# Benchmarking Severe Accident Computer Codes for Heavy Water Reactor Applications

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## Abstract

Consideration of severe accidents at a nuclear power plant (NPP) is an essential component of the defence in depth approach used in nuclear safety. Severe accident analysis involves very complex physical phenomena that occur sequentially during various stages of accident progression. Computer codes are essential tools for understanding how the reactor and its containment might respond under severe accident conditions.

International cooperative research programmes are established by the IAEA in areas that are of common interest to a number of Member States. These co-operative efforts are carried out through coordinated research projects (CRPs), typically 3 to 6 years in duration, and often involving experimental activities. Such CRPs allow a sharing of efforts on an international basis, foster team-building and benefit from the experience and expertise of researchers from all participating institutes. The IAEA is organizing a CRP on benchmarking severe accident computer codes for heavy water reactor (HWR) applications. The CRP scope includes defining the severe accident sequence and conducting benchmark analyses for HWRs, evaluating the capabilities of existing computer codes to predict important severe accident phenomena, and suggesting necessary code improvements and/or new experiments to reduce uncertainties. The CRP has been planned on the advice and with the support of the IAEA Nuclear Energy Department's Technical Working Groups on Advanced Technologies for HWRs.

## 1. Introduction

The postulated severe core damage accidents that result in the loss of core geometry have very low frequency in HWRs. There are reliable heat sinks and safety systems to prevent an accident from entering into the severe core damage realm. A severe core damage accident typically requires a significant loss of moderator, which would otherwise act as a heat sink for voided fuel channels. There are two categories of accidents for HWRs which involve fuel or core damage:

- Limited core damage accidents (LCDAs)

  In this type of accident, a single channel is affected by a stagnation feeder break or a severe flow blockage causing the fuel to overheat in the affected channel. During the accident, the integrity of the core geometry is not compromised except for the affected channel. Although the fuel is unlikely to melt, a limited amount of sheath or end plates from fuel bundles may melt. The relocation of melt in the affected fuel channel may cause channel rupture if sufficient melt accumulates in the pressure tube. The ejected melt will be adequately cooled by the moderator and the fuel bundles within the channel will be cooled by the discharge flow through the rupture.

- Severe core damage accidents (SCDAs)

  For an accident to proceed to significant core damage there must be failure of multiple fuel channels. This is possible with the loss of moderator as heat sink. The severe core damage accident may be initiated from an LCDA and progresses to significant core damage with the non-availability of moderator as heat sink. The SCDA is characterized by the loss of moderator heat removal capability. For example a loss-of-coolant accident (LOCA) with a loss-of-emergency-core-coolant (LOECC) in conjunction with loss of moderator cooling leads to a SCDA, resulting in slow boil-off lasting for many hours of the moderator and eventual core collapse.

The HWR plants have many distinctive and inherent safety related characteristics in terms of prevention and mitigation of severe accidents, for example:

- Large and subcooled inventory of moderator is an ultimate heat sink, preventing severe core degradation under accidents,
- Additional inventory of light water in the reactor vault surrounding the calandria vessel retains and cools core debris within the calandria under severe accident conditions,
- Reactivity and shutdown mechanisms in the low-temperature low-pressure moderator are un-affected by the disturbance in the high-temperature high-pressure primary coolant loop, i.e., the reactivity mechanisms do not penetrate pressure boundary in HWRs,
- On power refuelling of HWR maintains low excess reactivity, and
- The large inventory of water and multiple barriers provided in HWRs have the inherent capability to delay corium debris falling on the concrete minimizing core-concrete interaction under severe accident.

Severe accident analysis has been performed to support probabilistic safety assessments (PSAs), to resolve specific severe accident issues, and to support severe accident research programmes. However, with the rapid progress in computer technology and maturation of severe accident codes, accident analysis has increasingly been focused on the use of these codes for training purposes, the development and validation of accident management programmes, the design and validation of severe accident mitigation systems, and most recently, plant simulators.

The IAEA published a guidance document on severe accident analysis of HWRs, with involvement of experts from Canada, India and Korea [1]. This publication provides a set of suggestions, on the basis of current international practices on how to perform deterministic analysis of severe accidents in HWRs by means of the available computer codes. A more general framework for these suggestions is also provided, including a description of factors important to the analysis, an overview of severe accident phenomena and status in their modelling, categorization of available computer codes, and practical examples of various applications of analysis. This publication also provides information on severe accident management for HWRs. An overview of three main procedural elements of accident management, i.e., emergency operating procedures, severe accident management guidelines and the emergency plan is also introduced.

At present, there are a few severe accident computer codes used in different countries for analyzing the severe accident phenomena relevant to heavy water reactors, for example, the codes MAAP-CANDU and ISAAC. Some severe accident codes developed for light water reactors (LWRs) are also in use for the analysis of severe accidents in HWRs, for example, SCDAP, ASTEC, and MELCOR. These codes have never been compared with one another for severe accident phenomena relevant to HWRs. Besides, phenomenological models have been developed to simulate separate effects, like steam explosion, debris coolability, melt-pool heat transfer, vessel failure, etc. Separate effect tests also have been conducted by different laboratories relevant to specific phenomena of severe accident progression with either simulant or prototypic materials.

In case of design basis accidents, computer codes are validated against integral and separate effects tests, whereas in the case of severe accident computer codes it is rather impossible, or at least quite expensive, to carryout a validation exercise against integrated experiments. Consequently, the code capabilities have to be assessed based on benchmarking against other severe accident computer codes. In view of this, a benchmarking exercise becomes necessary to assess the results from various computer codes to provide an improved understanding of modelling approaches, strengths and limitations. The exercise could also suggest ways to overcome code limitations and thereby increase the confidence in severe accident code predictions. A benchmarking exercise encompassing various severe accident codes in use within HWR community is therefore proposed not only for providing confidence in the overall performance of codes but also for the reduction of uncertainties in their predictions.

## 2. Status and Trends for Severe Accident Analyses in HWR Countries

It is expected that the new reactor projects will explicitly and systematically consider severe accidents during the design phase to minimize the likelihood of severe core damage and large releases. For the currently operating plants, the risk posed by severe accidents is minimized through the implementation of severe accident management (SAM). SAM programmes are built on the existing symptom-based procedures for design basis accident management, emergency management framework, and available equipment and systems. Potential design changes for the operating plants would be primarily driven by the SAM needs, and assessed using benefit-cost considerations.

Currently different countries follow different regulatory requirements for severe accident considerations in HWRs. For example in Canada, the Canadian Nuclear Safety Commission is moving towards a systematic risk-informed regulation. This necessitates balanced consideration of severe accidents in the design and operation of nuclear reactors through implementation of specific design provision, accident management programmes, probabilistic and deterministic analyses. Safety analysis of severe accidents for CANDU reactors traditionally has not been a licensing requirement in Canada. Recently, however, the needs of PSA, development of SAM, and the new reactor design activities have led to the increased regulatory attention to the analysis of accidents resulting in severe core damage and releases of large amounts of fission products.

In India, the currently severe accidents considered are limited to dual failure such as LOCA with LOECC and LOCA with failure of containment. For the new reactors, inclusion of regulatory requirements with regard to severe accidents is under consideration.

The Korean regulatory authority issued the Severe Accident Policy in August 2001 in order that the owner of nuclear power reactor should prepare measures against severe accidents to minimize the possibility for severe accidents and their risk to the public. This policy is applicable not only for LWRs but also for all other types of nuclear power plants in Korea. It consists of four elements: safety goal, probabilistic safety assessment, severe accident prevention and mitigation capability, severe accident management programme.

In most other countries operating HWRs such as Argentina, China, Pakistan and Romania, their regulatory bodies strongly recommended to perform severe accident analysis and to establish SAM procedures for NPPs.

The safety goals for HWR severe accident events are in line with international practice where the severe core damage frequency (SCDF) for operating plants is accepted as less than $1x10^{-4}$/year. This includes contributions from all events and all operating states. For new designs the goal for the SCDF is less than $1x10^{-5}$/year. In addition, the sum of frequencies of all event sequences that can lead to release to the environment of more than $10^{15}$ Bq of $Cs_{137}$ should be less than $10^{-6}$/year. For the currently operating plants the large release frequency goals are accepted to be an order of magnitude higher than for the new designs.

## 3. CRP Objectives and Expected Outcomes

The overall objective of the CRP is to promote international collaboration among IAEA Member States through the benchmarking exercise to improve severe accident analysis capability for heavy water reactors. An improvement of safety for currently operating plants and a facilitation to more economic and safe designs for future plants are expected through the CRP. The final expected output is a TECDOC which will document the progress achieved through the collaboration with regard to code-to-code comparisons, code-to-separate-effects-test data comparison, assessment of model capabilities, identification of further model refinements and experiments.

The expected outcomes are:

- Improved understanding of the importance of various phenomena contributing to event timing and consequences of a severe accident,
- Improvement of emergency operating procedures or severe accident management strategies and
- Advanced information on computer code capabilities to enable the analysis of advanced HWR designs.

## 4. CRP Activities

The activities ongoing or planned within the CRP are:

**Activity 1:** Evaluate research contracts and research agreements.

Six institutes from five countries were determined as the participants of this CRP: Korea Atomic Energy Research Institute (Rep. of Korea), Shanghai Jiao Tong University (China), Politehnica University of Bucharest (Romania), Atomic Energy of Canada, Ltd. (Canada), Bhabha Atomic Research Centre (India), and Nuclear Power Corp. of India Ltd. (India).

**Activity 2:** Convene the first research coordination meeting – Prepare plan for overall CRP period.

The first research coordination meeting was held in Vienna on 25-27 February 2009. The objectives of the meeting were for the participants to share experience and expertise and to develop an integrated research plan indicating how the participating organizations will collaborate during the CRP and specifying their contributions to the various activities of the CRP.

**Activity 3:** Assess the existing models, correlations, experiments, and computer codes applicable to HWR severe accident analysis.

Many models and correlations originally developed for LWR could be applicable to HWR directly or with minor modification. Existing models and correlations would be categorized by three groups: 1) directly applicable, 2) applicable with modification, and 3) not applicable, so HWR specific is required. The first CRP progress report will include the description and evaluation for HWR applicability for models and correlations used in benchmark analysis.

**Activity 4:** Determine the severe accident sequences for CANDU 6 benchmark analysis. Collect and share sets of CANDU 6 design parameters such as primary heat transport system geometries, flow rates, power transients, fission product inventory, bulk temperature, mass of various heat sinks, calandria vessel geometry, reactor vault geometry, etc.

Typical postulated initiating events (PIEs) along with failure of subsequent systems which could lead to severe accident scenarios in HWRs are station black out (SBO), LOCA with LOECC, and steam generator tube rupture (SGTR) (containment bypass event).

SBO was selected as a reference PIE for CRP because of the following reasons:

- Envelope numerical calculations can be made for comparison
- Transients are slow
- Captures most of the severe accident phenomena
- Sequence can be anticipated and tracked: moderator temperature increase, secondary side boiloff, coolant inventory boiloff, primary side liquid relief valve opening, fuel channel heatup, calandria vessel rupture disk opening, core heatup and collapse.

The following assumptions were agreed to be used for SBO scenario. It should be noted that the assumptions are given to result in severe core damage status to study the capability of models used in severe core damage accident analysis in HWRs and therefore the assumptions made are not based on real design:

- AC Power and all outside standby emergency power (Class III and IV) are unavailable
- Reactor shutdown is initiated immediately after accident initiation
- Moderator cooling and shield cooling are unavailable
- Shut down cooling is not available
- Main and auxiliary feed water are unavailable
- Steam generator main steam safety valves are available, they open and close at set point to relieve pressure
- Turbine main stop valves are closed after accident initiation. The valve closure time is assumed to be 20 s.
- Containment leakage is modeled (1% of free space volume per 24 h)
- ECCS (high, medium and low pressure) is unavailable
- Steam generator crash cool-down is not credited
- Dousing is credited
- Local air coolers are not available
- Air-operated atmosphere steam discharge valves have no back-up air and springs. They are fail-closed and are not considered in the case of SBO.
- All operator interventions are not credited

**Activity 5:** Generate severe accident analysis input parameters and compare among benchmarking codes. Determine that the input parameters are consistent and accurate.

Reference plant data were provided to all participants to minimize the difference in the result caused by the deviation in input parameters.

**Activity 6:** Establish criteria for fuel failure, fuel channel failure, fuel channel disassembly, core collapse, calandria vessel failure and containment failure, and reactor vault failure. The failure criteria and the variables to be used in the comparison require consensus and understanding of limitations among all participating organizations.

<u>Fuel failure criteria</u>: A typical approach for fuel failure is to assume that the fuel cladding fails if the average fuel element temperature is higher than a specified value. For example, a value of 1000 K is used for LWR fuel, based on PHEBUS-FPT0 experimental results. This value is conservative for CANDU fuel, as it has a lower internal gas pressure. At high temperatures, the fuel elements in a fuel bundle will sag and the fuel bundle will disassemble. The temperature at which this occurs will depend on the extent of oxidation of the fuel cladding, as the oxide layer will provide support. One approach is to assume that the fuel bundle will disassemble when the temperatures reach zircaloy melting.

Fuel channel failure criteria: Fuel channel failure is defined as a perforation of its pressure boundaries that result in mass transfer between the inside of the pressure tube and the inside of the calandria vessel. This means by definition that both the pressure tube and the calandria tube have to be perforated for the fuel channel to fail. The mechanism for fuel channel failure depends on the primary heat transport system (PHTS) pressure, some mechanisms being applicable at low PHTS pressure (local melt through or sagging) and others at high PHTS pressure (non-uniform circumferential temperature gradient).

Fuel channel disassembly criteria: Fuel channel disassembly is another complex process, during which fuel and channel structural materials separate from the original fuel channel position and relocate downward, forming a suspended debris bed. The suspended debris bed will transfer to the calandria vessel bottom when the core collapses.

Core collapse criteria: Core collapse is a massive relocation of core material and some intact fuel channels within the moderator onto the bottom of the calandria vessel. When a large amount of core debris becomes lodged above the water level on top of the supporting channels (submerged channels) and the total debris mass exceeds the load-bearing capacity of the supporting channels, the supporting channels along with the debris bed can collapse. Two main calandria tube failure mechanisms can be considered: pullout from the rolled joint and shearing of the calandria tube. The former failure mechanism, however, is the most dominant failure mechanism since it requires significantly less load. The suspended debris bed mass per PHTS loop, which will trigger core collapse, can be estimated from the load required to cause calandria tube pullout from the rolled joint.

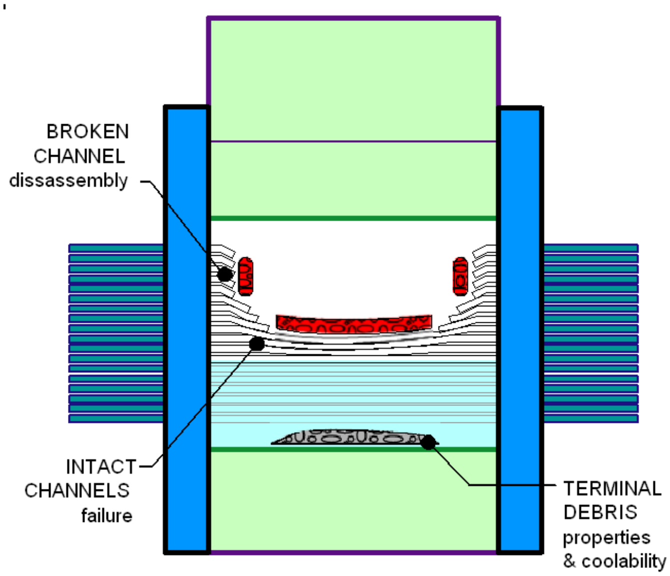


Figure 1 Fuel channel disassembly phenomena (conceptual – not to scale).

Calandria vessel failure: A number of failure criteria may be employed to determine the failure of calandria vessel, such as failure by creep, failure by high pressure, failure due to debris impingement, failure due to molten metal layer attack, failure of the drain line by hot molten debris on the vessel bottom (vessel type reactor only), and failure due to reduced external cooling.

End shield failure: Failure of an end shield is not anticipated.

Calandria vault (concrete) failure: During a severe accident with core collapse, water in the reactor vault can play an important role as a heat sink that can impact core debris behaviour within calandria vessel. A boil-off of calandria vault water below the core debris bed level could eventually cause the failure of calandria vessel wall. Calandria vault could be failed by overpressurization (differential pressure between inside and outside), by rupture disk rupture, or by corium-concrete interaction.

Containment failure: Direct containment heating is not an issue for PHWRs, as there are no accident sequences that lead to ejection of large amounts of molten material directly into containment at high pressures. Any molten material ejected from fuel channels at high pressures will have to pass through the calandria vessel and shield tank/calandria vault, and their bodies of water. Two containment failure criteria are proposed: pressure induced failure and strain induced failure.

**Activity 7:** Convene the second research coordination meeting – Review progress, resolve problems and plan future collaboration.

The second research coordination meting will be held in Mumbai on 23-26 February 2010.

**Activity 8:** CANDU 6 benchmark analysis – Phase 1 (Accident initiation to fuel channel dryout) **:** Compare the thermal-hydraulic behaviour before fuel channel dryout. Develop a strategy to generate consistent predictive results of best estimate codes at the onset of fuel channel dryout.

Important phenomena to be considered in Phase 1 are clad oxidation, fission product release from fuel to PHTS, PHTS thermalhydraulics, and fuel & fuel channel thermal response.

Parameters to be compared are PHTS pressure (outlet header), PHTS mass flow rate, maximum sheath temperature, mass inventories in the primary HTS and steam generator, the time when channel void reaches 0.9 for the entire length of channel for the first time in a channel, PHTS water level, times when liquid relief valve first opens, MSSV opens, first fuel channel rupture, and moderator rupture disk rupture, the time when the first channel uncovery occurs, secondary side pressure transient, amount of hydrogen produced, and the amount of heat transferred to the moderator form the fuel channels (kW).

**Activity 9:** CANDU 6 benchmark analysis – Phase 2 (Fuel channel dryout to core collapse) **:** Compare the prediction results for fuel channel heatup/failure/collapse, moderator boil-off, suspended debris formation, and fission product/hydrogen behaviour.

Important phenomena to be considered in Phase 2 are clad oxidation, core heatup, fission product release, aerosol transport and deposition, in-vessel cooling, vessel external cooling, molten debris heat transfer, debris fragmentation, debris dispersal, and debris coolability.

Parameters to be compared are moderator level in the calandria vessel, maximum temperature of any fuel channel and maximum sheath temperature, amount of hydrogen produced, time of core collapse, steam flow rate from the calandria vessel to the containment, the amount of heat transferred to the moderator form the fuel channels (kW), and the amount of heat transferred to the reactor vault water (kW).

**Activity 10:** CANDU 6 benchmark analysis – Phase 3 (Core collapse to calandria vessel failure) **:** Compare the prediction results for debris heat transfer, debris melting and molten pool formation, calandria vessel cooling by reactor vault water, calandria vessel heatup and failure, fission product/hydrogen behaviour, and in-vessel fuel coolant interaction.

Important phenomena to be considered in Phase 3 are fission product release, aerosol transport and deposition, in-vessel cooling, vessel external cooling, molten debris heat transfer, debris fragmentation, debris dispersal, and debris coolability.

Parameters to be compared are reactor vault water level, time of the calandria vessel water depletion, maximum calandria vessel wall temperature, maximum debris temperature, maximum heat load on the calandria vessel (kW), molten fraction of the debris, time and location of calandria vessel failure, mass of corium formed, amount of hydrogen produced, and amount of fission products released.

**Activity 11:** Convene the third research coordination meeting – Review progress, resolve problems and plan future collaboration.

**Activity 12:** CANDU 6 benchmark analysis – Phase 4 (Calandria vessel failure to containment failure) : Compare the prediction results for fission product/hydrogen concentration, ex-vessel fuel coolant interaction, molten corium concrete interaction, and containment pressurization.

Important phenomena to be considered in Phase 4 are fission product release, aerosol transport and deposition, hydrogen combustion (complete and incomplete), core-concrete interaction, wall ablation, containment natural circulation, and containment strain.

Parameters to be compared are amount of fission products released at the containment boundary, containment pressure, amount of hydrogen produced, amount of concrete ablation, amount of non-condensables and steam produced, temperature of the concrete, temperature of the debris bed, failure time of containment, mass of steam condensing on the containment wall, and the time when vault water inventory is depleted.

**Activity 13**: New experiment : Collaboratively define and potentially conduct new experiments[1] to obtain necessary experimental data. The experimental data could be used to develop new correlations or to verify the analysis results of computer codes.

The Canadian nuclear power generation industry conducted recently an experimental program in the molten fuel moderator interaction facility at Chalk River Laboratory to study the

[1] Experiments already in progress or planned in respective participant countries.

interaction between molten material ejected from a fuel channel and the moderator [2]. BARC and NPCIL of India are conducting a channel heat up test and a molten material coolant interaction test. Major results and the status of experimental programs will be presented in the second research coordination meeting.

Table 1 Schedule for All Activities

| Activity | 2008 | 2009 | 2010 | 2011 | 2012 |
|---|---|---|---|---|---|
| 1 | X | | | | |
| 2 | | X | | | |
| 3 | X | X | | | |
| 4 | | X | | | |
| 5 | | X | X | | |
| 6 | | X | X | X | |
| 7 | | | X | | |
| 8 | | X | | | |
| 9 | | X | X | | |
| 10 | | | X | X | |
| 11 | | | | X | |
| 12 | | | X | X | |
| 13 | X | X | X | X | X |
| 14 | | | X | X | X |
| 15 | | | | | X |
| 16 | | | | | X |

**Activity 14**: Benchmark analysis for experiment : Collaboratively define “standard problems” for conditions in which codes used in different countries can be used to model the same physical problems and the results can be compared. These could include, for example, core disassembly test (AECL), CHAN test – CS28-1 to 3 (AECL), channel heatup test (India), debris bed heat transfer test (India), melt-pool heat transfer test (India), melt-coolant interaction test (India) or new experimental data.

Many participants show interest in the simulation of CS28-3 experiment using their severe accident analysis codes. Detailed scope and schedule will be discussed in the second research coordination meeting.

**Activity 15:** Convene the fourth research coordination meeting – Review progress, resolve problems and plan future collaboration.

**Activity 16:** As the CRP is being completed, the IAEA together with the participating organizations will prepare a TECDOC synthesising the experience and technology advancements achieved by the CRP for dissemination to Member States. The content of the TECDOC will be planned during the CRP as the technology advancement occurs and becomes clear.

## 5. Summary and Conclusion

As part of the IAEA's overall effort to foster international collaborations that strive to improve the economics and safety of future water-cooled nuclear power plants, an IAEA CRP on "benchmarking severe accident computer codes for HWR applications" is being organized with participation of six institutes from five HWR countries. Validation of severe accident computer codes based on benchmarking against other severe accident codes is a practical method for providing the confidence in the overall performance of codes but also for the reduction of uncertainties in their predictions. One of the major outputs from this CRP will be an evaluation on the applicability of severe accident codes which are originally developed for severe accident analyses of LWRs. The CRP is being conducted on schedule with cooperation of participating institutes.

## 6. References

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Analysis of Severe Accidents in Pressurized Heavy Water Reactors, IAEA-TECDOC-1594, IAEA, Vienna (2008).

[2] T. NITHEANANDAN, et al., The results from the First High-Pressure Melt Ejection Test completed in the Molten Fuel Moderator Interaction Facility at Chalk River Laboratories, Proceedings of ICAPP 06, Reno, USA, June 4-8 (2006).

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