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STATUS OF VALIDATION MATRICES FOR CANDU® POWER PLANTS

by

E.O. Moeck[1], J.C. Luxat[2], M.A. Petrilli[3], and P.D. Thompson[4]

[1] Director, Fuel and Thermalhydraulics, Office of the Chief Engineer, Atomic Energy of Canada Limited, Chalk River, Ontario, Canada

[2] Senior Technical Consultant, Reactor Safety and Operational Analysis Department, Ontario Hydro Nuclear, Toronto, Canada

[3] Chef de section Analyse, Centrale nucléaire de Gentilly 2; Région Mauricie, Hydro Québec, Montréal, Québec, Canada

[4] Technical Superintendent, Safety Analysis, Reactor Physics & Fuel, Point Lepreau Generating Station, New Brunswick Power, New Brunswick, Canada

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ABSTRACT

As reported at the 1996 CNS Annual Conference, in mid-1995 the CANDU® industry began to develop validation matrices for CANDU power plants. Of the eight matrices required to address all physical phenomena that could occur in all relevant accident categories, two have been prepared and tabled with the Atomic Energy Control Board, and the remaining six are targeted for submission during 1997. The matrices provide the generic, code-independent knowledge base that will be used to validate major safety analysis codes over the next four years. The unique achievement reported in this paper is the identification and listing of all physical phenomena in all relevant accident categories .

1. INTRODUCTION

Computer codes for the analysis of accidents in CANDU power plants have been in use since the 1960s. With time, many of these codes have been revised and improved and some new ones have been written, to capture greater detail and/or new information from research laboratories and operating plants. To meet today's quality assurance standards, such codes, often referred to as 'scientific computer codes', must be qualified and used according to defined procedures.

Qualification of Scientific Computer Codes

The Canadian approach to code qualification covers several elements in a broadly based, integrated approach. The main elements include:

- a review of codes in current use, to target those that are to be used for the long term;
- a review and identification of safety analysis function needs, including future needs;
- the development of code migration plans to arrive, as far as possible, at a set of industry standard tools for safety analysis;

and for the targeted codes,

- an assessment of their current level of qualification,
- development of verification and validation plans for their further qualification,

- execution of verification plans,
- development of a knowledge base (validation matrices) for their systematic validation,
- execution of validation plans, and
- documentation of the verification and validation work.
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This paper provides an update on the development of the knowledge base and briefly mentions some of the code validation plans. The other elements are being addressed separately by the individual organizations, although the Industry Standard Toolset initiative, currently under way, provides an opportunity to join forces on some elements. The industry's target date for completion of the validation program is late 2000/early 2001.

<u>Validation Matrices</u>

The validation aspect can be considered in two phases: the generic, i.e. knowledge-based, code independent component, and a code-specific component. The Nuclear Energy Agency (NEA) of the Organization for Economic Co-operation and Development (OECD) developed a methodology for addressing the generic component for Light Water Reactors[1]. It is based on a 'validation matrix' that has two tables. The first identifies physical phenomena that could occur in the specified accident categories. The second identifies data sets that exhibit the physical phenomena and could be used to validate specific codes. The OECD/NEA produced a validation matrix for system thermalhydraulics of pressurized water and boiling water reactors[1], and it is currently working on a State-of-the Art-Report (SOAR) on Containment Thermalhydraulics and Hydrogen Distribution[2], which is proposed to include a sample matrix for containment behaviour phenomena under a PWR severe accident scenario. AECL is an active participant in the development of the SOAR, as the lead author for a main chapter on Recent Experimental Activities (Chapter 4).

In mid-1995, the Canadian CANDU industry, comprising Atomic Energy of Canada Limited (AECL), Hydro Quebec (HQ), Ontario Hydro Nuclear (OHN), and New Brunswick Power (NBP), decided to adopt the principles of the validation-matrix methodology and adapt them to CANDU power plants, to address all aspects of its safety analysis, not just system thermalhydraulics and containment. In particular, the industry chose eight scientific disciplines to cover the entire safety analysis:

(i) System Thermalhydraulics;
(ii) Fuel and Fuel Channel Thermal-mechanical Behaviour;
(iii) Fission Product Release and Transport;
(iv) Containment Behaviour;
(v) Reactor Physics*;
(vi) Radiation Physics,
(vii) Atmospheric Dispersion; and
(viii) Moderator and Shield System Thermalhydraulics.

To manage and perform the work, the Canadian CANDU industry decided to create an Industry Validation Team. The Team comprises a Steering Group of eight senior managers, to co-ordinate the overall effort, and 11 Working Groups and a sub-group, currently of ~90 specialists and technical managers, to develop the validation matrices, develop a technical basis, address uncertainties in code predictions, and develop the knowledge base for small reactors. The lead Working Group, on System Thermalhydraulics, has developed its validation matrix, which was the example used in the 1996 CNS paper on the industry-wide validation effort[3]. Since then, the Working Group on Fuel and Fuel Channel Thermal-mechanical Behaviour has also produced its validation matrix. The other Working Groups have developed, as a minimum, their lists of accident categories and physical phenomena, covering all aspects of CANDU safety analysis. To the authors' knowledge, this is a unique achievement for any nuclear reactor. The lists are the principal subject of this paper, and progress is reported on the identification of data sets and documentation of all aspects of generic validation. Future plans in this multi-year, industry-wide code qualification program are also addressed briefly.

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* In the 1996 CNS paper[3], Reactor Physics, Radiation Physics, and Atmospheric Dispersion were shown as Sub-groups of Physics. In reality, specialists in these three areas have been working autonomously.

Definition of Phenomenon

Webster defines a phenomenon as - *Any event, circumstance, or experience that is apparent to the senses and that can be scientifically described or appraised.* This definition is difficult to apply in the present context, and therefore the following working definition was used[4].

A phenomenon is an event or circumstance that:

a) characterizes the process of changing the physical state of a system, and

b) is either directly apparent to the senses or is indirectly apparent by means of measurements of the physical state of a system.

With this definition as a "filter", all phenomena relevant to the eight scientific disciplines were compiled. The definition was followed rigorously, to prevent confusion with properties, mechanisms, behaviours, mathematical correlations, effects, etc. Thus, for example, drift flux in two-phase flow is a mathematical representation of different phase velocities, not a physical phenomenon. Phase separation is the appropriate phenomenon for this example.

The relevant accident categories and physical phenomena are presented in Lists 1 to 17.

2. TECHNICAL BASIS DOCUMENT

The Technical Basis Document provides the overall 'road map' to the validation-matrix methodology. It identifies the accident categories, and for each accident category, the safety concerns, behaviours of systems and radionuclides, and main physical phenomena, as described in more detail in Reference 3. The Technical Basis Document is being written, and its target completion date is the end of 1997. The table of contents has been drafted and is shown in the Attachment. Section 1, the large loss-of-coolant accident (LOCA), has been documented and reviewed[5] , and it is being used as a model for the production of the remaining sections. A lengthy excerpt from section 1 is shown in the Attachment, to illustrate the descriptive style adopted for this document.

3. SYSTEM THERMALHYDRAULICS

The validation matrix in System Thermalhydraulics was on hand in 1995 December and was used to illustrate the methodology adopted for the industry-wide validation work[3]. For completeness, Lists 1 and 2 are presented here, showing the relevant accident categories and the physical phenomena, respectively[4]. The next steps in the validation methodology, namely code-specific validation plans, validation exercises, and validation manuals, are currently being developed and executed for the two-fluid systems codes CATHENA and TUF. The former is being used by AECL, HQ, and NBP, and the latter by OHN. This part of the code qualification program is tentatively scheduled for completion by late 2000/early 2001.

4. FUEL AND FUEL CHANNEL THERMAL-MECHANICAL BEHAVIOUR

The Working Group on Fuel and Fuel Channel Thermal-mechanical Behaviour has submitted revision 0 of its validation-matrix report to the Atomic Energy Control Board in 1996 December. The report identifies 23 physical phenomena that could occur in eight accident categories, Lists 4 and 3. The phenomena are ranked for one of them, the large LOCA. The data sets include: 19 accidents in reactors, one analytical solution, 5 cross-code comparisons, 33 out-reactor integrated tests, 49 in-reactor tests, and 55 separate-effects tests. All phenomena synopses and most of the data set synopses have been produced, and drafting of the remaining ones is under way.

4.1 Fuel Channel Thermalhydraulics

A Sub-group on Fuel Channel Thermalhydraulics has identified 20 physical phenomena in seven accident categories, Lists 17 and 16, and produced short descriptions of the phenomena. The Sub-group has updated its phenomenon/ accident table, draft ranked the phenomena, and preliminarily identified relevant experiments. The work of the Sub-group is being reformatted so that it can be integrated with future revisions of the validation matrices in System Thermalhydraulics and in Fuel and Fuel Channel Thermal-mechanical Behaviour.

5. FISSION PRODUCT RELEASE AND TRANSPORT

The Working Group on Fission Product Release and Transport has identified 19 physical phenomena in the sub-discipline of fission product release and 23 in fission product transport, List 6, and produced synopses of all of them. The relevant accident categories are shown in List 5. The Working Group has also identified 120 data sets and produced synopses of them. Their validation-matrix report is in the final stage of industry review and approval. This Working Group was the first to adopt the Microsoft relational data base ACCESS for their work and used it to great advantage in the course of their peer review and resolution of comments. They are now also in an excellent position to automatically manage revisions to, and control the configuration of their validation matrix. In addition to facilitating review and production of the 1200 page document, the ACCESS data base proved to be very space efficient in terms of storage. The single-file data base is less than three megabytes in size, and following compression, will fit on a single '3.5 inch' floppy diskette. Since the Industry Validation Team decided in 1997 April to eventually convert all matrices to ACCESS format, automatic conversion macros are being developed in MS Word 6.0. In addition, support of tables, figures, and other graphics is being actively explored.

The contributions from the Working Group to the Technical Basis Document are being drafted and reviewed, with a target completion date of mid-1997.

6. CONTAINMENT BEHAVIOUR

The Working Group on Containment Behaviour has identified 10 physical phenomena in the sub-discipline of containment thermalhydraulics, nine in hydrogen behaviour, seven in iodine chemistry, and 16 in aerosol behaviour, List 8. Combinations of these phenomena could occur in seven accident categories, List 7. Because of the multi-disciplinary nature of containment analysis, the list is divided into four sub-disciplines that have traditionally used different analysis codes. These sub-disciplines are Thermalhydraulics, Hydrogen Behavior, Iodine Chemistry, and Aerosol Behavior. Fission products other than iodine appear as aerosols in containment and are treated under the aerosol behavior sub-discipline. The Working Group has also identified seven numerical/analytical tests, 25 separate effects tests, and 17 integrated effects tests. The experimental database available for use in the validation of CANDU containment codes encompasses experiments and test facilities from around the world. Some of the tests were designed to be CANDU specific, while most are used worldwide for generic containment code validation. Synopses of phenomena and data sets and the contribution to the Technical Basis Document are being drafted. The target date for the completion of revision 0 of the validation-matrix report is mid-1997.

7. REACTOR PHYSICS

The Working Group on Reactor Physics has identified 16 physical phenomena that could occur in 15 accident categories, Lists 10 and 9. All phenomena synopses have been written and reviewed, and synopses of data sets from experiments in research reactors (primarily ZED-2 and NRU at the Chalk River Laboratories) and commissioning tests in Canadian CANDU reactors have been drafted. The Working Group was the second to adopt ACCESS for the production and configuration management of its validation-matrix report, which has been assembled and sent for industry review. The target date for its submission to the AECB is mid-1997.

8. RADIATION PHYSICS

The Working Group on Radiation Physics has identified 10 physical phenomena in five accident categories, Lists 12 and 11, and is currently drafting synopses of the phenomena. The target date for the completion of revision 0 of the validation-matrix report is the end of 1997.

9. ATMOSPHERIC DISPERSION

The Working Group on Atmospheric Dispersion has identified 15 physical phenomena, List 13, that need to be considered in the calculation of radiation doses to humans exposed to radioactive emissions, and has drafted synopses of the phenomena. Many phenomena related to atmospheric dispersion are independent of the accident that led to the release. The relative importance of other phenomena has been found to be more closely related to containment response rather than accident type. Containment response is itself dependent on the containment design concept, for example, whether a negative-pressure or positive-pressure design is employed. The final form of the atmospheric dispersion matrix is expected to reflect these considerations. The target date for the completion of revision 0 of the validation-matrix report is the end of 1997.

10. MODERATOR AND SHIELD SYSTEM THERMALHYDRAULICS

The Working Group on Moderator and Shield System Thermalhydraulics has identified 19 physical phenomena that could occur in 15 accident categories, Lists 15 and 14, and has drafted all phenomena synopses. The Working Group has also produced flow charts of safety concerns, behaviours, and phenomena that will be useful in the preparation of their contribution to the Technical Basis Document. They are presently preparing data set synopses. The target date for the completion of revision 0 of the validation-matrix report is the end of 1997.

11. SMALL REACTORS

While the main focus of the Industry Validation Team is the generic validation of computer codes for CANDU analyses, the Team also has a Working Group on Small Reactors which is developing a Technical Basis Document and validation matrices for pool reactors, principally those of the MAPLE family. Typically, the documents being produced by that Working Group are addenda to the documents arising from the work of their CANDU colleagues.

12. UNCERTAINTY ANALYSIS

A Working Group on Uncertainties in Code Predictions is developing practical methodologies that promise to be broadly applicable to the estimation of uncertainties in key outputs from safety analysis codes. Because of its exploratory nature, that work has not been planned in detail as yet and is expected to continue several years into the future.

13. LESSONS LEARNED

The Industry Validation Team first met in mid-1995 as a small group of senior managers from AECL, HQ, OHN, and NBP, with a common interest: to address systematic validation of major computer codes used in safety analyses of CANDU power plants. The group quickly realized that a large, industry-wide effort was required and that flexible collaboration arrangements were desirable, to maximize productive deployment of scare resources. The number of partcipants in the activity grew to ~100, most of whom are senior specialists and technical managers in their respective disciplines and some of whom have been assigned full time to this work. As the work progressed, some working protocols were adopted, and decisions were made, usually by consensus, to achieve as high a degree of uniformity as possible in the end products, i.e., the validation matrices. Some of the main lessons learned from the effort to date are itemized briefly below.

Organizational Aspects

- The Industry Validation Team adopted Terms of Reference and Working Protocols, to define roles and working relationships among the Steering Group, Working Groups, line managers in the participating organizations, and the regulator, i.e., the Atomic Energy Control Board, for communications, interactions, and reporting requirements. Working Groups were given a large degree of autonomy in defining their mode of operation, choosing their members, assigning responsibilities to their members, etc. Generic validation-matrix reports were seen as the end product of the industry-wide effort, and once these were on hand, the Industry Validation Team would have fulfilled its mandate. Subsequent validation of individual computer codes was seen as the responsibility of each participating organization. This flexible organizational structure and work by consensus have been, and continue to be highly effective in developing validation matrices in parallel on a short schedule.

- It was agreed that the Industry Validation Team would have no official status vis-a-vis the regulator, but would provide information and be available for informal discussions. Formal commitments and official submissions would continue to be the prerogative of the participating organizations, via existing communication channels.

- As the generic validation work is approaching completion, executive line management of the industry has recognized that the Industry Validation Team has become a valuable resource and should not disband, once it has completed its generic validation matrices. The Team is in the best position to provide continued leadership on validation activities. Thus, the Steering Group has been given the mandate to lead the process of selecting an Industry Standard Toolset, for safety analyses of CANDU power plants. The intent is to choose appropriate computer codes for use by the industry and to focus further development effort on them, including validation. To date, five Working Groups under the Industry Standard Toolset initiative have been formed and charged with examining in detail specific computer codes in their respective disciplines, with a view of recommending standard sets. Some successes have been achieved already, and prospects are good for consensus on additional codes. However, it is likely that several separate codes will remain in use in the industry.

<u>Validation-Matrix Completeness and Interfaces</u>

- As the Working Groups identified their respective lists of accident categories, physical phenomena, and data sets, it became important to ensure completeness, avoid duplication of effort, and use common definitions. One senior analyst was given the responsibility of collecting these draft lists, reviewing them, and assigning responsibility to a lead Working Group for the definition of each phenomenon that was common to one or more Working Groups. The Working Groups themselves were charged with reviewing draft lists and synopses of 'adjacent' Groups, to ensure consistency in the usage of overlapping accident categories, phenomena, and data sets. The Working Group on the Technical Basis Document was assigned overall responsibility for co-ordinating inputs from the other Working Groups into that document. The success to date of these interactions is apparent from the completed lists of accident categories and phenomena given in Lists 1 to 17. Detailed phenomena and data set synopses, too voluminous to be reproduced here, provide specific cross-links within and among the validation matrices.

- Development of validation-matrices was, and continues to be a learning experience for all participants. As in any first-of-its-kind endeavour, the developers and reviewers, including AECB staff, identified improvements that could be made. Rather than expending resources on successive iterations and improvements, the Industry Validation Team decided to complete the entire validation cycle. Thus, upon completion of the initial validation matrix in each discipline, effort and priority was, and is given to producing code specific validation plans, exercises, and validation manuals.

<u>Configuration Management</u>

- Each validation-matrix report captures a large volume of information that is written, assembled, and reviewed by many specialists over a period of time of a year or two. Such an endeavour naturally raises issues of resolution of comments, version control, and overall configuration management. The lead Working Groups managed these issues as they arose and produced revision 0 (and in one instance revision 1) reports. The Working Group on Fission Product Release and Transport spearheaded a radically different approach. Part way into its validation-matrix development work, the Group decided to adopt the Microsoft relational data base ACCESS, to convert the existing records into it, and to complete the remaining work in ACCESS. This decision turned out to be a resounding success. The Group executed the conversion in a very short time and reaped downstream benefits during the review and record keeping stages. ACCESS lends itself naturally to auditable resolution of comments, version control, and configuration management. Individual records and linkages among them are entered once, and thereafter the data base keeps track of them. A custodian keeps the master version and controls revisions. All these features should make it easier for the regualtor to review the validation-matrix document.

On the basis of the excellent experience described above, the Industry Validation Team decided to convert all validation-matrix reports to ACCESS, with the timing of that conversion left to the Working Groups.

- As part of the process of generating the validation matrices, unique identifiers have been assigned to phenomena and data. In some instances, as the work progressed, it became apparent that some phenomena or data needed to be removed. Instead of changing the identifiers on all subsequent phenomena or data and searching for cross-references in the entire set of documents to make corrections, it was decided to leave gaps in the sequence of identifiers. Thus, for example, in List 12, there is no phenomenon RAD10. At some future point, when all validation matrices are in the ACCESS data base, it would be relatively easy to re-number phenomena and data to remove gaps.

Data Sets

- The issue of qualification of data sets shown in validation matrices was resolved as follows. The matrix developers need to satisfy themselves, via inspection of the data and a preliminatry qualification of them, that they may be suitable for code validation. A more detailed qualification of data selected for the validation of a specific computer code is to be performed for the code validation plan.
- During the search for data sets, some Working Groups have identified data that are known to exist but are not readily available to the Group, mainly because they are owned by other organizations. It was agreed that such data sets would not be shown in the validation matrices, although their existence would be acknowledged in working documents, such as minutes of meetings, to provide a trail to show that the Working Group was aware of the data. If such data are 'more of the same', i.e. do not add significantly to the available data sets, then their omission is no great loss. If such data are unique or the only existing experimental information, then the Steering Group decides on the most appropriate way of addressing them.
- In some data searches, data have been identified that are, or could become unavailable because of neglect, i.e. because they are about to be abandoned or otherwise destroyed. Typically, such data are old, difficult to access with today's electronic technology, and would require an investment of expert staff time to make them readily available. Working Parties of the CANDU Owners Group (COG) already have a mandate and action to identify such data and to preserve them.
- Some data sets have been identified that are not directly applicable to the phenomena of interest, for example because they lie outside the range of CANDU analyses. The Industry Validation Team decided that, if data within the range are available, then there is no need to include data outside the range. If not, then data outside the range should be included, provided that they exhibit the phenomena of interest. The phenomenon description is to address this issue under 'State of Knowledge and Uncertainties'. A given code should be validated with best available data, even when outside the range of interest.
- Overlapping data sets present no problem and are shown in the validation matrices. During the code-specific validation stage, data are selected and that selection is described in the code validation plan.
- When Working Groups identify data gaps in validation matrices, they use the COG process to set priorities for new work, call for proposals, and invite R&D proponents to respond.

14. SUMMARY AND CONCLUSIONS

In summary, the Industry Validation Team is on track in its program to develop a generic knowledge foundation, based on validation matrices, for the validation of computer codes used in safety analyses of CANDU plants. The Team has ~100 participants from the Canadian CANDU industry, comprising Atomic Energy of Canada Limited, Hydro Quebec, Ontario Hydro Nuclear, and New Brunswick Power, organized into a Steering Group of eight senior managers and 11 Working Groups and a Sub-group of technical specialists and technical managers. Two of eight validation matrices have been submitted already to the regulator, the Atomic Energy Control Board, and the remaining six are targeted for completion during 1997. The 'road map' for the validation matrices, i.e., a single

Technical Basis Document, is also being drafted and targeted for completion before the end of 1997. Based on this generic foundation, code-specific validation plans are being developed and executed by the individual industry organizations, with a target completion date of late 2000/early 2001.

Code validation is one element of a broadly based, integrated program of code qualification undertaken by the individual industry organizations and targeted for completion by late 2000/early 2001.

15. ACKNOWLEDGMENT

The authors have assumed the role of Rapporteur for work done by others, namely, the ~90 specialists and technical managers throughout Atomic Energy of Canada Limited, Ontario Hydro Nuclear, Hydro Quebec, and New Bruswick Power who have, and are developing the validation matrices with great dedication and enthusiasm. Their efforts are hereby acknowledged.

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List 1: Accident Categories Relevant to CANDU System Thermalhydraulics

Large LOCA
- Power Pulse/Reactor Trip
- Early Blowdown Cooling
- Late Blowdown Cooling/Emergency Coolant Injection
- Refill

Large LOCA/LOECI
- Power Pulse/Reactor Trip
- Early Blowdown Cooling
- Steam Cooling/Heat Rejection To Moderator

Small LOCA
- Depressurization
- Reactor Trip
- ECI
- Refill

Loss of Flow
- Loss of Class IV Power - Pump Rundown
- Two-Phase Thermosiphoning
- Intermittent-Boiling-Induced Flow

Loss of Regulation
- Power Increase/Reactor Trip
- Fuel Channel Quench

Loss of Feedwater
- PHTS Pressurization/Reactor Trip
- Long Term Cooling

Steam Line Breaks

Steam Generator Blowdown
Reactor Trip
Loss of Class IV Power
ECI

List 2: Physical Phenomena Relevant to CANDU System Thermalhydraulics

ID	Phenomenon
TH1	Break Discharge Characteristics and Critical Flow
TH2	Coolant Voiding
TH3	Phase Separation
TH4	Level Swell and Void Holdup
TH5	Heat-Transport Pump Characteristics (Single-and Two-Phase)
TH6	Thermal Conduction
TH7	Convective Heat Transfer
TH8	Nucleate Boiling Heat Transfer
TH9	CHF and Post-Dryout Heat Transfer
TH10	Condensation Heat Transfer
TH11	Radiative Heat Transfer
TH12	Quench/Rewet Characteristics
TH13	Zirconium/Water Thermal-Chemical Reaction
TH14	Reflux Condensation
TH15	Counter-Current Flow
TH16	Flow Oscillations
TH17	Density Driven Flows: Natural Circulation
TH18	Fuel Channel Deformation
TH19	Fuel String Mechanical-Hydraulic Interaction
TH20	Waterhammer
TH21	Waterhammer: Steam Condensation Induced
TH22	Pipe Thrust and Jet Impingement
TH23	Non-Condensable Gas Effect

List 3: Accident Categories Relevant to CANDU Fuel and Fuel Channel Thermal-mechanical Behaviour

Large LOCA
Large LOCA/LOECI
Small LOCA
End fitting failure
Stagnation Feeder Break
Flow Blockage
Fuel Handling Accidents
Loss of Flow
Loss of Regulation

List 4: Physical Phenomena Relevant to CANDU Fuel and Fuel Channel Thermal-mechanical Behaviour

ID	Phenomenon
FC1	Fission and Decay Heating
FC2	Heat Diffusivity in Fuel
FC3	Fuel-to-Sheath Heat Transfer
FC4	Fuel-to-End-Cap Heat Transfer
FC5	Fission Gas Release to Gap and Pressurization
FC6	Sheath Deformation
FC7	Sheath Failure
FC8	Fuel Deformation
FC9	Sheath Oxidation/Hydriding
FC10	Fuel Oxidation/Reduction
FC11	Fuel, Sheath Melting and Relocation
FC12	Bundle Mechanical Deformation
FC13	Sheath-to-Coolant and Coolant-to-Pressure Tube Heat Transfer
FC14	Flow Mixing and Bypass
FC15	Local Melt Heat Transfer to Pressure Tube
FC16	Pressure Tube to Calandria Tube Heat Transfer
FC17	Calandria Tube to Moderator Heat Transfer
FC18	Pressure Tube Deformation and Failure
FC19	Calandria Tube Deformation and Failure
FC20	Pressure Tube Oxidation and Hydriding
FC21	Element/Pressure Tube Radiative Heat
FC22	Element/Bearing Pad/Pressure Tube Contact Heat Transfer
FC23	Failed Channel Interaction With Core Components

List 5: Accident Categories Relevant to CANDU Fission Product Release and Transport

Large LOCA
Small LOCA
End Fitting Failure
Stagnation Feeder Break
Flow Blockage
Large LOCA/LOECI
Secondary Side Breaks
Fuel Handling Accidents

List 6: Physical Phenomena Relevant to CANDU Fission Product Release and Transport

ID	Phenomenon
	Fission Product Release

FPR-1 Athermal Release
FPR-2 Diffusion
FPR-3 Grain Boundary Sweeping/Grain Growth
FPR-4 Grain Boundary Coalescence/Tunnel Interlinkage
FPR-5 Vapor Transport/Columnar Grains
FPR-6 Fuel Cracking (Thermal)
FPR-7 Gap Transport (Failed Elements)
FPR-8 Gap Retention
FPR-9 UO2+x Formation
FPR-10 U_4O_9 - U_3O_8 Formation
FPR-11 UO_{2-x} Formation
FPR-12 UO_2 Zircaloy Interaction
FPR-13 UO_2 Dissolution by Molten Zircaloy
FPR-14 Fuel Melting
FPR-15 Fission Product Vaporization/Volatilization
FPR-16 Matrix Stripping
FPR-17 Temperature Transients
FPR-18 Grain Boundary Separation
FPR-19 Fission Product Leaching

Fission Product Transport

FPT-1 Fuel Particulate Suspension
FPT-2 Vapour Deposition and Re-vaporization of Deposits
FPT-3 Vapour/Structure Interaction
FPT-4 Aerosol Nucleation
FPT-5 Gravitational Agglomeration in the Primary Heat Transport System (PHTS)
FPT-6 Brownian Motion (Diffusional) Agglomeration in PHTS
FPT-7 Turbulent Agglomeration in PHTS
FPT-8 Laminar Agglomeration
FPT-9 Electrostatic Agglomeration
FPT-10 Aerosol Growth/Revapourization
FPT-11 Thermophoretic Deposition in PHTS
FPT-12 Diffusiophoretic Deposition
FPT-13 Gravitational Deposition
FPT-14 Brownian Motion Deposition
FPT-15 Turbulent Deposition in PHTS
FPT-16 Laminar Deposition
FPT-17 Electrostatic Deposition
FPT-18 Inertial Deposition
FPT-19 Photophoretic Deposition
FPT-20 Aerosol Resuspension
FPT-21 Pool Scrubbing
FPT-22 Transport of Deposits by Water
FPT-23 Chemical Speciation
FPT-24 Transport of Structural Materials

List 7: Accident Categories Relevant to CANDU Containment Behaviour

Large LOCA
Small LOCA
Pipe Breaks
In-Core Breaks
Large LOCA/LOECI
Secondary Side Breaks
Fuel Handling Accidents
Auxiliary System Failures

List 8: Physical Phenomena Relevant to CANDU Containment Behaviour

ID Phenomenon

Thermalhydraulics

C1 Flashing Discharge
C2 Evaporation from Pools
C3 Convection Heat Transfer
C4 Conduction Heat Transfer
C5 Condensation Heat Transfer
C6 Air Cooler Heat Transfer
C7 Heat Removal by Dousing Water
C8 Laminar/Turbulent Leakage Flow
C9 Choked Flow through Pressure Reducing Valves
C10 Liquid Re-entrainment

Hydrogen Behaviour

C11 Buoyancy Induced Mixing
C12 Jet Momentum Induced Mixing
C13 Hydrogen Stratification
C14 Hydrogen Deflagration
C15 Flame Acceleration
C16 Flame Quenching by Turbulence
C17 Standing Flame
C18 Deflagration Detonation Transition
C19 Mixing and Removal by Recombiners

Iodine Chemistry

C21 Interfacial Mass Transfer
C22 Partition Coefficient
C23 Adsorption
C24 Carbon Filter Removal Efficiency
C25 Total Waterborne Iodine
C26 Fraction Airborne Organic Iodine
C27 Total Airborne Iodine

Aerosol Behaviour

C28 Jet Impingement
C29 Plateout (Gravitational Settling)
C30 Thermophoresis
C31 Diffusiophoresis
C32 Diffusional Agglomeration
C33 Removal in HEPA Filters
C34 Removal in Demisters
C35 Removal in Leakage Paths
C36 Condensation
C37 Evaporation
C38 Turbulent Agglomeration

C39 Turbulent Deposition
C40 Formation in a Flashing Jet
C41 Formation in a Steam Jet
C42 Gravitational Agglomeration
C43 Inertial Deposition

List 9: Accident Categories Relevant to CANDU Reactor Physics

Large LOCA
Emergency Coolant Injection and Class IV Power Intact
Loss of Emergency Coolant Injection
Loss of Class IV Power
Transition Break LOCA
Small Out-of-Core LOCA
Small In-Core LOCA
Pressure Tube/Calandria Tube Failure
Stagnation Feeder Break
End-Fitting Failure
Loss of Flow
Loss of Regulation
Slow
Fast
Loss of Feedwater
Steam Line Break
Moderator System
Loss of Moderator Inventory
Loss of Moderator Heat Sink

List 10: Physical Phenomena Relevant to CANDU Reactor Physics

ID	Phenomenon
PH1	Coolant-Density-Change Induced Reactivity
PH2	Coolant-Temperature-Change Induced Reactivity
PH3	Moderator-Density-Change Induced Reactivity
PH4	Moderator-Temperature-Change Induced Reactivity
PH5	Moderator-Poison-Concentration-Change Induced Reactivity
PH6	Moderator-Purity-Change Induced Reactivity
PH7	Fuel-Temperature-Change Induced Reactivity
PH8	Fuel-Isotopic-Composition-Change Induced Reactivity
PH9	Refuelling-Induced Reactivity
PH10	Fuel-String-Relocation Induced Reactivity
PH11	Device-Movement Induced Reactivity
PH12	Prompt/Delayed Neutron Kinetics
PH13	Flux-Detector Response
PH14	Flux And Power Distribution (Prompt/Decay Heat) in Space and Time
PH15	Lattice-Geometry Reactivity Effects
PH16	Coolant-Purity-Change Induced Reactivity

List 11: Accident CategoriesRelevant to CANDU Radiation Physics

Large LOCA
Fuel Channel Decay Heat
Moderator Heat Load
Containment Activity Monitor
Small LOCA
End Fitting Failure
Nuclear Criticality
Inadvertent Nuclear Excursion

List 12: Physical Phenomena Relevant to CANDU Radiation Physics

ID	Phenomenon
RAD1	Radiation Emission
RAD2	Isotope Generation and Depletion
RAD3	Neutron Transport and Streaming
RAD4	Photon Transport, Streaming and Skyshine
RAD5	Electron Transport
RAD6	Heating
RAD7	Internal and External Exposure
RAD8	Radiolysis
RAD9	Damage
RAD11	Criticality and Sub-Critical Multiplication

List 13: Physical Phenomena Relevant to Atmospheric Dispersion from CANDU Plants

ID	Phenomenon
AD-01	Plume Rise
AD-03	Downwash
AD-04	Modification of Effective Release Height Due to Building Entrainment
AD-05	Plume Broadening Due to Building Entrainment
AD-06	Fumigation
AD-07	Formation of the Thermal Internal Boundary Layer
AD-08	Reflection from an Elevated Inversion
AD-09	Plume Advection
AD-10	Plume Diffusion
AD-11	Wet Deposition
AD-12	Dry Deposition
AD-13	Plume Depletion
AD-14	Exposure to Cloudshine

AD-15 Exposure to Groundshine
AD-16 Internal Exposure due to Inhalation

List 14: Accident Categories Relevant to CANDU Moderator and Shield System Thermalhydraulics

Loss of Moderator Heat Sink
Loss of Moderator Inventory
Loss of Moderator Temperature Control Low
Loss of Shield Tank/End Shield Inventory
Loss of Shield Tank Temperature Control Low
Loss of Shield Cooling
Small LOCA
- In-Core Breaks
- In-Core Breaks from a Guaranteed Shutdown State
- Out-of-Core Breaks

Small LOCA/LOECI
- In-Core Breaks

Large LOCA
Large LOCA/LOECI
Secondary Side Breaks
Loss of Flow
Loss of Regulation

List 15: Physical Phenomena Relevant to CANDU Moderator and Shield System Thermalhydraulics

ID	Phenomenon
MH3	Moderator Degassing
MH4	Mass and Energy Transfer in Moderator Cover Gas
MH9	Moderator Pump Cavitation
MH10	Interaction of Moderator Flow with Calandria Tubes
MH11	Moderator Flow Turbulence
MH12	Moderator Buoyancy
MH13	Moderator Inlet Jet Development
MH15	Displacement of Poison from Containers
MH16	Injection of Poison along Nozzles
MH19	Moderator/Coolant/Poison Mixing
MH22	Calandria Tube/Moderator Heat Transfer
MH30	Failed Channel Interaction with Core Components
MH34	Hydrogen Deflagration
MH39	Moderator Heat Exchanger Response
MH41	Liquid, Vapor and Two-Phase Discharge
MH42	Moderator Swell
MH43	Thermal Conduction
MH44	Convective Heat Transfer
MH45	Radiative Heat Transfer

List 16: Accident Categories Relevant to CANDU Fuel Channel Thermalhydraulics

Large LOCA
Large LOCA/LOECI
Small LOCA
- Pressure Tube Failure, Calandria Tube Intact
- In-Core Breaks
- Out-of-Core Breaks

Loss of Flow
Loss of Regulation

List 17: Physical Phenomena Relevant to CANDU Fuel Channel Thermalhydraulics

ID	Phenomenon
FCT1	Convective Heat Transfer
FCT2	Onset of Vapor/Void Generation
FCT3	Pre-Critical Heat Flux (CHF) Boiling Heat Transfer
FCT4	Dryout (CHF)
FCT5	Transition and Film Boiling
FCT6	Quench and Rewet
FCT7	Inter-Subchannel Single- and Two-Phase Mixing
FCT8	Inter-Subchannel Turbulent Flow Scattering
FCT9	Inter-Subchannel Diversion Cross-Flow
FCT10	Phase Separation
FCT11	Single-Phase and Two-Phase Density-Driven Flow
FCT12	Single-Phase and Two-Phase Wall Shear and Form Losses
FCT13	Radiative Heat Transfer
FCT14	Steady-State and Transient Heat Conduction (Heat Diffusivity)
FCT15	Non-Condensable Gas Effect
FCT16	Zirconium/Steam and Zirconium/Air Thermal-Chemical Reaction
FCT17	Fuel and Channel Deformation
FCT18	Counter-Current Flow
FCT19	Waterhammer
FCT20	Flow Oscillations

ATTACHMENT

Excerpt from the Technical Basis Document[5]

Table of Contents

SECTION 1

TECHNICAL BASIS OF LARGE LOCA ANALYSES

1. INTRODUCTION

A large Loss of Coolant Accident (LOCA) involves a break in the heat transport system pressure boundary of sufficient magnitude that the normally operating reactivity control system, RRS, is incapable of maintaining reactivity balance and, as a result of the coolant void reactivity feedback, an immediate reactor power excursion occurs.

A large LOCA is characterized by the following general features:

1. an immediate power excursion driven by rapid coolant voiding in many channels,

2. a large rate of coolant discharge from the break into containment,

3. the potential for early impairment of fuel cooling, leading to possible pressure tube deformation,

4. the potential for fuel failures

5. a spike of iodine release from previously defected fuel into the coolant during the blowdown period

6. a potential increase in heat load to the moderator

7. the Emergency Coolant Injection System (ECIS) is available and coolant injection occurs.

8. an overpressure period in containment during which there can be a pressure driven release from containment.

The range of break sizes that are encompassed includes ones for which:

- the channels in the affected flow pass experience reduced flow in the normal flow direction (less than critical break size),
- channels in the affected core pass experience early, rapid reduction in flow to very low levels which are sustained for a limited duration (critical break size), and
- channels in the affected core pass experience sustained reverse flow during the blowdown (greater than critical break size).

2 KEY SAFETY CONCERNS

The safety concerns of relevance to large LOCA events whose consequences are quantified through the safety analysis are:

- public and in-plant dose related to fission product releases from the fuel,
- core coolable geometry related to fuel channel integrity, and
- containment integrity related to pressurization and hydrogen combustion.

3 ACCIDENT BEHAVIOUR

Quantification of the consequences associated with these safety concerns involves analysis of phenomena which can be grouped into sets of behaviour characterizing the physical processes that come into play during a large LOCA. These groups of behaviour typically evolve over limited time periods and proceed either in parallel with one another, or in a specific order determined by external sequences of events such as shutdown system initiation and ECIS initiation. For example, the early stages of blowdown cooling and the neutronic overpower transient evolve as parallel and inter-related behaviour, with the neutronic overpower transient behaviour occurring over a shorter time duration than blowdown cooling; whereas, ECIS delivery behaviour develops some tens of seconds following the neutronic overpower transient and the initial ECIS delivery proceeds in parallel with the later stages of blowdown cooling. Therefore, uncertainties in the modelling the phenomena associated with the different behaviour groupings are of relevance to the safety analysis only during those periods of time in which the behaviours exert a governing influence.

Phases of the Accident

The phases of a LOCA accident are defined according to the major time periods during the accident progression during which characteristic groups of behaviour are exhibited. For each of the major disciplines involved in a large LOCA the following phases are defined and the dominant behaviour during these phases are identified. Note that the time periods for each phase are approximate and do not imply specific limits on the start and end times for a phase.

Reactor Physics

1. *Power Pulse* (0-5 seconds) - the initial period following the break during which the reactor power increases due to positive coolant void feedback and which is terminated by shutdown system action. the dominant behaviour during this period is the *neutronic overpower transient.*

2. *Post shutdown* (5 seconds onwards) - the period following reactor shutdown in which the reactor is brought subcritical, the spatial neutron flux distribution stabilizes and the power distribution becomes governed by decay heat.

System Thermalhydraulics / Fuel & Fuel Channel Thermal Mechanical Behaviour / Fission Product Release

1. *Power Pulse* (0-5 seconds) - the initial period following the break during which the reactor power increases due to positive coolant void feedback and which is terminated by shutdown system action. the dominant behaviours during this period are *heat transport system depressurization, neutronic overpower transient and fuel heatup and axial fuel expansion.*

2. *Early Blowdown Cooling* (5 - 30 seconds) - the period during which the heat transport system blowdown continues prior to ECIS initiation. The dominant behaviours during this period are *heat transport system depressurization, blowdown cooling, fuel deformation, pressure tube deformation, fuel heatup, pressure tube heatup, fuel failure and fission product release.*

3. *Late Blowdown Cooling/ECIS Injection* (30 - 200 seconds) - the period of ongoing heat transport system blowdown with ECIS injection into the heat transport system. The dominant behaviours during this period are *heat transport system depressurization, blowdown cooling, ECIS delivery, fuel deformation, pressure tube deformation, fuel heatup, pressure tube heatup and fission product release.*

4. *Refill* (> 200 seconds) - the period during which refill of channels in the core proceeds and a quasi-steady state is attained. The dominant behaviours during this period are, *ECIS delivery, heat transport system refill, fuel cooling and fission product release.*

VALIDATION OF SOPHAEROS V1.3 CODE BASED ON FALCON EXPERIMENTS

M. Missirlian, G. Lajtha

Institut de Protection et de Sûreté Nucléaire (IPSN)
Département de Recherche en Sécurité (DRS)
C.E.A. de Cadarache, 13108 S^{T}-Paul-Lez-Durance CEDEX, France.

SUMMARY

A key problem in source term evaluation in nuclear reactor severe accidents is determination of the transport behaviour of fission products (f.p.) released from the degrading core. The SOPHAEROS computer code is being developed aiming at predicting in a mechanistic way f.p. transport in LWR circuits. In the first version of the code (1), released in 1994, the main aerosol (coagulation and numerous deposition mechanisms) and fission product vapour (condensation, evaporation and sorption) phenomena were modelled for 12 fission product species. The large number of mass balance equations with non-linear terms were solved using an efficient implicit numerical method.

In the version of the code used in this study, version 1.3 (2), progress has been mainly made with the introduction of vapour-phase chemistry. The basis of the chemical modelling is the calculation of thermodynamic chemical equilibrium of vapour-phase species from elements in the vapour-phase.

Before adding the chemistry module different approaches were considered to determine the fastest numerically robust method. A transient approach based on the deviation from equilibrium was chosen for solution of the equations with 'ideal kinetics'. This means that an artificial mathematical transient is applied to reach the real steady state (equilibrium). The 'kinetics' associated with equilibrium are directly related to the deviation from equilibrium.

The thermodynamic equilibrium assumption means that chemical speciation changes with temperature, carrier gas composition, and the concentrations of vapours. The number of species considered in the calculation can be easily increased or modified by adding data for the new species (name, molar mass, stoechiometric coefficients, equilibrium constant, etc) to the data bank. The default data bank comprises 30 elements and the compound species from these elements.

The SOPHAEROS code has in the past been validated on analytical experiments such as DEVAP, LACE, TRANSAT, TUBA as well as on integral tests of the PHEBUS FP program. In this paper, an example of validation for the fission product vapour transport (chemistry and condensation/evaporation) models is shown. Some tests of the Falcon (3) thermal gradient tube experiments (FAL-17, FAL-18, ISP-1, etc) are used.

The results of the SOPHAEROS 1.3 calculations are compared with the results of the VICTORIA code (4) and the measured data, see Table 1. The input data of the two codes were determined in the same way. It is seen that SOPHAEROS can predict the dominant chemical species and the element retention factors in the silica pipe quite well.

These applications of the SOPHAEROS code to the Falcon experiments, along with others which are not presented here, indicate that the numerical scheme of the code is robust, and no convergence problems are encountered. The calculation is also very fast being 3 times longer on a Sun SPARC 5 workstation than real time and typically about 10 times faster than an identical calculation with the VICTORIA code.

The study demonstrates that the SOPHAEROS 1.3 code is a suitable tool for prediction of vapour chemistry and f.p. transport with a reasonable level of accuracy.

Furthermore, the flexibility of the code « material databank » allows improvement of understanding of fission product transport and deposition in the circuit.

Table 1

Comparison of retention the results for the calculations and measured values

		Cs	I	Mo	Ba	Cd	In	Ag	Te	B	Si
FAL-17	**Experiment**	**45%**	**20%**	**65%**	**56%**	**57%**	**72%**	**76%**	**-**	**24%**	**22%**
	SOPHAEROS	**64%**	**26%**	**62%**	**76%**	**41%**	**61%**	**57%**	**-**	**19%**	**19%**
	VICTORIA	**28%**	**24%**	**27%**	**34%**	**18%**	**30%**	**27%**	**-**	**32%**	**-**
FAL-18	**Experiment**	**55%**	**75%**	**-**	**-**	**31%**	**57%**	**57%**	**21%**		**-**
	SOPHAEROS	**47%**	**29%**	**-**	**-**	**31%**	**58%**	**31%**	**58%**		**-**
	VICTORIA	**18%**	**30%**	**-**	**-**	**18%**	**30%**	**28%**	**29%**		**-**

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