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PERSPECTIVE ON ANALYSES OF CORE DAMAGE IN CANDU REACTORS

C. Blahnik[1], M.J. Brown[2], T. Nitheanandan[2] and S.M. Petoukhov[2]

[1] CBA Inc., Toronto, Ontario, Canada
[2] AECL, Chalk River, Ontario, Canada

Abstract

This paper overviews deterministic analyses of core damage in CANDU reactors. Reactor states with core damage at decay powers are described and modeling of accident progressions to these states is discussed. A perspective on core damage analyses is provided.

1. Introduction

Reactor cores are damaged by the effects of sustained mismatches between heat generation and removal. All nuclear reactors have engineered safety systems that shut down the reactor and establish a timely emergency cooling of the core to prevent core damage. More recently, provisions were added to existing reactors and were specifically designed for advanced reactors to mitigate consequences of accidents that progress into core damage. Analyses are used to study such accidents and the resulting insights guide provisions for mitigating core damage.

Four Reactor Damage States (RDSs) that could be encountered in CANDU reactors are identified in Figure 1. These states are grouped and labelled differently in the industry (see Core Damage States (CDSs) in the figure). Accidents progress from top to bottom of Figure 1 to an RDS row defined by the availability of heat sinks. Within the RDS row, differences can exist in fission product (FP) transport routes and in the environments to which debris is exposed, which affect the characteristics of radioactivity release from the reactor. The RDSs have been further grouped into broader categories of ‘limited’ and ‘severe’ core damage. The Limited Core Damage (LCD) states are unique to CANDU reactors. In these states, a portion of reactor fuel transforms to debris but most if not all of this debris remains within the regularly spaced fuel channels. The fuel debris is readily coolable by water applied to the inside or outside of the fuel channels. Severe Core Damage (SCD) is analogous to ‘Core Damage’ in Probabilistic Safety Assessments (PSAs) of Light Water Reactors (LWRs). In this category, the overheated fuel and other core materials transform into core debris of irregular geometry and unstable properties located within a metal vessel (i.e., the calandria vessel (CV) for CANDU reactors). A molten ‘corium’ forms if debris dries out; this corium may remain within the externally cooled CV (in-vessel retention) or relocates into the surrounding concrete vault or metal shield tank if the CV fails (ex-vessel accident progression).

This paper reviews the current understanding of CANDU damage states with emphasis on modelling and gives a perspective on deterministic analyses of core damage in shutdown reactors.

2. Limited Core Damage (LCD)

LCD is briefly discussed for completeness. It may be caused by short-lived power-cooling mismatches at elevated reactor powers (the first row in Figure 1) or by a prolonged loss of Emergency Core Cooling (ECC) at decay powers (the second row in Figure 1). LCD states were investigated before worldwide examinations of high-temperature core behaviour intensified following the accident at Three Miles Island (TMI) Nuclear Power Plant (NPP) in 1979.

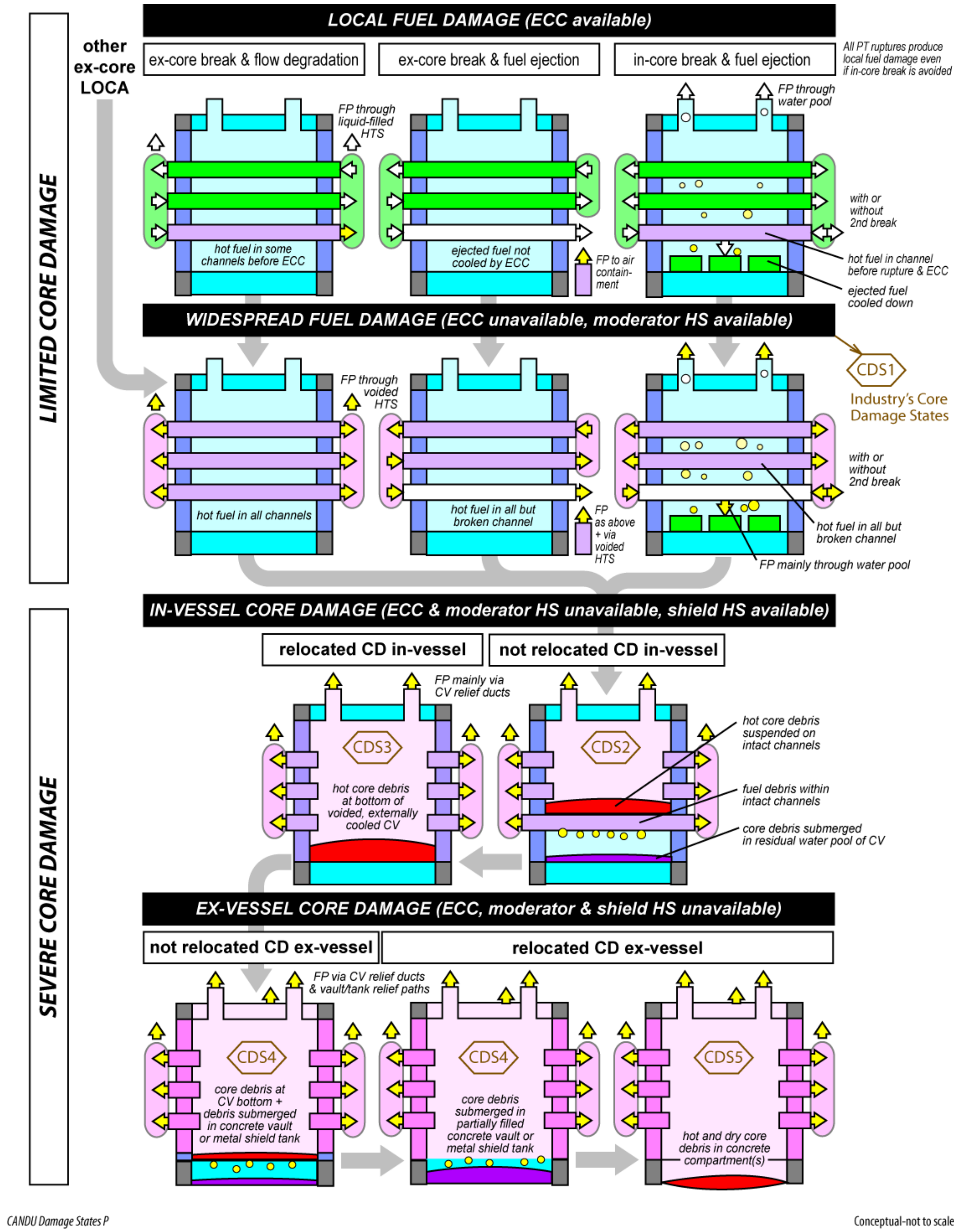

Figure 1 CANDU Reactor Damage States

(based on Figure 5 in [1])

2.1 Local Fuel Damage (LFD)

Safety concerns related to LFD include (i) maintenance of fuel channel integrity during brief power-cooling mismatches within the Heat Transport System (HTS) and (ii) propagation of damage following in-core channel ruptures. The simulations involve interplay of nuclear physics, thermal-hydraulics, and thermal-chemical and thermal-mechanical phenomena. They tend to be sensitive to details of the local environment that is rapidly changing due to the actions of reactor protective systems. Early analyses employed linked computer models to represent the multi-disciplinary phenomenology in conjunction with conservative assumptions regarding triggers of the transients. Extensive Research and Development (R&D) programs were undertaken to study fuel and channel behaviour at high temperature (discussed further in Section 2.2). As described in [2], advances in understanding void reactivity feedbacks reduced estimates of the safety margin in Large Loss of Coolant Accident (LLOCA; top left RDS in Figure 1). An improved understanding of break opening dynamics in ductile piping indicates a dramatic increase in the estimated safety margin during a LLOCA. Changes were made to the reactors and operating limits in order to improve the safety margins. CANDU reactors meet the defence-in-depth expectations of international standards and, for existing reactors, discussions among stakeholders are ongoing regarding future activities in this area [2]. In terms of integrity challenges stemming from in-core fuel channel ruptures (top right RDS in Figure 1), R&D results confirm that the original analyses are conservative (e.g., [3] to [5]). More realistic models of challenges to the integrity of CANDU cores due to in-core ruptures have been proposed [6].

2.2 Widespread Fuel Damage (WFD)

The damage states in the second row of Figure 1 involve a sustained loss of both normal cooling and ECC. The liquid moderator in the CV acts as the alternate heat sink in reactors other than Pickering A, where the moderator may be dumped as part of emergency shutdown. Fuel channels become internally voided in a large portion of the reactor core (up to all channels, depending on HTS loop isolation). The fuel and pressure tubes (PTs) heat up and deform, interacting with the surroundings. The sagged configuration (bottom left in Figure 2) is typical of long term conditions in WFD accidents since the HTS is typically depressurized by the time the PTs become hot enough to deform [7]. The ballooned PT configuration in the same figure illustrates the conditions in a few high power channels, following a LLOCA with an early power-cooling mismatch (top left corner of Figure 1) or other postulated circumstances where some PTs have reached high temperatures and deformed in the radial direction before the HTS has depressurized. During a sustained loss of ECC, the long-term convection heat transfer amongst the deformed fuel elements and bundles in the channel, and from the deformed channel to the HTS or moderator, is minor and is not illustrated in Figure 2. As already noted, extensive experimental and analytical data are available for CANDU fuel and channel behaviour at high temperatures. For example, a summary of open publications in [9] contains 124 papers on fuel-channel

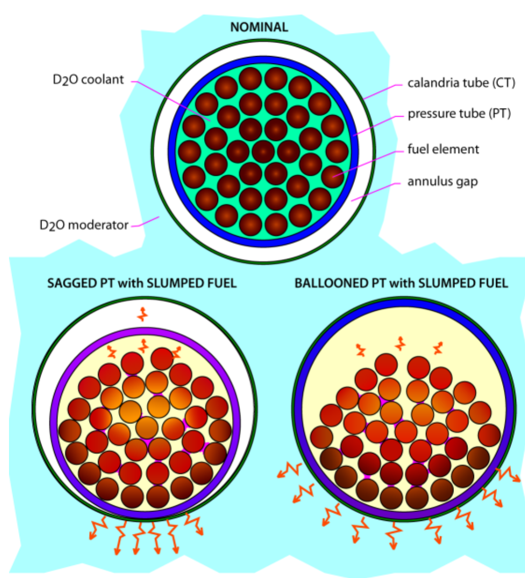

Figure 2 Channel Deformations
(e.g., Figure 5 in [8])

high-temperature behaviour and 60 papers on fuel behaviour, fission product release and fission product transport covering three decades of research work. Appreciable amounts of chemical heat can be generated by interactions of hot zirconium alloys with D_2O or H_2O, which supplements the decay heat and affects the temperature excursions. Flammable gas is also produced by these interactions, which is a potential threat to containment integrity. An extensive database is also available on flammable gas behaviour and mitigation, and the R&D on these topics is ongoing.

Notwithstanding the strong knowledge base on phenomena that come into play in WFD, realistic simulations of long term WFD progression by HTS codes in the Industry Standard Toolset (IST) are not readily obtainable. The chemical energy that needs to be dissipated from the fuel channels is sensitive to local conditions, which are affected by geometry changes within the channels (Figure 2) and also by the magnitudes and compositions of degraded flows through the parallel fuel channels (Figure 3). Furthermore, protective ZrO_2 layers can be altered by exposure to non-oxidizing

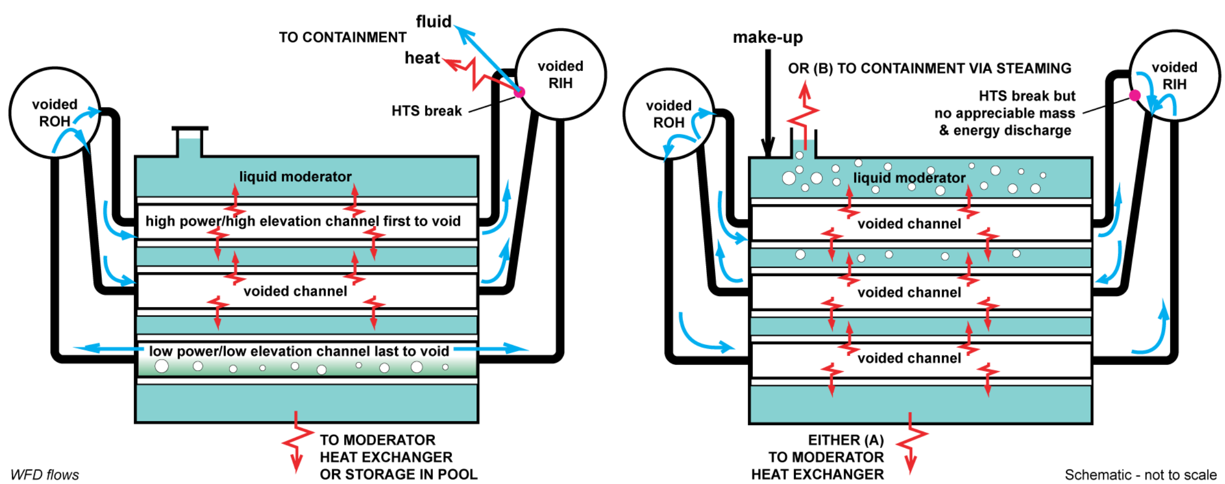

Figure 3 Degraded Channel Flows during WFD

environments (i.e., steam starved conditions) and by inter-element relocation of metal present under the oxidized cladding (illustrated by 'conduction bridges' in Figure 2). Although all these phenomena have been observed in experiments, and some mechanistic models were developed to describe them, the practical problem is representing thousands of parallel flow paths (channels and sub-channels) in the models. Each flow path has different decay and chemical heat sources due to different elevations in the core, connections to reactor headers and local fuel and channel deformations. All system codes typically use grouped, 'representative' channels to calculate the distribution of degraded channel flows through the core. The fuel geometry changes are not mechanistically simulated; they may be emulated via user inputs or neglected. The conductive heat transfer between the deformed fuel elements, and from the slumped fuel bundles to the moderator, may be emulated by inputs, smeared models or neglected. Isothermal correlations derived for Zr oxidation by fresh steam are employed even when the oxidation potential of the fluid within the channels and sub-channels is significantly reduced. Understanding WFD is not so much obtained from simulations by system codes as from broader assessments of all available data.

3. Severe Core Damage (SCD)

A capability to dynamically model accident progressions beyond WFD was acquired by Ontario Hydro (now Ontario Power Generation, OPG) following a recommendation of the Ontario Nuclear Safety Review [10] that severe accidents be analysed for Ontario's NPPs. The Modular Accident Analysis Program (MAAP) for CANDU reactors (MAAP-CANDU) was developed in late 1980s. It retained the benefit of post-TMI R&D on severe accident phenomena embedded in MAAP-LWR models and benchmarking while representing the unique features of CANDU NPPs [11]. The MAAP modelling approach is systematic and practical. The intent is to represent all reactor systems and structures as well as all known severe accident phenomena with process and phenomenology models, all integrated together to dynamically capture feedback effects. The modular code structure is flexible to allow more detailed representation should a particular process or phenomenon significantly affect the accident progression or consequence. The latter aspect recognizes that the knowledge of severe accident topics is evolving and that model refinements or additions are inevitable outcomes of improved understanding. The flexibility is particularly relevant to CANDU-specific models used beyond WFD conditions for which (i) no observations are available from actual reactor accidents, (ii) only limited experimental data are available and (iii) only a relatively short experience exists with dynamically modelling severe accidents and with the associated expert reviews of simulations (the first integrated analysis was performed in the early 1990s for the Darlington NPP [12]).

Relatively simple models of CANDU systems were developed from MAAP-LWR routines to describe the NPP behaviour up to the onset of severe core damage. These models provide scenario-specific, initial plant conditions for core degradation analysis; the code design intent was that the WFD conditions would be 'fine-tuned' to conditions estimated by the IST HTS codes when more precision at the onset of core break-up is desired (e.g., [13]). A generalized containment model was developed for analysing multi-unit CANDU containments, which has since been adopted by all MAAP code versions for all containment designs. Flexible models were developed for the investigation of unique CANDU core degradation processes based on separate assessments of plausible thermal-mechanical responses of core constituents (Figure 4). Recognition of these unique processes and phenomena led to the development of the complex CANDU core model (Figure 5), which is capable of representing the coexistence within the CV of liquid water, vapour, gases and core materials in various physical forms during the core break-up stage of severe accident progression. The MAAP-CANDU core model integrates pre-existing models of voided fuel channels with original models of coarse, suspended debris beds and the MAAP-LWR models of terminal debris at the bottom of the vessel. The fluid conditions within the still-intact channels are calculated with a user defined pressure gradient across the core (i.e., between the reactor headers). The simple HTS model of MAAP4-CANDU cannot provide meaningful estimates of core flow distribution and realistic, long term pressure gradients are not easily quantified as discussed in Section 2.2. The transformation of discrete fuel channels into various forms of debris are modelled as mass and energy transfers within and between core

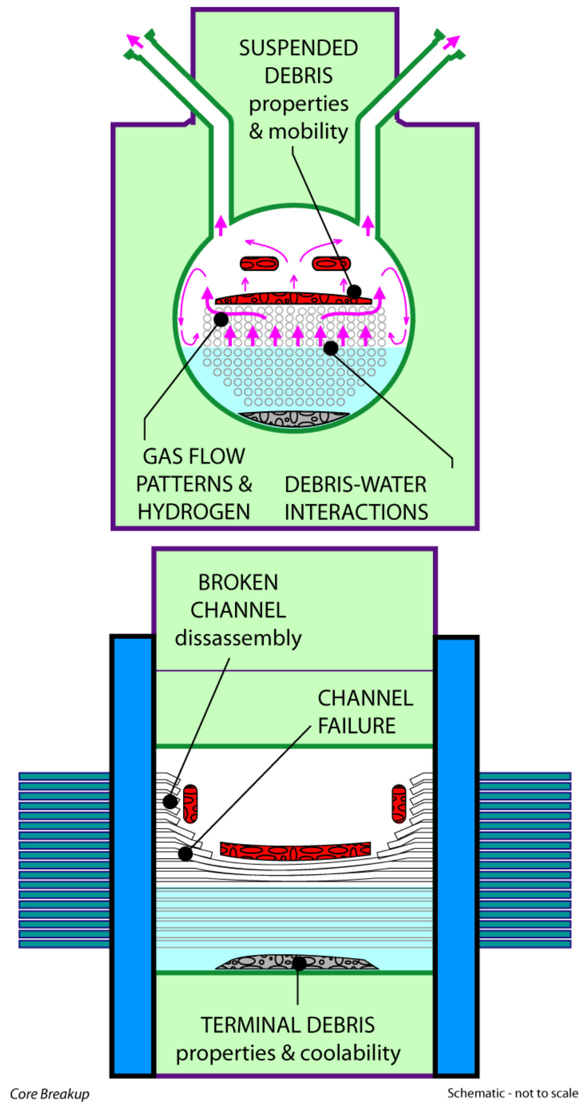

Figure 4 CANDU Core Degradation Phenomena (e.g., Figure 2 in [11])

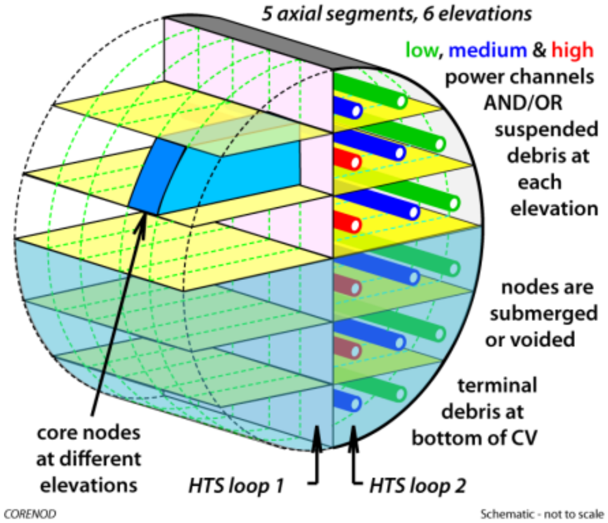

Figure 5 Typical Core Representation in MAAP-CANDU

nodes illustrated in Figure 5. These transfers are triggered by user-defined criteria (inputs) taking into account calculated local boundary conditions of channels and debris, which are separately derived. The core model is flexible enough to permit assessments of the conceivable modes of debris formation and relocation to the bottom of the vessel. The long term behaviour of core debris at the bottom of CV is modelled using MAAP-LWR routines. In line with the practical MAAP modelling philosophy, the processes and phenomena with a significant impact on the consequences can be identified and improved.

The current version of MAAP4-CANDU [14] is 4.0.7C. The CANDU-specific core models are defined and maintained by OPG, which has delegated this task to AMEC NSS. CANDU 6 and advanced/enhanced CANDU models are defined and maintained by the Chalk River Laboratories of AECL under a licence from OPG and EPRI in cooperation with AMEC NSS. Code modifications and enhancements are made by Fauske and Associates Inc. (FAI) which maintains and refines the MAAP-LWR models and assures the quality and consistency of all code versions in the MAAP family on behalf of EPRI. FAI has recently released MAAP5-Pressurized Water Reactor (PWR) code, which includes new reactor coolant system model and associated improvements of thermal-hydraulic models, numerous enhancements to other phenomenology models and a new capability of calculating on-site and off-site doses. Beta versions of MAAP5-CANDU and MAAP5-Boiling Water Reactor (BWR) are currently available, which include some of the MAAP5-PWR features. It is anticipated that the first formal version of MAAP5-CANDU will be available in about two years.

MAAP4-CANDU models have now been assembled for all existing NPPs in Canada, as well as advanced and enhanced CANDU reactor designs. Plant parameter files define the NPP physical systems and control certain phenomenological events (e.g., criteria for failure of components or boundaries, choices of alternate models, etc.). The input files define particular accident sequences for reference and sensitivity cases. The parameter and input files are compiled and documented by the users under quality assurance programs of the organizations that performs the analyses. These files are not in the public domain and the generic, phenomenology-related input parameters may not be harmonized among the users in different organizations.

Severe accident analyses of CANDU reactors are performed in support of PSAs. Analyses were recently completed for Point Lepreau by AECL ([15] to [19]), for Gentilly-2 by Candu Energy and for Darlington by AMEC-NSS (public references are not yet available). Analyses are in progress for Bruce A/B and Pickering A (AMEC-NSS), for Pickering B (ERIN/Kinetics) and for Embalse and Enhanced CANDU 6 (Candu Energy). They have also been performed but not concluded for the now discontinued development of AECL's Advanced CANDU Reactor. These are proprietary works for security and other reasons, which udergo internal and regulatory reviews but may not be readily available for reviews by other parties.

3.1 In-Vessel Core Damage (IVCD)

The damage states in the third row of Figure 1 are progressions from WFD following depletion or loss of the liquid water heat sink in the CV. The IVCD states can have (i) intact fuel channels, suspended debris and terminal debris coexisting in a partially voided or re-flooded CV (right side of IVCD row in Figure 1) or (ii) dried out debris relocated to the bottom of the CV (left side of IVCD row in Figure 1). With enhancements of defence-in-depth provisions in existing reactors and Severe Accident Management (SAM) design provisions in future reactors, the 'not relocated' IVCD state is the viable end point of severe accident progression. The MAAP4-CANDU code currently does not model the quenching of suspended debris beds; upon SAM re-flooding of the CV, it halts the suspended debris heat-up. Arresting the accident progression in this state can be emulated by artificially 'sending' the suspended debris into the water pool in the CV by means of user inputs. In such emulations, the transient rates of energy transfer to water are distorted (exaggerated) while the long term energy balance is maintained.

Technical issues are encountered in analyses of IVCD states, which involve the energy balance constituents, challenges to vessel integrity during core degradation and high-temperature phenomena during long term retention of molten debris. The chemical heat generation in suspended debris beds of variable and difficult-to-establish properties is a generic uncertainty in all reactors. This uncertainty tends to be more prominent in CANDU reactors, due to the large mass of Zr in pressure and calandria tubes that can be suspended by the still-intact, submerged fuel channels while exposed to oxidizing steam (Figure 4). The release of chemical heat affects the energy balance and accident progression. The associated flammable gas releases pose challenges to hydrogen mitigation systems and containment integrity. Calculations can examine the full range of conceivable suspended debris configurations in Figure 6, which reflects different mobility of suspended debris. Constraints on the various modes of CANDU core degradation were assessed. These assessments indicate that cores would break up in a series of channel column collapses under the weight of suspended debris (public references are not yet available). The collapses occur relatively early, at high liquid inventories in the CV (see the middle column of Figure 6). The implication of core collapse is that the exposure of suspended debris to steam, and the accompanying exothermic oxidation of Zr, are limited in time as well as extent. Following a core collapse, the terminal debris bed also contains previously submerged fuel channels. Hence, the terminal debris can contain significant amounts of unoxidized Zr and volatile FPs.

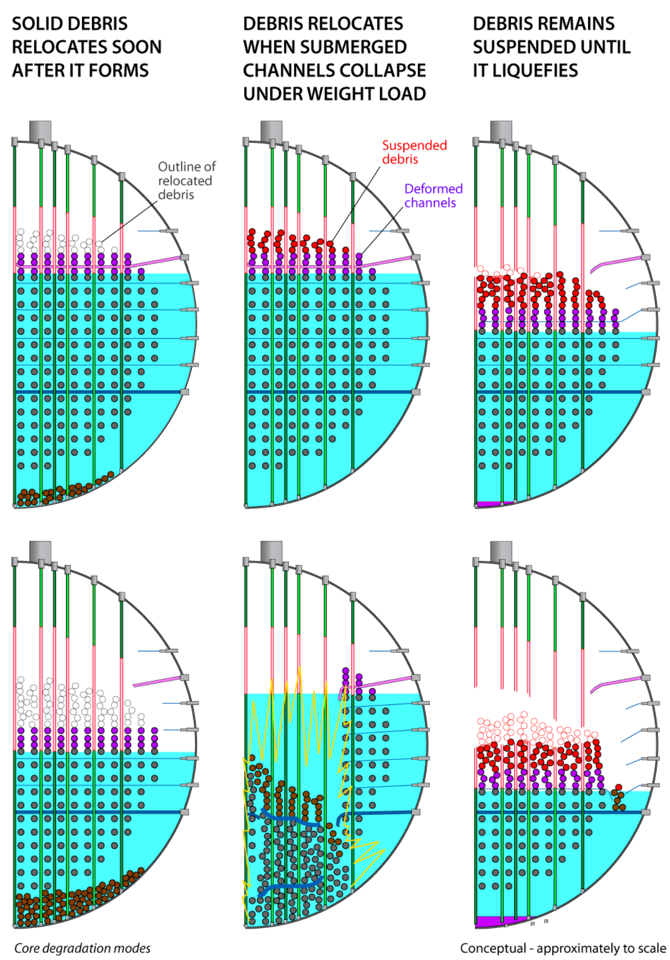

Figure 6 Range of Core Degradation Modes

Core collapses can be emulated by MAAP-CANDU with some limitations due to the relatively coarse core nodalization and channel assignment schemes that are typically employed (Figure 5 shows the common schemes). These emulations conserve the mass and energy but do not capture the dynamics of suspended debris quenching in the residual water pool within the CV. Due to its LWR heritage, relocations of suspended debris into the residual water pool are currently represented as pours of homogeneous, liquefied material and thus tend to exaggerate the pressure surges. Analyses do not report CV or containment integrity problems due to the exaggerated pressure surges. Therefore, more realistic modelling of debris relocation and quenching is viewed by some analysts as having a low priority.

When the SAM actions submerge core materials in water during core break-up and keep them submerged, this is the 'not-relocated' IVCD state in Figure 1. Intact fuel channels and coarse debris formed by disassembly of uncovered channel and collapses of submerged channel columns are coolable by submersion in water [20]. A shield water jacket around the CV provides 'insurance' that any localized debris 'hot spots' do not endanger the CV retention boundary. If the terminal debris dries out, it compacts and liquefies as illustrated in Figure 7 for the CANDU 6 layout. This is the 'relocated' IVCD state in Figure 1, which is also the viable end point of severe accident progression as long as the external cooling of CV boundaries is maintained indefinitely by SAM actions. CANDU reactors have pre-existing shield water volumes around the CV boundaries; thus, there are relatively few obstacles to be overcome in order to develop SAM strategies and provisions that facilitate the in-vessel retention of core debris by external water cooling (provisions are being made in existing and future reactors to maintain this state). Data from international R&D is available on water cooling vessel walls in contact with molten corium beds, and is available as correlations within MAAP. No assessments are reported on the heat transfer processes in the massive end shields of CANDU reactors where water passes through a porous medium of steel balls. Shield water flow paths are complex (see, for example, the normal flow paths within metal shield tanks in Figure 8). In existing reactors, the end shields are not designed for passive cooling.

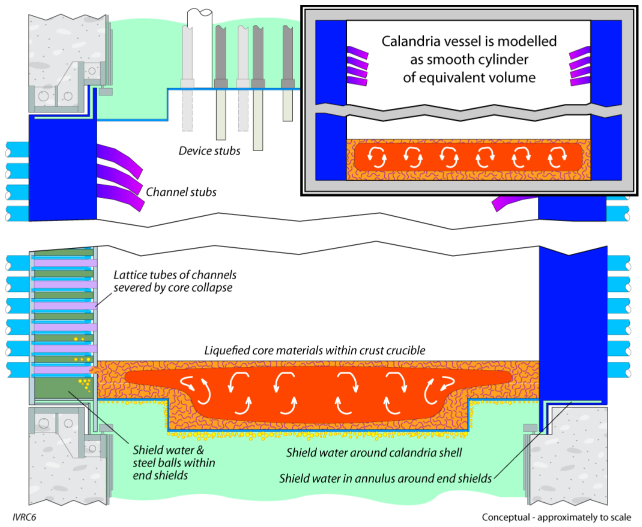

Figure 7 Dry Terminal Debris Bed

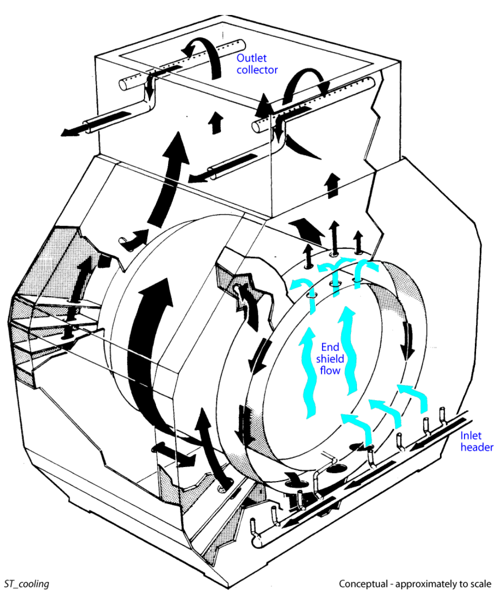

Figure 8 Flow in Shield Tanks

Correlations for FP releases in MAAP are empirical; they represent releases from small fuel pins or fragments and not from sizeable debris beds. The MAAP user can adjust the release rates but supplementary assessments are needed to produce realistic FP release transients from the terminal debris beds containing appreciable amounts of volatile FPs such as the beds formed by early core collapses.

Assessments of CV responses to the presence of molten debris were performed (e.g., [21] and the rationale for the failure criteria in simulations listed in Section 3), which evaluated the ability of the CV to retain dried-out core debris. These assessments were hampered by sparse information on thermal-chemical interactions of corium constituents within molten pools and on the associated changes of properties that can affect the long-term heat loads to vessel boundaries. These is a generic concern for all reactors. Recent tests of CANDU corium properties indicate that the melt within the CV will be homogeneous as modelled in past and present analyses (public references are not yet available).

Ambiguity emerges in the long-term responses of end shields when the pumped shield water flow is unavailable. The end shields are currently modelled by MAAP4-CANDU using (i) a dedicated end shield model in reactors with metal shield tanks or (ii) the generalized containment model in reactors with concrete calandria vaults. The porous medium is represented by a series of lumped parameter volumes (i.e., nodes with smeared, spatially-uniform properties) and junctions that evaluate unidirectional and counter-current gas and water flows. Given the complex, three-dimensional interface with corium (Figure 9), this modelling is likely too simple for assessments of the long term survivability of the end shields. Furthermore, the heat loads from the corium into the end shields are currently exaggerated by the simplified idealization of calandria shells, which averages (smears) the stepped profiles of CV shells (see inset of Figure 7). This idealization neglects the reduced depth of corium (and thus the locally reduced heat fluxes) in the proximity of the end shields. An R&D program is in progress at Chalk River Laboratories to characterize the heat transfer processes within the end shields under prototypical heat loads from corium. The results of this program will be reflected in updated models of end shield behaviour during the IVCD stages of severe accident progression when available.

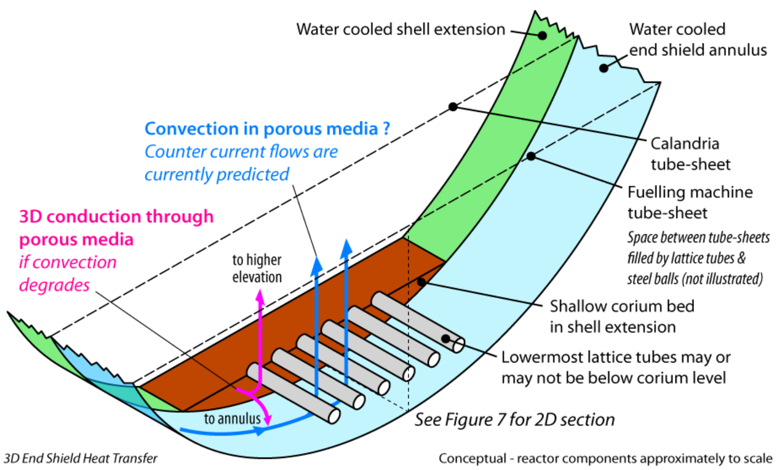

Figure 9 Corium/End Shield Interface

The technical issues reflect the practical aspects of integrated modelling (e.g., a need for simple nodalization schemes and models) as well as the uncertainties in CANDU-specific processes and phenomena during the IVCD states (e.g., suspended debris permeability and mobility, properties of porous medium in end shields, etc.). These issues affect the estimates of probability, timing and characteristics of accident progression branches (e.g., boundary failures, FP releases, etc.). Sensitivity analyses are typically performed on effects of phenomenology-related issues.

3.2 Ex-Vessel Core Damage (EVCD)

The damage states in the fourth row of Figure 1 depict a progression from the relocated IVCD state, when the CV retention boundaries cannot be maintained. These states reflect the prevailing judgement that corium would relocate downwards (i.e., through a failed calandria shell) when the shield water around the calandria shell is depleted. The onset of EVCD is reactor and sequence specific. In NPPs with metal shield tanks, and with over-pressure provisions that are not designed to accommodate the boiling of shield water, the external cooling of the CV would be lost by a relatively early drain of the water inventory through a failed weld joint at the bottom of the shield tank. This draining is followed by a gross failure of the calandria shell [12]. The resulting onset of EVCD is illustrated in Figure 10. This ex-calandria-vessel configuration of debris allows thermal-chemical interactions of hot core materials with (i) moist air from the containment atmosphere and (ii) steel shielding balls at the bottom of the shield tank. No MAAP4-CANDU models are available for the heat and fission product transport processes illustrated in Figure 10. Enhancements of shield tank relief capacity, which could avoid these conditions, were examined (e.g., [22]). However, the open literature does not report the outcomes of these examinations (i.e., the SAM enhancements that are implemented in, or planned for, reactors with metal shield tanks).

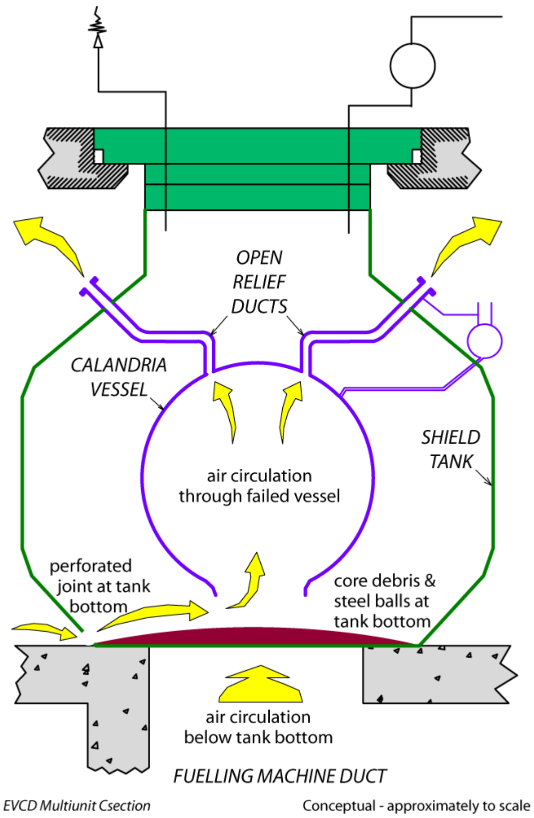

Figure 10 Onset of Ex-Vessel Damage in Multi-Unit NPPs (Figure 6 in [12]

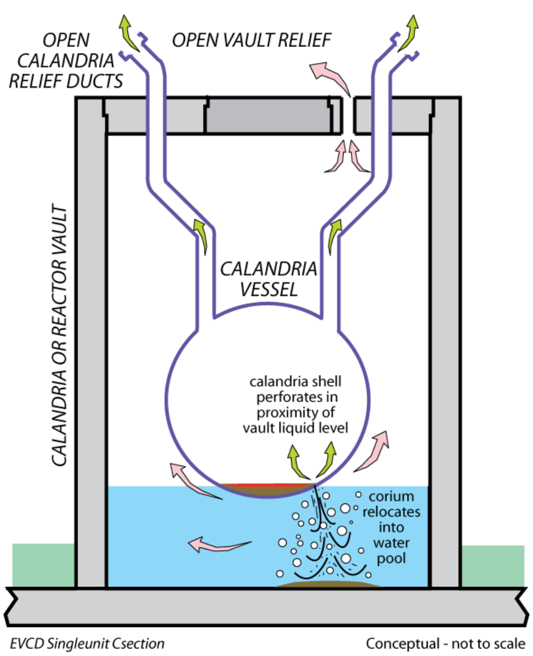

Figure 11 Onset of Ex-Vessel Damage in Single Unit NPPs

The pressure relief of concrete calandria vaults was enhanced as part of CANDU 6 refurbishment. The conditions at the onset of EVCD in the refurbished reactor are illustrated in Figure 11 as described in [18] and other papers. They are broadly similar to conditions in all CANDU reactors with adequate vault pressure relief (engineered or non-engineered) as well as to conditions in LWRs where the corium relocates from a partially submerged reactor vessel. The corium relocation is modelled in MAAP4-CANDU as a coherent pour of melt from the bottom of the CV that hydro-dynamically fragments within the remaining water in the calandria vault. While this model is conservative from the standpoint of calculated pressure surges, it is not prototypical for failures of partially submerged vessels. A partially submerged PWR vessel would fail near the top of terminal corium bed with the corium relocating by a relatively "benign dripping" into the residual water pool (e.g., [24]). A similar behaviour is expected for the partially submerged CVs as illustrated in Figure 11 but the open literature does not provide the rationale for this expectation. With the reactor vaults and containments being able to withstand the exaggerated pressure surges, an improved model of corium relocation from the failed CVs is also considered to be of low priority by some analysts.

The progression of an accident between the EVCD states in Figure 1 depends on SAM actions and on NPP provisions available to SAM. The 'not-relocated' EVCD state (bottom left of Figure 1) is the viable end state for reactors with an adequate pressure relief of concrete calandria vaults or metal shield tanks, and a water make-up into the interconnected vault/tank and CV volumes that can rapidly refill these volumes to a level well above the corium elevation within the CV. The 'relocated' EVCD state in the middle of Figure 1 is the viable end state when a sufficiently large and even floor area is available for corium spreading outside of the failed CV, in conjunction with a water make-up that keeps the ex-vessel debris submerged in water. The spreading area determines the depth (thickness) of this debris. Only a limited thickness of corium can be stabilized by one-sided water cooling. Multi-unit NPPs with fuelling machine ducts have very large spreading areas below the reactors but the duct floors have non-uniform contours (e.g., sumps), which could locally produce thick corium layers (not illustrated in Figure 1).

When ex-vessel debris is not, or cannot be, cooled by water, it eventually interacts with concrete as schematically illustrated in Figure 12. Corium-Concrete Interaction (CCI) is a complex thermal-chemical process that has been extensively researched for LWRs. Besides attacking a containment boundary, it produces 'side effects' by generating non-condensable and flammable gases and by providing driving forces for the generation and release of radioactive aerosols. As with all heterogeneous chemical reactions, the CCI is sensitive to the availability of reagents (compositions of corium and concrete) and the conditions of reacting surfaces (surface areas, temperatures and local corium and gas flows), which depend on the history of the accident progression. LWR models were developed for a broad spectrum of corium and concrete properties and are sufficiently generic to cover CANDU conditions; they are used by MAAP-CANDU. Mechanistic models of CCI are also available to examine details as needed. International R&D in this area is ongoing in support of SAM strategies that employ the ex-vessel corium retention provisions.

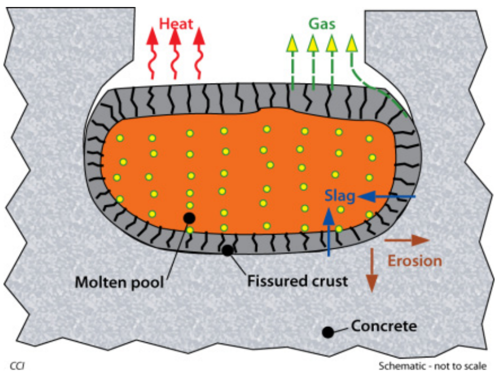

Figure 12 Corium-Concrete Interaction

4. Perspective on Core Damage Analyses of CANDU Reactors

Analyses of accidents where fuel, channel or reactor core geometry (configuration) is changing are multi-disciplinary, involving different fields of science. Phenomena with incomplete knowledge are encountered. System responses are triggered and feedbacks are encountered that depend on, and affect, the accident progression. Practical constraints are faced in modelling, resulting in the use of simplified models and empirical correlations that are valid only for specific boundary conditions.

When faced with complexity and uncertainty, different analytical approaches have been employed depending on the purpose of analysis. In assessments of NPP safety provisions, highly conservative or 'bounding' estimates are more readily obtainable. They have a downside of protracted analysis refinements and/or costly modification to physical NPP provisions if the adequacy of the original provisions is not confirmed using the extreme assumptions (e.g., the LLOCA in Section 2.1). More balanced yet reliably conservative or best estimate analyses are sought, which both require observations from actual events and/or prototypic tests, as well as synergy in the interpretation of modelling results and experimental data. The conservative analyses have been successfully performed for many of the LCD states illustrated in Figure 1. LCD analyses for future plants aim to be the best estimate analyses. Although the tools (i.e., the computer codes) used for these analyses may not precisely describe all details of a LCD accident, trends and variations of outcomes are understood. A potential problem is the maintenance of the understanding. The knowledge base is decades old for some aspects of LCD. New analysts and some decision makers may not be closely familiar with the data or with the limitations of the IST codes under conditions encountered in LCD accidents, with the attendant potential for treating the results of computer calculations as reality.

In the SCD realm, analyses are far more complex relative to the LCD realm. As already noted, CANDU reactors do not have observations from actual accidents and have only a limited base of CANDU-specific tests to guide the analyses. LWR data and models are applicable for some aspects of CANDU SCD but not all of them. As discussed in Section 3.1, most of the differences that arise in IVCD states are due to differences in reactor core and vessel layouts. The integrated severe accident codes are expected to be used for assessments of accident progression and consequence (e.g., [23]). Branch points (bifurcations) of severe accident sequences (e.g., SAM interventions, failures of equipment or boundaries) are typically evaluated by separate analyses, which may employ the results of integrated codes for the definition of boundary conditions at failure (e.g., heat generation and distribution). The failure assessments typically aim for 'balanced yet reliably conservative' estimates and employ expert opinions (e.g., [24] to [26]). Insights from these separate assessments are incorporated into the integrated system codes as simplified phenomenology models and failure models or criteria. In other words, the multi-disciplinary analyses of severe accident progressions are iterative and involve some form of expert judgment.

Obtaining a synergy of expert assessments is a slow process. This process requires that SCD analyses be published and disseminated, but the dissemination is held back by the proprietary nature of reports. In the LWR community, considerable consensus has been attained on the feasibility and limitations of core debris retention in externally-cooled reactor vessels. This is evidenced by deployments of this SAM concept in existing and advanced reactors (Figure 13). The consensus is not universal or all encompassing. Some advanced LWRs have chosen ex-vessel corium retention strategies for SAM (mainly large reactors with high power outputs), which were deemed to be more reliable for a variety of reasons. Uncertainties in high-temperature properties and interactions of core materials were one of these reasons. In the CANDU community, the expertise on the CANDU-specific aspects of SCD is growing. With the introduction of probabilistic safety requirements for licensing of new Canadian NPPs within the past decade [27], SCD modelling activities in Canada have evolved from simple assessments and development/refinement of models before these requirements were formulated to performing integrated simulations. The iterative process is in progress, which examines the predicted NPP boundary conditions, models and input data to produce assessments/judgements of the branch points in accident progression sequences. This is the analysis (as compared to performing simulations that are only inputs to analysis). The expertise is developed, and understanding of CANDU-specific SCD topics is advanced, by this process. The availability of forums for the interchange of opinions and ideas among analysts, phenomenology experts and NPP operators/designers affects the speed at which the expertise is developed and the understanding is advanced. Widespread discussions of model and input details that are embedded in severe accident calculations typically lag behind the production of simulation results.

CANDU operators and designers have somewhat different uses for the understanding of SCD. The probabilistic safety goals in [27] do not strictly apply to existing and refurbished NPPs although analyses are performed that allow comparisons with internal safety goals. Advances in the understanding of SCD provide inputs to the operators for refining SAM programs, which were developed using a PWR SAM guidance as a basis and the simple assessments of CANDU SCD states [28]. These programs are currently being re-examined as part of the post-Fukushima reviews of defence-in- depth provisions. Of main interest to these programs are the insights related to protecting debris-retention boundaries from challenges posed by severe accident phenomena.

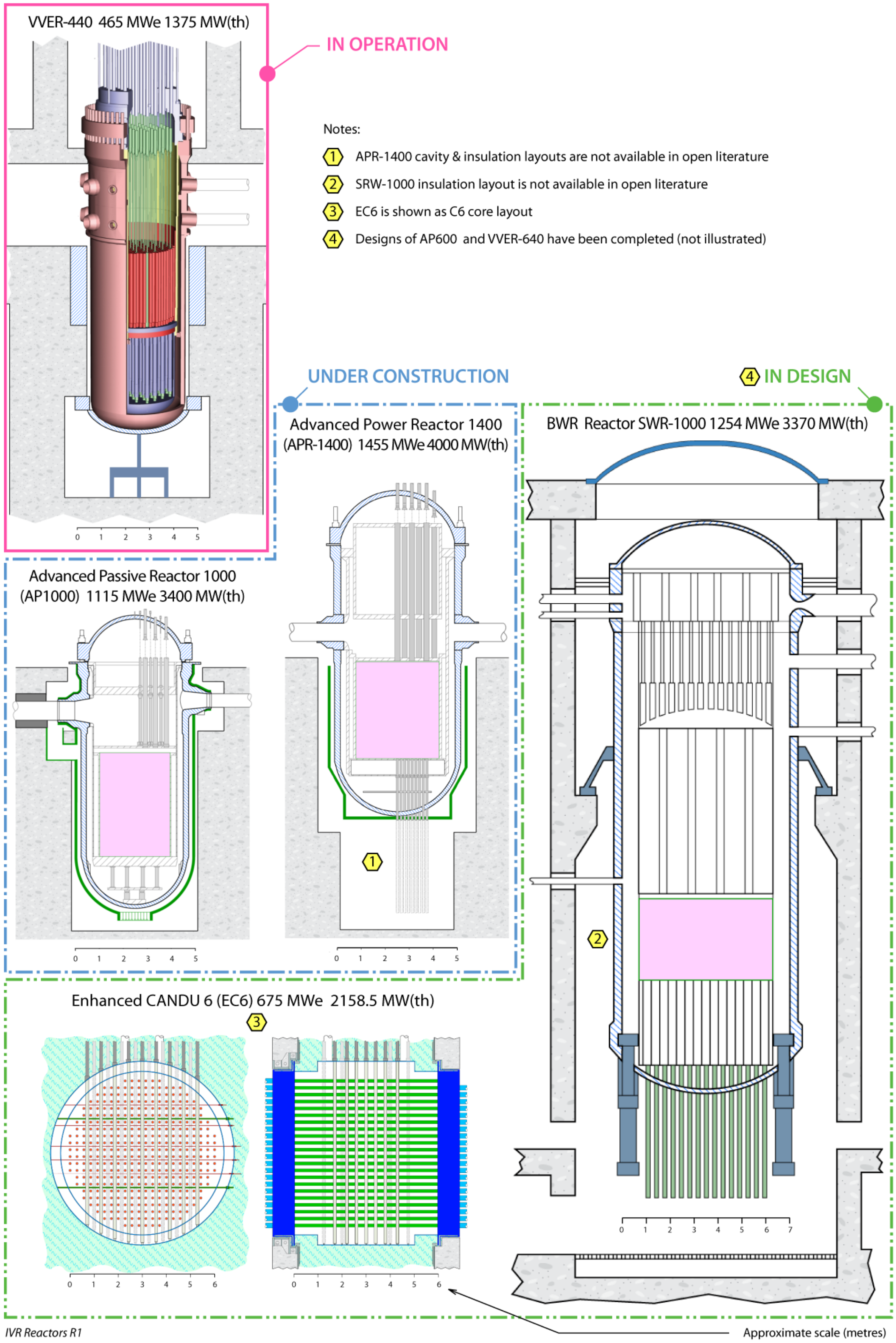

Figure 13 Reactors with In-Vessel Retention of Core Materials

For SAM-improvements, the enhancements of MAAP-CANDU models noted in Section 3.1 will be more useful than commonly perceived since they would provide more realistic symptoms of the various RDSs. The designers of future CANDU NPPs must meet the requirements of [27]. In addition, they are influenced by the evolution of 'advanced' LWRs, which have provisions for passive or automated cooling of degraded cores for considerable time after the SCD has occurred. Manual SAM actions are still required in the long term but the reliability of these actions is enhanced in modern designs. The configurations and properties of the retention boundaries are not 'frozen' in future NPPs; however there is an economic incentive to use the existing layouts, designs and technologies. Options for passively cooling all CV boundaries are inherent in the concept of CANDU reactors. Options for reliably cooling ex-vessel debris are also available for future reactors. Modelling and analyses are the main means of assessing design options. The technical issues noted in Section 3.1 broaden the range of boundary conditions to be considered in design assessments. The designers typically need more detailed information regarding challenges to which retention boundaries might be exposed. Analyses of specific topics are performed, or when practical, experiments are commissioned to provide missing information.

5. Conclusions

The analysis of core damage is an iterative process that involves modelling and interpretations of results by knowledgeable people in light of available observations and theoretical bases. In all reactors, modelling local changes in core conditions and geometry, and the attendant interactions of hot materials with local environments, is challenging as exemplified by discrepancies in hydrogen generation between observations from, and initial simulations of, severe accidents in LWRs.

In CANDU reactors, configuration changes and interactions during unique WFD states are well understood from extensive R&D; nevertheless, simulating WFD is challenging due to complex local geometry and environment changes. The WFD is a predecessor of SCD; the variability in the local properties of core materials is propagated to the IVCD states. Models of IVCD states are theory-based; no large-scale observations of CANDU core degradation exist and only few small-scale tests are available to corroborate the current modelling concepts (additional tests are ongoing or planned). A broad range of core reconfigurations (i.e., core break-up pathways) can be envisaged. These reconfigurations are affected by SAM actions and they affect the chemical energy releases and hydrogen generation. Assessments of core reconfigurations have been performed but consensus on the phenomenology awaits the dissemination of information and discussions amongst experts. Meanwhile, the range of core reconfigurations during IVCD is being examined analytically in order to understand the implications of CANDU core relocation phenomena.

Currently, we can dynamically calculate a progression of SCD in CANDU reactors given externally-defined branch points in the accident sequence (e.g., SAM interventions, boundary failures) and properties of core debris (e.g., geometry, permeability). Modelling issues exist as noted in Section 3.1, which need to be accounted in the interpretation of results. Some of these issues can be alleviated by model enhancements. Perhaps the most significant topics in terms of realistically quantifying SCD consequences are (i) the behaviour of core debris in failed shield tanks of multi-unit CANDU reactors and (ii) the long-term behaviour of end shields in CANDU reactors with adequately vented concrete vaults or shield tanks that can maintain the external cooling of calandria shells indefinitely. The latter is being addressed by an R&D program.

Assessments of challenges to debris retention boundaries and survivability of these boundaries have been performed. Challenges were examined using existing data as well as some mechanistic simulations of interactions between hot core materials and CANDU structures. Survivability assessments to date are based on simple calculations and expert judgements of structural responses to boundary conditions calculated by MAAP-CANDU or published correlations.

The iterative nature of SCD analyses with many independent and dependent variables results in practical challenges to the management of analysis extent, detail and cost. This is a universal contest between the desire for complete understanding and the practical limitations on obtaining this understanding. Probabilistic techniques are employed to aid with sorting out the technical issues. These techniques are not yet mature in dealing with severe accident phenomenology issues. The ingenuity of the human mind (i.e., expertise) is ultimately called upon to remove, or reduce the significance of, SCD issues through focused experiments or SAM provisions.

In the CANDU community, the database of SCD simulations is accumulating. These calculations are sometimes labelled as best-estimate analyses. Expert assessments and interpretations of voluminous results, which are important elements of best-estimate analyses, will synthetise the accumulated information into generalized insights. These insights are the end products of core damage analyses. They recognize, explain and prioritize SCD issues, direct future modelling and test activities, and guide practical improvements to safety of NPPs.

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