Analysis of CEA Withdrawal Accident Using BEPU Approach

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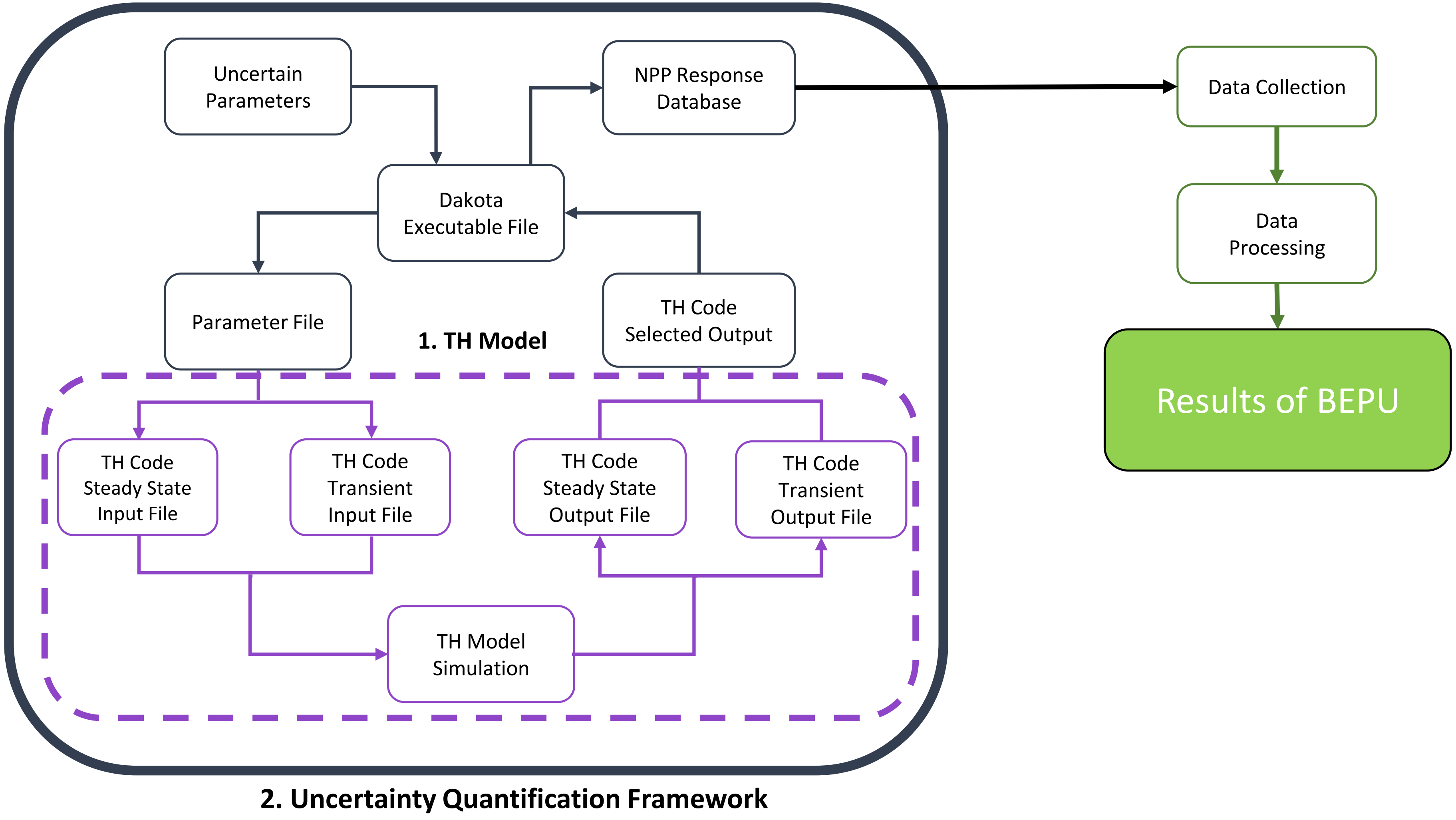
Introduction

- Uncontrolled Control Element Assembly (CEA) Withdrawal – Reactivity Initiated Accident (RIA) – is analyzed for APR-1400 nuclear power plant (NPP) system response using the Best Estimate Plus Uncertainty (BEPU) approach.
- A thermal-hydraulics model of APR-1400 is developed via one-way coupling with point kinetics model using MARS-KS.
- Uncertainty quantification is conducted by coupling MARS-KS with DAKOTA using a Python interface.
- Results of the BEPU analysis are intended to reflect the realistic system response which provides for better economy and more operational flexibility compared to the conservative approach.

Methodology



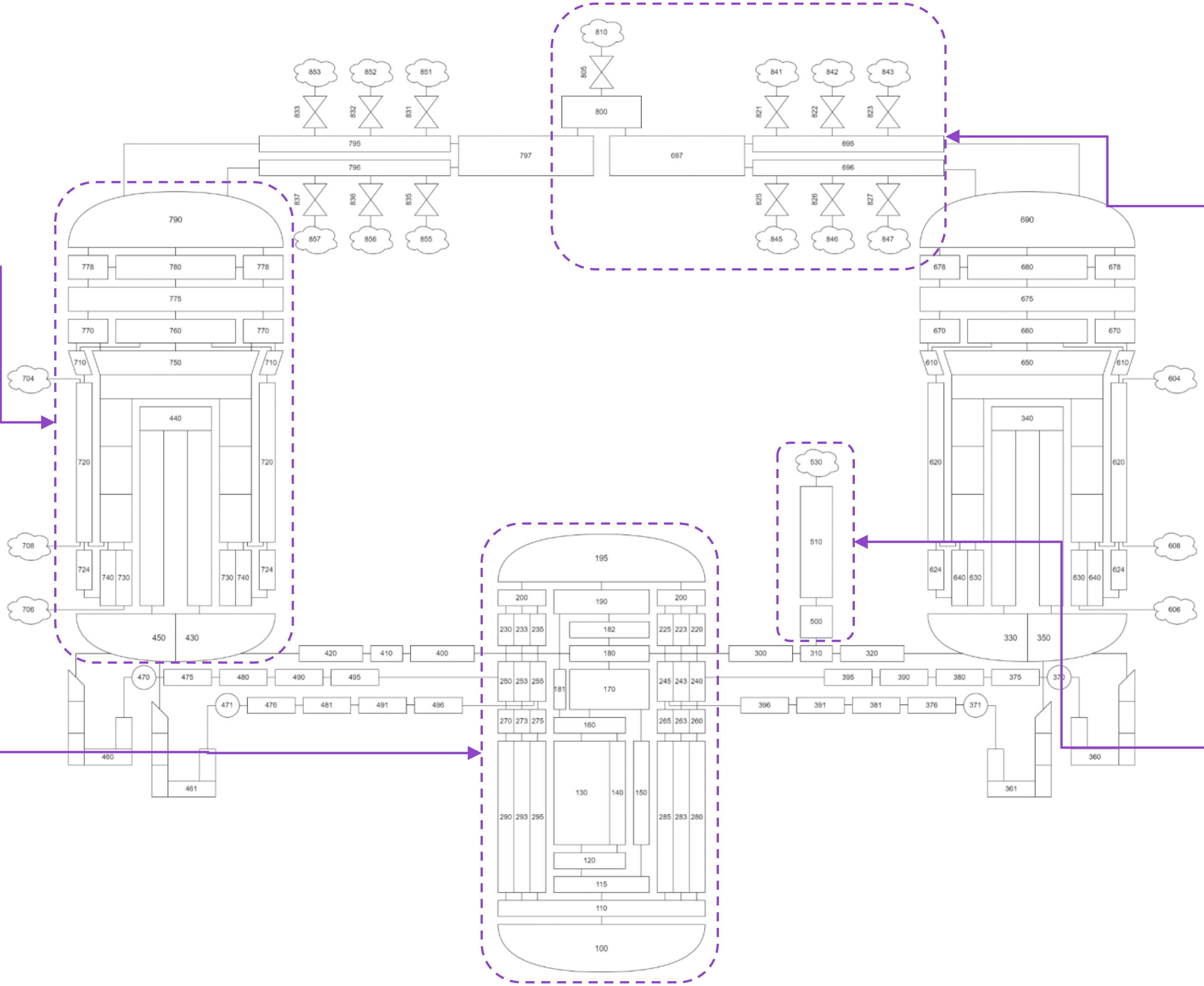
TH

Thermal-Hydraulic Model

APR-1400 Nodalization

Steam Generators (SGs)

- Two SGs - each connected to the RPV via one hot leg and two cold leg
- Heat generated on the primary side is transferred to the SGs via the u-tubes
- The u-tube section is modeled with equivalent heat transfer and pressure drop conditions
- Secondary water is provided by the Main Feedwater System (MFWS) as boundary condition
- Steam generated in the SGs is directed via the main steam line to the turbine modeled as a boundary condition
- Other important components of the SGs are: evaporator, separator, dryer, dome



Main Steam System (MSS)

- The Main Steam System (MSS) has four main steam lines leading from the two SGs to a common header, and then to the turbine through an isolation valve.
- Each line is connected to a set of Main Steam Safety Valves (MSSVs) to protect the system against over-pressurization.

Reactor Pressure Vessel (RPV)

- The core is represented using an average and a hot channel, surrounded by an annular core shroud together with the core bypass
- The core connects to an upper plenum and a lower plenum
- Two hot legs lead the coolant from the RPV to the SGs u-tubes, four cold legs connect the RCPs to the downcomer
- The downcomer is modeled using annulus six components

Pressurizer (PZR)

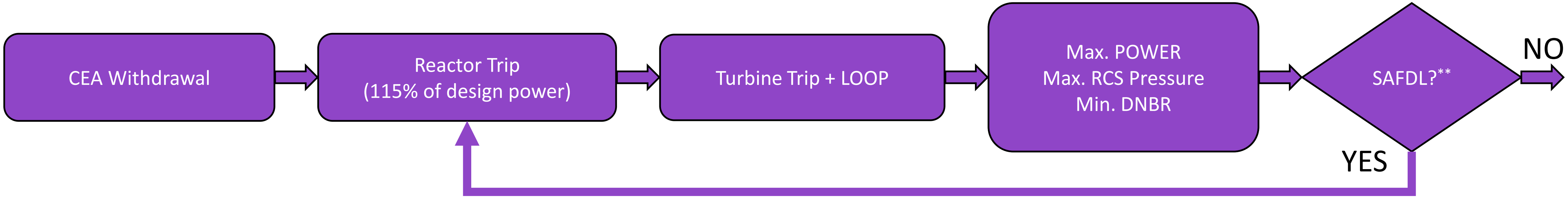
- Maintains operational pressure in the primary system loop.
- In steady-state, the pressurizer pressure is imposed by a boundary condition. In transient, the pressure is determined by the system conditions and Pilot-Operated Safety Relief Valves (POSRVs) operation.

Thermal-Hydraulic Model

Accident Description*

- An uncontrolled CEA withdrawal at full power is assumed to result from a single failure in the digital rod control system (DRCS), reactor regulating system (RRS), or an operator error. Such an event with a concurrent loss of offsite power (LOOP) is considered to be the most limiting case. The event is classified as an anticipated operational occurrence (AOO).

- Sequence of events for the analyzed accident is as follows:



- Initial conditions are set in a conservative manner. It means that the core inlet temperature, pressurizer pressure, core mass flow, and radial peaking factor are tuned to make the reactor to operate at a power operating limit (POL) when the transient is initiated.

*APR-1400 Design Control Document, Tier 2, Chapter 15, "Transient and Accident Analyses" (APR1400-K-X-FS-14002-NP, Revision 3, August 2018)

**Specified Acceptable Fuel Design Limit

Thermal-Hydraulic Model

Initial Conditions

- The CEA withdrawal at full power accident scenario is analyzed from the two following perspectives:

a. First, the initial parameters are set to drive the system to reach the lowest possible value of MDNBR in order to challenge the system thermal margin

Parameter	Model	DCD	Deviation
Core power, MWt	4062.66	4062.66	0.00%
Core inlet coolant temperature, °C	287.74	287.8	0.02%
Core mass flow rate, 10^6 kg/hr	69.64	69.64	0.00%
Pressurizer pressure, kg/cm^2	163.5	163.5	0.00%
Integrated radial peaking factor	1.49	1.49	0.00%
Initial core minimum DNBR	1.839	1.72	6.85%
Steam generator pressure, kg/cm^2	68.262	68.26	0.00%

b. Second, the initial parameters a modified to maximize value of peak RCS pressure at the most vulnerable location to test the system barrier performance

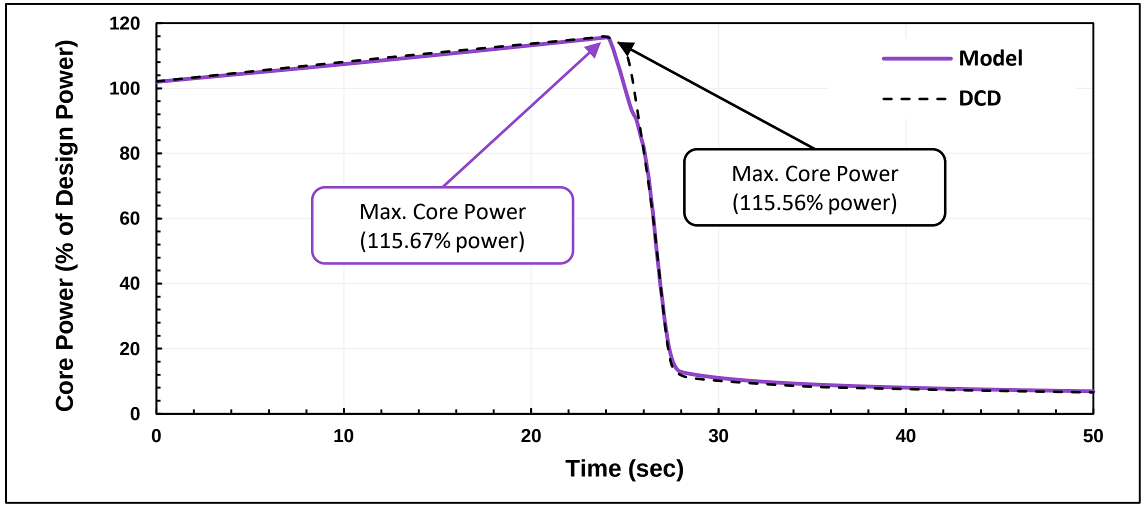
Parameter*	Model	DCD	Deviation
Core inlet coolant temperature, °C	294.7	295.0	0.00%
Steam generator pressure, kg/cm^2	75.85	75.86	0.00%

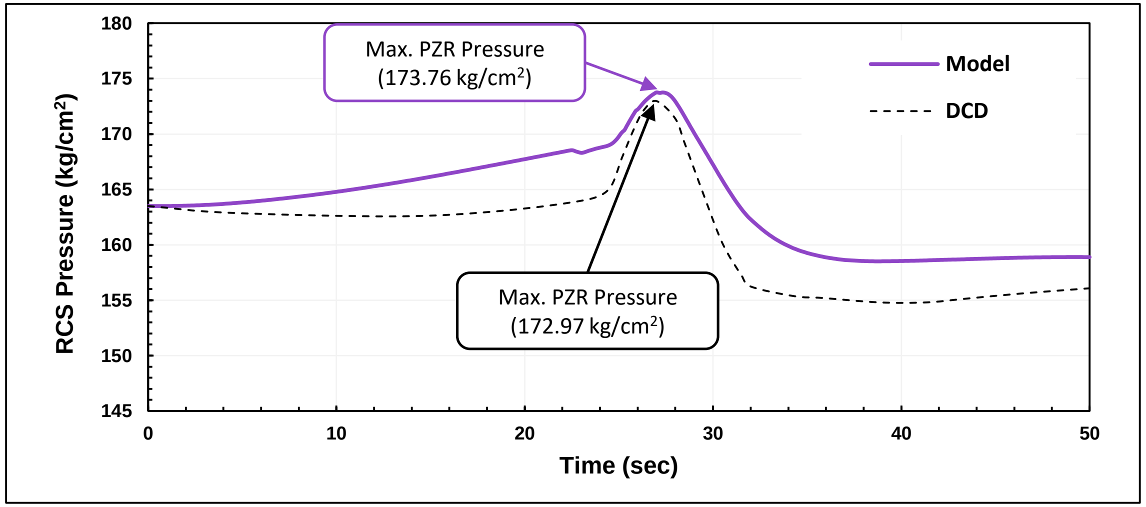
*Only the modified parameters are mentioned in the table. All the remaining parameters have kept their original values

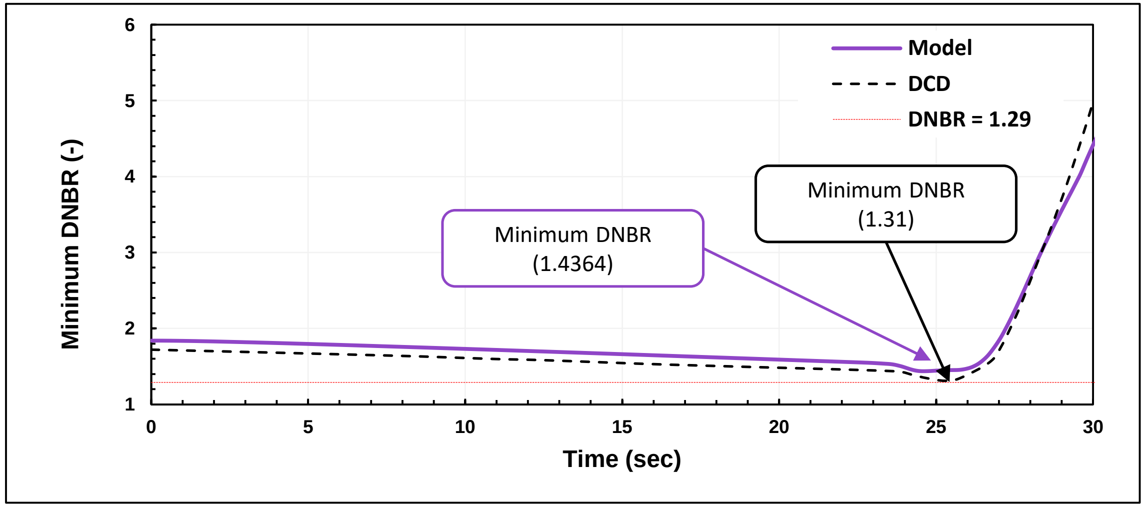
Thermal-Hydraulic Model

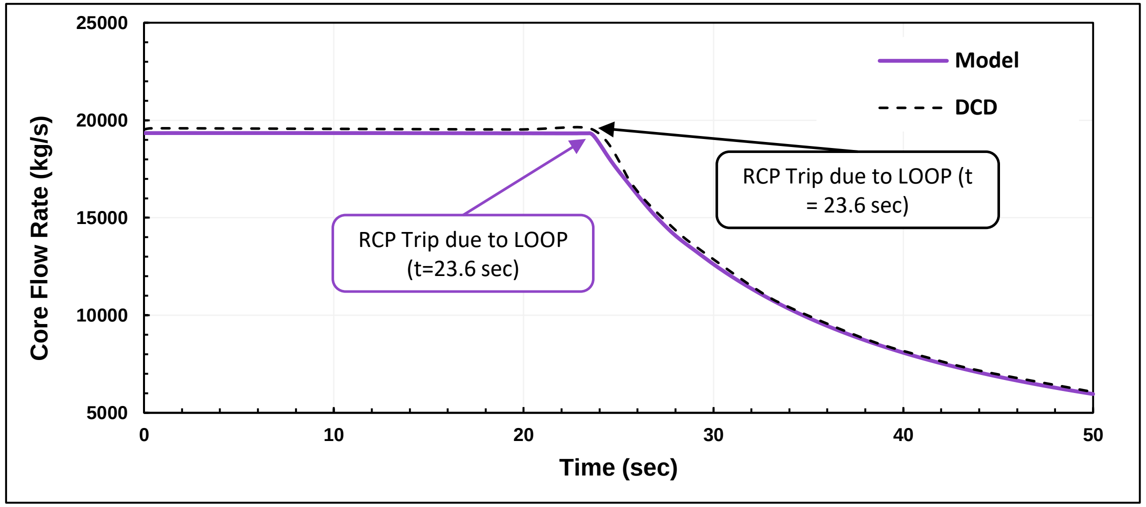
TH

NPP Response for MDNBR Case



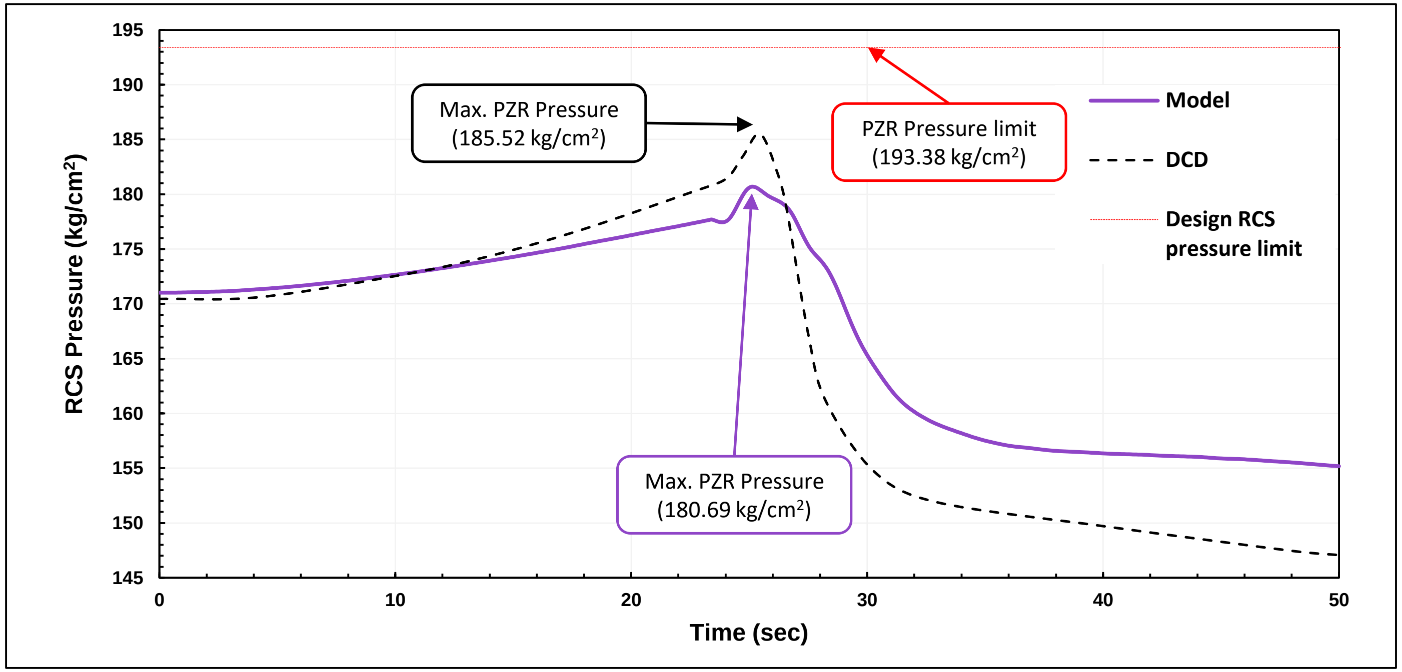






Thermal-Hydraulic Model

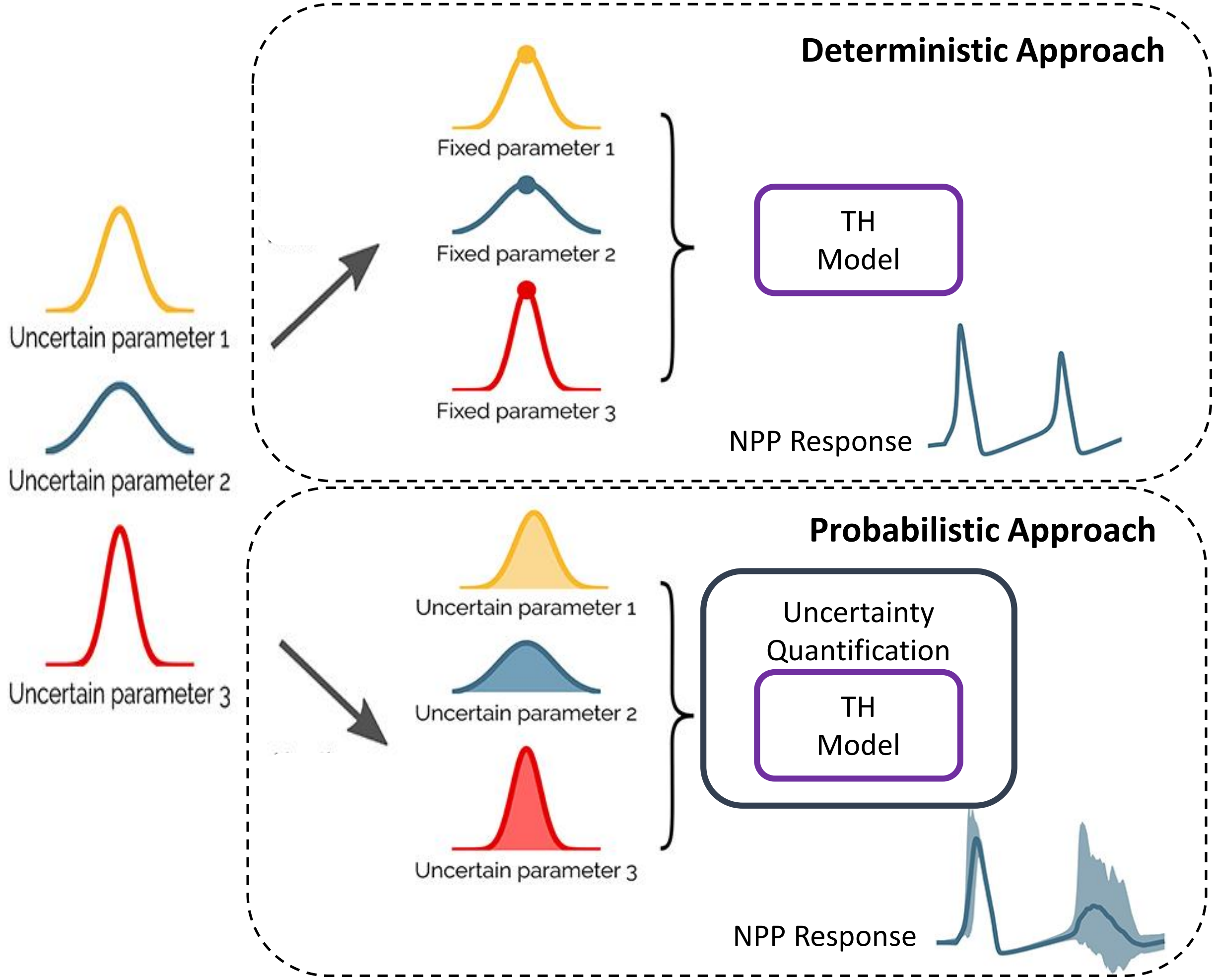
NPP Response for Peak RCS Pressure Case



○ Based on the validation against DCD conservative assumptions, it was concluded that the developed TH model provides accurate system response. Accordingly, the model was adjusted to the nominal conditions and perturbed using the statistical details of the uncertain parameter within the framework for uncertainty quantification.

Uncertainty Quantification*

Identify Key Phenomena (PIRT) → Derive Key Uncertain Parameters (UP) → Propagate the Uncertainty



Key Phenomena (PIRT)	Input Uncertain Parameters	Statistical Information				
		Mean	σ	PDF	Lower bound	Upper bound
Fuel rod manufacturing tolerances	Cladding outside diameter (mm)	4.7298	0.01	Normal	4.7156	4.7345
	Cladding inside diameter (mm)	4.1556	0.01	Normal	4.1390	4.1598
	Fuel theoretical density (kg/m³ at 20°C)	10970	50	Normal	10870	11070
	Fuel porosity %	4	0.5	Normal	3	5
	Cladding roughness (μm)	0.1	1	Normal	10^{-6}	2
	Fuel roughness (μm)	0.1	1	Normal	10^{-6}	2
	Filling gas pressure (MPa)	5.6911	0.05	Normal	5.6626	5.8277
Thermal hydraulic conditions	Coolant pressure (MPa)	15.550	0.075	Normal	15.390	15.710
	Coolant inlet temperature (°C)	290	1.5	Normal	289	292
	Coolant velocity (m/s)	4.00	0.04	Normal	3.92	4.08
Core power conditions	Injected energy in the rod (Joule)	300000	1500	Normal	27000	33000
Thermo-physical properties/key heat transfer models	Fuel thermal conductivity model (multiple coefficient)	1.00	5%	Normal	0.90	1.10
	Clad thermal conductivity model (multiple coefficient)	1.00	5%	Normal	0.90	1.10
	Fuel enthalpy/heat capacity (multiple coefficient)	1.00	1.5%	Normal	0.97	1.03
	Clad-to-coolant heat transfer (multiple applied for all flow regimes) coefficient-Same coefficient	1.00	12.5%	Normal	0.75	1.25

*Marchand, Olivier, Jinzhao Zhang, and Marco Cherubini. "Uncertainty and sensitivity analysis in reactivity-initiated accident fuel modeling: synthesis of organization for economic co-operation and development (OECD)/nuclear energy agency (NEA) benchmark on reactivity-initiated accident codes phase-II." Nuclear Engineering and Technology 50.2 (2018): 280-291.

Uncertainty Quantification*

Wilks' Theorem

- Wilks' one-sided formula deriving from the Wilks' theorem was applied to determine the minimum amount of computational work required to provide sufficiently large sample for uncertainty analysis.

$$1-\sum_{k=n-p+1}^{n} {}_{n}C_{k}\,\alpha^{k}(1-\alpha)^{n-k} \geq \beta$$

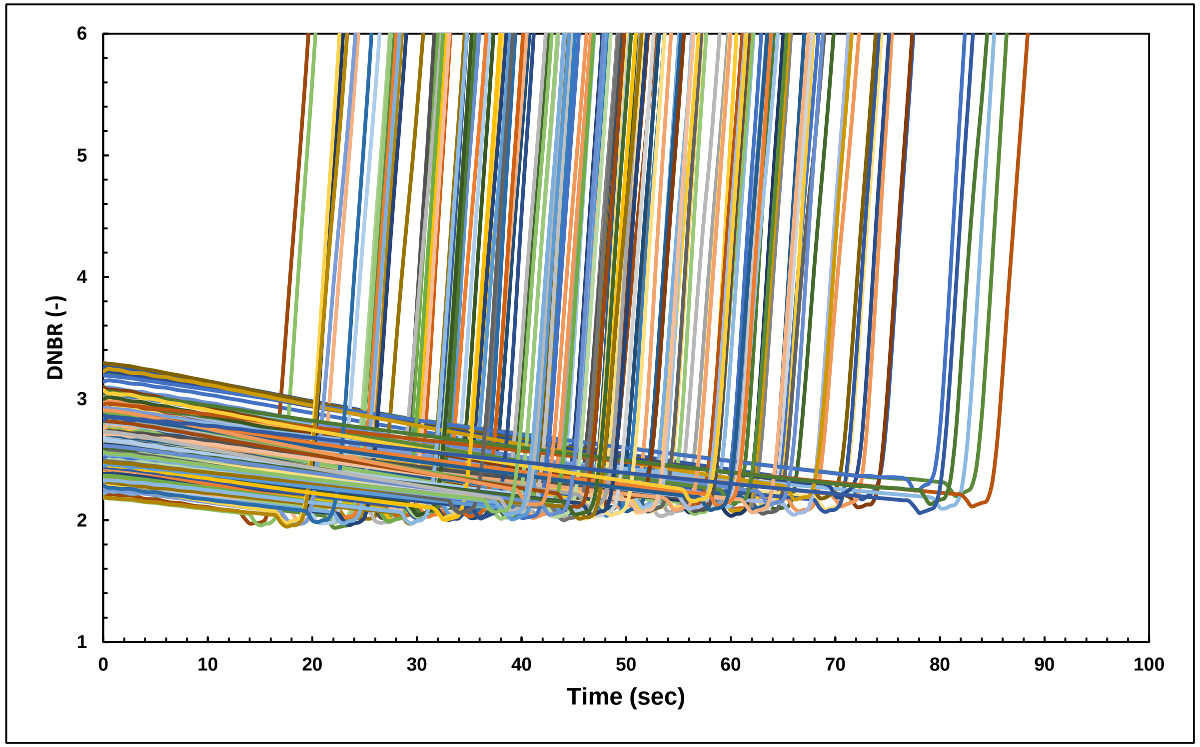
α – tolerance limit ($\alpha = 0.95$)
β – confidence level ($\beta = 0.95$)
n – sample size (number of simulations)
p – order ($p = 5$)

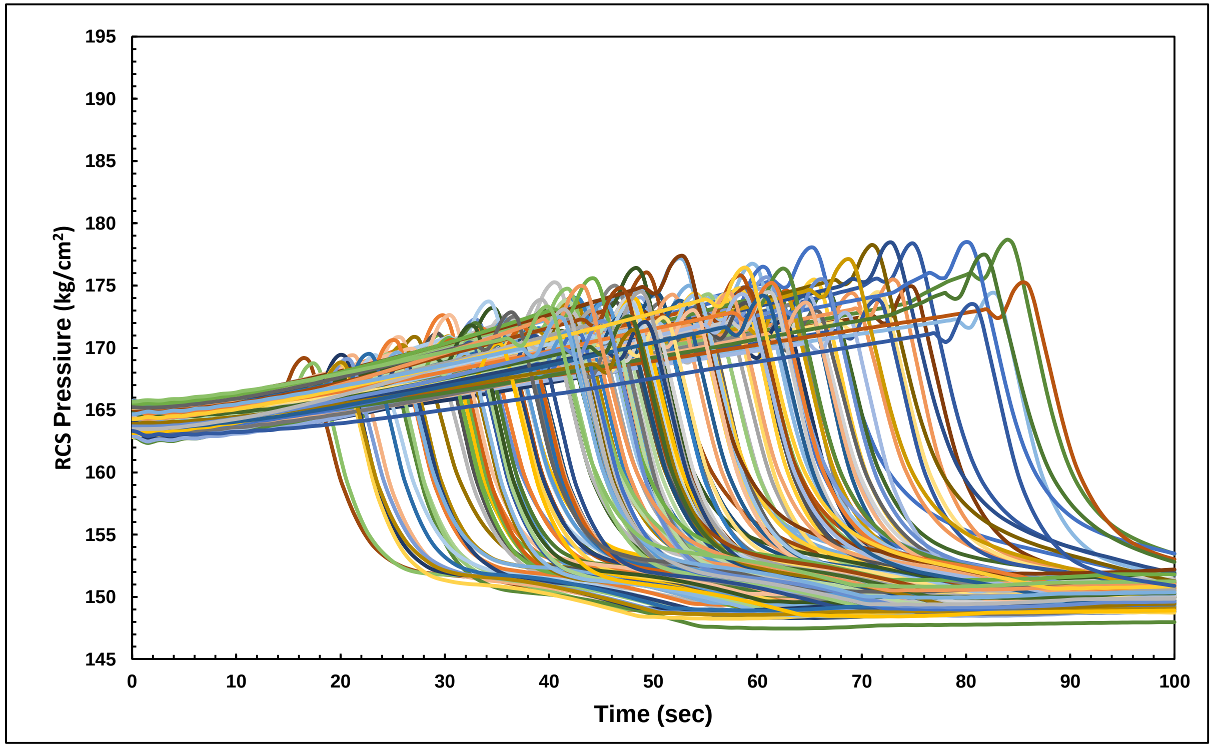
- Fifth order Wilks formula ($p = 5$) was used to ensure results credibility. Therefore, following the formula, the number of required simulation runs has been determined to be equal to $n = 181$.

- Since the order of the formula applied is $p = 5$, the fifth minimum and maximum value curves from among all the runs were selected as relevant for MDNBR and peak RCS pressure, respectively.

Uncertainty Quantification

Results



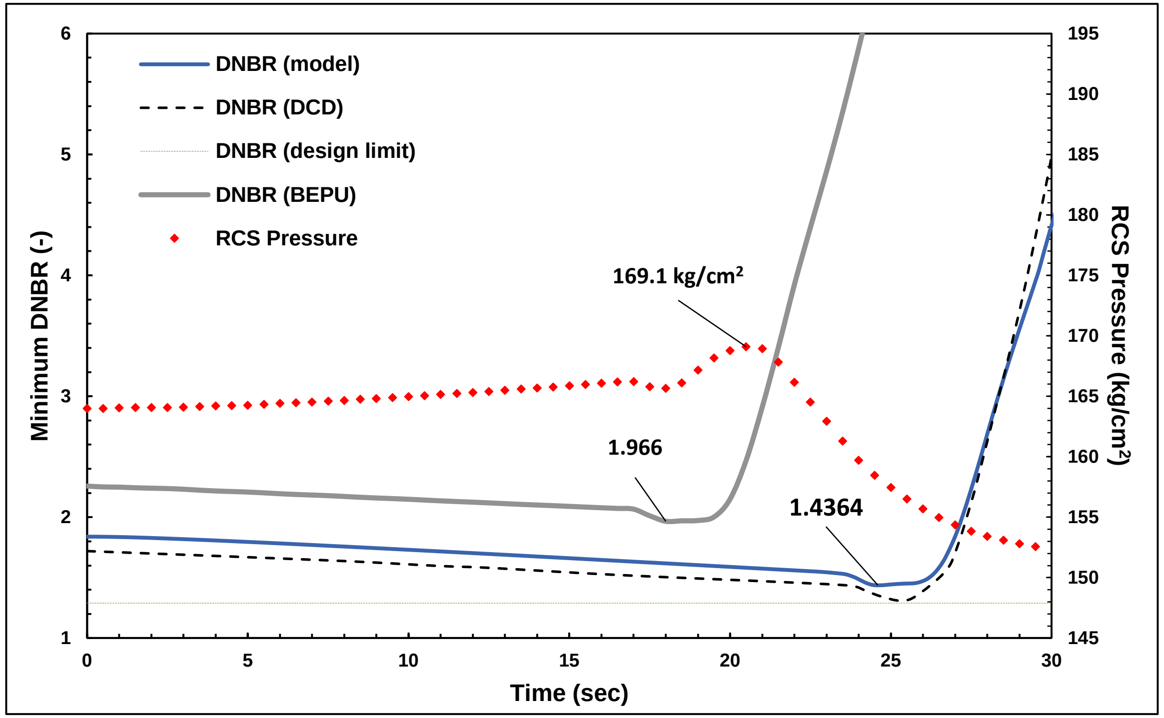


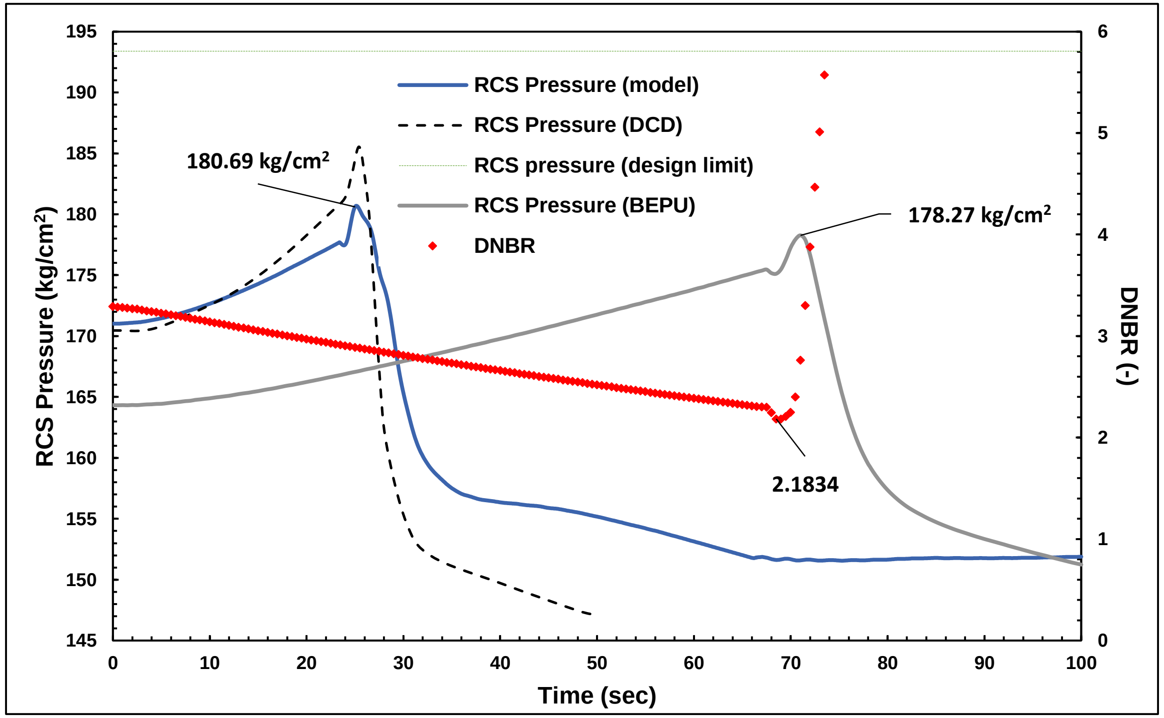
- The above graphs depict results generated from 181 runs (Wilks' 5th order) for MDNBR and peak RCS pressure cases.
- For each run a set of uncertain parameter values was randomly selected by the DAKOTA software within the UQ framework.

Uncertainty Quantification

UQ

Results





- From among the results generated in the UQ framework, the most probable system response was selected using the 95% confidence and 95% probability criterion.
- The final result is compared to the result achieved in conservative analysis. The increased safety margin notably enhances the system operational flexibility.

Conclusions

- In this work, CEA withdrawal at full power accident scenario was investigated using the best-estimate-plus-uncertainty (BEPU) approach.
- Starting with the phenomena identification and ranking table (PIRT) for reactivity initiated accidents (RIAs), key uncertain parameter are identified and propagated using the non-parametric Monte Carlo approach for random propagation of uncertain parameters; specifically Wilks' fifth order statistics along with the Latin Hypercube Sampling (LHS) technique.
- An uncertainty quantification framework was developed to assess the NPP response under different initial, boundary, operating conditions, as well as thermo-physical properties, and manufacturing tolerances.
- The NPP response was successfully predicted using one-way coupling approach. More realistic results are expected with two-way coupling between RELAP and 3DKin which is currently being developed*.

*Jan Hruškovič and Aya Diab, Multiphysics Analysis of CEA Withdrawal at Power for the Korean APR1400 Reactor, KNS Autumn Meeting, 2022.

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