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Development Status and Simulation Results for ROSHNI - A New Integrated Severe Accident Simulations Code for PHWR Severe Accident Simulator Development

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1. SUMMARY

Integrated PHWR severe accident progression and consequence assessment code ROSHNI [1], developed and tested initially for CANDU-6 reactors such as at Wolsong, improves significantly on the other computer codes that were developed over 30 years ago when modeling decisions were constrained by limited computing power. Those codes served well the requirements of 20-30 years ago. For today they are severely lacking. Other existing integrated PHWR severe accident computer codes have gross simplifications in reactor core modelling, materials, fluids and phenomena rendering the results academically interesting but practically suspect and wanting for easily implementable improvements. They do, however, excel in their computational speed (minutes of computations instead of days and hours for ROSHNI for a 24 hour station blackout accident progression assessment) because they compute so little.

ROSHNI differs from codes such as ISAAC [2] and MAAP-CANDU [3] and adaptations of LWR codes [4], in firstly the detail in which the reactor is modeled, and secondly in its more comprehensive consideration of important PHWR related severe accident phenomena, material behaviour and progression pathways.

ROSHNI incorporates a more detailed modelling of the reactor core. All horizontal PHWR reactor channels with all their fuel bundles, end fittings and feeders are modelled and in far greater detail than in other codes. Thermo-chemical transient behaviour of about 80,000 different fuel channel entities within the core is considered simultaneously. With that detail ROSHNI is able to provide information on phenomena and source terms that could not previously be generated. It has an advanced, more CANDU specific consideration of solid debris behaviour in the Calandria vessel. The code is designed to grow into and/or use its voluminous results in a severe accident simulator for training. It uses a multi-step predictor-corrector numerical solution scheme as opposed to the simple one time step Euler method in other codes.

Compared to risk profiles of modern power reactor designs, operating PHWRs are relatively obsolete in their inherent design with documented deficiencies in material selection, overpressure protection and instrumentation. They also exhibit more severe accident consequences; have undergone almost no design upgrades and effectiveness of some ad-hoc accident mitigation measures like PARS and Filtered Venting is tenuous at best. A best effort code such as ROSHNI can be instrumental in identifying the risk reduction benefits of undertaking design improvements and provide guidance on optimization of measures that have been proposed to fix the known design and accident management deficiencies.

Special features of the ROSHNI code

- No adaptation of any LWR code; direct, dedicated horizontal channel PHWR core models.
- Integrated reactor and system modelling with appropriate feedbacks
- Heat Transport System model dedicated to severe accidents– include P&IC behaviour
- Channel and Core models – significantly improved detail – all channels and all fuel bundles modelled simultaneously.
- End fittings and feeders modelled specific to each fuel channel
- Core debris modelled with consideration of in-core devices
- Hydrogen source term – includes steel oxidation in feeders, end fittings and piping
- Fission products – tracks all risk sensitive species
- Failure mechanisms clearly documented and new accident progression pathways identified and enabled
- Heavy water properties and light water properties as needed
- Containment – more detailed model
- Integrated Post processing and simple optimization
- Graphics for display of results and visualization of accident progression

3. ROSHNI MODELLING

A single unit CANDU-6 reactor is from a 700 MWe class of Generation II reactors. It has 380 horizontal, about 6m long, ~10 cm diameter fuel channels with 37 element fuel bundles. Channels are located in a 22x22 matrix with fuel channels differing from each other by a number of factors including power, elevation within the Calandria vessel and feeder geometry. Nominal channel powers vary (*Figure* 1) from about 2500 kW to 6600 kW (average 5430 kW) with channels separated by a square pitch of about 28.6 cm, the elevation difference between the top and bottom channels is about 6.65m. The inlet (outlet) feeders range in size from 3.8 cm (4.9 cm) to 5.9 cm (8.2 cm) and their lengths range between 6 to 17m (*Figure 2*), for a total length of 9,200m. Volume of water in the feeders ranges between 7 and 93 liters for a total volume of about 27,000 liters. The feeders are present ample surface area (~1,740 m^2) for steel oxidation and a metal mass of over 100 tons. Thus modelling of feeders individually for all aspects of their thermo-chemical interactions is important.

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22
A									3187	3299	3378	3378	3298	3187								
B						2862	3421	3919	4206	4394	4437	4437	4394	4205	3917	3419	2860					
C					3194	3808	4437	4908	5215	5363	5333	5333	5362	5213	4906	4434	3805	3191				
D				3319	4014	4716	5316	5727	5971	6059	5970	5969	6058	5969	5725	5313	4713	4010	3314			
E			3188	4012	4748	5390	5901	6216	6356	6387	6278	6278	6386	6355	6213	5899	5386	4744	4006	3179		
F			3868	4689	5320	5857	6241	6437	6369	6350	6285	6284	6350	6367	6435	6239	5853	5315	4683	3859		
G		3497	4410	5224	5693	6111	6362	6479	6415	6397	6400	6400	6396	6414	6478	6360	6108	5689	5219	4404	3491	
H		3983	4925	5665	6002	6303	6469	6536	6460	6441	6456	6456	6440	6459	6535	6467	6301	5999	5662	4921	3984	
J	3213	4317	5302	5977	6192	6329	6447	6485	6450	6425	6414	6414	6425	6449	6484	6447	6328	6191	5977	5303	4322	3207
K	3434	4611	5584	6207	6301	6362	6463	6475	6432	6383	6320	6320	6383	6432	6475	6463	6362	6301	6209	5587	4620	3432
L	3577	4763	5739	6365	6493	6515	6526	6495	6427	6346	6228	6228	6346	6427	6495	6527	6516	6495	6368	5744	4775	3578
M	3564	4768	5762	6413	6572	6593	6575	6521	6440	6350	6221	6220	6350	6441	6522	6576	6595	6574	6417	5768	4782	3568
N	3394	4610	5627	6323	6549	6608	6592	6542	6467	6395	6295	6296	6395	6468	6543	6593	6609	6552	6327	5364	4624	3399
O	3184	4334	5365	6113	6467	6596	6585	6557	6495	6458	6444	6444	6549	6495	6558	6586	6598	6469	6116	5369	4345	3186
P		3975	4937	5713	6100	6409	6539	6584	6499	6485	6513	6514	6485	6500	6584	6540	6410	6102	5715	4939	3981	
Q		3470	4368	5173	5639	6070	6343	6482	6439	6447	6481	6480	6447	6439	6483	6343	6071	5640	5173	4367	3471	
R			3762	4534	5096	5656	6129	6390	6372	6402	6392	6393	6402	6273	6390	6130	5656	5096	4534	3761		
S			3045	3818	4479	5156	5784	6182	6385	6459	6372	6372	6459	6385	6182	5784	5156	4478	3817	3044		
T				3073	3756	4485	5155	5640	5944	6076	6003	6003	6076	5944	5640	5155	4485	3756	3072			
U					3073	3756	4485	5155	5640	5403	5384	5384	5403	5206	4832	4288	3596	2926				
V						2548	3187	3761	4132	4384	4463	4463	4384	4132	3761	3187	2518					
W									2961	3174	3300	3300	3174	2961								

Figure 1: Channel power distribution

It is evident that channels with the same power but different feeder geometry would have a different state of coolant starvation at any given time. For example, in a station blackout scenario following loss of boilers as a heat sink and reduction of HTS inventory by boiloff down from header level, the reactor core may contain channels that are full of water while others are boiling off with yet others degraded and heating up (Figure 4).

INLET FEEDER LENGTH [m]

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22
A									6.65	6.59	6.19	6.19	6.59	6.65								
B						7.41	7.35	7.41	7.34	7.40	7.35	7.35	7.40	7.34	7.41	7.35	7.41					
C					7.07	7.01	7.07	7.01	7.66	7.61	6.90	6.90	7.61	7.66	7.01	7.07	7.01	7.07				
D				7.51	7.62	7.68	7.62	7.68	7.62	7.68	7.62	7.62	7.68	7.62	7.68	7.62	7.68	7.62	7.51			
E			7.79	8.21	7.81	7.75	7.81	7.75	7.81	7.76	7.90	7.90	7.76	7.81	7.75	7.81	7.75	7.81	8.21	7.79		
F			7.90	8.52	8.16	8.21	8.16	8.24	8.18	8.24	8.16	8.16	8.24	8.18	8.24	8.16	8.21	8.16	8.52	7.90		
G		8.03	8.80	8.47	8.51	8.45	8.53	8.47	8.53	8.47	8.21	8.21	8.47	8.53	8.47	8.53	8.45	8.51	8.47	8.80	8.03	
H		8.55	8.79	8.66	8.91	8.99	8.93	8.99	8.93	8.98	8.90	8.90	8.98	8.93	8.99	8.93	8.99	8.91	8.66	8.79	8.55	
J	9.26	8.75	9.41	9.03	9.54	9.48	9.54	9.48	9.54	9.48	8.95	8.95	9.48	9.54	9.48	9.54	9.48	9.54	9.03	9.41	8.75	9.26
K	9.70	9.26	9.80	9.37	9.94	10.02	9.94	10.00	9.94	10.02	9.92	9.92	10.02	9.94	10.00	9.94	10.02	9.94	9.37	9.80	9.26	9.70
L	9.64	10.16	10.56	10.45	10.68	11.00	11.43	11.74	12.44	12.75	9.65	9.65	12.75	12.44	11.74	11.43	11.00	10.68	10.45	10.56	10.16	9.64
M	10.06	10.88	10.99	11.14	11.12	11.71	11.85	12.45	12.84	13.47	10.05	10.05	13.47	12.84	12.45	11.85	11.71	11.12	11.14	10.99	10.88	10.06
N	10.59	11.19	11.61	11.47	11.73	12.01	12.47	12.76	13.48	13.78	10.67	10.67	13.78	13.48	12.76	12.47	12.01	11.73	11.47	11.61	11.19	10.59
O	10.87	11.80	11.89	12.07	12.04	12.63	12.75	13.38	13.78	14.39	11.07	11.07	14.39	13.78	13.38	12.75	12.63	12.04	12.07	11.89	11.80	10.87
P		11.61	12.54	12.62	12.82	12.78	13.37	13.50	14.11	14.50	15.12	15.12	14.50	14.11	13.50	13.37	12.78	12.82	12.62	12.54	11.61	
Q		12.22	12.82	13.24	13.09	13.37	13.64	14.10	14.40	15.11	15.41	15.41	15.11	14.40	14.10	13.64	13.37	13.09	13.24	12.82	12.22	
R			13.43	12.92	13.70	13.66	14.26	14.39	15.01	15.40	16.02	16.02	15.40	15.01	14.39	14.26	13.66	13.70	12.92	13.43		
S			13.72	13.54	13.98	14.28	14.53	15.00	15.29	15.99	16.28	16.28	15.99	15.29	15.00	14.53	14.28	13.98	13.54	13.72		
T				13.83	14.60	14.56	15.14	15.27	15.89	16.30	16.92	16.92	16.30	15.89	15.27	15.14	14.56	14.60	13.83			
U					14.89	15.16	15.43	15.88	16.17	16.89	17.19	17.19	16.89	16.17	15.88	15.43	15.16	14.89				
V						15.92	15.75	16.19	16.50	16.72	17.18	17.18	16.72	16.50	16.19	15.75	15.92					
W									16.74	17.46	17.75	17.75	17.46	16.74								

Figure 2: Variations in feeder lengths

The feeder geometry differences affect a typical severe accident progression (e.g. station blackout) by virtue of not only the differences in coolant mass within the feeders but also the resistance to fluid flows based on feeder diameters, elevations and lengths. Given that a necessary condition for a fuel channel to lose its integrity (heatup, deformation and melting of fuel channel segments under stress / weight) is a loss of cooling from within and a loss of moderator water enveloping the channel, the elevation of the channel within the depleting moderator is also the governing factor for channel heatup and degradation to debris in absence of large heat sinks. Additionally, disassembly of a channel is affected by physical contact with overlying channels deformed down or disassembled during the accident. Movement of debris within the core is also affected by presence of in-core devices (*Figure 3*), both horizontal and vertical. Therefore, the core needs to be modelled with the greatest detail. Previous attempts at modelling the fuel channels included simplifications in modelling of fuel channels and fuel bundles. Some codes modelled the fuel bundle with a single pin, while others did not model fuel sheaths explicitly. None modelled feeders for oxidation or in-core devices for influence on motion of each individual bundle after disassembly into suspended debris.

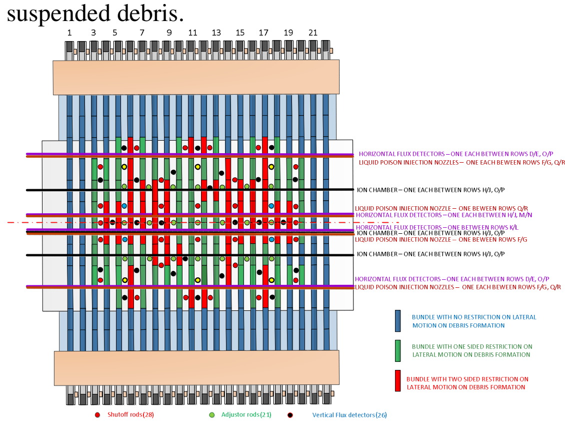
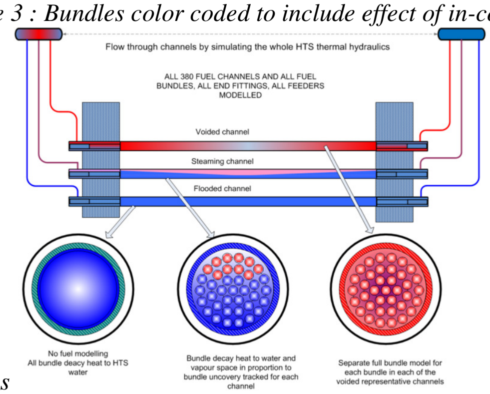


Figure 3 : Bundles color coded to include effect of in-core



devices

Figure 4: Range of channel behaviors after depletion of water inventory to the headers.

Thermal hydraulic and thermo-mechanical-chemical response of all fuel channels and all fuel bundles is modelled in ROSHNI with considerations of variations in power, axial power profiles, fuel burnup, feeder geometries and external boundary conditions including presence of debris, proximity to in-core devices and other thermal hydraulic boundary conditions. For a CANDU 6 reactor this requires modelling of 12 bundles in each of the 380 channels along with their feeders and end fittings. Within each channel all fuel bundles are modelled individually and each fuel bundle is modelled with 2 nodes for every fuel ring and associated fuel sheaths. Zircaloy associated with end plates, appendages and fuel element end caps (14% of the total Zircaloy in a bundle) is smeared over the fuel sheath. With 16 node radial model (*Figure 7*) for each bundle (or part thereof) thermal response of 380x12x16=54720 volumes is computed at each time step with consideration of oxidation, growth of oxide layer, fission product inventories and deformations as necessary. In addition, thermal response of all 760 end fittings and 760 feeders is also computed with 10 subnodes each (*Figure 5*).

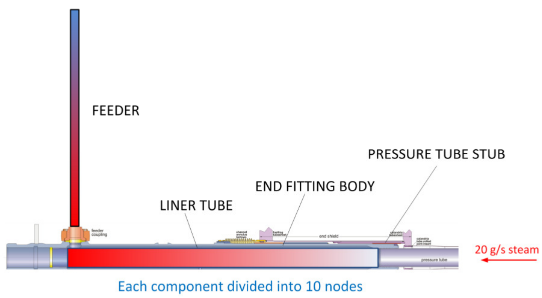


Figure 5: End-fitting & feeders modelling in ROSHNI

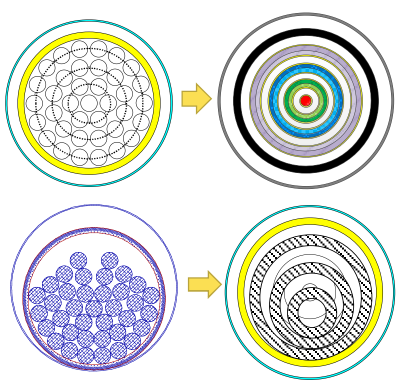


Figure 6: Bundle model for un-deformed and deformed bundle geometry.

Bundles are modelled in their deformed state with a ring model similar to the un-deformed state (*Figure 6*) Fuel sheath failures as well as perforations of Pressure and Calandria tubes under stress are also modelled *(Figure 8).* Suspended debris at each of the 4,560 bundle locations (*Figure 9*) is tracked on each bundle basis including the effect of bundle transfer from higher to lower locations. Terminal debris (*Figure 10*) behaviour is differentiated with consideration of water level and melt formation. Fission product species are tracked based on their risk importance (*Figure 11*)

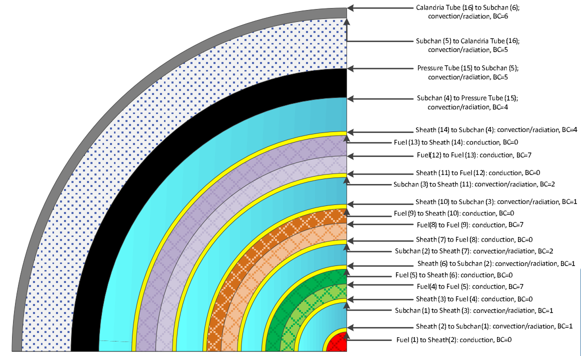


Figure 7:

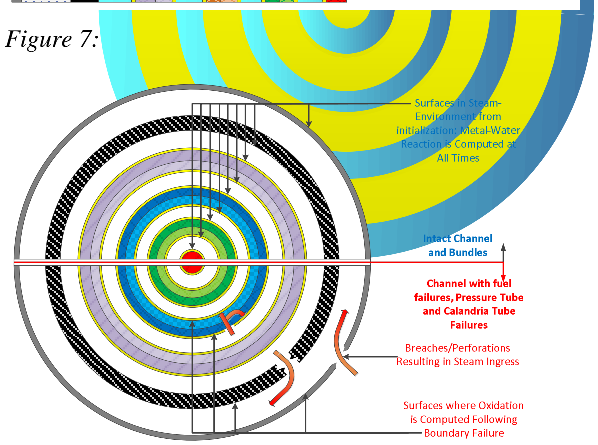


Figure 8: Oxidation surfaces before and after fuel / channel failures

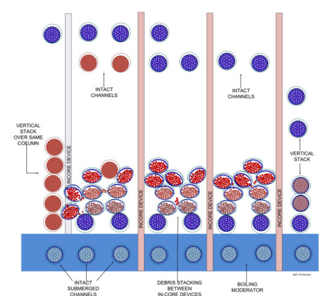


Figure 9: Debris behaviour tracking each disassembled bundle

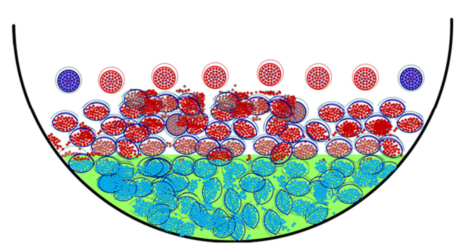


Figure 10: Terminal debris with water and/or melt.

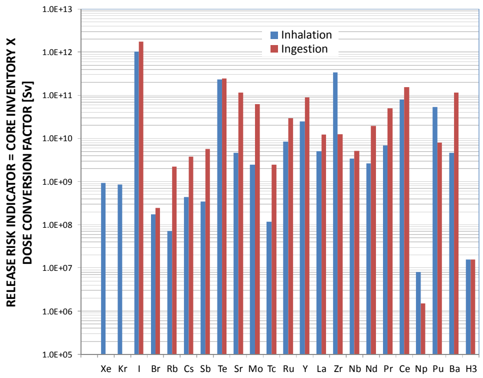


Figure 11: Core inventory times dose conversion factors for inhalation (Blue) and ingestion (red) to examine relative risks of releases for 23 elements

4. Sample Results

Figures 12 through 15 present sample results for a core behaviour following a station blackout event without operator intervention or recovery actions.

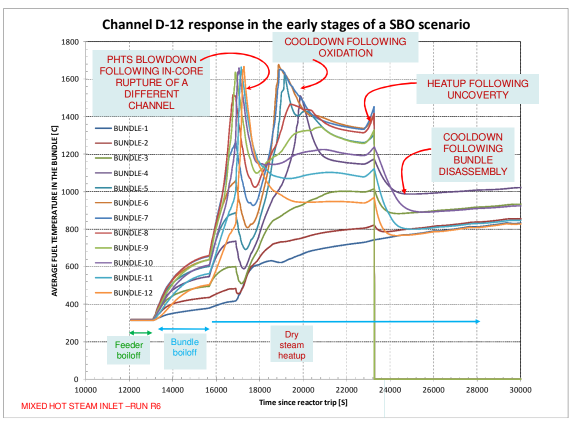


Figure 12: Thermal transients for 12 bundles of a typical channel

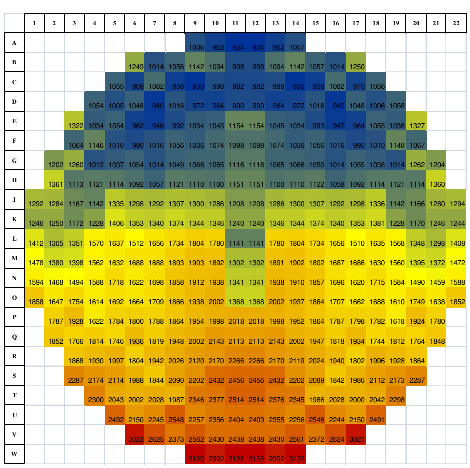


Figure 13 : Time taken to boiloff water in each feeder

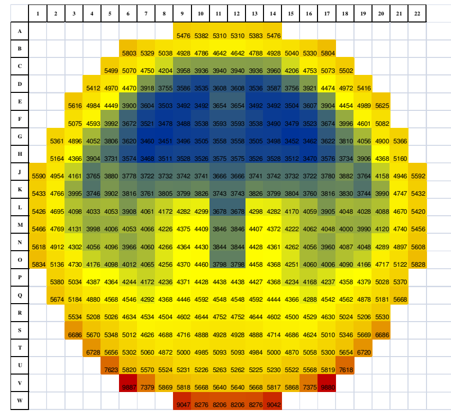


Figure 14: time taken to boiloff water within each channel and begin dry heatup

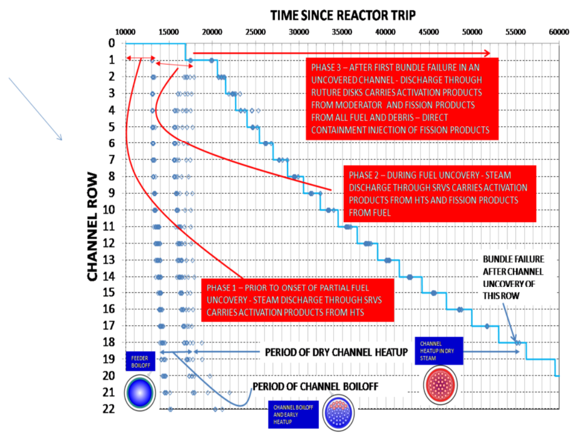


Figure 15: Core heatup and disassembly summary for whole core

5. Conclusions

A new integrated computer code to model reactor behaviour to a severe core damage accident has been developed and is being tested for single unit CANDU-6 reactor such as one at Wolsong. Preliminary results are exciting, revealing and interesting and very different from the currently used codes such as ISAAC and MAAP-CANDU and adaptations of LWR codes reported in TECDOC 1727. PHWR severe accidents can now be analyzed more realistically and in greater detail.

6. References

[1] ROSHNI - a new integrated severe accident simulations code for PHWR level 2 PSA applications and severe accident simulator development, Proceedings of ICONE-23, Paper ICONE23-1054,23rd International Conference on Nuclear Engineering May 17-21, 2015, Chiba, Japan

[2] Korea Atomic Energy Research Institute, Development of Computer Codefor Level 2 PSA of CANDU Plant, KAERI / RR-1573/95 (1995).

[3] Modular Accident Analysis Program for CANDU Reactors, C.Blahnik, C.Kim, S.Nijhawan, R.Thuaisingham, ANS 1992

[4] IAEA TECDOC 1727, Benchmarking Severe Accident Computer Codes for Heavy Water Reactor Applications , 2013 (Caanad, Korea, India, Rumania, China participated)