Analysis of a Postulated Main Steam Line Break Accident using a Multi-Physics Simulation

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1. Introduction

This paper aims to analyze the postulated main steam line break (MSLB) accident of the APR1400. The accident is analyzed in steps, first is thermal-hydraulic modeling with RELAP5/SCDAPSIM/MOD3.4 and a point kinetics model to simulate core neutronics, secondly, 3DKIN (3D Neutron Kinetics/Thermal-hydraulic code) is used to replace the one-dimensional point kinetics model with a three-dimensional representation of the core for real time reactivity feedback.

MSLB is a postulated accident where a double ended pipe break occurs on one of two main steam lines, the steam generator towards the turbine. The break creates loss of secondary flow inventory, immediate depressurization of the steam generator secondary side, and results in excessive reactor coolant system (RCS) cooldown, this in turn increases core reactivity due to negative moderator temperature reactivity coefficient. MSLB accident analyses are grouped into two categories, cases that maximize post trip return to power (RTP) and those that maximize pre-trip degradation in fuel performance and radiation doses [1]. The analysis chosen for this project is the MSLB that maximizes the potential for post trip RTP Case 1). This case occurs inside containment, at full power operation with loss of offsite power (LOOP) concurrent with the initiation of the break, with a combination of single failure and a stuck control element assembly (CEA).

MSLB is associated with notable space-time effects in the core, which occurs because of asymmetric cooling [2]. The steam line break occurs in one of the two loops, the affected loop, which accordingly experiences rapid decrease in temperature and pressure compared to the other loop. This induces three dimensional affects in association with the asymmetric cooling of the core. By definition, a point kinetics model is incapable of representing such asymmetry due to the inherent averaging process which masks key three dimensional phenomena. It is therefore common practice in safety analysis to simulate such accidents with three-dimensional analysis codes. Codes such as RELAP5/SCDAPSIM/MOD 3.4 [3] and MARS-KS [4] are typically used in simulating thermal-hydraulic behavior of the plant one dimension, however this accident requires a high fidelity nuclear system code to model the behavior of the reactor core, control element assemblies, fuel assemblies and other reactor internals, as these affect accident progression [5]. Furthermore, temperature changes in the core result in variations in the distribution of reactivity feedback mechanisms due to variations in moderation and absorption cross sections within the core. Monitoring of these spatial variations is only possible via a Multiphysics simulation.

2. Methodology

Two models are explored in this study, first is the thermal-hydraulic (TH) model using RELAP5/SCDAPSIM/MOD3.4 [3] developed by Innovative Systems Software (ISS). Second is the Neutron Kinetics (NK) Model in 3DKIN based on NESTLE (nodal eigenvalue, steady-state, transient, le core evaluator) code [6] which was developed at North Carolina State University.

2.1. Thermal-hydraulic model using RELAP5/SCDAP/MOD3.4

APR1400 is a PWR with two closed loops. Each loop comprises of one hot leg, a steam generator, two cold legs and two reactor coolant pumps. The loops are connected in parallel to the reactor pressure vessel. The pressurizer is connected to one of the loops. The core is modelled with inlet and outlet nozzles, downcomer, and lower and upper plenums as part of the reactor vessel. The reactor core is represented using 10 volumes, as seen in Figure 1 and Figure 2. On the primary side of the plant, the reactor coolant is introduced into the reactor vessel with a downward flow between the shell of the reactor vessel and the core barrel, then flows upwards into the reactor core, and exists the core through hot legs into the tube side of the steam generators, where heat is transferred to the secondary system. After exiting the tubes of the steam generator, the flow returns to the suction side of the RCPs and then to the cold legs of the reactor vessel in the same manner as it was introduced.

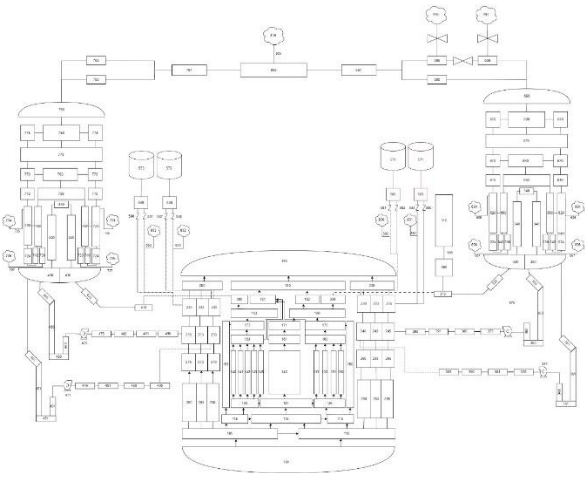


Figure 1: APR1400 Nodalization

On the secondary side, the heat that was transferred to the secondary side reactor coolant through the tubes of the steam generator, produces steam to drive the turbine-generator set. The steam of the secondary side is at saturation quality. The moisture content of the steam is controlled by moisture separators and dryers of the steam generator. Each main steam line is equipped with an integral flow restrictor in the event of a steam line break.

2.2. Nodal kinetics model using 3DKIN

To simulate the interactions and feedback between the thermal hydraulic phenomena during accident progression, it is common practice to couple TH and NK codes, where 3D neutronics codes accurately model the moderator conditions such as boron, density and fuel conditions, [5] and TH codes model the response of the NSSS, which is affected by the feedback from core neutronics.

The core neutronics is modelled with 3DKIN with 10 radial nodes as illustrated in Figure 2. There are 241 fuel assemblies in the APR1400 reactor, each with a 17 x17 Fuel Assembly (FA) array. The core was loaded with 241 FA according to the loading pattern in Figure 3.

The core region is divided into 10 volumes, categorized into four zones for outer, inner, central and the stuck control rod, and communicating via crossflows as shown in Figure 2. Surrounding the core region is the reflector region. Axially the core has 20 nodes with 2 for top and bottom reflectors.

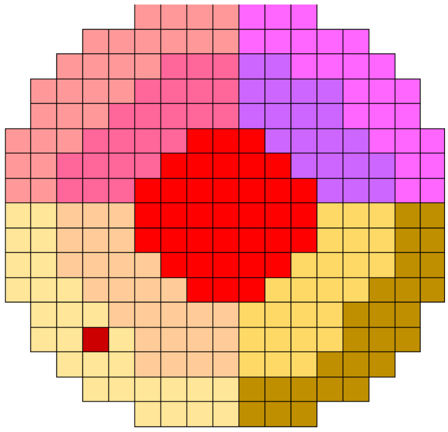


Figure 2: Core Radial Map

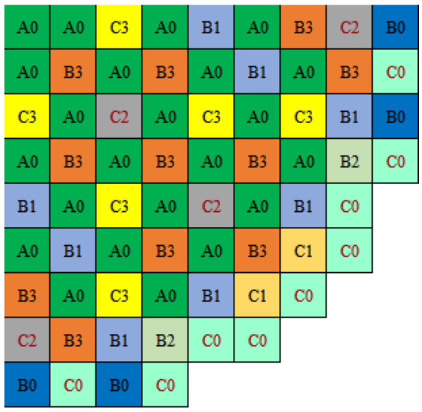


Figure 3: Fuel Loading Pattern for a Quarter Core

Nine different types of FAs exist in the first cycle loading pattern according to their enrichments level (ranging from 1.71 to 3.64% by weight). Additionally, some FAs contain Gadolinium oxide as burnable absorber with enrichment of 8% as shown in Table 1 and Figure 4.

Table 1: Fuel Assembly Data

FA Type	No. of Fuel Assemblies	Fuel Rod Enrichment (w/o)	No. of rods per assembly	No. of Gd_2O_3 rods per assembly	Gd2O3 enrichment (w/o)
A0	77	1.71	236	-	-
B0	12	3.14	236	-	-
B1	28	3.14/2.64	172/52	12	8
B2	8	3.14/2.64	124/100	12	8
B3	40	3.14/2.64	168/52	16	8
C0	36	3.64/3.14	184/52	-	-
C1	8	3.64/3.14	172/52	12	8
C2	12	3.64/3.14	168/52	16	8
C3	20	3.64/3.14	120/100	16	8

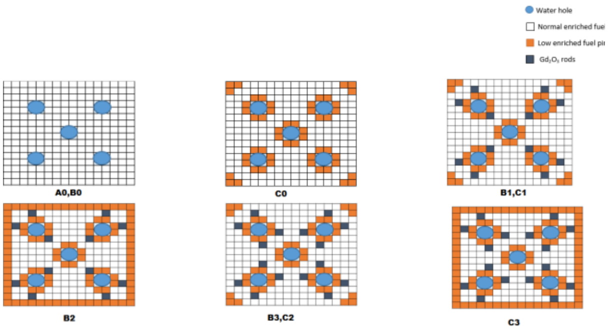


Figure 4: First Cycle FAs Configurations

3. Results

In this section, the results of the model will be presented: starting with the point kinetics model followed by the coupled model using nodal kinetics.

3.1. Point Kinetics Model Results

3.1.1. Steady State Validation

As is common practice, the TH model must be verified and validated for both steady state and transient scenarios. Table 2 below provides the result of the validation and verification of the TH model in RELAP5/SCDAP/MOD3.4 under steady state conditions.

Table 2: Steady State Validation

Parameter	DCD	Model
Initial Power level (MWt)	4062	4062
Initial core inlet coolant temperature, ºC	295	290
Initial core mass flow rate kg/s	19344.44	19318
Initial pressurizer pressure, kg/cm2A	163.46	163.13
Initial pressurizer water volume, m3	39.91	39.94
Axial Shape Index	0.3	0.3
CEA worth for trip %Δρ	-9.3	-9.3
Moderator coefficient	most negative	most negative
Doppler coefficient	most negative	most negative
Initial steam generator liquid inventory per SG, kg	124113	124595
Two safety injection pumps	Inoperable	Inoperable
Core burn up	End of cycle	End of cycle

3.1.2. Transient Validation

As for the transient validation, the MSLB was initiated, and the system response compared to that reported in DCD. The sequence of key events have been identified and compared to the DCD as listed in Table 3. As illustrated, the model captures key events with reasonable accuracy compared to values and timing in comparison with the results reported in DCD.

Table 3: Sequence of events for the MSLB analysis

Time (s)		Sequence of Key Events	Value	
DCD	Model		DCD	Model
0.00	0.00	Steam line break + LOOP occurs	-	-
0.67	0.71	Reactor coolant pump reaches CPC low RCP shaft speed (%)	94.83	94.83
1.02	0.00	CPC low RCP shaft speed trip signal generated and AFW flow initiated	-	-
1.12	0.80	Reactor trip breaker opens, and breaker	-	-
9.34	164.40	Void begin to form in RV upper plenum	-	-
12.07	19.21	steam generator pressure reaches main steam isolation signal kg/cm2A	52.73	52.73
18.42	25.56	MSIV close completely	-	-
23.42	30.56	MFIV close completely	-	-
203.83	157.60	Pressurizer empties	-	-
251.05	92.09	Pressurizer pressure reaches safety injection actuation signal analysis setpoint, kg/cm2	109.32	109.32
291.05	78.06	Safety injection flow begins	-	-
343.96	-	Safety injection boron begins to reach reactor core	-	-
373.96	600.00	Maximum transient reactivity, %Δρ	-0.361	0.147
1800.00	1800.00	Operator initiates cooldown	-	-

Additionally, the time response of key system parameters have been cross-validated against the evolution of corresponding parameters from DCD as illustrated in Figure 5 to Figure 11, and as displayed, these results show reasonable qualitative agreement between the DCD and the model. However, some quantitative differences can be observed, especially with regards to the RCS pressure. As per DCD result, an initial rapid decrease in pressure is expected, followed by slow depressurization of the RCS. However, the model does not exactly follow this behavior; instead, a slow depressurization is exhibited.

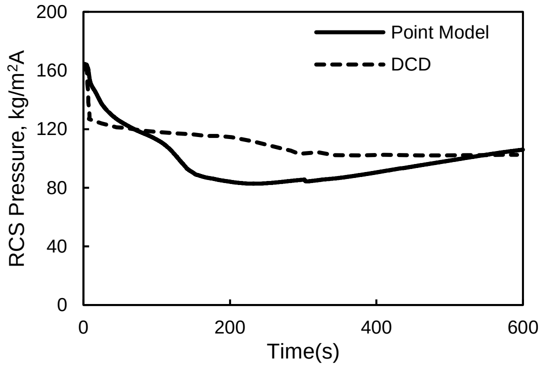


Figure 5: RCS Pressure

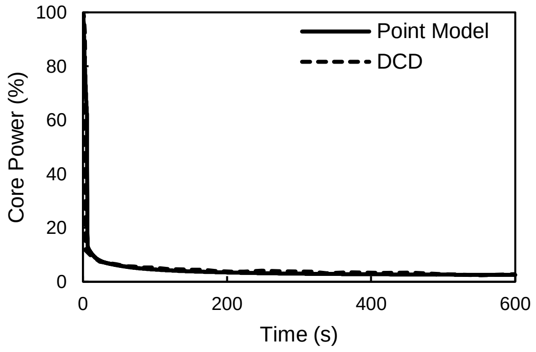


Figure 6: Core Power

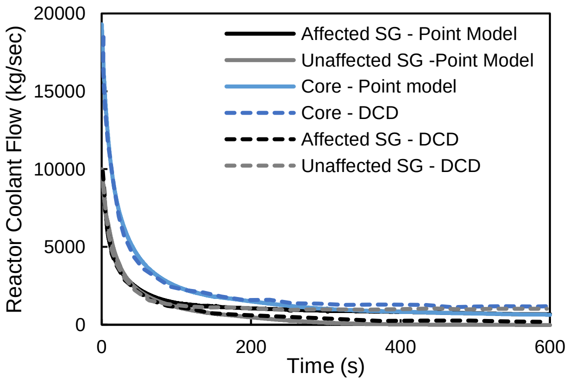


Figure 7: RCS Flowrate

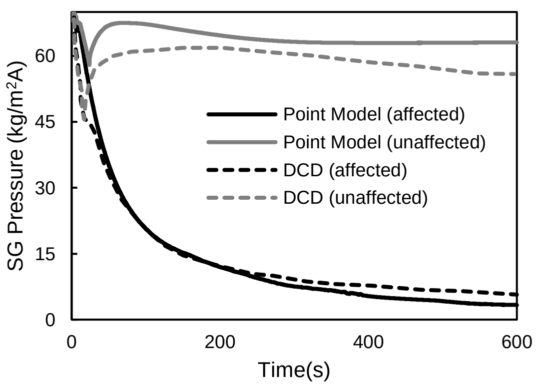


Figure 8: SG Pressure

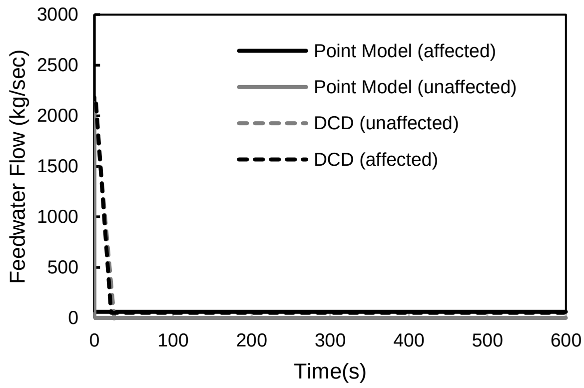


Figure 9: Feedwater Flowrate

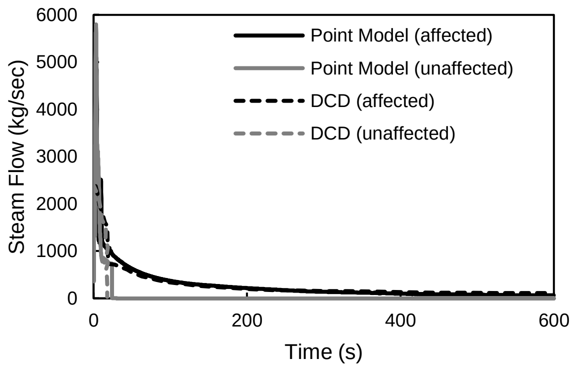


Figure 10: SG Steam Flowrate

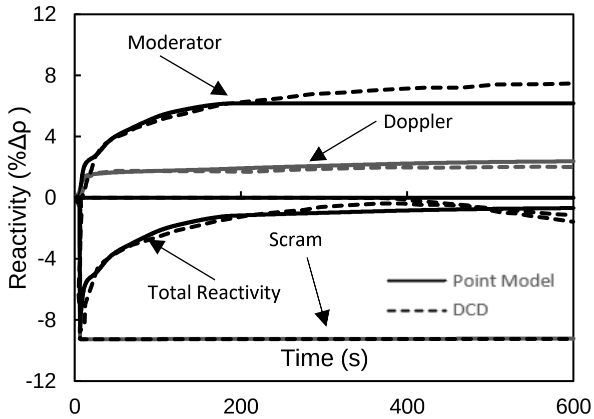


Figure 11: Reactivity

3.1.1. Nodal Kinetics Model Results

Similar to the validation and verification that was performed for TH model, the NK model was also validated by comparing the model's radial power distribution with the radial power distribution from the DCD [1] as shown in Figure 12 and Figure 13.

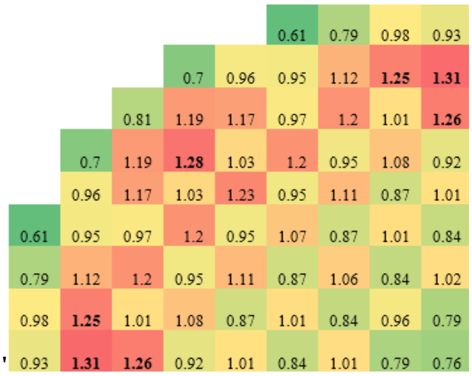


Figure 12: Radial Power Distribution (Nodal Model)

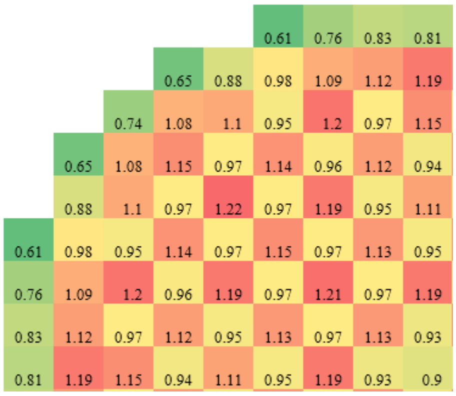


Figure 13: Radial Power Distribution (DCD)

Though the radial power distribution deviated from the average rod power distribution; however, since the deviation is within 16% of the radial power distribution reported in the APR1400 DCD, further improvement of this result is still needed.

For conservatism, it is assumed that LOOP occurs concurrent with the turbine trip, which renders the RCPs and feedwater pumps immediately unavailable. As a result, the heat removal capacity is diminished and a slight increase in primary system temperature and pressure ensues as can be observed in Figure 14 and Figure 17.

Figure 14 to Figure 19 illustrate the system response obtained from the coupled model in contrast to the results of the point kinetics model and the corresponding values reported in the DCD with reasonable agreement.

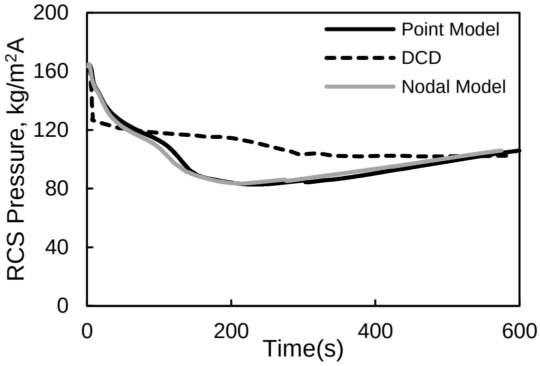


Figure 14: RCS Pressure (Nodal Kinetics)

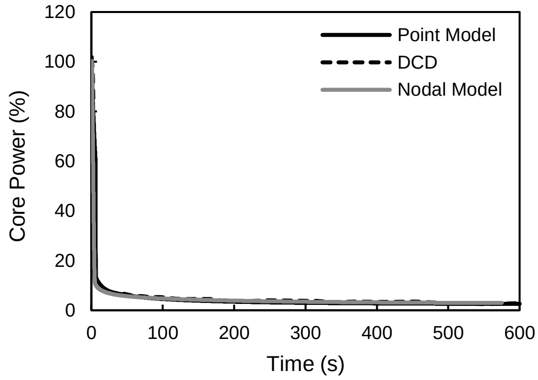


Figure 15: Core Power (Nodal Kinetics)

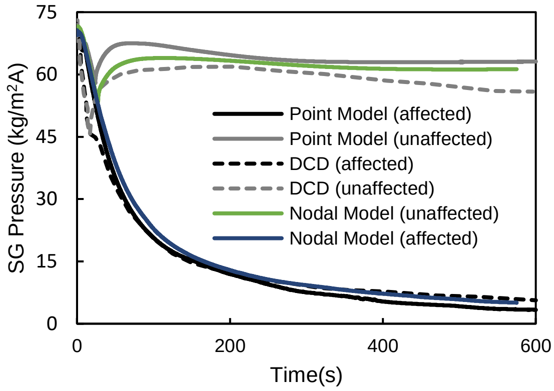


Figure 16: SG Pressure (Nodal Kinetics)

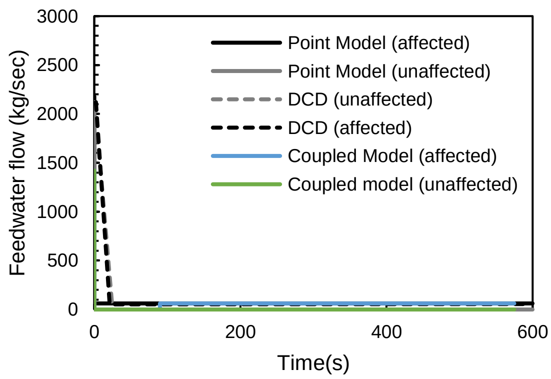


Figure 17: Feedwater Flowrate (Nodal Kinetics)

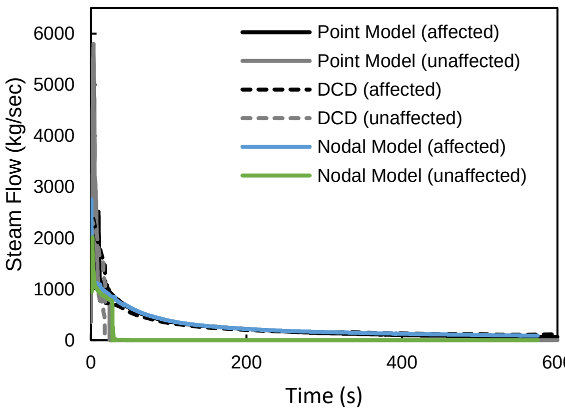


Figure 18: SG Steam Flowrate (Nodal Kinetics)

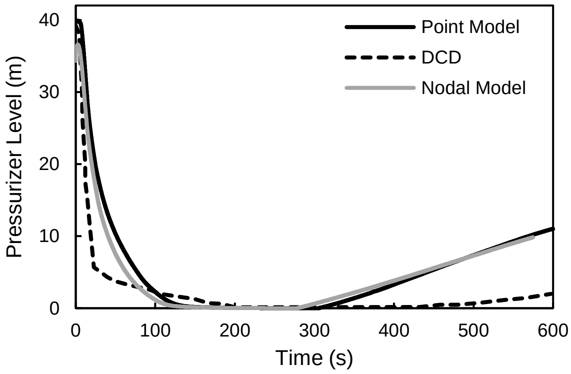


Figure 19: Pressurizer Level (Nodal Kinetics)

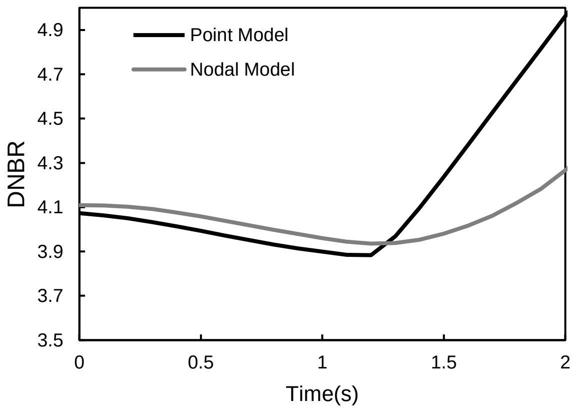
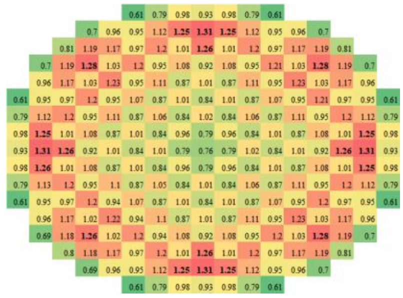


Figure 20: DNBR

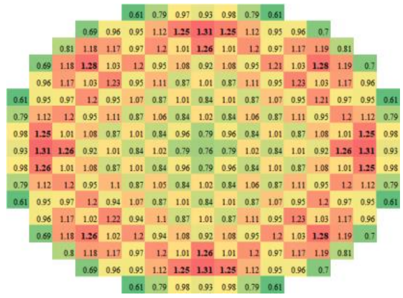
As shown in Figure 20, the DNBR drops as a result of the accident until the reactor is tripped at 0.67 seconds; subsequently, the DNBR recovers at around 1.2 seconds. The results also indicate that the coupled model with nodal kinetics shows more margin compared to the point kinetics model.

Figure 21 illustrates the full core map showing the radial power distribution at different times during the accident: at the onset of MSLB, at the minimum DNBR, at the SG isolation and at the point of potential RTP. As can be observed, the coupled model with three-dimensional core can clearly show the asymmetric cooling and hence three-dimensional effects introduced due to the uneven reactivity feedback mechanisms as a result of the accident.

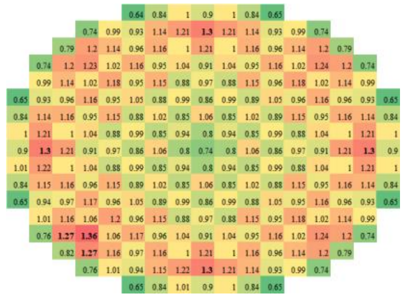
The nodal kinetics model results show some deviation; therefore, model improvement is currently being investigated. The safety concern under investigation for this accident is RTP, did not occur in the 600s of transient. Overall, the results were found acceptable as the system remains safe post reactor trip. Preliminary BEPU results show that the plant remains safe and that no RTP occurs for this accident.



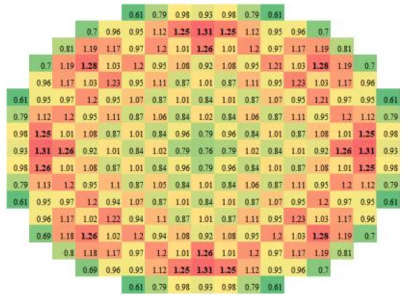
(a) core map at time of accident initiation



(b) core map at time of lowest DNBR



(c) core map at time of SG isolation



(d) core map at time of potential RTP

Figure 21: Radial power distribution at different times

4. Conclusion

This paper is an initial report on the ongoing study to analyze the postulated MSLB accident on the APR1400 using Multiphysics simulation. This accident scenario maximizes the potential for post trip fuel degradation. Preliminary results indicate that post-trip RTP does not occur, but the model needs further improvements before solid conclusions can be drawn.

5. Acknowledgement

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