Computer code ROSHNI's predictions of CANDU severe accident progression and its confirmation of severity of reactor design omissions, source term vulnerabilities and avoidable risks

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1. Introduction

ROSHNI is a new integrated computer numerical simulation package that models in unprecedented detail all risk sensitive core components, actual fluids as well as all relevant thermo-physical-chemical phenomena for severe accidents in PHWR type reactors such as those at Wolsong.

The most essential element of determination of severe accident consequences is detailed evaluation of degraded reactor core thermal-chemical behaviour with full consideration of severe accident phenomenology. In case of a CANDU reactor at Wolsong, it means determination of not only fuel behaviour within each of the 380 fuel channels and the 4560 fuel bundles but also the 760 end fittings and 760 feeders. All channels are similar in geometry in the horizontal section between the end fitting connections to the feeders but different from each other by a large number of parameters that affect their subsequent behaviour (power, power profile, feeder geometry, feeder volumes, vertical location within the Calandria vessel - to name a few). Therefore they contribute differently to the overall plant response and source term of fission products, energy and combustible gases; must be treated individually and cannot be lumped together.

A number of computer codes have been used over the years to evaluate accident progression and estimate the source terms. There even have been attempts (reference 1) to compare their performance and the large differences between them attest to the incomplete treatment of several important phenomenology and modelling detail limitations. The most widely used computer code MAAP-CANDU (I developed most of it during 1988-1993 attaching CANDU specific models to the LWR code MAAP at Ontario Hydro) suffers from the following errors and deficiencies:

1. No consideration of heavy water, deuterium gas (light water and H_2 properties used)
2. No momentum equation for HTS – no fluid flows
3. Channel degradation during channel boiloff before dry steam/D2 heatup not modelled - Initial fuel temperatures at onset of heatup are arbitrary.
4. Channel hydraulics based on assumed header to header Δp and no overall T/H. No intra channel flows. No consideration of fluid discharge paths.
5. A limited number of channels modeled (typically only 18 fuel channels modeled).
6. No explicit fuel sheath modeling.
7. No modeling of out of flux pressure tube lengths.
8. No modeling of water retention in end fittings after boiloff or blowdown
9. No thermal modeling of feeders and end fittings
10. No consideration of differences in burnup and power profiles between various channels
11. No modeling of in-core devices and their effect on individual fuel bundle displacements.
12. No modeling of piping into Calandria vessel.
13. Crude modeling of core disassembly & a physically impossible model of 'core collapse'
14. Primitive modeling of suspended solid debris
15. Solid debris interactions with air not modelled
16. Deuterium / Hydrogen generation by steel oxidation and Uranium-steam oxidation ignored.
17. Fission product releases from debris crudely modelled.
18. Fission products do not decay.
19. As 'engineered' codes with specific accident progression pathways – many scenario paths not considered.
20. Difficult I/O; primitive post processing

Given the lessons of Fukushima, is incomprehensible that above deficiencies have not been rectified in the 25 years since the control of that code development left my control and then Canadian hands. No SAMGs, design assist or other accident management or training measures are possible without properly modeling the reactor, its phenomenology and all potential accident progression pathways. Without modeling the behaviour of each fuel channel individually, for example, the erroneous conclusions drawn from bizarre models such as for a core collapse can give misleading and dangerously inaccurate results.

KAERI used another coupling with the MAAP code in its computer code ISAAC. Representing the reactor core with fuel bundles modeled as single fuel pins, that code does not model the reactor bundles with any credible detail; omits very important phenomena and is unsuitable for any serious evaluation of accident progression or source term predictions.

2. ROSHNI is a unique, modern code that takes no shortcuts in reactor core modelling

Following features of the ROSHNI code (reference 2) are considered an improvement over all existing PHWR code packages for severe accident progression analyses (including ISAAC, MAAP-CANDU and attempts with MELCOR, RELAP and the multi code Indian package used in IAEA TECDOC 1727):

1. ROSHNI is not an adaptation of a LWR code; it is a direct, dedicated PHWR CANDU model package.
2. Maintains integrated approach to reactor and system modelling
3. Channel and Core models – significantly improve detail – all fuel bundles in all 380 fuel channels modelled. Each bundle represented by 16 different thermal-chemical entities for over a 100,000 discrete locations.

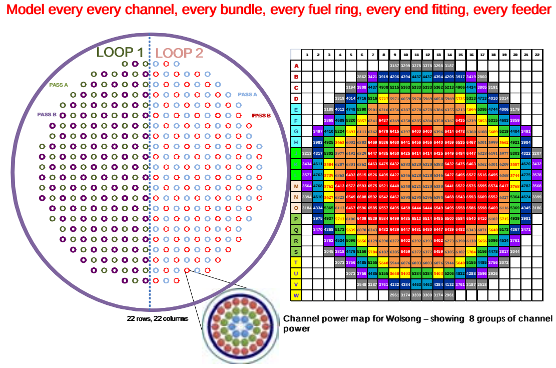


Figure 1: ROSHNI models every fuel bundle in every channel

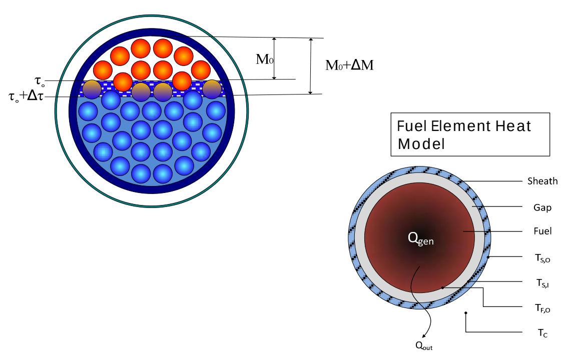


Figure 2: ROSHI models fuel bundle uncovery and heatup prior to dry heatup (no other code does).

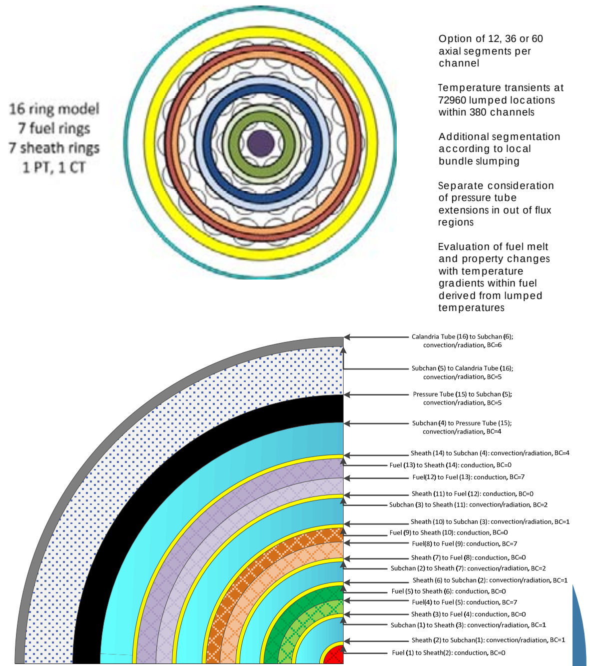


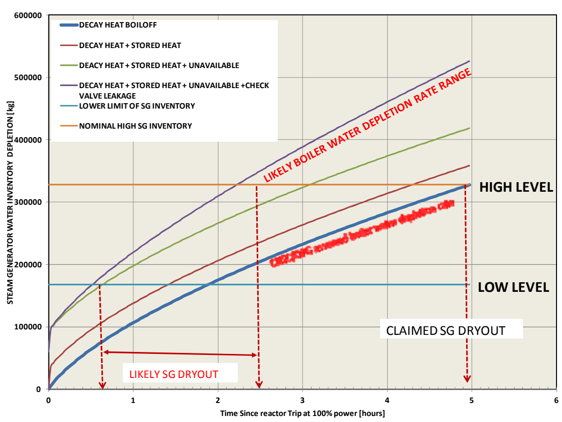
Figure 3 : ROSHNI models each bundle in great detail

4. End fittings and feeders modelled specific to each channel
5. Core bundle dislocation and debris modelled with consideration of in-core devices
6. 'Hydrogen' source term – include steel oxidation of feeders and end fittings
7. Fission products – track all risk sensitive species and their transport. FP decay model under development
8. Failure mechanisms clearly documented and new accident progression pathways identified and enabled
9. Heavy water properties for liquid and superheated D_2O and light water (H_2O) properties as needed. Corresponding release and behaviour of Deuterium gas.
10. Verified and qualified material properties
11. Fully documented input data with QA and direct link of derivation of data to code
12. Containment – more detailed model
13. Recovery actions and interventions planned
14. Modularity, validation, benchmarking planned
15. Integrated Post processing and simple optimization
16. Graphics for display of results and visualization of accident progression
17. Clean and modern Fortran coding; links to 3D visualization software
18. Full documentation of each code segment, phenomena and code limitations

19. Clear User's manuals, theory manuals and upgrades with active feedback from participants
20. Open source code and training to participating organizations

3. ROSHNI Code predictions of Severe Accident progression pathways, timing and challenges

One of the first indicators of a pressurized water power reactor vulnerabilities is the behaviour of its built in heat sinks - specifically the time over which the built in semi-passive heat sinks are available for maintaining core cooling. By consideration of all heat sources, heat sinks, boiler inventory leak pathways, ROSHNI shows that the heat sinks in reactors at Darlington are only about an hour and a half as opposed to 5 hours predicted by the MAAP-CANDU and only about an hour for Wolsong as opposed to more than two hours predicted by ISAAC.



The analytical results show the location of the first channels to fail by overheating after the overpressure induced uncontrolled PHTS boundary rupture failures deplete by draining the primary inventory prematurely to the headers. The overpressure failures may occur in the boiler tube creating a containment bypass or occur in a fuel channel. Recall that the uncontrolled failures are caused by a yet unresolved design error of insufficient steam relief capacity in the DCRVs. The code also shows the effect of differences in channel powers, feeder geometries and elevations on channel thermal-chemical response.

By demonstrating the differences in bundle failures and channel segment disassembly in the core, it belies the MAAP-CANDU concept of a 'core collapse' that has been surreptitiously used to knowingly limit source terms and thus ill serves the need for enhanced emergency measures at CANDU reactors in Canada.

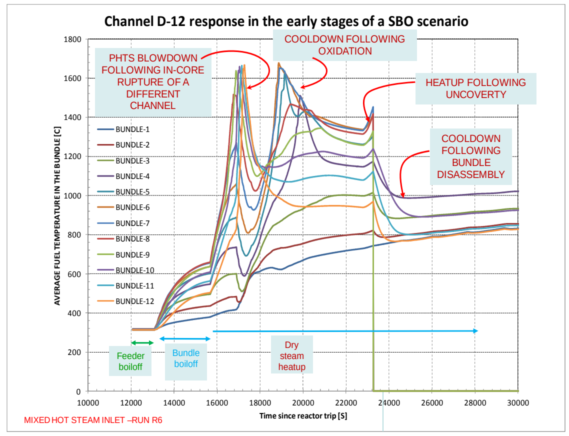


Figure 4: Sample fuel channel response to a loss of heat sinks

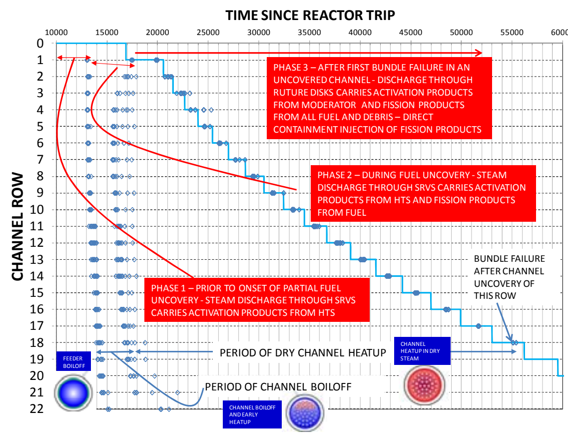


Figure 5 : Sample core disassembly transient without early overpressure failures

4. ROSHNI Code confirmation of reactor design errors and need for design fixes

- Due to a faulty overpressure protection design, a reactor pressure boundary damage is forced even before an ECC injection loss leading to severe core damage.
- A manual depressurization of PHTS after SBO can reduce consequences. Absence of a super high pressure ECC or water inventory makeup intervention / injection.
- Onset of a severe core damage in a CANDU reactor puts activity directly into the containment. There is no holding of activity in a vessel like in a PWR pressure vessel.
- Significantly higher sources of hydrogen from large amounts of carbon steel and Zircaloy. Recombiners, inadequate in numbers, capacity and performance metrics for Deuterium, will cause explosions.

- Energetic interactions with enveloping water in shield tank due to Calandria weld failures will cause steam explosions
- Inadequacy of pressure relief in ALL relevant reactor systems (PHTS, Calandria, Shield Tank, Containment) will cause enhanced source terms to the environment
- Containment bypass from reactivity device failure a likely outcome after a severe core damage
- Calandria vessel cannot contain debris and can fail catastrophically at welds.
- Shield Tank cannot contain pressure upon boiling and can fail. Restoration of cooling after water depletion problematic as flow outlet at the top of vessel.
- Currently planned PARS inadequate and potentially dangerous.

5. ROSHNI Code prediction of enhanced source terms

The code predicts the source term of risk sensitive fission products with consideration of their dose conversion factors and inventory depletions. It shows that the releases into the atmosphere will be two orders of magnitude higher than those predicted by the Canadian nuclear industry where a release no larger than the 100 TBq Cs-137 which is really only a definition large release has been used for emergency management.

The code also predicts the Deuterium source term at least twice as high as that from other codes by including the source term from steel oxidation.

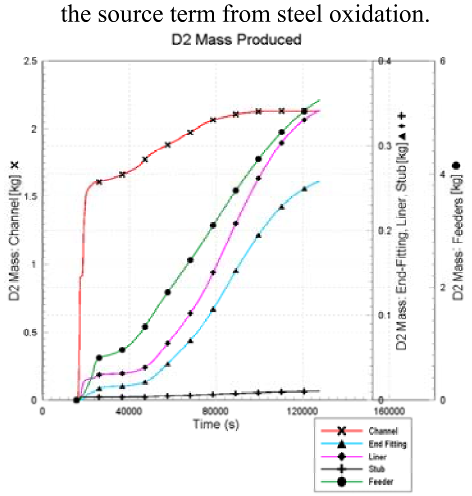


Figure 6: Prediction of Deuterium source term from fuel and feeders in a single channel.

SUMMARY

The purpose of this paper is to illustrate the importance of information a comprehensive computer simulation can provide and how the results of ROSHNI differ from those from other computer codes such as MAAP-CANDU and ISAAC . It is also to illustrate that these PHWR reactors that were never designed for severe accidents can have large source terms of fission products and combustible gases into the relative weak containments they sport and as a result potentially cause significant off-site damage. It shows the timing of progression of accident and the limited opportunities for successful accident management. It becomes evident soon that the Wolsong PHWRs have inherent design issues that must be resolved immediately to reduce risk or the reactor operation be only continued with open and full acceptance of the severe accident risk. At the moment there are no simulation techniques available to the reactor operators in Korea that can benefit any real risk reduction measures or support operator training in realistic management of severe accidents. The source terms and challenges that these reactors pose can only be reduced by developing a common understanding between the analysts, the regulators and the operators of power reactors. Real safety needs more than just lip service. In addition :

- Past attempts to model CANDU reactor severe accidents with obsolete integrated codes and patch work imported codes has failed
- Severe accident phenomena in a CANDU reactor differ from other reactors in critical areas to require dedicated modelling
- It is irresponsible not to improve design in light of known deficiencies and emerging understanding of severe accident phenomena
- Design improvements cannot be mandated; they have to be reasonable and effective
- Proper accident management requires proper analytical backup. It requires a deep understanding of potential accident progression pathways and recovery actions exercised well in advance.
- Simulator based training is an essential accident management training tool
- Present codes were good for requirements 20 years ago. For today they are severely lacking.

REFERENCES

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1. Benchmarking Severe Accident Computer Codes for Heavy Water Reactor Applications, IAEA TECDOC 1727, November 2013

2 . "ROSHNI - A new integrated severe accident simulations code for PHWR level 2 PSA applications and severe accident simulator development" published in International Conference on Nuclear Engineering, Proceedings, ICONE 23