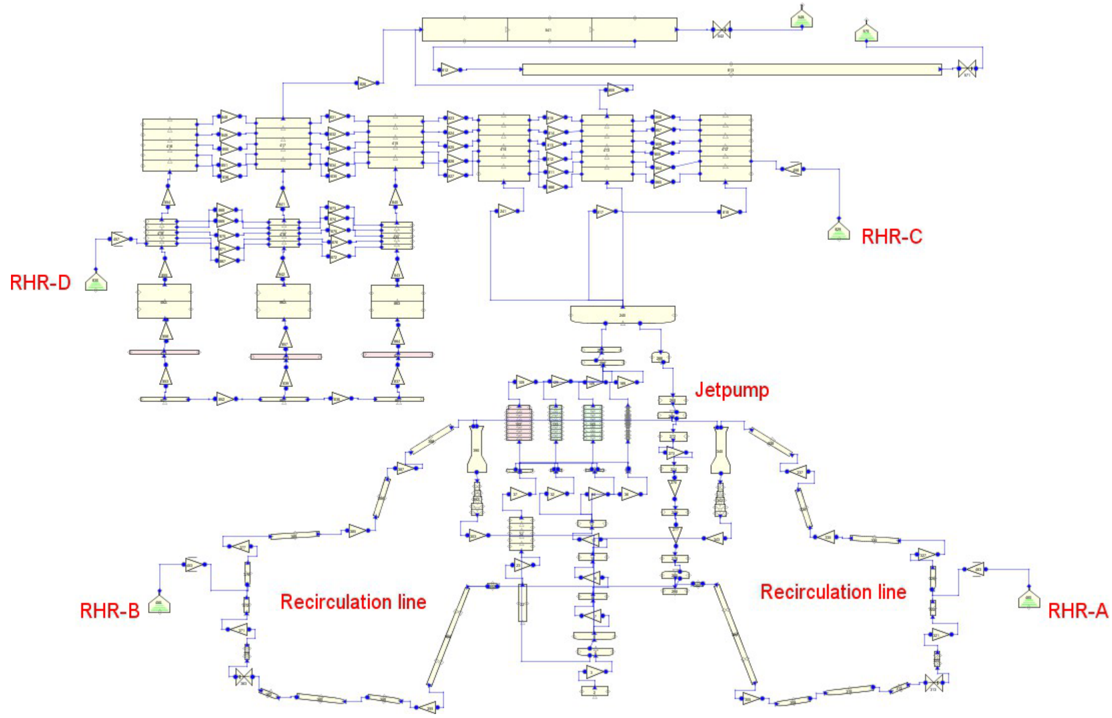

Figure 2-4 The RELAP5/SNAP of Chinshan NPP Decommissioning Model

Table 2-1 Comparison of Decay Heat Application Scope and Contribution Factors

	Range (After Reactor Shut Down)	Contribution Factor
ASB 9-2	$0 \sim 10^7$ s (~115 day)	Fission products Heavy elements
RG 3.54	1 ~110 year	Fission products Neutron capture Actinides Activation of structural materials
ANSI/ANS 5.1	$0 \sim 10^{10}$ s (~320 year)	Fission products Neutron capture Actinides
ISO 10645	$0 \sim 10^9$ s (~32 year)	Fission products Neutron capture Actinides

Table 2-2 Table of Publication Years of Decay Heat Models

Decay Heat model	Year of publication
ASB 9-2 (NUREG-0800)	1978
ANSI/ANS 5.1 (American Standards)	2014
ISO 10645 (European Standards)	2022
RG 3.54 (Nuclear Regulatory Commission) Revision 3	2022

Table 2-3 Table of RG 3.54 Decay Heat Numerical Ratio

After reactor shutdown	SFP (MW)	Reactor core (MW)	Total (MW)
7 days	1.676	1.536	3.212
30 days	1.477	1.334	2.811
60 days	1.305	1.159	2.464
90 days	1.167	1.019	2.186
180 days	0.850	0.699	1.549
365 days	0.489	0.333	0.822
2 years	0.238	0.082	0.320
3 years	0.181	0.029	0.21
4 years	0.167	0.018	0.185
5 years	0.163	0.015	0.178
6 years	0.160	0.015	0.175
7 years	0.159	0.015	0.174

3 THE ANALYSIS RESULTS OF THE BASE CASES

3.1 Description of Energy Conservation Formula

The comparative data analysis will employ heat transfer equations to calculate relevant parameters, serving as a reference for validating the results obtained from simulations. The formula used is derived from a report published by the Electric Power Research Institute (EPRI) in 2012, which analyzed the spent fuel pool of Japan's Fukushima Daiichi Nuclear Power Plant following the March 11 incident. Based on the principle of energy conservation, a simplified heat transfer equation was developed to estimate the changes in pool water temperature, the evaporation time, and the time required for the water level to drop to the top of the fuel rack in the event of a loss of cooling water.

The overall system equations are expressed in formulas (4-1) to (4-3), where Q is the decay heat released by the spent nuclear fuel (kJ); m is the mass of water (kg); s is the specific heat of water (4.184 kJ·kg/℃); ΔT is the temperature change (℃); P is the decay heat power (kW/s); t is the time required to reach the saturation temperature (s); V is the volume of water (m³); ρ is the water density (kg/m³); and T_init is the initial system water temperature (℃).

For decay heat corresponding to different cooling periods, the detailed heating sequence is shown in Table 8, and the temperature rise trend is illustrated in Figure 13.

$$Q = ms\Delta T \quad (4\text{-}1)$$

$$Pt = V\rho s(100 - T_{init}) \quad (4\text{-}2)$$

$$t = \frac{V\rho s(100 - T_{init})}{P} \quad (4\text{-}3)$$

After the system water temperature reaches the saturation point, the spent fuel continues to release decay heat. This heat is absorbed by the cooling water and converted into latent heat of vaporization. Therefore, the pool water evaporation can be calculated using the equations shown in formulas (4-4) to (4-6), where Q represents the power (kJ); m is the mass of water (kg); L is the latent heat of vaporization of water (kJ/kg); P is the decay heat power (kW/s); t is the time required for the water level to reach a specific height (s); and V is the volume of water (m³). For decay heat corresponding to different cooling durations, the detailed heating sequence is provided in Table 9

$$Q = mL \quad (4\text{-}4)$$

$$Pt = V\rho L \quad (4\text{-}5)$$

$$t = \frac{V\rho L}{P} \quad (4\text{-}6)$$

3.2 The Results and Discussion

The results show that, within one day, the outlet pressure of the upper pool remained stable at approximately one atmosphere, as shown in Figure 3-1. The reactor core temperature and the spent fuel pool temperature were around 30 °C and 29 °C, respectively, consistent with the conservatively defined initial conditions, as illustrated in Figures 3-2 and 3-3. The overall system temperature was approximately 27 °C, as shown in Figure 3-4. Furthermore, the variations in inlet and outlet flow rates were consistent, confirming that the system was in a steady-state equilibrium condition, as shown in Figures 3-5 and 3-6.

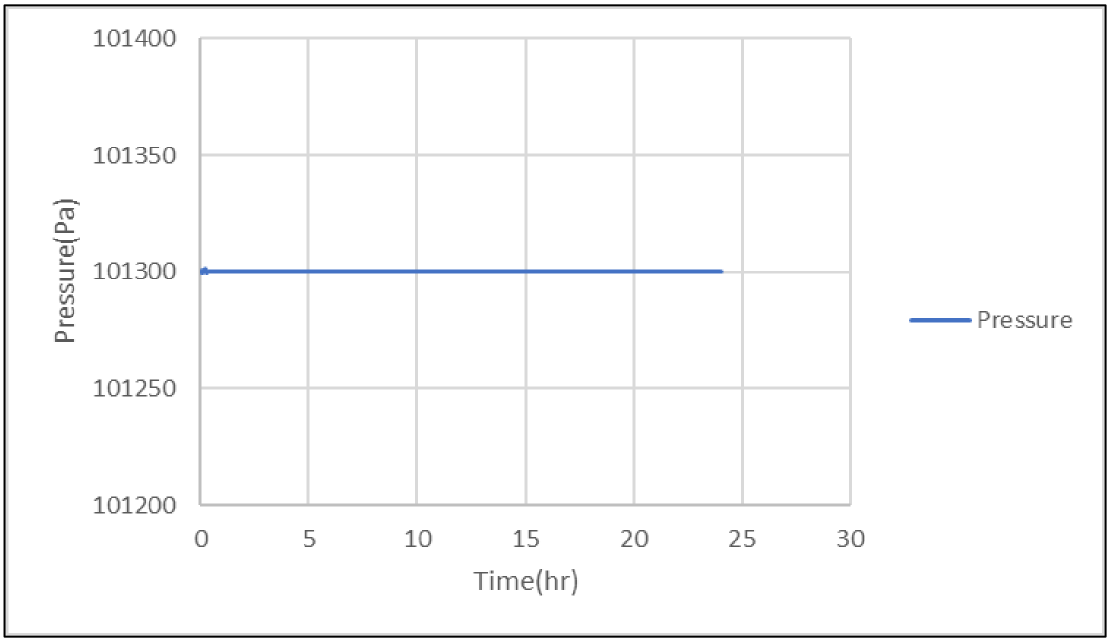

Figure 3-1 Upper Pool Pressure

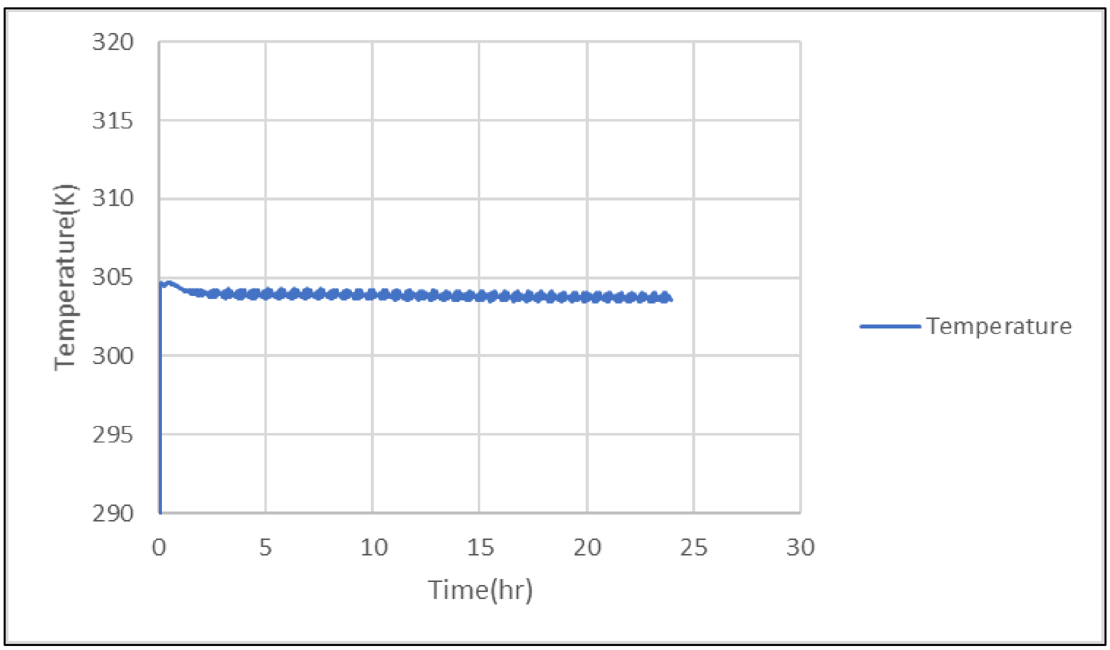

Figure 3-2 Temperature of Reactor Core

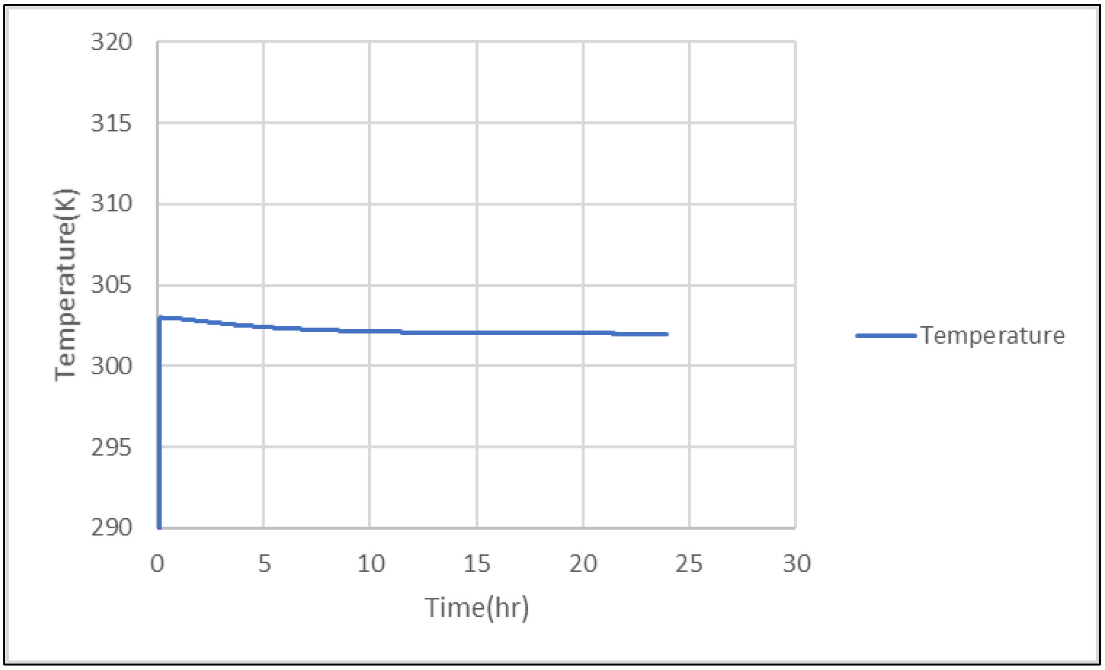

Figure 3-3 Temperature of SFP

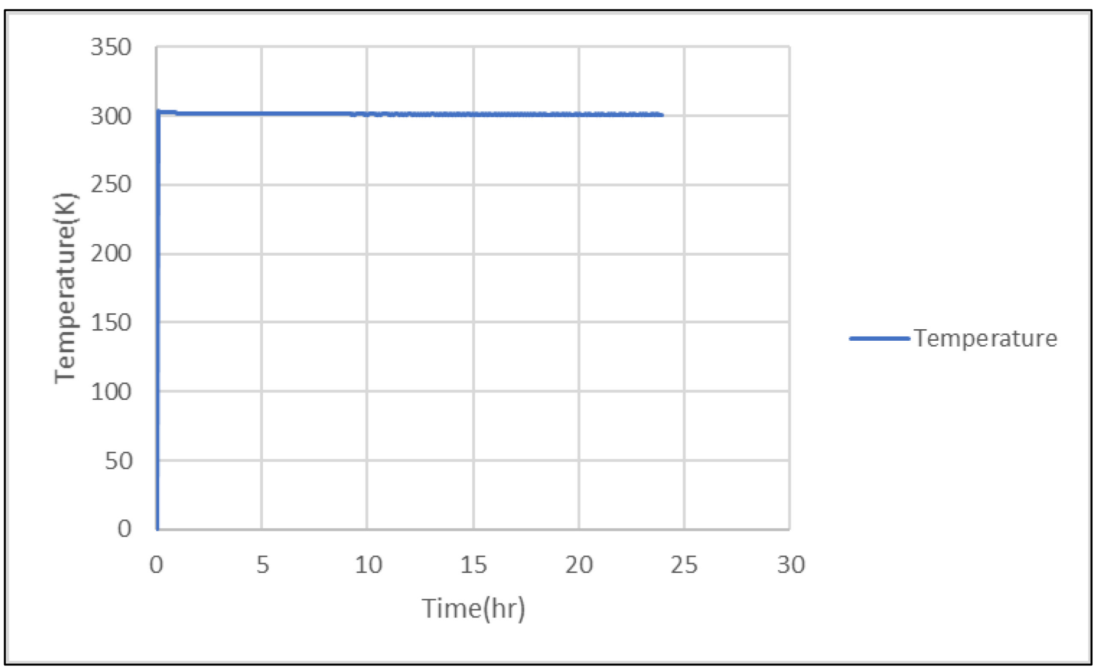

Figure 3-4 The Average of Total Temperature

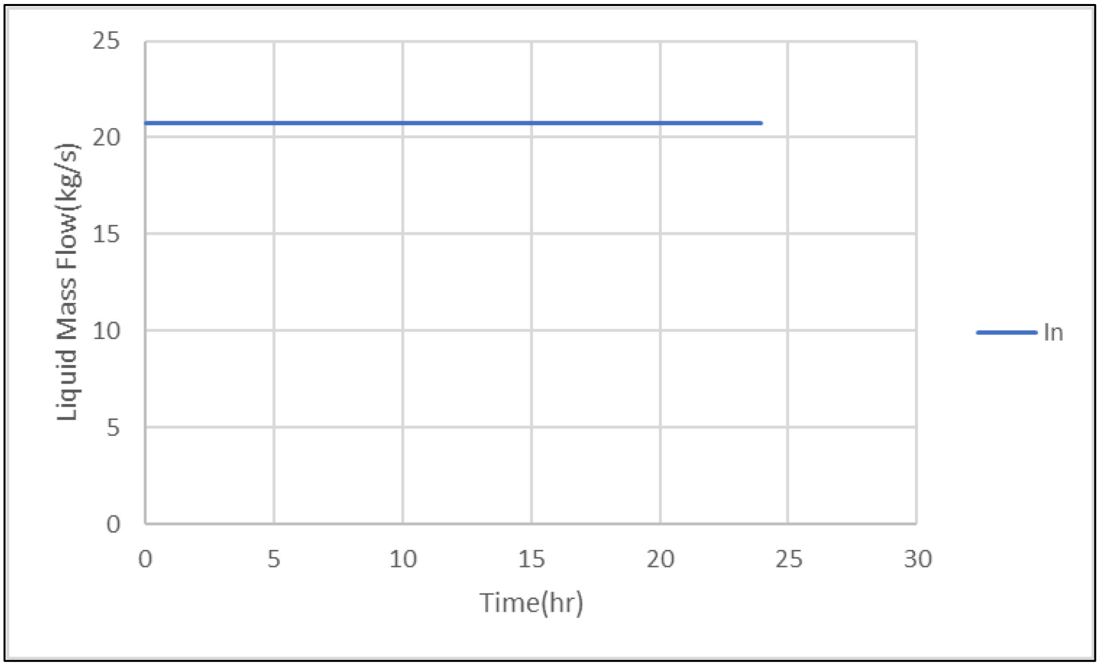

Figure 3-5 Mass Flow of Water Injection

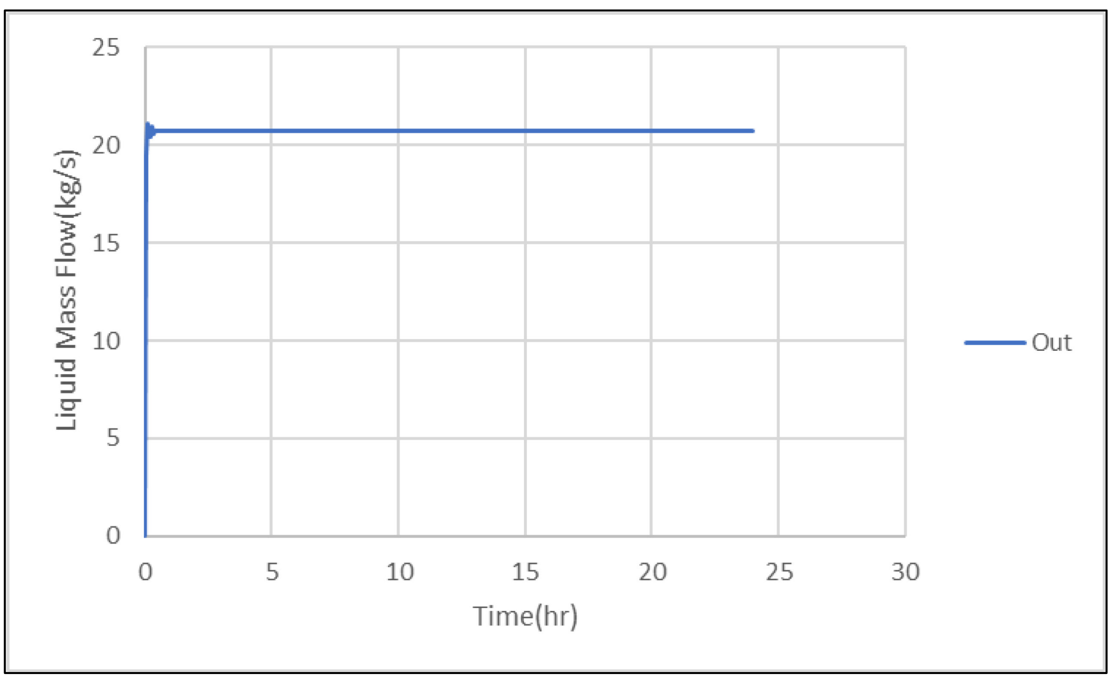

Figure 3-6 Mass Flow of Exit

4 THE ANALYSIS RESULTS OF THE CASES

First, an analysis was conducted on the boil-off time at the bottom of the reactor gate. Table 4-1 presents the comparison of gate-bottom dry-out times under different decay heat levels. As the decay heat decreases, the time required for the water level to drop to the gate bottom increases. When the decay heat is 3.212 MW (corresponding to seven days after shutdown), the gate-bottom dry-out time is 8.34 days. When the decay heat decreases to 0.173 MW (corresponding to seven years after shutdown), the dry-out time extends to 153 days, indicating that lower decay heat results in a significantly longer water boil-off duration.

Next, the boil-off time at the top of the spent fuel rack was analyzed. Table 4-2 shows the comparison of dry-out times at the fuel rack top under different decay heat levels. A similar trend is observed—the required time increases markedly as decay heat decreases. When the decay heat is 3.212 MW (seven days after shutdown), the dry-out time is 9.56 days. When the decay heat decreases to 0.173 MW (seven years after shutdown), the dry-out time extends to 165.38 days. The detailed trend is illustrated in Figure 4-1, where the two reference elevation lines represent the gate bottom (5.95 m) and the fuel rack top (4.93 m). Overall, the SNAP/RELAP5 simulation results show a good agreement in trend with the EPRI analytical formulas.

Finally, the core water level variation was examined. Table 4-3 presents the comparison of fuel top dry-out times under different decay heat levels. When the decay heat is 3.212 MW (seven days after shutdown), the dry-out time is 13.16 days, whereas when the decay heat decreases to 0.173 MW (seven years after shutdown), the dry-out time extends to 706.36 days. The detailed trend is shown in Figure 4-2, where the two reference elevation lines represent the gate bottom (24.996 m) and the top of the reactor core fuel (12.83 m).

In summary, as the decay heat decreases, the time required for the water level to drop to a specific height becomes significantly longer. Under the open-vessel mode, the SNAP/RELAP5 model can reasonably capture the water-level change trends in the spent fuel pool, showing good consistency with the EPRI analytical results. This demonstrates that under varying decay heat conditions, the model can successfully predict the water-level reduction behavior and provide reasonable estimates for dry-out times.

For comparison using the simulation results corresponding to five years after shutdown, the transient results are as follows:

1. Time to reach saturation temperature: SNAP/RELAP5 ≈ 28.51 days, EPRI ≈ 35.74 days.
2. Time for water level to reach gate bottom: SNAP/RELAP5 ≈ 137.00 days, EPRI ≈ 159.60 days.
3. Time for pool water level to reach rack top: SNAP/RELAP5 ≈ 148.83 days, EPRI ≈ 171.74 days.
4. Time for reactor water level to reach fuel top: SNAP/RELAP5 ≈ 690.20 days, EPRI ≈ 694.77 days.

To provide a clearer observation of the various data in the simulation, an animation of the SNAP/RELAP5 simulation process is provided, as shown in Figure 4-3.

Table 4-1 Table of Time for Water Level to Reach the Bottom of the Gate

After Reactor shutdown	Decay Heat (MW)	SNAP/RELAP5 (Days)	EPRI (Days)
7 days	3.212	8.34	8.84
30 days	2.811	9.79	10.11
60 days	2.464	11.09	11.53
90 days	2.186	12.69	13.00
180 days	1.549	17.78	18.34
365 days	0.822	33.66	34.56
2 years	0.320	84.57	88.78
3 years	0.210	123.77	135.28
4 years	0.185	132.68	153.56
5 years	0.178	137.00	159.60
6 years	0.175	140.96	162.34
7 years	0.174	153.00	164.21

Table 4-2 Table of Time for Water Level to Reach the Top of the Fuel Rack

After Reactor shutdown	Decay Heat (MW)	SNAP/RELAP5 (Days)	EPRI (Days)
7 days	3.212	9.56	10.02
30 days	2.811	11.13	11.44
60 days	2.464	12.65	13.04
90 days	2.186	14.43	14.68
180 days	1.549	20.17	20.65
365 days	0.822	37.95	38.58
2 years	0.320	93.87	97.04
3 years	0.210	133.06	146.15
4 years	0.185	144.68	165.34
5 years	0.178	148.83	171.74
6 years	0.175	153.66	174.63
7 years	0.174	165.38	176.58

Table 4-3 Table of Time for Water Level to Reach the TAF

After Reactor shutdown	Decay Heat (MW)	SNAP/RELAP5 (Days)	EPRI (Days)
7 days	3.212	13.16	14.07
30 days	2.811	15.32	16.13
60 days	2.464	17.44	18.46
90 days	2.186	19.83	20.87
180 days	1.549	28.11	29.82
365 days	0.822	54.22	58.67
2 years	0.320	174.09	186.67
3 years	0.210	408.61	412.09
4 years	0.185	596.64	599.54
5 years	0.178	690.20	694.77
6 years	0.175	694.48	697.51
7 years	0.174	706.36	699.38

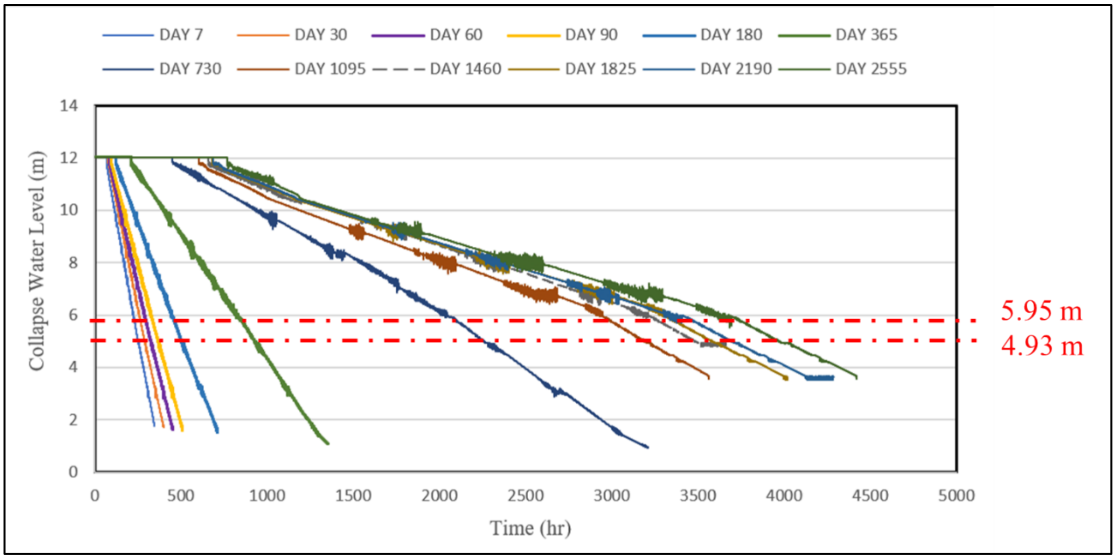

Figure 4-1 SFP Water Level (Bottom of Gate 5.95m, Top of Fuel Grid 4.93m)

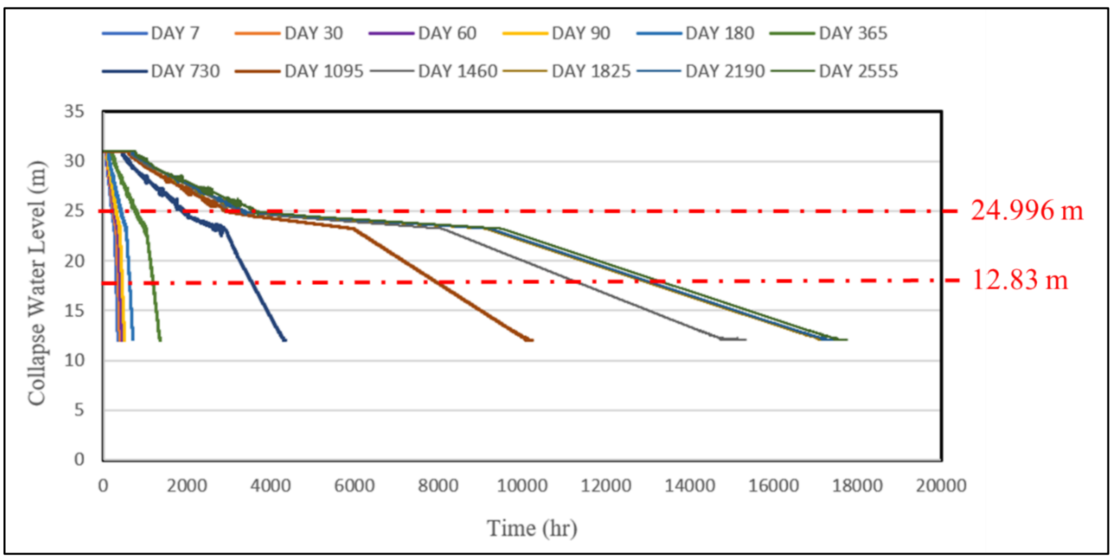

Figure 4-2 Core Water Level (Bottom of Gate 24.996m, TAF 12.83m)

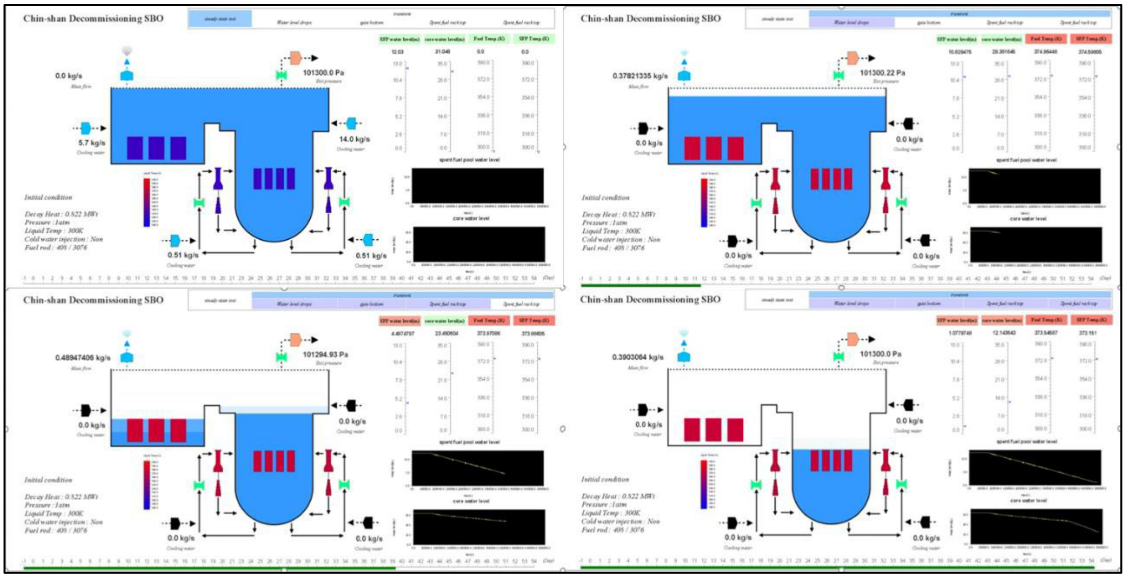

Figure 4-3 SNAP/RELAP5 Chinshan Decommissioning SBO Animation

5 CONCLUSION

This study completed the open-vessel mode model for the decommissioning transition phase of the Chinshan Nuclear Power Plant using RELAP5/MOD3.3 with the graphical user interface SNAP. The NRC-approved decay heat calculation standard RG 3.54 Rev. 3 was adopted, and the results were compared with those obtained using thermal-hydraulic formulas from EPRI, in order to more accurately assess the effects of reactor water level and temperature variations. The analysis verified that this model is capable of simulating a SBO accident under decommissioning conditions.

1. Model Design:

In this study, the graphical user interface SNAP was employed to make the analysis more visualized and consistent with actual operational requirements. In particular, the decay heat distribution between the reactor core and the spent fuel pool at different locations was considered, resulting in more reliable simulation data.
For the open-vessel mode, multilayer longitudinal and transverse nodalization was applied to enable more uniform heat transfer and to simulate natural convection. This stratified design contributes to stable heat flow during long-term storage, enhancing safety.

2. Decay Heat Calculation:

The decay heat in this simulation was calculated using RG 3.54 Rev. 3, which is more suitable for long-term shutdown conditions compared to other decay heat models. The results indicate that after 5–6 years of shutdown, the decay heat from the spent fuel pool decreases significantly, delaying the effects on water level and temperature. This confirms the stability of long-term storage and ensures sufficient time for emergency response following an accident.

3. Numerical Analysis:

In comparing the results from SNAP/RELAP5 with those obtained using EPRI formulas, it was observed that when a smaller time step was applied, the simulated water level in the spent fuel pool reached the top of the fuel racks more quickly than when using the default value. According to “Some Comments on the Behavior of the RELAP5 Numerical Scheme at Very Small Time Steps” published in Nuclear Science and Engineering [10], this phenomenon can be attributed to reduced numerical diffusion with smaller time steps, which allows faster physical phenomena to be captured and minimizes potential numerical oscillations. This further verifies the sensitivity of the simulation time to numerical computation parameters.

4. Safety Analysis:

During the early shutdown period, the reactor core contained fuel from the final operating cycle, resulting in higher decay heat; thus, core cooling was the primary concern at that stage. However, as the shutdown period extended, the core’s decay heat gradually decreased while the spent fuel pool’s decay heat became relatively higher, making the spent fuel pool the main focus for safety considerations in the later stages.

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NRC FORM 335
(12-2010)
NRCMD 3.7

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

1. REPORT NUMBER
(Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG/IA-0558

2. TITLE AND SUBTITLE

Chinshan BWR Decommissioning SBO Analysis with RELAP5/MOD 3.3

3. DATE REPORT PUBLISHED

MONTH	YEAR
February	**2026**

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

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6. TYPE OF REPORT

Technical

7. PERIOD COVERED (Inclusive Dates)

8. PERFORMING ORGANIZATION - NAME AND ADDRESS (If NRC, provide Division, Office or Region, U. S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)

National Tsing Hua University and Nuclear and New Energy Education and Research Foundation
101 Section 2, Kuang Fu Rd.,HsinChu, Taiwan
Department of Nuclear Safety, Taiwan Power Company
242, Section 3, Roosevelt Rd., Zhongzheng District, Taipei, Taiwan

9. SPONSORING ORGANIZATION - NAME AND ADDRESS (If NRC, type "Same as above", if contractor, provide NRC Division, Office or Region, U. S. Nuclear Regulatory Commission, and mailing address.)

Division of Systems Analysis
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

10. SUPPLEMENTARY NOTES

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11. ABSTRACT (200 words or less)

This four-year project focuses on safety analysis and decommissioning planning for Taiwan Power Company's nuclear power plants. Using SNAP/RELAP5, the project will develop a transitional decommissioning model for the Kuosheng Nuclear Power Plant. Heat transfer calculations and TRACE simulations will be used for validation and comparison. The results will support decommissioning assessments for the Chinshan Nuclear Power Plant and related activities.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)

Decommission, SBO, SNAP/RELAP5, Decay Heat

13. AVAILABILITY STATEMENT

unlimited

14. SECURITY CLASSIFICATION

(This Page)

unclassified

(This Report)

unclassified

15. NUMBER OF PAGES

16. PRICE

Printed on recycled paper

Federal Recycling Program

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, DC 20555-0001

OFFICIAL BUSINESS

@NRCgov

NUREG/IA-0558 **Chinshan BWR Decommissioning SBO Analysis with RELAP5/MOD 3.3** **February 2026**