GENERIC VALIDATION OF COMPUTER CODES USED IN SAFETY ANALYSES OF CANDU® POWER PLANTS

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Abstract

Since the 1960s, the CANDU® industry has been developing and using scientific computer codes, validated according to the quality-assurance practices of the day, for designing and analyzing CANDU power plants. To provide a systematic framework for the validation work done to date and planned for the future, the industry has decided to adopt the methodology of validation matrices, similar to that developed by the Nuclear Energy Agency of the Organization for Economic Co-operation and Development for Light Water Reactors (LWR). Specialists in six scientific disciplines are developing the matrices for CANDU plants, and their progress to date is presented.

1. INTRODUCTION

Since the 1960s, the CANDU industry has been engaged in the development and validation of safety-related computer codes. The codes have been used in support of safety analyses of CANDU reactors, and in some instances to assist in the planning and understanding of experimental work done at the laboratories. The focus of the industry's validation approach was to gain knowledge through experimental and theoretical studies and implement that knowledge in mathematical models that are validated, to the extent possible, in separate-effects tests. The models were then installed in computer codes that are tailored to meet current quality assurance practices of reliability and user friendliness, and the codes were validated against integrated tests.

During the fifteen years leading up to 1990, there was an intense effort on code development and validation to support the CANDU reactors in operation and those under development. The task of code validation was supported by an R&D program, presently known as the Safety and Licensing R&D Program of the CANDU Owners Group (COG). The program was jointly funded and reflected the interests that were common to the three Canadian utilities operating CANDU power plants (Ontario Hydro Nuclear (OHN), Hydro Quebec (HQ), and New Brunswick Power (NBP)) and Atomic Energy of Canada Limited (AECL).

Since 1990, the R&D has become more focused on ensuring that code validation is carried out to satisfy both the needs of the industry, for its current design activities and plant operations, and the demands of the regulators. The R&D programs are reviewed both by COG Technical Committees and in-house by AECL. In 1995 June, the industry formed a Code Validation Team, to coordinate code-validation activities in the four partner organizations (OHN, HQ, NBP, and AECL). More recently, the Validation Team has been restructured into a Steering Group and several Working Groups. Building upon work initiated at Ontario Hydro Nuclear, the Team's focus is the generic validation of the major codes used in safety analyses of CANDU reactors in operation and those under development. Generic validation refers to those activities that are code independent and provide the knowledge base necessary for the systematic validation of specific codes, as explained further in Section 3. One of the Team's first outputs was agreement on six main disciplines into which physical phenomena can be grouped conveniently for validation purposes. These disciplines are:

- i) System Thermalhydraulics;
- ii) Fuel and Fuel Channel Thermal-mechanical Behaviour;
- iii) Fission Product Release and Transport;
- iv) Containment Behaviour;
- v) Physics (comprising reactor physics, shielding, and atmospheric dispersion); and
- vi) Moderator and Related Thermalhydraulics.

Working Groups of specialists in each discipline carry out the work. Overviews of the current status of validation activities and planning to date in this multi-year validation program are given below.

2. FORMAL APPROACH TO VALIDATION

While the industry's traditional approach to code validation, as outlined in the Introduction, has been in line with international practice, recent developments domestically and internationally have provided the stimulus for a re-examination. Increasingly, the CANDU industry and its regulators expect computer codes to be formally validated within a systematic framework that can be readily audited. Such a framework exists, and its foundations are validation matrices. The Nuclear Energy Agency of the Organization for Economic Co-operation and Development (OECD/NEA) has recently published [1] validation matrices for LWRs that represent an international consensus in the LWR community on (i) the major, hypothetical accidents, (ii) physical phenomena that might occur during these accidents, (iii) experimental facilities, and (iv) data from separate-effects experiments suitable for the validation of computer codes used in safety analyses and licensing submissions. These matrices address thermalhydraulic phenomena in the primary heat-transport circuit, and for pressurized water reactors, also the secondary heat-transport circuit.

The CANDU industry has decided to utilize the validation-matrix methodology for its validation activities, and to adapt it as necessary, taking into account the state of the art internationally, available expertise, and cost/benefit considerations. Where no international precedents exist, the industry is proceeding with prudence. The steps are typically as follows:

- i) identification of accident scenarios to be analyzed;
- ii) identification and ranking of physical phenomena relevant to these accidents;
- iii) description of the phenomena;
- iv) identification of experiments that exhibit the phenomena;
- v) description of the source facilities/tests; and
- vi) generation of a cross-reference table of phenomena versus relevant experimental data.

The validation matrix comprises the tables in items (ii) and (vi) above.

The industry is examining its suite of safety-analysis codes, with a view to selecting the most appropriate ones for long-term development (if needed), application, and support. The validation matrices will provide the basis upon which to plan further code validation, if needed, to bring code development to closure. The above activities comprise a multi-year validation program, the front end, i.e. generic portion of which is described in the next sections.

3. VALIDATION MATRICES AND THEIR ROLE IN CODE VALIDATION

The validation-matrix methodology has five basic steps, illustrated in Table 1. In the first step, a Technical Basis Document is produced that provides a total overview of all postulated accidents in the design basis of the nuclear plant and the associated main physical phenomena governing the behaviour of plant systems and radionuclides. In the second step, validation matrices are produced for each discipline, relating all relevant physical phenomena to the relevant subset of accidents and to data from experiments, operating plants, mathematical solutions, and benchmark codes. Steps one and two provide the generic knowledge base which is code independent.

Steps three to five are code specific. In step three, a validation plan is produced for each code. The plan identifies validation work that is believed to be necessary to provide sufficient validation of the code for its intended applications. The execution of the plan demonstrates that the code version accurately represents the governing phenomena for each phase of the selected accident scenario. In step four, validation exercises are performed to compare model predictions with selected data sets. Uncertainties in code predictions are estimated. In step five, a validation manual is produced, summarizing code accuracy, sensitivities, and uncertainties for specific applications. The manual addresses the question whether the validation is adequate.

While the validation methodology shows a linear progression through five steps, actual work is being performed in parallel, on steps one and two, and in all six disciplines, to maximize progress on as many fronts as possible and to engage specialists in all disciplines. The Steering Group ensures that the activities are coordinated and that experience gained is shared among participants. The achievements to date and the near-term plans are summarized in the sections below.

3.1 Technical Basis Document

Draft sections of the Technical Basis Document are being produced by specialists in the six disciplines, with some sections being in an advanced state of preparation and undergoing peer review. An example is the technical basis for analyses of large loss-of-coolant accidents (LOCA). The logic of that technical basis is illustrated in Figure 1, which relates the safety concerns, behaviours of plant subsystems and radionuclides, and main physical phenomena. Similar descriptions are being produced for other accidents in the design basis.

3.2 System Thermalhydraulics

A validation matrix for system thermalhydraulics has been developed that is based on the physical phenomena that might occur during accidents which form the design basis of CANDU power plants. Seven accident categories have been identified and addressed. They are: (i) large LOCA, (ii) LOCA with loss of emergency coolant (EC) injection (LOECI), (iii) small LOCA, (iv) loss of flow, (v) loss of regulation, (vi) loss of feedwater, and (vii) steam-line break. For this ensemble of postulated accidents, 23 phenomena have been identified, assigned an identification number from TH1 to TH23, and their relative importance during the different phases of the accidents has been estimated. That work has been summarized in a 23 x 7 matrix, an excerpt of which is illustrated in Table 2. For each of the seven accident scenarios, a table has been produced that divides the accident into a number of phases in the accident progression and identifies primary and secondary phenomena in each phase. Table 3 is an excerpt from the large-LOCA tabulation in which seven primary and three secondary phenomena have been ranked in four significant time phases. Similar rankings have been produced for the other six postulated accidents.

In the next step, relevant available tests, both experimental and numerical, were identified and tabulated. Identification numbers were assigned to separate-effects tests (SE1 to SE25), component tests (CO1 to CO5), integrated tests (INT1 to INT17), and numerical tests (NUM1 to NUM10). An excerpt from this tabulation is illustrated in Table 4. At this point, the quality of the data was not judged; the data were simply identified as being potentially suitable and available for validation purposes. In the next step, the data were reviewed and assessed for suitability for code validation. One of three grades was assigned to each data set as it relates to each of the 23 thermalhydraulic phenomena: (i) not suitable, (ii) suitable for indirect validation, or (iii) suitable for direct validation. An excerpt from this tabulation is illustrated in Table 5.

To complete the generic part of the validation methodology, descriptions have been produced of the: (i) 23 phenomena, (ii) 37 experimental facilities, (iii) 25 separate effects tests, (iv) 5 component tests, (v) 17 integrated tests, and (vi) 10 numerical tests. The validation matrix comprises the two cross-reference tables: phenomena to postulated accident scenarios (illustrated in Table 2) and phenomena to tests (illustrated in Table 5).

Staff from the Atomic Energy Control Board (AECB) examined the validation-matrix document for system thermalhydraulics for CANDU power plants and discussed it informally with industry representatives. The staff's view was that the work done represents a significant advancement of generic validation, however, they expressed strong interest in the specifics in the validation plans for individual codes.

The industry's future work will focus on individual computer codes and their interface with the validation matrix. The partner organizations may opt to retain their preferred codes and to identify potential gaps, if any, in the data base and the possible need for additional code development and validation against selected tests from the data base. The specific tests will be selected to ensure that all phenomena that are likely to be encountered during an accident are addressed. The selection of these tests will be done on the basis of a thorough understanding of the thermalhydraulic phenomena and their rank or relative importance during a postulated accident.

Although the focus of the above work was on CANDU safety analyses, the phenomena have broader applications to other thermalhydraulic systems such as research reactors and experimental loops.

3.3 Thermal-mechanical Behaviour of Fuel and Fuel Channels

The Working Group decided to construct the validation matrix in stages. The Group agreed that the initial data sets compiled for inclusion in the matrix would be those potentially suitable for validation of analytical tools used to assess channel-integrity concerns of large LOCAs.

Twenty three phenomena, representing all those expected to occur in any of the design-basis accidents, have been identified. In some cases, mutually dependent phenomena have been grouped and are represented by one observable process. This list has been cross checked for completeness for application to large LOCAs, via a detailed review of the relationships between safety concerns, parameters that are used to define margins for each safety concern, and the phenomena that determine the behaviour of each parameter. The latter information will represent the Group's contribution to the Technical Basis Document.

Synopses of all phenomena are being prepared. Initial definitions have been compiled, the task of preparing detailed descriptions has been distributed to Group members according to their area of expertise, and 14 descriptions have been produced. A preliminary ranking of phenomena, as either of primary or secondary importance, has been completed for each phase of the large-LOCA scenario. An initial draft list of 99 data sets has been compiled. Drafting of synopses for an initial selection of 30 of these is underway, with synopses of 29 of the in-reactor data sets completed. A draft matrix has been prepared that cross references the 23 phenomena to each of the 99 data sets. This initial correlation is based on preliminary expert judgment and still requires confirmation, following the preparation of data-set synopses.

3.4 Fission-Product Release and Transport

Due to the complexity and clear differences between the phenomena that control the fission-product release and the fission-product transport processes, for simplicity, the discipline was divided into these two sub-disciplines, and Sub-groups were formed in each. To avoid superposition, it is necessary to define the region of application for each sub-discipline. The following definitions have been adopted.

i) The Fission-Product Release sub-discipline includes all fission-product phenomena occurring in a fuel element up to the release of radionuclides via sheath failure.
ii) The Fission-Product Transport sub-discipline includes all fission-product phenomena occurring between sheath failure and release of radionuclides into containment.

Lists of 20 fission-product release phenomena and 23 fission-product transport phenomena have been produced. The lists of phenomena are under review by the team members and other members of the Canadian nuclear industry. Synopses that describe each of these phenomena and the identification of their key parameters are in preparation. As a trial case, the large LOCA combined with LOECI was selected for the phenomena-ranking process. The fission-product release phenomena were ranked as of primary or secondary importance with respect to their perceived impact on the amount of fission-product releases during a particular phase of the accident.

Preliminary identification of available experimental information on fission-product release indicates that the following tests are possible choices for the validation matrix: (i) 45 in-reactor tests, (ii) 200 in-cell tests, and (iii) 15 laboratory tests. Each test will be assessed to determine which phenomena occurred during the course of the test. This experimental data base includes experiments performed around the world. Some of these experiments, primarily in-reactor tests, were CANDU specific. The in-cell and laboratory tests have a wider application area.

In the area of fission-product transport, identification of relevant validation data sets is in progress. The data sets for code validation will include experiments performed in Canada, e.g., laboratory aerosol-transport tests, hot-cell fission-product-transport tests, and in-reactor tests performed in the Blowdown Test Facility at the Chalk River Laboratories. The data sets for the validation of fission-product-transport codes will also include international separate-effects and integral experiments such as those from the PHEBUS-FP program. After appropriate tests have been identified, the data sets will be summarized and the uncertainties in the data will be quantified.

3.5 Containment Behaviour

The discipline was divided into the sub-disciplines of (i) Containment Thermalhydraulics and Hydrogen Behaviour, and (ii) Fission Product Chemistry and Aerosol Behaviour, and Sub-groups were formed in each.

The current status of the draft chapter for the Technical Basis Document is as follows. Postulated accident scenarios have been identified, and one is described in detail. Safety concerns for the chosen accident scenarios have been identified, described, and tabulated. Fundamental phenomena have been identified along the sub-discipline lines. Six phenomena have been described, as examples of the detail required for the final document. A table showing the relative importance of the phenomena for the accident scenarios has been produced.

The current status of the draft Validation Matrix Report is as follows. The available data base has been organized into categories, with 25 separate-effects tests, 13 integrated tests, and 7 numerical tests covering the areas of containment thermalhydraulics, hydrogen combustion, fission product chemistry, and aerosol behaviour. An additional category, inter-code comparisons, is included, but no data sets have been identified because the benefit of this category to code validation is not clear at this time. Separate-effects tests, integrated tests, and numerical tests have been described briefly. Validation-base data sets and the number of individual tests in each set have been tabulated. The cross-reference table of the validation matrix that relates data sets to the phenomena identified in the Technical Basis Document has been prepared.

3.6 Physics

A Working Group has been assembled to define a validation matrix for the sub-discipline of reactor physics, seen as the area of high priority. While ad hoc validation work in the sub-disciplines of shielding and atmospheric dispersion of radionuclides is ongoing, it does not yet follow the validation-matrix methodology.

Preparation of the validation matrix for reactor physics is under way, and the steps outlined in Section 2 above are being followed.

In advance of the above work, AECL experts in physics produced preliminary documents on validation of physics codes, in all three sub-disciplines, that are in common use at AECL. These documents collect in one place information that has been generated over many decades and is dispersed in many references. These documents are useful now and are expected to make it easier to develop the validation matrix reports in the physics area.

3.7 Moderator and Related Thermalhydraulics

A Working Group has been formed to address moderator and related thermalhydraulics, and the Group has identified its scope of work. To date, the following tasks have been completed. A preliminary list of accidents involving the systems has been prepared. A preliminary list of concerns, behaviours, and phenomena for each accident has been developed. A preliminary table relating the major phenomena to accident has been prepared. Results of the above are being circulated for comment.

4. FUTURE VALIDATION WORK

The methodology described in the preceding sections defines the course of action adopted by the Canadian CANDU industry to achieve the end point, which is computer codes, validated according to a structured methodology, and suitable for future safety analyses of CANDU plants and licensing decisions. That end point will bring to closure some of the code-development work and R&D, which in some instances has been ongoing for decades. The end products of the generic work presently under way will be a Technical Basis Document and six Validation Matrix Reports, the first of which has been completed and commented upon by staff from the Atomic Energy Control Board. These documents will provide the basis for planning the next steps in the validation program. In the next steps, the most appropriate computer codes will be selected and, if needed, validation plans for them will be defined. Any further code development will be focused on identified shortcomings. If gaps exist in the validation data base that can be addressed by additional R&D, such R&D will be specified and executed.

5. NUCLEAR SYSTEMS OTHER THAN CANDU PLANTS

The preceding sections address the needs, with respect to validated computer codes, of the operators, designers, builders, and regulators of CANDU plants. AECL also operates other nuclear facilities, notably research reactors, and AECL designs, submits for licensing, and builds small reactors of the MAPLE family. The computer codes used in much of that work are often versions of those used in the CANDU business and hence require similar levels of quality assurance, including validation. The validation program described here provides a solid foundation to which specific validation work can be added to meet AECL's needs in the non-CANDU line of business. To foster close interactions between the CANDU and non-CANDU validation activities, a Working Group on Small Reactors has been formed within the scope of the Validation Team.

6. SUMMARY AND CONCLUSIONS

The Canadian CANDU industry has 35 years of experience in the development and application of computer codes used in safety analyses and licensing submissions. While these computer codes were validated as a matter of course during their development, that validation was performed according to the practice of the day. No single, systematic validation methodology was used because none existed. Recently, the OECD/NEA developed and published a validation matrix for system thermalhydraulics in LWRs, comprising two cross-reference tables: the first identifying physical phenomena that might occur in design-basis accidents, and the second identifying experimental and numerical tests that exhibit the physical phenomena. The validation matrix is generic to the chosen type of nuclear plant and serves as the basis for the validation of specific computer codes.

The Canadian CANDU industry adopted the fundamentals of the validation-matrix methodology for LWRs and is adapting and extending it to CANDU power plants. Industry-wide Working Groups have been formed to develop validation matrices in six scientific disciplines:

- i) System Thermalhydraulics;
- ii) Fuel and Fuel Channel Thermal-mechanical Behaviour;
- iii) Fission Product Release and Transport;
- iv) Containment Behaviour;
- v) Physics (comprising reactor physics, shielding, and atmospheric dispersion); and
- vi) Moderator and Related Thermalhydraulics.

These disciplines cover a much broader range of phenomena than those addressed by the OECD/NEA.

The Working Group in System Thermalhydraulics has the lead and has produced a validation matrix document. Working Groups in the other disciplines are at various stages in developing their validation matrices which will be generic in each discipline. The validation program is expected to span several years and to bring to closure the development of computer codes, validated according to a structured methodology, and suitable for safety analyses of, and licensing decisions on CANDU power plants. While this is the primary focus for the work currently under way, the methodology and results will also provide a basis for the validation of computer codes used in safety analyses of nuclear and experimental facilities other than CANDU power plants, notably small reactors of the MAPLE family.

7. ACKNOWLEDGMENT

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8. REFERENCES

[1] OECD/NEA, "Separate Effects Test Matrix for Thermal-Hydraulic Code Validation, Volume I, Phenomena Characterisations and Selection of Facilities and Tests; Volume II, Facility and Experiment Characteristics", Report OECD/GD(94)82, also NEA/CNSI/R(93)14/Part.1/Rev., Paris (1993).

Table 1: VALIDATION METHODOLOGY

(1)	Technical Basis Document	Relate safety concerns to main phenomena governing behaviour during each phase of specific accident.
(2)	Validation Matrices (6 in total)	Relate all relevant phenomena to accidents and data sets.
Generic (Code Independent) Knowledge Base		
Code Version Specific		
(3)	Validation Plan	Demonstrate that code version accurately represents governing phenomena for each phase of the selected accident scenario.
(4)	Validation Exercises	Compare model predictions with selected sets (uncertainty).
(5)	Validation Manual	Summarize code accuracy, sensitivities, and uncertainties for specific applications.

Table 2: EXCERPT OF THERMALHYDRAULIC PHENOMENA RELEVANT TO CANDU ACCIDENT ANALYSIS

ID NO	PHENOMENON	ACCIDENT SCENARIO (7)		
		(1) LOCA	(2) →→ LOCA/ LOECI	(7) STEAM LINE BREAK
TH1 ↓	Break discharge characteristics and critical flow	√	√	√
TH12 ↓	Quench/Rewet characteristics	√		
TH23	Noncondensible gas effect	√	√	

Table 3: EXCERPT FROM RANKING OF PHENOMENA FOR LARGE LOCA

PHASE	POWER PULSE/REACTOR TRIP	EARLY BLOWDOWN COOLING	LATE BLOWDOWN COOLING/EC INJECTION	REFILL
Time(s)	0 - 5	5 - 30	30 - 200	>200
		PHENOMENA		
PRIMARY (7) ↓	Break discharge characteristics & critical flow.	Break discharge characteristics & critical flow.	Break discharge characteristics & critical flow.	Counter-current flow.
SECONDARY (3) ↓	Critical heat flux & post-dryout heat transfer	Critical heat flux & post-dryout heat transfer	Phase separation	Waterhammer

Table 4: EXCERPT OF SEPARATE EFFECTS TESTS, COMPONENT TESTS, INTEGRATED EXPERIMENTS, NUMERIC TESTS, AND INTER-CODE COMPARISONS RELEVANT TO THERMALHYDRAULIC CODE VALIDATION

SE1 ↓	Edwards Pipe Blowdown	2 tests
SE25	WL Waterhammer Tests	about 48 tests
CO1 ↓	Stern Labs End Fitting Characterization Tests	about 600 tests
CO5	MR-2 Air-Water Test Loop	about 225 tests
INT1 ↓	Stern Pressure-Tube Burst Tests (IBT Series)	6 tests
INT17	RD-14M Shutdown Cooling Tests	9 tests
NUM1 ↓	JUICE Standard Problems	3
NUM10	Tank Bottom Discharge Test	1
No Inter-Code Comparisons Identified at this Stage		

Table 5: EXCERPT OF THERMALHYDRAULIC PHENOMENA AND RELEVANT TEST DATA FOR CODE VALIDATION: Separate Effects Test

ID NO.	PHENOMENA	SE1→	SE4 →	SE16→	SE22→	SE25
TH1 ↓	Break discharge characteristics & critical flow	□				
TH12 ↓	Quench/Rewet characteristics		■	■		
TH20 ↓	Waterhammer				■	
TH23	Noncondensible gas effect					

■ Suitable for direct validation
□ Suitable for indirect validation

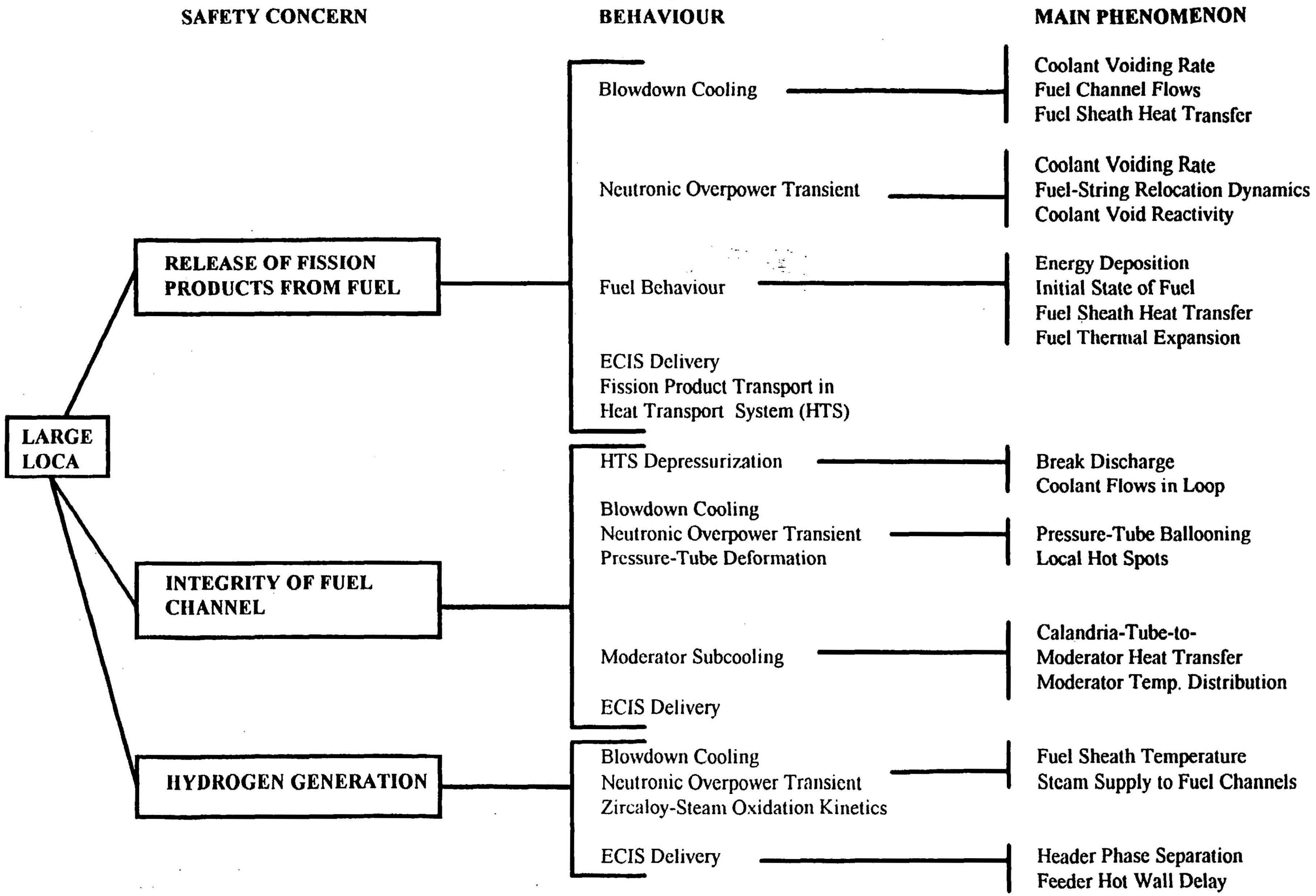

FIGURE 1: TECHNICAL BASIS FOR LARGE LOCA