Automated F-C Curve Generation: (1) Event Sequence Frequency Quantification Using PCTRAN

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Abstract: The nuclear industry's transition toward advanced and next-generation reactors has created a growing need for a flexible, technology-inclusive licensing framework that can accommodate diverse reactor designs beyond conventional Light Water Reactors (LWRs). Nuclear Energy Institute (NEI) 18-04 presents a Technology-Inclusive, Risk-Informed, and Performance-Based (TI-RIPB) methodology developed primarily for non-LWR reactor types, capable of providing a Frequency-Consequence (F-C) curve-based analytical framework applicable to the licensing of advanced and various reactor types, including LWRs. In this context, the F-C curve can serve as a tool that integrates the radiological consequence of event sequences informed by Deterministic Safety Analysis (DSA) and the event sequence frequency quantified by Probabilistic Safety Assessment (PSA) into a single diagram, enabling Licensing Basis Event (LBE) selection and providing a quantitative basis for Structures, Systems, and Components (SSC) safety classification. Meanwhile, in the process of F-C curve generation, calculating sequence-specific frequencies and their associated consequences across numerous initiating events and event sequences can entail substantial effort, such that automation of a consistent procedure may be warranted.

In this study, PCTRAN was adopted as the physics module within a dynamic event tree analysis and automatic event sequence generation with frequency calculation for each sequence was implemented. Furthermore, leveraging the structural advantage that PCTRAN output is directly compatible with the input for RadPuff of the same developer, Micro-Simulation Technology, a foundation is established that can be extended to automated consequence calculation in follow-on research, with source term and Total Effective Dose Equivalent (TEDE)-based consequence calculation and full F-C curve generation addressed in a companion paper presented at this conference. As a suitable non-LWR thermal-hydraulic model could not be secured at the time of this study, a case study is presented using the Korean OPR1000 model of PCTRAN to demonstrate the full process from automatic event sequence generation to sequence-specific frequency calculation. The proposed automated procedure aims to demonstrate an automated workflow and a practical implementation approach for integrating PCTRAN within a dynamic event tree analysis, thereby improving the efficiency and reproducibility of analytical procedures applicable to future F–C curve generation.

Keywords: Automated PSA, PCTRAN, Frequency-Consequence Curve

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1. INTRODUCTION

In the nuclear field, Probabilistic Safety Assessment (PSA) has been widely used to generate risk information. WASH-1400 is an early representative example that combined accident probabilities, radioactive material release, meteorological conditions, and population distributions and presented the results as a Complementary Cumulative Distribution Function (CCDF) [1]. PSA is divided into Levels 1-3, which evaluate Core Damage (CD) frequency, containment response and release characteristics, and offsite radiological exposure and health effects, respectively. These consequence evaluations have also been linked to evacuation strategy and emergency response optimization studies [2, 3].

Meanwhile, PSA and licensing frameworks have been developed mainly from Light Water Reactor (LWR) experience. Therefore, for next-generation reactors, including non-LWRs, the purpose, role, and application of these frameworks need to be adjusted to reflect their design-specific characteristics. To address this, the Licensing Modernization Project (LMP) and NEI 18-04 proposed a Technology-Inclusive, Risk-Informed, and Performance-Based (TI-RIPB) approach. In this framework, the Frequency (F)-Consequence (C) Target curve, hereafter referred to as the F-C curve, integrates the occurrence frequency and radiological consequence of Licensing Basis Events (LBEs) and supports risk-significant LBE identification, Required Safety Function (RSF) derivation, Structures, Systems, and Components (SSCs) safety classification, and Defense-in-Depth (DID) evaluation [4].

However, applying the F-C curve requires the repeated evaluation of numerous event sequences, uncertainties, source terms, and consequences, resulting in significant computational and analytical burdens [5]. This raises the need for an automated procedure that can quantify both the frequency and consequence aspects required for F-C curve evaluation in a reproducible manner. As a preliminary step toward automated F-C curve generation, this study leverages the advantages of Dynamic Event Tree Analysis (DETA), whose branching concept is well aligned with the event-sequence-based nature of F-C curve evaluation. Based on this concept, a DICE–PCTRAN procedure was established to automatically generate event sequences and quantify sequence-specific plant responses and occurrence frequencies.

In this framework, DICE, as a DETA tool, is used to quantify the frequency aspect of each event sequence, while PCTRAN provides time-dependent plant responses and source-term information that can be coupled with RadPuff-based consequence assessment. PCTRAN was adopted as the physics module because it is an accessible and computationally efficient simulator that can be used without separately constructing detailed plant input data, and because consequence quantification can be facilitated through RadPuff, which was developed by the same organization. Although F-C curve-based evaluation was originally proposed as part of a technology-inclusive licensing methodology for non-LWRs and advanced reactors, this study selected the OPR1000, a relatively well-understood LWR, as the case application to first examine the feasibility of implementing the automated procedure. Through this implementation, this study aims to establish a practical basis for applying the automated DICE–PCTRAN procedure and for examining and understanding the NEI 18-04 F-C curve evaluation process. The results of this study will be coupled with RadPuff-based consequence assessment in follow-on research to support full automated F-C curve generation.

2. RESEARCH BACKGROUND

2.1. F-C Target Curve in NEI 18-04

The F-C curve relates the consequence and occurrence frequency of each event sequence. In NEI 18-04, the consequence axis is defined as the 30-day Total Effective Dose Equivalent (TEDE) at the Exclusion Area Boundary (EAB), while the frequency axis is quantified through PSA using initiating event frequencies, SSC availability, and related probabilistic information [4]. Consequence is evaluated from the Mechanistic Source Term (MST), including radionuclide inventory, transport, and release characteristics, and converted to TEDE at the EAB through atmospheric dispersion analysis. Each LBE is then assessed based on its position relative to the F-C Target shown in Figure 1.

Figure 1 F-C Target Curve and Anchor Points in NEI 18-04

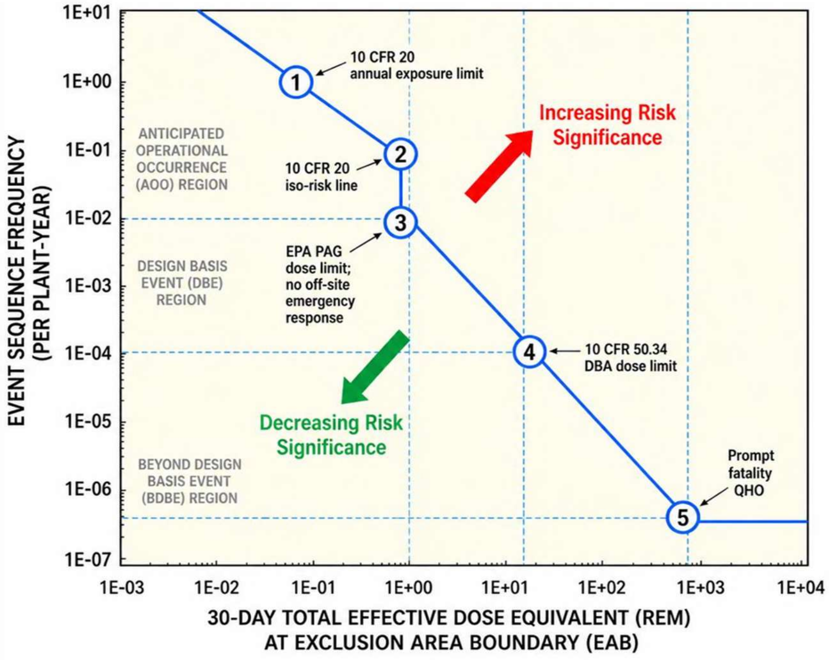


Table 1 summarizes the anchor points of the F-C Target and their corresponding regulatory bases. These anchor points define the relationship between event-sequence frequency and acceptable radiological consequence, providing the reference criteria for evaluating whether an LBE remains within the acceptable risk region.

Table 1. F–C Target Anchor Points and Regulatory Basis

Anchor	Frequency (per plant-year)	TEDE (rem)	Regulatory Basis
1	1E+0	0.1	10 CFR 20 annual exposure limit
2	1E−1	1	10 CFR 20 iso-risk line
3	1E−2	1	EPA PAG dose limit; no off-site emergency response
4	1E−4	25	10 CFR 50.34 DBA dose limit
5	5E−7	750	Prompt fatality QHO

The F-C Target is used as a criterion that links LBE classification with design decision-making. In NEI 18-04, LBEs are classified as Anticipated Operational Occurrences (AOOs), Design Basis Events (DBEs), or Beyond Design Basis Events (BDBEs) based on their occurrence frequency and consequence level. An LBE that approaches or potentially exceeds the F-C Target is evaluated as a risk-significant LBE.

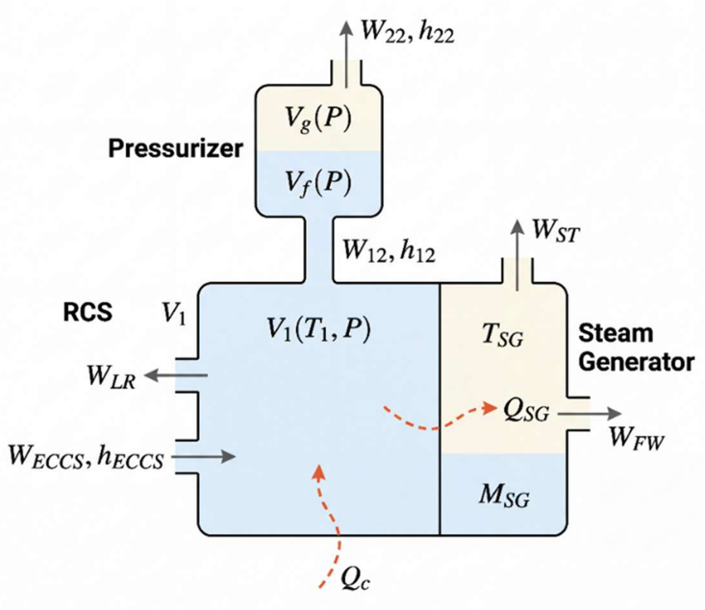
Consequently, the F-C curve in NEI 18-04 enables the risk of LBEs to be compared and evaluated on a common basis by simultaneously quantifying the likelihood of event sequences and their radiological consequences. This evaluation is directly linked to the derivation of RSFs, SSC safety classification, and DID adequacy evaluation. Therefore, the F-C curve can be regarded as a central analytical tool for translating PSA results into design and regulatory decision-making within the NEI 18-04 methodology.

2.2. Overview of PCTRAN

To generate an F-C curve, the frequency and consequence of each event sequence, that is, the (F, C) pair, are required. F is quantified by taking advantage of DICE, as described in Section 2.3, whereas C is obtained through event-sequence-specific source term and radiological dose assessment. PCTRAN is a PC-based reactor transient and accident simulator developed by Micro-Simulation Technology, and it can calculate both major Thermal-Hydraulic (TH) responses and radiological source-term evolution. Major TH variables stored in Microsoft Access database format as PCTRAN outputs (PlotData.mdb) are used for plant event sequence analysis, while the radioactive release rate (DoseData.mdb) is used for subsequent consequence assessment. PCTRAN includes simplified models for the reactor core, reactor coolant system, pressurizer, Steam Generator (SG), safety systems, fuel, containment, and radioactive release pathways. Reactor power is calculated using the Point Kinetics Equation (PKE), with reactivity feedback from control rods, moderator temperature, fuel temperature, and void effects [6].

As shown in Figure 2, PCTRAN represents the Reactor Coolant System (RCS) using a reduced-node approach, in which the pressurizer, SG, break or leakage path, and Emergency Core Cooling System (ECCS) injection path are modeled as representative control volumes and connections.

Figure 2 PCTRAN reduced-node model



By solving mass and energy balance relationships at each time step, PCTRAN evaluates key variables such as RCS pressure, coolant inventory, core level, SG heat removal, pressurizer response, fuel and cladding temperature, containment response, and radioactive release characteristics. In the source-term calculation, fission products are released according to the degree of fuel and cladding damage and transported through possible release pathways, including the containment, SG secondary side, auxiliary building, or atmosphere. The radioactive release rate is calculated based on radionuclide concentration, leakage or ventilation flow rate, filter efficiency, and removal effects.

2.3. Dynamic Event Tree Analysis using Dynamic Integrated Consequence Evaluation (DICE)

PSA has been widely used to quantify Nuclear Power Plant (NPP) event-sequence likelihood and system availability. However, when equipment actuation timing, operator actions, and TH transient responses interact during accident progression, time-dependent information must be considered. This is particularly important for advanced reactors with passive safety systems, where the boundary between system success and failure can depend on accident progression; therefore, DETA may be required [7].

Dynamic PSA (DPSA) directly couples probabilistic models with plant simulators to reflect these time-dependent interactions. DICE consists of multiple functional modules, including a Scheduler and a physics module, and performs event-sequence generation, branching decisions, system availability assessment, and frequency quantification within a single computational loop.

DICE supports both the Discrete Dynamic Event Tree (DDET) and Monte Carlo Event Tree (MCET) methods. DDET generates multiple branches at each branching point, while MCET explores the event-sequence space by sampling from probability distributions. In DDET, branches are constructed to satisfy the Mutually Exclusive and Collectively Exhaustive (MECE) principle, allowing event-sequence frequencies to be quantified with the sum of branch probabilities equal to 1 [8]. In this study, PCTRAN is coupled as the DICE physics module. By defining success criteria and branching rules in the DICE inputs, automated event-sequence analysis and frequency quantification can be performed while retaining the binary branching structure of ET.

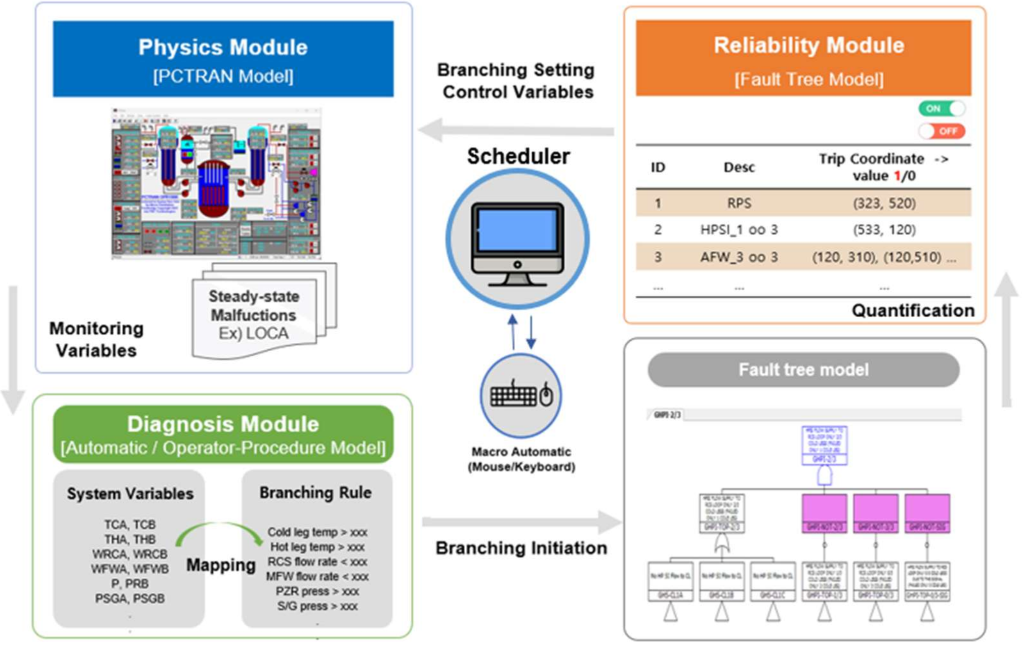
3. DICE-PCTRAN

3.1. DICE-PCTRAN Architecture

PCTRAN is a User Interface (UI)-based simulator and does not provide direct external control interfaces such as the MARS-KS DLL, MELCOR ACF, or Flownex SE API. Therefore, in the DICE-PCTRAN coupling, the physics module is controlled by using UI macros that replace the user's mouse and keyboard operations, rather than by directly controlling PCTRAN at the code level [9].

The coupled architecture is shown in Figure 3. DICE-PCTRAN consists of the Scheduler, diagnosis module, reliability module, and PCTRAN as the physics module. The DICE Scheduler sets the initiating event and simulation conditions and performs PCTRAN malfunction settings, run/pause control, and pump and valve manipulations through UI macros. The TH variables calculated by PCTRAN are then output through PlotData.mdb, and the Scheduler reads them and transfers them to the diagnosis module.

Figure 3. DICE-PCTRAN Architecture



ID	Desc	Trip Coordinate -> value 1/0
1	RPS	(323, 520)
2	HPSI_1 oo 3	(533, 120)
3	AFW_3 oo 3	(120, 310), (120,510) ...
...	...	...

The diagnosis module determines whether monitored variables such as pressure, level, temperature, flow rate, and trip signals have reached predefined branching conditions. When a branching condition is satisfied, the reliability module determines the success or failure of the corresponding safety system or operator action based on probabilistic information. The result is then transferred back to the Scheduler, which reflects the corresponding branch state in PCTRAN through UI macros. Through this information-exchange procedure, DICE-PCTRAN can perform automated event sequence generation while using PCTRAN as the physics module, even though PCTRAN has limited external interfaces.

3.2. DICE-PCTRAN Inputs

As shown in Table 2, the DICE-PCTRAN inputs used in this study consist of physics module inputs, diagnosis module inputs, and reliability module inputs, with a total of 17 input items. The physics module inputs were based on PCTRAN/OPR1000, and the diagnosis module inputs were developed with reference to the PCTRAN OPR1000 manual. The reliability module inputs were simplified rather than directly reflecting the entire plant-specific Fault Tree (FT) model of the selected plant.

Table 2. Summary of DICE–PCTRAN Input Structure

Module	Input File	Description
Physics Module	OPR 1000 PCTRAN (Basic ListData.mdb: 100% POWER EOC, Basic IC Thermo Data, built-in Isotope Data table)	Defines the plant model, initial TH conditions, initiating events, and source term inputs for PCTRAN simulation
Diagnosis Module	Rules, Rules Auto, Rules Manual, TC List, TC Status, Sim Varlist, IE List, IE Status, CV List, CV Status	Defines process-variable conditions, automatic/manual branching rules, initiating event status, and target component actions for DICE branching
Reliability Module	FM List, FM Status, Cutset Auto, Cutset Manual, Koon Auto, Koon Manual	Defines basic event probabilities, failure mode status, cutset logic, and K-out-of-N configurations for branch probability quantification

(1) Physics Module Inputs

In this study, the 100% POWER EOC condition in the “IC Thermo Data” of PCTRAN was adopted as the reference Plant Operating State (POS) for full-power internal event analysis. It represents 100% rated power at End of Cycle (EOC), with initial conditions of 2,815 MWt, approximately 15.5 MPa primary pressure, 311.2 °C average coolant temperature, 15,309 kg/s primary flow rate, and 7.8 MPa SG pressure. For radiological source-term calculation, the “IsotopeData” table in “ListData.mdb” of PCTRAN/OPR1000 was used as the default input without modification. It contains data for 52 isotopes, including fuel/core inventory, release-related coefficients, half-life, gamma energy, dose conversion factors, and initial RCS and SG secondary-side radioactivity concentrations.

(2) Diagnosis Module Inputs

The diagnosis module inputs were configured to translate PCTRAN calculation results into DICE branching conditions. “Rules” define process-variable setpoints; “Rules Auto” and “Rules Manual” define automatic safety-function actuation and manual operator actions; “TC List”/ “TC Status” specify the target components and their initial status; and “SimVarlist” specifies the PCTRAN variables read by DICE. “IE List”/ “IE Status” and “CV List”/ “CV Status” define the initiating events, control variables, activation status, frequencies, and branching status. The setpoints and actuation logic were established based on the OPR1000 PCTRAN manual. Table 3 presents representative target components manipulated in PCTRAN during branching. The configuration follows the existing DICE input structure, with additional screen coordinates defined for macro-based Scheduler operation because PCTRAN is a UI-based simulator.

Table 3. Representative Target Components in TC List

ID	TC_Index	Type	Coordinate	Description
1	HPSI_A	PUMP	(35, 186)	HPSI train A
2	HPSI_B	PUMP	(35, 216)	HPSI train B
...	...	...	...	...
19	SDS_A	VALVE	(412, 125)	Safety depressurization valve
41	MSIV_A	VALVE	(240, 228)	Main steam isolation valve A

(3) Reliability Module Inputs

The reliability module inputs were configured to calculate the success and failure probabilities of each branch. "FM List" and "FM Status" define the probability information and activation status of basic events used in the PSA FT logic. "Cutset Auto" and "Cutset Manual" define combinations of basic events that cause failures of automatic safety functions and manual actions, respectively. "Koon Auto" and "Koon Manual" define the component actuation, failure, or delay states to be reflected in PCTRAN according to each branch outcome, together with the k-out-of-n configuration.

The FT for each safety system was simplified for demonstration purposes by considering only component-level front-line systems, such as pumps, valves, and fans. The train configuration was identified based on the PCTRAN OPR1000 UI; however, detailed success criteria and train-level redundancy were not explicitly credited in the simplified branch probability model. Instead, a conservative binary success/failure criterion was applied, where the failure of any modeled train or required front-line component was treated as the failure of the corresponding safety-system branch. Accordingly, the simplified FT logic was constructed using OR logic among the modeled train and component failure events, as illustrated for the HPSI system in Figure 4. Only demand failures were considered, and the resulting cutsets were directly enumerated and entered into “Cutset Auto” and “Cutset Manual.” This simplified FT structure was used to demonstrate the DICE–PCTRAN procedure, while a full plant-specific FT model should be incorporated for actual applications.

Figure 4. Simplified HPSI FT for the DICE–PCTRAN Case Study

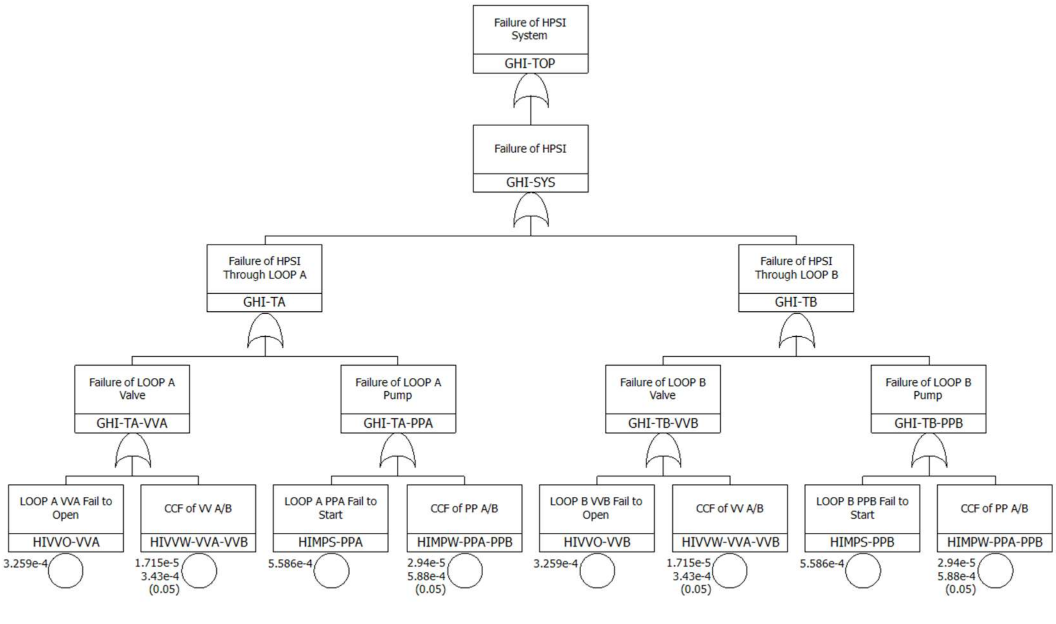


Table 4 presents selected examples of failure mode data from the "FM List" used in the reliability module. Each example includes the applied distribution or method and the 5th percentile, mean, and 95th percentile values, which serve as basic inputs for branch probability calculation. The probability values in the reliability module were treated as assumed inputs for the case study.

Table 4. Representative Failure Mode Data in FM List

ID	Name	Distribution/Method	5th	Mean	95th
1	HPSI_A_PUMP	Beta	1.04E-4	5.59E-4	1.31E-3
5	HPSI_PUMP_CCF	Alpha factor	5.45E-6	2.94E-5	6.90E-5
...	...	...	...	...	...
17	SIT_CCF2	Alpha factor	7.17E-9	1.83E-6	7.00E-6
60	SDS_A_VALVE	Beta	7.41E-5	3.25E-4	7.23E-4

Based on the mean values of the failure mode data and the directly enumerated cutsets, the mean branch probabilities used in the DDET simulation were calculated and summarized in Table 5. For automatic safety-system actions, the listed mean probabilities represent the conditional probabilities of the corresponding failure branches. The success-branch probabilities were determined by complementing the listed failure probabilities, thereby normalizing the branch probabilities at each branching point to sum to unity. For manual actions, the mean branch probabilities include failure, delayed success, and immediate success states, depending on the assumed operator action timing. The Reactor Protection System (RPS) branch was treated as a success branch in this case study, and its probability was therefore assigned as 1. These mean branch probabilities were subsequently multiplied along each event-sequence path, together with the initiating event frequency, to quantify the mean occurrence frequency of each sequence.

Table 5. Mean Branch Probabilities of Safety Systems for the Case Study

Rule	Branch	Systems	Mean Branch Probabilities
Auto_1	A_1_1	RPS SUCCESS	1
Auto_2	A_2_1	ISOL_FAIL	2.027E-03
Auto_3	A_3_1	HPSI_FAIL	1.816E-03
Auto_4	A_4_1	LPSI_FAIL	1.816E-03
Auto_5	A_5_1	AFW_FAIL	1.816E-03
Auto_6	A_6_1	CS_FAIL	1.816E-03
Auto_7	A_7_1	SIT_FAIL	1.184E-05
Auto_8	A_8_1	CAC_FAIL	9.284E-04
Manual_1	M_1_1	SDS_FAIL	6.690E-04
Manual_1	M_1_2	SDS_DELAYED	3.600E-02
Manual_1	M_1_3	SDS_SUCCESS	9.633E-01
Manual_2	M_2_1	HPSI_R_FAIL	8.856E-04
Manual_2	M_2_2	HPSI_R_DELAYED	4.600E-02
Manual_2	M_2_3	HPSI_R_SUCCESS	9.531E-01

4. CASE STUDY

The case study was conducted by applying the DDET method based on the DICE and OPR1000 input structures described in Section 3.2. Hot-leg Loss of Coolant Accident (LOCA) was selected as the initiating event. The initiating event frequency was assumed to follow a Gamma distribution, and the branch probability cutoff was set to 1.0×10^{-10}. The initiating event frequency of Hot-leg LOCA was set to 1.54×10^{-4}/rx-yr. The automatic safety system branches considered in this case study included

High-Pressure Safety Injection (HPSI), Low-Pressure Safety Injection (LPSI), Containment Spray (CS), isolation (ISOL), Auxiliary Feedwater (AFW), Safety Depressurization System (SDS), Containment Air Cooling (CAC), and Safety Injection Tank (SIT). For operator action branches, Feed-and-Bleed (F&B) operation and HPSI recirculation were considered, with branch states defined as failure, delayed success, and immediate success. The delayed-success branch was assumed to represent operator action completed after the corresponding demand.

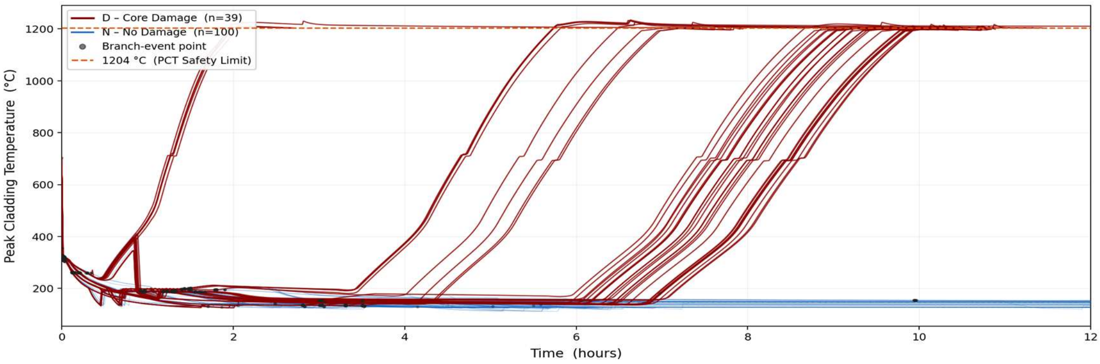
To reduce the computational burden, the effective capacity of each PCTRAN safety system was set to 50%, and the mission time was limited to 12 h. Therefore, the end of the sequence states and the corresponding consequence value C evaluated using RadPuff may depend on these analysis assumptions. PCTRAN was run at 16× speed, and the UI-based Scheduler sequentially executed, stored, and extracted the sequences. The total runtime was approximately 254 h, or 10.6 days. The generated sequences were classified as Normal (N), Core Damage (D), Cutoff (C), Unknown (U), or Branch Point (R). Here, C indicates sequences terminated by the probability cutoff, U indicates uncertain cases caused by thermal-hydraulic code or UI-macro errors, and R indicates branch points used for branch restart. Table 6 summarizes the number of sequences and total frequency for each end state. Because a high-probability successful sequence was classified as U and excluded, the total frequency distribution may not be fully represented; therefore, caution is required in interpreting the results.

Table 6. Number of Sequences and Frequencies for Case Study

End State (Tag)	Number of Sequences	Sum (5th)	Sum (Mean)	Sum (95th)	Description
Normal (N)	100	1.689×10^{-7}	3.122×10^{-7}	5.475×10^{-7}	< 1,477 K at 12 h
Core Damage (D)	39	5.081×10^{-11}	7.950×10^{-10}	3.923×10^{-9}	>1,477 K at 12 h
Cutoff (C)	171	N/A	N/A	N/A	$< 1.0\times10^{-10}$
Unknown (U)	6	N/A	N/A	N/A	TH code or UI-macro error
Branch Point (R)	447	N/A	N/A	N/A	Restart

Figure 5 compares representative TH responses of the N and D sequences. HPSI failure was identified as the dominant contributor to CD. In the D sequences, core level decreased with increasing Peak Clad Temperature (PCT) and Clad Failure Fraction (FRCL), whereas the N sequences maintained or recovered core level with low PCT and FRCL. The gray markers indicate the branch occurrence times.

Figure 5. Thermal-Hydraulic Response for Case Study: PCT

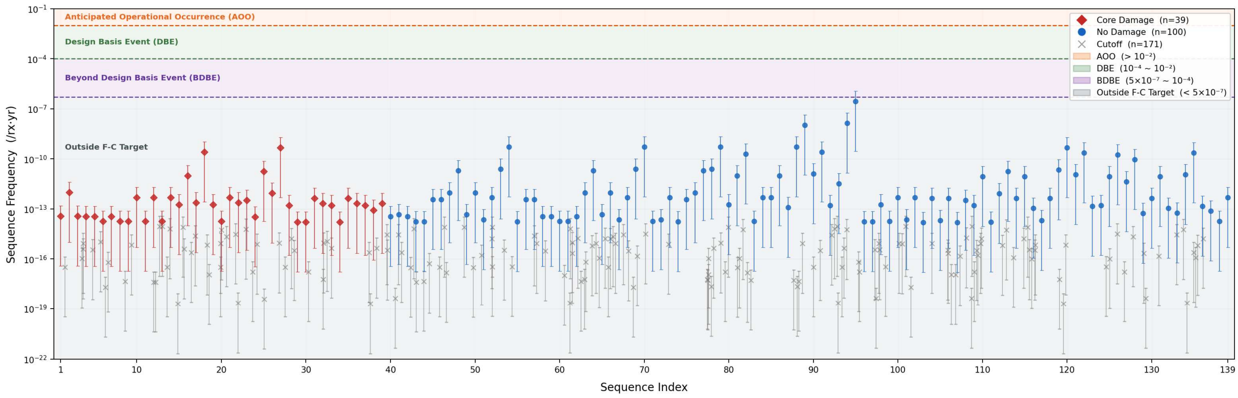


Among the four radiation monitors examined in the case study, the Containment Air Monitor (RM1) and the Primary Coolant Leakage Pathway Monitor (RM2) exhibited clearly distinguishable trends between the two sequence groups: both remained near background levels in the N sequences but increased sharply in the D sequences after CD. The SG Area Monitor (RM3) and the Auxiliary Building Air Monitor (RM4) remained near background levels throughout all sequences.

Among the three modeled release pathways, the Containment Leakage Pathway (STRB) dominated the radioactive material release characteristics. STRB release remained low in the N sequences but increased significantly in the D sequences after CD, mainly due to increased coolant radioactivity and changes in containment conditions. In contrast, the SG Pathway (STSG) and the Turbine Building/Condenser Off-Gas Pathway (STTB) remained negligible because SG tube damage and major secondary-side release pathways were not considered in this analysis. Therefore, the external source term in this case study is dominated by the containment release pathway.

Figure 6 presents the occurrence-frequency results for the 139 sequences with determined end states and the 171 sequences pruned by the cutoff condition in this case study. The frequency of each sequence was calculated by sequentially multiplying the initiating event frequency by the success or failure probabilities of the safety system branches encountered along the corresponding sequence path. Each point represents the mean occurrence frequency of an individual sequence, while the vertical error bars indicate the frequency range resulting from uncertainties in the initiating event frequency and branch probabilities. Sequences classified as U were excluded from the frequency calculation because their end states could not be reliably determined due to TH code or UI-macro errors.

Figure 6. Sequence Frequency for Case Study



Sequence Index 17 (D) and 101 (N) were selected from Figure 6 to illustrate the sequence frequency calculation. Sequence Index 17 consists of RPS success (A_1_1, P = 1.0), ISOL success (A_2_2, P = 0.9980), HPSI auto-injection failure (A_3_1, P = 1.816×10^{-3}), LPSI success (A_4_2, P = 0.9982), SDS immediate operator action success (M_1_3, P = 0.9633), CS success (A_6_2, P = 0.9982), CAC failure (A_8_1, P = 9.284×10^{-4}), SIT success (A_7_2, P = 0.99999), and HPSI recirculation hardware failure (M_2_1, P = 8.856×10^{-4}). By multiplying these branch probabilities by the initiating event frequency, IE = 1.545×10^{-4}/reactor-yr, the resulting sequence frequency is calculated as 2.21×10^{-13}/reactor-yr. In contrast, Sequence Index 101 consists of RPS success (A_1_1, P = 1.0), ISOL success (A_2_2, P = 0.9980), HPSI auto-injection success (A_3_2, P = 0.9982), LPSI success (A_4_2, P = 0.9982), SDS hardware failure (M_1_1, P = 6.690×10^{-4}), CS failure (A_6_1, P = 1.816×10^{-3}), CAC failure (A_8_1, P = 9.284×10^{-4}), SIT success (A_7_2, P = 0.99999), and HPSI recirculation immediate operator action success (M_2_3, P = 0.9531). The resulting sequence frequency is calculated as 1.65×10^{-13}/reactor-yr.

The sequence frequencies in this case study were calculated under several simplifying assumptions. In this preliminary case study, the mission time was limited to 12 h and the effective capacity of each PCTRAN safety system was set to 50% to demonstrate a diverse set of sequences and end states. In addition, the reliability inputs were based on assumed branch probabilities derived from simplified FT structures, rather than on a detailed plant-specific FT model. Furthermore, not all elements required for a complete DETA-based risk assessment, such as detailed train-level dependencies, recovery actions, and full uncertainty propagation, were explicitly considered in this preliminary case study. These assumptions may also affect the source-term results and the subsequent RadPuff-based consequence quantification in follow-on work.

5. CONCLUSION

As a preliminary step toward automated NEI 18-04 F-C curve-based evaluation, this study proposed a DICE–PCTRAN/OPR1000 procedure for quantifying event-sequence-specific plant responses, source terms, and occurrence frequencies. In the Hot-leg LOCA case study, the DDET method was applied to take advantage of its capability for simulating safety system actuation, RCS depressurization, core cooling, containment response, and source-term evolution. The occurrence frequency of each event sequence was quantified by combining the initiating event frequency with branch probabilities, demonstrating the feasibility of automatically generating the frequency axis of the F-C curve. The sequence-specific source terms and DoseData.mdb generated in this study will be used with RadPuff-based consequence assessment in follow-on work to complete F-C curve generation. Although the case study was performed using the LWR model due to the limited availability of a suitable non-LWR TH model, the proposed procedure is expected to be applicable to future analyses of non-LWRs and next-generation reactors.

The proposed methodology still has several limitations. Repeated calculations for numerous branches can cause a significant computational burden as the number of initiating events and event sequences increases. Therefore, future work should incorporate high-performance computing techniques, such as parallel computing, and define appropriate branch pruning criteria by considering the analysis objective, initiating event characteristics, and potential consequence significance. Furthermore, because DICE–PCTRAN coupling relies on UI-based macros, stable execution conditions are required for long-duration simulations. In addition, the reliability module was based on a simplified FT structure; therefore, detailed logic and inter-system dependencies should be incorporated for plant-specific applications. Although PCTRAN is effective for rapid event sequence simulation and source-term extraction, it may have limitations in modeling severe accident phenomena compared with severe accident analysis codes such as MELCOR or MAAP.

Future applications should extend the scope beyond Hot-leg LOCA to SG Tube Rupture (SGTR) and other transient initiating event groups. This extension would enable evaluation of not only containment release pathways but also source-term pathways through the SG, turbine building, and auxiliary building. The proposed procedure is expected to provide a basis for automated F-C curve generation to support LBE identification, risk-significant sequence screening, SSC safety classification, and DID evaluation for non-LWRs and next-generation reactors.

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