IAEA-TECDOC-1594

Analysis of Severe Accidents in Pressurized Heavy Water Reactors

IAEA
International Atomic Energy Agency

June 2008

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The originating Section of this publication in the IAEA was:

Safety Assessment Section
International Atomic Energy Agency
Wagramer Strasse 5
P.O. Box 100
A-1400 Vienna, Austria

ANALYSIS OF SEVERE ACCIDENTS
IN PRESSURIZED HEAVY WATER REACTORS
IAEA, VIENNA, 2008
IAEA-TECDOC-1594
ISBN 978–92–0–105908–6
ISSN 1011–4289

Printed by the IAEA in Austria
June 2008

FOREWORD

Certain very low probability plant states that are beyond design basis accident conditions and which may arise owing to multiple failures of safety systems leading to significant core degradation may jeopardize the integrity of many or all the barriers to the release of radioactive material. Such event sequences are called severe accidents. It is required in the IAEA Safety Requirements publication on Safety of the Nuclear Power Plants: Design, that consideration be given to severe accident sequences, using a combination of engineering judgement and probabilistic methods, to determine those sequences for which reasonably practicable preventive or mitigatory measures can be identified. Acceptable measures need not involve the application of conservative engineering practices used in setting and evaluating design basis accidents, but rather should be based on realistic or best estimate assumptions, methods and analytical criteria.

Recently, the IAEA developed a Safety Report on Approaches and Tools for Severe Accident Analysis. This publication provides a description of factors important to severe accident analysis, an overview of severe accident phenomena and the current status in their modelling, categorization of available computer codes, and differences in approaches for various applications of severe accident analysis. The report covers both the in- and ex-vessel phases of severe accidents. The publication is consistent with the IAEA Safety Report on Accident Analysis for Nuclear Power Plants and can be considered as a complementary report specifically devoted to the analysis of severe accidents. Although the report does not explicitly differentiate among various reactor types, it has been written essentially on the basis of available knowledge and databases developed for light water reactors. Therefore its application is mostly oriented towards PWRs and BWRs and, to a more limited extent, they can be only used as preliminary guidance for other types of reactors such as PHWRs and RBMKs with the most important potential differences in severe accident behaviour of other reactor types.

This publication provides a set of suggestions, on the basis of current international practices on how to perform deterministic analysis of severe accidents in PHWRs 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 PHWRs. An overview of the main procedural elements of accident management, i.e. emergency operating procedures and severe accident management guidelines is also introduced.

The IAEA officer responsible for this publication was S. Lee of the Division of Nuclear Installation of Safety.

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1. INTRODUCTION

1.1. Background

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 accidents have a very low probability, but may have significant consequences resulting from nuclear fuel degradation.

Severe accidents involve very complex physical phenomena that take place 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. The codes are used as a tool to support engineering judgement, based on which specific measures to mitigate the effects of severe accidents are designed. They are also used to determine accident management strategies and for probabilistic safety assessment (PSA). It is very important to use these sophisticated tools in accordance with certain rules derived from knowledge accumulated worldwide.

The aims of severe accident analysis are stated in Ref. [1]:

- To evaluate the ability of the design to withstand severe accidents and to identify particular vulnerabilities. This includes assessment of equipment that could be used in accident management and instrumentation that could monitor the course of the accident;
- To assess the need for features that could be incorporated in the plant design to provide defence in depth for severe accidents;
- To identify accident management measures that could be carried out to mitigate accident effects;
- To develop an accident management programme to be followed in beyond design basis accidents and severe accident conditions;
- To provide input for off-site emergency planning.

The term severe accident can have different connotations. For example, it sometimes refers to beyond design basis accidents, or accidents that fall below a certain cut-off frequency. In other cases, severe accidents are those that involve fuel damage or core damage. The IAEA definitions are:

Beyond design basis accidents: Accident conditions more severe than a design basis accident. It may or may not involve core degradation.

Severe accidents: Accident conditions more severe than a design basis accident and involving significant core degradation.

For a pressurized heavy water reactor (PHWR), accidents that result in damage to the reactor core fall naturally into two classes — those for which the core geometry is preserved, limited core damage accidents (LCDAs), and those for which the core geometry is lost, severe core damage accidents (SCDAs). SCDAs are within the above definition of a severe accident, and are the primary focus of the guidance provided in this publication. On the other hand, LCDAs are typically considered as part of the design basis for PHWRs. As such, there are provisions for their prevention and mitigation, and severe accident management is not required. While LCDAs are not within the definition of severe accidents, for completeness, and because LCDAs are typically precursors to SCDAs, their phenomenology will be described in section 4.

There is no widespread agreement on the best approach to severe accident analysis and acceptance criteria. Severe accident analysis is generally carried out using best estimate assumptions, data, methods and decision criteria. Where this is not possible, reasonably conservative assumptions should be made which take account of the uncertainties in the understanding of the physical process being modelled.

The analysis would typically involve a multi-tiered approach using different codes, including detailed system and containment analysis codes, mode simplified risk assessment and 'separate effects' codes, and source term and radiological impact studies. Use of a full selection of codes will ensure that all the expected phenomena are adequately analysed.

Accident management is a set of actions during the evolution of beyond design basis accidents to prevent the escalation of the event into a severe accident, to mitigate the consequences of a severe accident, and to achieve a long term safe stable state.

The basic features of the accident management programme are preventive and mitigatory features, accident progression and degree of severity, assessment of plant vulnerabilities and capabilities, accident management strategies, information needs, plant equipment performance and material support needs.

Severe accident analysis and accident management are addressed in a number of IAEA publications [1−9]. IAEA Safety Requirements publication on Safety of Nuclear Power Plants: Design [2] specifies that "Consideration shall be given to the severe accident sequences, using a combination of engineering judgement and probabilistic methods, to determine those sequences for which reasonably practicable preventive or mitigatory measures can be identified." The methodology for deterministic analysis of severe accidents is also addressed in the IAEA reports devoted to PSA level 2 analysis in Refs [10, 11]. In order to ensure that the most important design accidents can be mitigated effectively, the need to consider design accidents beyond the design basis is also addressed in Ref. [12].

Among IAEA publications, in particular, the IAEA Safety Report [4] provides practical guidance for performing accident analysis based on the present good practices worldwide. All the steps required to perform such analysis are covered in the report, e.g. selection of initiating events, acceptance criteria, computer codes and modelling assumptions, preparation of input data and presentation of calculation results are covered. Specific suggestions applicable for pressurized water reactors (PWRs) and PHWRs are given in Refs [5, 6]. The reports cover both design basis accidents and beyond design basis accidents including severe accidents. However, only basic guidance is provided for severe accident analysis. Therefore, a specific report on accident analysis for severe accident is needed and provided through this report.

Recently, the IAEA has developed a Safety Report [13] on the severe accident analysis for in-vessel and ex-vessel phenomena. This publication provides a description of factors important to severe accident analysis, an overview of severe accident phenomena and the current status in their modelling, categorization of available computer codes, and differences in approaches for various applications of severe accident analysis. Although the reports do not explicitly differentiate among various reactor types, they have been written essentially on the basis of available knowledge and databases developed for light water reactors (LWRs). Therefore their application is mostly oriented towards pressurized water reactors (PWRs) and boiling water reactors (BWRs), and to a more limited extent, they can be only used as a preliminary guidance for other types of reactors such as PHWRs and high-power boiling reactor with pressurized channels (RBMKs) with the most important potential differences in severe accident behaviour of other reactor types. Thus, it is necessary for the IAEA to develop an analysis of severe accident in PHWRs to support the Member States operating or constructing the PHWRs, with the best international practice, at individual NPPs.

The IAEA Safety Report on implementation of accident management programme for NPPs [8] provides a description of the elements which should be addressed by the team responsible for preparation, development and implementation of a plant specific accident management programme at an NPP. The issues addressed include formation of the team, selection of accident management strategies, safety analysis required, evaluation of the performance of plant systems, development of accident management procedures and guidelines, staffing and qualification of accident management personnel, and training needs.

This publication complements two IAEA Safety Reports listed above and supports the IAEA Safety Guide on Safety Assessment and Verification for Nuclear Power Plants [1] and the Safety Guide on Severe Accident Management Programmes for Nuclear Power Plants [7].

1.2. Objectives and scope

The objective of this publication is to provide information on the analysis of severe accidents in PHWRs, on the computer codes available for analysis of severe accidents (both in the reactor coolant system as well as in the containment), their capabilities and limitations, and the development of accident management programmes.

This publication provides a set of suggestions, on the basis of current international practices, on how to perform deterministic analysis of severe accidents 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 the 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 guidance for PHWRs. An overview of main procedural elements of accident management, i.e. emergency operating procedures (EOPs) and severe accident management guidelines (SAMGs) is also introduced.

This publication is intended primarily for code users or reviewers involved in analysis of severe accidents for PHWRs. In the preparation of this publication, it was assumed that the users of this publication have some knowledge of severe accident phenomena and also of the use of computer codes for accident analysis but may have not previously been actively involved in severe accident research or analysis activities. Although this publication is intended as a self-standing report, it is suggested for the user to read the previous general reports on the accident analysis [5, 6].

This publication focus primarily on the existing PHWRs, however, it may also be applicable to future PHWRs such as Advanced CANDU reactor (ACR) or Indian advanced PHWRs.

1.3. Structure

The structure of the present publication is as follows. Section 2 describes briefly the design characteristics of the PHWRs, such as reactor, heat transport system, moderator system and special safety system. Section 3 provides an explanation of phenomena important to the analysis of the in-vessel and ex-vessel phases of severe accidents, including thermal-hydraulics of the primary heat transport system (PHTS) during the accident, fuel and fuel channel behaviour, reactor core and containment behaviour, fission product transport during a severe accident and other important factors. Basic steps in performing severe accident analysis are described in Section 4, including selection of methodology, computer codes, appropriate models for reactor channel, moderator and containment and nodalization scheme. Basic steps in developing input data and presentations of results are also addressed. Section 5 provides failure criteria and their effects on simulators for heat transport system, fuel, fuel channel, calandria vessel and the containment. Section 6 describes applicable areas of severe accident analysis such as PSA and SAMGs. Section 7 introduces basic approaches to severe accident management. Proposed criteria for transition from EOPs to SAMGs, a method to select and develop the accident management strategies, and basic steps to develop the SAMGs are elaborated in this section with practical examples. Section 8 provides the summary and conclusions of the report. Recent experimental programmes such as system thermal-hydraulics, fuel channel and fission product behaviours are summarized in Appendix I. A discussion of frequently used computer codes and the status of their validation are provided in Appendix II. Appendix III illustrates severe accident progression in a CANDU reactor. Annex I elaborates various examples of the severe accident analysis results of CANDU and Indian PHWRs. Annex II provides an overview of computer codes for design basis accidents and beyond design basis accidents developed by Atomic Energy of Canada Ltd (AECL), Canada.

2. DESIGN CHARACTERISTICS OF PHWRS

This section briefly outlines some of the PHWR plant features with special reference to its severe accident mitigation features. Distinctive /inherent safety related characteristics of PHWRs are as follows:

- On power refuelling helps in maintaining low excess reactivity. Thus no/low poison in moderator.

- There are two independent fast acting shutdown system of diverse nature.

- High pressure and high temperature coolant is separated from the low pressure and low temperature moderator. Reactivity and shutdown mechanisms in the moderator are unaffected by the disturbance in the coolant loop.

- Large and subcooled inventory of moderator can act as an ultimate heat sink. Thus preventing severe core degradation under accidents. Water surrounding the calandria in calandria vault/shield tank system can hold the fuel channels in debris within the calandria under loss of coolant accident (LOCA) with failure of emergency core cooling system (ECCS) and moderator system.

Thus a large inventory of water and multiple barriers have inherent capability to prevent corium debris falling on the concrete floor leading to core-concrete interaction under severe accident.

2.1. Canadian PHWRs

2.1.1. CANDU

2.1.1.1. Reactor and auxiliaries

Fuel channels

The CANDU 6 reactor core has 380 fuel channels. These fuel channels span the calandria horizontally between the two end-shields and are located within two equal figure-of-eight loops that can be isolated from each other under certain accident conditions. Each fuel channel consists of a zirconium/niobium alloy pressure tube surrounded by a zircaloy-2 calandria tube with CO_2 gap (annulus gap) in between. Each fuel channel contains twelve fuel bundles. The fuel bundles are made up of 37 zircaloy-4 tubes containing natural UO_2 pellets. The fuel channels exhibit a large variability in total channel powers and in axial bundle powers. Therefore, any heatup of the core due to loss of heat sinks is not expected to be monolithic.

Inherent features of CANDU reactors ensure that the fuel channels maintain their integrity and that the fuel remains below melting even when heat removal by coolant flowing through the bundles is not available. The low temperature, low pressure D_2O moderator surrounding the fuel channels provides an effective heat sink under some accident scenarios, so accident progression to severe core damage depends on the availability of the moderator and heat removal by the moderator cooling system.

Since the fuel channels are located horizontally in the calandria, their failure mode will be different from those for the vertical fuel assemblies in PWRs. The long fuel channels will sag when they are heated and dislocate to the lower rows where deformation of channel configuration would occur, causing severe core damage.

Calandria

The calandria houses the fuel channels that span it horizontally and reactivity mechanisms that span it vertically. The calandria shell is a horizontal, stepped single walled cylinder made of austenitic stainless steel. The ends of the calandria shell are enclosed by stainless steel tube-sheets which form a common boundary between the calandria and the end-shields.

The heavy water moderator in the calandria is a unique CANDU feature which provides a passive heat sink for some accident scenarios. A cover gas system maintains a pressure of less than 27.6 kPa (4 psia) above the moderator. Over pressure protection is provided by rupture discs at the top of calandria at the upper ends of four pressure relief pipes which are designed to provide an adequate discharge area for heavy water flow to the containment (boiler room) during a postulated simultaneous pressure tube/calandria tube rupture at full system pressure.

End shields

The two end-shields are integral parts of the reactor assembly. Each end shield is a cylindrical shell bounded by the calandria tubesheet and the fuelling tubesheet and spanned by 380 lattice tubes. To provide biological shielding and cooling, the end shields are filled with carbon steel balls and demineralized light water cooled by the end-shield cooling system. The end-shields are not part of the moderator pressure boundary, but may provide a thermal barrier to severe accident progression by resisting thermal attack by any debris contained within the calandria shell under severe accident conditions. The end-shields share a common cooling system with the calandria vault.

When the corium pool level is high, the molten corium overflows from the main shell of the calandria into the sub-shell and contacts the calandria tubesheet which is in contact with the end shield. Then the tubesheet will be easily heated up and may fail, allowing a molten corium to flow into the end shield which is filled with steel balls. It is not expected, however, that the corium will be released into the fuelling machine room after melting all of the steel balls in the end shield.

Calandria vault

The calandria vault is built of ordinary concrete and is filled with light water which functions as a biological shield under normal operating conditions and a passive heat sink under certain severe accident scenarios. For example, if hot dry debris is collected in the bottom of calandria after core disassembly and heats up, the calandria vault water will remove the decay heat from the calandria through the calandria wall. External vessel cooling considered to be an important accident management programme in PWRs, is inherent in the CANDU plant.

The calandria vault floor has a significant spreading area for any potential core debris. The corium pool formed in the vault will not touch the calandria. There are two layers of concrete below the calandria vault floor. The thick concrete floor provides a significant time (many days) for ablation by the hot molten debris.

Reactor building ventilation system provides venting of calandria vault, end shields, and delay tanks. In addition to this venting, rupture discs are provided on the combined vent lines to relieve over pressure caused by boiling of the vault water or failure of the cover gas system.

Shield cooling

The end-shield cooling system is common to the two-end-shields and the calandria vault. The coolant flow is modulated through its heat exchangers to maintain specified temperature distributions in the end-shields and the vault. At normal full power operation, the recirculated cooling water removes the anticipated cooling load of about 7.3 MW for CANDU 6. While lower than the decay heat under normal conditions, the heat removal capacity is enhanced under higher coolant temperatures. For example, if a loss of moderator cooling contributes to a severe core damage and dry hot debris in the calandria tank, the heat removal by the shield water cooling system becomes the dominant heat sink and is expected to successfully do so at elevated water temperatures.

2.1.1.2. Heat transport system and auxiliaries

The PHTS is comprised of two loops. Each loop serves 190 of the 380 reactor fuel channels. The fuel channels are divided for this purpose about the vertical centre plane of the reactor. Each loop contains two pumps, two steam generators, two inlet headers and two outlet headers in a 'figure-of-eight' arrangement. Feeders connect the inlet and outlet ends of the fuel channels to the inlet and outlet headers respectively. Pressurized heavy water circulates through the reactor fuel channels to remove the heat produced by fission of natural uranium fuel. The heat is transported by the reactor coolant to steam generators where it is transferred to light water to generate steam, which subsequently drives the turbine generators. The steam generators, PHTS pumps and headers are located above the reactor; this permits the heat transport system coolant to be drained to the head elevation for maintenance of the PHTS pumps and steam generators, and also facilitates thermosyphoning (natural circulation) when the PHTS pumps are unavailable and the reactor is shut down.

Two closed loops are generally interconnected with isolation valves. An automatic loop isolation reduces the rate of reactor coolant loss in the event of a loss of coolant accident. Isolating valves are automatically closed when PHTS pressure drops to below certain pressure whether or not a loss of coolant accident has occurred and whether or not the ECCS is implemented.

The reactor outlet headers at one end of reactor are connected to a common pressurizer. The pressurizer is the principal component in the pressure control of the heat transport system.

2.1.1.3. Moderator system

The heavy water moderator in the calandria is used to thermalize fast neutrons produced by fission. The heavy water moderator is circulated through the calandria and moderator heat exchangers to remove the heat generated in the moderator during reactor operation; moderator heat is rejected to the recirculated cooling water system. The operating pressure at the moderator free surface, which is maintained within a specified range above the top row of fuel channels, is slightly above atmospheric.

The moderator system is fully independent of the heat transport system. The moderator system includes two 100 per cent pumps and two 100 per cent tube and shell heat exchangers. The moderator system head tank maintains the moderator level in the calandria within the required range by accommodating moderator swell and shrink resulting from temperature fluctuations.

The heavy water in the calandria functions as a heat sink in the unlikely event of a loss of coolant accident coincident with failure of ECCS. The capability of this heat sink is assured by controlling the heavy water temperature in the calandria within specified limits.

2.1.1.4. Emergency core cooling system

The function of the ECCS is to provide an alternate means of cooling the reactor fuel, in the event of an LOCA which depletes the normal coolant inventory in the PHTS to an extent that fuel cooling is not assured.

There are three stages of ECCS according to the operating pressure: high, medium, and low pressure. As soon as the PHTS header pressure drops to below 4.04 MPa(g) (586 psig), water flows from the high pressure ECCS tanks into headers of the failed loop. Though the flow rate from the water tanks depends on the break size, the high pressure injection lasts for a minimum 2.5 minutes for a 100% header break size.

The medium pressure injection valves open when the low high pressure ECC water tank level signal on. The ECC pump injects water from the dousing tank to all reactor headers when the pump discharge pressure is higher than the reactor header pressure. The medium pressure injection lasts for a minimum of 12.5 minutes for a 100% header break size.

As the dousing tank water depletes, the medium pressure injection is automatically terminated and low pressure injection starts. Long term low pressure injection is provided by collecting the mixture of H_2O and D_2O from the reactor building sump and recirculating it into the PHTS via the emergency cooling system heat exchangers. Screens and strainers are provided at each of the pump suction inlet lines to prevent debris entrainment. The low pressure ECCS circuit is designed to operate at least three months following a LOCA. Figure 1 shows a simplified diagram of the ECCS. For large breaks, the ECCS recovery heat exchanger is the main heat sink. For small breaks, the steam generator continues to be the main heat sink.

2.1.1.5. Shutdown system

The CANDU 6 reactor incorporates two diverse, passive, shutdown system which are independent of each other and from the reactor regulating system. Each shutdown system is capable of tripping the reactor and has sufficient negative reactivity to maintain it in a shutdown state. Each shutdown system is designed to have no effect in tripping the reactor from the failure of support system including dual computer control failure. Especially even though any class of electrical power fails, shutdown system has no effect in tripping the reactor.

Shutdown system No.1 consists of mechanical shutdown rods which drop into the core when a trip signal de-energizes clutches which hold them out of the core.

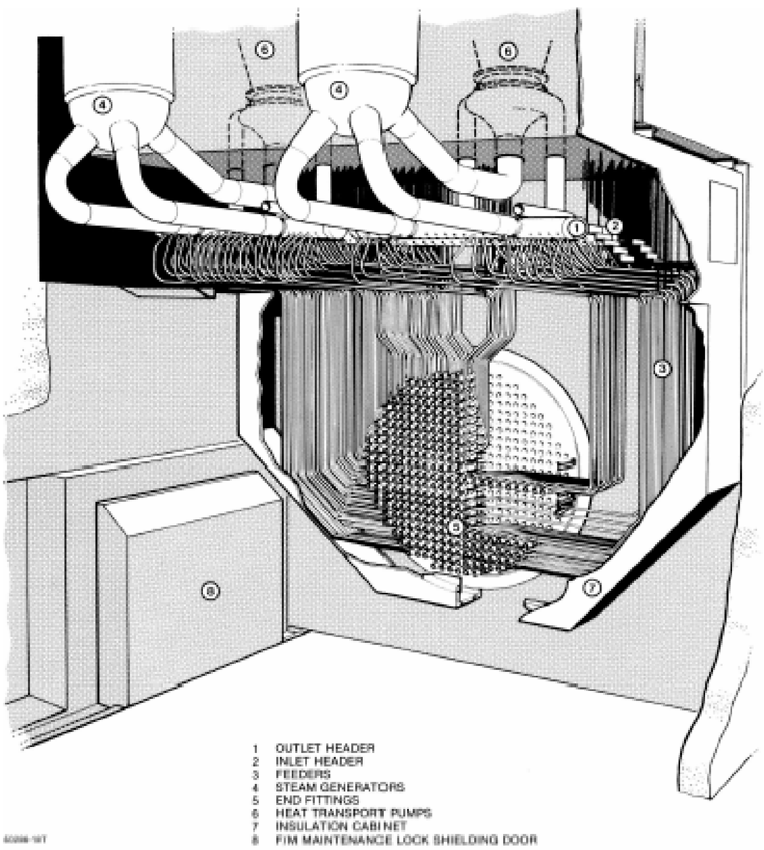


FIG. 1. CANDU 6 reactor assembly.

Shutdown system No.2 injects a concentrated solution of gadolinium nitrate into the low pressure moderator to quickly render the core subcritical. The injection is initiated by opening fast acting valves to pressurize the individual poison tanks associated with each of the injection nozzles with helium.

2.1.1.6 Containment system

1) Single unit CANDU containment system

The chief function of containment system is designed to confine the release of fission products into the environment during accident, to within the acceptable limits. The containment system consists of a leaktight envelope around the reactor and associated nuclear systems, and includes containment isolation systems, containment atmosphere energy removal systems and cleanup systems. Hydrogen control is provided in the newer, larger PHWRs in order to cater for the long term build-up of hydrogen resulting from radiolysis after a LOCA, and for severe accident such as LOCA plus loss of ECCS.

The CANDU 6 single unit containment is a pre-stressed concrete building consisting of three principal structural components: a base slab, a cylindrical wall, and a spherical segmental dome. The concrete provides strength and shielding. The inner surface of reactor building is lined with an epoxy coating, instead of steel plate, to improve leaktightness.

Airlocks are provided for personnel and equipment transfer to the interior of the reactor building without breaching containment. They are designed to withstand LOCA or main steam line break conditions with the door on the reactor building end open or closed. Electrical and pneumatic interlocks ensure that only one door of an airlock can be open at a time. Each door is provided with dual inflatable seals whose condition, together with door closure, is continuously monitored to assure that a fully capable barrier exists at all times.

The containment structure is penetrated by pipes and ducts to allow passage of process fluids and ventilating air during normal operation. The function of the containment isolation system is to close these pipes and thus seal the containment envelope.

The spray dousing system is designed to limit the magnitude and duration of containment over pressure caused by a LOCA or a main steam line break inside the reactor building. The system is automatically initiated when the containment pressure exceeds 14 kPa(g) (2 psig). The flow rate is designed to limit the maximum building pressure to less than the design pressure following a guillotine rupture of the heat transport system header. When containment pressure decreases to 7 kPa(g) (1 psig), the dousing valves close in 7 seconds. According to the size of the break, there is a continuous or cyclic operation of dousing valves. Manual operation of the dousing system is possible from either control centre.

The dousing tank is located directly below the dome of the reactor building and holds water for both dousing and medium pressure ECCS supply. Separation is achieved by placing the inlet of the dousing downcomers at the higher position than the inlet of ECCS line above the bottom of the tank.

Long term pressure control and heat removal, after the dousing water is depleted, is achieved through local air coolers. Local air coolers are located in the various places in the reactor building such as steam generator rooms and fuelling machine rooms.

In recent CANDU 6 plants (Wolsong 2, 3, and 4, Qinshan 1 and 2), a network of hydrogen igniters is provided to burn any local concentrations of hydrogen formed in the long term post-LOCA, and in dual failures (LOCA plus loss of ECCS), preventing hydrogen from reaching deflagration or detonation levels by using controlled ignition as soon as the hydrogen concentration reaches flammable limits.

2) Multi unit vacuum building containment system

In the multiunit vacuum system, four or eight reactors, each with its own local containment, are connected by large scale ducting to a separate, common vacuum building kept, as its name implies, at near zero absolute pressure. Should steam be released from a pipe break in the reactor building, the pressure causes banks of self-actuating valves connecting the vacuum building to the ducting to open. The steam and any fission products are then drawn along the duct by suction; the steam being condensed by dousing in the vacuum building resulting in soluble fission products such as iodine being washed out. The dousing is passively actuated by the difference in pressure between the main body of the vacuum building and the vacuum chamber; it does not require electrical power or compressed air supplies for its operation.

This concept, which was developed as a result of the economic benefits of multiunit sites, has a number of unique safety characteristics:

- After an accident, the entire containment system pressure is sub-atmospheric for several days; thus leakage is inward, rather than outward.

- The overpressure period in the reactor building is very short, of the order of a couple of minutes, and therefore the design pressure is reduced and design leak rate can be increased relative to single unit containment.

- Even if the vacuum building is not available, the large interconnected volume of the four or eight reactor building provides an effective containment.

- Several days after an accident, when the vacuum is gradually depleted and the containment pressure rises towards atmospheric pressure, an emergency filtered air discharge system is used to control the pressure.

2.1.2. ACR design

The ACR design is based on the use of modular horizontal fuel channels surrounded by a heavy water moderator, the same feature as in all CANDU reactors. The major innovation in ACR is the use of slightly enriched uranium fuel and light water as the coolant, which circulates in the fuel channels. This results in a more compact reactor design and a reduction of heavy water inventory, compared to CANDU reactors that employ natural uranium as fuel and heavy water as coolant. Figure 2 shows the relative size of ACR core compared to other CANDU reactors. The design also features higher pressures and temperatures of reactor coolant and main steam, thus providing an improved thermal efficiency than the existing CANDU plants.

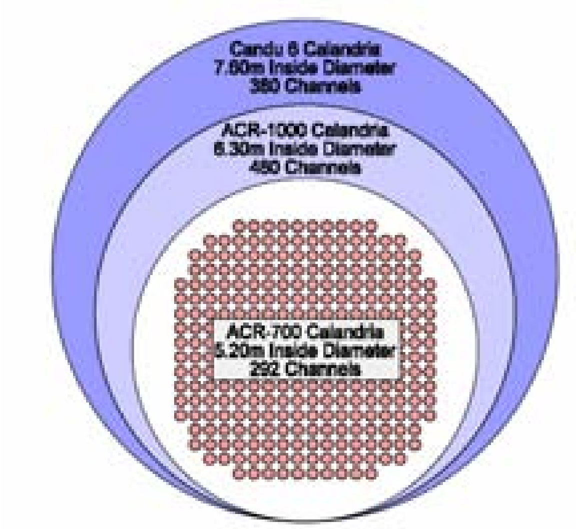


FIG. 2. ACR reactor size versus other CANDU reactors.

The above changes and other evolutionary design improvements are well supported by the existing knowledge base and build on the traditional characteristics of the CANDU system, including: proven, simple and economical fuel bundle design; on-power fuelling; separate cool, low pressure moderator with back-up heat sink capability; and low neutron absorption for good fuel utilization. The safety enhancements made in ACR encompass safety margins, performance and reliability of safety related systems. In particular, the use of the CANFLEX fuel bundle, with lower linear rating and higher critical heat flux, permits increased operating and safety margins of the reactor. Passive safety features draw from those of the existing CANDU plants (e.g. the two independent shutdown systems) and other passive features are added to strengthen the safety of the plant (e.g. a gravity supply of emergency feedwater to the steam generators). Additional important aspects of the ACR include: the adoption of state-of-the-art engineering methods and tools and construction techniques; the consideration of the feedback from the existing CANDU plants to improve operability and maintainability; and a full integration and optimization of nuclear steam plant and balance of plant.

Key ACR safety systems are:

- Two fast-acting, fully capable, diverse, and separate shutdown systems, which are physically and functionally independent of each other and from the reactor regulating system;
- ECCS consisting of the emergency coolant injection system and the long term cooling system, and
- Containment system: strong containment structures, containment isolation system, containment heat removal system, etc.
- Systems that provide reliable support services are referred to as safety support systems, which include:
- Reserve water system (see Fig. 3) provides an emergency source of water by gravity to the containment sumps for recovery by the long term cooling system, to the steam generators, moderator system, shield cooling system, and heat transport system if required.
- Two auxiliary feedwater pumps.
- Electrical power systems supply. The safety related portions of the systems are seismically qualified and consist of redundant divisions of standby generators, batteries, and distribution to the safety related loads.
- Recirculated cooling water system circulates demineralized cooling water to different loads in the plant. The safety portions of the recirculated cooling water system are seismically qualified and comprised of two redundant, closed-loop divisions. One division is sufficient to cool the plant in a shutdown state.
- Raw service water system disposes of the heat from the recirculated cooling water system to the ultimate heat sink. The safety related portions of the system are seismically qualified and comprised of two redundant, open-loop divisions.
- Compressed air system provides instrument air and breathing air to different systems in the plant.
- Chilled water system supplies water to air conditioning and miscellaneous equipment.
- Secondary control area contains monitoring and control capability to shut down the reactor and to maintain the plant in a safe shutdown condition following events that may render the main control room unavailable.

The safety enhancements made in the ACR encompass safety margins, performance, and reliability of safety related systems.

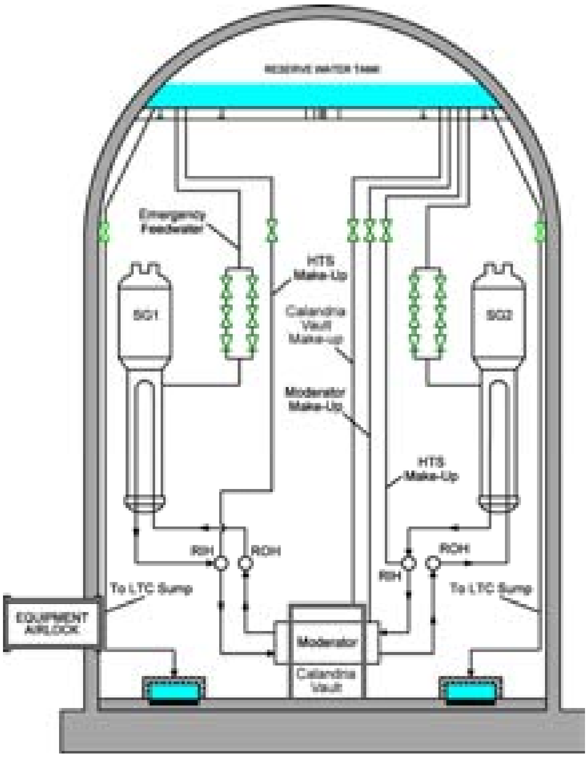


FIG. 3. Schematic diagram of reserve water system.

2.2. Indian PHWRs

India has two kinds of PHWRs: 220MW(e) PHWR and 540MW(e) PHWR. In addition, design of 700 MW(e) PHWR is in progress. The following sections describe some of the important features of these reactors.

2.2.1. Primary heat transport system

A standardized 220 MW(e) Indian PHWR has 306 horizontal fuel channels in a figure-of-eight loop configuration, with one pair of steam generators and pumps in each bank of PHTS loop. In case of 220 MW(e) each fuel channel has 12 fuel bundles and each bundle consists of 19 fuel elements. A 540 MW(e) PHWR has two PHTS loops with 392 fuel channels. Each loop has one pair of steam generators and pumps. In 540 MW(e) number of bundles in a channel is 13 and a bundle consists of 37 fuel elements.

2.2.2. Emergency core cooling system

The ECCS assures fuel cooling following a loss of coolant accident by refilling the core and recirculating coolant through the core for long term heat removal from the fuel. The heat picked up by ECCS water is rejected to the process water in the ECCS heat exchangers.

In the Indian 220 MW(e) PHWRs, the ECCS incorporates:

- High pressure heavy water injection;
- Intermediate pressure light water injection;
- Low pressure long term recirculation.

The high pressure heavy water injection is provided by a system of accumulators containing D_2O pressurized by nitrogen. Intermediate pressure light water injection is provided by a system of two accumulators and a pressurized nitrogen gas tank. When the PHTS pressure falls further, low pressure light water injection occurs, followed by recirculation provided by pumps. The ECCS pumps draw water from the suppression pool and this water is cooled in the ECCS heat exchanger by process water system.

The provision of a heavy water accumulator for the initial high pressure injection permits its use during certain non-LOCA transients to make up for fast shrinkage in the PHTS, but without downgrading the PHTS heavy water.

Injection of ECCS water takes place through two of the four headers, which are selected on the basis of the size and location of the break. In the case of small breaks, or breaks on the outlet header side, in which the flows continue in the normal direction, injection takes place into the reactor inlet headers. In large breaks on the inlet header side, which result in reversal of flow in half of the core, injection is directed to the inlet and outlet headers on the side away from the break so as to assist the flow direction. Selection of the headers takes place automatically and is based on a pressure signal between the headers, indicating the flow direction. All actions up to and including the establishment of long term recirculation are automatic. The ECCS of 540 MW(e) PHWRs is similar to 220 MW(e) PHWRs.

2.2.3. Shutdown systems

Indian PHWRs have two diverse, passive shutdown systems which are independent of each other like CANDU 6. A typical 540 MW(e) unit consists of;

- Shutdown system No.1 of 28 cadmium sandwiched stainless steel rods of worth 72 mk, which move into the low pressure moderator system;
- Shutdown system No.2 injects the liquid poison into the moderator.

2.2.4. Containment system

Double containment is employed in Indian PHWRs and the containment structures are of concrete. The primary containment is a pre-stressed concrete structure, consisting of a perimeter wall topped by a pre-stressed concrete dome. The outer or secondary containment envelope is a reinforced, cylindrical concrete wall topped by a reinforced concrete dome. The primary containment uses an epoxy coating to form a liner for added leak tightness and ease of decontamination.

Primary containment houses reactor and its various associated systems. In case of postulated loss of coolant accident, the containment holds the fission products released from the reactor and/or its coolant in such a way that the release of fission products to the environment is within the acceptable and prescribed limits. The secondary containment does not house any plant system. The high enthalpy steam lines passing through the secondary containment space are designed to pass through pipe sleeves to eliminate the need for designing the secondary containment for postulated pipe break of steam line resulting in internal pressure and temperature loading. The nuclear containment structures provide biological shielding to limit the radiation dose to the public and plant personnel in the case of design basis accidents.

The annulus between the two containment walls is maintained under vacuum with a provision of continuous monitoring for any accidental release of fission products to the annulus space from the inner primary containment. This double containment design ensures almost zero ground release to the environment. Another notable feature of the containment structure is that the double barrier also serves to effectively resist external and internal missile impact loads.

3. IMPORTANT PHENOMENA TO CONSIDER FOR SEVERE ACCIDENTS

Several experiments in PHWRs have been performed to elucidate the phenomena for severe accidents in PHWRs, and to provide data for the development and validation of computer codes to model severe accident behaviour. A brief description of the experimental facilities is introduced in Appendix I.

For a PHWR, accidents that could result in damage to the reactor core fall naturally into two classes — those for which the core geometry is preserved, limited core damage accidents (LCDAs), and those for which the core geometry is lost, severe core damage accidents (SCDAs).

The LCDAs can involve single channels or the entire core. For example a feeder break can result in overheating of the fuel in the affected channel. Or a LOCA with loss of ECCS could lead to widespread fuel damage. In both cases, however, the presence of the moderator as a secondary heat sink prevents failure of the fuel channels and core degradation. In this sense, the LCDAs are distinct from accident sequences for a LWR as they can involve fuel damage without core relocation.

For an accident to proceed to severe core damage there must be multiple fuel channel failures leading to significant core degradation. Such a situation is only possible if the heat sink behaviour of the moderator is lost. Typically, SCDA initiates as a LCDA, but there is no moderator cooling to remove heat and there is no moderator make-up. In this case, the moderator boils off allowing uncovered fuel channels to heat up and fail. Thus the early stages of a SCDA for a PHWR are different from those of a LWR in that the presence of the heavy water moderator slows down the progression of core disassembly. Once the core has become fully disassembled and has collected on the bottom of the calandria vessel, subsequent behaviour is similar to that of a LWR with the core relocated into the lower vessel head.

3.1. Limited core damage accidents

As mentioned above, LCDAs in a PHWR can be further subdivided into two types, single channel events and full core events.

The distributed nature of the core of a PHWR, with a network of feeder pipes circulating coolant between large diameter headers and individual fuel channels, requires consideration of accident sequences for which the coolant flow to an individual channel is disrupted. These are commonly referred to as single channel events. These accidents can be initiated by partial or complete blockage of the flow, or a break in a feeder or a fuel channel (see Fig. 4). Since only a limited number of channels are instrumented, the event can continue at full power until a reactor trip point is reached (containment pressure, for example). The outcome of the event depends on the degree to which cooling for the affected channel is reduced. In the extreme case of complete blockage or flow stagnation, the fuel in the affected channel can be damaged. The consequences of a single-channel event are determined by the extent of fuel damage in the affected channel. Assessments are performed to ensure there are no phenomena that can lead to propagation to other fuel channels.

For the fuel in more than one fuel channel to be damaged, the primary cooling flow must be interrupted and emergency core cooling must be impaired. Interruption of the primary coolant flow to more than one channel requires an initiating event such as a LOCA that will trip the reactor. Thereafter, these events proceed at decay power. The presence of a secondary heat sink, in the form of the moderator around every fuel channel, limits the consequences of these accidents. Under extreme conditions such as large LOCA with loss of ECCS, the fuel in many fuel channels can undergo a high temperature transient. Heat loss from the fuel to the moderator, via the pressure and calandria tubes, prevents gross melting of the fuel and preserves the channel core geometry of the reactor.

3.1.1. Single channel events

3.1.1.1. Thermal-hydraulic behaviour

When the coolant flow to an individual channel is reduced, the remaining flow will be increasingly converted to steam as the mass flow decreases. The reduced flow is still capable of removing heat from the fuel, and high reductions in mass flow (equivalent to blockages greater than 90% of the flow area) are required before fuel temperatures increase significantly. The events of interest from a core damage perspective are those that result in significant reduction in mass flow with the consequent temperature excursions for the fuel and fuel channel. The primary thermal phenomena are the heat transfer within the fuel, from fuel to coolant, from coolant to pressure tube, thermal radiation, and the hydraulic phenomenon of concern is the resulting phase separation of the coolant in the channel and its feedback on thermal behaviour. Eventually the fuel channel will fail allowing for reintroduction of coolant from the unbroken or unblocked side. This cools the fuel and trips the reactor as for a small-break LOCA.

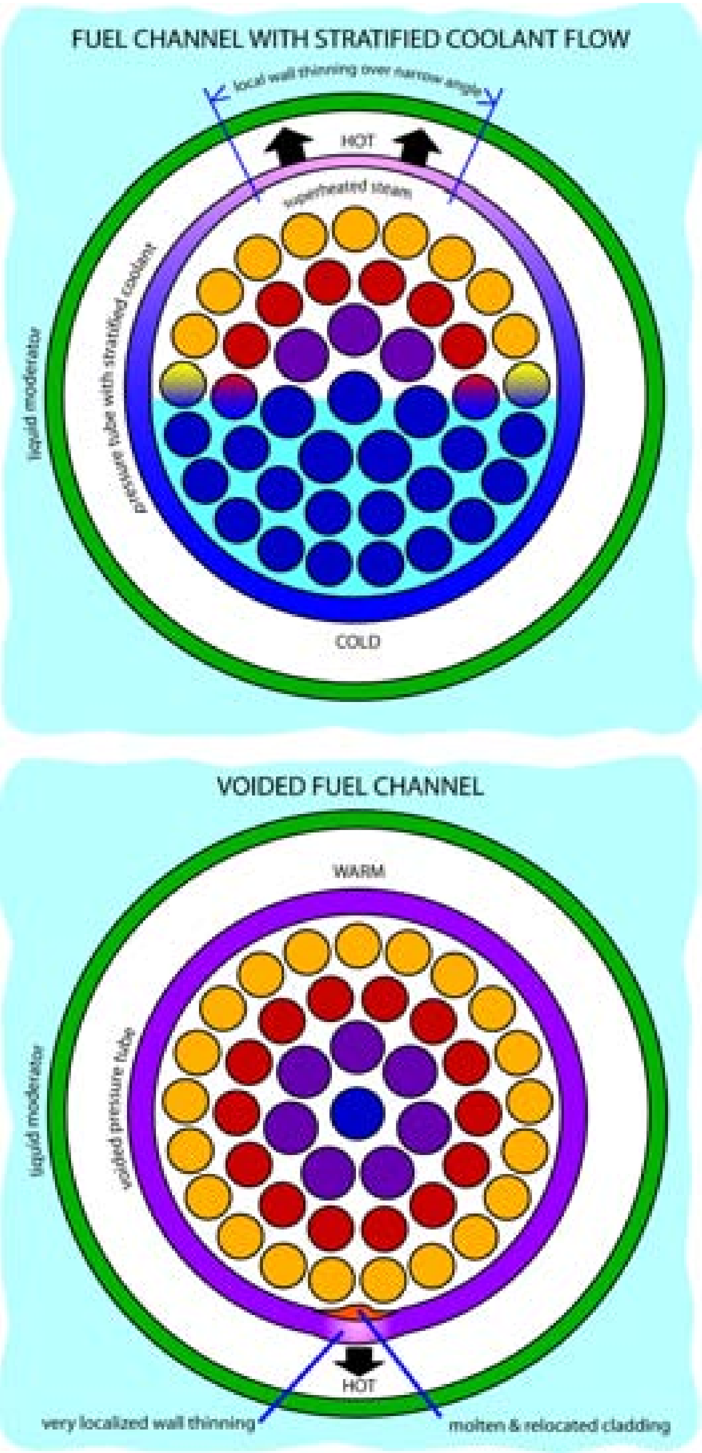


FIG. 4. Examples of single channel events (above) and pressure tube failure in single channel events (below).

3.1.1.2. *Fuel behaviour*

For severe reductions in coolant flow in a single channel, the fuel is subjected to a rapid temperature excursion. The phenomena of interest are those that lead to fuel bundle deformation and to failure of the fuel cladding. These include fuel element bowing, zircaloy oxidation, and Be-braze penetration. The high coolant pressure tends to prevent clad ballooning. If the affected pressure tube does not fail due to circumferential temperature gradients, there can be clad melting and relocation onto the pressure tube, which is predicted to cause failure. If delayed failure is assumed, there can be fuel melting.

3.1.1.3. *Fuel channel behaviour*

As heat is transferred to the pressure tube, its temperature will rise and the tube will start to deform [14] under the influence of the full-system pressure as shown in Fig. 4. Stratification of the coolant in the channel will keep fuel temperatures lower in the bottom of the fuel channel and lead to a top to bottom circumferential gradient. Under these conditions, the top of the pressure tube will deform more quickly and fail [15–18]. If the temperature around the tube is more uniform (perhaps due to rapid expulsion of the coolant with little stratification), the pressure tube will expand uniformly (balloon) into contact with the calandria tube.

Contact of the pressure tube with the surrounding calandria tube provides support and cools the pressure tube. The fuel will continue to heat up and either molten zircaloy (most likely) or a combination of molten zircaloy and UO_2 (less likely) will form and relocate to the bottom of the fuel channel. The combined pressure and calandria tubes will then heat up locally and fail. In principle, localized 'hot spots' on the pressure tube wall could also be caused by contact of fuel appendages or of bowed fuel elements. These hot spots could only lead to earlier failure of the pressure tube, thereby avoiding the formation of molten clad or fuel.

Calandria tubes are thinner than the pressure tubes, but are directly cooled by the surrounding liquid moderator. If the pressure tube fails prior to ballooning into contact with the calandria tube, the annulus between the tubes will become pressurized. The calandria tube is not expected to survive pressurization, particularly when taking into consideration impingement of superheated steam and hot fuel material.

3.1.1.4. *Reactor core behaviour*

One of the primary considerations in analyzing single-channel LCDAs is the potential for propagation of the event to additional channels (see Fig. 5). Three means of propagation are assessed. The first is failure of the initiating channel leading to failure of the calandria vessel. The second is propagation of the initiating failure to neighbouring channels. The third is impairment of the ability to shutdown by damage to control and shutoff rods and by displacement of poisoned moderator by heavy water coolant.

If a fuel channel fails, the coolant discharging into the calandria vessel forms a large steam bubble that displaces the liquid moderator and pressurizes the calandria vessel. The ejection of hot fuel materials can deposit additional energy into the moderator, producing more steam (see Fig. 6). If there is liquid material in the effluent, the possibility of a steam explosion should be considered (although it is believed to be unlikely for high pressure ejection). Large relief ducts are provided for the calandria vessel, which are designed to accommodate sustained coolant discharge from the primary system. There is, however, an initial pressure spike in the calandria vessel, when the rapidly growing steam bubble displaces the incompressible liquid. This pressure spike is attenuated by the collapse of neighbouring calandria tubes onto their pressure tubes. The resulting early loads on the walls of the calandria vessel are approximately the value of the calandria-tube collapse pressure, and the vessels of PHWRs have been shown to be able to withstand these loads.

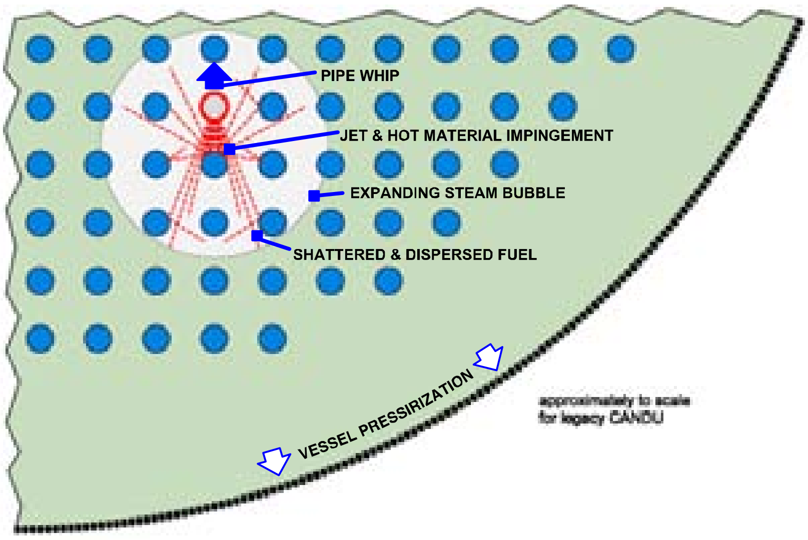


FIG. 5. Core behaviour phenomena in single channel events.

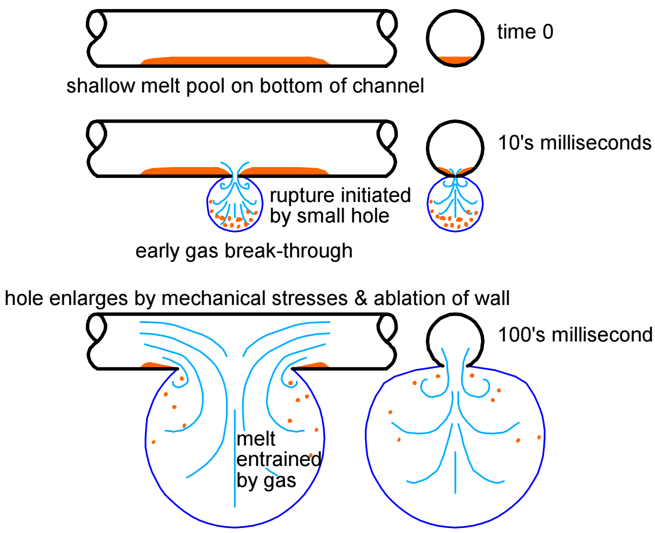


FIG. 6. Forced melt/water interaction (conceptual — not to scale).

The discharge of high-enthalpy coolant into the calandria vessel also produces strong jet impingement loads onto the surrounding channels and in-core devices. The surrounding components can also be struck by the ruptured channel (i.e. pipe whip) and by projectiles consisting of fuel bundles or fragments thereof. These phenomena have been assessed and there is no damage propagation to the adjacent fuel channels and enough shut-off rod guide tubes are undistorted to maintain adequate shutdown capability [5]. In addition, the poison inventory of the second shutdown system is adequate to ensure shutdown even if poisoned moderator is displaced by primary coolant.

3.1.1.5. Fission product behaviour

For a single channel event, the maximum fission product release is the complete inventory of the fuel channel. For a severe reduction in channel flow, the release could occur very rapidly. With the wide range of possible conditions, it is simplest to evaluate the fuel releases parametrically, assuming varying magnitudes of release up to the total channel inventory and release rates up to an essentially instantaneous release of the volatile fission products. While some fission products could remain dissolved in the PHTS or in the moderator, such retention is not typically credited. Fission product behaviour in containment is determined by wet aerosol phenomena and the complex chemistry of radioiodine [19] (see Section 3.1.2.6 for further discussion).

3.1.1.6. Containment behaviour

There are no particular safety issues related to containment behaviour in this family of accidents. In terms of containment thermal-hydraulics, the single channel events are small LOCAs. Some hydrogen could be produced during the transient (up to the amount equivalent to all zircaloy in the fuel bundles of one channel), but the maximum amount is too small to cause flammable concentrations in the containment.

3.1.2. Full core events

Extreme scenarios for this family of accidents are LOCAs with loss of ECCS that typically involve core voiding, fuel overheating, fuel and fuel channel deformations at moderate and low PHTS pressures, hydrogen generation and release of fission products from the fuel. Deformation of the fuel channel creates heat transfer paths from the fuel to the moderator, thereby limiting the consequences.

3.1.2.1. Thermal-hydraulic behaviour

The early stages of a full core LCDA are dominated by the blowdown behaviour, Fig. 7. The transient will start with a pulse in reactor power as the voiding coolant has a positive feedback on reactivity. This pulse is quickly terminated by shutdown of the reactor, and adds to the stored energy in the fuel at the start of the transient. A characteristic of PHWRs is the figure of eight configuration for the PHTS. For a LOCA, the pass with the break will start depressurizing first, followed by the unbroken pass. Once the PHTS has depressurized, the remaining liquid coolant boils off, and the resulting steam flows out the break. The flows in the primary system can be complex with buoyancy forces and steam-induced pressure gradients producing inter-channel flows through the connections provided by the headers. When steaming ceases, the remaining vapour in the PHTS, consisting of steam and hydrogen, continues to flow driven by buoyancy forces. Not much hydrogen is produced, because there is no fresh steam available to maintain the zircaloy oxidation. Fission products released from the fuel are distributed through, and largely retained in, the PHTS during this stage.

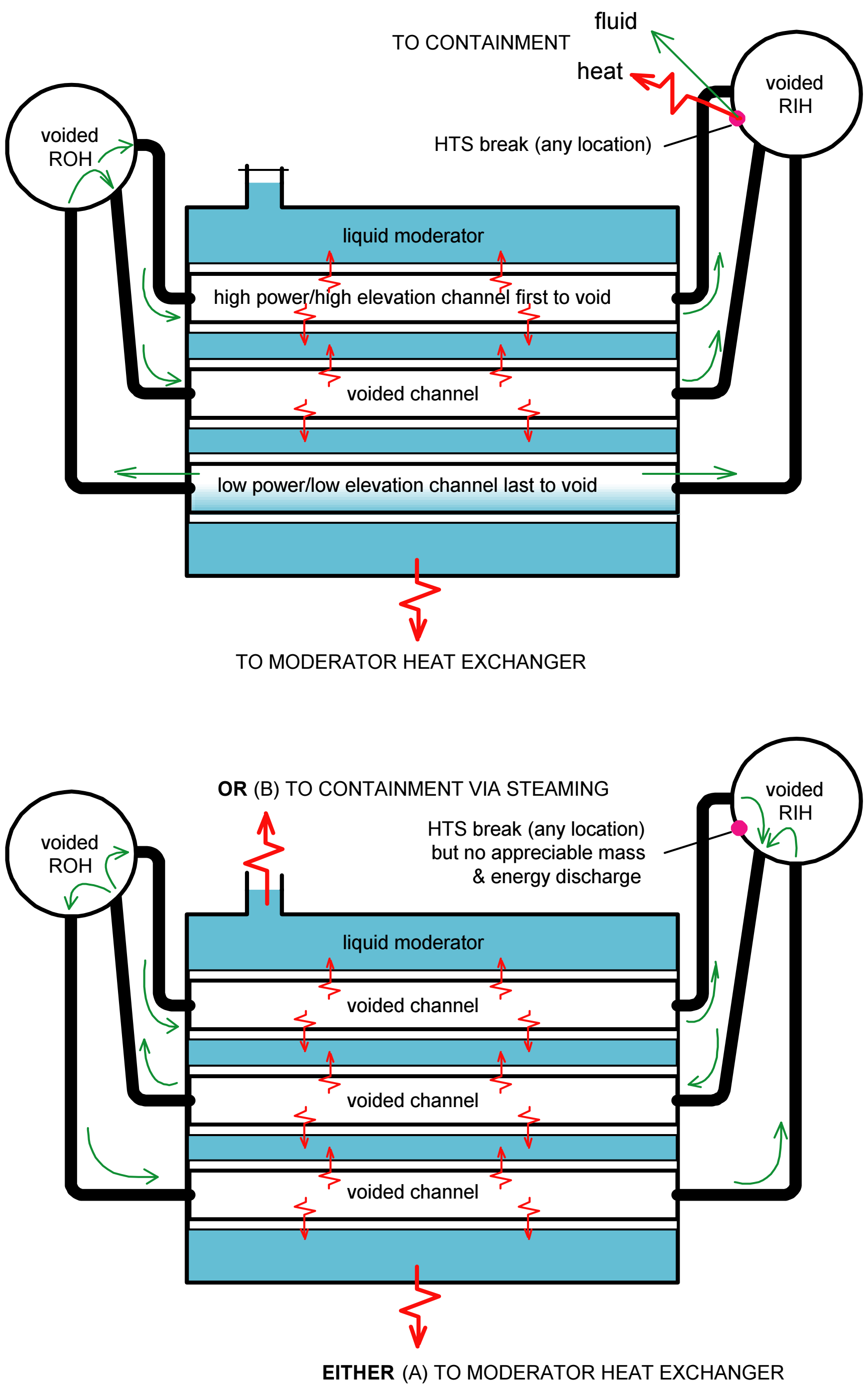


FIG.7. Thermal-hydraulic behaviour in a full core LOCA.

It follows from the preceding explanation that appreciable amounts of fission products and hydrogen are released from a break in the PHTS only during the period when liquid water is available below the reactor headers. The duration of this period is accident-scenario specific. Large LOCAs with their flashing-induced, early voiding of the PHTS will have a short period. Small LOCAs with a boil-off induced voiding of the PHTS will have a longer period, but at low decay power levels and correspondingly modest peak fuel temperatures.

If the fuelling machines are connected to the reactor during a LCDA, they form part of the primary heat transport system, and are therefore subject to the depressurization transient. In the longer term, any fuel in the fuelling machine will heat up, and in the absence of the secondary heat sink provided by the moderator, the hot fuel can breach the fuelling machine boundary and provide a release path to containment for the fission products and any hydrogen generated in the fuelling machine.

A special consideration for assessing the consequences of a LCDA is the presence of pre-existing leaks in steam generator tubes. These leaks provide a path beyond containment for any portions of the transient where the secondary system is not isolated.

3.1.2.2. Fuel behaviour

In a LOCA with loss of ECCS, the fuel is at decay power and the PHTS pressure is reduced. As the residual liquid in the fuel channels boils off the fuel bundles become uncovered. While there may be an initial spike in fuel temperatures due to the pulse in reactor power, the fuel does not reach very high temperatures until most of the water has been removed. Certain large breaks can cause rapid voiding of channels in a portion of the core. Under these conditions there could be rapid heat-up of the voided channels, but most accidents in whole core LCDA category start with the boil-off of stratified coolant.

During the early stages of a LOCA with loss of ECCS, changing thermal-hydraulic conditions can result in flow reversals that forcibly translate the fuel bundles within the fuel channel. Since the fuel is still relatively cool, break-up of the fuel bundles is not expected. As the accident proceeds, the fuel will heat up and axially expand. If there is not enough free axial gap, the expansion will be constrained, leading to deformation and possible break up of fuel bundles.

As a channel voids, the uncovered fuel heats up in steam. When the fuel reaches sufficiently high temperatures, the chemical heat released by the exothermic reaction of steam with zirconium-alloy fuel sheath supplements the decay heat. Convective heat removal from the channel by a mixture of superheated steam and hydrogen is small, and the fuel is cooled mainly by thermal radiation to the surrounding pressure tube. Therefore, for severe temperature excursions, the hottest fuel rods are in the interior of the fuel bundle because these rods radiate to hot, surrounding rods instead of to a relatively cool pressure tube wall.

Fuel and pressure tube temperatures continue to increase until a heat balance is reached where the heat generated in the fuel is removed by the steam flow (a smaller fraction) and radiation/conduction to the externally cooled calandria tube (a larger fraction). The transient is complicated by deformation of the fuel and the pressure tube, which alter the heat conduction paths as well as the hydraulic characteristics of inner subchannels in the fuel bundle.

Peak fuel rod temperatures are reached around the time the fuel bundle starts slumping. The values of peak temperature are accident-scenario specific. The interior of high-power fuel bundles could reach the zircaloy melting point if the channel were to void rapidly and remain voided thereafter. The additional energy deposition into the fuel caused by the positive void reactivity feedback before the reactor is tripped contributes to high fuel rod temperatures. In the long term, the temperatures decrease slowly with decreasing decay power.

A redistribution of stored energy in the fuel after the reactor is shutdown, and the fuel temperature rise later on, lead to fission gas release from the fuel grains and grain boundaries to the fuel-to-cladding gap. This increases the internal pressure in the fuel rod.

Hot fuel cladding will balloon due to the difference between the internal gas pressure and the coolant pressure. The free volume in a CANDU fuel rod is relatively small, and therefore ballooning in a localized region with the highest cladding temperature will balance the internal and external pressures. The fuel cladding may fail during ballooning by various mechanisms, including over-strain, oxide cracking, oxygen embrittlement and beryllium-braze crack penetration. The ballooning of fuel cladding causes some flow area obstruction. Following ballooning, the fuel pellets relocate to the bottom of ballooned fuel cladding and cracking during the temperature excursion may lead to further relocation of fuel fragments.

Prolonged exposure to high temperatures causes sagging of the fuel rods and deformation of the softened end plates. While end plates may maintain some spacing between the ends of fuel rods, within a short distance, the fuel rods sag into contact and fuse with each other [20–22]. For very high temperature excursions, the bundle becomes a coarse debris pile at the bottom of the pressure tube. The fuel pellets are retained in perforated and oxidized cladding shells. All distorted bundle geometries impede access of steam to the interior subchannels, which reduces the energy contribution from zircaloy oxidation. Fuel geometries may be further compacted by thermal fragmentation of embrittled fuel rods during refilling of the fuel channels when emergency cooling is eventually introduced. The fuel debris bed at the bottom of the pressure tube will be quite shallow (a few centimetres deep), so there is no issue of fuel coolability after the liquid water enters the channel.

Gross fuel bundle deformation is accompanied by thermal-chemical interactions of UO_2 and zircaloy, particularly if the peak temperature approaches the melting point of zircaloy. Limited amounts of molten material may be formed due to interaction between UO_2 and zircaloy below their melting points, depending on the interface temperature and contact pressure. UO_2/zircaloy interaction also leads to reduction of the fuel as uranium oxide fuel dissolves in molten zircaloy. Experiments show that surface tension forces retain the small amounts of liquefied materials in the inter-element voids. Some molten material (largely from the end caps and end plates) may relocate onto the pressure tube, causing intense localized hot spots on the pressure tube.

3.1.2.3. Fuel channel behaviour

The behaviour of a pressure tube in an accident at decay power is different from the behaviour in an accident at full power as described in Section 3.1.1.3. The reduced pressure means that higher circumferential temperature gradients and larger local hot spots (for example where bearing pads contact the pressure tube) can be tolerated [17, 23, 24].

For LOCA transients that lead to pressure tube heat-up while the internal pressures remain relatively high, the dominant mode of pressure tube deformation is ballooning [25, 26]. If the channels void more gradually, the pressure tubes will not heat up until the residual pressure is low. Under these conditions, the pressure tube will sag into contact with the calandria tube. Figure 8 illustrates the modes of channel deformation for a LOCA.

Upon contact between the pressure and calandria tubes, the stored energy in the hot pressure tube augments the heat transfer from radiation and conduction. If there is insufficient subcooling margin during ballooning contact, there can be extensive film boiling on the outside of the calandria tube. Under these conditions the temperatures of the combined calandria and pressure tubes can escalate to the point of failure. Such failures are not a concern for sagging contact because any film boiling is localized and does not lead to temperature escalation.

Extreme temperature excursions to the zircaloy melting point result in the relocation of molten metal onto the composite wall of the submerged tube. An unstable dryout patch arises at the point of molten material contact that does not produce any noticeable deformation of the composite tube.

If in the longer term, the pressure tube is exposed to a hydrogen/steam mixture with very little steam, the protective nature of the oxide layer may be diminished allowing hydrogen to be picked up by the zirconium alloy pressure tube. This hydrogen may embrittle the pressure tube, impairing its ability to withstand loads induced at low temperatures, such as those imposed by quenching due to late introduction of emergency cooling.

3.1.2.4. Reactor core behaviour

During normal operation, the local temperature distribution of the moderator is quite complex because of the interplay of the forced and natural circulation fields in a large liquid volume that has significant internal heat generation. After the reactor shuts down, the moderator temperature generally decreases due to the combined effect of temperature homogenization (mixing) and continued heat removal from the moderator with a reduced heat load. The temperature homogenization is rapid and therefore the available subcooling margin is uniform throughout the core. Continued heat removal from the moderator lowers the alternate heat sink temperature in the longer term (tens of minutes).

3.1.2.5. Containment behaviour

The initial discharge of a high-enthalpy coolant from the PHTS is considered as part of the design basis and containment can withstand it with margin to spare. Some designs employ dousing to condense steam and wash out fission products. In the longer term, core decay heat is transferred to the moderator and dissipated through the moderator heat exchangers with no appreciable energy released into the containment. If the moderator heat exchangers are unavailable, and there is make-up to the calandria vessel, decay heat can be removed by boiling of the moderator. Under these conditions, local air coolers in containment prevent any long term pressurization.

Hydrogen produced by oxidation of the hot core components will be discharged into containment during the blowdown and steaming phases of the accident. Additional hydrogen will be generated when emergency coolant is eventually injected into the core to restore the internal heat sink. On a long term time scale (days), radiolysis of water pools in containment releases hydrogen at a slow, but sustained rate. Hydrogen released from the PHTS and liquid pools in the containment is dispersed into the containment atmosphere by buoyancy-induced flow patterns, aided by the effects of local air coolers s. Depending on the reactor design, the potential for highly energetic hydrogen burns [26–32] can be diminished through dilution, the use of igniters and passive autocatalytic recombiners.

Some multi-unit CANDU stations have a small containment around individual reactors, all of which are connected through large ducts to a common vacuum building. In the event of a LOCA, self-actuating valves connect the vacuum building to the ducting. Effluent is then drawn from the reactor building to the vacuum building. Dousing in the vacuum building is used to condense steam, and to wash out soluble fission products. In the longer term, an emergency filtered air discharge system is used to control pressure, while filtering out fission products.

Current Indian PHWRs use a double concrete containment. The inner containment is a cylinder and dome of pre-stressed concrete, with an epoxy lining for leak tightness. The outer containment is a cylinder and dome of reinforced concrete. The intervening space is maintained at a negative pressure with a purging arrangement. A suppression pool between drywell and wetwell volumes in containment is used to limit peak pressures. The suppression pool also provides a source of long term low pressure emergency core cooling. Local air coolers provide pressure control and heat removal, and there is a filtered system for controlled gas discharge in the longer term.

3.1.2.6. Fission product behaviour

Fission products are present in fuel grains, fuel grain boundaries and fuel-to-cladding gap. The proportions of the total inventory in these locations depend on the fuel operating history. The on-power refuelling of PHWRs means that there is a broad spectrum of rod fission product inventories in any given channel. Upon failure of the fuel cladding, gap inventories are available for immediate release — the nature of the released fission products depends on their chemical form and associated volatility as a function of temperature [33]. Grain-boundary and fuel-grain inventories releases are driven by temperature-dependent diffusion, or mechanical breakup of the fuel.

Once released from the fuel, fission products are carried by thermal-hydraulic flows to the break in the PHTS. A unique aspect of PHWRs is the large surface area of relatively cool metal components immediately adjacent to the fuel channels (i.e. end fittings, feeder pipes and headers) through which the fission products must travel. These assemblies can retain significant quantities of fission products through deposition, or through pool scrubbing if they remain water filled. The actual transport, deposition and re-suspension phenomena for fission products are not unique to PHWRs [34].

Much of the fuel will remain well below the zircaloy melting point in LCDAs at decay power. As a result, noble gases and a subset of fission products that could be volatilized at intermediate fuel temperatures (i.e. iodine, caesium, tellurium, ruthenium and strontium) will be the dominant species entering containment. There are no structural materials in the PHTS that could be volatilized in this accident family and complicate the fission product chemistry (e.g. control rod materials are in the liquid moderator, not the fuel channels).

The behaviour of fission products in containment is determined by the interactions of aerosol and gaseous species as shown in Fig. 9. The radioisotope of primary concern is iodine for its relatively high abundance and biological interactions. A good understanding of the chemical behaviour of iodine in PHWR containments has been developed.

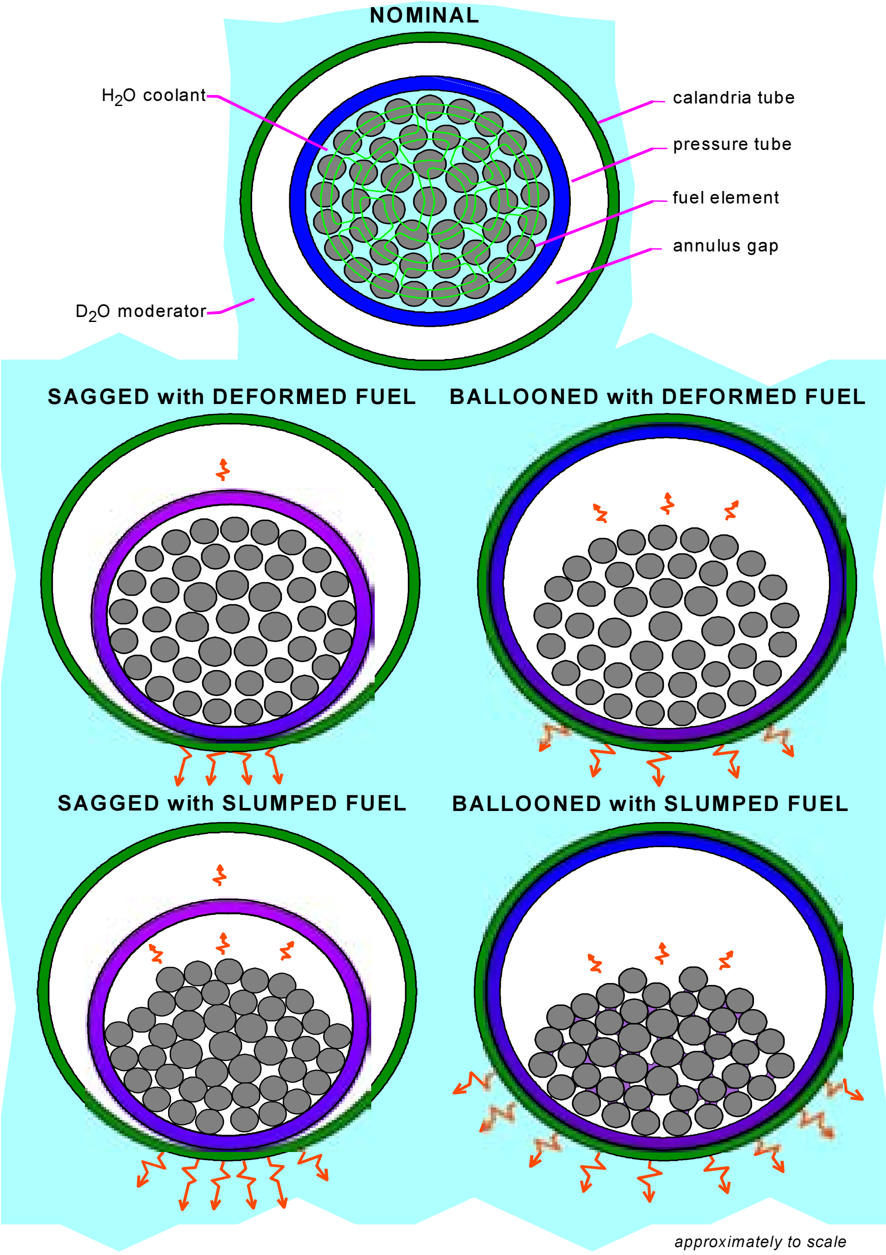


FIG. 8. Channel deformation models in a LOCA.

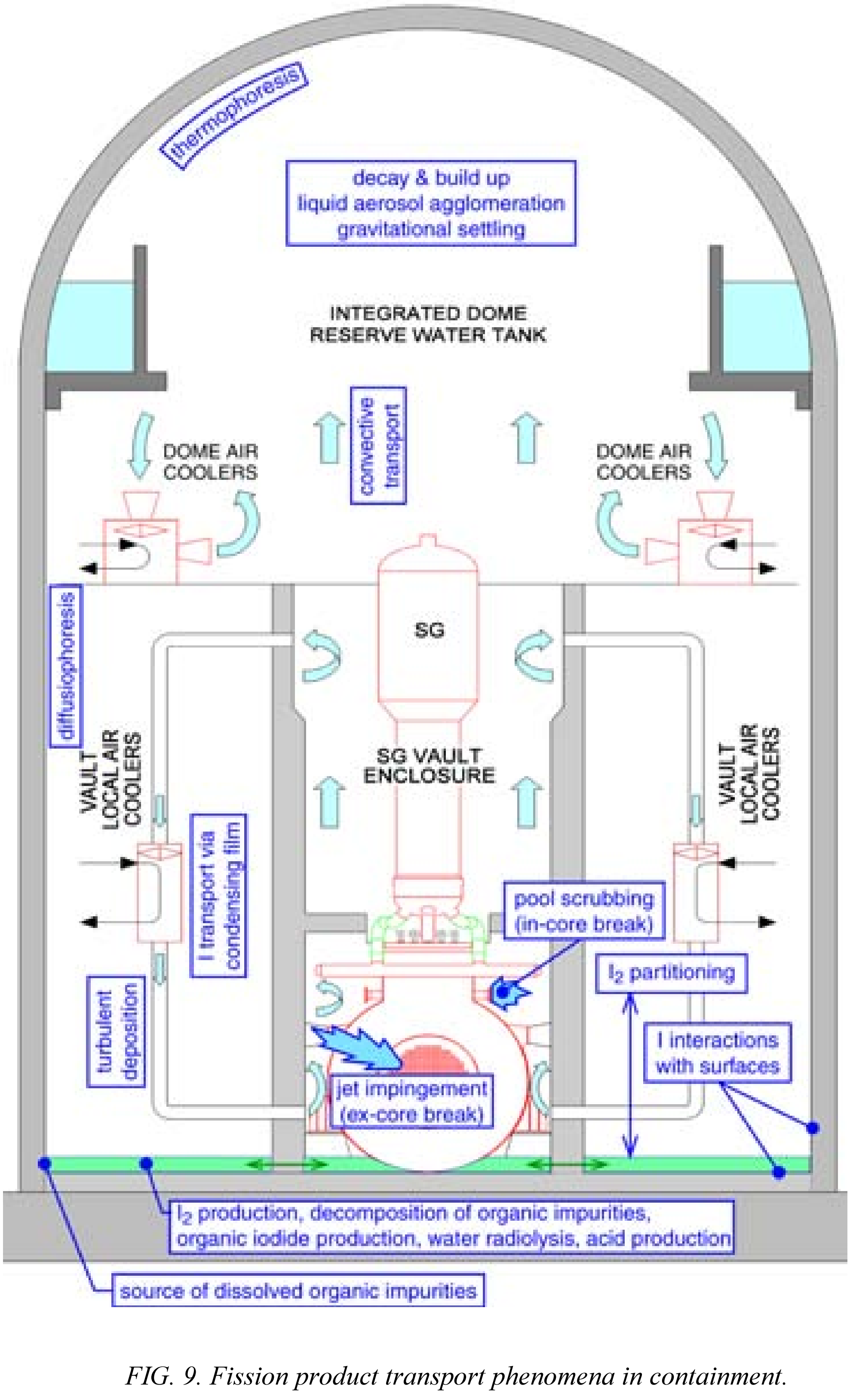


FIG. 9. Fission product transport phenomena in containment.

The kinetic processes governing iodine behaviour following an accident are complex. Based on experiments and modelling studies, it has been determined that iodine will be released to containment primarily as non-volatile caesium iodide (CsI), with a small fraction of gaseous species (assumed to be molecular iodine, I_2, for conservatism). While caesium iodide dissolves readily in water forming non-volatile I^-, radiation fields cause radiolytic processes that can form volatile I_2, which will partition from solution to the gas phase. In the gas phase, I_2 can be oxidized to non-volatile iodine oxides (e.g. HOI, IO_3^-) or be reduced back to dissolved I^- by various thermally and radiolytically driven reactions. Dissolved iodine can also react with organic species to form organic iodides that can also partition to the gas phase. Finally, iodine species can be deposited on surfaces, or participate in surface-catalyzed reactions. The presence of radiation fields in containment (after a postulated accident) prevents chemical equilibrium from being achieved. Instead time-dependent chemical, mass transfer and surface sorption reactions determine iodine behaviour. The rates of these reactions are functions of concentration, radiation dose and dose rate, pH, temperature and impurities.

3.2. Severe core damage accidents

The phenomenology of severe core damage accidents is unique to PHWRs until a corium bed is formed at the bottom of the calandria vessel. The subsequent behaviour of corium is similar to that of LWRs, but there are differences in corium composition (i.e. proportions of UO_2, Zr and other materials) and corium geometry (i.e. a surface to volume ratio, which is given by the shape of the vessel or the containment compartment).

A SCDA can result from a LCDA with loss of the moderator heat sink, or from events where loss of primary coolant flow is associated with loss of multiple safety systems. The initial stages of the LCDA are described in Section 3.1.2.

In the latter sequence, the PHTS is initially at full pressure, with heat from the fuel causing pressure rises that are relieved by the liquid relief valves. Eventually the volumes above the reactor headers are voided, and the upper, high power, fuel channels start to void and overheat. Under the influence of the full system pressure and temperature gradients, one of the voided channels will rupture, depressurize the PHTS, and initiate the automatic injection of emergency coolant from the ECCS accumulator tanks. Once ECCS has been depleted the reactor core will continue to heat up in a configuration similar to the late stages of a whole core LCDA.

3.2.1. In-vessel phenomena

3.2.1.1. PHWR core disassembly

As described in Section 3.1.2, as long as the fuel channels remain surrounded by moderator, the core geometry will be maintained. If moderator cooling and makeup are unavailable, the water level in the calandria will start to drop and uncover the upper fuel channels. The moderator level may also drop suddenly for an in-core break that leads to discharge of the moderator.

Uncovered fuel channels heat up and deform by sagging. Eventually, channel segments break off and form a coarse debris bed, which rests on still-intact lower channels [35–37], Figure 4.8. This suspended debris bed imposes a load on the channels below and alters the steam flow patterns in the calandria vessel. In time, the suspended debris also includes materials from uncovered in-core devices.

When the weight of the suspended debris exceeds the load-bearing capacity of the fuel channel plane just below the water level, most of the core collapses into the water pool at the bottom of the calandria vessel (see Fig. 10). Only some channel stubs at high core elevations remain in the voided portion of the calandria vessel. These stubs may join the debris bed later in time. Materials at the bottom of the calandria vessel are called terminal debris, because they are at their final (terminal) location in the context of the core disassembly.

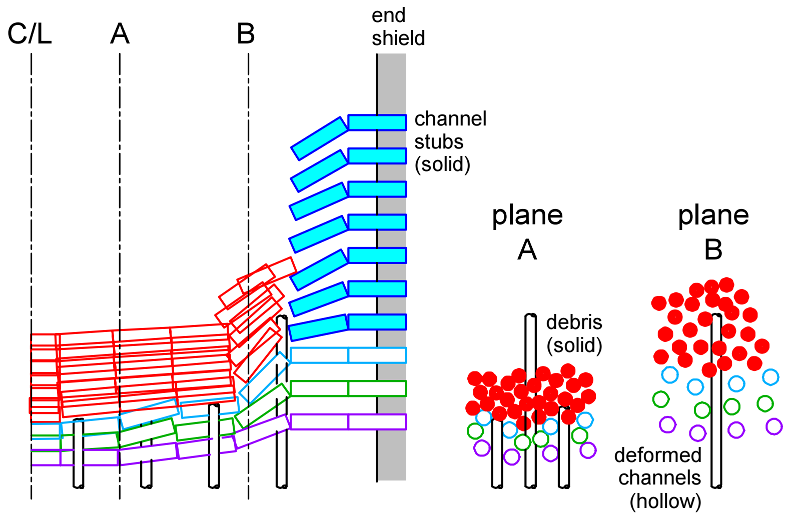


FIG. 10. Channel break-up into debris (conceptual — not to scale).

Perforations of calandria and pressure tube walls allow steam access into the annulus gap, which contains zirconium-alloy surfaces that are not protected by appreciable ZrO_2 layers. Steam also gains access to fuel surfaces within the pressure tube, which may have been steam-starved before the channel has broken apart. There is ample supply of fresh steam in the calandria vessel and the metal channel components are hot at this point in time. Such conditions are favourable to the Zr-steam reaction. The main impediment to a 'runaway' reaction is the absence of pressure gradients across the length of channel debris segments to drive the steam into the interior of tubular debris.

Steam supply into the interior of debris is a source of uncertainty for subsequent chemical heat and hydrogen generation rates. Sensitivity analyses show a range of possible thermal responses of the debris bed. At one end of the spectrum, oxidation rates are modest and the debris remains solid until the core collapses. At the other end of the spectrum, chemical heat released within the debris causes the zirconium alloys to melt while the debris is still suspended. Molten cladding is largely retained in the slumped fuel bundle by surface tension forces. Any liquid metal formed in the annulus gap is mobile and can relocate into the water pool at the bottom of the calandria vessel.

Local zirconium-alloy melt relocation processes (before the liquid metal is spilled from the debris) feed back on hydraulic characteristics in the interior of tubular debris and alter the surface area available for metal oxidation. Further feedback arises due to debris compaction. Hot tube shells are weight-loaded as well as ‘shaken’ when channels at lower elevations become uncovered and deform, causing the suspended debris bed to shift downwards. All these feedback processes tend to reduce the rate of exothermic oxidation within the debris pile.

A special case of suspended debris is a channel stub at the reactor end shield face, left behind after the channel breaks up. Depending on the accident sequence, these stubs can expose the interior of broken pressure tubes to fresh steam, but little steam enters the annulus gap, which is open on one side only. These stubs invariably contain low-power fuel and can conduct heat laterally to the end shields. Therefore, they can remain solid and suspended for a long time.

There are two modes of suspended debris relocation to the bottom of the calandria vessel:

- Intermittent, small pours of liquid zirconium alloy and/or a dropping of a small mass of fragmented solid debris;
- A sudden drop of a large mass of hot, solid material.

The pours of liquid zirconium alloy are relatively small amounts (kilograms to tens of kilograms) of melt. Upon contact with liquid water, the melt reacts with it and partially oxidizes. The chemical reactivity of the melt makes it highly improbable for the molten material droplets to form a stable suspension in liquid water, which is a precondition of steam explosion. Some hydrogen is generated during the pour. If fragmented, a molten stream solidifies while travelling through water. Otherwise, the stream can reach the calandria vessel wall where the liquid metal spreads and freezes. There are no particular safety concerns with these small, intermittent pours of reactive, molten metal into water at saturation temperature.

A large mass of solid materials relocates rapidly (tens of seconds) when enough suspended debris accumulates above the water level to exceed the load-bearing capacity of the first plane of submerged channels. The load is carried mainly by the calandria tubes, which are relatively cool. The thicker pressure tubes within the calandria tubes are hot and thus weaker. The failure mechanism is calandria tube pullout from the rolled joint at the tube-sheet of the end shield. Once the first submerged plane of channels cannot support the weight of the suspended debris, the planes at lower elevations invariably cannot support the load. A ‘cascading’ process occurs in which the load is transmitted to channels below while being increased by the mass of channels just pulled out. The whole core, except the channel stubs left behind during the formation of the suspended debris bed, collapses on a rather short time scale (minutes) into the residual liquid pool (see Fig. 11).

Once the core has collapsed all core materials are under water. The terminal debris consists of the pulled off channels at temperatures well below the melting point of zircaloy and the solid channel debris below the melting point of UO_2. The embrittled debris is likely to fragment during the core collapse. The channels that were submerged and failed by pull out, maintain their tubular geometry.

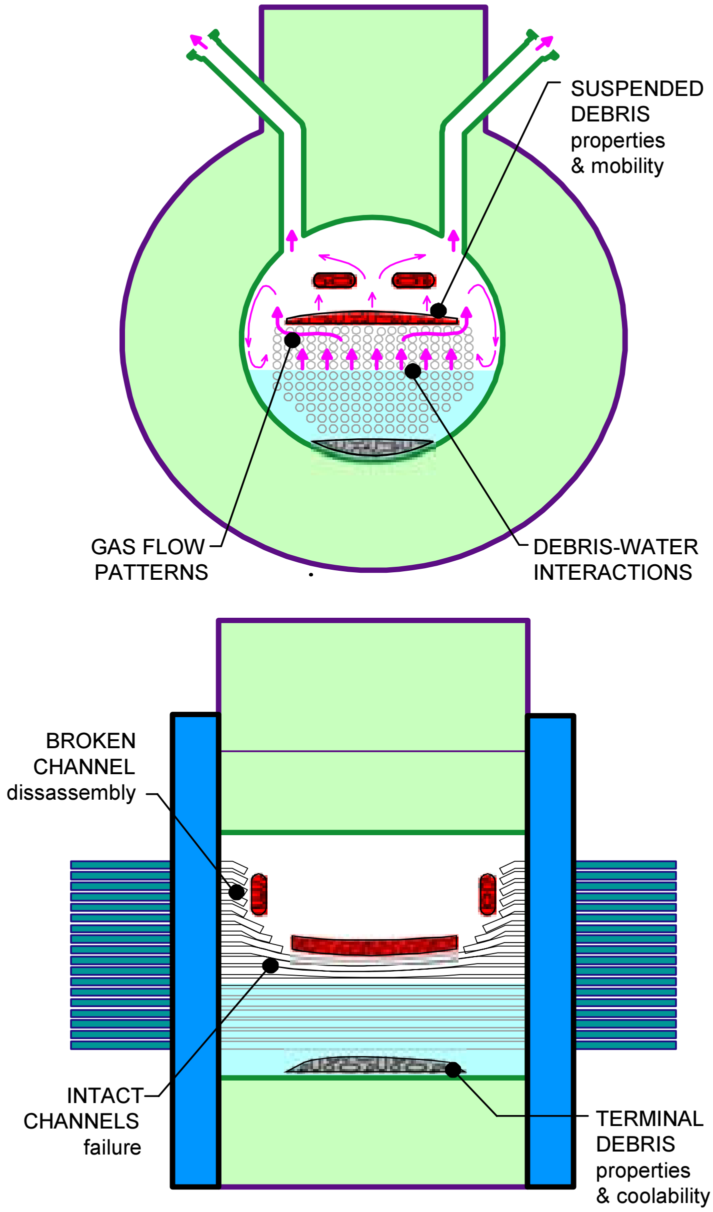


FIG. 11. CANDU core disassembly phenomena (conceptual — not to scale).

Strong steam surges arise as the stored heat is transferred to water. The calandria vessel has four large relief ducts, but even with the large relief flow area, the transient pressurization could pose an integrity challenge. The quenching of long channel segments is complex, because their exterior is not hot and their interior can only be accessed via a limited flow area at the ends of the segments. Eventually, the stored heat in the terminal debris bed is removed by vaporization of water.

The quenched debris is a coarse mixture of ceramic and metallic materials at low temperatures. As water evaporates, this mixture is gradually uncovered. The processes are similar to water boil-off from a fuel channel. Steam flow rates through the debris decrease with decreasing water level, but the uncovered materials do not reach very high temperatures until essentially all water is evaporated. Exothermic oxidation of zircaloy could come into play during the process of drying out the debris.

Eventually, the dry terminal debris reheats in a non-oxidizing environment and compacts. A ‘crucible’ is formed where materials adjacent to the externally cooled calandria vessel wall and materials at the top of the debris pile are solid. The interior of the crucible may contain molten or liquefied materials.

With the formation of the terminal debris bed, the question of possible recriticality arises. While the volume of the reactor is smaller, the probability of recriticality is slight because the moderator is boiling off, control rod materials are mixed with the fuel debris and there is a large upper surface area for neutron leakage.

3.2.1.2. In-vessel corium retention

Corium can be retained in the calandria vessel or the shield tank, Fig. 12. The surface-to-volume ratio of the debris bed is large, resulting in low heat fluxes as well as short heat conduction distances within the corium bed. Low heat fluxes avoid possible problems with convective heat removal on the waterside. Water-cooled walls completely surround the top debris surface to ensure effective heat transfer by thermal radiation.

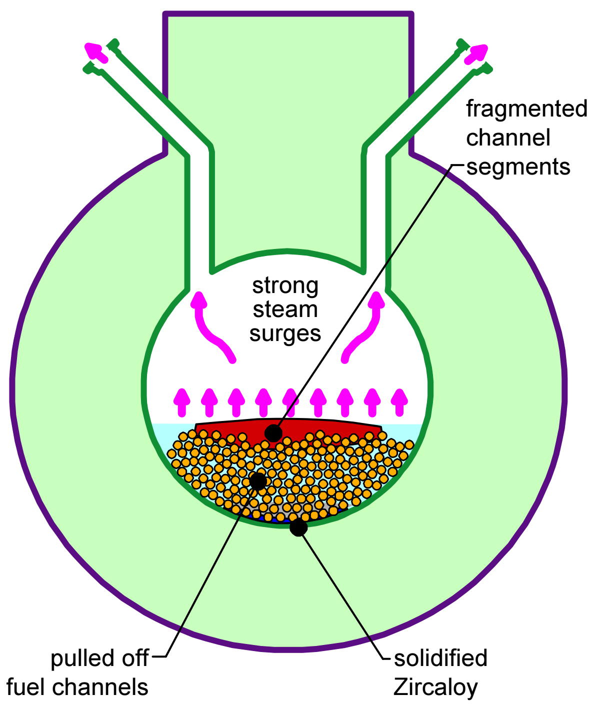


FIG. 12. Calandria vessel conditions after core collapse (conceptual).

The corium configuration within the calandria vessel is stable. Two alternate heat sinks are available (i.e. active heat removal by heat exchangers and passive heat removal by boiling and make up). For the corium bed to relocate to the shield tank/calandria vault, the water level around the calandria vessel must decrease to approximately the elevation of the corium bed surface (see Fig. 13). The radiation heat sink will have deteriorated during voiding of the shield water, so the corium crust at the top surface has become thin. Eventually, the corium thermally attacks the calandria vessel wall just above the water surface. Liquid corium flows out of the hole, ablating the crust as well as the wall. Interactions of molten corium with liquid water cause steam surges and associated pressure loads on the walls of both vessels (see Fig. 14).

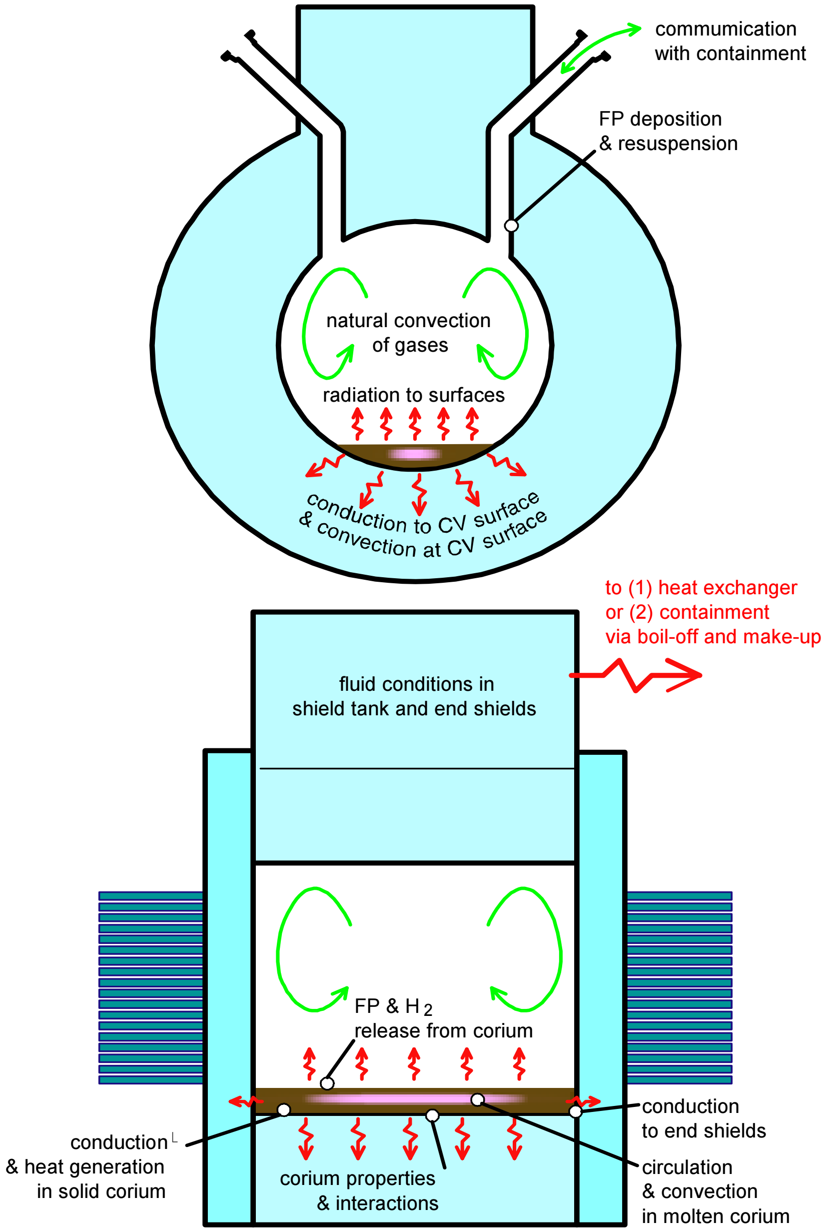


FIG. 13. Corium retention phenomena in calandria vessel (conceptual — not to scale).

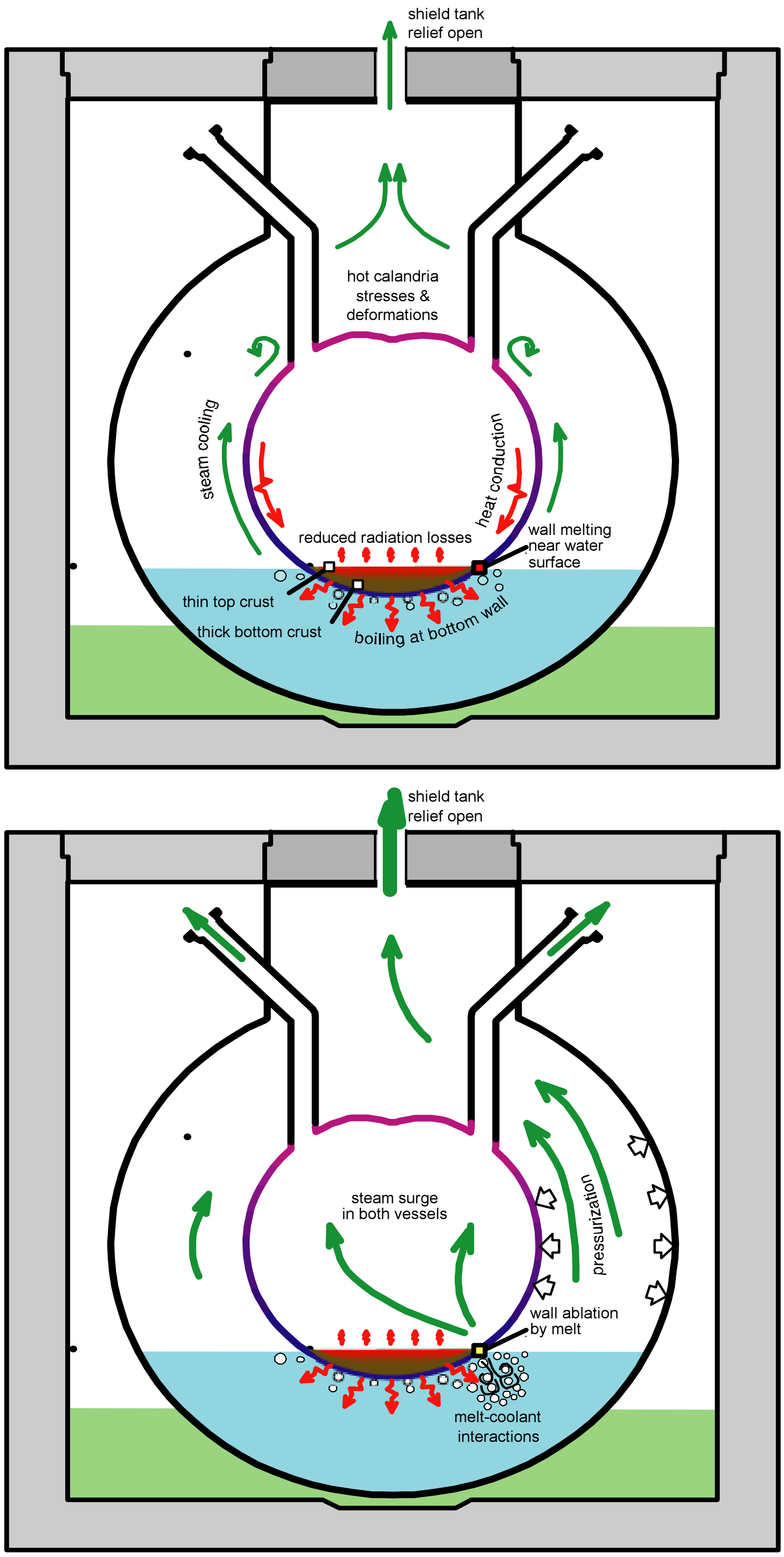


FIG. 14. Vessel-to-vessel relocation phenomena (conceptual — not to scale)

The relocation of molten corium from the calandria vessel to the shield tank/calandria vault will proceed relatively slowly. The melt pool near the corium surface is not particularly deep (so there is no significant head to drive the melt flow) and the pressure acts in opposition to the melt flow. These conditions can be expected to produce slow transfer rates and moderate steam surges that can be accommodated by combined relief paths of the calandria vessel and the shield tank. On the other hand, the potential for steam explosions arises as fragmented corium droplets can form quasi-stable slurries in water. If the shield tank/calandria vault survives the integrity challenges brought about by the corium interactions with water, a stable configuration is again reached.

3.2.1.3. Fission product and hydrogen releases

Fission products and hydrogen are released gradually before the reactor core starts to disassemble and in intermittent bursts after the accident progresses into a severe core damage sequence. Steam flow into the debris, which also sets the transport of products from the debris, predominantly affects the releases of fission products as well as hydrogen in these later stages.

The boundary conditions in the suspended debris bed are amenable to additional fission product releases from fuel as well as additional hydrogen production. Relatively hot channel materials are reconfigured during the channel disassembly such that heat loss from the debris is limited to that provided by steam/hydrogen flow through the interior. Meanwhile, the availability of fresh steam in the interior produces chemical heat and hydrogen.

Temperature excursions of the debris are controlled by how much steam gets into the interior of the debris (i.e. by chemical heat generation). If the steam flow is optimal there could be a temperature excursion to the melting point of zirconium. The coarse debris geometry (i.e. long channel segments) provides mitigating effects to prevent runaway oxidation. The duration of releases from the suspended debris is capped by the collapse of the core under the suspended debris load. Hydrogen will be produced during any relocation of the molten zirconium alloys into the water pool prior to the collapse.

Fission product and hydrogen releases are small during the debris quenching and re-heating stages. Volatile fission products have largely been released at this stage of a SCDA. Hence, fragmentation of the fuel during quenching is not an important fission product release mechanism. As noted earlier, temperatures during debris dry-out and reheating are modest such that no appreciable hydrogen is produced.

As temperatures rise in the dry corium bed (up to a liquefaction point), the less volatile fission products become mobile, but there are no driving forces (other than concentration gradients at the top surface) to drive vapours from the debris bed. No hydrogen is produced in a dry corium bed that is contained in a metal vessel.

3.2.1.4. Fission product transport

There could be multiple release pathways from a corium-retaining vessel. The principal pathway is through the pressure relief ducts of the calandria vessel or the relief paths of the two interconnected vessels. The secondary pathway is through an ex-core break in the PHTS, which might lead into containment or outside containment if a containment bypass is the postulated initiating event. Fission products will deposit on metal surfaces of release pathways. A resuspension of the deposited fission products upon any subsequent heat up of the metal walls is then possible. Nevertheless, in the absence of purging gas source in the vessel, the only release mechanism is diffusion through the relief paths, which is negligible on the scale of severe core damage accidents. Hence, any appreciable transport of fission products from the corium-containing vessel is during steam surges, which accompany corium motions. Any fission products present in the vessel atmosphere are purged by the steam surge.

With the exception of coolant jet and pool interactions, containment transport phenomena are the same as those identified in Section 3.1.2 for LCDAs. Small amounts of non-fuel materials with low melting points (from in-core devices) might supplement fission product aerosols. However, the aerosol concentrations are still 'lean', because there are no mechanisms to produce airborne particles of materials with high melting points that can enter containment.

If forced circulation of the containment atmosphere remains available, transport of the aerosols would be dominated by turbulent deposition in the local air coolers s and their associated ducts. If forced circulation were unavailable, the distribution of aerosols in containment would be governed by natural circulation of the containment atmosphere.

3.2.1.5. Containment behaviour

Additional hydrogen will be released intermittently during periods of core debris motion in the calandria vessel, always in conjunction with significant steaming. Hydrogen surges may temporarily produce elevated concentrations in the containment atmosphere. Otherwise containment behaviour is as described for LCDAs in Section 3.1.2.

3.2.2. Ex-vessel phenomena

Accidents involving ex-vessel phenomena must progress through the stages of the in-vessel SCDAs before corium can enter containment. This accident category is typically associated with severe challenges to the integrity of the containment (i.e. generation of non-condensable and flammable gases to pressurize the containment, and degradation of containment structures through interaction with corium) in conjunction with significant airborne fission products burden brought about by the same processes that cause pressurization. Decay of the radionuclide inventory in containment provides an additional heat load. A pressure-induced failure of the containment boundary under these conditions would release large amounts of radioactive materials into the environment. The water inventories of the calandria and shield systems must be evaporated for the corium-concrete interactions to come into play. This vaporization would over-pressurize the containment well before the core/concrete interactions could commence, unless the steam pressure is relieved by other means. Any subsequent core-concrete interactions occur at reduced pressures and failed or otherwise vented containment maintains some fission products retention capabilities.

3.2.2.1. *Melt release*

By the time an accident has reached the point that molten material is released into containment, the primary system will have been depressurized for a long period of time, so there is no mechanism for pressurized melt ejection. Thus the mechanism for melt release will be a 'pour' through the failure point(s) of the shield tank or calandria vault. The amount of melt released will depend on the location of the failure point(s).

3.2.2.2. *Fuel coolant interactions*

Depending on the specific design of the reactor, there can be water present underneath the stream of molten corium exiting the shield tank/calandria vault. The interaction between the molten material and water can result in energetic reactions causing pressure loadings, further oxidation of molten metals, and release of additional fission products. There is the potential for steam explosion, triggered for example by molten material impacting on the submerged containment floor.

3.2.2.3. *Molten corium concrete interaction*

If there is insufficient cooling of the melt released from the shield tank/calandria vault (melt is too deep for cooling by conduction and not enough water to provide additional cooling), the molten corium will react with the concrete floor. This interaction leads to release of steam, non-condensable gases such as carbon dioxide and combustible gases such as hydrogen and carbon monoxide. If containment is still intact, the erosion of the containment structures and the resulting pressure rise and potential for energetic burns can pose a late-phase threat to containment integrity.

4. BASIC STEPS IN PERFORMING ANALYSIS OF SEVERE ACCIDENTS

For carrying out an analysis it is important to visualize the accident scenario. The current severe accident codes are highly user dependent. The user should be familiar with the type of multi disciplinary phenomena, likely to occur during accident progression. It helps in deciding how to go about in carrying out the analysis. Sometimes one need to do a parametric study as the sensitivity to input parameters is significant and the sequence of events may have cliff hang effects. The various steps involved in an analysis are

1) Selection of methodology;

2) Selection of the computer codes;

3) Selection of appropriate models for different phenomena likely to occur;

4) Selection of appropriate nodalization scheme;

5) Preparation of input;

6) Interpretation of results;

7) Assessment of uncertainties/sensitivities.

Subsequent sections of this section describe these subjects briefly.

4.1. Establishment of methodology

Methodologies to be adopted in severe accident analyses are to be based on realistic basis and normally best estimate codes are to be used for the purpose. This is contrary to the design basis analysis where conservative computer codes are used for licensing purposes. However, the trend is now changing to use best estimate codes with conservative input conditions.

The methodology to be adopted largely depends on the type of integrated or component codes available in a utility. A user having codes which are 1-D codes in thermal-hydraulics and point kinetics for physics is certainly going to have a different philosophy from the user who has an access to a 3-D code in thermal-hydraulics and 3-D code in physics. In principle having one integral computer code for the entire spectrum of severe accidents containing detailed physics, thermal-hydraulics, mechanics and other disciplines connected with PHWR is not advisable as it would be impracticable in terms of computer time and memory.

However, depending on the type of accident sequence it is possible to have computer codes which deal with one or two disciplines in detail. The remaining inputs are provided in the form of initial and boundary conditions. For obtaining these inputs additional computer programmes may have to be run. This practice has been followed in several design basis accident analysis and is more applicable to severe accidents where the dominance of disciplines changes significantly as the accident progresses.

The clad ballooning may change the flow area of the coolant or increase the fuel sub channel bypass. The code calculating the clad ballooning may derive its external pressure conditions and temperatures from the main thermal-hydraulic analysis, but the strain calculations in the clad, a different code may have to be used.

The multiple computer codes required for analysis may be connected in series or may be connected in parallel. A parallel connection of computer code is needed when some phenomena in some part of the NPP has to be studied in detail. Very often this is a part of the core where the fuel is approaching a safety consideration which is going to alter the course of events. This change of course may be affected through mid course correction in the main solution.

For this the initial/boundary conditions are borrowed from the output of the main analysis and fed to the simulation where that part of the core is accounted in detail. For example a hot channel may have to be analysed in detail. In a global analysis it will only be apart of the lumped channels and effects such as vapour pull through or carry under at the inlet of the channel in a stratified header cannot be accounted.

For studying this channel in detail the boundary conditions may be borrowed from the global solution and then properly account for vapour pull through and study the channel in detail through a separate simulation. While doing this, it should be ensured that the part chosen for detailed attention is relatively small part of the earlier solution and even larger changes in the portion, are not going to affect the main solution. If it is not possible to ensure this than the main sequential calculation may have to include this detailed modelling.

Generally methodologies having computer codes in a sequentially progressive manner are preferred. The results of each computer code form the end of one phase of the accident. For example, a thermal-hydraulic code calculates the break discharges into the containment. Then a second containment code takes up these discharges as boundary conditions and calculated the containment transients. This is possible because the discharges into the containment are most of the time not affected by the containment pressure.

4.2. Selection of appropriate codes

The selection of the computer code for analyzing a particular sequence should depend on the type of phenomena encountered in that sequence. Although the codes available in this field are limited, it should be seen that the integrated code is well validated. Often validation results are difficult to interpret.

Procedures for quantifying the validation results are still not well established. However, a code which captures all the trends of a particular sequence is definitely a good code to use. The codes based on phenomenological evolution are better than the codes which are empirically related to the experiments, though such codes are normally more computer time consuming.

The codes which are modular in structure give much more scope to the user for experimentation with the use of the code. Currently the expert elicitation is the only method available for deciding the suitability of the codes. International standard problem exercises are the alternate means of collectively deciding the suitability of the codes.

Appendix II gives the list of the codes currently available for doing severe accident analysis and a short description of these codes together with its associated limitations. However, there are other codes dealing with heat transfer, mechanical structures, etc. These codes, in general, are easier to validate. An integrated computer code is generally better than a number of codes used sequentially.

4.3. Selection of models

In computer codes there may be more than one model available for simulating a particular phenomenon. Users are required to select a model depending on the progression of the accident. The following may be taken into account before a model is selected:

1) A model based on mechanistic development is preferable;

2) The model is applicable in the zone of analysis parameters;

3) Of the many models, the one with a larger database or a larger validation database is preferable;

4) The model should have monotonic trends outside its zone of applications. It should not have cliff edge effects outside its zone of application;

5) Predictions of the model should be bounded within the range of physical reasoning;

6) Predictions should be continuous within the zone of its application;

Following are some examples in the selection of models.

4.3.1. System models

Overall system models are required to provide the initial conditions and overall reactor response to the accident sequence.

4.3.1.1. Reactor physics

Physics models provide the reactor power transient and the initial conditions for fuel and fission product calculations.

4.3.1.2. System thermal-hydraulics

Based on the power transient and the response of reactor, system thermal-hydraulic models provide the overall thermal and coolant flow transients. Simplified representations for reactor components, fuel, etc. can be used to obtain the overall response.

4.3.2. Detailed fuel channel models

The reactor channel of a PHWR consists of multi rod bundles surrounded by a pressure tube which is enclosed in a calandria tube, submerged in a relatively cold moderator. The following models are required to simulate the channel behaviour during a severe accident.

4.3.2.1. Thermal conduction

Thermal model should be a two dimensional conduction model for multi rod, multi material with temperature dependent thermo-physical properties and with internal volumetric sources of heat generation in cylindrical geometry. It should also be able to simulate surface heat generation sources with convective and radiative heat transfer as boundary conditions. Emissivities are required as a function of material, temperature and oxide layer thickness. For the purpose of analysis, an equivalent hollow cylinder may represent the fuel rods in a particular pitch circle. However in such a case a suitable compromise in the initial stored heat, the surface area for radiative and convective heat transfer and thermal capacity, etc. should be arrived at. The ability to evaluate the feedback of deformed (e.g. sagged) fuel bundles on thermal-hydraulic behaviour may be required.

4.3.2.2. *Hydrodynamics for the reactor channel*

The reactor channel during a severe accident goes through from single phase liquid flow to two phase flows and single phase steam flow. An appropriate flow model coupled with constitutive models derived from heat transfer and pressure drop considerations are needed for estimating the coolant conditions. Heat transfer ranging from subcooled nucleate boiling through nucleate, transition and film boiling needs to be covered.

Conservation models

Two phase, two fluid partial differential equations representing models for the conservation of mass momentum and energy need to be used to estimate the coolant conditions in the channel. The corresponding constitutive models for pressure drop and void fraction, etc. have to be incorporated.

Flow pattern models

Flow pattern in the channel may influence the channel heat up significantly. A stratified flow may result in a quick heat up of the exposed rods of a bundle leading to early failure Two phase flow pattern models for horizontal channels with internals should be incorporated.

4.3.2.3. *Metal water reaction model*

The fuel cladding, calandria tubes and pressure tubes are made of zirconium alloys which react with steam. This is an exothermic reaction dependent exponentially on temperature, and on the oxide phase. A model, potentially based on Arrhenius equations, should be incorporated for determining not only the hydrogen produced and the oxide layer thickness, but also the amount of additional heat released.

4.3.2.4. *Fuel channel deformation model*

The pressure tubes undergo relatively rapid creep deformation at higher temperatures. These may be either ballooning at higher internal pressure or sagging at low internal pressure under the weight of fuel. In some cases simultaneous ballooning and sagging may also occur. Normal creep laws are applicable, with consideration of circumferential temperature gradients taken into account. When the calandria tube is cold, it restricts further deformation as the pressure tube comes into contact. The deformation model can also be applied to the creep rupture of the fuel channel at high pressures, as discussed in Section 3.2.

4.3.3. *Detailed fuel models*

Detailed fuel models may be required to assess the timing of fuel failures, and to provide initial conditions for fission product models. These would consider the thermal and mechanical response of the fuel to the temperature and pressure transients. Consideration of fission gas release is required to assess fuel clad ballooning and over-strain failure. Models for other failure mechanisms include oxidation, oxygen embrittlement and beryllium braze penetration.

4.3.4. *Fission product models*

Based on input from fuel and thermal-hydraulic models, fission product models calculate release kinetics, and transport behaviour, providing the source term for containment calculations. Fission product models can be correlations based on experiments, or detailed considerations of high-temperature thermo-chemistry, transport and deposition mechanisms.

4.3.5. *Models for moderator behaviour*

Moderator thermal-hydraulic models need to take into account the mode of heat transfer from the fuel channels, ranging from nucleate to film boiling. If moderator cooling is impaired, models are required to account for moderator boil-off — properly simulating core uncovery, and for vapour flow through the relief ducts.

4.3.6. *Models for uncovered calandria tubes*

After the calandria tubes uncover the fuel channel temperatures rise rapidly. Under predefined conditions the channel may be assumed to fail. These conditions may be set by temperatures in the channel or from a detailed calculation. A separate detailed model can evaluate the mode and location of channel failure. It is expected that a certain length in the middle will fail leaving the two short end portions in place.

4.3.7. *Models for core disassembly*

The disassembly of a PHWR core is determined by the mechanical failure of the fuel channels. Failure of individual channels leads to the formation of a debris bed. Eventually the debris bed becomes too large to be supported and the remaining fuel channels fail, causing full core collapse. Consequently, models for core disassembly can be based on detailed deformation models, or on a simplified temperature criterion for single channel failure and a load criterion for final core collapse.

4.3.8. *Debris and corium models*

4.3.8.1. *Thermo-chemical models*

The debris collected at the bottom of the calandria vessel initially will be solid. But slowly as the remaining moderator gets evaporated and heat in the form of decay heat and metal water reaction heat continues to be generated, the debris will heat up and melt. Models are required for the thermal and chemical behaviour of the debris, and for the formation of corium. These should include separation and stratification phenomena, and formation and ruptures of crusts.

4.3.8.2. *Vessel failure*

As long as the shield tank/calandria vault water level remains above the level of the corium, the calandria vessel should remain intact. Models for calandria vessel failure should take into account creep rupture of the vessel, or attack by corium. On failure, models are required for the corium migration to the shield tank / calandria vault water, including ablation of the opening, and interaction with the water. In the case of designs with a shield tank, the corium will interact with the vessel wall, leading to failure in a similar fashion to the calandria vessel, and corium migrating to containment concrete floor, possibly under water.

4.3.8.3. *Interaction of debris with concrete*

With a calandria vault design, failure of the calandria vessel results in corium migrating to the concrete floor, similar to failure of a shield tank. In either case, any water present will interact with the corium, eventually boiling off. Models are required for the interaction between water and corium, including the effects of vapour formation (e.g. hydrogen, corium vapours, etc.) limiting the access of water to the centre of the corium. Models are also required for core concrete interaction, and the production of gases such as carbon monoxide and dioxide. Models are also required for the spreading of corium as it migrates to the concrete, and as it further heats up.

4.3.9. *Containment models*

Effluent from the primary heat transport system and core-concrete reaction products, water, steam, hydrogen, non-condensable gases and fission products, will end up in containment. Containment performance and fission product leakage are assessed to determine the consequences of the accident.

4.3.9.1. *Containment thermal-hydraulics*

Containment thermal-hydraulics should model the behaviour of air, steam, hydrogen and non-condensables. Heat transfer to containment walls, structures and other heat removal systems, and heat generation from fission product decay should be taken into account. Intentional and unintentional leakage should be allowed for. Failure of containment is typically based on an over-pressurization criterion.

4.3.9.2. *Combustible gas behaviour*

Hydrogen from metal water reaction and radiolysis, and carbon monoxide from core-concrete reactions redistribute in containment. The containment thermal-hydraulics model will predict the combustible gas distribution. The pockets of hydrogen depending on the concentration of steam, air and carbon dioxide form mixtures which may deflagrate or form explosive mixtures. Appropriate models for hydrogen combustion based on composition and mitigating systems may be used.

4.3.9.3. *Fission product behaviour*

Aerosols are generated and released to containment because of disintegration of the core and release of fission products. The models should account for agglomeration, spray deposition, settling, atmospheric transport, diffusiophoresis, thermophoresis and diffusion. In addition models are required for the chemical behaviour of iodine, focusing on the behaviour of volatile iodine species, etc. redistributes itself in the containment.

4.4. Selection of nodalization scheme

Most of the thermal-hydraulic computer codes employ finite volume based computer codes. Mechanical structure codes are normally finite element codes while heat transfer codes are finite difference scheme based computer codes. Some of the 3-D fluid flow codes are computational fluid dynamics codes. All these schemes need the system to be divided into a number of nodes. For 1-D thermal-hydraulics these nodes are connected through junctions.

It is important the nodalization chosen should be based on the spatial and temporal gradients expected during the transient. It should also take into account the zones where important phenomena are likely to occur. While spatial and temporal discretization must be carried out, one should not decide on very fine nodalization lest it may introduce other errors due to computer limitations and phenomena such as water packing. Limitation on aspect ratio between adjacent volumes should not be violated. If a computer code validation for a reactor is carried out with a particular nodalization, then attempt should be made to use the same nodalization scheme. All the process systems, structures and other systems which may affect accident progression, should be represented. This is unlike, nodalization for design basis accident analysis where some of the non safety grade systems may not be taken credit of. Provisions should be made so that operator action can be simulated adequately in terms of boundary conditions. A change in nodalization for mid course correction should consider overall balances such as energy and mass balance.

4.5. Input data preparation

The quality of an analysis is mostly dominated by the features and correctness in the input for a given computer code. If possible, it should be started from an already existing input, which is familiar to the user, and from which he knows, which are leading to a reliable results.

4.5.1. Input data sources

Input data sources should be plant specific and with a reference to the exact sources of the data used (document titles and numbers, drawing numbers, etc.). The reference nuclear unit of the input deck and the state reflected by the input deck (e.g. current status, after future back-fitting, etc.) should be clearly defined.

In some cases when there are several plants of the same design, it may be useful to use a generic database for a reference plant and perform only a verification/update for credibility of the input data to a particular nuclear unit, reflecting the plant specific deviations towards the reference plant.

4.5.2. Documentation of input data preparation

The conversion of the plant data into an input deck for a particular code should be properly documented (e.g. in the form of calculation notes). The major features of the document should be sufficiently descriptive and illustrative, allowing reproducibility of the calculations included in the calculation notes. The document should also contain the major assumptions, the simplifications, neglected features, the initial conditions, and the boundary conditions.

Input data description should refer to a properly identified input deck (name, date of creation, etc.). A system of keeping track of changes in the input deck and the relevant documentation (part of the quality assurance plan) should be elaborated and operated. Also, it is advisable to provide comments into the input deck to ease the interpretation, handling and identification of the input decks. An input deck can be considered well commented if the comment lines are about 30% of the total input deck.

4.5.3. Input deck qualification

The quality of the input can be enhanced by an internal/external review process. For time dependent (transient) simulations, the input deck is qualified if steady state conditions are easily achievable and by the ability to keep these conditions before the beginning of the transient. Non-physical oscillations may often be attributed to a poor balance of physical quantities — temperature and pressure distribution, mass flow rates, junction form loss coefficients, boundary conditions, etc. Note that also for containment analyses the stable initial conditions are required before starting the transient analysis.

Input decks for real plants can be qualified by comparison to relevant calculations of transients, which have happened in the particular or a similar plant. The quality of the input deck is also greatly enhanced if the possible range of the input parameters is identified (e.g. by operating states or known measurement errors) and if their influence on the predicted results is quantified. This approach is especially recommended, if uncertainty in the input parameters is significant. Remaining unresolved issues should be clearly identified by the input deck developers.

4.6. Interpretation of the results

A severe accident analysis involves prediction of a scenario where a number of phenomena which may be significantly different from each other are connected through thermal-hydraulic and other processes. It is often difficult to interpret the results without the help of a post processor. The post processor should be able to bring out the following in a graphical form for easy comprehension.

- A 3-D/2-D graphical representation of thermal-hydraulic parameters and structural geometry of corium may be obtained. An animation of in vessel structure and temperature profiles may be obtained to give a very clear understanding of how and why an accident has progressed in a particular manner.
- A list of significant events should be identified which are the corner stones of accident progression. Time of occurrences of these events may be specified.
- It is important to differentiate numerical instabilities from physical instabilities. The normal procedure is to estimate the time constants of the systems participating in a

particular instability. From the comparison of the time constant (or frequency) of the instability and the time constant of the system parameters responsible for, it is often possible to differentiate between two types of instabilities.

- False diffusion may creep in many solutions if the numerical techniques employed are not carefully chosen to avoid it. Normally it would be difficult to quantify the degree of false diffusion which has crept in.

4.7. Assessment of uncertainties/sensitivities

The uncertainties involved in severe accident analyses are significantly large as compared to those involved in design basis accident analyses. This is because of immaturity in the understanding and modelling of the phenomena, development of computer codes and estimation of material properties. A sensitivity study identifies the parameters for which uncertainty analysis may be carried out. It also identifies if cliff edge effects are present. The objective of uncertainty analysis in context of severe accident analysis is to facilitate accident management strategies. This is unlike that for design basis accident analyses where it is done to estimate safety margins available.

5. FAILURE CRITERIA AND THEIR EFFECTS

In addition to the analysis assumptions and models employed, the results of the analysis will depend on the failure criteria used for various PHWR components, such as containment, calandria vessel, fuel channel, reactor vault and so on. Some of the criteria used in the analysis require user input values. Other criteria are calculated and applied by the severe accident analysis code. These failure criteria should be developed from experimental results, when available. If experimental data are not available, engineering judgement is used. Examples of some of the failure criteria that have been used by the modular accident analysis program for CANDU (MAAP4-CANDU) code [38] and the integrated severe accident analysis code for CANDU (ISAAC) code [39] are summarized here.

Failure criteria may be identified as limiting temperatures, masses, pressures, fluid levels, and the status or geometry of plant systems or components, which specify the conditions that this component does not perform the initially prescribed function. The failure criteria may also be a combination of factors, and there may be multiple sets of criteria for the same final state of a particular plant component.

5.1. Fuel failures

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 [40, 41]. 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. A more realistic value would be 1273 K. Currently both MAAP4-CANDU and ISAAC use the more conservative value of 1000 K.

5.1.1. Fuel bundle disassembly criteria

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 — this is the current approach in MAAP4-CANDU.

5.2. Fuel channel failure

Fuel channel failure is defined as a perforation of its pressure boundaries that results 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 PHTS pressure, some mechanisms being applicable at low PHTS pressure and others at high PHTS pressure.

High PHTS pressure

High temperature fuel channel experiments have shown that non-uniform circumferential temperature distributions lead to pressure tube rupture at high pressures. The calandria tube then fails due to pressurization and impingement of hot steam, etc. from the pressure-tube break. Predicting failure under these conditions requires a two-dimensional thermal mechanical calculation. An approximation is to assume that the pressure tube fails when it reaches a temperature at which it starts to deform (e.g. balloon) — approach used in both MAAP4-CANDU and ISAAC. This will be close as the failure due to non-uniform temperature gradients occurs during the deformation process.

Low PHTS pressure

At low PHTS pressure, the fuel channels may fail because of a local melt-through or sagging of pressure and calandria tubes. For melt-through, the channel can be assumed to fail when a portion of the fuel channel exceeds the melting temperature for zircaloy (approach used in both MAAP4-CANDU and ISAAC). Sag is more complex. Again, a simplified approach is to assume that the fuel channel fails if a certain temperature is exceeded (e.g. 1473 K is used in MAAP4-CANDU).

5.3. Fuel channel disassembly and reactor core collapse

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 is a massive relocation of core material and some intact fuel channels within the moderator onto the bottom of the calandria vessel.

The primary heat transport system of a CANDU reactor is divided into one or two loops. A two loop reactor core (e.g. a CANDU 6) is divided vertically such that channels in the left and right halves of the core are in separate loops, when viewed from the reactor face. Thus any channels in a vertical column of channels are in the same loop. Regardless of the number of loops, the channel flow direction is opposite in adjacent channels, whether above, below, or on either side of the present channel.

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. It is assumed that the suspended debris is carried by calandria tube rather than pressure tube since the latter could be hot and weak. 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.

It has been estimated that a submerged channel can support up to 7 additional channels before pullout from the rolled joints occurs, and it can support about 42 additional channels before shearing occurs. For one PHTS loop (one vertical core half), the number of supporting channels would be about 10, i.e. the width of one loop. Each of these ten supporting channels (located in the same horizontal row) can support the weight of seven additional channels. So, the total number of channels to be supported would be $7 \times 10=70$.

The dry channel mass is the sum of the UO_2 mass and Zr sheath mass for all 12 bundles in a channel, plus the mass of the pressure tube and calandria tube:

$M_{channel} = (21.327 + 1.966) \times 12 + 56.22 + 21.892 = 357.6$ kg.

Mass of 70 empty channels, which are supported by submerged channel:

$70 \times 357.6 = 25032$ kg.

Thus about 25 000 kg maximum load is equivalent to 70 dry and fueled channels. As a result, a possible failure criterion is to assume the core collapses when the debris mass exceeds 25 000 kg — current approach in MAAP4-CANDU. ISAAC assumes that the suspended debris bed relocates through the formation of molten material that falls to the bottom of the calandria vessel.

The maximum suspended mass (25 000 kg as estimated above) may be an overestimate of the amount of debris that can be supported, since estimates have been made that 2 to 6 rows of channels above the water level will lose their strength and transfer their weight to the submerged channels by sagging. Therefore, a significant fraction of the load on the submerged channels may be due to sagged intact channels, so that less debris might be required to cause core collapse. Also, the core debris may not be evenly distributed across all the supporting channels.

5.4. Calandria vessel failure

The calandria vessel can fail due to creep, over-pressurization or attack from molten corium. As a result, a number of failure criteria may be employed to determine the failure of calandria vessel.

- 1st Failure Criteria: Failure by creep — This can be based on the Larson-Miller parameter — This involves calculating the hoop-stress on the calandria vessel wall at a given temperature and pressure. It then uses the stress and Larson-Miller parameter to obtain the creep rupture time and the conditions to fail the calandria vessel by creep.

- 2nd Failure Criteria: Failure by high pressure — The easiest approach is to determine that the calandria vessel fails if a certain pressure has been exceeded (e.g. 2.25 MPa is used in MAAP4-CANDU).

- 3rd Failure Criteria: Failure due to debris impingement — This requires calculation of the erosion due to impingement of the jet, until the wall is breached.

- 4th Failure criterion: Failure due to molten metal layer attack — This can be modelled by assuming the wall fails when the temperature reduces the local strength to below the hoop stress.

- 5th Failure criterion: Failure of the drain line by hot molten debris on the vessel bottom — Can assume the drain line fails if it is subjected to high temperatures.

- 6th Failure criterion: Failure due to reduced external cooling — Once the shield tank/calandria vault water is reduced to the level of the corium, the wall will soon fail due to local overheating.

5.5. End shield failure

Failure of an end shield is not anticipated for the following reasons:

Core disassembly introduces a number of closed 'cavities' in the calandria vessel end shields (i.e. openings resulting from a pull-out of pressure and calandria tube rolled joints). However, the lowest 'cavity' is about 1 meter above the calandria vessel bottom (i.e. at the elevation of the lowest channel row) while the liquid corium top surface will be at a lower elevation. Hence, the cavities do not come into contact with the molten material.

The heat load from the corium to the end shields is about an order of magnitude smaller the heat load from the corium to the shield tank. Since the end shields are interconnected with the main shield water volume (i.e. any liquid loss is compensated for by an inflow of shield water), they will remain liquid-filled longer than the walls of calandria shell.

5.6. Calandria vault (concrete) failure

The calandria vault is a water-filled steel-lined rectangular concrete compartment that supports the whole reactor assembly through embedment in its walls. The vault also provides thermal shielding and cooling for the calandria vessel by means of its light water inventory. The calandria vault and the end-shield have combined vent lines. Two rupture disks are connected to the combined vent lines to relieve over pressure (during reactor transient) caused by the heating and boiling of the reactor vault and end shield water.

During a severe accident with reactor core disassembly, 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.

The calandria vault itself can fail in two different modes under severe accident conditions. The first failure mode is defined as the interconnection of the vault volume with the remaining containment volume causing the expulsion and/or draining of the vault water, thus affecting the heat sink status on the outside surface of calandria vessel. This pressure- induced failure occurs due to internal pressurization. It normally involves a gradual pressure excursion, so no dynamic load needs to be examined.

The second failure mode can be defined as the lost of the structural integrity (normally because of the molten corium-concrete interaction) of the calandria vault concrete bottom or sidewall that facilitates a relocation of core debris out of the vault. This failure mode involves a high temperature excursion of core debris and the vault structural materials (so called 'temperature-induced' failure mode). There is little existing information regarding the reaction of calandria vault shielding structures to the thermal and mechanical loads exerted on them by the debris in the calandria vessel. Hence, analysts should be careful with the selection of the failure criteria for vault failure and, in most cases, rely on engineering judgement.

Failure criteria for the calandria vault are as follows:

- 1st Failure Criterion: Overpressurization — the calandria vault can be deemed to have failed if a preset pressure differential between outside and inside is achieved.

- 2nd Failure Criterion: Rupture disk failure — again a preset differential pressure limit can be used for rupture disk failure.

- 3rd Failure Criterion: Corium — concrete interaction — using a model for ablation by the attack of corium, the calandria vessel can be deemed to have failed either as the wall is breached, or when a certain thickness remains. Failure in the latter case would be due to the imposed loads of the corium mass.

5.7. Shield tank failure

If the shield tank does not expel its water inventory, there can be a long delay before the hot core debris comes into contact with shield tank wall However, major temperature gradients and attendant thermal stresses can be anticipated as the shield thank water level decreases. This situation has been assessed, and the shield tank can accommodate the thermal stresses during the boil-off.

Once the shield tank drains its water content, the hot debris will come into contact with its wall soon thereafter. The water level on the outside will be below the bottom of the shield tank, so the bottom wall will fail by temperature-induced creep or melting. In severe accident simulations, this type of failure is adequately represented by a short delay (say 10 minutes) relative to the time of shield tank draining.

5.8. Containment failure

The present understanding of the failure of CANDU containment under pressure loads is largely based on results of experiments and analysis sponsored by the Canadian regulatory body at the University of Alberta [42]. Experiments were conducted to determine the effect of overpressure on a 1/14 scale containment model at room temperature. The test results were analyzed and applied to the Gentilly-2 containment. The analysis results showed that at a containment pressure of 430 kPa (a), the first through-wall cracks appeared. At that pressure the reinforcing horizontal tendons were still intact. The horizontal tendons began to yield at 555 kPa (a). At 630 kPa (a) the reactor building failed as a result of the rupture of the horizontal tendons.

The containments for newer CANDU 6 designs (e.g. Wolsong and Qinshan NPPs) evolved from earlier designs of the Gentilly-2 plant. The modern CANDU 6 containment has more rebarred and more vertical and horizontal tendon pre-stressing, than does the Gentilly-2 plant. Therefore, the ultimate pressure capacity for these containments is expected to be higher than the Gentilly-2 containment.

Note that 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:

1) 1st Failure Criterion: Pressure-induced

Containment failure is based on a exceeding a predetermined pressure. For example, a value of 500 kPa (a) is currently used in the MAAP4-CANDU analysis of CANDU 6 plants and ISAAC uses 519 kPa (a), both based on the information discussed above.

Note that these values are below the calculated failure pressure of 630 kPa (a) for the failure of horizontal tendons for the University of Alberta tests. If containment leakage is modelled, then containment failure may be delayed. Such effects should be considered in the analysis. It may be possible that noble gases are released to the outside environment through containment wall cracks starting at about 430 kPa (a), when through-wall cracks appear. This phenomenon also needs to be considered in the analysis.

2) 2^{nd} Failure Criterion: Strain-induced

A simplified model can be used to determine containment stresses due to internal pressure and compared to the ultimate stress of the containment structure.

6. USES OF SEVERE ACCIDENT ANALYSIS AND BASIC APPROACHES

Severe accident analysis has been used to support PSAs, to help 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 (a) training purposes, (b) the development and validation of accident management programmes, (c) the design and validation of severe accident mitigation systems, and (d) most recently, plant simulators. In addition, these codes are being used for the analysis of a wider range of different reactor designs.

6.1. Support for PSA

As discussed in more detail in the IAEA report [10], one of the earliest uses of the integral codes was to support PSA activities. Plant specific calculations were performed using the integral codes for representative groups of sequences to establish the (a) results for important variables as a function of time and (b) timing for major events. These calculations were supported by sensitivity studies, expert opinion, and, in selected cases, mechanistic code calculations to estimate the overall uncertainties of the results. These results are then combined to determine the accident progression event trees and associated probabilities for different branch points. Additional plant specific calculations were also performed using the integral codes, and in limited cases, a combination of mechanistic codes, to determine source terms for high frequency release sequences or those sequences were expected to include relatively large releases of fission products. These results were then combined as part of a level 2 PSA.

6.2. Resolution of severe accident issues/severe accident research

Most of the internationally recognized codes have been used to design and analyse severe accident experiments and to support international research programmes as well as help to resolve any outstanding technical issues (such as reflooding, ex-vessel cooling). In some cases, these applications may result in additional modelling improvements and the release of new versions of the severe accident codes. For example, improvements in models to treat the re-flooding of a damaged core, cooling of the debris and vessel during the later stages of the accident, and the formation and slumping of molten fuel are likely.

6.3. Development of training programmes

Code specific user training in combination with generalized training on severe accident phenomena and research can also be an effective way to train technical support staff and engineering analysts. In particular, engineering analysts familiar with system thermal-hydraulic codes used for design basis analysis can be relatively quickly trained to use mechanistic codes for severe accident conditions. In addition, since most plant models developed for system thermal-hydraulic design basis accident analysis can easily be extended for use in the mechanistic codes, experienced thermal-hydraulic analysts can quickly convert between data for design basis and severe accident calculations. For experienced analysts of system thermal-hydraulic codes, modifying the plant models to include core degradation aspects takes generally only a few days. The training for the integrated codes is somewhat more involved since these codes cannot use input models developed for design basis accident analysis and thus require training programmes different from training for design basis accident system thermal-hydraulic analysis. However, generalized training on severe accident phenomena can apply equally well to analysts using either the mechanistic or integral codes.

6.4. Analytical support for accident management programmes

Development of accident management programmes is one of the most frequent applications of severe accident analysis. Obviously, the first priority for reactor safety has always been to prevent any accident from occurring, in line with the aim to achieve very low core melt probabilities. Should the accident progress into a severe condition despite all preventive measures, and then the priority is to arrest or slow accident progression and to attenuate or mitigate the releases of radioactive material by utilizing all means of accident management available at the site.

Accident analysis related to accident management programme is important to understand plant response to beyond design basis accident and severe accident, to understand which accident phenomena are important for the plant in question, to understand and rank challenges to fission product boundaries, and to provide a sound basis for the investigation of preventive and mitigatory measures of accident management programme . The analysis should be done with a suitable and reasonably validated code, and similarly as other kinds of severe accident analysis should be performed on a best-estimate basis.

There are specific IAEA reports devoted to accident management programme and to their review [8, 13]. These reports include also basic requirements on how to perform analysis of severe accidents needed to support preparation, development and implementation of accident management programme .

Three categories of analysis are identified in Ref. [8]:

- Preliminary analyses; these are informative in nature and provide an understanding of the response of the plant to various types of accidents and basis for selection of recovery strategies. In particular, analysis should be made of sequences that, without operator intervention, would lead to core damage, core melt, vessel failure, and release of fission products.

- Procedure and guideline development analyses; these are needed for detailed confirmation of the choice of recovery strategies adopted, to provide necessary input to set-point calculations (where appropriate), and to resolve any other open items identified during the previous step. Recovery strategies include preventive measures to halt or to delay onset of the core damage and mitigatory measures to mitigate the consequences of the core damage.

- Validation analyses for procedures and guidelines; they are performed in order to demonstrate the capabilities and choice of appropriate strategies and optimize some aspects of these.

6.5. Use of computer codes in simulators for severe accidents

Application of simulators and simulation techniques in general in accident management training is described in the IAEA report [14]. In this report, a simulator is characterised as 'a computer based assembly of software and hardware, which is capable of presenting the physical behaviour of the whole NPP or the part of it during various operational states and malfunctions. The simulators are typically equipped with an advanced user interface (graphical or hardware interface) suitable for interactive operation and particularly suitable for training purposes.' Generally, the simulators are subdivided into engineering simulators (used for design purposes and in particular for justification of the design) and training simulators.

6.6. Support for new designs

For new designs, reactor designers are contemplating the possibility of developing the mitigatory measures to cope with severe accidents. Such measures have been also the subject of many international research programmes.

In this regard, many activities are being performed to understand the main phenomenological aspects and to develop the most appropriate severe accident management, both of the preventive and of the mitigatory type. For this reason, many research projects in this area have a strong 'evolutionary' flavour based on the consensus around the safety approach of evolutionary reactors, such as EPR, AP1000, ABWR, ESBWR, ACR. In different extent, some of the evolutionary designs incorporate severe accidents into their design and licensing approach.

Combination of calculations by means of mechanistic system computer codes with detailed computational fluid dynamics codes is typically needed. Calculations often have to be complemented by special experiments. Large uncertainties in calculations should be compensated by more robust design.

7. PHWR ACCIDENT MANAGEMENT

7.1. Introduction:

Accident management programme of the PHWRs follows the general defence in depth philosophy to achieve the four main objectives of accident management:

1) Prevention of the accident from leading to core damage;

2) Termination of core damage;

3) Maintaining the integrity of the containment as long as possible;

4) Minimizing on-site and off-site releases and their adverse consequences.

The first objective, prevention of core damage accident, is achieved by the inherent and designed safety features of the PHWR. Procedural lines of defence are provided in the EOPs to guide the operating staff to manage the design basis accidents should one occur. The latter three objectives constitute what is generally referred to as severe accident management. The emergency plan is provided to deal with mitigating the radiological consequences of releases of radioactive materials as a result of accidents.

This section suggests a summary overview of the three main procedural elements of accident management for PHWRs: EOPs, SAMGs, and emergency plan, based on Canadian practice. More elaboration is provided in the area of severe accident management strategies, as this is the key component for the mitigation of severe accidents.

7.2. Emergency operating procedures

In PHWR plants, EOPs (or similarly named documents) are used to deal with plant events and accidents that result in, or require automatic or manual power reduction. The entire design basis set of accidents is well covered by the EOPs. The operating personnel are well trained in the use of the EOPs, which form part of the licensed regulatory certification process [43].

There are two types of EOPs — event based (specific EOP) and symptom based (generic EOP). Event based EOPs provide optimum response to plant upsets and accidents where they can be diagnosed, and the resultant plant response and required corrective actions can be predicted. Typical events considered are derived from the accidents examined in the plant safety analyses. When the upsets cannot be clearly diagnosed, when they cannot be identified in advance, when the plant or operator response to any diagnosed upset proves inadequate, or when the plant response or corrective actions cannot or have not been predicted, then event based EOPs are not appropriate. Symptom based EOPs are provided under these circumstances to recover the critical safety parameters status; In PHWRs, the status of a relatively small set of parameters can direct the operating staff to ensure adequate fuel cooling and containment of fission products.

A typical set of EOPs include the following procedures:

7.2.1. *Event based EOPs*

- Diagnostics and power reduction actions;
- Large LOCA with automatic ECCS initiation;
- Small LOCA/heat transport system leaks;
- Loss of steam generator feedwater;
- Main steam line break;
- Loss of electrical power;
- Steam generator tube rupture;
- Loss of service water;
- Loss of instrument air;
- Dual control computer failure.

7.2.2. *Symptom based EOPs*

- Generic main control room (critical safety function restoration procedure);
- Generic secondary control room (seismic/common mode events procedure).

7.3. Severe accident management guidance:

The transition from using EOPs to SAMGs occurs when the EOP (specifically — symptom based EOP) is not effective and core damage has occurred or is imminent. In the Westinghouse Owners Group approach [44], this transition is based on a single measurement of the exit plenum thermocouple. It is a condition in a PWR where the accident is likely to involve uncovering of at least part of the core and to rapidly progress to the point where fission product release barriers are challenged. In PHWR, such a clear-cut criterion does not exist.

A loss of heat sink or an in-core LOCA can lead to fuel channel rupture, loss of coolant and consequential draining of the moderator at relatively low temperatures. The EOP can manage this situation. However, with the moderator inventory loss, this in-core LOCA can progress to a severe accident if the ECCS also fails. For other accident sequences such as LOCA plus loss of ECCS, the EOP can be successful in maintaining adequate decay heat removal by the moderator. Once the EOP can no longer maintain the moderator level or if a large release is indicated, given dried out fuel channels, then the transition should be made to SAMGs.

There are three conditions, certain combinations of which would be appropriate for moving from the preventative regime of the EOP to the mitigatory regime of SAMGs. As illustrated in Table 1, they are:

- Indication of severe degradation of core cooling;
- Reduction in moderator level, or Indication of a large ongoing release of fission products to containment.

TABLE 1. PROPOSED CRITERIA FOR TRANSITION FROM EOP TO SAMG

Condition	Parameter	Measurement device
1. Loss of core cooling	No subcooling margin in inlet headers for several minutes	Heat transport system temperature and pressure instrumentation
AND either		
2. Loss of moderator cooling to fuel channels OR **3.** Excessive release of fission products from fuel	Moderator level below highest channels Plant radiation levels	Moderator level instrumentation In-containment fixed area gamma monitors or field surveys

Degradation of cooling is indicated by sustained loss in subcooling at the inlet header, consistent with a loss of coolant. If the moderator level remains high, then the EOPs are being effective. A drop in moderator level can occur for many design basis accidents, but in combination with degraded cooling, it indicates the potential progression of core damage state and hence, need for SAMGs. The high radiation is an indication that the EOPs are not effective in preventing core damage. It is also an indication of an associated release of hydrogen, which is a potential threat to containment integrity.

7.4. Strategies for the severe accident management

One of the aims of severe accident analysis is to identify accident management measures that could be carried out to mitigate accident effects and to develop severe accident management programme to be followed in beyond design basis accidents and severe accidents conditions (see Ref. [45]).

A necessary step in accident management planning is to identify the vulnerabilities of the plant which are likely to cause challenges to the safety functions, and the mechanisms by which the barriers preventing the release of radioactive materials can be challenged.

On the basis of the vulnerability assessment and understanding of accident behaviour, as well as of the plant capabilities to cope with accidents, the next step is to develop accident management strategies. The objectives of the strategies should be specified and related to the basic safety functions, e.g. to protect the core integrity by maintaining subcriticality and restoring core cooling, to protect the PHTS integrity, to protect the containment integrity and to minimize the radioactive releases in the event the containment fails or is bypassed. One of the first steps in developing strategies should be the establishment of criteria which uses identifiable physical states in the plant as either action levels or thresholds for the various steps in the operator response. These steps are aimed at preventing or delaying each of the phases of progressing severity.

Failure of a strategy to achieve the objectives at one phase should still leave options for achieving the objectives at subsequent phases. There may be a systematic evaluation of the possible strategies which can be adopted at each phase. Suitable strategies should be capable of being carried out under the physical plant conditions associated with the particular challenge to the safety function which the strategies are intended to restore. It should be investigated what the impact is of the strategies on the different plant conditions during the subsequent phases of a severe accident. Both positive and negative consequences should be investigated, so that it can be decided which strategies constitute a proper response under a given plant damage condition.

7.4.1. Selection of severe accident management strategies.

Selection of severe accident management strategies should take place after a review of all severe accident insights that are relevant for the particular plant or group of plants. These insights are derived from various sources, including the analyses:

- Severe accident research at a variety of institutes and laboratories;
- Candidate accident management strategies from other sources;
- Industry studies on severe accident management guidance;
- PSA or individual plant examination of that plant or group of plants.

Severe accident analysis to identify severe accident phenomena, criteria to define onset of severe accident, plant damage states requiring different accident management strategies, etc.

Based on this material, the different stages and processes of a severe accident are studied to determine whether they are relevant for the plant considered. A binning process may be followed, in which consequences of phenomena and countermeasures are considered. An example of such a binning process is given in Table 2.

Once the insights have been determined, a path can be set out to obtain suitable strategies, where due consideration should be given to the remaining uncertainties in severe accident insights. Such strategies are single actions or a group of actions that are initiated after a certain degraded condition has been identified. The degraded condition is often called 'plant damage state' or ‘core damage state’, for which several approaches exist. Examples are given in Appendix III, where the damage states refer to the core. It may be needed to do some calculations to conclude to a certain plant damage state, as measured parameters may need interpretation. In order to avoid doing such calculations during an actual event, pre-calculated curves and graphs may be used in the form of computational aids. E.g. measuring the containment pressure and reading the hydrogen concentration may give an immediate insight whether or not the containment is challenged.

TABLE 2. EXAMPLE OF THE DEVELOPMENT OF SEVERE ACCIDENT MANAGEMENT INSIGHTS FOR A PHWR

1	steam explosion	in-vessel: (high pressure steam explosion in calandria vessel is highly unlikely) ex-vessel: will or will not fail containment
3.	core concrete interaction	can/cannot lead to containment overpressurization can lead to combustible gas (CO) will/will not continue after flooding of debris
5.	in-vessel debris cooling	submerging debris will/will not keep debris in-vessel
6.	external vessel cooling	will/will not retain debris in-vessel
7.	ex-vessel debris cooling	Not applicable for PHWRs
8.	hydrogen generation	hydrogen deflagration may/may not occur deflagration may/may not challenge the containment integrity
10	determination of accident progression	onset of core melting will/will not be observed by control room relocation of debris to calandria vault/shield tank will/will not be identified by control room calandria breach will/will not be observed by control room

Strategies are based on actions that are either still available to the operator or are available first after certain systems have been restored to service. They are high-level actions, as they should primarily protect fission product boundaries (containment, steam generator U tubes) and restore core/debris cooling to the extent practical. Sometimes, these actions are therefore called 'Candidate High Level Actions'. In general, they provide responses to the plant damage states defined above and are either initiated after recognition of such plant damage states or after certain parameters exceed their safety thresholds, depending on the approach chosen.

The actions should not be executed before both positive and negative consequences have been carefully considered, with all their side-phenomena, is contained in the SAMGs. Where quantitative information is needed or useful, again use is made of computational aids.

7.4.2. Development of severe accident management strategies

The first step of the Severe Accident Management Programme development is to decide and document the basic SAM strategies to be applied to the specific plant. The selected strategies and their implementation may depend on the basic approach chosen based on the national requirements. In case that plant modifications are involved to enhance the accident management programme, the degree of confidence in successful accident management actions will be increased.

In case that the accident management programme is developed from the generic programme based on application of the reference plant concept, the team should check that the differences between the actual plant and the reference plant are not so important that they would invalidate the strategies. Often it may be of crucial importance for the preventive strategies that the reactors as well as primary and relevant secondary system designs are similar. Respectively, for the mitigatory aspects the containment designs should be similar. If this is not the case, it may happen that the generic actions are still valid, but that they should be executed in a different order or initiated from other values of set-points.

When developing individual severe accident management strategies, interaction between various strategies may have influence both in case of a generic approach and in case of developing the accident management programme from the scratch. Feedback effects may appear among such actions as primary circuit depressurization, hydrogen management, filtered venting, and long term heat removal from the containment, and thus they should not be developed in separation. For some plant designs even slight differences may have a big impact on the interaction of selected strategies.

The resulting strategy/technical basis document will be applied for development of accident management procedures and guidelines.

7.4.3. Severe accident management guideline approach

The SAMG approach developed by the Westinghouse Owners' Group is widely used as a basis for developing guidelines. This approach for PHWRs was considered by CANDU Owners Group in Canada with modifications due to design differences. The principles for establishment of an effective severe accident management programme include:

- Ensuring a balance between organizational measures and design enhancements;
- Identifying the roles and responsibilities of the operating staff and special emergency teams;

- Identifying and evaluating plant systems and features suitable for use during severe accident management;
- Providing adequate training to the operating staff and special emergency teams.

7.4.3.1. Developing the guidelines

An important feature of SAMGs is that it is symptom based. That is, the exact sequence of events may not be known or identifiable from an analysis of plant status, or at least with certainty. Furthermore, relevant phenomena such as sheath melting, core disassembly or molten core concrete interaction could be progressing but there is no specific instrumentation to confirm it reliably. Symptoms are measurable or otherwise quantifiable conditions. For example, molten core concrete interaction generates gases that would pressurize containment. Here, SAMG would provide strategies to manage containment pressure thereby protecting this important fission product barrier. In this manner, the guidance is mitigatory and does not require knowledge of the event sequence.

Once it is decided to enter SAMG, the focus shifts to diagnosing the plant conditions that can challenge the ultimate fission product barrier — the containment. The key to the identification of the appropriate strategies to follow in response to an actual or imminent severe accident is the diagnosis of the nature of the challenges to the plant. Therefore, the SAMG diagnostic tools, the diagnostic flow chart (DFC) and severe challenge status tree (SCST), represent the cornerstone of effective severe accident management. Figure 15 shows the decision logic for the DFC and SCST.

Inherent in the structure of these tools is a hierarchy based on the immediacy and severity of the threat. The diagnostic tools contain plant parameters that are used for three purposes:

- Identification of current (or very imminent) challenges (in the SCST);
- Identification of potential or anticipated challenges (in the DFC);
- Identification of having achieved a controlled, stable state (in the DFC).

The highest priority is to protect containment. The SCST is the diagnostic tool that will 'call' a severe challenge guide (SCG) from a step in the SCST based on an urgent threat. These have been prioritized based on protecting the containment boundary, and include the following:

- SCG-1, Mitigate fission product releases (Objective: control site releases);
- SCG-2, Reduce containment pressure (Objective: depressurization without fission product release);
- SCG-3, Control containment atmosphere flammability (Objective: prevention of containment failure due to overpressure resulting from a Hydrogen burn at high concentrations;
- SCG-4, Control containment vacuum (Objective: prevention of containment failure due to vacuum).

These challenges are mutually exclusive so only one would occur at one time. The process loops until all challenges are mitigated, after which the priority shifts to maintaining the plant in a controlled stable state.

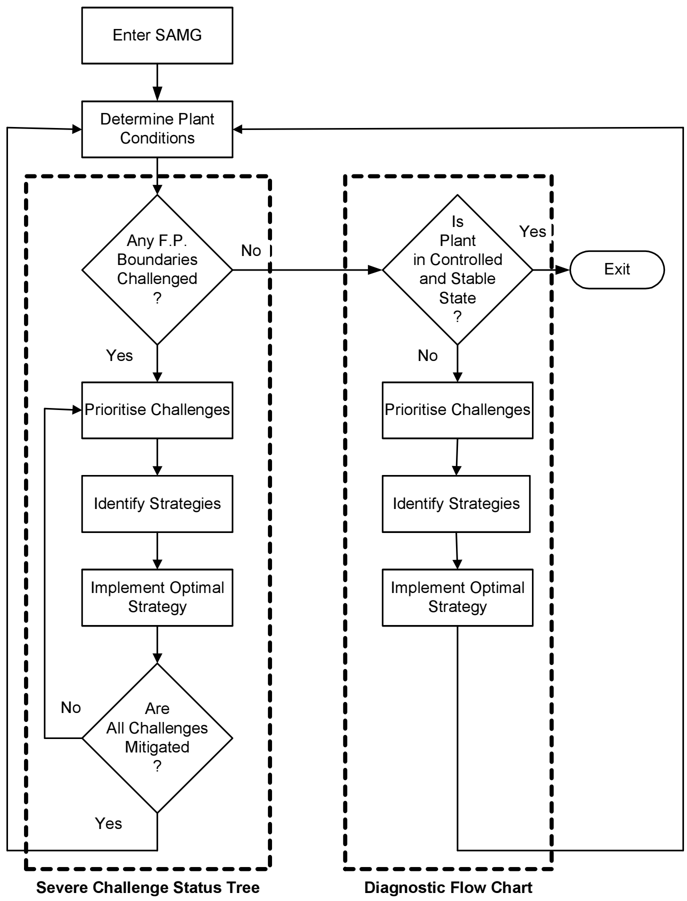


FIG. 15. Decision logic for the DFC and SCST.

The DFC is the diagnostic tool that monitors the critical safety parameters and ‘calls’ a severe accident guide (SAG) should conditions warrant it, based on a potential or anticipated challenge. The status of barriers is checked to identify any that are at risk based on the controlled, stable state parameters. If one were found to be at risk, then the relevant SAG would be called. The priority order of the SAGs is based on addressing the earliest expected challenges first, given that there are no short term challenges to containment (if there were it would be addressed in the SCG). The list of SAGs is as follows:

- SAG-1, Control shield tank conditions (Objective: prevent or delay rupture of calandria vessel);
- SAG-2, Control moderator conditions (Objective: prevent or delay fuel channel failure);
- SAG-3, Inject into the heat transport system (Objective: maintain channel integrity;

- SAG-4, Reduce fission product releases (Objective: remove or reduce the driving force for releases);

- SAG-5, Control containment conditions (Objective: prevention of future challenges to containment due to over-pressurization);

- SAG-6, Reduce containment hydrogen (Objective: prevention of containment failure due to overpressure resulting from a Hydrogen burn at high concentrations);

- SAG-7, Inject into containment (Objective: additional cooling water for heat removal from containment atmosphere, prevention of core-concrete interaction).

Unlike the SCST, there may be more than one challenge being mitigated at one time in the DFC loop. Thus, more than one SAG may be in effect at one time; however, only one strategy (called within a SAG) would be implemented at one time, in general. This allows for the evaluation of effectiveness of the selected strategy. If the strategy is not sufficiently effective, a second strategy may be added to supplement the first, if needed. Only certain key parameters need to be stable before SAMG can be exited.

The 'call' to any SCG or SAG is based on a given parameter reaching a setpoint. The entry 'setpoint' for each SCG is given a 'high enough' value to warrant urgent action, but also 'low enough' to allow some margin before the containment boundary is expected to fail. One example is very high containment pressure. The setpoint could be well above the design pressure of containment but below that at which through-wall cracking would occur.

The entry setpoint for a SAG would be the point where the condition exceeds a controlled and stable state. Continuing with the containment example, this could be set below the design pressure, taking into account station specific features.

Both the SCG and SAG have a similar structure illustrated in Table 3. The difference is in the evaluation of the benefits and negative impacts of strategies before they are implemented. The SCG has fewer evaluation steps due to its urgency, whereas the SAG includes assessing the negative impacts. The SAG requires comparing the benefit with the negative impact with the result that a strategy may not be implemented if the negative impact is judged to be unacceptable compared to the benefits. This option is not part of the SCG due to the urgency involved and the severe negative consequence (large fission product release) if the strategy is not implemented (i.e. benefits always outweigh the negative impacts in SCG regime).

7.5. Validation of severe accident management guidelines

Upon its establishment, a SAMG should be validated to confirm its effectiveness, usability, technical accuracy and scope. Periodic reviews of severe accident management programme should be undertaken to reflect changes in plant design or organizational responsibilities as well as new information derived from training programmes safety analyses

TABLE 3. STEPS IN THE GENERIC SCG AND SAG

Severe Challenge Guide (SCG)	Severe Accident Guide (SAG)
1) IDENTIFY available strategies	1) IDENTIFY available strategies
	2) DETERMINE capability of available equipment
	3) IDENTIFY and evaluate negative impacts
	4) DETERMINE if strategy should be implemented
2) IDENTIFY the preferred mitigation strategy and equipment line-up	5) IDENTIFY the preferred mitigation strategy and equipment line-up
3) IDENTIFY any limitations	6) IDENTIFY any limitations
4) DIRECT the control room staff to implement strategy	7) DIRECT the control room staff to implement strategy
5) VERIFY strategy implementation	8) VERIFY strategy implementation
	9) DETERMINE if additional mitigating actions are necessary to mitigate actual negative impacts
6) DETERMINE if challenge is being mitigated or if another strategy is required	10) DETERMINE if challenge is being mitigated or if another strategy is required
7) IDENTIFY long term concerns	11) IDENTIFY long term concerns
8) RETURN to Severe Challenge Status Tree (SCST) at step in effect	12) RETURN to Diagnostic Flow Chart (DFC) at step in effect

7.6. Implementation of severe accident management guidelines

After developing the SAMG, the next step is the effective implementation of the SAMG for achieving the severe accident management goals. Implementation of severe accident management effectively reinforces the defence in depth approach by enhancing the operator ability to cope with accidents well beyond the plant design basis. For optimum results in the implementation of severe accident management programme, it is crucial to assemble organizational groups consisting of experts in various disciplines. Adequate infrastructure must be available for implementation of SAMG. The infrastructure covers authority, organization, co-ordination of response, plans, procedures, logistic support and facilities, training, drills and exercises and quality assurance programmes. Detailed information on the implementation of the SAMG is in Ref. [8].

8. SUMMARY AND CONCLUSIONS

The analysis of severe accidents requires a proper understanding of different phenomena taking place during the progression of the severe accidents. These phenomena could take place 'in-vessel' or 'ex-vessel' (where in the case of a PHWR, the 'vessel' is the combination of calandria vessel and surrounding shield tank or calandria vault). While most of the phenomena taking place 'ex vessel' are common to all water reactors, there is a significant difference in the 'in vessel' phenomena between light water reactors and pressurized heavy water reactors. This difference is primarily due to the core of a PHWR consisting of horizontal reactor channels immersed in relatively cold moderator.

In PHWRs, some accidents that would be considered severe accidents in LWRs (e.g. LOCA with loss of ECCS) are included as part of the design basis events, and as such, the reactor is designed so that their consequences are mitigated. For PHWR severe core damage accidents, the sequence of events is identified through analysis so that accident management measures can be prescribed to minimize the risk to the public.

The analysis of severe accidents in general needs a multi disciplinary approach. This is also true for a PHWR where physics, thermal-hydraulics, structural integrity, chemical reactions and metallurgical considerations are closely interlinked with each other for the determination of accident progression. A large LOCA may result in a pulse in reactor power because of the positive void coefficient. The concerns in a stagnation channel break may be the integrity of the adjacent channels in addition to the fuel integrity of the affected channel. High temperature diffusion of oxygen in fuel, internally and externally leads to metallurgical transformations which change the melting point, and therefore the point at molten material relocation starts. These are just a few examples of the linkages between disciplines.

Most of the linkages between different disciplines have a significant role in determining the progression of a severe accident. A significant amount of world-wide research has been conducted, and is ongoing, to understand the different phenomena and their interdependencies. As this knowledge base grows, the uncertainties associated with the analysis of severe accidents diminish, as does the consequent sensitivity of the path of accident progression.

The computer codes available for carrying out the analysis of severe core damage accidents were originally developed for vessel-type reactors. While suitable for ex-vessel phenomena, these codes have to be significantly modified to be able to analyse in-vessel accident progression in PHWRs, most notably in the area of channel disassembly and core collapse. These modifications have been made in producing MAAP4-CANDU and ISAAC for severe accident analysis of PHWRs.

This report provides guidance on the analysis of severe core damage accidents in PHWRs, on the computer codes available for analysis of severe accidents, their capabilities and limitations, and lessons learned from performing severe accident analysis, with the aim, in particular, of supporting hardware modifications in existing nuclear power plants and the development of accident management programmes.

This report also provides a set of suggestions, using current international practices, on how to perform deterministic analysis of severe accidents by means of the available computer codes.

A general framework for these suggestions is provided, including description of factors important to the analysis, overview of severe accident phenomena and status in their modelling, categorization of available computer codes, and differences in approaches for various applications of analysis.

Guidance on the development of severe accident management is also provided in this report. The relationships with standard and emergency operating procedures are described, noting in particular the types of 'entry requirements' that dictate the transition from EOPs to SAMGs. Strategies for severe accident management are also described, with a focus on symptom based approaches.

The ability to include severe accident conditions in the design and operation of plants, in the development of accident management and severe accident mitigation strategies, has been dramatically enhanced over the past few years as many severe accident research programmes have been completed and mature codes have become available for general use. The rapid growth in the speed of computers and the enhanced performance and reliability of these codes has also been a significant factor making such activities much more affordable. However, systematic training of analysts in severe accident phenomena and the severe accident codes being used, the use of systematic methodologies to insure the validity of any calculations or plant simulations, and the participation in technical exchanges on severe accident research and code applications as well as the participation in international research projects and International Standard Problems are crucial to any successful application of severe accident technology.

In addition, the continued application of these tools and the lessons learned from the last two decades of severe accident research to the development of advanced plants, operating procedures, and training will help insure that the future work is devoted strictly to hypothetical severe accidents.

Appendix I

BRIEF DESCRIPTION OF MAIN EXPERIMENTAL FACILITIES

The following is a brief description of the main experimental facilities used to elucidate the phenomena for severe accidents in PHWRs, and to provide data for the development and validation of computer codes to model severe accident behaviour.

I.1. Experimental facilities in Canada

RD-14 M loop

The RD-14M loop is an 11MW, full elevation, scaled representation of a PHWR heat transport system located at AECL Whiteshell laboratories, as shown in Fig. I.1. The reactor core is simulated by ten, 6 m-long horizontal channels each containing seven electrically heated fuel element simulators. Each of the channels has end fitting simulators which are connected to headers by full length feeder pipes. Other key PHWR components represented in this facility include full height steam generators and heat transport pumps. These components are arranged in PHWR figure of eight geometry. The facility operates at typical primary system pressures (up to 10 MPa) and temperatures (up to 310^{o}C). Experiments are conducted in RD-14M to gain an improved understanding of the thermal-hydraulic behaviour of a PHWR during loss-of coolant accidents, under forced and natural circulation conditions and during shutdown scenarios. The data collected from this extensively instrumented facility are used to identify and examine phenomena observed in the heat transport system and forms a database for use in developing and validating thermal-hydraulic computer codes used to predict PHWR behaviour.

High temperature heat transfer laboratory

The high temperature heat transfer laboratory, located at the AECL Chalk River Laboratories, is used to investigate the integrated thermal-chemical-mechanical response of a CANDU fuel channel under accident conditions. Equipped to perform high-temperature (up to 1700^{o}C), high pressure (up to 10 MPa), transient heat-transfer experiments, this facility has experimental rigs that can handle pressure tubes 1.5–2.5 m long (example shown in Fig. I.2). These rigs can use graphite heaters, or fuel element simulators where the fuel elements influence fuel channel behaviour, for example through fuel element to pressure tube contact. Superheated steam can be flowed through the pressure tubes, and the calandria tube is submerged in a water tank simulating moderator conditions. Instrumentation is provided for pressure, mass flow, and temperature measurements, deformation of the pressure tube or calandria tube, and hydrogen production. Test rigs are equipped with view ports to record a test progress with high-speed cameras. Test series have been performed to investigate heat transfer phenomena and to establish the conditions under which a pressure tube and calandria tube may fail.

Large scale vented combustion test facility

The large scale vented combustion test facility (Fig. I.3), located at the AECL Whiteshell laboratories, is a 10 m long, 4 m wide, and 3 m high (120 m^3) rectangular structural steel building designed to withstand 600 kPa (impulse) and 300 kPa (static) internal pressures. The building is insulated and electrically-heated up to 120°C to simulate the wide range of thermodynamic conditions relevant to post-accident containment atmospheres. The end panels of the building are designed to provide a variable vent area from 0.4 m^2 to 7.2 m^2. Internally, the building has removable walls to create two or three vented sub-volumes of 60 m^3 and 30 m^3. The facility has three separate gas addition systems for steam, hydrogen, and inert gases. Hydraulic fans inside the test chamber are used to mix the gases and can be used to provide turbulent conditions during combustion. Instrumentation includes pressure transducers, thermocouples, and gas sampling by a mass spectrometer, at various locations inside the test chamber.

The large scale vented combustion test facility was designed to systematically quantify effects of key thermodynamic and geometric parameters affecting pressure development during vented combustion under conditions relevant to ignition. The facility has good control of initial thermodynamic conditions, is sufficiently large to capture the effects of scale, and is geometrically similar to rooms (e.g. flat walls and square corners).

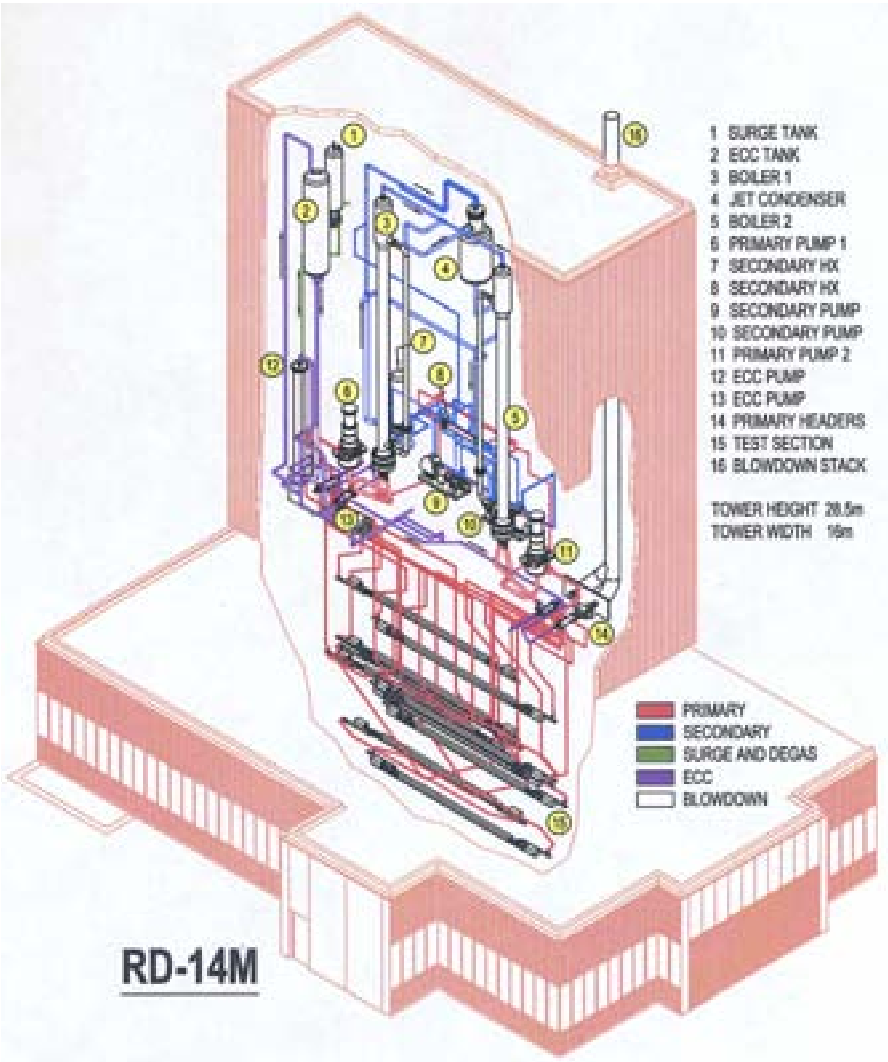


FIG. I.1. Schematic diagram of the RD-14M thermal-hydraulic test facility.

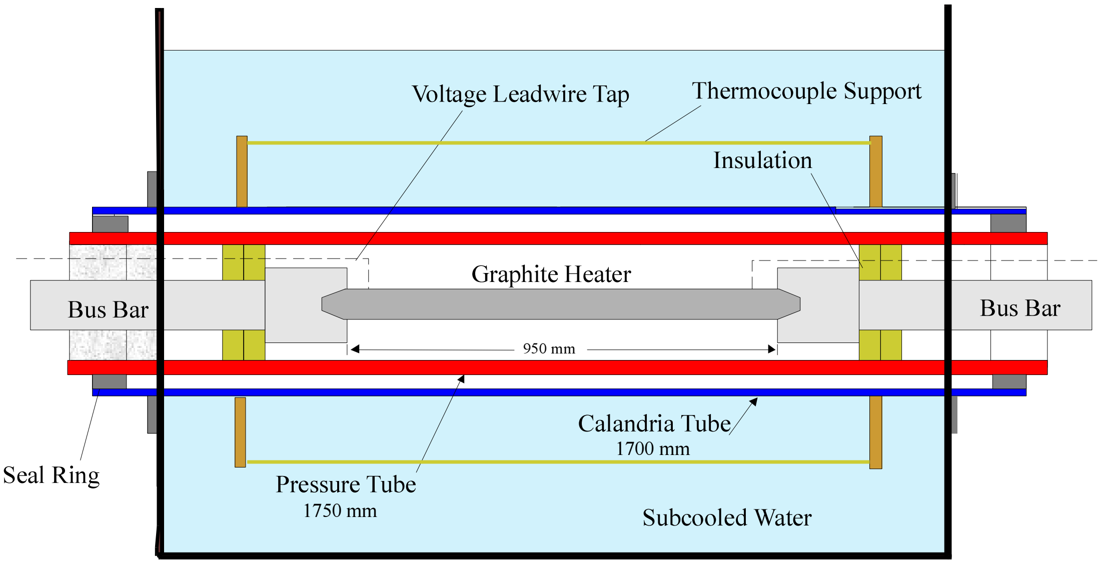


FIG. I.2. Schematic diagram of an apparatus to investigate fuel channel thermal-mechanical behaviour under accident conditions.

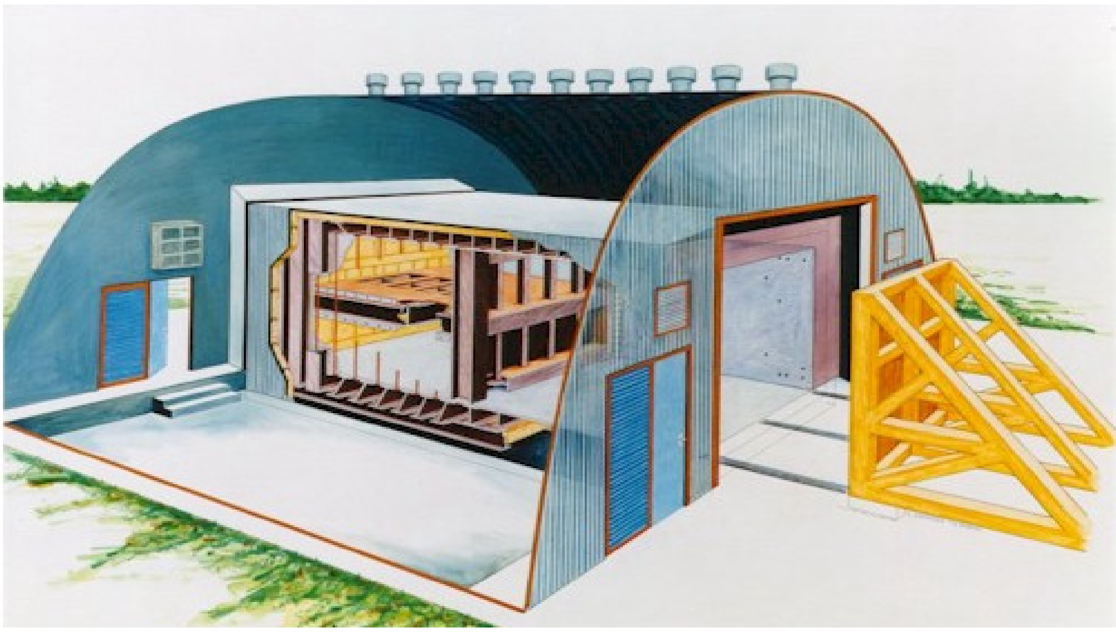


FIG. I.3. Schematic diagram of the large scale vented combustion test facility.

Large scale gas-mixing facility

The large scale gas mixing facility was a 1000 m^3 room at the AECL Whiteshell laboratories used to study containment thermal-hydraulics, in particular the behaviour of steam and air mixed with helium (to simulate hydrogen) as shown in Fig. I. 4. The room was insulated and had injection points for steam and helium. Internal partitions were used to simulate sub-compartments. Instrumentation included temperature measurements, condensate collection and gas phase sampling. The facility was used to study mixing, buoyancy-induced flows, stratification, condensation, and effects of containment partitions, and has provided data for use in validation of containment thermal-hydraulic codes and predictions of hydrogen distributions. The large scale gas mixing facility was closed in 2001, and has been replaced by the large scale containment facility at the AECL Chalk River laboratories.

FIG. I.4. Schematic diagram of the large scale gas mixing facility

Radioiodine test facility

The radioiodine test facility operated at AECL Whiteshell laboratories over the period of 1988 to 1999. The radioiodine test facility was an intermediate-scale facility that provided for many combinations of potential reaction media (gas phase, aqueous phase and a variety of surfaces) and conditions (pH, temperature, radiation, various initial concentrations and initial speciation of iodine) to simulate the chemical system expected in a reactor containment building following an accident. The schematic flow chart and a photograph of the radioiodine test facility are shown in Fig. I.5. Over 50 tests were performed in the radioiodine test facility before it was decommissioned in 1999. The radioiodine test facility programme revealed the importance of a number of unexpected phenomena relating to iodine behaviour that were not previously observed in large-scale studies or bench-scale studies. The integrated tests in the Radioiodine Test Facility have also allowed for the evaluation of the relative importance of physical and chemical processes in determining speciation and volatility of iodine, and have provided critical data for the development and validation of the models for predicting iodine behaviour in an accident. Various chemical and physical measurements performed in each test (such as iodine speciation and measurements of water radiolysis products and impurities) have provided data with which to examine the validity and self-consistency of various subsets within the mechanistic models for predicting iodine behaviour.

Molten fuel moderator interaction facility

The molten fuel moderator interaction facility has recently been constructed at the AECL Chalk River laboratories to investigate the high pressure ejection of corium melt into the moderator as shown in Fig. I.6. The experiments consist of heating up a mixture of UO_2, Zr, and ZrO_2, representative of the molten material expected in a fuel channel, inside a short length of pressure tube. Once the molten material has reached the desired temperature, ~2400^{o}C, the pressure inside the tube is raised to about 10 MPa, and the pressure tube fails at a pre-machined flaw, releasing the molten material into the surrounding tank of water. The planned experiments will cover two different amounts of molten material, and will investigate the effects of the material interacting with tubes representing neighbouring fuel channels.

Blowdown test facility

The blowdown test facility experiments were performed in the NRU reactor at the AECL Chalk River laboratories. Four blowdown test facility in-reactor experiments were performed to improve the understanding of PHWR fuel and fission-product behaviour under accident conditions, and to provide data for use in reactor safety code validation.

The insulated test assemblies were oriented vertically in a zircaloy re-entrant flow tube, which fitted inside a thick-walled stainless-steel pressure tube (test section) located in the reactor core (see Fig. I.7). Test assemblies in the blowdown test facility were cooled with pressurized water or saturated steam. An accident sequence was initiated by isolating the in-reactor test section from the rest of the coolant loop, and voiding the coolant through an instrumented blowdown line and a wire mesh filter into a sealed tank. Steam, inert gas and cold water were used for post-blowdown cooling in the blowdown test facility. The blowdown line was instrumented to measure coolant thermal-hydraulic parameters and fission product gamma emissions.

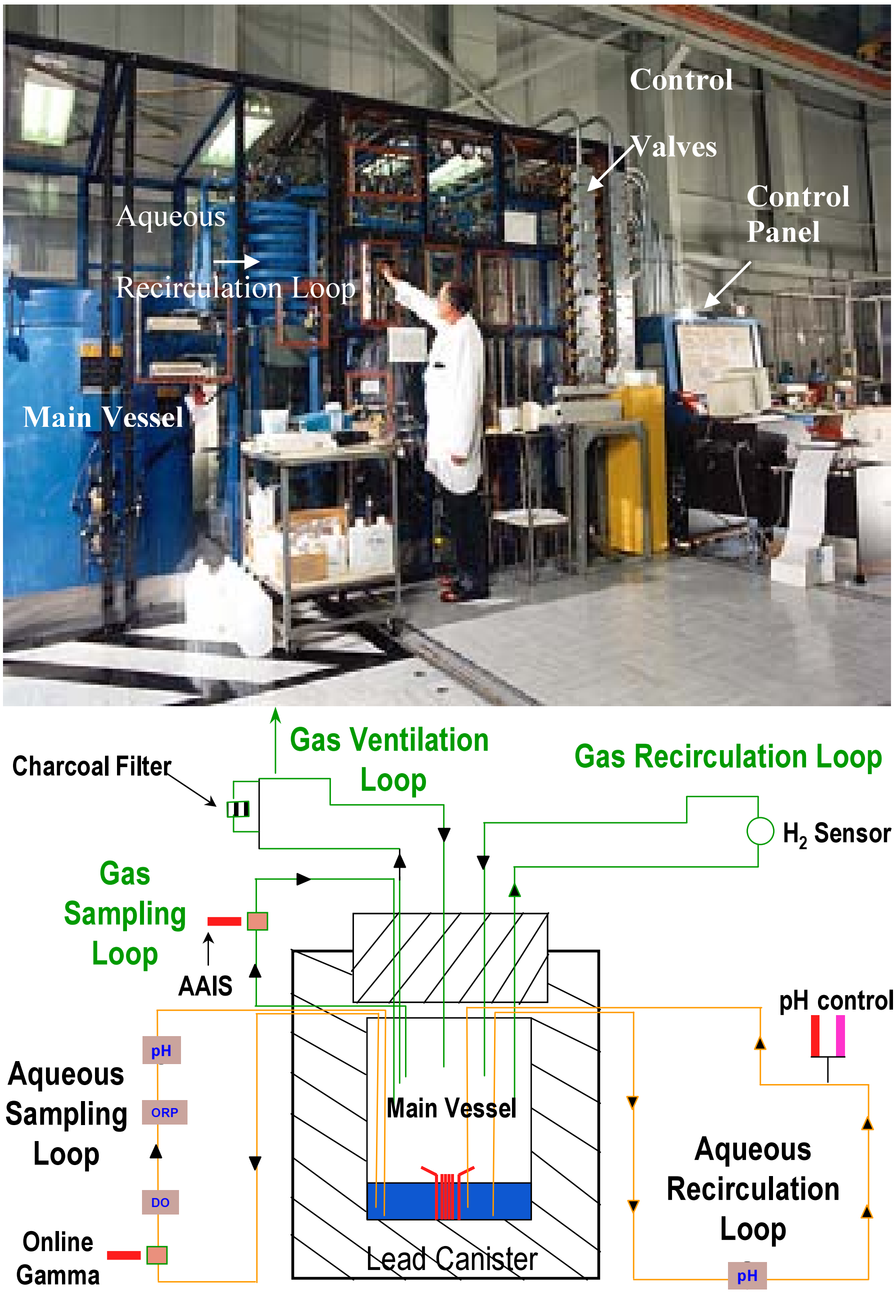


FIG.I.5. The exterior and the schematic flow chart of the radioiodine test facility the automated airborne iodine sampler, oxidation/reduction potential probe and dissolved oxygen probe.

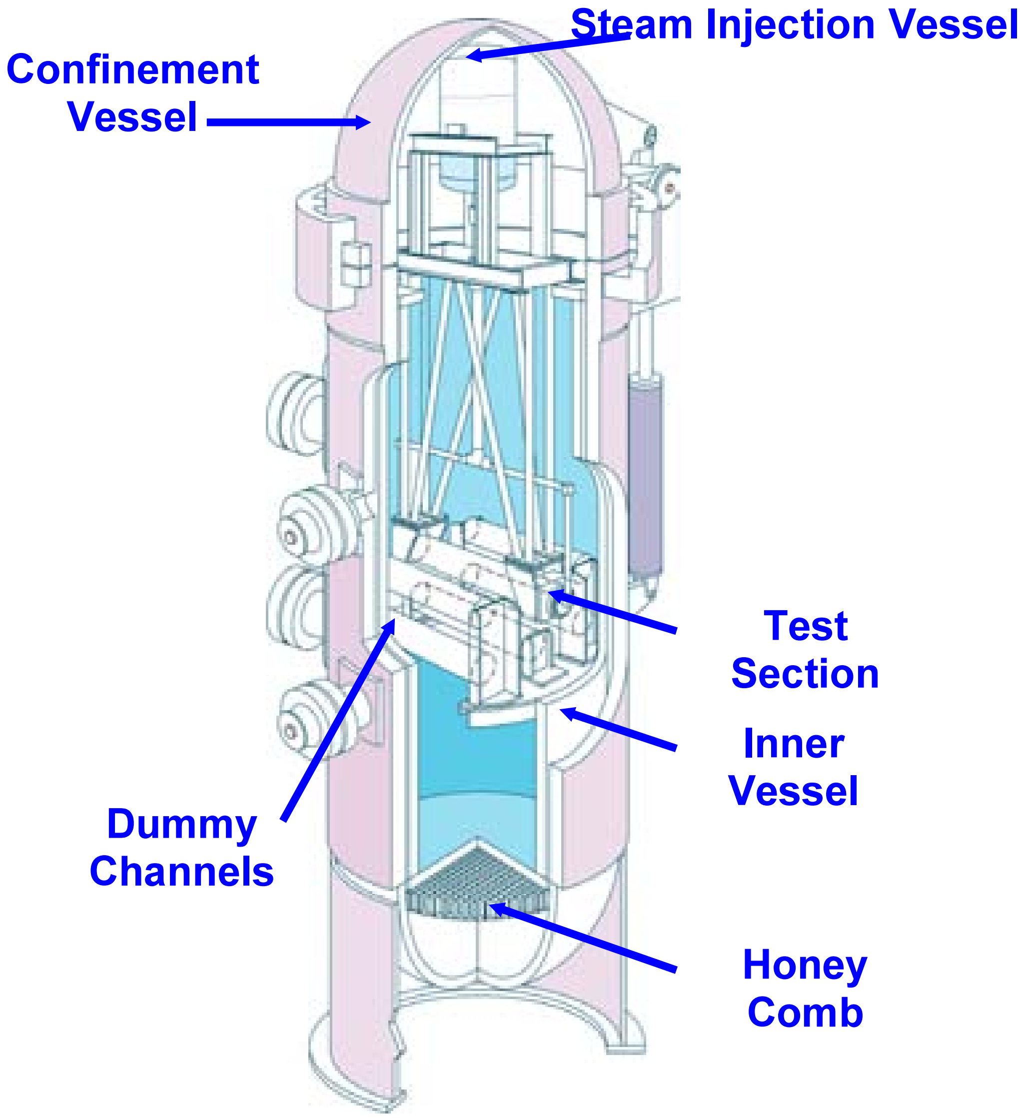


FIG. I.6. Schematic diagram of the molten fuel moderator interaction facility.

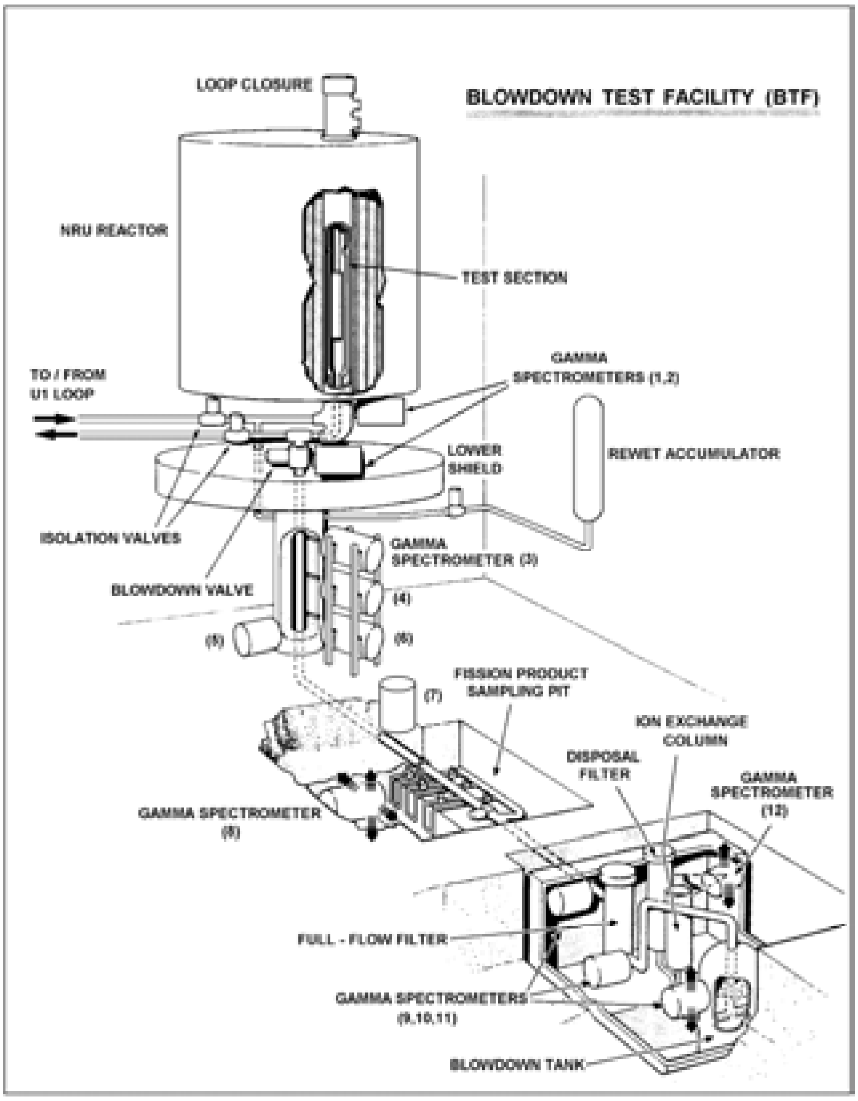


FIG. I.7. Schematic diagram of the blowdown test facility in the NRU reactor.

In the BTF-107 experiment, a three-element cluster of PHWR-sized fuel elements was subjected to severely degraded cooling conditions resulting in a high-temperature (≥2400°C) transient. A partial flow blockage developed during the test due to relocation of a molten U-Zr-O alloy and the high-temperature transient was terminated with a cold-water quench.

The other three experiments in the blowdown test facility programme, BTF-104, BTF-105A and BTF-105B, were conducted with single PHWR-sized fuel elements at maximum temperatures of 1500–1900°C in a steam-rich environment. The tests were performed to evaluate the behaviour of a PHWR fuel element and the resultant fission-product release and transport in a LOCA/Loss of ECCS scenario.

- The BTF-104 experiment provided data on fuel behaviour, and volatile fission-product release and transport (Kr, Xe, I, Cs, Te and Ba) from a previously irradiated fuel element at a volume-averaged fuel temperature of about 1500°C.

- The BTF-105A experiment used an internally instrumented fresh fuel element and provided data for validation of transient fuel performance codes and tested instrumentation for the BTF-105B experiment.

- The BTF-105B experiment investigated fission-product release and transport from a previously irradiated fuel element at an average fuel temperature of 1800°C. Due to improved measurements of fuel-cladding temperature, flow and neutron flux, and better control of steam condensation in the test section, the thermal-hydraulic boundary conditions for the BTF-105B test were better quantified than for previous tests at the blowdown test facility.

Core disassembly facility

The progression of a severe core damage accident in a PHWR reactor is typified by boil-off of the moderator and sequential failure of fuel channels as they become uncovered. An experimental programme has been initiated at AECL Chalk River laboratories to investigate this core disassembly process. It is based on a facility with 1/5 scale fuel channels, heated by individual tungsten heaters to simulate fuel bundles. An array of 4 such channels can be subjected to a predicted transient to investigate the formation of suspended debris through the interaction of failed channels with lower intact channels. Figure I.8 shows the results of a typical test. To date, tests completed in an inert atmosphere have shown that the debris that is formed will typically be long lengths of fuel channel that are suspended by lower channels, leaving short stubs connected to the calandria vessel end wall. Future tests will investigate the effects of an oxidizing environment on channel disassembly.

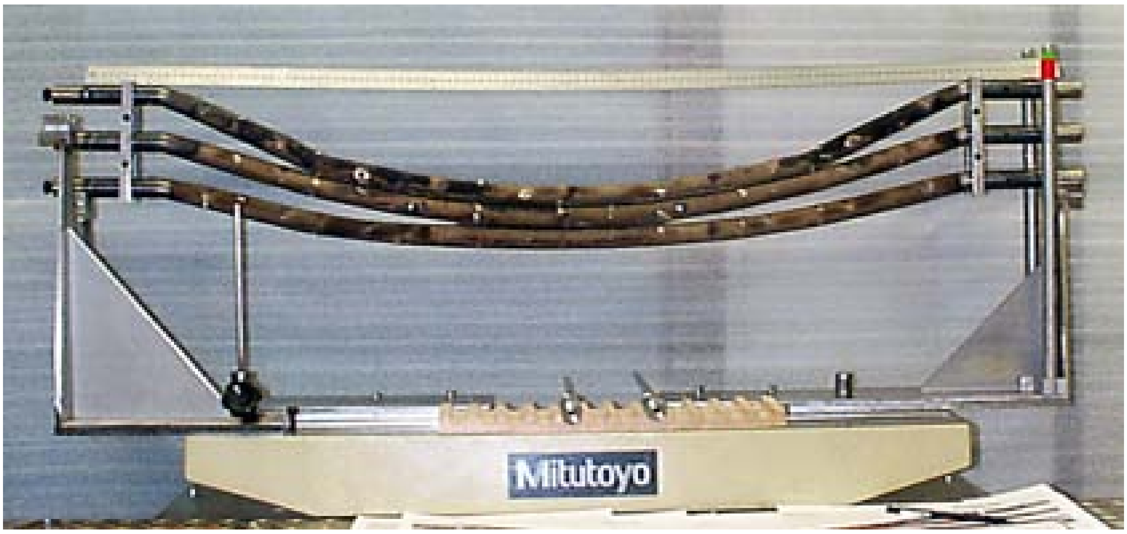


FIG. I.8. Post-test view of a core-disassembly test.

I.2. Experimental facilities in Republic of Korea

LAVA facility

In-vessel corium retention measure is considered as one of the most promising severe accident management strategies. The in-vessel corium retenrion can be achieved by in-vessel re-flooding and/or flooding outside of the reactor vault in the course of a severe accident. To investigate the possibility of in-vessel corium retention, two kinds of experiments were performed using LAVA (**L**ower-plenum **A**rrested **V**essel **A**ttack) facility. One is done to investigate the possibility of gap cooling (LAVA experiment and LMP200 experiment) and the other is done to investigate the feasibility of the external vessel cooling (LAVI-ERVC experiment).

The LAVA facility consists of a thermite melt generator, a melt separator, and a test section of the lower head vessel. A schematic diagram of the LAVA experimental facility is shown in Fig. I.9. The LAVA facility can be pressurized up to 3.0 MPa by the gas supply system, as an experimental parameter of accident condition. The hemispherical test vessel for LAVA experiment for which 70 kg of thermite is used, is a 1/8 linear scale mock-up of the reactor vessel lower plenum. The inner diameter and the thickness of the vessel are 50 cm, 2.5 cm, respectively. The dimension of the vessel was determined by preserving the average membrane stress across the vessel which is a main parameter affecting the creep rupture failure of the lower head vessel induced by thermal and pressure load. The cylindrical parts of the lower head vessel was equipped with two 20 kW band heaters to heat water inside the vessel up to the desired temperature. LMP200 experiments were conducted in the test vessel simulated with a 1/5 linear scale mock-up and of the reactor vessel lower plenum and used 200 kg of thermite.

LAVA and LMP200 experiments showed that the gap is formed between the corium and the inner surface of reactor vessel but the possibility of in-vessel corium retention through in-vessel gap cooing depends on the size of gap formed and the amount of corium relocated to the lower plenum.

From the LAVI-ERVC experimental results, it could be concluded that the insulation design which allows sufficient water ingression and steam ventilation could increase the possibility of in-vessel corium retention through the external reactor vessel cooling.

TROI facility

Whether the steam explosion will occur or not during severe accident was investigated using TROI (**T**est for **R**eal c**O**rium **I**nteraction with water) facility in Korea Atomic Energy Research Institute. Figure I.10 shows a schematic diagram of the TROI facility.

After preliminary tests using ZrO_2, experiments using a mixture of ZrO_2 and UO_2 were performed. A molten corium in the form of a jet was poured into a sub-cooled water pool at an atmospheric pressure. Spontaneous steam explosions were observed in the experiments which used pure zirconia and mixture of 70% UO_2 and 30% ZrO_2. For the mixture of 80% UO_2 and 20% ZrO_2, the spontaneous steam explosion was not observed. This means that the occurrence of an energetic steam explosion is highly dependent on the composition of the melt.

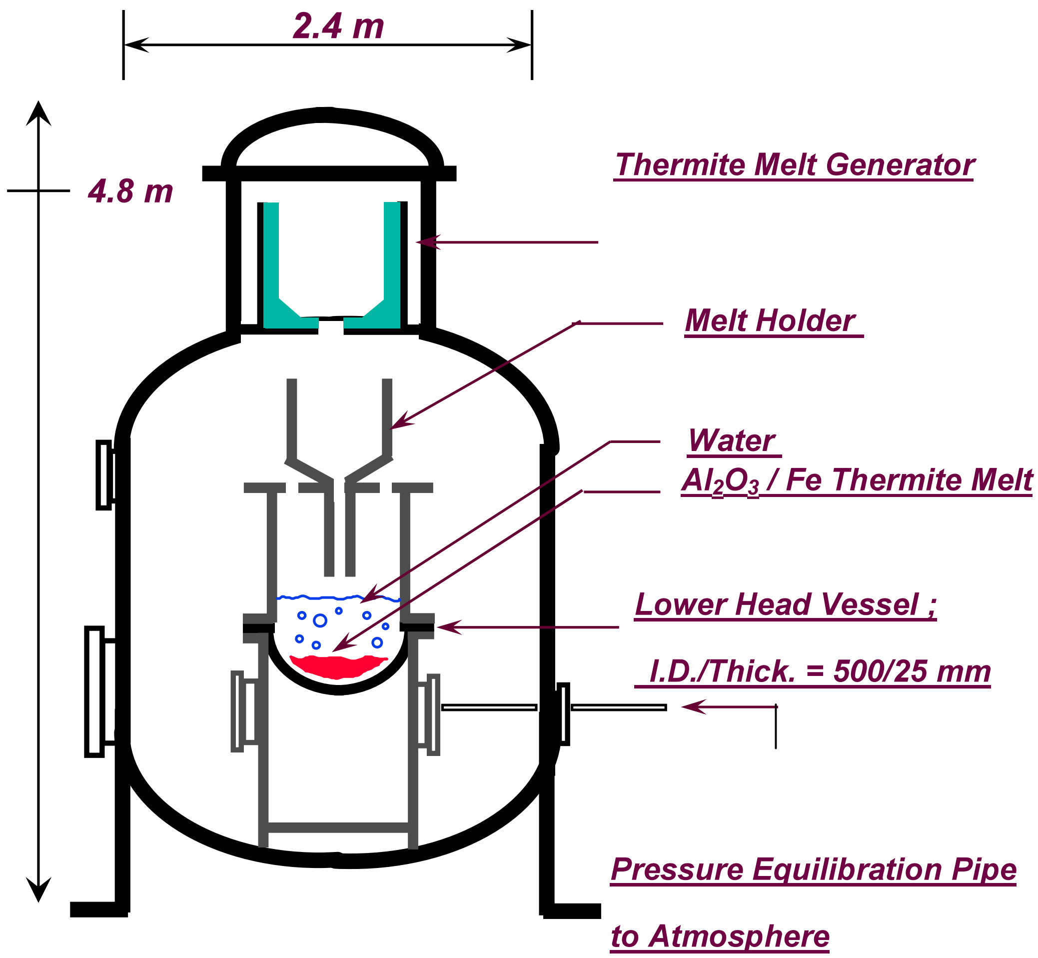


FIG. I.9. Schematic diagram of the LAVA experiment facility.

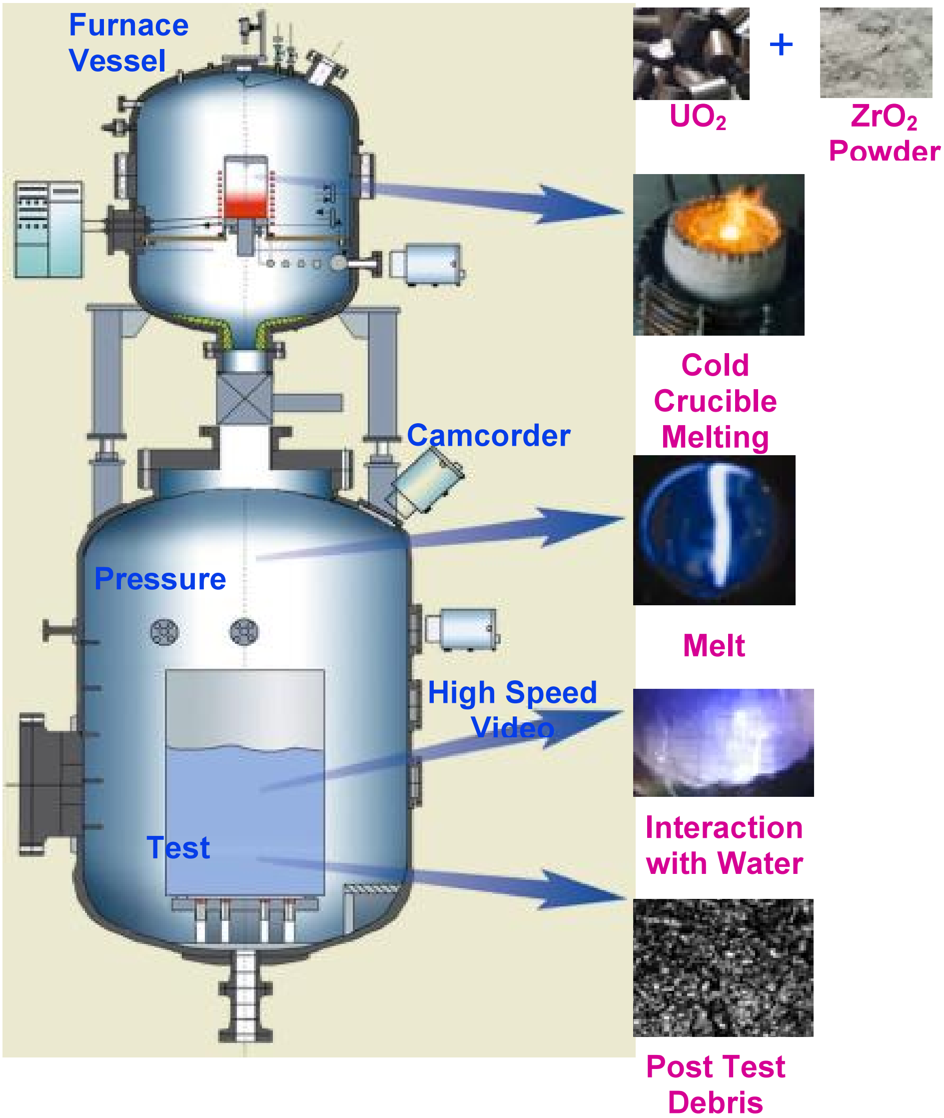


FIG. I.10. Schematic diagram of TROI facility.

Test facility for hydrogen burn

The survivability of the essential equipment is very important to manage the accident progression during the severe accident. The availability of the existing equipment and instrumentations depends on the temperature profile during the severe accidents. Hydrogen burn in a large dry containment may pose a serious threat to the equipment inside the containment. A quenching mesh was suggested to stop the propagation of hydrogen flame.

A performance test of the quenching mesh that is for hydrogen gas was conducted within a closed model compartment near an atmospheric pressure for an application in nuclear power plants. The experimental set up is shown in Figs. I.11 and I.12.

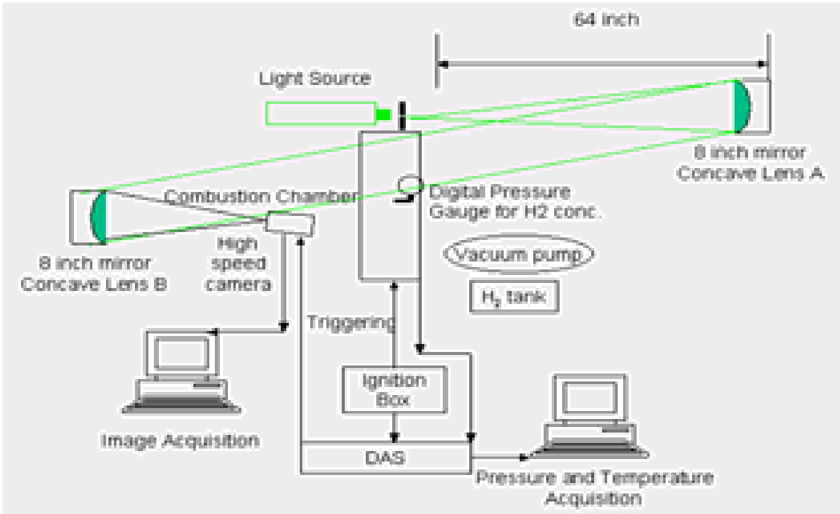


FIG. I.11. A schematic diagram of the experimental apparatus.

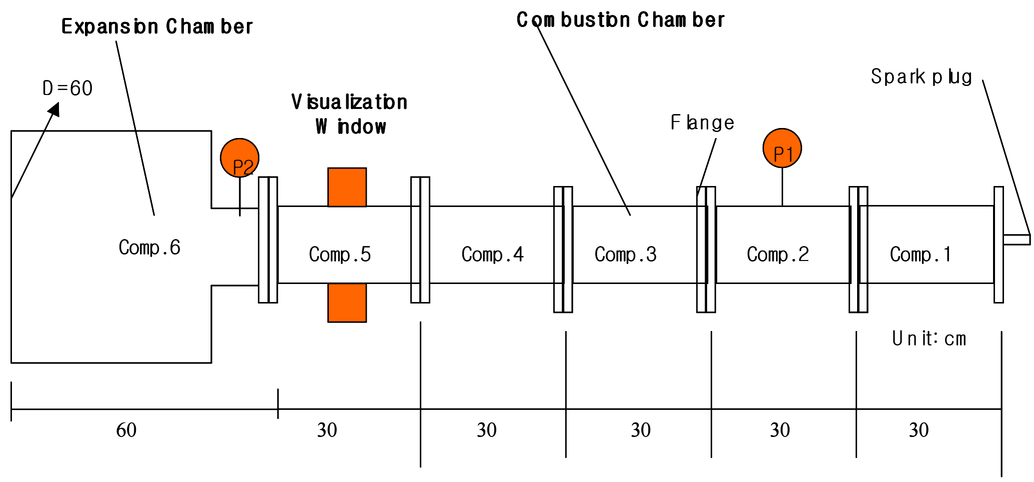


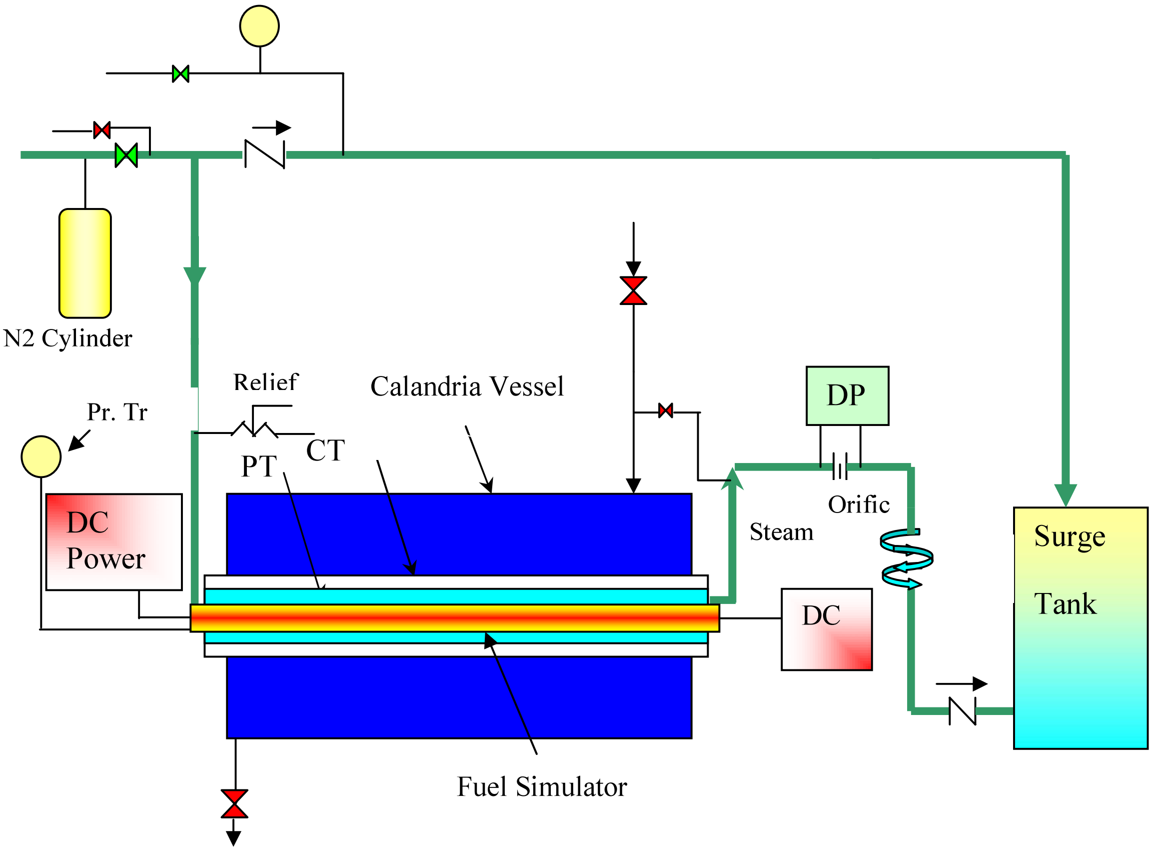
FIG. I.12. The combustion chamber.

I.3. Experimental facilities in India

Channel heat up facility

A channel heat up experimental is set up at Indian Institute of Technology, Roorkee to study the channel deformation behaviour at high temperature. A schematic of the set up is shown in Fig. I.13. It consists of a pressure tube enclosed in a calandria tube. The length of the tube is equal to the length between two garter springs in Indian PHWRs. The calandria tube is immersed in proportionate amount of water. Fuel weight is simulated. The pressure tube is electrically heated. Temperature is monitored at various locations in the pressure tube, calandria tube and moderator. The facility is capable of being flushed with an inert gas prior to starting the experiment. In the first phase of experiments pressure tube sagging experiments has been carried with and without water.

Experiments without water carried out to measure pressure tube deflection with water to study the boiling heat transfer. Subsequently the facility has been modified for carrying out symmetric and asymmetric pressure tube ballooning experiments.



PT: pressure tube, CT: calandria tube, DP: differential pressure, and DC: direct current

FIG. I.13. Experimental set-up for molten material–coolant.

Molten material coolant interaction

An experimental facility is being set up to study molten material coolant interaction in number of phases to validate different modules of molten fuel coolant interaction code. The facility provides to study film stability experiments, limited molten material release. A schematic of the experimental set up is shown in Fig. I.14. It consists of a melt generator where a low melting point would be melted in an inert atmosphere. Scaled down calandria together with dummy reactor channels are simulated. Pressure and temperature are measured at various locations along with water level and steam flow. The facility will be extended for large amount of molten material release along with high pressure steam discharge.

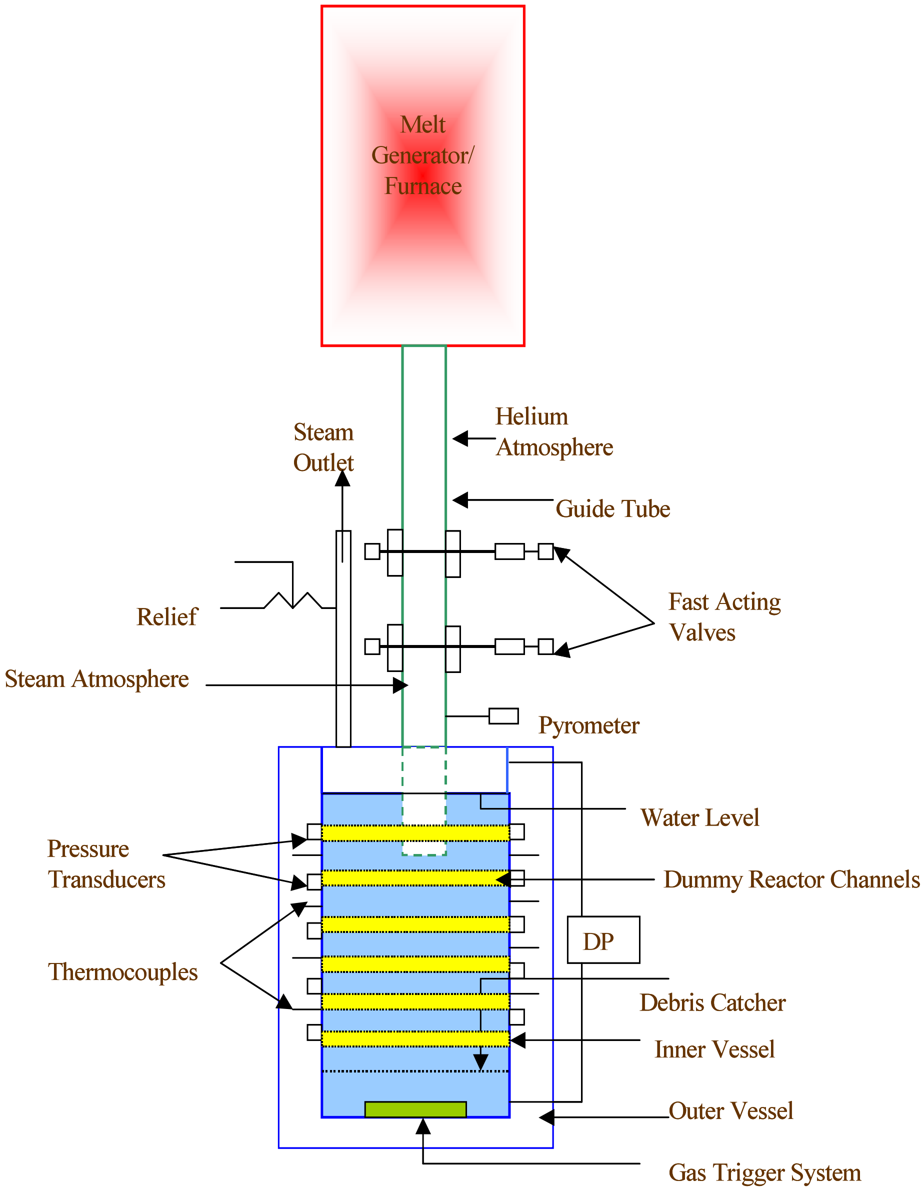


FIG. I.14. Experimental set-up for molten material–coolant interaction.

Appendix II

MAIN FEATURES OF INTEGRATED SEVERE ACCIDENT CODES

II.1. MAAP4-CANDU

The Modular Accident Analysis Program for CANDU Nuclear Generating Station MAAP4-CANDU is a computer code that can simulate the response of the CANDU NPPs during severe accident conditions, including actions undertaken as a part of accident management. MAAP4-CANDU Code was developed on the base of MAAP4 code (used for PWRs and BWRs);

Use of MAAP4-CANDU allows predicting quantitatively the evaluation of severe accidents starting from normal operating full power conditions and implying a set of system faults and initiating events through events such as PHTS inventory blowdown, core heat up and melting, PHTS failure, calandria tank, reactor vault failure and containment failure. Furthermore, some models are included in the code that allow to analyze opportunity to stop the accident (or mitigate its sequences) by cooling debris in the calandria vessel or containment.

II.1.1. MAAP4-CANDU code structure

MAAP4-CANDU has moduler structure. The subprogrammes comprising MAAP4-CANDU are at four levels:

1) The high level (executive) subroutines;
2) The system and region subroutines;
3) The phenomenology subroutines;
4) The property and utility subroutines.

The **high level subroutines** include the main programme, the input-output subroutines, the data storage and retrieval subroutines and numerical integration subroutines. The time integration subroutines INTRT and DIFFUN control the time step and call the system and region subroutines at each time step during an accident transient.

The **system and region subroutines** include the EVENTS subroutine which set the event flags (Boolean variables) giving the status of the system and the status of operator interventions. The event flags control code execution. Region subroutine defines the differential equations for the conservation of internal energy and mass. System subroutine examines inter-region flows. The system and region subroutines pass global variables by common blocks and operate on them by calling the phenomenology subroutines.

The **phenomenology subroutines** describe the rate of the physical processes taking place in each region of the NGS model. The phenomenology subroutines are generic in nature and can be called by any of the system or region subroutines or by other phenomenology subroutines.

This modularization allows the fundamental physical models to be changed by altering or rewriting a subroutine independent of the rest of the MAAP4-CANDU subroutines.

The **property and utility subroutines** give the physical properties (e.g. specific heat, saturation pressure, viscosity, etc.) of the important materials (e.g. steam, water, air, etc.). Property and utility subroutines are generally called by the phenomenology subroutines.

Code is written on FORTRAN77 language and contains 514 subroutines.

II.1.2. MAAP4-CANDU model

- Two-loops (for CANDU 6) or one loop (CANDU 9) PHTS including piping, pumps, outlet headers and inlet headers, feeders;
- Pressurizer and pressure and inventory control system;
- CANDU reactor core assembly: fuel channels, calandria vessel, shield tank, reactor vault;
- Steam generators — primary and secondary sides;
- Containment building including a number of compartments;
- moderator cooling system;
- End shields cooling system;
- Shutdown cooling system;
- Reserve water system (for CANDU 9, ACR);
- Emergency core cooling system (high, medium and low pressure components);
- Dousing system;
- Local air coolers;
- Crash cooldown system;
- Power operated and passive (spring loaded) relief valves.

II.1.3. Physical processes and phenomena

MAAP4-CANDU treats the spectrum of physical processes that might occur during an accident such as steam formation, core heat up, calandria and reactor vault failure, core debris-concrete interaction, ignition of combustible gases, steam explosions, fluid entrainment by high velocity gas, fission product release, transport and deposition. The most important distinguishing feature of the M4C is the model of the CANDU horizontal reactor core with fuel channels situated inside pressure and calandria tubes. The important processes and phenomena which control the core behaviour are modelled in the CANDU Channels System, including:

- PHTS heat transport;
- PHTS loss of water inventory and accumulation in the containment;
- ECCS sources, sinks and paths;
- Uncovering of the core, heat up and hydrogen and fission product release into the PHTS;
- Hydrogen release into the containment;
- Fission product transport and deposition in the PHTS;
- Core material migration, fragmentation, steam generation and additional hydrogen formation in calandria vessel and shield tank;
- Calandria vessel and/or shield tank failure and ablation;
- Material creep and possible rupture of PHTS components, calandria vessel and shield tank walls;
- Ex-vessel steam generation and hydrogen formation;
- Core debris entrainment and coolability, concrete attack and carbon monoxide generation;
- Hydrogen and carbon monoxide combustion in containment;
- Temperature excursion and deformation of fuel and fuel channels and interactions with the moderator system;
- Zircaloy-steam reaction;
- Thermal mechanical failures of fuel channels ;
- Disassembly of fuel channels;
- Formation of suspended solid debris beds;
- Motion of solid and molten debris bed;
- Interaction of the core debris with the calandria vessel;
- Ex-vessel heat transport, water inventories and containment cooling ;
- Long term PHTS hating due to deposited fission products;
- Fission product transport and deposition in the containment compartments;
- Containment failure or venting and depressurization.

II.1.4. CANDU 6 nodalization for MAAP4-CANDU analysis

II.1.4.1. Core

CANDU 6 core has 380 fuel channels arranged in 22 rows and 22 columns. For the CANDU 6 core a simplified fuel channel model was used. For this purpose, the 22 rows were divided into 6 vertical nodes, with 4, 4, 3, 3, 4, 4 rows in each node. Within each vertical node, we further divide fuel channels into 3 groups according to their fuel powers. The 22 columns of fuel channels are divided into 2 loops symmetric to the vertical axis. The left 11 columns are in one loop and the right 11 columns are in the other loop. Thus, we have reduced 380 fuel channels to 2 loops, and each loop has 18 representative channel groups.

In CANDU 6, each fuel channel has 12 fuel bundles. These 12 fuel bundles are divided into 5 axial nodes, with 2, 3, 2, 3, and 2 bundles in each axial node. Thus, the total 380 channels, each containing 12 fuel bundles are modelled by a total of 90 nodes in 3-dimensions for each loop (see Fig. II.1).

In CANDU 6, each fuel channel consists of a calandria tube, a pressure tube, and 37 fuel elements arranged in 4 rings (central, inner, intermediate and outer rings) (see Fig. II.2). Each of inner, intermediate and outer fuel rings as 2 rings are modelled. Thus, CANDU 6 fuel channel is represented by 9 rings (see Fig. II.3).

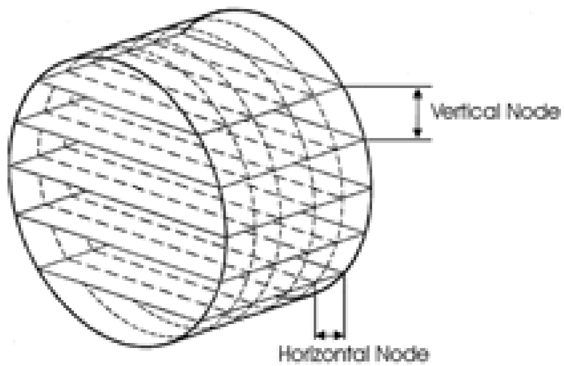


FIG. II.1. Nodalization scheme for CANDU 6 fuel channels according to the elevations and channel axial direction.

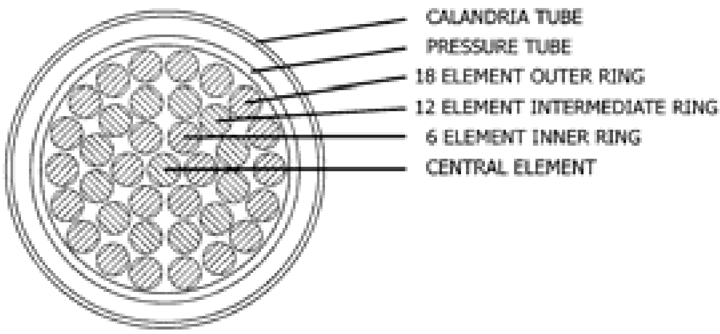


FIG. II.2. CANDU 6 fuel bundle consisting of a calandria tube, a pressure tube and 37 fuel elements

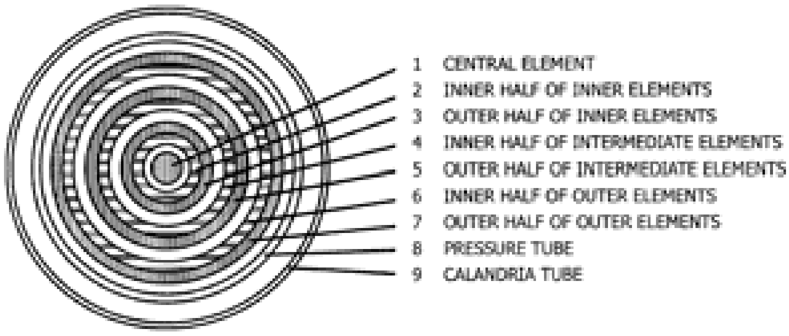


FIG. II.3. Nodalization scheme for CANDU 6 fuel channel.

II.1.4.2. Primary heat transport system

The PHTS (see Fig. II.4) comprises of two loops, each loop serving 190 of the 380 fuel channels. Each loop contains two steam generators, two pumps, two inlet headers, and two outlet headers. Feeders connect the inlet and outlet end of fuel channels to inlet and outlet headers respectively. The CANDU 6 design is such that the flow through the fuel channels in one loop follows the shape of 'figure of eight' with some channels carrying the flow inward and others outward from the reactor face.

The PHTS was modelled as 15 nodes (see Fig. II.5). Feeder pipes connecting headers and fuel channels are also modelled in the Code.

II.1.4.3. Calandria vessel wall

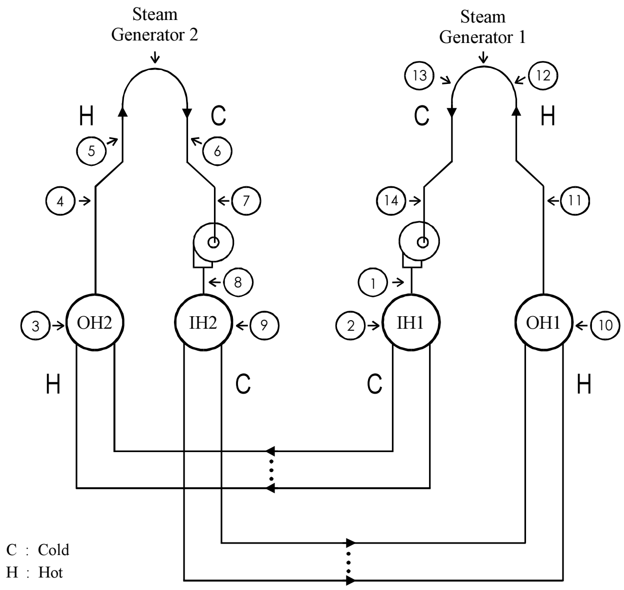
MAAP4-CANDU tracks the calandria wall temperature at different elevations. For this purpose, the calandria vessel walls at different elevations were nodalized. Two types of walls are considered: one wall-type represents the front and back face of the vessel and the other represents the shell of the vessel. Both types of walls are sliced horizontally to 15 nodes from the vessel floor to the ceiling along the elevation levels (see Fig. II.6). Each node has the same temperature along the calandria vessel shell length.

II.1.4.4. CANDU 6 containment

There are many rooms (or compartments) inside the CANDU 6 containment. It is impractical to model each room even as one node, because of the computer run time and such details are not used in MAAP4-CANDU analyses. It is customary to combine several rooms as one node based on some rationale. 3 nodes and 31 flow junctions are used to represent CANDU 6 containment (see Fig. II.7). The node numbers and their corresponding containment room numbers are given in Table II.1. A total of 90 wall heat sinks in containment are modelled in the present work.

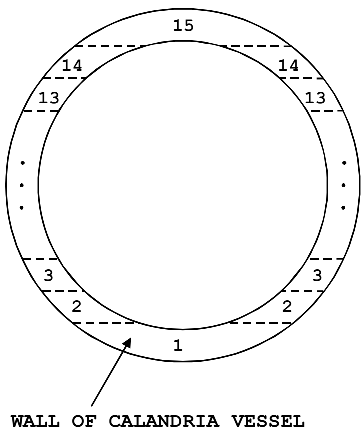
II.1.4.5. Steam generator

The steam generator is hard-wired as one node in MAAP4-CANDU, so the user does not need to enter node numbers. However, the user needs to provide some variable values for the primary and secondary sides of the steam generator, such as, pressure set point for safety relief valves, total volume of the primary side, total number of U-tubes, inside diameter of the U-tube, the diameter of tube sheet, the height of the shell above the tube sheet, a table of volume versus height in the secondary side, etc.



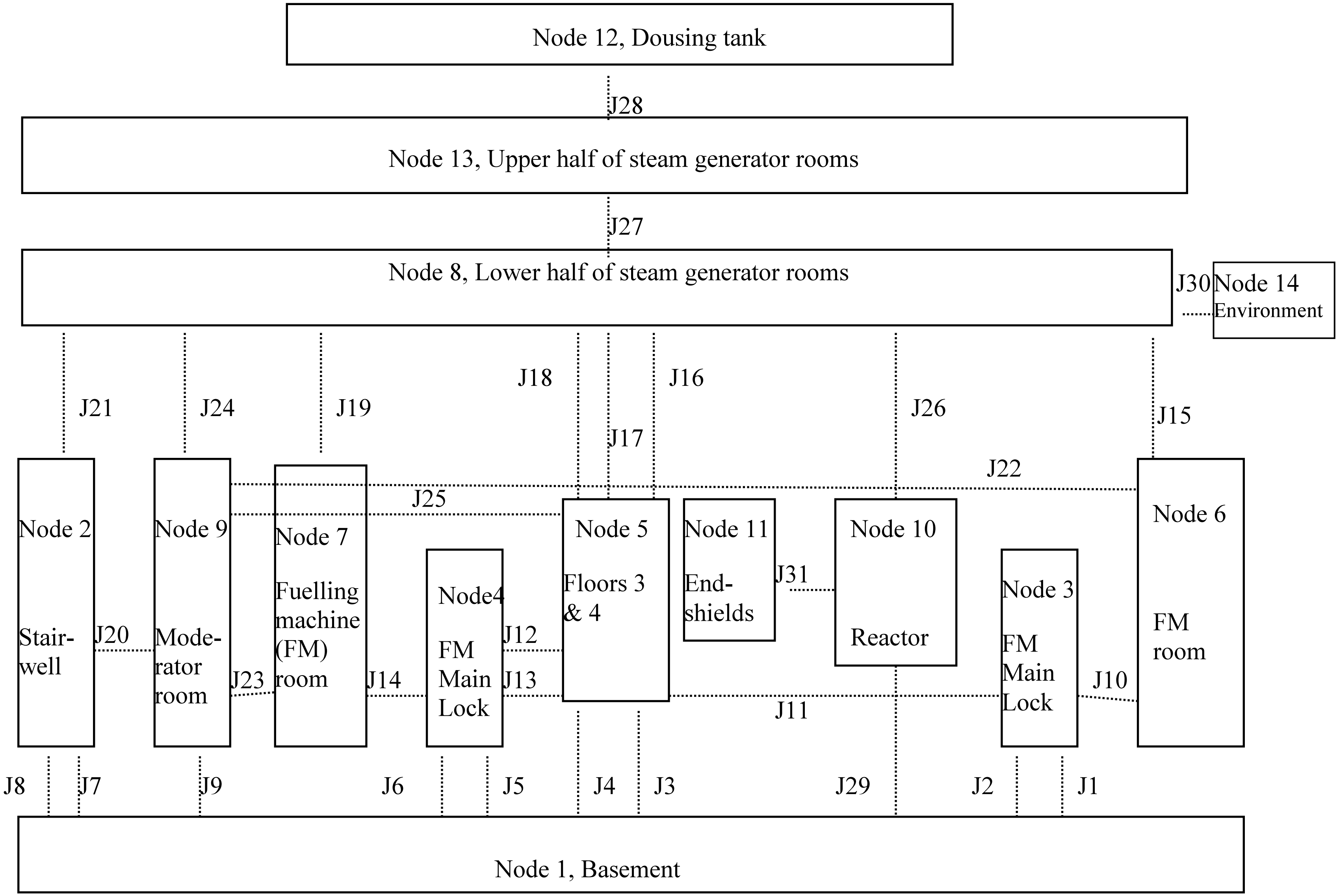
OH: outlet header, IH: inlet header, and 1~14: node number

FIG. II.4. Nodalization scheme for CANDU 6 primary heat transport system



1 ~ 15: node number

FIG. II.5. Nodalization for CANDU 6 calandria vessel wall.



J1 ~ J30: junction number

FIG. II.6. Containment nodalization for CANDU 6.

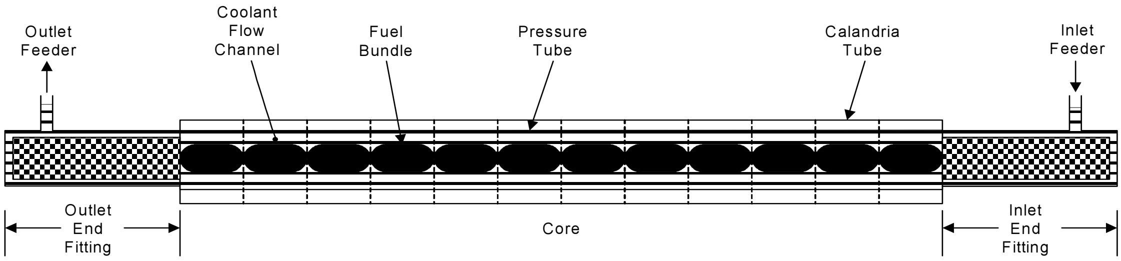


FIG. II.7. Axial core configuration modelled in ISAAC.

TABLE II.1. EVENTS OF LOW FREQUENCY

Event 1	Decrease in PHT System Inventory
1.1	Small or large LOCA Coupled with any one of the following: 1. Failure of ECCS (in injection or recirculation mode) 2. Failure of steam generator auto-crash cooling
1.2	Failure of tube(s) in PHTS heavy water heat exchanger other than steam generator coupled with any one of the following:- 1. Failure of ECCS 2. Failure of steam generator auto-crash cooling actuation 3. Failure to close the isolation devices on the pipes carrying process water to and from the heat exchangers
Event 2	Others
2.1	Station blackout (Simultaneous failure of Class III and Class IV electrical power supply) for specified duration
2.2	Safe shutdown earthquake simultaneous with LOCA: This is to be considered only for the purpose of design of those equipment/system/structures whose failure could impair integrity of containment
2.3	Fuel handling failure coupled with containment impairment

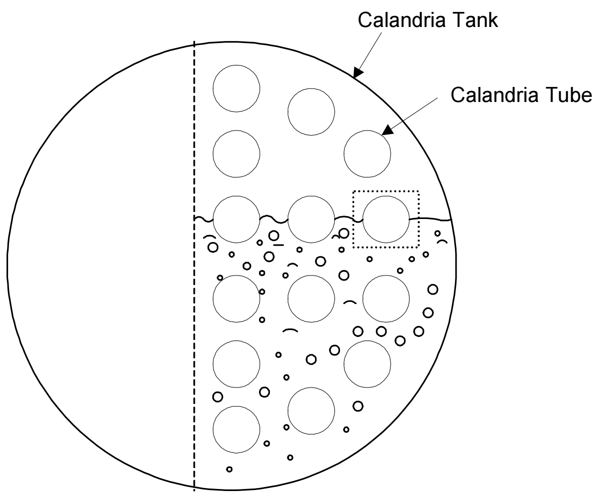
II.2. ISAAC code

The ISAAC (Integrated *S*evere *A*ccident *A*nalysis code for *C*ANDU plants) computer code [39] has been developed to simulate accident scenarios that could lead to a damaged core and eventually to containment failure at the Wolsong NPPs of the Republic of Korea. The MAAP4 computer code [46], which was developed by EPRI for pressurized water reactors, was used as a reference code. As the Wolsong NPPs, which are CANDU 6 type reactors, differ from typical PWRs, the Wolsong-specific features are newly modelled and added to the ISAAC code. The code was used to assist in quantifying the containment event tree and estimating source terms by analyzing the accident progression beyond core damage to the containment failure. The ISAAC code is constructed in modules covering for individual regions in the plant: primary heat transport system, pressurizer, steam generators, calandria, calandria vault, end-shields, degasser condenser tank, and the containment. Every major engineered safeguard features are represented in the code: shutdown cooling system, emergency core cooling system, moderator and shield cooling system, reactor building local air coolers, igniters, and containment dousing spray system. Phenomena modelled in the code are: thermal-hydraulics, core heatup and pressure tube rupture, relocation of damaged fuel to the bottom of calandria, debris behaviour in the calandria including debris quenching, corium/concrete interaction causing calandria vault floor melt-through, hydrogen burn, and fission product transport.

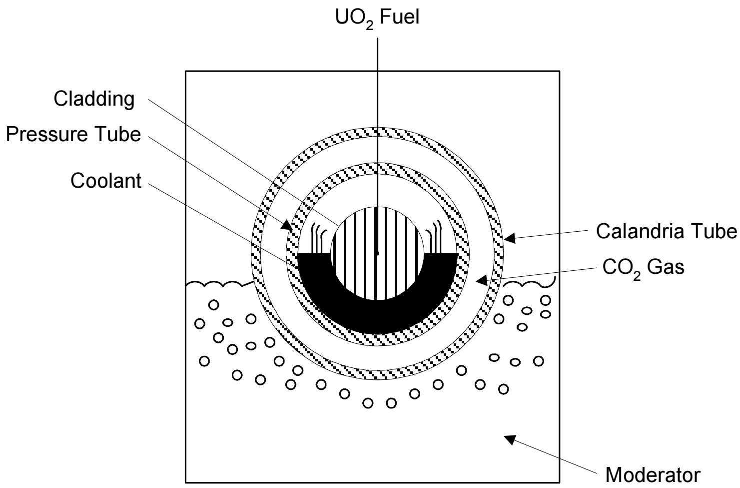
As ISAAC was derived from MAAP4 for PWRs, it adopts most of the MAAP models for severe accident phenomena in general. Wolsong-specific models for the horizontal core, figure-of-eight primary heat transport system, calandria, and safety systems, are briefly described here. Figures II.7 and II.8 show the axial and radial core configuration in the code. Each fuel channel can be nodalized into up to 12 axial horizontal nodes. Each node contains a representative fuel rod with the cladding (fuel sheath), the pressure tube, the calandria tube, CO_2 gas between pressure tube and calandria tube and the coolant inside the pressure tube. Heat transfer between the components including the coolant and the moderator is modelled in the code. The code calculates the temperature of each component depending on the boundary conditions.

The fuel cladding may balloon due to pressurization as a result of the heating up the internal gases and fission products. When the fuel rod internal pressure exceeds the primary system pressure, the cladding fails based on both cladding temperature and oxidation thickness criteria. A Larson-Miller relationship is used to correlate the rupture or sagging time with cladding temperature and oxidation layer thickness [46]. When the cladding in the core node is determined to be ruptured, the fuel pellets are assumed to accumulate on the bottom of the pressure tube, resulting in a heat transfer area reduction between the coolant and the core material.

During the high pressure accident sequences, the pressure tube may balloon and contact the calandria tube causing a direct contact heat transfer. This will significantly increase the heat transfer rate from the fuel to the moderator and increase the calandria tube temperature. The increase in the calandria tube temperature will weaken the calandria tube and lead to the sagging of the tube causing rupture of the pressure tube due to the lack of support of the calandria tube. For a low pressure sequence, the pressure tube may not balloon, but nonetheless may still rupture or melt through due to high temperature. The code simulates the impact of calandria tube sagging by disabling heat transfer to surrounding coolant and moderator. This accelerates the temperature escalation and eventually leads to the relocation of the core material to the bottom of the calandria.



(a) Example of a 14-representative-channel configuration in Loop 1
(the maximum can be 37 channels)



(b) Radial configuration of a representative channel

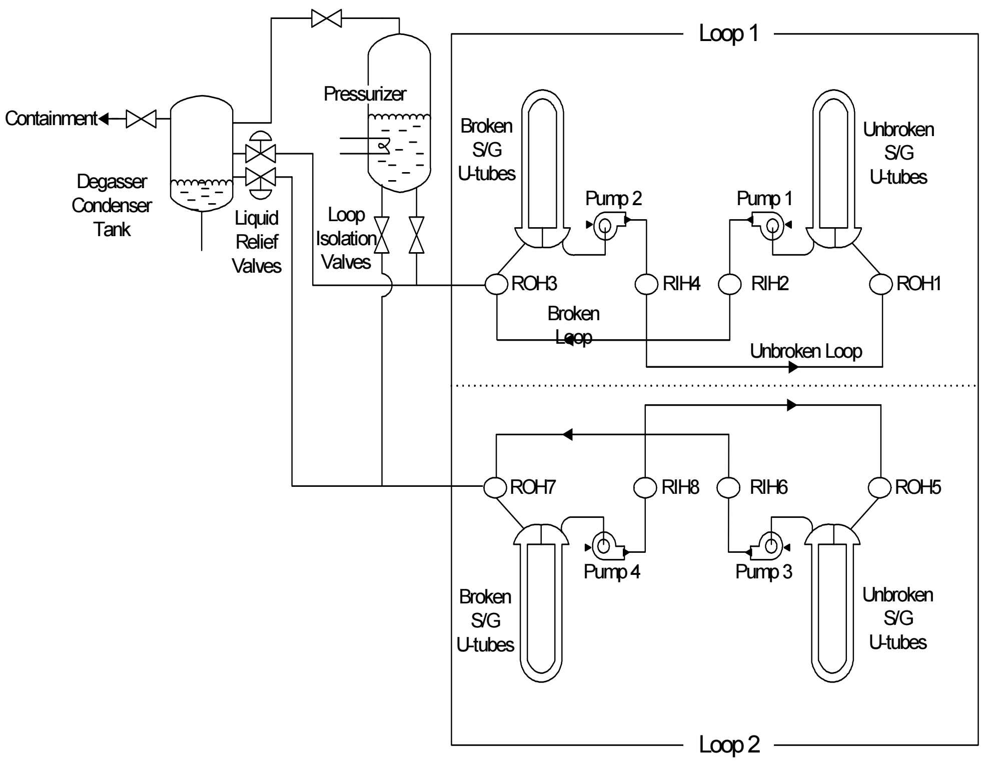
FIG. II.8. Radial core configuration model in ISAAC.

The PHTS has two independent figure-of-eight loops. Figure II.9 shows a schematic diagram of ISAAC modelling of Wolsong PHTS. Both loops and all four steam generators/pumps are modelled individually. The broken steam generator is defined to have a broken U-tube in that steam generator and the broken loop has a pipe break along the loop. When the primary system pressure boundary is intact at the beginning, the broken steam generator loop is still defined in the code, but the code does not distinguish the broken steam generator loop from the rest. The arrow in the figure shows the direction of coolant in the figure-of-eight flow configuration. The code allows the user to group 380 fuel channels into up to 74 core channels based on their elevations, power levels, core passes and loops.

The coolant in the primary heat transport system receives heat from the fuel and loses heat through the pressure tubes (transferred to the calandria), through the steam generator U-tubes (transferred to the steam generator shell side) and to the containment atmosphere through the PHTS wall. For the purpose of tracking the temperature of these heat sinks, the primary heat transport system is nodalized into up to eighty-eight nodes. The primary heat transport system modelled in the code is shown in Fig. II.10 for 3 × 3 core passes per loop.

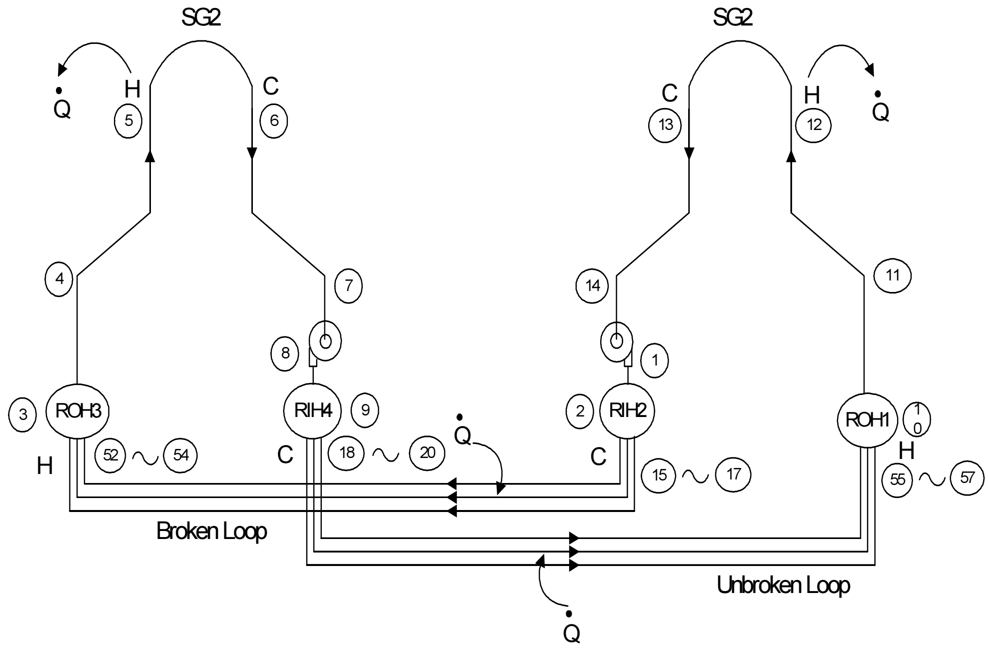
The calandria is modelled as a separate region in the code. Besides usual thermal-hydraulics and debris quenching, the behaviour of debris bed on the bottom of the calandria is modelled. The calandria vault and two end-shields are modelled in the code. Corium quenching after debris is expelled from the calandria to the calandria vault and erosion of the calandria vault concrete floor are both modelled.

In addition, the following Wolsong-specific safety features are also modelled in the code: three stages of ECCS, containment dousing spray system, cooling system for moderator and calandria vault shield water, shutdown cooling system, igniters, and local air coolers. The liquid relief valves, pressurizer relief valves, and degasser condenser tank relief valves are also modelled for PHTS to protect from over-pressurization.



ROH: reactor outlet header; RIH: reactor inlet header; and S/G: steam generator.

FIG. II.9. Schematic diagram of ISAAC modelling for Wolsong primary heat transport system.



ROH: reactor outlet header; RIH: reactor inlet header; SG: steam generator;
H: hot; C: cold, Q: heat flux, and 1 ~ 57: node number

FIG. II.10. Schematic diagram of heats sinks numbering scheme in PHTS of 3 × 3 core passes.

II.3. Severe accident code used in India

II.3.1. Introduction

A multi step approach has been adopted to analyze the core damage for PHWR. In the multi step approach, some of the computed data in one step is used as a boundary condition to the next step. In each step a specific phenomena is addressed. This procedure helps to reduce the large computational time required for a very large code system, reduce numerical problems and option for selecting specific step as per the requirement. Various steps and the corresponding codes used are described below. Recently the codes RELAP5/SCADAP and ASTEC are also being used for severe accident analysis. ANSYS is used to calculate the channel failure

II.3.2. HWR specific events

The events of low frequency and multiple failure events that may led to severe accident for PHWR type reactors are summarized in Table II.1 and Table II.2, respectively.

TABLE II.2. MULTIPLE FAILURE EVENTS (BEYOND DESIGN BASIS EVENTS)

Event No.	Events
BDBE-1	LOCA plus failure of both the reactor shutdown systems
BDBE-2	LOCA plus failure of ECCS followed by loss of moderator heat sink
BDBE-3	Failure of coolant channel seal plug or end fitting leading to ejection of fuel bundle from coolant channel coupled with containment impairment logic

II.3.3. PHWR specific phenomena

As the core damage progression models are different for PHWR as compared to PWRs and BWRs, a different methodology is being adopted for the assessment calculation. The methodology involves multi step calculation to simulate all the probable phenomena that come during an accident condition. The phenomena which are currently modeled in severe accident analysis includes the following aspects during an accident conditions:

- Single phase and two phase thermal-hydraulics in the PHTS and containment thermal-hydraulics;
- Reactor header flow stratification and vapour pull through and reactor;
- Channel flow stratification;
- Radiation heat transfer among the fuel pins, pins to the pressure tube and also from pressure tube to the calandria tube;
- Pressure tube deformation by sagging or by symmetric/asymmetric ballooning;
- Calandria Tube outer surface boiling heat transfer;
- Fuel element heat up and metallurgical deformations and release of fission gas from fuel matrix;
- Ballooning of the fuel pin and its failure;
- Bundle behaviour during asymmetric fuel pin heating;
- Steam-zircaloy-UO_2 reaction and formation of the eutectic of U-Zr alloy and hydrogen generation;
- Transportation of the radioactive material in the PHTS and the release into the containment;
- Ex-channel molten fuel-coolant (moderator) interaction;
- Debris bed–molten pool behaviour.

II.3.4. Analysis methodology

Following are the brief discussion on the computer codes used to assess severe accident scenario for PHWR.

II.3.4.1. System thermal-hydraulic behaviour

The thermal-hydraulic analysis of the plant is carried out with codes like RELAP5/MOD3.2, ATMIKA (NPCIL, India), etc. The plant model includes simulation of PHTS, containment, calandria vessel and calandria vault. Thermal-hydraulic phenomena related to initial phase of the accident namely blowdown and core uncovery can be addressed with these codes. Phenomena like header stratification and vapour pull through in the reactor channel is calculated with code BFQ. Vapour pull through is of significance as it changes the thermal-hydraulic condition in the channel. During the course of blowdown some of the channels experience flow stratification due to low flow and high voided condition. An asymmetric heating is expected as the fuel bundle experiences different heat transfer environment. For a detailed flow stratification calculation code HFLOWR has been used. The code uses horizontal flow regime map along with incorporation of wall friction of the fuel pins. This code is well validated and used for 19 and 37 pin fuel bundle. Calandria vessel and calandria vault specific RELAP5 model is used to estimate the moderator boil-off period and heat transfer from moderator to calandria vault

II.3.4.2. Channel behaviour

Under severe core damage condition for PHWR, the slumping of the fuel bundle is an important phenomenon, which leads to high heat transfer from the fuel pins to the pressure tube as a result high contact conductance. To achieve the above objective a thermal analysis is carried out. The effect of metal-water reaction has been considered in all cases. The modelling of heat transfer from pressure tube to Calandria tube is considered for two scenarios.

- Model-1: The pressure tube is not in contact with the calandria tube
- Model-2: The pressure tube is in contact with calandria tube at a local hot spot region (where the pressure tubes temperature has exceeded the limit required for sagging). The contact resistance between pressure tube and calandria was taken as zero.

For Model-1, the heat rejection to the calandria tube is considered by natural convection, conduction and radiation modes through the CO_2 annulus gap. For Model-2, the top 300^O heat rejection is by above mode whereas for the bottom 60^O, the transfer is through conduction to the calandria tube. Heat transfer from calandria tube to moderator is by natural convection. Wherever the calandria tube surface temperature exceeds moderator temperature, local pool boiling is found adequate for the temperature range. The importance of Pressure Tube ballooning and its contact with calandria Tube is of significance in PHWR severe accident. The thermo-mechanical creep behaviour for ballooning as well as for sagging is studied by a separate model (PTCREEP). This model estimates transient elastic-plastic strain for Pressure Tube

II.3.4.3. Fuel behaviour and hydrogen generation

Model OXYCON has been developed for the estimation of oxidation of zircaloy cladding above α/β transformation temperature. At these temperatures zircaloy-steam reaction is very rapid and oxygen picked up by the cladding diffuses rapidly inside the cladding. A partially oxidized cladding at these temperatures shows three distinct layers: an oxide layer, an oxygen stabilized alpha zirconium layer and beta zirconium region containing an increased amount of dissolved oxygen. The relative thickness of these layers and concentration of dissolved oxygen in the beta region depends on the time and temperature of oxidation. OXYCON predicts the thickness of various layers and oxygen concentration profile in the cladding. The basic assumptions made are the following,

- The diffusion coefficient of oxygen in oxide, alpha and beta phases is dependent only on temperature. The effect of oxygen concentration is negligible;

- Equilibrium concentration of oxygen exists at the phase boundaries;

- The volume expansion associated with oxidation of zircaloy is normal to the sample surface i.e. in the radial direction;

- The temperature is constant across the cladding thickness, i.e. there is no temperature gradient across the cladding thickness. The computational model has been validated by the authors with the published literature.

The model has been validated for oxygen weight gain and oxide growth using available oxidation data on PHWR cladding. Validation for oxygen distribution has also been done using published literature data. The agreement between calculation and experiment is satisfactory. OXYCON has been used to analyse the oxidation behaviour of PHWR fuel cladding at temperature above 1000 ^{o}C and to estimate time for attaining a given concentration of oxygen in the cladding.

To address the clad internal oxidation from UO_2, SFDCPA code is being used. The code estimates internal and external stoichiometric and non-stoichiometric oxidation layers along with U-Zr eutectic formation. This model is validated against out of pile experiments with different heating and cooling rates.

II.3.4.3. Fission product release from fuel and transport

Estimation of fission product release and transport in the PHTS is carried out with code PHTACT. This code is having option of two models namely CORSOR and CORSOR-M. Brief description of these two models are given below,

(a) CORSOR

The computer code CORSOR is a FORTRAN programme that calculates fractions of 23 reactor core material species released during a degraded core accident in light water reactors. It calculates aerosol and fission product escape from the core as a function of time during core damage accidents. The user supplies plant–specific information including core initial species inventories, geometric distribution of material and core power peaking factors. At each time-step, the current temperature of each node and the extent of Zr oxidation at each node are supplied as input to the code. The programme calculates the release of each species at specified time steps and combines appropriate releases to track the recommended groups in Ref. [47].

The code is based on a model developed by the ORNL staff and described in Ref. [48]. CORSOR provides release rates for eleven fission products (Cs, I, Xe, Kr, Te, Ag, Sb, Ba, Ru, Mo, Sr), two cladding components (Sr, Sn), one structural component (Fe), and the UO_2 Fuel.

With one exception, the fission product release rate coefficients in CORSOR depend only on temperature and have the mathematical form A exp (BT), where A & B are constants over specific temperature ranges and T is temperature. However, to account for the hold up of Te by unoxidized Zr in the cladding, the fractional release rate coefficient calculated on the basis of the above expression, is reduced by a factor of 40 if the nodal extent of Zr oxidation is less than 70%.

Gap release

A small fraction of the volatile fission product species resides in the fuel-cladding gap during normal reactor operation and is subject to a one-time release at 900°C. This temperature corresponds to an initial fracture of the fuel rod cladding and represents the so-called gap release. This gap release mechanism is simulated in CORSOR by releasing a particular fraction of the inventory of various species from every axial node at a given radial position as soon as the temperature of any axial node exceeds 900°C. This corresponds to the emission through the break in the fuel rod of the 'gap inventory' found along the entire length of the rod. Following this release the radial position is not subject to any further gap release.

Transient release

Two methods for calculating the transient release of all species except 'control rod materials' are available to the user of CORSOR. Both methods assume a first order release rate from each node for each species such that

FFP = FP*(1-exp (-FRC * DTIME))

where FFP is the mass of the species released from the node during time period DTIME, FP is the mass of the species present at the node at the start of the time step, and FRC is the fractional release rate coefficient.

The value of FRC used in the code calculations depends upon the method selected by the user. These two methods are described below.

Default method

For the default method, the value of FRC is species and temperature dependent, given by a relationship of the form:

FRC = A(I,J) * EXP [B(I,J) * T]

where T is the temperature in ^oC and A & B are constant where values are selected for the I^{th} species and the J^{th} temperature range. The three temperature regimes for which these constants are defined are 900–1400°C, 1400–2200°C, and 2200–2760°C, with all temperature values greater than this latter value set to 2760 °C since this has been taken to be the maximum credible temperature for any node in the core. This method of calculation of the release rates is recognized as being non-mechanistic, and no attempt has been made to account for any scaling effects to which these release coefficients may be subject in the transition from the experiments to the accident situation.

(b) CORSOR-M method

A second method of calculating the release rate coefficients is denoted by M-version. M-Version makes use of a more physical description of the release process. In this method the release rate coefficient is given by an Arrhenius type equation:

FRC = KO(I) * EXP (-Q(I)/1.987-3*T))

where KO(I) and Q(I) are species–dependent constants, T is the nodal absolute temperature, and 1.987E-3 is the value of the gas constant multiplied by a unit conversion factor. For the release of the noble gases, i.e. Te, Cs, and I, it is assumed that the release rates are controlled by the fuel matrix, and so have the same KO and Q values. The resulting release rates are almost identical to the release rates obtained from the default method for these species. The release of refractory fission products and structural materials is assumed to be controlled by vaporization, so that the Q values are heats of vaporization for these released species. Ba, Sr, La, and fission product Zr are assumed to be released in oxide form and so the heats of vaporization of these oxides have been used for Q. The KO values are determined by adjusting the curve to the existing data.

It is recognized that the releases predicted in this way still do not have a mechanistic basis, but this approach does have the advantage of incorporating the available data into a simple framework that has some foundation in physical phenomena.

Control rod release

For the calculation of the control rod release in the code, three different empirical correlations/constants are used on the basis of three different temperature zones

Fission product species mass continuity is solved in conjunction with mass flow rates and mass inventory calculated by thermal-hydraulic code at different axial nodes over the PHTS to estimate evolution of fission product in PHTS as well as in containment.

II.3.4.4. Molten fuel coolant interaction

To address this phenomena the code MFCI will be used. The code solves mass, momentum and energy continuity for steam water, large and fine molten droplets with interfacial closure laws to compute the energy released to moderator. With the generated pressure pulse the neighboring channel and control rods integrity will be assessed Empirical molten jet fragmentation models involving thermal-hydraulic fragmentation are incorporated in this code. This code is under numerical and experimental validation programme.

II.3.4.5. Debris bed molten pool behaviours

With the core uncovery the reactor channel are expected to be heated up and may collapse in a cascading way. This may lead to accumulation of debris bed at the calandria vessel bottom and heatup calandria vessel with time. Calandria Vessel may fail due to thermal creep or material ablation. Code MPOOL has been developed with finite volume approach to calculate heat transfer in the molten pool. The code is under experimental validation programme.

Codes used in at various stages of calculation to address different severe accident phenomena are given below. Codes related to Molten Fuel Coolant Interaction and Debris Bed-Molten Pool is not listed as they are under validation programme.

- Step 1 : Global Simulation: A global simulation is carried out using a thermal-hydraulic code such as ATMIKA or RELAP5/Mod 3.2;

- Step 2: Header Stratification Level and Vapour Pull through Simulation The header stratification and the vapour pull through quality has been calculated with ‘BFQ’ code;

- Step3: With these channel boundary conditions in the header (Flow and quality and pressure) the slave channel analysis is carried out;

- Step 4: Channel Flow Stratification — A high quality and low flow leads to flow stratification in the channel. Code HFLOWR predicts flow stratification in channels with internals;

- Step 5: Detailed Thermal Analysis of the channel is carried out with HT/MOD4;

- Step 6: Pressure tube Deformation: The pressure tube deformation computed with PTCREEP model works in an interactive mode with HT/MOD4;

- Step 7: Steam-Clad interaction with OXYCON and Steam-Clad-Fuel Interaction with SFDCPA models metallurgical interactions within fuel rods;

- Step 8: Activity Release — PHTACT calculates the fission product release with CORSOR model and transport of activity in PHT System and release to the containment;

An iterative approach between SCADAP and ANSYS has been adopted so that these internationally available codes can be used for PHWR severe accident analysis. The code ASTEC is being used for ex-vessel analysis.

Appendix III

SEVERE ACCIDENT PROGRESSION IN A CANDU REACTOR

Design basis accidents can progress into a severe accident if critical barriers fail. Once the accident has progressed to a severe accident, there is a lot of commonality in how the accident progresses, to the extent that a small number of generic ‘core damage states (CDSs)’ can be defined as a convenient way to represent severe accident progression. A CDS is simply a quasi-steady state during which the decay heat is absorbed into its surrounding environment. These core damage states are independent of the initiating event and generally independent of station design (there are a few specific differences unimportant to this discussion). However, the timing of progression from one state to the next can be affected by the initiating event, the detailed design and operator actions, that latter of which are aimed at successfully terminating the progression.

There are five such typical CDSs for PHWRs as follows and the related parameters on each CDS are described in Table III.1:

CDS1: The fuel channels have lost water inventory, dried out and heated up. The fuel sheath is oxidized and the pressure tubes have ballooned into contact with the calandria tubes. The moderator removes most of the decay heat. This is the terminal state of a LOCA plus loss of ECCS. It is sustainable as long as the moderator level can be maintained.

CDS2: The moderator level has dropped exposing several upper channels (due to moderator rupture disk bursting due to boiling or in-core LOCA). The exposed channels have heated up, sagged, oxidized and broken apart collapsing onto lower submerged channels or dropping to the bottom of the calandria vessel. Most of the decay heat is removed from submerged channels as well as some of the decay heat of the collapsed fuel channels that are now submerged. Adding water to the calandria vessel can prolong this state.

CDS3: The moderator inventory is gone (boiled off slowly or drained quickly due to type and location of break). All channels have heated up, sagged, oxidized and broken apart leaving a rubble pile of ‘corium’ (mix of fuel and core structural materials) at the bottom of the calandria vessel. The steel calandria vessel and surrounding biological shielding materials (water and steel in the shield tank, or concrete depending on design) remove some of the decay heat. The structure is not capable of removing all decay heat and the corium will eventually melt through; however, adding water to the calandria vessel can prolong this state.

CDS4: Corium has penetrated through the calandria vessel and biological shield and is now on the concrete floor. Accumulated water will quench the molten corium.

CDS5: Due to lack of water or insufficient contact area for boiling, or due to formation of an upper crust, the corium attacks the concrete referred to as molten core concrete interaction. Ablation of concrete produces steam, H_2, CO and CO_2. The degree to which the molten core concrete interaction can be terminated depends on the decay heat (which diminishes with time), the surface area of the melt (affects rate of cooling by a water layer, limited by the critical heat flux) and the availability of water. Since the rate of ablation is slow (about 2 cm/hour with decay power at 1%) and the basemat is thick (>1m), and decay power diminishes with time, basemat penetration is unlikely, and certainly not expected within a few days.

TABLE III.1. THE RELATED PARAMETERS ON EACH CDS

Core damage state	Parameters
CDS 1	• Moderator level • Moderator outlet temperature
CDS 2	• Moderator level • Activity monitoring • Monitoring of calandria vault water temperature
CDS 3	• Calandria vault temperature monitoring • Activity monitoring
CDS 4	• Area monitoring • Containment pressure
CDS 5	• Area monitoring • Containment pressure (Generation of gasses by molten core concrete interaction increase the pressure of the containment)

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ABBREVIATIONS

AECL	Atomic Energy of Canada Limited
BWR	Boiling water reactor
CANDU	Canadian deuterium uranium
CDS	Core damage state
DFC	Diagnostic flow chart
ECCS	Emergency core cooling system
FRC	Factional release rate coefficient
ISAAC	Integrated severe accident analysis code for CANDU
LAVA	Lower-plenum arrested vessel attack
LCDA	Limited core damage accident
LOCA	Loss of coolant accident
LWR	Light water reactor
MAAP4	Modular accident analysis program for CANDU
OECD	Organisation for Economic Co-operation and Development
PHWR	Pressurized heavy water reactor
PHTS	Primary heat transport system
PSA	Probabilistic safety assessment
PWR	Pressurized water reactor
RBMK	High-power boiling reactor with pressurized channels (Russian design)
SAG	Severe accident guide
SAMG	Severe accident management guideline
SCDA	Severe core damage accident

SCG	Severe challenge guide
SCST	Severe challenge status tree
TROI	Test for real corium interaction with water
WWER	Water cooled, water moderated power reactor (Russian Design)

Annex I

EXAMPLES OF SEVERE ACCIDENT ANALYSIS RESULTS

I-1. Examples of severe accident analysis results for CANDU 6 plants

Sample analysis results of severe accidents in CANDU 6 plants are described in this section. There are two main sources of the assessment results that have been included and summarized in this report:

1) Earlier level 2 PSA results published in 1988 (also know as the KEMA study or TTR-221) based on a suite of computer codes that contains both CANDU specific computer programmes and those derived from the US NRC source term code package;

2) Later severe accident analyses using the MAAP4-CANDU computer code in support of the generic CANDU PSA programme initiated in 1998. Severe accident analysis work using MAAP4-CANDU is ongoing.

The above two sources of severe accident analysis results provides for a good overview understanding of severe accidents in the CANDU reactors deriving from various computer codes.

There are a number of other sources of CANDU severe accident analysis results in Refs [I.1–I.3]

I-1.1. KEMA Study or TTR-221 results

I-1.1.1. Backgrounds

In September 1986, a study was undertaken to determine the frequencies and consequences for severe accidents in a CANDU reactor. This initial study was done cooperatively between AECL and KEMA (N-V. Tot Keuring van Elekrouxhnische Materialen Arnhem, the Netherlands). The goal was to produce source term data for a CANDU 6 reactor which could be compared to the results of light water reactor probabilistic risk assessments. Since this initial study was completed in early 1987, AECL continued with a more detailed study of the consequences of severe accidents for CANDU. The combined results of these studies are documented in Ref. [I–4].

Analytic tools and the supporting R&D have been upgraded since the original issue of the document. In particular, improved models of heat transfer within the calandria and the timing and mechanism of channel failure during the late core assembly have been developed. Modifications have been made to the reactors as well so that some detailed results may not apply. None of these improvements are reflected in the Ref. [I-4]. Nevertheless, it still has significant generic value in that it addresses key phenomena which are involved in CANDU severe accident analysis.

I-1.1.2. Computer codes

The computer codes used in this analysis form the basis of a source term code package for CANDU (Fig. I-1). The computer programmes which are derived from the U.S. NRC source term code package are: ORIGEN (fission product inventories), CORSOR-C (the equations from CORSOR for transient fission product releases from fuel were used), NAUA (fission product transport and deposition in containment), and CORCON (the core-concrete interaction computer programme CORCON, was not used directly in the analysis, rather information from previous LWR analyses was used).

CANDU-specific computer programmes are those directly related to describing the core melt progression, i.e. those codes which are more core geometry dependent. These involved: first, the computer programmes developed at Carleton University, i.e. MODBOIL (moderator water discharge and steaming, and channel uncovery) and DEBRIS (fuel and fuel channel debris temperatures), second, the AECL computer programmes FIREBIRD (heat transport system thermal-hydraulics), MODSTBOIL (moderator and calandria vault water discharge and steaming and channel uncovery), CHAN (fuel and fuel element temperatures for degraded cooling conditions) and PRESCON (containment thermal-hydraulics).

Note that the plateout and washout of fission products in the calandria vessel and heat transport system were not modeled before release to the containment environment. This would result in lower releases from the core region to the containment atmosphere.

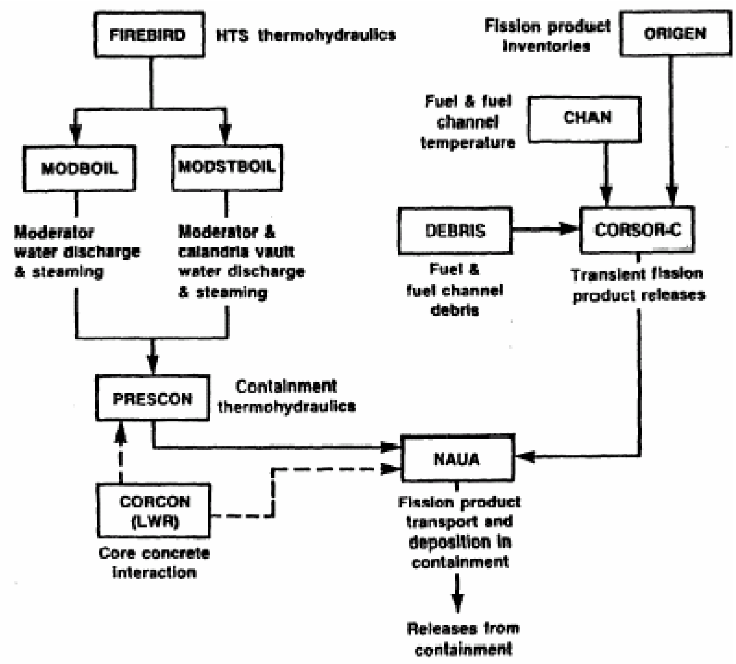


FIG. I-1. Computer code interaction.

I-1.1.3. Accident scenario and results

A best estimate evaluation of a representative event sequence was performed. The event is a loss of service water (i.e. loss of cooling to the moderator, calandria vault, ECCS heat exchanger, etc.), combined with a loss of class IV power as a consequence of the reactor trip, and a loss of steam generator heat sink. The basis for choosing this representative event is that it is consistent with a dominant frequency event sequence in this event category (late core disassembly), and it is expected to result in containment release predictions which will also represent other events in the category. A qualitative description of the event sequence is as follows. Table I-1 provides a summary of the event sequence.

TABLE I-1. SUMMARY OF LATE CORE DAMAGE EVENT SEQUENCE

Approximate Time (hour)	Event
0	Reactor trip (after loss of power and feedwater)
0 to 0.75	Steam generator boil-off
0.75 to 0.83	Primary heat transport system relief to degasser condenser (10.34 MPa) and containment
0.83 to 0.85	A few channels fail (~3 in each loop assumed), allowing primary heat transport system blowdown to moderator, moderator overflow to containment, and initiation of high pressure ECCS
0.85 to 1.0	High pressure ECCS refills primary heat transport system; medium and low pressure emergency coolant injection assumed unavailable
1.0 to 1.1	Both primary heat transport system loops drain; decay and zircaloy/steam reaction heat transferred to calandria liquid
1.1 to1.7	Calandria liquid heatup
1.7 to 5.1	Calandria liquid boil-off
5.1 to7.2	Core debris heatup
1.1 to 8.4	Shield water heatup
8.4 to 26	Shield water boil-off
26	Core material relocates to concrete shield tank
26 to 60	Core-concrete interaction
60 to end	Debris in basement water

Initially, it is postulated that feedwater flow to the steam generators is lost and shortly thereafter a reactor trip (shutdown system No. l) occurs. Previous analysis indicated that without operator action there is about 3/4 of an hour (2700 seconds) available before the steam generators boil dry at decay power.

After 2700 seconds, the steam generators no longer act as a heat sink. Therefore, the primary circuit temperature and pressure would rise. The pressure would rise to the liquid relief valve setpoint of 10.34 MPa(a). One or more liquid relief valves would open and discharge primary coolant into the degasser condenser. The degasser condenser fills and then relieves either via the degasser-condenser relief valves or from the heavy water storage tank to the containment. The discharge would remove decay power from the primary circuit, and arrest the pressure rise, but the flow would become two phase as the inventory decreased. Steam could also start to penetrate into the reactor inlet header and into the inlet of the fuel channels. Thus, at about one hour the primary circuit is intact, at high pressure, and has a small leak. Because of the loss of service water, the moderator has lost its cooling and slowly warms up to about 100°C due to the direct gamma heating at decay power.

The combination of flow reduction and steam at the inlet of some channels leads to degraded fuel cooling. As the fuel temperature rises, radiative heat transfer to the pressure tubes would also cause their temperature to rise. Since the primary circuit pressure is at the liquid relief valve setpoint (10.34 MPa(a)), the pressure tubes would strain significantly. A pressure tube could strain in a balloon-like fashion, so that it contacts the surrounding calandria tube around the entire circumference. Heat would be transferred from the pressure tube through the calandria tube to the moderator. If the calandria/moderator surface does not dry out, then the pressure tube would be supported by the calandria tube and the moderator would function as the heatsink. However, at full system pressure it is considered that a small number of pressure tubes could fail (due to non-uniform pressure tube strain) prior to contact with the Calandria tube and the calandria tube could also fail. The time between the steam generators boiling dry and some pressure tubes failing is estimated to be 300 seconds. Only a few pressure tubes are likely to fail, since the primary circuit depressurizes quickly after the first few failures.

The primary circuit coolant then discharges into the moderator. Fuel in the failed channels would be cooled by the discharge flow and/or by the cool moderator water. The primary coolant would mix to some degree with the moderator, and the moderator pressure would rise enough to discharge through the relief ducts and into containment. It is estimated that, after about 80 seconds of primary circuit blowdown, the ECCS would be initiated on signals of low primary circuit pressure and high moderator level.

The ECCS would refill the primary circuit and calandria vessel in a few minutes. The discharge into containment during this time removes the stored energy in the primary coolant, piping and fuel.

By 3750 seconds, the high pressure stage of ECCS (from accumulators) is estimated to be depleted. The medium (pumps) and low pressure stages are postulated to be unavailable. Therefore, the primary circuit would drain until about 4000 seconds when it would be empty.

After 4000 seconds, the fuel end pressure tubes of those channels that are intact at this time would begin to heat-up again. The previously undeformed pressure tubes would strain or sag into contact with the calandria tubes, depending on the primary circuit pressure. Heat would be transferred to the mixture of light water and heavy water that would be in the calandria at this time. During this time the fuel sheaths would heat-up enough to be partially oxidized with any steam present in the channel. Part of the free fission product inventory in the fuel gap is released. Fuel temperatures do not reach the melting point, however temperatures are high enough that additional inventory beyond the gap inventory is released.

The moderator cooling system is postulated to be unavailable for long term heat removal, and therefore the inventory would boil-off. Initially the water heats up and about 30% of the water is expelled due to swell and carryover. The decay power is relatively low at this time, so it takes a number of hours to boil-off the inventory. The channels will sag and continue heating up. The calandria tubes which are uncovered weaken and pieces of the pressure tube, calandria tube and fuel sheaths drop to the bottom of the calandria vessel. Pressure tube and calandria tube temperatures are expected to remain below 1000°C, so oxidation of zircaloy is limited.

The progressive boil-off of the moderator allows more channels to disintegrate. The debris is very porous but as the moderator boils away, the channel components melt and zirconium (principally) penetrates to the calandria shell and refreezes. The debris bed is cooled by the remaining moderator water, and heat is transferred to the calandria vault water surrounding the calandria vessel. Moderator water (about 260 Mg) is boiled off at about 5 hours. It is postulated that the calandria vault water cooling system is also not available for long term cooling, so that water also boils off. However, at this time the decay power is very low, so that the vault water boils off slowly over a period of hours.

Eventually (after about 25 hours), the calandria vault water (about 500 Mg) has boiled away to a level near the top of the debris bed in the calandria and the calandria vessel begins to overheat. The mode of vessel failure is likely to be localized at the points where temperatures are high. The mode of vessel failure is then overheating and tensile failure of the shell. The overheated area is probably along a line somewhat above the waterline. Should this line of shell material fail, it would release material into the bottom of the calandria vault.

The released contents would enter the remaining water at the bottom of the pool to rest on the thick 2.4 m (8 feet) concrete calandria vault floor. At this time, a series of small steam explosions are possible, but unlikely due to the relatively slow process involved. Hydrogen may be produced in significant quantities. A molten core-concrete interaction is likely for the core material in direct contact with the concrete. This process produces a significant amount of gases, such as CO_2, CO, etc., and in addition some heavier nuclides from the molten core debris can be released. The concrete floor is so thick that a period of days would be needed to penetrate it. At that point some material could penetrate into the water-filled basement of containment and would be quenched. The water in the basement is expected to provide a long term heat sink.

The pressure transient in the containment atmosphere is shown in Fig. I-2 which reflects all the major events listed in Table I-1. The pressure response briefly exceeds the cracking pressure threshold of 330 kPa(g).

A key feature of the CANDU 6 containment structure is that it is a pre-stressed concrete building. Experiments at the University of Alberta in the 1980s have demonstrated that at 330 kPa(g) internal pressure (2.3 times the proof test pressure), cracks would penetrate through the wall. Leakage through the cracks is negligible at pressures below 345 kPa(g), and increases exponentially as the pressure is increased beyond that. At pressures approaching but still below the predicted failure load of around 530 kPa(g), the experiments suggest a leakage rate sufficiently high that the internal pressure is relieved; so it is difficult to have a condition in which the containment fails due to internal pressure loading. This has a significant advantage — the structure would be unlikely to fail in a catastrophic way, and hence fission products would be largely retained inside the containment structure. The 'wet' atmosphere therein will immobilize them further.

The predicted source transient and the corresponding concentration of hydrogen in the containment atmosphere are plotted in Fig. I-3. There are three distinct periods of significant hydrogen generation. The first occurs between 1 and 3 hours, when the core is still largely intact, and all the fuel still resides within the channels. The zircaloy/steam reaction is fed by the remaining water in the primary cooling circuit. The second period is associated with molten core relocation from the calandria vessel at 25.8 hour and is quite brief. The reaction at this time is fed by the remaining shield water. The third and final period begins shortly after core relocation as the molten debris begin to ablate the concrete of the calandria vault.

Little hydrogen is generated between the first and second periods, i.e. between 3 and 25.8 hours. The reactions are as follows. Channel sag and failure occur at temperatures that are too low to initiate the zircaloy/steam reaction. The debris falls into the remaining water and remains submerged and cool until most of the water has boiled off at 5 hours. A minor amount of hydrogen is produced only near the end of this boil-off period. From 5 to 25.8 hour, there is no hydrogen produced because there is no steam or water available to the core debris while the calandria is intact.

The relatively low concentration of hydrogen in containment is the result of the large fraction of steam in the containment atmosphere. Ignitable concentrations are not attained for extensive period.

I-1.2. MAAP4-CANDU results

I-1.2.1. Background

To support the generic PSA programme at AECL, in particular to perform level 2 PSA of a CANDU 6 plant undergoing a postulated severe accident, the capability to conduct severe accident consequence analysis for a CANDU plant is required. For this purpose, AECL selected MAAP4-CANDU from a number of other severe accident codes. The necessary models for a generic CANDU 6 station have been implemented in the code, and the code version 4.0.4A was tested using station data, which were assembled for a generic CANDU 6 station.

A number of severe accident scenarios have been evaluated for a generic CANDU 6 plant using the MAAP4-CANDU code, including:

- Large LOCA;
- Station blackout;
- Stagnation feeder break.

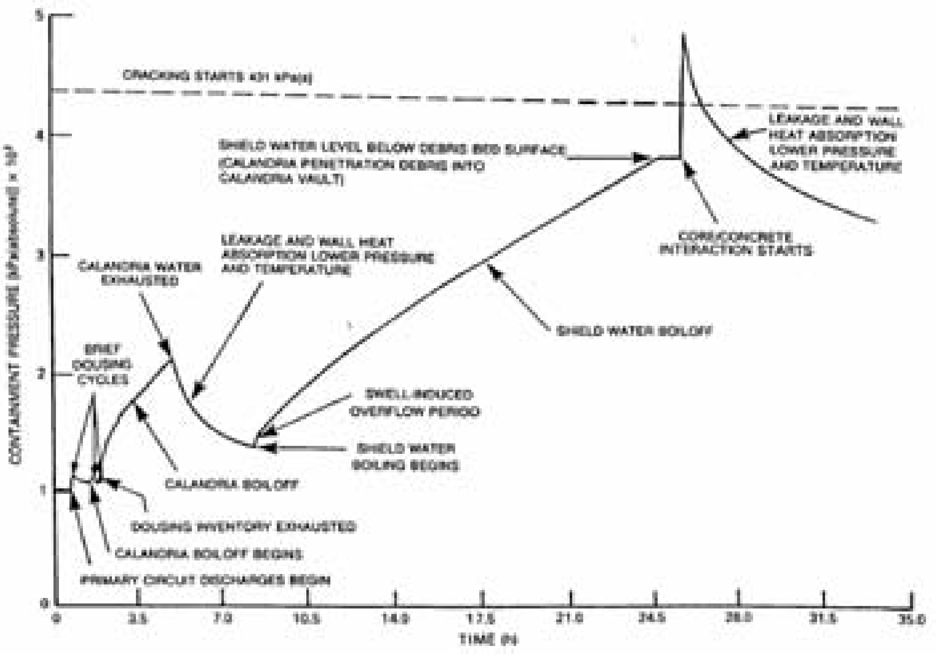


FIG. I-2. Containment pressure response.

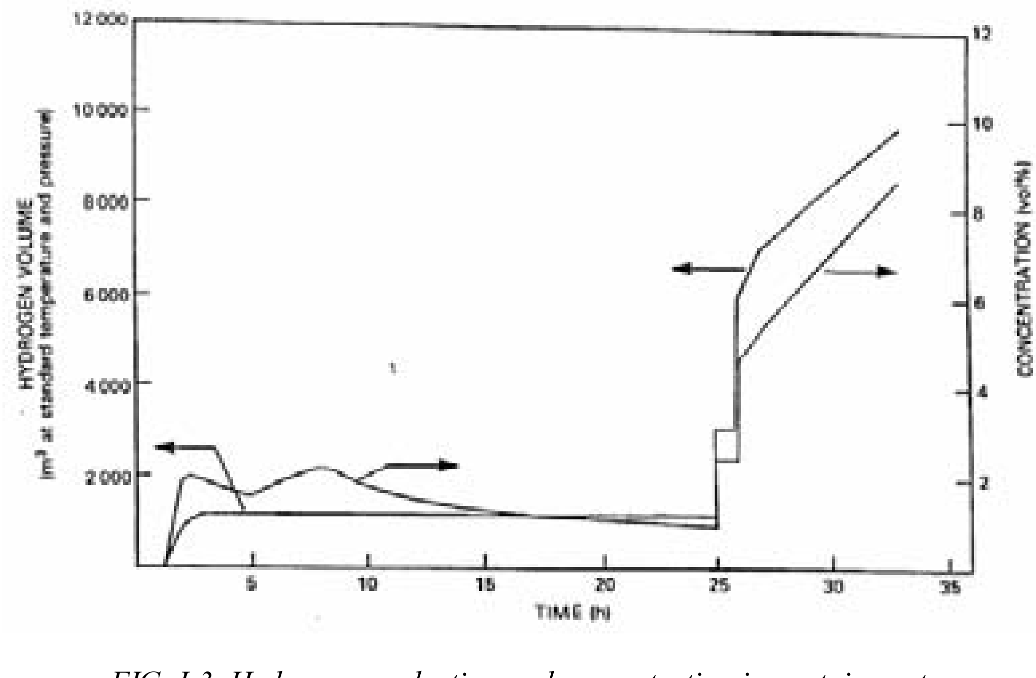


FIG. I-3. Hydrogen production and concentration in containment.

The purpose of these studies was not to produce final results for level 2 PSA, but to demonstrate the code capability for level 2 PSA applications.

For purposes of this report, summary results of a severe accident initiated by a large LOCA event are presented. The results are taken from in Ref. [I-5]

I-1.2.2. Nodalization of CANDU 6 station

MAAP4-CANDU simulates only the most significant systems, components and processes that are deemed necessary to demonstrate the overall response of the plant to a severe accident. For reference, some details of the nodalization scheme used in the present work to simulate some of the systems and components of CANDU 6 are given below.

MAAP4-CANDU has a generalized containment model, which was used to model the CANDU 6 containment. The containment was represented by 13 nodes, 31 flow junctions and 90 wall heat sinks representing horizontal and vertical containment walls.

The PHTS is represented by two symmetric loops, with the flow through each loop following a 'figure of eight' configuration, and with some channels carrying the flow inward and others outward from each reactor face. Fourteen nodes in each PHTS loop represent the following components: pump discharge lines, reactor inlet headers, reactor outlet headers, inlet piping of steam generators, hot leg tubes of steam generators, cold leg tubes of steam generators, and pump suction lines after cold leg tubes of steam generators. MAAP4-CANDU has a simple PHTS thermal-hydraulics model; the coolant pressure is the same in all nodes within the same PHTS loop.

The CANDU 6 core has 380 fuel channels arranged in 22 rows and 22 columns. A simplified core nodalization was used to represent the total number of fuel channels. The 22 rows of the core were divided into 6 vertical nodes, with 4, 4, 3, 3, 4, 4 rows in each node. Within each vertical node, the fuel channels were divided into 3 groups (low, medium and high power) of characteristic channels according to their fuel powers. The 22 columns of fuel channels were divided into 2 loops, symmetric about the vertical axis. The 12 bundles in a CANDU channel were modeled as 12 axial nodes. In a fuel channel, the calandria tube and the pressure tube are modeled as two concentric rings. The 37 fuel elements of the fuel bundle are modeled as 7 concentric rings. Thus, 9 rings represent a CANDU 6 fuel channel. The secondary side of the steam generator is represented as one node and the primary side of the steam generator contains two nodes.

The pressure and inventory control system is represented by a pressurizer joined with two PHTS loops. Each line connecting the pressurizer with the PHTS contains a motor-operated pressurizer loop isolation valve, which can be closed in case of a LOCA. All basic thermal-hydraulic processes, such as boiling and condensation, flashing and rain out, and the behaviour of fission products are modeled in pressurizer. Pressurizer heaters are also modeled. The degasser condenser is not currently modeled.

I-1.2.3. Modelling assumptions

Several assumptions were made in the current analysis. These assumptions are either embedded in the code as models with input control parameters, or they are assumed in the present analysis:

- Reactor shutdown is initiated immediately after accident initiation;
- Moderator cooling and shield cooling are unavailable;
- Shutdown cooling system is unavailable;
- Main and auxiliary feed water are unavailable;
- No containment leakage or ventilation is modelled;
- Governor and main steam isolation valves are closed after accident initiation;
- Liquid relief valves and pressurizer relief valves discharge the PHTS inventory into containment. In reality, these valves should discharge into the degasser condenser; but the degasser condenser is not currently modeled;
- Steam generator main steam safety valves are available; they open and close at the set point to relieve pressure;
- Crash cool down system is available;
- ECCS: high pressure injection and medium pressure injection are available;
- ECCS low pressure injection is unavailable;
- Containment dousing spray system is available;
- All operator interventions are not credited.

I-1.2.4. Analysis results

In the present analysis, we consider a large LOCA scenario initiated by a guillotine rupture of the reactor outlet header in loop 1, followed by a double-sided blowdown of the PHTS coolant. The break area of the reactor outlet header considered is 0.2594 m^2. Table I-2 lists the sequence of significant events observed during this simulation.

TABLE I-2. SEQUENCE OF SIGNIFICANT EVENTS FOR LARGE LOCA

Time (hour)	Time (second)	Event
0	0	Reactor outlet header guillotine rupture on loop 1
0.0015	6	ECCS high pressure injection is on
0.0016	6	Dousing system is on
0.0068	24	Pressurizer and PHTS loops are isolated
0.0094	34	Steam generator main steam safety valves are open, crash cooldown system is on
0.027	98	ECCS high pressure injection is terminated
0.027	98	ECCS medium pressure injection is on
0.08	298	Dousing tank water is depleted for containment sprays
0.31	1100	ECCS medium pressure injection is off
0.60	2614	Steam generator is dry, loop 2
0.80	2860	Fuel bundles are uncovered inside fuel channels in loop 1
4.0	14 386	Steam generator is dry, loop 1
5.0	17 940	At least one channel is dry in loop 1
5.5	19 826	Calandria vessel water pool is saturated
6.1	22 060	Fuel bundles are uncovered inside fuel channels in loop 2
7.0	25 180	At least one channel is dry in loop 2
7.3	26 236	Calandria vessel rupture disk #1 is open
9.7	35 066	Pressure tube and calandria tube rupture, loop 2
10.1	36 381	Beginning of the core disassembly
16.8	50 781	Core collapse onto the calandria vessel bottom
18.9	68 068	Water is depleted inside calandria vessel
55.4	199 308	Calandria vessel bottom wall failed due to creep
55.4	199 328	Energetic core debris-steam interaction occurred in reactor vault
55.4	199 351	Containment failed
55.4	199 480	Corium is discharged into reactor vault
58.1	209 109	Water is depleted in reactor vault
122.0	439 019	reactor vault floor failed because of concrete erosion

I-1.2.4.1. Primary heat transport system and ECCS response

As a result of the break in the ROH in loop 1, the PHTS pressure in loop 1 decreases faster than in loop 2. When the pressure in loop 1 reaches 5.5 MPa (a), the loop isolation valves are closed at about 24 s to isolate the pressurizer from the two loops, and loop 2 from loop 1. High pressure injection into loop 1 starts when its pressure reaches 4.14 MPa (a), and terminates at about 98 s. Since the loop 2 pressure is greater than loop 1, water from high pressure injection ECCS goes mainly into loop 1. After ECCS injection terminates, the pressure in the intact loop (loop 2) increases as a result of core decay heat, and because of the unavailability of the auxiliary feed water and shutdown cooling systems. The pressure in loop 2 reaches a constant value of about 10 MPa (a) and oscillates as a result of the periodical opening and closing of the liquid relief valves. The pressure in loop 2 then drops rapidly at about 35 100 s, due to fuel channel failure in loop 2.

When the pressure difference between the medium pressure injection water source (dousing tank) and the PHTS reaches the set point of 114 kPa, medium pressure injection starts at about 98 s and water from the dousing tank is pumped into the PHTS. Medium pressure injection continues until 1100 s, when water in the dousing tank is no longer available.

Because MAAP4-CANDU uses a simple PHTS thermal-hydraulics model, very good agreement between results obtained from a detailed thermal-hydraulics code and MAAP4-CANDU cannot be expected during the short term accident progression events. The scope of the MAAP4-CANDU analysis is to provide results on the long term behaviour of a CANDU plant to severe accidents.

I-1.2.4.2. Steam generator response

The steam generator main steam safety valves are opened after receiving the LOCA signal to initiate crash cooldown at about 34 s, which decreases the pressure in the primary side of the steam generators. As a result of the blowdown through the open main steam safety valves and the boil-off of water from the secondary side of the steam generators, the water level in all four steam generators decreases. The steam generators dry out by about 2600 s in loop 2, and by about 14 400 s in loop 1. The water level in the steam generators in the broken loop (loop 1) is higher than in the unbroken loop (loop 2). Because very little high pressure injection water is injected into loop 2, the coolant in loop 2 is hotter than in loop 1, resulting in faster boil-off of water in the secondary side of the steam generators in loop 2. As a result, the water level in the secondary side of the steam generators in loop 2 decreases faster than in loop 1.

I-1.2.4.3. Fuel channel response

Table I-2 shows that the fuel bundles are uncovered inside the fuel channels at about 2900 s in loop 1 and at about 22 100 s in loop 2. The uncovering of the fuel bundles is the result of a combination of the following phenomena: (1) coolant boil-off due to decay heat from the core, (2) loss of coolant through the break, (3) loss of coolant through PHTS liquid relief valves, and (4) loss of heat sink in the steam generators due to the loss of the secondary side steam generator inventory.

As steam generators dry out by about 2600 s in loop 2, the PHTS pressure increases to the liquid relief valve set points and the PHTS coolant inventory is discharged into the containment. When the temperature of the pressure tube reaches about 900 K at the high PHTS system pressure in loop 2, one fuel channel in loop 2 ruptures at about 35 100 s.

The pressure tube, calandria tube and fuel temperatures remain constant up to about 20 000 s for the channel in loop 1 and up to about 27 000 s for loop 2, because the current code version assumes that the core decay heat goes directly into the PHTS coolant during that period. After the given fuel channel is dry, the channel module of the code is initialized and the fuel channel conditions and pressure tube, calandria tube and fuel temperatures are analyzed at every time step.

When the disassembly criteria are satisfied, channel sections relocate into the 'holding bins' and stay there temporarily as a suspended debris bed. When the suspended debris bed mass exceeds the user-input value of 25 000 kg/per loop, the core material in the suspended debris bed, and most of the intact channels relocate into the containment vessel bottom by core collapse at about 50 800 s.

I-1.2.4.4. Calandria vessel response

Following the initiating event, the moderator temperature and pressure in the calandria vessel increase as a result of the loss of moderator cooling and heat transfer from the core. The moderator in the calandria vessel reaches the saturation temperature at about 19 800 s. At about 26 200 s, the pressure inside the calandria vessel reaches the set point of the rupture disks, and the rupture disks fail, resulting in moderator expulsion through the relief ducts. The moderator continues to discharge into the calandria, resulting in a further gradual decrease of the calandria vessel water level.

Following the core collapse at about 50 800 s, the water inside the calandria vessel is depleted. Water in the reactor vault acts as a heat sink and cools the calandria vessel. Eventually, water in the reactor vault reaches the saturation temperature and boils off. Crusts are formed on the calandria vessel walls very soon after core collapse; the crust thickness on the calandria vessel walls is in the range of 5 to 10 cm. After water in the calandria vessel is depleted, the core debris in the calandria vessel begins to heat up.

When the water level in the reactor vault falls to the calandria vessel bottom level, which occurs at about 199 000 s, the calandria vessel bottom heats up rapidly and fails due to creep, at about 199 300 s. When the calandria vessel fails, the debris relocates into the reactor vault.

I-1.2.4.5. Reactor vault and end-shield response

The pressure and water level in the reactor vault and end-shields increase gradually after the initiating event, due to the unavailability of the shield and moderator cooling systems and the resulting thermal expansion of water in the reactor vault. The reactor vault and end-shields are connected to combined vent lines to relieve over pressure through rupture disks. At about 19 100 s, these rupture disks burst. Steam is discharged from the end shields to the calandria, resulting in a decrease in the end-shield water level. The water in the RV begins to boil-off at about 81 500 s, which results in a gradual water level decrease.

At about 199 300 s, the calandria vessel fails and the corium in the calandria vessel relocates to the RV floor. Energetic corium/steam interaction was predicted by the code at about 199 300 s in the RV, following the corium relocation. Eventually, all water in the reactor vault dries out and corium reacts with the concrete floor. When the eroded depth of the concrete reaches 2 m, the RV fails at about 439 000 s.

I-1.2.4.6. Containment response

Figure I-4 shows the pressure in the lower half of the steam generator enclosure. After accident initiation, the containment pressure increases, because the PHTS coolant is discharged into the containment through the outlet header break and the PHTS liquid relief valves. When the containment pressure reaches 114 kPa (a), the dousing sprays are turned on at about 6 s; the containment pressure is thus reduced. The sprays are turned off when the containment pressure decreases to 107 kPa (a). The rapid increase (or decrease) of containment pressure, as shown in Figure 5 at the approximate times of 26 000 s, 50 800 s, 199 300 s and 440 000 s, are due to the following events, respectively: (1) the opening of calandria vessel rupture disk, (2) core collapse, (3) corium relocation from the calandria vessel, corium/steam energetic interaction and the subsequent containment failure, and (4) corium relocation into the basement after relief valve failure and subsequent steaming. The assumed containment failure pressure of 500 kPa (a) is reached at about 199 400 s.

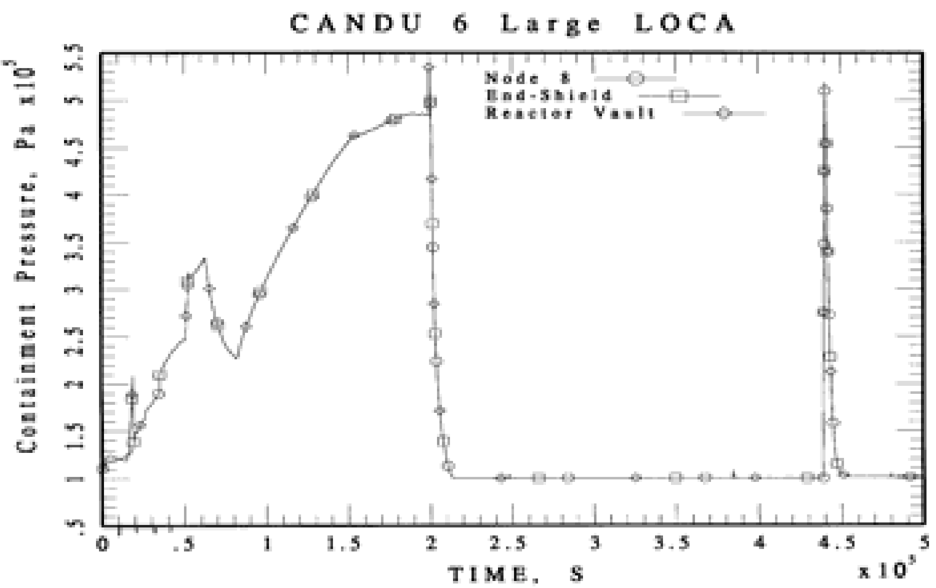


FIG. I-4. Pressure in containment, reactor vault and end shield.

I-1.2.4.7. Fission product and hydrogen release

The original inventory of the noble gases in the core is 57.7 kg, based on calculations using the coupled multi-region WIMS-AECL/ORIGEN-S code. Major portion of the noble gases is released into the CV from the fuel and the suspended debris bed during core disassembly and core collapse from about 36 000 s to about 51 000 s. Eventually, all noble gases are released into the environment, when the containment fails at about 199 400 s. No containment leakage or ventilation is modeled in the present analysis.

Figure I-5 shows the mass of CsI released in-vessel, ex-vessel (outside the calandria vessel), in the PHTS, in the calandria vessel, in containment and into the environment. The initial inventory of CsI is 27.96 kg. At about 22 000 s, fuel temperature for loop 1 is higher than 1000 K, when the fission product release from the fuel matrix begins. At about 35 100 s, the pressure tube and calandria tube rupture in loop 2, and the fuel element temperatures are greater than 1000 K; therefore, fission products are released. Because the calandria vessel rupture disks are already opened at about 26 000 s, the fission products are released through the calandria vessel rupture disks into the containment. The mass of CsI in the containment (including airborne and deposited) remains at about 1.4 kg until about 220 000 s. Because almost all of the CsI is retained in the containment by various fission product retention mechanisms, only a very small amount of CsI and CsOH totaling about 0.0068% is released into the environment, when the containment fails. When water in the RV is depleted at about 209 000 s, the corium reacts with the concrete, and fission products are released ex-vessel. At about 475 000s, 0.196 kg of CsI and 0.904 kg of CsOH are released to the failed containment and subsequently to the environment. The total amount of Cs and I released to the environment in the form of CsI and CsOH is about 0.996 kg or 3.6% of the initial Cs and I inventory.

Hydrogen is generated during the accident as a result of the following reactions: (1) Zr-steam reaction in fuel channels and during core debris oxidation in the suspended debris beds, (2) jet breakup of molten debris in the water pool of the reactor vault, and (3) molten core-concrete interaction. Analyses show that the mass of hydrogen generated in the PHTS and calandria vessel prior to calandria vessel failure is 261 kg, and the mass generated in the reactor vessel is about 2202 kg as a result of jet breakup and molten corium-concrete interaction.

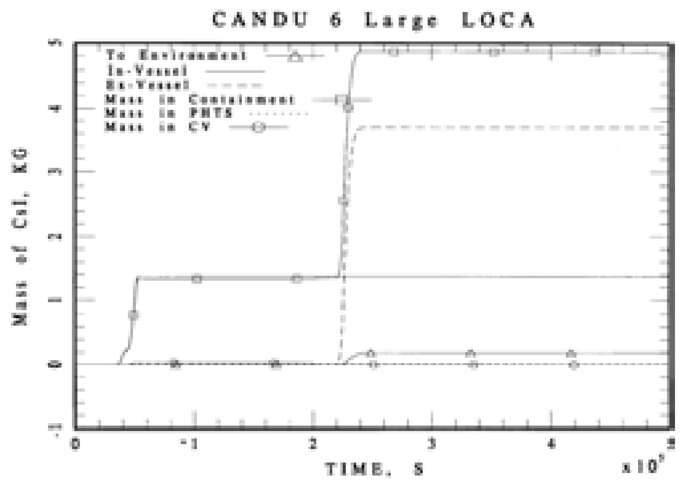


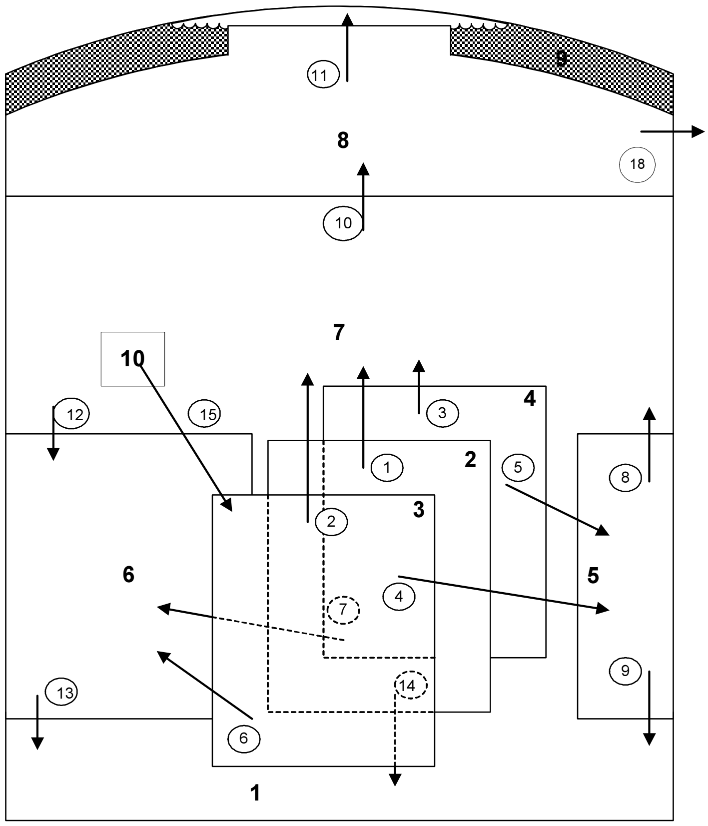
FIG. I-5. Mass of CsI released.

I-1.3. ISAAC code result

In order to demonstrate the capability of the ISAAC computer code, typical high- and low-pressure sequences, a station blackout and a large LOCA are selected and analyzed for the sample calculations. Also the nodalization scheme for the core and the containment is described.

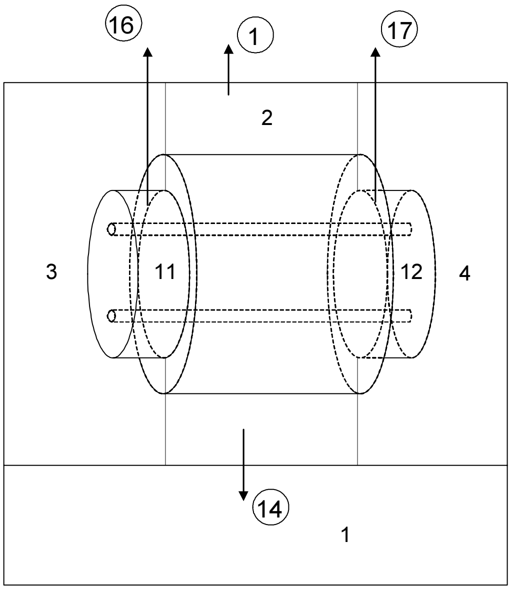
I-1.3.1. Core and containment nodalization

While the ISAAC computer code has a fixed primary system nodalization, the fuel channel configuration inside the calandria is flexible and the user is supposed to define the number of fuel channels in the broken and unbroken loop in loop 1 and loop 2, respectively. Though the code can simulate the maximum of 37 representative fuel channels in loop 1 and loop 2, a total of 6 channels (3 channels in the broken loop and 3 channels in the unbroken loop) is defined in loop 1 and the same configuration is assumed for loop 2 in this analysis. Figures I-6 and I-7 show 3 × 3 channel configuration in loop 1. Once the user sets up the core configuration, the structure of the variable heat sinks for inlet and outlet feeders is defined automatically in the code.



1 — 12: compartment number, ①–⑱: node number

FIG. I-6. Twelve compartment model for Wolsong compartment (1).



1–12: compartment number, 1– 18: node number

FIG. I-7. Twelve compartment model for Wolsong compartment (2).

The user is also required to nodalize the containment into separate compartments. The following nodalization scheme was used for the sample runs: basement (node #1), calandria vault (#2), front fuelling machine room (#3), back fuelling machine room (#4), moderator room (#5), access area (#6), boiler room (#7), upper dome (#8), dousing tank (#9), degasser condenser tank (#10), and two end-shields (#11 and #12). Figures I-6 and I-7 show the suggested nodalization. Flows of water, steam, non-condensable gases and molten corium between the compartments, are defined through a junction connecting two compartments. 18 junctions are defined, most of which are normally connected except the failure junctions that appear when certain conditions such as containment failure (#18) or concrete floor melt-through (#14) are satisfied.

I-1.3.2. Station blackout sequence analysis

In the station blackout sequences, loss of class IV power and emergency diesel generators cause all safety systems unavailable. That is, steam generator feedwater system, emergency core cooling system, moderator and shield cooling system, shutdown cooling system and other engineered safety systems fail. The liquid relief valve is assumed to fail open at the beginning and the dousing sprays are also assumed to be unavailable. The boiler pressure is controlled by the main steam safety valves which open and close at their set points. The containment is assumed to fail at 420 kPa(g) (519 kPa(a)). Table I-3 shows the major events during the accident.

When the accident occurs, the reactor scrams right away. Then only the decay power is generated from the fuel. The PHTS pressure, which is shown in Fig. I-8, drops at the beginning after reactor scram and increases due to the less heat transfer to the steam generators as the feedwater stops. The peak pressure is controlled at the set point of degasser condenser tank relief valve until the fuel channel fails at 3.3 hours into the accident. The pressurizer pressure shows the independent behaviour after the pressurizer is isolated from the PHTS. The water mass in each loop, which is about 32.5 tons initially, increases up to about 42 tons from the pressurizer inflow and then drops down due to the discharge through the liquid relief valve. The steam generator pressure increases to the set point of main steam safety valves (5.11 MPa) and then decreases following the channel tube failure at 3.3 hours. The steam generators are depleted around 2.5 hours.

TABLE I-3. MAJOR EVENTS DURING STATION BLACKOUT

Time		Major events
Hours	Seconds	
0	0	Loss of AC power and diesel generators ECCS off Main/auxiliary feedwater system off Moderator, shield cooling system off Local air coolers off
0	0	Reactor scram
0	0	Steam generator main steam isolation valves closure
0.001	3	Steam generator main steam isolation valves start open
2.5	9059	Four Steam generators dryout
3.0	10804	Core starts uncovery in loop 1 and loop 2
3.3	12042	Calandria rupture disc opens
3.3	12008	Pressure tube/Calandria tube fail (rupture)
3.8	13534	Beginning of core disassembly
4.1	14863	Corium relocation onto calandria bottom
9.0	32293	Calandria vessel depletion
24.8	89358	Containment failure
37.3	134604	Calandria vessel failure

The fuel is overheated between 4 and 9 hours and then the damaged fuel is relocated from the channel to the calandria. As shown in Fig. I-9, the initial fuel relocation occurs around 4 hours and about 140 tons are collected in the calandria. When the calandria fails at 37 hours, fuel mass in the calandria is delivered into the calandria vault. The water inventory in the calandria and the calandria vault is shown in Fig. I-10. the water in the calandria decreases when the calandria rupture discs fail (~3.3 hours) and then 217 tons of moderator is evaporated completely in 9 hours. The water in the calandria vault starts to decrease around 12 hours and then gets depleted after 42 hours. Calandria vessel failure at 37 hours causes sudden drop of water inventory in the calandria vault.

The containment pressure is shown in Fig. I-11. As long as the moderator is available as a heat sink, the pressure increases. When the water in the calandria vault starts evaporation, the containment pressure increases again and eventually exceeds the containment failure pressure which is assumed as 519 kPa. In this scenario, the containment fails at 24.8 hours, while the calandria vessel fails at 37.3 hours. This result indicates the importance of the containment heat removal system like local fan coolers in CANDU 6 type plants.

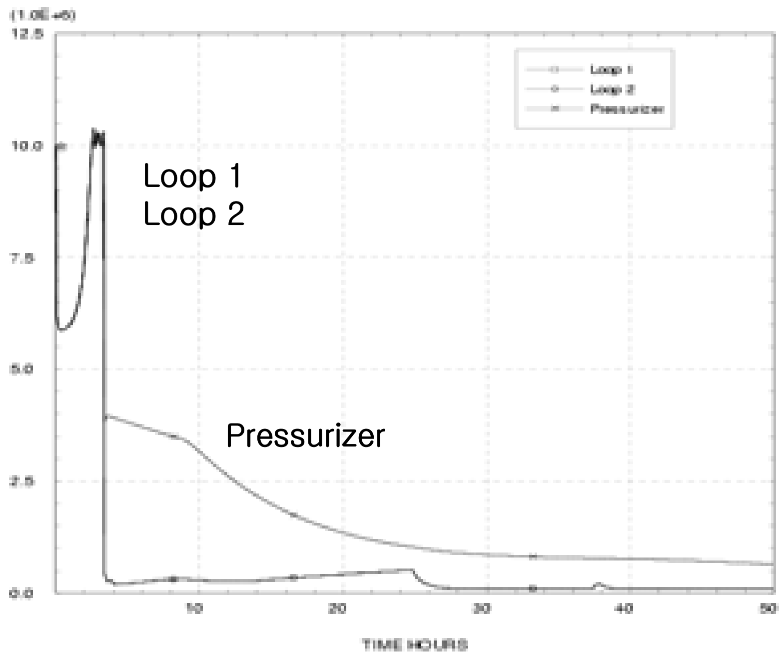


FIG. I-8. Pressure behaviour in PHTS and pressurizer during station blackout [Pa].

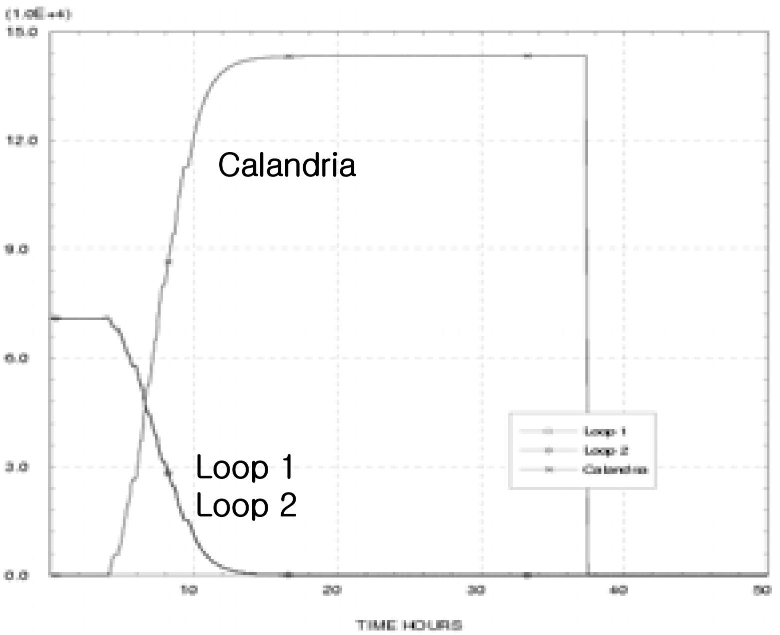


FIG. I-9. Fuel relocation behaviour at core and calandria vessel during station blackout (g/s).

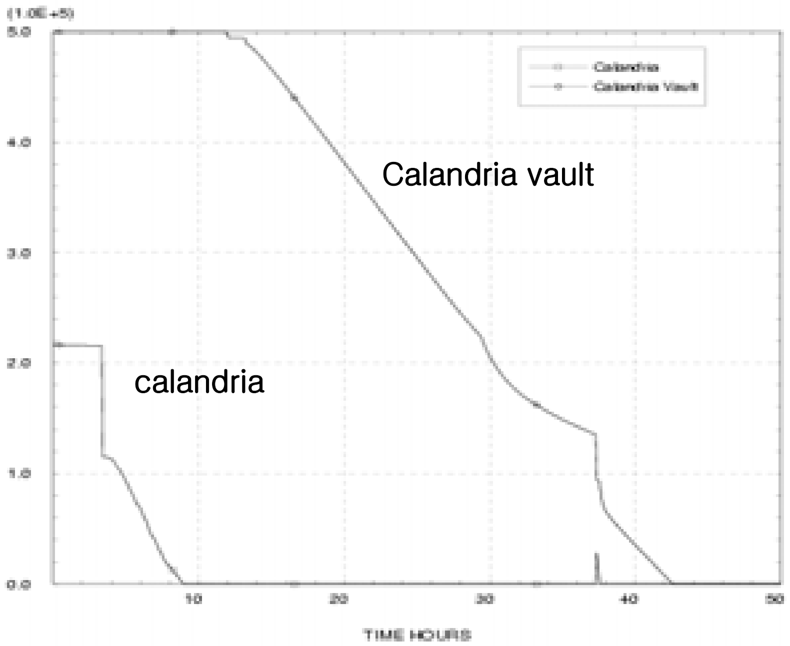


FIG.I-10. Water mass behaviour at calandria and calandria vault during station blackout (g/s).

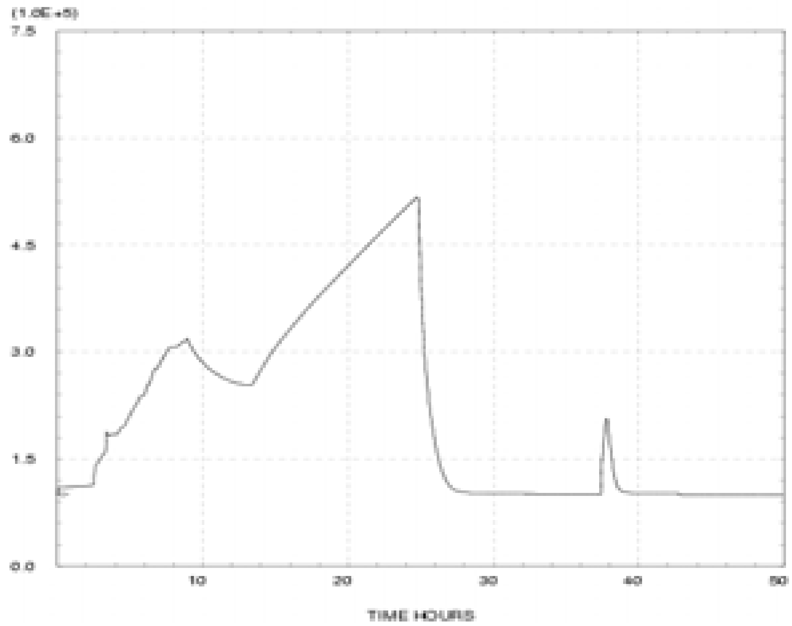


FIG.I-11. Containment pressure behaviour during station blackout (Pa).

I-1.3.3. Large LOCA sequence analysis

As a typical low pressure sequence, large break at reactor outlet header in loop 1 is assumed whose break area is 0.2594 m^2. To investigate the severe core damage, safety systems like the core cooling system, moderator and shield cooling system are assumed to fail. Crash cooldown and dousing sprays are assumed to be available. The reactor scrams at 0.87 seconds based on final safety analysis report of Wolsong 2 NPP. Table I-4 shows the major events during the accident.

The PHTS pressure is shown in Fig. I-12. Although the break occurs in loop 1, both loop pressures drop down quickly as long as they are connected via the loop isolation valves. When the loops are isolated, the PHTS pressure in loop 2 increases as no feedwater is available to remove the decay heat. Then the intact loop pressure follows the similar trend to the high pressure sequence like a station blackout sequence. That is, loop 2 pressure is controlled by the D_2O tank relief valve set point and then drops down after fuel channel failure. The water inventory in loop 2 decreases when the fuel channel fails at 2.5 hours. As the crash cooldown operation is successful, steam generator pressure drops down 30 seconds after the LOCA signal. The loop 1 and loop 2 steam generators are depleted around 40 hours and less than an hour respectively.

The fuel is overheated between 2 and 7 hours and then the damaged fuel is relocated from the channel to the calandria. Figure I-13 shows the status of fuel material distribution in loop 1, loop 2, and in the calandria. When the calandria fails at 36.7 hours, fuel mass in the calandria is delivered into the calandria vault. The water inventory in the calandria and calandria vault is shown in Fig. I-14. The moderator is depleted around 6.7 hours and the water in the calandria vault gets depleted after 40 hours. The containment pressure is shown in Fig. I-15. In this scenario, the containment and the calandria vessel fail at 36.7 hours

TABLE I-4. MAJOR EVENTS DURING LARGE LOCA

Time		Major Events
Hours	Seconds	
0	0	Large LOCA initiates ECCS off, local air coolers off Moderator, shield cooling system off
0	0	Steam generator main steam isolation valves closure
0	0.87	Reactor scram manually Main/auxiliary feedwater forced off
0.006	21	Loop 1 core uncovery starts
0.007	24	Pressurizer isolated
0.009	33	Steam generator main steam isolation valves manually open for crash cooldown
0.04	133	PHTS loop 2 pumps off
0.93	3361	Loop 2 steam generator dryout
1.6	5649	Calandria rupture disc opens
1.9	6856	Loop 1 fuel channel failure due to creep
1.9	6962	Beginning of core disassembly (from loop 1)
2.1	7709	Corium relocation onto calandria bottom
2.5	9047	Loop 2 fuel channel failure due to creep
2.9	10565	Beginning of core disassembly (from loop 2)
6.7	24027	Calandria vessel depletion
36.7	132130	Calandria vessel failure
36.7	132221	Containment failure

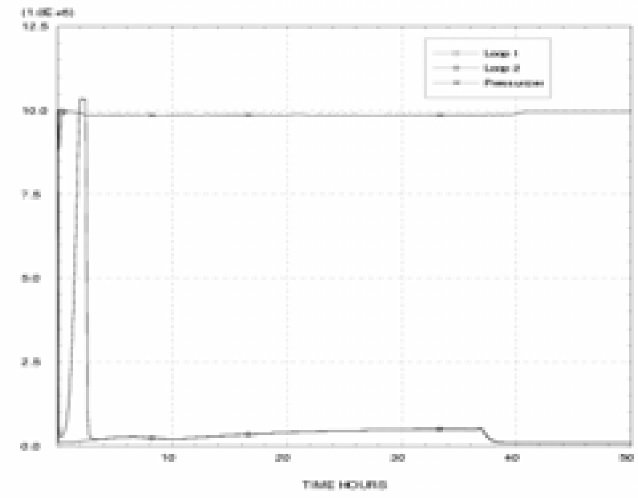


FIG.I-12. Pressure behaviour in PHTS and pressurizer during large LOCA (Pa).

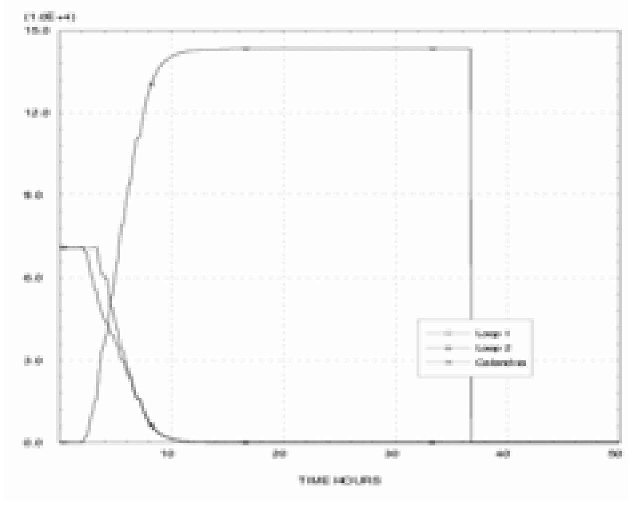


FIG.I-13. Fuel relocation behaviour at core and calandria vessel during large LOCA (g/s).

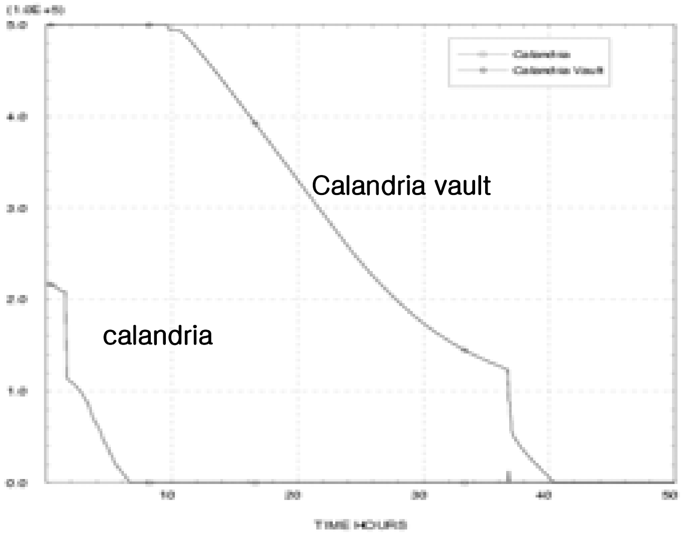


FIG. I-14. Water mass behaviour at calandria and calandria vault during large LOCA (g/s).

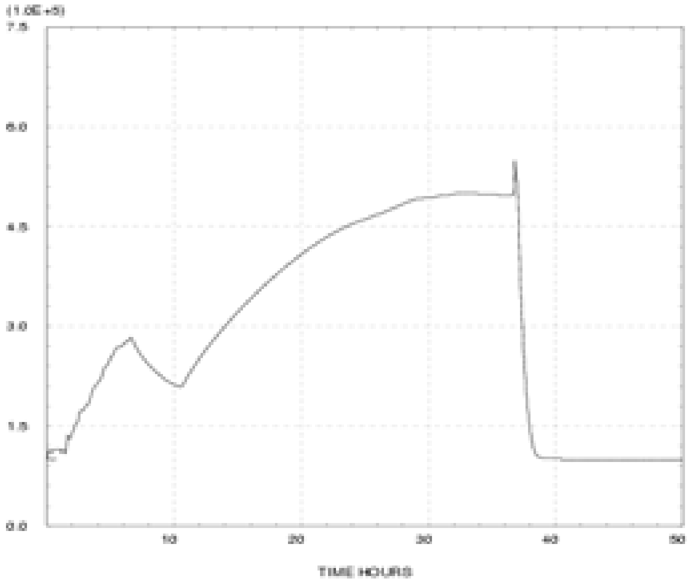


FIG. I- 15. Containment pressure behaviour during large LOCA (Pa).

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Annex II

OVERVIEW OF COMPUTER CODES FOR LIMITED CORE DAMAGE ACCIDENTS

II-1. Introduction

This Annex provides a description of a typical code suite used to model Limited Core Damage Accidents as shown in Fig. II-1. As mentioned in Section 4, LCDAs are usually considered as part of the design basis for PHWRs. As a result, they are analyzed in detail, using mechanistic computer codes to fully capture all major phenomena and their interactions. The disciplines involved in modeling LCDAs are:

- Reactor physics: supplies the initial core state (e.g. fuel burn-up) and the power transient;
- Thermal-hydraulics: provides the coolant conditions transient, the temperature distribution in the moderator, and any pressure transients in the moderator from in-core breaks;
- Fuel channel: models fuel channel thermal, chemical and mechanical behaviour;
- Fuel: provides the fuel thermal, chemical and mechanical response, and fission product release and transport behaviour;
- Containment: assesses the containment thermal-hydraulic conditions, hydrogen combustion behaviour and fission product behaviour;
- Atmospheric dispersion: calculates the dose to public based on release characteristics from containment and weather conditions;

The following sections will provide brief descriptions of the computer codes used in each of these discipline areas.

II-2. Physics analysis

II-2.1. WIMS-AECL

WIMS-AECL is a multi-group transport code used for general-purpose lattice calculations. Cross sections are provided from a data library, e.g. ENDF/B-6. Inputs include geometry, cell dimensions and compositions, and temperatures and densities of cell materials. The code provides 2-group cell-averaged parameters for RFSP calculations, tabulated as function of burnup, fuel temperature and coolant density.

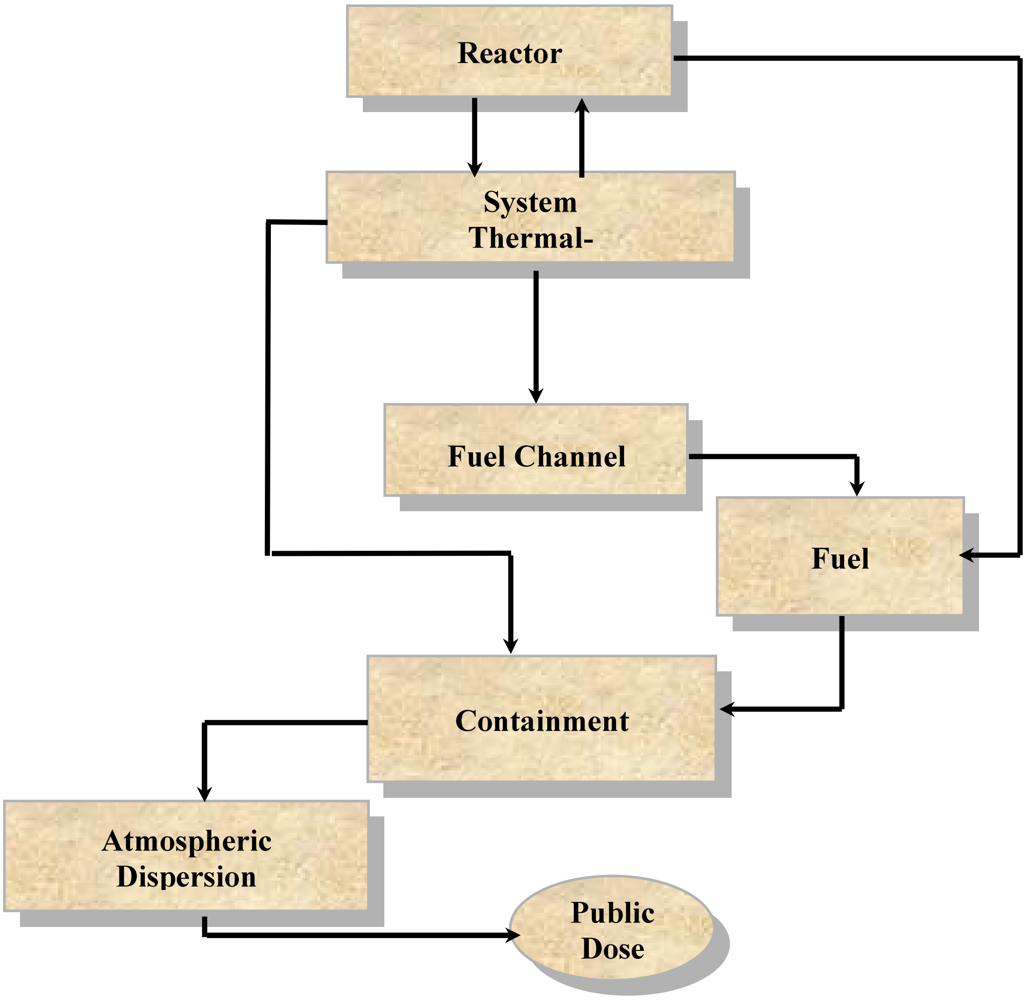

FIG.II- 1. Schematic showing links between the various disciplines used to model LCDAs

II-2.2. RFSP

RFSP calculates space-time-dependent flux and power distributions using 2-group lattice parameters from WIMS-AECL. It can be coupled to CATHENA to capture feedback between thermal-hydraulic parameters (e.g. coolant void transient) and reactor power. Inputs include geometry, device incremental cross-sections, and coolant and fuel conditions from CATHENA. Outputs include bundle and channel powers, and thermal and fast flux distributions.

II-3. Thermal-hydraulic analysis

II-3.1. CATHENA

CATHENA is a non-equilibrium two fluid thermal-hydraulics code. It is used to calculate the overall system response to a transient, and the detail channel response. CATHENA includes a generalized heat transfer package for modelling detailed temperature responses in fuel channels and has both D_2O and H_2O properties. In its system mode of operation, inputs are piping geometry, component specifications, system controls and properties, and channel and reactor powers (e.g. from RFSP). Outputs are the system thermal-hydraulic behaviour, boundary conditions for detailed channel analysis and coolant and fuel conditions. In its single-channel mode of operation inputs are fuel and channel geometry, axial and radial power distributions, and transient header boundary conditions from CATHENA fuel circuit analysis. Outputs include fuel boundary conditions for ELOCA, pressure tube temperatures and strain, initial conditions for TUBRUPT, and hydrogen generation for GOTHIC.

II-4. Fuel analysis

II-4.1. ORIGEN

ORIGEN calculates the time-dependent concentrations of isotopes from neutron transmutation, fission, radioactive decay, input feed rates, and physical or chemical removal rates. Its inputs include cross-sections from WIMS, and fuel composition and power / burnup history from RFSP. Outputs are the total inventories of fission products used as input to fission product calculations.

II-4.2. ELESTRES

ELESTRES models the thermal and mechanical behaviour of an individual fuel element during its irradiation life under normal operating conditions. It calculates two-dimensional deformation behaviour, and one-dimensional thermal and fission product behaviour. Inputs include fuel geometry and composition, fission product inventory from ORIGEN, and power / burnup history from RFSP. ELESTRES provides heat generation rates, clad strain, and fuel to clad gap conductance for ELOCA, and fission product distributions for SOURCE.

II-4.3. ELOCA

ELOCA calculates fuel element thermal and mechanical behaviour during a postulated transient. It takes as input fuel geometry; heat generation rates, clad strain, and fuel-to-clad gap conductance from ELESTRES; pressure, temperature, and fuel-to-coolant heat transfer coefficient from CATHENA; and the power transient from RFSP. ELOCA provides the mechanical response, including cladding failure; and fuel and clad temperatures as input to SOURCE.

II-5. Fission product analysis

II-5.1. SOURCE

SOURCE is used to calculate the amount and type of fission products released from fuel. To do so it models diffusion, grain boundary sweeping, vapour transport, gap transport, fuel/zircaloy interaction, temperature transients, grain boundary separation, etc. SOURCE takes as input fuel temperature transients from ELOCA, and the power transient from RFSP. It provides the fission product release transients used as input by SOPHAEROS.

II-5.2. SOPHAEROS

SOPHAEROS calculates the fission product deposition and transport through the heat transport system components and piping. Its inputs are the fission product release transients from SOURCE and thermal-hydraulic boundary conditions (steam pressure and temperature, materials and temperatures) from CATHENA. SOPHAEROS provides the amount of fission products retained in the primary heat transport system, and the release transients into containment.

II-6. Moderator analysis

II-6.1. MODTURC_CLAS

MODTURC_CLAS is a computational fluid dynamics code used to calculate the moderator velocity and temperature distributions. Its input includes moderator circuit geometry and conditions, and heat load to the moderator from CATHENA. Its output is the moderator subcooling margin used to determine whether or not there is adequate cooling to prevent fuel channel failure.

II-7. In-core damage analysis

II-7.1. TUBRUPT

TUBRUPT is used to determine the pressure transients within the calandria vessel caused by an in-core channel break. Its inputs are core and channel geometries, and thermal-hydraulic conditions at the time of channel failure (pressure and temperature) from CATHENA. TUBRUPT predicts the resultant damage to the calandria vessel, adjacent fuel channels and shut-off rod guide tubes.

II-8. Containment analysis

II-8.1. GOTHIC

GOTHIC is a 3-dimensional, 2-fluid thermal-hydraulic code used to determine the transient conditions inside the reactor containment. Its inputs are containment geometry and heat sinks; and the break discharge and hydrogen release transients from CATHENA. It provides thermal-hydraulic conditions as input to SMART, and containment pressure transients and hydrogen distributions for assessments of threats to containment integrity.

II-8.2. SMART

SMART calculates fission product behaviour inside containment and release via containment leak paths. It includes models for aerosol and iodine behaviour. SMART take as input thermal-hydraulic conditions from GOTHIC and fission product release transients from

SOPHAEROS. Its output is fission product release transients to be used in dose assessments by ADDAM.

II-9. Dose assessment

II-9.1. ADDAM

- ADDAM uses a Gaussian dispersion model to calculate fission product dispersion factors, with corrections due to nearby buildings, release height, and deposition. Inputs include weather patterns, local topography and fission product release transients from SMART. The code calculates:

- Internal dose due to inhalation and skin absorption of radioactive material

- External dose from exposure to radioactive material in the cloud (cloud shine)

- External dose due to exposure to radioactive material from ground deposition (ground shine)

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