

Integration Strategy of the Domestic CAP Code for CANDU Reactor Building Analysis

Bong-Jin Ko*, Dong-Sik Jin, Kang-Moon Lee, Eun-Bin Mo, Sook-Kwan Kim, Young-Chan Park, Jun-Hyun Park
STANDARD Testing & Engineering Inc., 168, Gajeong-ro, Yuseong-gu, Daejeon, 34129

*Corresponding author: bjko@stdte.co.kr

***Keywords :** CANDU, Reactor Building, GOTHIC, CAP

1. Introduction

Thermal-hydraulic analysis of reactor buildings in domestic pressurized heavy water reactors (CANDU) has long been performed using GOTHIC code. GOTHIC is a representative code used to evaluate pressure and temperature behavior, hydrogen mixing, and the performance of Engineered Safety Features (ESFs) within the reactor building. In the current regulatory environment, GOTHIC code has evolved up to Version 8.5[1]. However, the domestic CANDU sector has only acquired licensing for Version 7.2a.[2] This limitation presents challenges in adopting the latest features and effectively addressing the safety evaluations required for the continued operation of long-term operating nuclear power plants.

Recently, to secure approvals for the continued operation of CANDU plants, the demand for reactor building analysis has increased and diversified. This includes safety analyses for Design Basis Accidents (DBAs) and Beyond Design Basis Accidents (BDBAs), thermal-hydraulic analysis for Probabilistic Safety Assessments (PSAs), compartment pressure analysis, and long-term cooling evaluations. Developed by the Electric Power Research Institute (EPRI), GOTHIC code is subject to usage restrictions, such as licensing issues, which hinder response to these expanding and diversifying demands. Consequently, there is an urgent need to break away from dependence on foreign commercial codes and introduce a technologically self-reliant reactor building analysis code optimized for the domestic nuclear power plant environment.

CAP code, developed primarily by the Korea Hydro and Nuclear Power (KHNP)[3], is a domestic reactor building analysis code capable of performing pressure and temperature evaluations, hydrogen behavior analysis, and ESF modeling for pressurized water reactors (PWRs). This study reviews the technical feasibility of introducing CAP code to the field of CANDU reactor building analysis and proposes future development directions through a comparative analysis with GOTHIC code.

2. Current Status and Limitations of GOTHIC Code

2.1 Code Overview and History of Domestic Applications

GOTHIC code is a computational platform designed to analyze the multidimensional thermal-hydraulic behavior of reactor buildings and large-scale fluid systems. In domestic CANDU plants, it has been primarily utilized for safety analysis and licensing in the following areas:

- Analysis of reactor building pressure and temperature during DBAs
- Analysis of reactor building long-term cooling and compartment pressure
- Analysis of hydrogen behavior and average hydrogen concentration

2.2 Technical Characteristics and Limitations

GOTHIC code provides a high degree of modeling flexibility by offering multidimensional flow and natural circulation analysis capabilities, along with a rich set of physical models such as condensation, evaporation, spray systems, and fan coolers. However, the fact that the licensing foundation in the domestic CANDU sector is tied to Version 7.2a acts as a technical constraint.

In particular, one of the core objectives of reactor building thermal-hydraulic analysis is to evaluate the environmental release of radionuclides during an accident to perform dose analysis. Currently, the domestic CANDU sector relies on a manual linkage, where the thermal-hydraulic results from GOTHIC are used as input data for the SMART code[4] (for radionuclide behavior analysis) and the ADDAM code [5](for external dose analysis). While the latest GOTHIC Version 8 incorporates these radionuclide behavior analysis functions natively, actual application is impossible in domestic and foreign CANDU fields because the Industry Standard Toolset (IST) certification and licensing foundation have not been established.

3. Characteristics and Model Configuration of CAP Code

3.1 Code Overview and Connectivity

CAP code was developed to evaluate reactor building integrity, Emergency Core Cooling System (ECCS) performance, and Environmental Qualification (EQ) envelopes for large dry reactor buildings currently under operation and construction in Korea. Notably, it supports not only standalone calculations but also

bidirectional coupled calculations with SPACE, a state-of-the-art system safety analysis code. It establishes an integrated analysis framework where it receives mass and energy release rates (source terms) from the SPACE code and feeds back the calculated reactor building pressure to SPACE as a backpressure condition.

3.2 Structure of Input Data

The input file for CAP code is structured in an ASCII-based text format consisting of title lines (=), input lines, comment lines (*), and termination lines (End). The input lines follow the format of <Keyword + Input Value> and exhibit sequence dependency. Since memory is allocated immediately as input elements are read sequentially, elements referenced during connectivity modeling must be declared in advance. For example, before defining a junction, which represents a flow path, the two volumes to be connected must first be defined at the top of the file to prevent errors. Input format errors and missing mandatory data are verified through the system verification file (.chk).

3.3 Key Physical and ESF Models

CAP code maximizes computational efficiency by simplifying the internal space of the reactor building using a lumped-parameter volume model as its core. It provides models for pipes and junctions as flow paths, mass/energy source terms, and hydrogen generation sources. Furthermore, to simulate engineered safety features, it includes models for pumps/fans, valves/doors, spray nozzles, heaters/coolers, fan coolers, heat exchangers, and Passive Autocatalytic Recombiners (PARs). It also allows for the flexible implementation of complex control logic through trips and control variables.

4. Characteristics and Model Configuration of CAP Code

4.1 Comparison of Modeling Methods and Input Structures

GOTHIC code is capable of detailed grid modeling at a multi-dimensional CFD level, giving it a strong advantage in analyzing localized mixing phenomena, though at the cost of a heavy computational load. Conversely, CAP code is based on a lumped-parameter approach, offering excellent computational efficiency, which makes it advantageous for safety evaluations requiring numerous sensitivity analyses. In terms of user input, GOTHIC provides the convenience of a GUI, whereas CAP code uses a structured ASCII keyword method, which excels at automated input generation via scripting and managing large amounts of data.

4.2 Comparison of Detailed Physical Models and Correlations

The two codes exhibit several key differences in their physical models and correlation schemes, which significantly influence the long-term behavior of the reactor building.

First is the availability of a wall leakage modeling function. GOTHIC code provides a feature to directly model micro-leakage through structural walls, whereas the current CAP code lacks this capability. Wall leakage plays a critical role in long-term reactor building analysis by evaluating radionuclide releases caused by leakage and simulating the containment cooling effect via fluid discharge from compartments. Therefore, this function must be supplemented in future CAP code developments.

Second is the material inventory available for structural thermal conductors. CAP code only provides key material properties for major reactor building structures, such as carbon steel, stainless steel, and concrete. In contrast, GOTHIC code additionally offers aluminum and copper. However, because these metallic materials are rarely utilized in standard reactor building thermal analyses, the impact of this difference on the analytical results is expected to be negligible.

Third is the diversity of convective heat transfer correlations at structural walls. CAP code only provides the Dittus-Boelter correlation for forced convection analysis and is restricted to the McAdams and Yovanovich correlations for natural convection conditions. On the other hand, GOTHIC code built-in an extensive range of standard correlations and employs an open architecture that allows users to input arbitrary, user-defined equations. This gives GOTHIC a distinct advantage in terms of modeling flexibility.

4.3 Technical Self-Reliance and Feasibility of CANDU Application

Since CAP code is a domestic, independently developed code, it is free from licensing restrictions associated with foreign software. It also allows for the immediate reflection of user requirements and domestic regulatory guidelines, offering high flexibility in licensing responses.

Although originally developed for PWR analysis, its flexible lumped-parameter modeling allows it to be technically applicable to CANDU reactor building analyses, such as compartment pressure evaluations, long-term cooling analyses, and average hydrogen concentration assessments. However, it is essential to first establish an input framework tailored to the unique multi-compartment structure of CANDU reactors and specific CANDU accident scenarios (e.g., reactor building behavior during a large-break LOCA), followed by benchmarking against existing GOTHIC results. In particular, the aforementioned absence of a wall leakage model and the limited convective correlations may cause quantitative discrepancies during rigorous analyses of long-term compartment

behavior in CANDU plants, necessitating prior sensitivity evaluations.

4.4 Scalability for Radionuclide and Dose Assessments

Currently, CAP code focuses heavily on thermal-hydraulic and hydrogen behavior analyses and lacks the capability to calculate the transport, deposition, and environmental release of radionuclides during an accident. Consequently, in its current state, there are limitations to establishing an organically integrated dose assessment framework with the SMART and ADDAM codes.

However, since the code can be modified and expanded internally, it possesses strong scalability. If it is coupled with CINEMA—a domestic severe accident analysis code or if internal radionuclide transport models are developed in the future, it can evolve into an integrated safety analysis platform that satisfies CANDU regulatory requirements (REGDOC 2.4.1)[6].

5. Conclusions

This study evaluated the technical feasibility of integrating the domestic CAP code into the field of domestic CANDU reactor building analysis. The evaluation demonstrates that CAP code, backed by its outstanding computational efficiency and comprehensive ESF models, is fully viable for reactor building pressure-temperature behavior evaluations and PSA thermal-hydraulic analyses.

To establish CAP code as the primary tool for the continued operation licensing and safety assessments of operating CANDU plants, the following stepwise development and verification are required:

Verification of CANDU-specific models: Establishment of a CANDU reactor building geometric database and benchmarking against existing GOTHIC 7.2a results (including impact assessments of wall leakage effects and variations in convective correlations).

Establishment of an integrated coupled platform: Consolidating the coupled analysis framework between the primary heat transport system thermal-hydraulic code and CAP, and developing modules to automate radionuclide transport and dose assessment capabilities (linking with SMART and ADDAM).

Securing a regulatory licensing foundation: Performing code verification and sensitivity analyses in compliance with Canadian regulatory requirements (REGDOC 2.4.1) and standard guidelines of the Nuclear Safety Act in Korea.

The successful introduction of CAP code to CANDU analysis is expected to significantly reduce reliance on foreign commercial codes and complete the technological self-reliance of domestic nuclear safety analysis capabilities.

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