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Technical Basis Document

Technical Basis for the Validation of Computer Programs Used for Safety Analysis of the ACR Design

Advanced CANDU Reactor

108US-03500-TBD-001
Revision 0

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ABSTRACT

This Computer Program Validation Technical Basis Document relates Advanced CANDU® Reactor (ACR)™* safety analysis scenarios to the fundamental physical phenomena that form the basis for computer modelling of these accident scenarios. The document describes the categorization of accident scenarios and physical phenomena of importance for computer code validation that is relevant to the ACR. It provides an organised approach to the management of detailed information on physical phenomena important for computer program validation.

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TABLE OF CONTENTS

SECTION **PAGE**

TABLE OF CONTENTS

SECTION **PAGE**

TABLE OF CONTENTS

SECTION **PAGE**

TABLE OF CONTENTS

SECTION **PAGE**

TABLE OF CONTENTS

SECTION **PAGE**

TABLE OF CONTENTS

SECTION **PAGE**

TABLE OF CONTENTS

SECTION **PAGE**

TABLES

FIGURES

APPENDICES

1. INTRODUCTION

This Technical Basis Document (TBD) provides a foundation for the safety analysis computer program[1] validation process established by AECL for application to the Advanced CANDU Reactor (ACR). It includes a brief overview of the entire validation process, beginning with the identification of postulated accident scenarios for the ACR. The main intent of the present document is to describe these postulated accident sequences and to identify the physical phenomena which dominate the accident phases.

The Technical Basis Document addresses design basis events and limited core damage accidents. In the ACR event classification approach, design basis events are categorized into three classes, 1, 2, and 3. The design basis events include seven event types; large LOCA, small LOCA and single channel accidents, secondary coolant failures, fuel handling accidents, loss of flow, loss of regulation, and auxiliary system failures.

Table 1 provides examples of class 1, 2, and 3 design basis events and the corresponding chapter in this report where they are covered.

[1] A glossary of selected terms and acronyms is provided in Chapter 12.

Table 1
Examples of Design Basis Accidents

Class 1 Events	**Report Section**
Loss of Pressure and Inventory Control	Ch. 7, LOR
Loss of Secondary Circuit Pressure Control	Ch. 5, Secondary Side Failures
Loss of Reactivity Control	Ch. 7, LOR
Total Loss of Class IV Power[2]	Ch. 8, LOF
Single HT Pump Trip	Ch. 8, LOF
Moderator Events (except pipe ruptures)	Ch. 9, Auxiliary System Failures
Loss of Feedwater Flow	Ch. 5, Secondary Side Failures
Class 2 Events	
Steam Generator Tube Rupture	Ch. 5, Secondary Side Failures
Pressure Tube Failure (CT intact)	Ch. 4, Small LOCA & Single Channel Accidents
Small LOCA	Ch. 4, Small LOCA & Single Channel Accidents
Off-Stagnation Feeder Break	Ch. 4, Small LOCA & Single Channel Accidents
End Fitting Failure	Ch. 4, Small LOCA & Single Channel Accidents
Feedwater Pipe Break	Ch. 5, Secondary Side Failures
Moderator Events (pipe ruptures)	Ch. 9, Auxiliary System Failures
Partial Single Channel Flow Blockage	Ch. 4, Small LOCA & Single Channel Accidents
On-Reactor Fuel Handling Accident	Ch. 6, Fuel Handling Accidents
Class 3 Events	
Large LOCA	Ch. 3, Large LOCA
Main Steam Line Break (inside containment)	Ch. 5, Secondary Side Failures
Heat Transport System (HTS[3]) Pump Seizure	Ch. 8, LOF
Pressure Tube/Calandria Tube Failure	Ch. 4, Small LOCA & Single Channel Accidents

In the ACR event classification approach, limited core damage accidents are categorized into two classes, 4 and 5. This includes any design basis accident combined with loss of emergency core cooling. This category (limited core damage accidents) includes two additional scenarios; severe flow blockage and stagnation feeder break.

2 "Class IV Power" is equivalent to the stations normal AC power supply.

3 The Heat Transport System (HTS) is equivalent to the Reactor Coolant System (RCS).

Table 2 provides examples of limited core damage accidents that are discussed in chapter 10 of this report.

Table 2
Examples of Limited Core Damage Accidents

Class 4 and 5 Events
Large LOCA +LOECC
Small LOCA +LOECC
Severe Channel Flow Blockage
Stagnation Feeder Break
Pressure Tube/Calandria Tube Failure +LOECC
End Fitting Failure +LOECC
Single Steam Generator Tube Rupture +LOECC

1.1 The Validation Process

The purpose of the validation process is to:

- Develop a consistent basis for validation among several disciplines associated with safety analysis of postulated accident scenarios;
- Identify and define phenomena that affect the physical behaviour of the reactor system during specific accident scenarios and other safety-related events;
- Cross-reference these phenomena to well documented and reliable experimental data sets which can be used to validate the modelling of physical phenomena; and
- Document a comparison of these data sets with computed results, which will provide guidance to computer program users as to the nature and magnitude of any uncertainty in the analysis results within the range of applicability of the computer program.

Computer program validation methodology can be organized as a multi-stage process associated with a structured set of documents, as shown in Figure 1. The first two stages are implemented with the preparation of a Technical Basis Document and Validation Matrix Documents, which are prepared without reference to specific versions of computer programs. The remaining stages of the process involve the completion of Validation Plans, Validation Exercises and Validation Manuals for specific computer programs.

1.2 A General Basis for Validation

Different approaches to computer program validation have been discussed to some extent in the literature. This section describes the general approach to computer program validation and documentation that has been established by national and international groups that have sought to guide, organize and retain consistency in the process (Reference [1]).

1.2.1 Technical Basis Document

The Technical Basis Document provides a technical breakdown of each of the accident categories relevant for the ACR accident analysis. The main objective of the TBD is to provide a foundation for selecting physical phenomena of importance for the validation of computer programs. Governing phenomena involved in each accident scenario are identified and grouped into technical specialities or disciplines for which computer program validation will have to be provided. The document focuses on the phenomena and disciplines involved, and does not specify the computer programs to be employed in the analysis. A phenomenon is listed as having a ‘primary’ importance if it has a significant impact on one or more of the event key safety concerns. A ‘secondary’ designation is used for phenomena which are not primary but have some impact on one or more of the event key safety concerns. The remaining phenomena, which are determined to be of neither primary nor secondary significance, during any phase of an accident, are left blank, and not discussed. If there is uncertainty in deciding whether a phenomenon should be ranked primary or secondary, a conservative approach is used and a primary ranking is assigned.

From another viewpoint, the TBD provides a summary of the accident analyses required to quantify the consequences of events for a given event type. It identifies the physical phenomena influencing reactor behaviour relevant to an accident safety concern. It also identifies key safety criteria, such as public dose, fuel channel integrity, containment integrity, etc., for which consequences are quantified and assessed through analysis. The governing phenomena associated with the various phases of an accident are identified by technical speciality or discipline (e.g., reactor physics, thermalhydraulics, etc.). The document discusses the rationale and relevance of the associated primary phenomena for each accident phase.

The accident-scenario-focused TBD thus serves to establish major accident types and events, the disciplines involved in assessing the accident, the safety concerns associated with the phases of the accidents, and the governing physical phenomena.

1.2.2 Validation Matrices

Validation Matrices relate, by discipline, the phenomena identified in the TBD to data sets that exhibit the phenomena. The data sets are taken from operational data, experimental measurements, analytic solutions, single effect tests, integral effects tests or results from other validated computer programs. The data sets themselves must also be verified, qualified, and their sources of error/uncertainty identified. To maintain the independence of the validation process, data sets which are the basis for developing the physical models should not be used for validation.

1.2.3 Validation Plans

Validation Plans are specific to a particular computer program version. The plan identifies the intended applications for which a computer program version is being validated, and uses the Validation Matrices to identify data sets for validation and any gaps in the data available for validation. The plan details what will be done to demonstrate that a particular computer program version accurately represents the phenomena occurring in selected accident scenarios for the intended range of application.

The plan identifies the series of Validation Exercises to be undertaken as the next stage of overall validation. Specification of the Validation Exercises may require specific selection from the data sets. The criteria for the selection of sub-matrices from the Validation Matrices are identified in the Validation Plans. The selection criteria are based on the key parameters and governing phenomena which dominate the key safety criteria for the computer program application(s) being validated.

The method by which input data uncertainties will be assessed is also addressed in the Validation Plan document.

1.2.4 Validation Exercises

The Validation Exercises compare the computer program results (modelling of governing phenomena) to the relevant data sets and quantify the uncertainties and biases in the identified key safety output parameters over the range of application. The test apparatus and procedure associated with the data sets are briefly described in an associated Validation Report and linked by cross-reference to original test results described in the Validation Matrix.

Errors and uncertainties associated with the data sets are documented in the Validation Report for a particular Validation Exercise. On completion of the simulations and comparisons, the sensitivity of key safety output parameters due to uncertainties in key input parameters is determined.

1.2.5 Validation Manuals

Validation Manuals summarize the results of Validation Exercises in the context of the Validation Plan. They document how the technical basis for validation of a particular computer program has been covered. Each document also summarizes the uncertainty associated with computer program predictions of the identified key safety output parameters for the range of intended applications. These documents are intended to be a guide to users in practical applications of the computer program.

1.3 Canadian Nuclear Industry Computer Program Validation

1.3.1 Background

There are several organizations within Canada that are involved in the development and application of computer programs for nuclear reactor safety analysis. Canadian reactor types currently include the CANDU heavy water pressure tube reactor, the Advanced CANDU Reactor (ACR) with H_2O coolant and D_2O moderator, as well as small pressure tube and pool-type reactors used for isotope production and reactor-design research. Although many of the accidents and phenomena are common to the CANDU system, this TBD is specific to the ACR. As a result of the co-operative development of information relevant to computer program validation, a particular adaptation of the foregoing generalized methodology, which is consistent with international practice [1], complies with quality assurance requirements [2], and takes into account regulatory guidelines [3], has been undertaken for nuclear power plant computer program validation.

1.3.2 Technical Basis Document

To provide a systematic basis for code validation a discipline structure for nuclear safety analysis has been established by the Canadian nuclear industry, as shown by the following breakdown into areas of reactor system, technical speciality or discipline:

1. Reactor Physics
2. Trip Coverage
3. Fuel Channel and System Thermalhydraulics
4. Fuel and Fuel Channel Thermal-Mechanical Effects
5. Moderator and Shield System Thermalhydraulics
6. Fission Product Release and Transport to Containment
7. Containment (Containment Thermalhydraulics, Fission Product Chemistry, Hydrogen and Aerosols)
8. Radiation Physics
9. Atmospheric Dispersion (including public dose assessment).

This Technical Basis Document is arranged in a matrix format. Primary and secondary phenomena are identified with respect to initial and accident conditions occurring in major design basis event types and limited core damage accidents across the above nine disciplines.

In an accident scenario, system behaviour that involves the overlap of two or more disciplines, is addressed in this document by the discipline responsible for the dominant behaviour, or by each discipline if none is clearly dominant. Thus, for example, System Thermalhydraulic phenomena are discussed along with Reactor Physics phenomena in postulated accident scenarios where the two interact.

1.3.3 Validation Matrix Documents

A number of Validation Matrix Documents should focus on phenomena relevant to the nine technical disciplines listed in Section 1.3.2.

The matrices are not generated based upon computer-program-specific considerations, but rather upon the underlying knowledge base of the specific discipline or technical speciality. This allows for flexibility when applying the Validation Matrices to cases where more than one computer program has been developed and adopted for a particular application.

The Validation Matrix Documents should be prepared to cross-reference phenomena with data sets from experiments, measurements and other tests [4]. This provides a convenient summary of the availability of data sets to validate computer programs in a selected discipline, and to identify any gaps in experimental data in the range required for validation.

Detailed descriptions of the phenomena and the test facilities are included in the Validation Matrix Documents for each discipline. The documents include:

1. Identification of experimental data sets in which the phenomena are known to occur;
2. A summary of the state of knowledge and uncertainties concerning the phenomena; and
3. Key references to papers or reports that describe or quantify the phenomenon.

The Validation Matrices provide an organized linkage between the physical phenomena of importance and the associated data sets. The Validation Matrix Documents can be modified as new data sets to quantify phenomena become available.

1.3.4 Validation Plans, Exercises and Manuals

Individual Validation Plans, Exercises and Manuals are computer program specific. Necessary actions are identified from the Validation Matrices and the validation status of the program.

The Validation Plans and Exercises for CANDU safety analysis computer programs take into account regulatory guidelines [3] and assure compliance with quality assurance requirements [2] for the validation of computer programs for nuclear power plants in Canada. They include the criteria for acceptance of computer program accuracy and ensure sufficient input for an uncertainty assessment.

1.4 Technical Basis Document Organization

This document is primarily devoted to the discussion of initial conditions, accident scenarios and relevant phenomena in accordance with the scope, described in Sections 1.2.1 and 1.3.2. The TBD provides a guide to the overall validation process and acts as an introduction to more complex details covered by the Validation Matrix documents.

The initial conditions for accident analyses are established by determining reactor parameters when operating in a steady state at a certain power level. This steady state is usually established through the application of computer programs using parameters that may be measured or predicted. Chapter 2 provides a discussion of phenomena important to the establishment of the initial conditions for use in each discipline.

Each chapter is devoted to a major design basis event type (Chapter 3 to 9). Limited core damage accidents are discussed in Chapter 10. Each chapter begins with an overview of accident conditions and accident sub-categories. Accident descriptions follow, and are organized chronologically or by logical sub-division for each discipline. This is followed by the identification of important safety concerns (public dose, fuel channel integrity, etc.), and the overall accident behaviour in the context of primary phenomena. Subsequently, the primary phenomena are discussed in detail for the accident types. In some cases, some or all of the phenomena governing system behaviour are identical to those discussed in another accident scenario, in which case repetition is avoided by referring to the appropriate section of this Technical Basis Document. Finally, a summary of the accident scenario is provided.

The Technical Basis Document describes basic accident sequences and relevant phenomena that arise during the progression of various accident types. A listing of phenomena identified for nine disciplines is given in column 1 of Table 3 along with the importance assigned to each of the eight accident scenarios as is discussed in Section 1.2.1. The matrix tabulation in Table 3, serves as an introduction to, and summary of, the discussion of governing phenomena in each discipline for the accident scenarios described in Chapters 3 to 10.

Chapter 11 is a closing summary to the TBD. Appendix A provides a discussion of atmospheric dispersion phenomena, since this is generic in nature and applies to all accident scenarios.

2. INITIAL CONDITIONS

2.1 Introduction

This section discusses phenomena that determine the initial reactor conditions. Examples of such phenomena are: neutron flux and power distribution in the core, fuel-to-sheath[4] heat transfer, fuel condition (e.g., sheath defects), fuel channel condition (e.g., pressure tube creep), buoyancy and inertial effects in the moderator, fission product distribution, etc. Examples of initial conditions are: normal operating conditions, shutdown conditions, reduced power conditions, etc.

For some disciplines, these phenomena are generic to all accident scenarios. Therefore, for these disciplines, the phenomena associated with initial reactor conditions are discussed only once (in Chapter 2) without reference to an accident scenario. Where these phenomena are different for different accident scenarios, the phenomena are discussed for each accident scenario separately.

2.2 Systems and Disciplines

2.2.1 Reactor Physics

A primary phenomenon in most cases is the flux and power distribution (prompt or decay heat) in space and time *(PH14)*[5]. A number of factors, including reactivity induced device movement, have an impact on the flux and power distribution. Fuel initial conditions, fission-product content and fuel properties are affected by the power/burnup history of the fuel.

2.2.2 Fuel Channel and System Thermalhydraulics

To facilitate safety analysis of fuel channel and system thermalhydraulics for a given accident scenario, different phases of an accident are analyzed separately. Typically, an initial steady state condition represents limiting operating conditions for the event. Phenomena specific to each accident scenario and phase are discussed in the appropriate sections of Chapters 3 to 10. Some of the factors influencing the initial steady state usually include pressure drop over the fuel string, steam generator tube heat transfer, system resistance distribution, and fuel element heat transfer. The fuel channel and system thermalhydraulics phenomena relevant to these parameters are thermal conduction *(TH6)* of heat in the fuel and fuel sheath, pump characteristics (TH5), condensation heat transfer (TH10), convective heat transfer *(TH7)* from the sheath to the primary coolant, and subcooled nucleate boiling *(TH8)* which is present to a small degree in the hotter sections of the channels.

Radiative heat transfer *(TH11)*, while present, is negligible at normal operating conditions and thus has little or no effect on the thermalhydraulic state prior to a transient. Similarly, other phenomena, such as coolant voiding *(TH2)* and phase separation *(TH3*), are usually of significance only during certain phases of a transient resulting from a postulated accident.

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4 "Fuel sheath" is equivalent to cladding.

5 The numbering system for phenomena is that used in Table 3.

2.2.3 Trip Coverage

For trip coverage assessment, the initial conditions for fuel channel and system thermalhydraulics are chosen to be conservative with respect to the event or parameter of interest.

Heat transport system pump characteristics (*TH5*), including the inertia of the flywheel and back-EMF (Electro-Motive-Force) and two-phase resistance braking effects, are of prime importance for all loss of flow transients (partial and total), loss of Class IV power, pump trip, and pump seizure.

The initial pressurizer level, involving level swell and void holdup (*TH4*), is important for all loss of pressure and inventory control transients as well as small LOCA and overcooling events such as main steam line breaks; level swell and void holdup are of secondary importance in overpressure transients such as loss of reactivity control or loss of flow.

Secondary circuit levels and volumes (in the steam generators and deaerators), involving level swell and void holdup *(TH4)*, are important for the feedwater failures and steam line break scenarios as they determine the inventory available for the heat sink and thus affect the timing of the event.

The initial flow (including the effect of instrument uncertainties that could cause the measured flow to be lower and closer to the setpoint) and power *(PH14)* in the instrumented channels (i.e., channels used for SDS1 and SDS2 low flow trip) have a significant effect on the timing of the low flow trip. Initial channel flows are influenced by coolant temperature and ageing effects such as pressure tube creep and magnetite deposits on the piping surface. This low flow trip is prominent in the loss of flow events (loss of class IV power, pump trip, pump seizure) as well as for the small LOCA scenario. It is also important for the large LOCA for reduced initial powers, as well as for the smaller break sizes.

2.2.4 Fuel and Fuel Channel Thermal-Mechanical Effects

The initial conditions for thermal-mechanical effects in the fuel and fuel channel are established in a similar manner to those for the thermalhydraulics of the fuel channel and system. Phenomena involved in heat transfer from the fuel are discussed in Section 2.2.2.

In addition, the condition of the fuel can be important for some scenarios as it may affect fuel temperatures, fission product release or sheath failure thresholds. Also, plant ageing can affect the channel geometry because of pressure tube creep and calandria tube sagging and changes in their material properties.

2.2.5 Moderator and Shield System Thermalhydraulics

Certain accident scenarios are influenced by the initial conditions or quasi-steady-state existing in the moderator prior to the start of an accident. Transient local moderator sub-cooling (determining the mode of heat transfer from the calandria tubes to the moderator *(MH22)* during periods of pressure tube/calandria tube contact) is dependent primarily on the pre-existing quasi-steady-state temperature distribution in the moderator, and on the combined cover gas and static pressure head.

The quasi-steady-state moderator temperature and flow distribution combines the effects of forced convection due to the inlet jets and buoyancy convection due to internal heat generation in the fluid within a complex geometry (calandria tube matrix). There are several phenomena that determine the moderator temperature and quasi-steady-state flow distribution. These parameters are primarily affected by the ratio of inertial to buoyancy forces *(MH12)* within the vessel (known as the Archimedes or Richardson number). Inlet jet strength and development *(MH13)* determine the inertial forces in the vessel whereas buoyancy forces *(MH12)* are a result of local heat generation. Turbulence *(MH11)* and interaction of the inertial and buoyancy forces with the calandria tubes *(MH10)* also influence the temperature and flow distribution. This, in turn, is determined by the velocity distribution across the face of the nozzle inlet which can be determined experimentally. Depending on the Archimedes number, the flow pattern can be dominated by either buoyancy or momentum. For ACR, the moderator inlet flow is directed to enhance the buoyancy-driven flow pattern. Inlet flow is injected downwards along the wall of the calandria and collides at the bottom of the vessel, resulting in a stable upward flow pattern through the core that is enhanced by buoyancy-driven flow, giving a stratified moderator temperature distribution.

2.2.6 Fission Product Release and Transport to Containment

Fission Product Release

During the fuel irradiation before an accident, several phenomena actively contribute to the fission product distribution in the fuel pellets and gap inventory. The primary phenomena in this phase are diffusion *(FPR2)*, grain boundary sweeping/grain growth *(FPR3)*, grain boundary coalescence/tunnel interlinkage *(FPR4)*, and vapour transport/columnar grains *(FPR5)*. The only secondary fission product release phenomenon applicable to the intact fuel is athermal release *(FPR1)* which is important only at very low temperatures.

If fuel element failures occur during the pre-accident irradiation, the change in the fuel stoichiometry will increase the fission product inventory available for immediate release due to UO_{2+x} formation *(FPR9)*, and possibly U_4O_9-U_3O_8 formation *(FPR10)* if the failures are sufficiently severe. The primary phenomena listed for intact fuel also apply to fuel that failed before the accident. Gap transport *(FPR7)*, gap retention *(FPR8)* and fission product leaching *(FRP19)* will also have primary roles in the fission product distribution prior to the transient for fuel failed before the accident.

Fission Product Transport

During the pre-accident fuel irradiation, the primary contributor to the transport of fission products and actinides from defected fuel is transport of deposits by water *(FPT22)*. Chemical speciation *(FPT23)* plays a secondary role by determining the form in which transport occurs (particulate or dissolved).

2.2.7 Containment

During normal operation, the containment building is maintained at a small negative pressure, in order to eliminate or minimize the leakage of gaseous fission and activation products to the outside atmosphere. The containment ventilation system normally operates to provide fresh air to those areas of containment (accessible areas) which are accessible by the station operators. Fresh air is drawn in through the ventilation inlet line, distributed to various accessible rooms inside containment and eventually expelled through the ventilation outlet line via the exhaust stack. Vault coolers normally operate to maintain the containment temperature at a level acceptable for station operators, and are also normally available in the event of a LOCA to limit containment pressurization.

In some cases, the consequences of a postulated accident are influenced by the initial containment operating conditions. Thus, the initial conditions inside containment are generally established conservatively for each initiating event, to maximize the event consequences and analysis objectives.

2.2.8 Radiation Physics

The radiation physics discipline is responsible for establishing the distribution of thermal energy deposition within the reactor core *(RAD1, RAD3, RAD4, RAD5, RAD6)*. Although most of the nuclear energy is released in the fission process and carried off by the recoiling fission fragments to be deposited in the ceramic fuel itself, some of the neutron and photon energy is deposited in other core components. These other components include the moderator heavy water, calandria tubes and pressure tubes, etc. The radiation physics discipline also establishes the inventories of radioactive material *(RAD2)* in the core due to fission products.

2.2.9 Atmospheric Dispersion

During normal operation, very low levels of radioactive contaminants are released in the gaseous phase from a nuclear power plant. Many of these contaminants are released from areas outside containment. The behaviour of these gaseous contaminants in the atmosphere is subject to the same phenomena as the airborne contaminants released following an accident. Including the contribution from food intake, the annual dose to the most exposed member of the public is small and is measured, typically, in terms of a few microsieverts per annum. Thus, the contribution to public dose is too small to be considered in the initial conditions to the accident analysis, and none are established.

Initial conditions for atmospheric dispersion and dose calculation (e.g., weather) are described in Appendix A.

2.3 Summary

Pre-transient conditions are a necessary input to accident analyses for all disciplines. For some disciplines, the initial conditions are the same for all accident scenarios. The quasi-steady-state moderator flow/temperature distribution determines the pre-transient state for moderator analysis. The inventory of radionuclides in fuel pellets and the gap establish the initial conditions for fission product release analyses and the decay heat analyses.

Other disciplines (reactor physics, fuel channel and system thermalhydraulics, fuel and fuel channel thermal-mechanical effects, and containment) typically tailor the initial conditions to particular accidents or phases of accidents. In cases where best estimate simulations are being performed, as in validation against integral test data, such tailoring of initial conditions should be minimal. The pre-transient assumptions for these disciplines are discussed with the corresponding accident scenarios described in Chapters 3 to 10.

3. LARGE LOSS OF COOLANT ACCIDENT

3.1 Description of Accident

A large Loss of Coolant Accident (LOCA) involves a break in the heat transport system pressure boundary that results in fast depressurization of the heat transport system and an associated degradation of fuel cooling. This scenario includes cases for which Class IV power may, or may not be, available.

The following general features characterize a large LOCA event:

1. A large rate of coolant discharge from the break into containment;
2. A decline in reactor power due to the negative coolant void reactivity;
3. A reactor trip resulting in reducing reactor power to decay power level;
4. The potential for early impairment of fuel cooling, leading to fuel failures;
5. A spike of Iodine (and other water soluble fission products) release from any previously defected fuel into the coolant during the blowdown period;
6. An available Emergency Core Coolant (ECC) system with successful coolant injection; and
7. 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 those for which channels in the affected core pass (downstream of the break) experience:

- reduced flow in the normal flow direction (for break sizes less than the critical break size);
- early flow reduction to very low level for a limited period of time (the critical break size); and
- sustained reverse flow during the blowdown (for break sizes greater than the critical break size).

3.2 Key Safety Concerns

The safety concerns relevant to large LOCA events whose consequences are quantified through safety analysis are:

- Public dose related to fission product releases from the fuel;
- Core coolable geometry related to fuel channel integrity;
- Effect of jet discharges and pipe whip on safety related systems; and
- Containment integrity related to pressurisation.

In addition, trip coverage analysis is performed for the entire range of break sizes, down to the small LOCA range (for which the reactor regulating system (RRS) is capable of maintaining reactivity balance), and for the entire range of initial reactor power, including low powers.

3.3 Accident Behaviour

3.3.1 Introduction

Quantification of the consequences associated with the above safety concerns involves analysis of phenomena that influence physical processes and determine system behaviour during a large

LOCA. These system behaviours 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 and ECC system initiation. For example, the early stages of blowdown cooling is accompanied by reactor shutdown; whereas, ECC delivery develops later and proceeds in parallel with the later stages of blowdown cooling. Therefore, uncertainties in modelling the phenomena determining a system behaviour are of relevance to the safety analysis only during those periods of time in which the phenomena exert a governing influence.

3.3.2 Phases of the Large LOCA Accident

The phases of a LOCA accident are defined according to the major time periods during the accident progression for which characteristic system behaviours are exhibited. For each of the disciplines involved in large LOCA analysis, the phases are defined in the following sections. 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.

3.3.2.1 Reactor Physics

1. Initial Conditions - the steady state power and flux distribution in the reactor just prior to the large LOCA.
2. Shutdown Period (0-5 seconds) - the initial period following the break during which the reactor power is being brought to a highly subcritical state due to shutdown system action.
3. Blowdown Period (5-60 seconds) - the period following reactor shutdown in which the spatial neutron flux distribution stabilizes in a subcritical state and the power distribution becomes governed by decay heat.

3.3.2.2 Trip Coverage

For a large LOCA, trip coverage applies mainly to the first portion of the transient up to reactor shutdown. Analysis is done for the entire range of break sizes and initial powers. The important trip parameters are high containment pressure, low HTS pressure, low flow and Regional Over-Power (ROP).

3.3.2.3 Fuel Channel and System Thermalhydraulics/Fuel and Fuel Channel Thermal-Mechanical Effects/Fission Product Release and Transport

1. Early Blowdown Cooling (0-30 seconds) - the period during which the reactor is being shut down and the heat transport system blowdown continues prior to ECC initiation. The dominant system behaviour during this period is a result of heat transport system depressurization, blowdown cooling, fuel and sheath heat-up, pressure tube heat-up, fuel failure and fission-product release.
2. Late Blowdown Cooling/ECC Injection (30-200 seconds) - the period of ongoing heat transport system blowdown with ECC injection into the heat transport system. The dominant system behaviour during this period again is due to heat transport system depressurization, blowdown cooling, ECC delivery, fuel bundle deformation, fuel heat-up, pressure tube heat-up and fission product release.
3. Refill (> 200 seconds) - the period during which the HTS pumps are tripped, refill of channels in the core proceeds and a quasi-steady-state is attained. The dominant system

behaviour during this period is determined by ECC delivery, heat transport system refill, fuel cooling and fission product release.

3.3.2.4 Moderator and Shield System Thermalhydraulics

From the moderator thermalhydraulics standpoint there is no need to define different accident phases and this discipline has insignificant impact to safety analyses of LOCA with ECC available.

The dominant system behaviour for moderator thermalhydraulics is established by the transient moderator temperature distribution, which is determined to a large degree by the quasi-steady-state moderator temperature distribution (see Section 2.2.5), the transient heat load and forced flow (pump) availability.

3.3.2.5 Containment

1. Short-term Pressurization (0-60 seconds) - the initial period of containment pressurization resulting from the coolant discharged out of the rupture in the heat transport system boundary. The dominant system behaviour during this period is due to coolant flashing from the break leading to containment pressurization, aerosol formation, fission product behaviour inside containment and fission product release from containment.
2. Long-term period (>60 seconds) - the containment pressure peaks immediately following the short-term period as the break discharge begins to decrease and local air coolers and building structures remove heat from the containment atmosphere (LACs are accident qualified for ACR and can thus be credited for long-term heat removal following a LOCA). The pressure subsequently decreases due to decreased flashing from the break discharge (less heat input to the containment atmosphere) and continued heat removal by the local air coolers. The containment pressure eventually reaches atmospheric, days after the initiating event. The dominant system behaviour during this period is determined by containment mass transfer, containment heat rejection, fission product behaviour inside containment, aerosol formation and removal from the containment atmosphere, and the behaviour of hydrogen generated through radiolysis of the coolant. Hydrogen recombiners act to remove this generated hydrogen.

3.3.2.6 Radiation Physics

Radiation physics makes a limited contribution to safety analyses of different accident scenarios compared with those of other disciplines such as reactor physics, thermalhydraulics and containment behaviour. The main role of this discipline in safety analysis is the calculation of total fission product inventories in the fuel and of the distribution of nuclear energy deposition. These quantities are used as inputs to the main safety analysis; an example is the decay power of the fuel. The calculation of nuclear energy deposition is also the starting point to quantify the amounts of hydrogen released by radiolysis to containment in the long term.

3.4 Governing Physical Phenomena

3.4.1 Reactor Physics

The primary reactor physics phenomena influencing the power transients of large LOCA, with and without the loss of Class IV power, are the shutdown-system device-movement induced reactivity *(PH11)*, and the flux and power distribution (prompt/decay heat) in space and time *(PH14)*. Some secondary phenomena are coolant density change induced reactivity *(PH1)*, and the flux-detector response *(PH13)*.

The flux and power distribution (prompt/decay heat) in space and time *(PH14)* is the primary phenomenon for the initial and blowdown phase of a large LOCA transient. The initial flux and power distributions characterize the reactor core configuration and have a strong influence on the subsequent transient. However the decay heat characterizes the reactor physics behaviour during the blowdown phase of the transient.

The coolant-density-change induced reactivity *(PH1)* is driven by the change in coolant voiding *(TH2)*. Large LOCA in ACR results in an induced small negative reactivity due to core voiding.

3.4.2 Trip Coverage

The main phenomena relevant for trip coverage are the heat transport system depressurization behavior *(TH1)* resulting from the break discharge, and coolant voiding (*TH2*). The amount of voiding affects the timing of the process and neutronics trips. The trip parameters for this event include high reactor building pressure, low HTS pressure, low flow and ROP.

3.4.3 Fuel Channel and System Thermalhydraulics

The heat transport system depressurization behaviour is determined by the:

a) State of the fluid,
b) Rate of coolant discharge from the pipe rupture (governed by break discharge characteristics and critical flow *(TH1)*),
c) Coolant mass and energy redistribution within the heat transport pipes and components that occurs due to the combined effects of the transient pressure gradients and the heat transport pump pressure heads and flows (governed by HTS pump characteristics *(TH5)*), and
d) Reactor power level at the time of the accident.

For a given break location, and break sizes below a particular size, the flow in the affected downstream pass will be maintained in the normal forward direction. At a "critical" break size the flow being delivered by the two pumps in a pass is balanced by the flow discharge out the break, resulting in one pass experiencing very low flows (a condition sometimes referred to as stagnation). For break sizes above the critical break, one pass will experience reverse blowdown flows while the other pass will maintain forward flow. In the early blowdown phase the heat transport pump characteristics (*TH5*) affects the balance between the forward flow and the reverse flows depending on whether the flow is single phase or two phase.

Depressurization behaviour is strongly influenced by the state of the fluid. For subcooled liquid, the pressure decreases very rapidly even when relatively small amounts of fluid have been discharged. The pressure of the subcooled fluid will decrease until it reaches the saturation pressure corresponding to the fluid temperature. At this point, flashing of liquid to steam will occur and the rate of pressure decrease, for the same rate of mass discharge, becomes significantly lower. This is a consequence of the lower density of the steam phase, relative to the liquid, associated with steam generation from the flashing liquid. Finally, if a steam filled system is depressurizing, the rate of pressure decrease, for the same mass discharge rate, becomes higher than that for two-phase conditions. In this case there is no liquid to generate steam that can displace the volume of fluid being discharged from the break.

Blowdown cooling behaviour is of primary importance in determining the magnitude and duration of fuel element and pressure tube temperature excursions. As such, it has a strong influence on the calculated fission product releases and the challenges to fuel channel integrity.

Important phenomena influencing the blowdown cooling behaviour are the rate at which coolant voiding *(TH2)* develops, the magnitude of coolant flows in the fuel channels, and the heat transfer between the fuel sheath, coolant and pressure tube *(TH6, TH7)*. Coolant voiding is determined by break discharge, flashing and heat transfer from the sheath to the coolant. Critical heat flux (CHF, dryout) and post-dryout heat transfer *(TH9)* from sheath to coolant dominates the fraction of void generation due to heat transfer form the sheath to the coolant. Zircaloy-water thermal chemical reaction *(TH13)*, while limited in extent has the potential to generate heat and hydrogen.

Coolant voiding *(TH2)* is calculated to occur rapidly for large break LOCA. Typically, channels are calculated to become highly voided in the first two seconds following the break and could remain essentially steam filled until ECC delivery is underway. Any phase separation *(TH3)* effects in this period will exist for only a short period. Phase separation during the late blowdown period could result in the development of stratified flow, which results in higher circumferential temperature gradients around the pressure tube. Fuel channel thermo-mechanical analysis addresses the effects of these issues on pressure tube strain behaviour.

With regard to heat transfer from the fuel during the early stage of blowdown (i.e., prior to ECC initiation), the cooling conditions rapidly become governed by a combination of convective heat transfer *(TH7)* to single-phase superheated steam, radiative heat transfer *(TH11)* and thermal conduction *(TH6)*. The increase in coolant void *(TH2)* during the initial rapid depressurization reduces coolant density and accordingly induces a small negative reactivity.

The Zircaloy-water reaction *(TH13)* has the potential to affect the sheath heatup and hydrogen generation for high sheath temperatures. However sheath temperatures do not reach levels at which the contribution of the exothermic reaction to fuel heatup becomes significant as discussed in Section 3.4.4.

The important phenomena influencing the late blowdown cooling phase and ECC injection phase are the break discharge characteristics (*TH1*), convective heat transfer (*TH7*), condensation heat transfer (*TH10*) and the quench and rewet characteristics (*TH12*). During the late blowdown cooling/ECC injection phase, the break discharge (*TH1*) continues to drive the system depressurization and flow patterns. Convective heat transfer (*TH7*) continues to be the main heat transfer mechanism from the fuel sheaths to the coolant, up to the time that ECC water arrives. When the ECC water initially arrives in the headers it condenses the steam (*TH10*) and causes the system pressure to drop further thus enhancing ECC flows. As ECC water begins to come into contact with the hot feeder piping, and subsequently the hot fuel sheaths, quench/rewet behaviour (*TH12*) of the hot surfaces occurs, resulting in a rapid drop in surface temperature, accompanied by the generation of steam.

ECC delivery to the heat transport system is determined solely by the pressure differential between the ECC high pressure tanks and the inlet headers. Delivery of ECC to the individual channels is governed by the pressure differentials that exist across the feeder connection points to pairs of headers, the extent and duration of phase separation *(TH3)* effects in the reactor headers, and the potential for hot wall delays, associated with counter-current flow *(TH15)* that has the potential to limit the penetration of water down individual feeder pipes.

For most large breaks, the delay of fuel channel refill due to hot wall delay effects does not play a significant role in large LOCA behaviour. ECC flow to the inlet headers and the availability of the interconnect between the outlet headers provides good flow circulation and facilitates refilling and ensures fuel rewet. When sufficient ECC water has entered either the inlet or outlet feeder of a channel, the hydrostatic head in this feeder drives the steam out from the channel through the opposite feeder and quench/rewet of the fuel sheaths proceeds. For a narrow range of break sizes, the so-called stagnation breaks, flow stagnation may occur in the fuel channels and feeders may become hot. For these break sizes, hot wall delay and related flooding phenomena may become important.

For breaks located at the inlet side of the HTS (e.g., inlet header break), an intermittent density driven flow may develop following core refill and HTS pumps trip. This would occur when the pressure drop across some channels approaches zero, and accordingly, flow temporarily stagnates in these channels. This stagnant subcooled condition is called standing start. This condition will lead to coolant evaporation followed by steam venting into one of the feeders. As a consequence, the channel refills again and the process repeats itself into intermittent flow cycles providing sufficient fuel cooling for the affected channels.

3.4.4 Fuel and Fuel Channel Thermal-Mechanical Effects

During the first phase of a large break LOCA, prior to reactor trip, there is a potential for an increase of fuel temperatures. If fuel temperatures were to increase sufficiently, molten UO_2 could form and could potentially lead to pressure tube rupture. However, a reactor trip occurs quickly enough to limit fuel temperatures and assure a large margin to fuel melting.

During the blowdown phase, the main channel integrity safety concern is the potential for the formation of local hot spots on the pressure tube by fuel element or bearing pad contact. If such a hot spot were to develop on the pressure tube, local strain to rupture could potentially occur. The key fuel and fuel channel parameters to assess the potential for local PT rupture due to hotspot formation are sheath/end cap temperatures, axial expansion/contraction of the pressure tube, and pressure tube temperatures. The phenomena which govern hotspot formation, are bundle mechanical deformation (*FC12*), heat transfer between the bearing pad or sheath and the pressure tube *(FC22*), sheath-to-coolant and coolant-to-pressure tube heat transfer *(FC13)* and element-to-pressure tube radiative heat transfer *(FC21)*.

Sheath non-uniform heating and the difference in sheath heat-up rate compared to the rate of pressure tube heat-up, determine the risk of fuel element (FE) contact. The dominant phenomena influencing fuel and sheath heatup behaviour are energy stored in the fuel (governed by fission and decay heating *(FC1)*), heat diffusivity in the fuel *(FC2)*, fuel-to-sheath heat transfer *(FC3*), fission gas release to the gap and internal pressurisation *(FC5)*, sheath-to-coolant-to-PT heat transfer *(FC13)*, and the initial state of the fuel which determines the fundamental thermal-mechanical properties, such as UO_2 thermal conductivity, and thermal expansion coefficients. Typically, during the early stages of blowdown of a critical break LOCA which results in channel flow stagnation, radiative heat transfer *(TH11)* and convective heat transfer to the pressure tube *(TH7)* are of the same order of magnitude. For non-critical large LOCA, the convective heat transfer *(TH7)* between the coolant and pressure tube will tend to dominate.

Coolant flow mixing and bypass (*FC14*) occurs in fuel bundles or at the junctions of the bundles and also affects fuel temperatures. This phenomenon is important for aged plants where a portion of coolant flow could bypass fuel bundles due to pressure tube diametral creep that could result in a larger upper gap between the bundles and pressure tubes.

Zirconium oxidation by steam (sheath oxidation *(FC9)*) begins to contribute slightly to fuel heating at temperatures in the vicinity of 800°C, and provides a significant contribution to fuel heat-up when fuel sheath temperatures rise above approximately 1200°C. This exothermic chemical reaction is governed primarily by fuel sheath temperature, since the fuel sheath is the hottest zirconium surface in the fuel channel, and by the supply of steam due to coolant flows in the fuel channels. However, since sheath temperatures rise above 800°C during the blowdown phase only over a short period of time, the contribution of the exothermic reaction to fuel heat-up is insignificant.

Sheath failure *(FC7)* is a phenomenon which governs the release of fission products from the fuel. A potential sheath failure mechanism is sheath strain at high temperature *(FC8)* driven by the pressure differential between the internal fission gas pressure *(FC5)* and the channel coolant pressure, which reduces rapidly during blowdown. The sheath temperature transient is governed by the phenomena described above.

The fuel-to-sheath heat transfer *(FC3)* reduces due to fuel sheath strain driven by coolant depressurization and contraction of the fuel pellet after reactor trip.. The reduction of fuel-to-sheath heat transfer acts to decrease the fuel sheath temperature and accordingly results in reducing pressure tube temperatures by reducing both the heat transfer to the coolant and the radiative heat transfer to the pressure tube. The reduction in fuel-to-sheath heat transfer *(FC3)* increases the fuel heat-up, which has a corresponding effect on the potential for fuel deformation *(FC8)*, sheath failure *(FC7)* and fission product release.

3.4.5 Moderator and Shield System Thermalhydraulics

Since the moderator system is of most importance during LOCA events in which ECC is unavailable, the description of governing phenomena for both small and large LOCA events is provided in Section 10.4.5.

3.4.6 Fission Product Release and Transport to Containment

At the beginning of the accident, the inventory and distribution of fission products in the core are defined by the irradiation history of each fuel element (see Section 2.2.6 on pre-transient modelling requirements). During the blowdown phase the fuel elements may experience an increase in their temperature due to cooling degradation and the associated fuel sheath temperature increase. The UO_2 in fuel elements with intact sheaths will experience non-oxidizing conditions. Fuel with failed sheaths may experience oxidizing conditions due to oxygen in equilibrium with steam from steam dissociation.

If the sheath fails, the gaseous fission products in the fuel-sheath gap will escape to the HTS. Gaseous fission products released from the fuel elements will be transported to the break location and released into containment through the break. During transport, some fission products may participate in one of several types of interactions, such as deposition on the HTS surfaces, chemical reactions, aerosol formation, etc. These mechanisms may inhibit or delay the release of certain fission products.

In the refill phase, the fuel temperatures decrease further. Consequently, additional fuel pellet cracking may occur due to the fast cooling caused by the refill. In addition, the fuel-sheath gaps of failed fuel elements will be filled with water after the fuel temperature decreases and the fission product releases from the fuel grains will be terminated. The Iodine and other soluble fission products in the fuel-sheath gaps, on pellet surfaces and grain boundaries will continue to be released to the HTS and transported to the break.

The fission product release and transport phenomena that contribute to the quantity of fission products released into the HTS during the large LOCA accident category are discussed for each accident phase separately.

3.4.6.1 Blowdown Phase

Fission Product Release

During both the early and late blowdown phases, the primary fission product release phenomena are diffusion *(FPR2)*, grain boundary sweeping/grain growth *(FPR3)*, grain boundary coalescence/tunnel interlinkage *(FPR4)*, gap transport *(FPR7)*, gap retention *(FPR8)* and UO_{2+x} formation *(FPR9)*. Releases associated with the phenomenon of UO_{2+x} formation are significant only for the case of previously defected fuel. For all failed fuel, gap transport *(FPR7)* and gap retention *(FPR8)* will have primary roles.

Because of the short time of fuel exposure to high temperatures, secondary contributions to the total releases are expected from fission product vaporization/volatilisation *(FPR15)*.

Fission Product Transport

Depending on the break size and location, some of the high-powered channels may be voided during the blowdown phase of a large LOCA. Some of the HTS components between the channels and the break location (e.g., end-fittings, feeder pipes, headers, and steam generators) may be partially or completely filled with liquid water. Thus, various fission product phenomena are important for different fission products and for different locations in the HTS. Fission products and structural materials will be released in vapour form from the fuel elements with defected/failed sheaths. The noble gases will not be affected by any transport phenomena other than simple gas transport considerations. The chemical speciation *(FPT23)* of some fission products is a primary phenomena as structural materials will allow their nucleation as aerosols when the temperature drops (in the coolant stream near the pressure tube, or when the coolant stream passes into a cooler region of the channel or feeder).

Fission product compounds may also deposit directly on surfaces from the vapour phase *(FPT2)*, once the appropriate condensation temperature is reached. Revaporization *(FPT2)* may occur if the surface temperature increases or the gas composition changes significantly.

In the end-fittings and feeder pipes, the temperatures will probably be suitable for interaction of fission products in the vapour phase with structural materials *(FPT3)*; most notably, Cesium, Iodine, and tellurium compounds will react with stainless steel, carbon steel, Inconel alloys and their oxides. The rates of these interactions will depend on the temperature, the chemical condition of the surface, and chemical speciation *(FPT23)* of the fission products.

Pool scrubbing *(FPT21)* of fission product aerosols and vapours will be important during flow through a partly-voided header or other water-containing components of the HTS between the channels and the break.

The deposited fission products may be released from the HTS when liquid water flows over the pipe surface *(FPT22)*. Liquid flows will occur during flow transients, or when emergency core cooling reaches the channel (see refill phase below). The dissolved or resuspended fission products will be transported out of the HTS in the break discharge flow.

Other phenomena occur at secondary importance level. Brownian motion agglomeration (*FPT6*) and deposition *(FPT14)* will also occur. Fuel particulate material may be formed from fuel elements that exhibit extensive sheath degradation, and can be transported *(FPT1)* by very rapid gas flows or by liquid flow.

3.4.6.2 Refill Phase

Fission Product Release

Because of the low fuel temperatures during refill, most release phenomena are ineffective during this phase. The primary contributors to releases are fuel cracking *(FPR6)*, gap transport *(FPR7),* gap retention *(FPR8),* and fission product leaching *(FPR19*). Fuel cracking *(FPR6)* occurs when the fuel is quenched from high temperatures, and the other phenomena apply to releases from failed fuel in liquid water.

Fission Product Transport

The primary phenomena governing fission product transport in the HTS during the refill phase are pool scrubbing *(FPT21)* in partially water-filled components between the channel and the break, and transport of deposits by water *(FPT22)* after the channel has refilled with water. Chemical speciation *(FPT23)* will only affect whether the fission products are transported in solution or as suspended particles.

3.4.7 Containment (Containment Thermalhydraulics, FP Chemistry, Hydrogen, Aerosols)

The primary safety concern with respect to containment is containment integrity under internal pressurization. The primary phenomena determining containment pressure relate to the total inventory and energy content of steam and water discharged during the event and the discharge rate. The coolant inventory depends on the behaviour of the heat transport and ECC systems. Since energy discharge *(C1)* from the primary circuit continues indefinitely, heat transfer to vault coolers *(C6)* and building structures *(C3, C4, C5)* are of primary importance in limiting the peak containment pressure and establishing the transient pressurization profile for containment. Another significant heat transport process is evaporation from pools *(C2)*. Laminar/Turbulent leakage flow *(C8)* and choked flow *(C9)* will control release rates form containment at high pressures. Heat transfer to the containment walls develops stresses in the structure over a long time period. These stresses are of secondary importance since they develop slowly because heat transfer through concrete is relatively slow compared with the discharge from the cooling circuit.

Since the large LOCA event discussed in this section does not involve failure of the ECC functions, phenomena related to fission product chemistry, hydrogen and aerosols are relatively unimportant. The blowdown jet influences the formation of liquid aerosols *(C37, C38)* and subsequently the transport of fission products in containment. With ECC available, little hydrogen is generated in the HTS and therefore hydrogen behaviour in the short term for this accident scenario is unimportant. However, a significant amount of hydrogen may be generated in the longterm by radiolysis of the coolant-ECC mixture. This hydrogen enters containment. The hydrogen-related containment phenomenon of primary importance is consequently removal by recombiners *(C17)*. Fission product releases to the coolant from failed fuel sheaths are transported through the break into the containment. Soluble fission products are carried with the coolant to the floor of the containment building. Some non-soluble fission products become airborne as aerosols. These fission products plate out on the walls and the internal surfaces of the containment. Some of the fission products will decay and some of them may leak through the containment wall during the containment over pressure period.

Iodines released into containment from failed fuel are highly soluble and will enter containment as iodide dissolved in the liquid phase. However, Iodines are also reactive. Organic iodide can be produced in the containment water pool or in the liquid film on the containment surfaces. Once organic iodide is produced, it comes out of solution, enters the containment atmosphere as a gas and is not easily removed from the atmosphere by any removal process (other than decay). The Iodine chemistry related phenomena of primary importance for large LOCA are interfacial mass transfer (*C18*), carbon filter removal efficiency *(C21)*, total waterborne Iodine *(C22)*, and surface adsorption.

Fission product attenuation from the containment atmosphere is governed by processes relating to water aerosol retention and include aerosol removal due to jet impingement *(C25)*, and gravitational settling *(C26)*. Fission product release from containment is also governed by these attenuation processes as well as phenomena relating to Iodine behaviour (gas-phase and water-borne Iodine) leakage from containment.

Section 10.4.7, which involves ECC failure, includes additional discussion of phenomena of more importance to containment for that accident.

3.4.8 Radiation Physics

The initial state of the reactor is set by events preceding initiation of the accident, yielding the radionuclide inventory in the fuel *(RAD1 and RAD2)*. Shutdown of the reactor is typically initiated by process trips and terminates the production of new fission products, but radioactive decay continues. Radiolysis *(RAD8)* and the generation of decay heat *(RAD6)*, which are time dependent, continue as they would normally, but since the HTS normal cooling mode is compromised, the decay heat phenomenon is of primary importance. Other phenomena have little influence on the radiation status of the plant following shutdown. Most radioactive material remains within the core following the LOCA when there is effective ECC. Thus shielding considerations are not significantly different than they would be normally.

Section 10.4.8, which includes failure of the ECC, provides additional discussion of radiation physics phenomena of greater significance to that accident.

3.4.9 Atmospheric Dispersion

Atmospheric diffusion is primarily governed by external atmospheric conditions as discussed in Appendix A. The large LOCA conditions relevant to dispersion include phenomena related to specification of the source term released to the atmosphere, the heat content of any release and the location of leakage paths through the containment structure, as these input factors from containment analysis affect dispersion via the phenomena of plume rise (buoyancy, *(AD1)*) and advection.

3.5 Summary

The large LOCA accident scenario is driven primarily by heat transport system depressurization and the associated degradation in fuel cooling. The blowdown phase and the associated fuel heat up is terminated on ECC injection.

Shutdown system-induced reactivity is the dominant influence on neutron kinetics, power and flux distributions in the reactor core. Fuel channel and system thermalhydraulics phenomena of most concern involve break discharge characteristics, fuel to coolant heat transfer regimes, and the delivery of emergency core coolant. Fission product releases are determined primarily by the number of fuel elements that fail, their normal-operation gap inventories, irradiation histories and transient temperature histories of previously defected fuel elements.

4. SMALL LOSS OF COOLANT ACCIDENTS AND SINGLE CHANNEL EVENTS

4.1 Description of Accident

A small break LOCA refers to a break where the operating reactor regulating system is capable of preventing a significant reactor power deviation. A small LOCA may occur due to a small break in a large diameter pipe (e.g., the header) or up to a 100% break of other smaller HTS pipes, including feeders and steam generator tubes. A special class of small break LOCA events are events that primarily affect a single channel. The single channel events in this chapter include end-fitting failure, feeder break, partial flow blockage and spontaneous failure of a pressure tube. Certain low probability single channel events fall into the category of limited core damage events covered in Chapter 10, including stagnation feeder break and severe flow blockage.

Single-channel breaks may occur outside of the core (feeders, end-fittings and other external piping) or within the core. Breaks within the core fall into two categories. In one category, only the pressure tube fails, leading to break discharge to containment via the PT/CT annulus after bellows failure. In the other category, the entire fuel channel (pressure tube and calandria tube) fails in which case the discharge to containment is via the moderator rupture discs and pressure relief ducts.

A single channel event or small LOCA is characterised by the following general features:

1. No significant power change. Reactor shutdown systems may not detect the failure immediately so that the reactor continues to operate at or slightly below full power.
2. A low rate of coolant discharge from the break into containment, moderator or other reactor subsystems, such as steam generator secondary side, depending on the break location.
3. Fuel failures do not occur or are restricted to a single affected channel.
4. Increases in the heat load to the moderator via coolant discharge are possible in the event of channel failure.
5. For breaks within the coolant makeup capacity, coolant feed automatically replaces the lost HTS inventory and the pressurizer level does not start to drop until the coolant storage tank is depleted. For larger breaks, the pressurizer level starts to drop on the initiation of the break. Eventually, reactor shutdown is initiated on a reactor trip parameter, for example, low HTS pressure or high containment pressure, temperature or activity. ECC is initiated on low heat transport system pressure.
6. For very small breaks, reactor shutdown and cooldown is in a controlled manner to avoid ECC activation. For larger breaks, ECC is initiated to cool and refill the HTS.
7. The maximum containment pressure is lower than for large LOCA due to the lower energy discharge rate. The initial pressurization period is followed by depressurization to near atmospheric conditions.

Additional features of importance to some accident categories include:

a) Ejection of fuel bundles into containment in the case of an end-fitting failure, or fuel elements into moderator for PT/CT failure.

b) Condensation of coolant by the cool moderator in the case of an in-core break.

c) Potential in-core structural damage for channel failure events.

d) Leakage of primary coolant into the steam and feedwater system in the case of a steam generator tube failure.

4.2 Key Safety Concerns

The safety concerns relevant to small LOCA events are:

- Public dose related to potential fission product releases from the fuel,
- Maintenance of core integrity following single fuel channel failures,
- Reactor trip effectiveness, and
- Containment integrity related to pressurisation.

4.3 Accident Behaviour

4.3.1 Introduction

Quantification of the consequences associated with the above safety concerns involves analysis of phenomena that influence physical processes and determine system behaviours during a small LOCA or single channel accident. These system behaviours 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 and ECC system initiation. Therefore, uncertainties in 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 phenomena exert a governing influence.

This discussion focuses on those small breaks that are sufficiently large that they lead to automated reactor trip and shutdown. Very small breaks that are amenable to detection after an extended time and manual shutdown are not formally considered, because their progression is benign and there are minimal uncertainties in the technical bases for quantifying the safety consequences within regulatory limits.

4.3.2 Phases of the Accident

The phases of a small LOCA or single channel accident are defined according to the major time periods during the accident progression in which characteristic system behaviours are exhibited. For each of the major disciplines involved in small LOCA analysis, the phases of the accident are defined and the dominant behaviours during these phases are identified.

4.3.2.1 Reactor Physics

1. *Pre-Shutdown* - the initial period of the event, lasting up to several minutes after the break initiation, prior to reactor shutdown. During this period, the RRS attempts to counteract the neutronic transient if any and maintain the reactor power constant. This phase is ultimately terminated by shutdown system action following a reactor trip on high containment pressure, low coolant pressure, or low flow. For an in-core break, the discharged light water coolant displaces the heavy water moderator. If the moderator contained a low level poison, this poison concentration would be diluted, but the net result is a decrease in reactivity because of the added H_2O in the core. Also shut-off rod guide tubes could become damaged due to impact of the coolant jet and any discharged fuel.

2. *Post-Shutdown* - the period following reactor shutdown. During this period, due to shutdown-system action, the reactor is brought to a subcritical state in which the spatial neutron flux distribution stabilizes and the power distribution becomes governed by decay heat.

4.3.2.2 Trip Coverage

Trip coverage analysis for a small LOCA is usually performed with the break in the reactor header, resulting in a discharge of coolant into containment.

The pressure and inventory control system responds to maintain nominal conditions. For break sizes beyond the capacity of this system, the HTS has a net inventory loss and depressurizes.

The loss of inventory causes voiding in the core which produces a negative reactivity feedback. The reactor regulating system, if available, acts to attempt to keep the power constant until trip. For the cases where failures of RRS are postulated, power will decrease due to voiding.

The reactor trips on one of the process trips (high reactor building pressure, low HTS pressure, or low flow) on one of the shutdown systems.

4.3.2.3 Fuel Channel and System Thermalhydraulics/Fuel and Fuel Channel Thermal-Mechanical Effects/Fission Product Release and Transport

1. *Full Power* - the initial period before and following an initiating event. The reactor power remains essentially constant as a result of regulating system action countering local neutronic transients.
2. *Blowdown Cooling (post shutdown to ECC initiation)* - the period during which the heat transport system blowdown occurs prior to ECC initiation. The dominant HTS behaviour during this period is depressurization and blowdown cooling.
3. *Refill (ECC initiation to thermosyphoning)* - the period during which refill of channels in the core proceeds and a quasi-steady-state is attained. The dominant HTS behaviour during this period is a refill controlled by ECC delivery, pump trip, possible "standing start"(see Section 3.4.3), fuel cooling and fission product release.
4. *Thermosyphoning* - the dominant HTS behaviour is stable cooling via single-phase thermosyphoning in the primary circuit. Cooling is dominated by convective flow established by heat transfer to the steam generators.
5. *Special cases* - small breaks and a number of single channel events involve unique behaviour.

    a) *End-Fitting Failure* - Stages 1 to 4 are complicated taking into account the potential for discharge of the fuel from a channel into the containment atmosphere. Cooling of that fuel is maintained by heat transfer via convection and radiation in air-steam-water mixtures.

    b) *Flow Blockage/Feeder Breaks* - phases of the circuit transient are those described in stages 1 to 4 above. The reduced flow in the impacted channel will result in degradation of fuel cooling.

c) *Pressure Tube Failure* - the event is initiated by a spontaneous pressure tube failure. This results in a small break LOCA where the coolant loss is limited by restricted flow through the end-fitting/lattice tube annulus and failed end-fitting bellows.

d) *Channel Failure* - This event is governed by mechanical behaviour influenced by calandria material impact strength, strength-temperature relationships and material failure phenomena. CT Failure may occur promptly as an immediate consequence of a spontaneous PT failure, or after a delay governed by thermal-mechanical phenomena of the CT. Break discharge will result into pressurizing the moderator and consequential flow out the failed calandria rupture discs. Primary circuit behaviour develops as in phases 1 to 4 as described above.

e) *Single Steam Generator Tube Failure* – Phases 2 and 4 occur through controlled cooldown followed by valving in the Long Term Cooling (LTC) system. As the event leads to a coolant discharge that is within the coolant feed system capacity, refill of the channels through the use of the ECC system is not needed and phase 3 is bypassed.

4.3.2.4 Moderator and Shield System Thermalhydraulics

The moderator thermalhydraulic behaviour is only a factor in the event of pressure tube or fuel channel failures.

1. *Pressure Tube Failure* – upon flooding of the gap annulus in a fuel channel, the heat transfer to the moderator will be enhanced locally.
2. *Channel Failure* - the moderator will be pressurized on break discharge followed by rapid break discharge through the rupture discs.

4.3.2.5 Containment

Discharges of coolant from small breaks may be sufficiently small such that local air coolers or other energy sinks may preclude significant pressurization of containment. ACR containment is ventilated during normal operation. Initiation of containment isolation depends on signals indicating rising pressure, temperature or radioactivity in containment.

1. *Ventilation* - the dominant containment behaviour during this period is governed by break discharge and flashing into containment. The operation of the ventilation system and condensation of steam by coolers and cool surfaces helps in reducing over-pressurization of the containment. Depending on the rate of discharge, this phase ends when a containment isolation trip signal occurs.
2. *Initial Pressurization (isolation)* - containment is isolated from the outside atmosphere and containment pressurization behaviour is determined by the rates of break discharge flashing, heat removal by the local air coolers, and heat transfer to the containment walls.
3. *Long-Term Period* - the dominant behaviour during this period is due to slow pressurization, containment leakage to the external atmosphere, containment heat rejection, and behaviour of the hydrogen generated by radiolysis.

4.4 Governing Physical Phenomena

The physical phenomena governing system behaviour during single channel events are grouped with respect to discipline and dominant behaviour.

4.4.1 Reactor Physics

The reactor physics phenomena, which play a role in small LOCA outside the core, are discussed under Trip Coverage in Section 4.4.2. This section focuses on the single channel events that result in in-core LOCA.

The primary reactor physics phenomena for both phases (see Section 4.3.2.1) of single channel in-core breaks are the device-movement induced reactivity *(PH11)* and the flux and power distribution in space and time *(PH14)*. Specifically for in-core breaks, the primary phenomena in both phases also include the moderator-temperature-change induced reactivity *(PH4)*, the moderator-poison-concentration-change induced reactivity *(PH5)* and the moderator-purity-change induced reactivity *(PH6*).

The moderator-temperature-change induced reactivity *(PH4)* is a primary phenomenon for single channel events where hot coolant is discharged into the moderator. In such cases, the moderator temperature rises due to the fluid mixing and this has a negative reactivity effect.

The moderator-poison-concentration-change induced reactivity *(PH5)* is a primary phenomenon for an in-core LOCA when the moderator contains some poison. As injection of the H_2O coolant displaces heavy water (out the pressure relief ducts), and reduces the poison concentration, the negative reactivity effect of the poison is reduced. However, the net result is a decrease in reactivity because of the added H_2O in the core.

The moderator-purity-change induced reactivity *(PH6)* is also a primary phenomenon for single channel events where heat transport coolant is discharged into the moderator. In these cases, the moderator heavy water purity decreases significantly due to injection of the H_2O coolant which displaces heavy water. This has a large negative reactivity effect.

The device-movement induced reactivity *(PH11)* is a primary phenomenon during both the pre-shutdown and the post-shutdown phases. For the pre-shutdown phase, reactor-regulating-system response is significant. This may include mechanical zone controllers or mechanical control absorbers being inserted. For the post-shutdown phase, the effect of the shutdown system is dominant.

The flux and power distribution in space and time *(PH14)* is changed by, and affects, the device movement and the thermalhydraulic conditions during the pre-shutdown phase. During the post-shutdown phase, decay heat becomes the dominant factor.

4.4.2 Trip Coverage

The primary phenomena are the HTS depressurization *(TH1)* and the channel voiding *(TH2)*, as well as the device-movement-induced reactivity *(PH11)*. The channel voiding as well as the coolant-density-change-induced reactivity affects the timing of the process trips. Some process trips are affected significantly, e.g., low HTS pressure, while others (e.g., high reactor building pressure and low flow) are less affected by the amount of voiding. CHF (dryout) and post dryout *(TH9)*, break discharge *(TH1)*, nucleate boiling *(TH8)*, and convective heat transfer *(TH7)* play an important part in determining trip effectiveness.

4.4.3 Fuel Channel and System Thermalhydraulics

In general, for small breaks in large pipes, the dominant phenomena are similar to those identified for large LOCA (Section 3.4.3). Zircaloy-water reaction *(TH13)* has a less significant role in small LOCA events because sheath temperatures are lower. On the other hand, in small LOCA events with loss of Class IV power or after HTS pumps trip, density driven flows *(TH17)* play a more important role than large LOCA.

The dominant phenomena during single channel events are dependent on the nature of the initiating event and could include channel voiding *(TH2)*, and fuel heat-up. Coolant voiding is determined by heat transfer from the sheath to the coolant. The important phenomena in this mode of heat transfer are CHF (dryout) and post-dryout *(TH9)*, and nucleate boiling *(TH8)*. These phenomena are related to the affected channel. Heat transfer to the coolant *(TH6, TH7)* remains a dominating overall phenomenon, providing adequate cooling to the core with the exception of the affected channel. Similar to large LOCA events, the phenomena, phase separation *(TH3)*, condensation heat transfer *(TH10)*; radiative heat transfer *(TH11)* and quench/rewet characteristics *(TH12)* can play an important role in the event transient.

The heat transport system depressurization behaviour is determined by the state of the fluid, the rate of coolant discharge (governed by break discharge characteristics and critical flow *(TH1)*), and the coolant mass and energy redistribution within the heat transport pipes and components. The coolant mass and energy redistribution occurs due to the combined effects of the transient pressure gradients and the heat transport pump pressure heads and flows (governed by HTS pump characteristics *(TH5)*).

For a given inlet feeder break location, and break sizes below a particular size, the flow in the affected channel will be maintained in the normal forward direction. At a “critical” break size in an inlet feeder, the flow is balanced by the flow discharge out the break, resulting in flow stagnation in the affected channel. The critical break event (stagnation feeder break) is included in the limited core damage accidents section (Chapter 10). For break sizes above the critical break, part of the channel will experience reverse blowdown flows, while the other part will remain in forward flow for a period of time causing significant void generation in the channel. Larger break sizes result in flow reversal in the affected channel.

Depressurization behaviour proceeds much as for the large LOCA case, but much more slowly. ECC delivery conditions are also similar to those of the large LOCA event but are protracted in time. Related phenomena are discussed in Section 3.4.3.

The major difference between single channel events and large LOCA relates to the possibility of continued full power operation with cooling to the affected channel degraded due to feeder break or flow blockage. This and other aspects significantly influence the affected channel behaviour.

4.4.4 Fuel and Fuel Channel Thermal-Mechanical Effects

In general, phenomena relating to large LOCA discussed in Section 3 are also relevant to small LOCA. As mentioned in Section 4.3, there are some additional specific system responses related to small LOCA. A notable difference in behaviour is that some single channel breaks have relatively little influence on reactor physics so some channels may be exposed to normal (high) power level for longer time periods as shutdown signals are slow to develop.

4.4.4.1 End-Fitting Failure

For the case of an end-fitting failure, the fuel may be ejected from the reactor and cooled only by air-steam-water mixtures. One of the key parameters governing fuel cooling is the geometry of the fuel bundle. The related radiative and convective heat transfer phenomena are well described in the literature.

4.4.4.2 Fuel Channel Failure

Should the failure of both a pressure tube and calandria tube *(FC19*) occur coincidentally, additional phenomena become important. Ruptured channel projectiles *(FC24)* and/or reaction forces including moderator hydrodynamics *(FC23)* from the failed channel have the potential to lead to failure of additional channels or to interfere with shutdown system *(FC25).*

Fission product release and transport phenomena associated with this event are discussed in Section 4.4.6.3.

High-temperature channel components (specifically fuel elements and fuel bundle components) can potentially be expelled into the moderator for the CANDU reactor design following channel rupture. The heat addition to the subcooled moderator due to Fuel-to-Moderator Interaction (FMI) could impact the hydrodynamic transients within the moderator. The intensity of the hydrodynamic transient is primarily determined by the rate at which the channel debris is delivered to the moderator water, and the rate of heat transfer from the debris to the moderator. Debris fragmentation influences the surface area available for interaction between the debris and moderator accordingly impact on the heat addition to the moderator and the subsequent magnitude of the hydrodynamic transients.

4.4.5 Moderator and Shield System Thermalhydraulics

The moderator system is of most importance during LOCA events in which ECC is unavailable, a discussion of the governing phenomena for both small and large LOCA events is provided in Section 10.4.5. However, some of the phenomena that are of primary importance for the large LOCA and LOCA with LOECC events are of secondary importance for small LOCA and single channel events.

4.4.6 Fission Product Release and Transport to Containment

At the beginning of the accident, the inventory and distribution of fission products in the core are defined by the irradiation history of each fuel element (Section 2.2.6).

4.4.6.1 Off-Stagnation Feeder Break

Off-stagnation feeder breaks result in low flow in the channel while maintaining channel integrity for a relatively extended period of time (i.e., sheath melting does not occur and the terminating event is not channel failure). The heat-up rate is slower than for the stagnation feeder break case. However, the duration of the event is long enough to cause fuel failures and significant release of fission products from the fuel. The fission products are transported in the two-phase steam-water flow to the break and released into containment.

The high-powered elements are expected to fail within about one minute for the worst-case off-stagnation break. Following sheath failure, steam exposure oxidizes the UO_2 to UO_{2+x}. The degree of oxidation is a function of time, temperature, exposed UO_2 surface area, and the amount of oxygen in the steam. The equilibrium deviation from stoichiometry, which is the highest achievable oxidation state, depends on the fuel temperature and the amount of oxygen in the steam.

In hyperstoichiometric UO_{2+x}, fission gases diffuse through the grains to the grain boundaries at an increased rate. This change in the fuel composition enhances the mobility of both gas atoms and intragranular bubbles. UO_2 oxidation in steam has also been reported to cause accelerated grain growth, which leads to enhanced release of fission products by grain boundary sweeping.

Fuel heat-up following an off-stagnation break is terminated by reactor shutdown (manual or otherwise). The fuel in the broken channel will be rapidly cooled as the channel rewets. Thermal shock following rewet may cause failure of fuel sheaths, as well as cracking and/or powdering of fuel pellets exposed to the rewet flow.

The off-stagnation feeder break may be segregated into three stages: Prior to Sheath Failure, Following Sheath Failure, and Rewet.

4.4.6.1.1 Prior to Sheath Failure

Fission Product Release

During this stage of an off-stagnation break, fuel may heat up to a high temperature, however, since the fuel sheath remains intact during this stage, the fuel environment is considered to be non-oxidizing. Prior to sheath failure, the fission product releases enter the fuel-sheath gap. If any previously defected fuel elements are present in the affected channel, oxygen-enhanced fission product release to the coolant begins immediately. Fission products from any defected fuel in the core are released from the break along with the coolant.

The release phenomena of primary importance in this phase include: diffusion *(FPR2)*, grain boundary sweeping/grain growth *(FPR3)*, grain boundary coalescence/tunnel interlinkage *(FPR4)*, vapour transport/columnar grains *(FPR5)*, temperature transients *(FPR17)*, and grain boundary separation *(FPR18)*.

If fuel with sheath failures is present in the affected channel during the pre-transient phase, UO_{2+x} formation *(FPR9)*, gap retention *(FPR8)* and gap transport *(FPR7)* in the failed fuel will be primary release phenomena for the failed fuel.

Fission Product Transport

The fission product transport phenomena of primary importance in this phase are pool scrubbing *(FPT21)* and transport of deposits by water *(FPT22)*. The transport properties are determined by the character of the fission products released from previously defected fuel (mainly noble gases and Iodine). Gaseous and aerosol fission products will be partitioned between the liquid phase and gas phase in partially water-filled primary circuit components (e.g., end-fitting, feeder pipe).

4.4.6.1.2 Following Sheath Failure

Fission Product Release

During this stage of the accident, the release phenomena of primary importance are diffusion *(FPR2)*, grain boundary sweeping/grain growth *(FPR3)*, grain boundary coalescence/tunnel interlinkage *(FPR4)*, vapour transport/columnar grains *(FPR5)*, UO_{2+x} formation *(FPR9)*, fuel melting *(FPR14)*, matrix stripping *(FPR16)*, temperature transient *(FPR17)*, and fission product vaporization/volatilisation *(FPR15)*.

The secondary phenomena include: athermal release *(FPR1)*, gap transport *(FPR7)*, gap retention *(FPR8)* and UO_2-Zircaloy interaction *(FPR12)*.

Fission Product Transport

The fission product transport phenomena for this phase are similar to the phenomena for the Heatup (following sheath melting) phase of the Stagnation Feeder Break/Severe Flow Blockage Section (10.4.6.2).

4.4.6.1.3 Rewet

Fission Product Release

The release phenomena of primary importance during this stage of the accident are fuel cracking *(FPR6)*, gap transport *(FPR7)*, gap retention *(FPR8)*, and fission product leaching *(FPR19)*.

During this stage of the accident, the fuel heat-up is terminated by reactor shutdown. The fuel in the affected channel will be rapidly cooled due to rewet, causing fuel cracking *(FPR6)*. The other phenomena are associated with releases from failed fuel in liquid water.

Fission Product Transport

Fuel particles from cracking or powdering will be entrained *(FPT1)* in the water flows and rapid steam flows associated with rewet. Aerosols will be resuspended *(FPT20)* by the turbulent steam in advance of the quench front, and deposited fission products will be transported by water *(FPT22)* after the quench front has passed by. Chemical speciation *(FPT23)* is considered to be of secondary importance, as it will only affect whether the fission products are transported in solution or as suspended particles.

4.4.6.2 End-Fitting Failure

In an end-fitting failure event, the fuel in the fuel channel will be ejected into the fuelling machine vault. The behaviour of the fuel in all of the other channels will be similar to that which would be expected for a small LOCA (i.e., no other fuel failures will occur) so that releases from the ejected fuel will dominate accident consequences. The fuel which is ejected will be damaged and oxidized in an air-steam environment. The temperatures may reach the point where the fuel oxidation rate is at a maximum.

Since the fuel from the affected channel is assumed to be ejected from the HTS into containment, fission products from the ejected fuel would be released directly into containment. Initial releases will be directly to the atmosphere. At some time after the initial break, water discharged from the break will cover fuel fragments on the vault floor. The ACR vault floor is at the basement level so that there will be water on the floor (from the ECC system and the HTS inventory).

At the beginning of the event, the inventory and distribution of fission products in the affected channel and the coolant are defined by the irradiation history of each fuel element in the channel, including any defective elements (Section 2.2.6). Iodine and other soluble fission products in the fuel-sheath gaps, pellet surfaces, and grain boundaries will continue to be released into the water.

The fission product release phenomena that contribute to the quantity of fission products released into containment are introduced by accident phase.

Fuel Ejection and Release Phase

During these two phases the primary phenomena are UO_{2+x} formation *(FPR9)*, U_4O_9-U_3O_8 formation *(FPR10)*, and fission product vaporization/volatilisation *(FPR15)*. In the case of previously defected fuel, it is not expected that any additional phenomena will be of primary importance. It is assumed that there will be substantial fuel breakup due to impact such that the above phenomena dominate further fuel degradation.

Secondary phenomena will include gap transport *(FPR7)* and gap retention *(FPR8)*. Fuel damage is expected to be sufficiently severe that the effect of the fuel-to-sheath gap will be minimal, since a large amount of the ejected fuel will not be well contained by the sheath.

Long Term Releases

Because of the low fuel temperatures during this phase, most release phenomena are minimal. A possible contributor to releases is fission product leaching *(FPR19)* to the flowing coolant. Gap retention *(FPR8)* may be of significance as well, for relatively undamaged fuel elements in which the fuel is still contained within the sheath.

4.4.6.3 Spontaneous Pressure Tube/Calandria Tube Rupture

The initiating event for this accident is the spontaneous rupture of a pressure tube and subsequent failure of the surrounding calandria tube. Some of the fuel elements may have sheath failures or be severely damaged to the extent that the UO_2 fuel is ejected from the fuel sheath. Since the fuel elements in the channel or calandria and fragments of fuel in the calandria are well cooled by the coolant and moderator, the fuel temperatures remain low.

Fission Product Release

Following the rupture of the channel, the fission product release phenomena of primary importance are fuel cracking *(FPR6)*, gap retention *(FPR8)*, gap transport *(FPR7),* and fission product leaching *(FPR19)*.

Fission Product Transport

There are no primary phenomena that would dominate fission product transport for this event.

4.4.6.4 Steam Generator Tube Failure

A failure of a single steam generator tube causes the primary coolant to leak into the steam and feedwater system. As consequential fuel failures are not expected for this event, the major risk to the public is from the release of any fission products (radioiodines and noble gases) present in the primary coolant from previously defected fuel. The only source of radionuclides in the primary coolant is the normal burden of fission and activation products and any increase of fission products due to spiking after reactor shutdown.

Fission Product Release

The primary phenomena applicable for fission product release through the secondary system, are diffusion of the radionuclides *(FPR2)*, vapour transport and columnar grains *(FPR5)*, and fission product vaporization/vitalization *(FPR15)*.

Fission Product Transport

Fission products released during a steam generator tube failure are transported outside containment. Fission products can be transported to the environment through the vapour phase (mentioned above), and by bypassing containment.

4.4.7 Containment (Containment Thermalhydraulics, FP Chemistry, Hydrogen, Aerosols)

The primary safety concern related to containment following initiation of small breaks and single channel events is fission product release from containment. In this regard, a primary safety concern is the prevention of containment leakage by the isolation systems, which depend on containment pressure and other signals.

Similar to the large LOCA events, containment pressure/temperature conditions are influenced by such phenomena as break discharge flashing *(C1)*, heat removal by local air coolers *(C6)*, and heat transfer to the containment walls *(C5)*. Heat transfer to coolers and walls are more important for small breaks because of their influence on trip signals. Fission product attenuation inside containment is governed by processes relating to water aerosol retention phenomena and includes aerosol removal due to jet impingement *(C25)*, resuspension of water borne aerosols and removal due to structural leak path retention. All the containment-thermalhydraulics phenomena which are considered important for large LOCA are of primary importance to small breaks and single channel events. Gravitational settling *(C26)* leakage from containment and filter performance are also important phenomena. For events involving in-core breaks, phenomena relating to fission product retention (e.g. *FPR15*, *FPR19*) in the moderator are also important, as they influence the amount and form of fission products entering containment.

For steam generator tube failures the release of radionuclides is effectively outside containment and containment is by-passed.

4.4.8 Radiation Physics

The initial state of the reactor is set by events preceding initiation of the accident, which is primarily concerned with the inventory of radionuclides in the fuel *(RAD1 and RAD2)*. Shutdown of the reactor or depressurization initiates a change of state to decay processes *(RAD6)* governed mainly by time. Other phenomena have little influence on the radiation status of the core following shutdown. Although some radioactive material may be released from the core, this is relatively unimportant to this accident. It is of more relevance for events including LOECC, which are discussed in Section 10.4.8.

Since one of the containment isolation signals is the detection of activity in the vapour recovery and ventilation air ducts, a calculation for the signal at the detector location is needed. The calculation has three aspects, the specification of the radiation source in the duct *(RAD1* and *RAD2*), the transport of radiation from the duct to the detector *(RAD4*), and the conversion of that radiation in the detector to a response of the detector *(RAD5* and *RAD7*).

For single channel events some phenomena such as external exposure (*RAD7*) and radiolysis (*RAD8*) are a primary concern.

4.4.9 Atmospheric Dispersion

Atmospheric diffusion is primarily governed by external atmospheric conditions as discussed in Appendix A. The single channel accident conditions relevant to dispersion include phenomena related to aerosol settling and the location of leakage paths through the containment structure as these input factors from containment analysis affect dispersion through natural convection and settling (Section 3.4.9).

In the case of a steam generator tube failure, atmospheric diffusion can occur through leakage of steam from the secondary side or through opening of the main steam safety valves.

4.5 Summary

Small LOCA events (including single channel accidents) are analyzed in phases established by containment pressurization, reactor shutdown, ECC injection, pressure tube deformation for the affected fuel channel and long-term containment pressure suppression. The effect of a small LOCA on bulk and spatial control can induce global changes in the flux and power distributions mainly through moderator temperature, device movements (RRS), and poison concentration effects. Once the reactor is shut down, however, radioactive decay (rather than fission) is the dominant heat source.

Fuel channel and system thermalhydraulic considerations are very dependent on the specific accident, but heat transfer to the secondary coolant, break discharge rates, and degraded cooling to the affected channel (as the reactor continues at high power), are the important considerations. Fission product release and transport phenomena are also very accident dependent. For the steam generator tube rupture scenario, fission products may bypass containment to the environment.

5. SECONDARY SIDE COOLANT FAILURES

5.1 Description of Accident

Secondary circuit failures include accidents involving the main steam and the feedwater system.

Breaks in the main steam line lead to slow or rapid loss of secondary coolant inventory. With the reactor operating at full power, it is necessary to initiate shutdown and demonstrate that alternative heat sinks are capable of removing decay heat. If the break is inside the reactor building, the discharge also causes the reactor building pressure to rise. The consequences are determined by the amount of time available before an alternative heat sink has to be brought in (if needed), and by the heat sink effectiveness. The alternative heat sinks to feed water, if required, are the Long Term Cooling system (LTC), and steam generator makeup water backed up by a Reserve Water System. The primary cooling system remains intact.

The following general features characterize steam line breaks:

1. Large breaks result in rapid cooling and depressurization of the secondary circuit.
2. Guillotine breaks of the large steam mains within the reactor building give the highest energy discharge rate and greatest total energy discharge of any piping failure in the reactor building. Steam line breaks can also lead to pipe whip, which, for breaks inside containment can jeopardize the containment boundary and internal safety-related components. For breaks outside the reactor building, pipe whip could damage safety support systems.
3. A small secondary coolant-piping break inside containment can also lead to high reactor building pressures.
4. Following large breaks some emergency coolant may be injected to the primary system as it is cooled and pressure falls.
5. The reactor will be shutdown quickly (seconds) for large breaks with variance in trip signals depending on whether the break is inside or outside containment.
6. The primary system remains intact (apart from a requirement to consider steam generator tube leakage) thus limiting the release of radioactive material.
7. Piping failures outside containment may increase pressure and temperature in the turbine building, and steam jets and pipe whip can create harsh local conditions.
8. Piping failures outside containment release any radioactivity which may be present in the secondary circuit, plus radioactivity of the primary coolant which is transferred to the secondary coolant via leaking steam generator tubes.

In general, feedwater breaks are similar to steam line breaks, except that the energy discharge rate is less (i.e., subcooled liquid instead of steam). Therefore, for containment overpressure integrity, feedwater breaks are less limiting than large steam line breaks. Breaks can be symmetric or asymmetric and can occur inside or outside containment. Asymmetric breaks can affect one steam generator more than the other.

A total loss of the main feedwater pumps would lead to an immediate loss of feedwater to both steam generators. For a loss of feedwater flow due to closure of the control valve(s), one or both of the steam generators are affected, depending on the failure (i.e., failure of one control valve, or a total loss of instrument air which would close all control valves).

A loss of secondary circuit pressure control could lead to either steam generator pressurization or depressurization, depending on the exact nature of the failure. Some events require either safety system or operator intervention to mitigate the consequences of the failure. Pressurization of the secondary circuit can be the result of an excessive amount of heat from the reactor or from an inadequate rejection of steam via the regulating valves. Steam generator depressurization can be caused by the heat removal rate via steam discharge exceeding the heat generation rate. This can be caused by the inadvertent opening of either the atmospheric steam discharge valves (ASDVs) or the main steam safety valves (MSSVs).

5.2 Key Safety Concerns

The relevant safety concerns to secondary side events, whose consequences are quantified through safety analysis, are:

- Demonstration of trip effectiveness for satisfactory fuel cooling;
- Establishment of a long term heat sink;
- Pressure tube integrity and heat transport system integrity must be retained for feedwater line breaks and loss of feedwater flow, because of the rise in primary heat transport system temperature and pressure due to degraded boiler heat removal;
- Jet discharges and pipe whip do not damage safety related systems;
- Containment structural integrity; and
- Public doses related to fission product releases due to the break and any steam generator tube leak or rupture which could result in releases bypassing the containment are acceptable.

5.3 Accident Behaviour

5.3.1 Introduction

The behaviour of the secondary cooling system in terms of discharge and depressurization, and the behaviour of the primary cooling system in response to sudden cooling, dominate the response of reactor systems following this accident. In case of control failures, the response of the steam generator pressure and level control, as well as the sizing of the steam relief valves is of primary importance.

5.3.2 Phases of the Accident

For the secondary circuit breaks, important behaviour is generally grouped with respect to whether the piping failure is inside or outside containment. Details of system response and consequences are determined by control system actions.

5.3.2.1 Reactor Physics

The rapid HTS cooling and void collapse associated with this accident scenario would result in a small increase in reactivity due to the small coolant-density-change induced reactivity *(PH1)*. Such increase in reactivity is within the RRS capacity and accordingly reactor power is controlled by RRS. If RRS is impaired, reactor power increase will result in limiting the increase in core coolant density and the resulting reactivity and power increase. However, reactor shutdown is primarily brought about by pressure changes and/or fluid level changes related to thermalhydraulic discharge and heat transfer phenomena.

5.3.2.2 Trip Coverage

For a steam line break, the relevant trips include high reactor building pressure, steam generator low level (SGLL), HTS high pressure, HTS low pressure, as well as the manual trip. The main criteria for this event are: (a) to prevent excessive containment pressurization, (b) to preclude fuel failures by providing adequate cooling, (c) to meet the dose limits for the event and (d) to provide an alternate long term heat sink. Details of the system behaviour are given in the thermalhydraulic section below.

For a feedwater failure (including feedwater breaks, loss of feedwater pumps, and valve closure to the steam generator), the relevant trips are SG low level, reactor building high pressure, HTS high pressure and the manual trip. The acceptance criteria for this event are the same as for the steam line break. However, the steam line break is limiting for the containment overpressurization as well as for the dose limits. The main criteria for this event are to provide adequate cooling and to preclude fuel failures. Again, the detailed system behaviour is described in the thermalhydraulic section.

For the loss of secondary circuit pressure and inventory control failures, the relevant trips are SG low level, HTS low pressure, HTS high pressure, and the manual trip. The high neutron power trip is also effective for the cases where the RRS is assumed to be frozen and power is allowed to increase. In some cases, the system reaches a new “steady state” and a trip is not required. The main criteria for this event are the same as for the steam line break, but again, the steam line break is more limiting for the containment overpressurization. Details of the system behaviour are presented below in the thermalhydraulics section.

For some events resulting in a system overcooling, there may be a slight increase in power prior to reactor trip.

5.3.2.3 Fuel Channel and System Thermalhydraulics/Fuel and Fuel Channel Thermal-Mechanical Effects/Fission Product Release and Transport

One initiating event for this type of accident is the rupture of a main steam line outside of the containment boundary. Pre-existing fuel defects contribute to the build-up of an inventory of fission products in the primary circuit to a maximum level, which is at, or below, that prescribed in the operating principles and procedures of the station.

As no failure of the primary coolant pressure boundary is expected for this accident category, activity present in the HTS before the transient and some additional activity released by the previously defected fuel will be confined to the HTS. Pre-existing leaks in the steam generator tubes provide a path for transport of the fission products contained in the primary coolant to the secondary side and hence through the steam line break. The noble gases, and some fraction of the Iodine, will quickly partition into the vapour phase when they are discharged into a steam-filled component or through the steam line break.

The rapid power reduction following the reactor trip, coupled with continued availability of forced convective cooling, quickly reduces fuel temperatures. The fuel-sheath gaps of any fuel that had failed before the accident will be filled with water and the fission product releases from the fuel grains will be terminated. The Iodine and other soluble fission products in the fuel-sheath gaps, pellet surfaces and grain boundaries will continue to be released to the HTS coolant.

Thermalhydraulic behaviour following steam line failures is initially dominated by the sudden discharge of steam via the piping break. Primary system thermalhydraulic behaviour is subsequently dominated by blowdown and cooling behaviour of the secondary circuit. Heat transfer from the hot primary circuit to the secondary circuit determines cooldown of the primary system, which remains intact and retains primary coolant. The time at which trip occurs depends on break size and whether the break is inside or outside containment. For large breaks inside containment a trip is initiated nearly instantly as a result of rising containment pressure. For large breaks outside containment, trip is initiated within a few seconds. Emergency core cooling may be initiated as the primary system pressure drops below the ECC pressure setpoint, however if ECC is initiated injection would be limited for a short period since it would only compensate for a coolant shrinkage (HTS intact). As the primary circuit continues to cool, pressure decreases, ultimately tripping the primary heat transport system pumps so that high velocity forced convection fuel cooling is no longer available. Thermosyphoning flow is maintained providing fuel cooling while the remaining feedwater inventory and RWS makeup allows depressurization and cooling of the HTS to LTC operating conditions.

The discharge rate from the secondary cooling system depends on the size and location of the break. The fastest depressurization would occur for a large symmetric break in the steam balance header. Such a break would accelerate the steam flow out of the boilers, causing the level to initially swell. Steam separation would be less effective when the swelled level reaches the steam nozzle. In this case, a two-phase mixture may discharge from the boilers. Once the swelled level falls below the nozzle outlet, mainly steam leaves the boilers.

Small breaks do not depressurize the primary circuit sufficiently to initiate boiler crash cooldown, emergency core cooling, or automatic primary coolant pump trip. The steam separators would be able to separate liquid from the low steam flows; hence, only steam is discharged from the break. This leaves more water in the boiler for removal of primary circuit heat generation and the stored heat in the primary circuit.

Feedwater piping breaks involve mainly liquid discharge, as opposed to the steam ejected in the event of steam line breaks. The limiting feedwater line break (with respect to heat sink availability) would be a break downstream of the check valve since it is involved in maximum drain rate of the available inventory of the secondary side. For events involving the failure of feedwater supply (for example, due to feedwater pump failure), the boilers start to lose inventory immediately and boiler levels begin to drop and will eventually reach the stepback or the boiler low-level trip setpoints.

A loss of feedwater pumps results in a trip on steam generator low level. In the case of feedwater pump trip, the normal response for a loss of feedwater due to a valve closure is a reactor setback on SG low level. If this fails to function, the SG low level trip is reached. For higher reactor powers, the HTS pressure increases and reaches the HTS high-pressure trip, if an earlier trip or setback had not already occurred.

The loss of secondary circuit pressure control could result in either pressurization or depressurization scenarios. The SG depressurization scenarios include the inadvertent opening of either the ASDV’s, condenser steam discharge valves (CSDVs), or MSSV’s, or due to a failure in the turbine governor valve to unload or close following a reduction in reactor power. For an inadvertent opening of all MSSV’s, the reactor is tripped on HTS low pressure. For a pressurization transient (e.g., loss of condenser vacuum), the SG overpressure protection system limits the pressure increase by opening the MSSV's. Fuel cooling capability is not normally impaired by these control failures.

For fission product transport purposes, the HTS may be considered to be full of water, perhaps containing some degree of void. During transport, some fission products may participate in one of several types of interactions such as deposition on the HTS surfaces and chemical reactions.

5.3.2.4 Moderator and Shield System Thermalhydraulics

There are no relevant moderator and shield system thermalhydraulics phenomena for this event.

5.3.2.5 Containment

Initial containment behaviour for secondary failures which discharge into containment is similar to that of the large and small primary coolant system breaks, as discussed in Chapters 3 and 4. However, in this event, an increased heat load to the containment occurs as energy from both the primary and secondary circuit is discharged to the containment. The following additional phases are introduced:

1. Containment Pressure Beyond Design Pressure - Breaks which discharge into containment may cause the "design pressure" of the containment boundary to be exceeded. Since the discharge of radioactive material from this accident is small, dose to the public can be maintained within allowable limits even if containment leakage goes beyond the designed limits. However, it must be demonstrated that containment integrity is sufficient to avoid consequential damage to the primary coolant circuit. This is demonstrated by analysis that shows specified failure pressures are not reached.
2. Breaks Outside Containment - Although the containment structure plays no role in this accident scenario, it is necessary to show that no safety systems are compromised by discharge into the turbine building and reactor auxiliary building or by discharges near other safety critical facilities.

Behaviour of radioactive materials within or outside containment does not dominate this accident sequence since quantities are small.

5.3.2.6 Radiation Physics

During a secondary coolant accident, most radioactivity remains in the intact core. The radiation from the core is thus much the same as during normal operation.

5.4 Governing Physical Phenomena

The physical phenomena that govern the behaviour of systems of particular importance to secondary side failures are discussed in the following sections. Where phenomena are the same as those encountered for HTS LOCA, reference is made to the appropriate section(s) in Chapters 3, 4 and 10.

5.4.1 Reactor Physics

The rapid HTS cooling and void collapse associated with this accident scenario would result in a small increase in reactivity due to the small negative coolant-density-change induced reactivity *(PH1)*. RRS response to control reactivity transients and the shutdown system action are important for this event, therefore, device movement induced reactivity (*PH11*) is an important phenomena. Coolant density change induced reactivity, fuel temperature change induced reactivity and flux and power distribution (prompt/decay heat) are all considered to only have secondary importance.

5.4.2 Trip Coverage/Fuel Channel and System Thermalhydraulics

The primary trips for these events include low SG level and high reactor building pressure. In addition, for events where the cooling of the HTS is affected, an induced HTS high or low pressure can also occur, depending on whether the HTS is over or under cooled by the event.

Phenomena of importance relate to the evaluation of the secondary coolant discharge to atmosphere and the blowdown of the circuit and heat transfer phenomena such as thermal conduction *(TH6)*, convective heat transfer *(TH7)*, nucleate boiling *(TH8)*, and condensation heat transfer *(TH10)*, which govern the cooling of the primary coolant through the steam generators. Several other system thermalhydraulic phenomena are relevant to secondary circuit failures, including break discharge characteristics *(TH1)*, phase separation *(TH3)*, level swell *(TH4)*, and pump characteristics *(TH5)*. Density driven flows *(TH17)* and thermosyphoning in particular is a primary phenomenon during the period when the HTS is being depressurized and cooled prior to LTC operation. Phenomena of particular importance in the primary circuit are those associated with cooling (i.e., shrinkage and flashing of the coolant and depressurization leading to trip signal generation).

There are no additional phenomena associated with fuel channels other than those relevant to large and small primary coolant system LOCA (Chapters 3 and 4).

5.4.3 Fuel and Fuel Channel Thermal-Mechanical Effects

Since the discharge from the secondary side may bypass containment, and some radioactive material may leak from the primary side to the secondary side, fuel behaviour is important for this event. Phenomena of importance focus on those associated with sheath integrity and trapping of radioactivity within the fuel matrix, including fuel-to-sheath heat transfer *(FC3)*, fission gas release to gap and internal pressurization *(FC5)*, sheath to coolant and coolant to pressure tube heat transfer *(FC13)*. Decay heating related phenomenon like fission and decay heating (*FC1*), and diffusion of heat in fuel (*FC2*) are also considered primary phenomena.

5.4.4 Moderator and Shield System Thermalhydraulics

Other than moderator phenomena related to SDS2, for example, injection of poison (*MH15*), and moderator/coolant poison mixing (*MH19*), no other moderator phenomena are important for this event.

5.4.5 Fission Product Release and Transport to Containment

At the beginning of the accident, the inventory and distribution of fission products in the core are defined by the irradiation history of each fuel element (Section 2.2.6). Since fuel cooling and fuel integrity are not issues for this event, fission product release and transport phenomena are not considered important and are ranked secondary in Table 3.

5.4.6 Containment (Containment Thermalhydraulics, FP Chemistry, Hydrogen, Aerosols)

For main steam line break inside the containment, containment integrity under internal pressurization is the main concern, however, for other types of secondary side failure the integrity of the primary cooling system is the primary safety concern. Since containment may be bypassed by a break outside containment (main steam lines inside containment may also provide such a by-pass) or by the opening of the MSSV's, the containment function is of secondary importance. For breaks inside containment, environmental qualification of safety related equipment inside containment is also of concern, in particular, the environmental qualification of the fuelling machine to withstand the harsh post-accident conditions and maintain the cooling of the irradiated fuel inside.

The basic containment thermalhydraulic response parallels that of the large LOCA analysis in Section 3.4.7. However, the fission product release to containment is much lower and of a different nature due to the intact HTS and the reduced heating of fuel compared to that during primary system failures. The phenomena relating to hydrogen behaviour and Iodine chemistry are also not considered relevant.

5.4.7 Radiation Physics

Radionuclides largely remain in the core during secondary side failures, and thus there are no additional phenomena of importance beyond those discussed in Section 3.4.8.

5.4.8 Atmospheric Dispersion

Atmospheric diffusion is primarily governed by external atmospheric conditions as discussed in Appendix A. Phenomena relevant only to secondary side failures include the possibility of discharge into the turbine building and consequent releases bypassing discharge stacks. Plume rise will be important for discharges from the MSSVs and ASDVs.

5.5 Summary

Secondary circuit failures, including steam line and feedwater breaks, loss of feedwater flow, or loss of secondary circuit pressure control, can involve discharge of the secondary side coolant into containment, depressurization and cooldown of the secondary side, followed by rapid primary coolant cooldown and shrinkage. Heat transfer through the steam generators is the dominant system thermalhydraulics consideration. For fuel and fuel channel thermal-mechanical effects and fission product release and transport analyses, phenomena determining sheath integrity and radionuclide retention in fuel are the primary concern, because there is the possibility of a discharge through a steam line break which bypasses containment. This can also occur for the loss of secondary circuit pressure control events involving the inadvertent opening of the MSSV's. Containment pressure increases for a steam line break can be higher (due to a larger heat load) than for primary circuit LOCA events, and thus it must be ensured that this poses no threat to the HTS and the reactor building.

6. FUEL HANDLING ACCIDENTS

6.1 Description of Accident

This category includes all accidents during fuel handling arising from loss of normal cooling of the fuel in the fuelling machine, or during fuel transfer to spent fuel storage. The fuelling machine is periodically connected to reactor channels, and thus accidents involving the fuelling machine are of two major types, on-reactor and off-reactor events. Fuelling handling design is on going at this time and accordingly off-reactor fuel handling events are not discussed in this report.

On-reactor events include a failure of piping or connection that may occur while the machine is connected to the reactor. In this case it essentially initiates a small loss of coolant event similar in severity to a small feeder failure (Chapter 4).

The fuelling machine may contain as much spent fuel as that contained in one fuel channel. In addition, should the machine be connected to the reactor, the fuel in the affected channel may also be damaged.

6.2 Key Safety Concerns

The relevant safety concerns for fuel handling events, with consequences quantified through safety analysis, are:

- Adequacy of fuel cooling to limit fuel temperatures under degraded cooling conditions;
- Fission product release from the affected fuel; and
- Public dose due to fission product transport.

6.3 Accident Behaviour

6.3.1 Introduction

Accident cases included in this report are on-reactor fuel handling events. In this case, the behaviour encountered following postulated failure of the piping to the fuelling machine is a special case (in terms of discharge flow rate) of the single channel events described in Chapter 4. The phenomena relevant for this case are similar to those covered in Chapter 4

6.3.2 Phases of Fuel Handling Accidents

The two phases relevant fuel handling accidents are;

- *Fuel heat up phase*, this is where the fuel heats up at a particular rate due to the reduction or loss in cooling. The rate at which the fuel heats up is dependent on the specific event.
- *Recovery phase*, this phase is where the affected fuel is immersed in water.

6.4 Governing Physical Phenomena

In general, the phenomena encountered for the on-reactor cases are similar to the phenomena that determine behaviour for a small feeder break scenarios of Chapter 4.

6.5 Summary

In general, the phenomena encountered for the on-reactor cases are similar to system behaviour determined for a small feeder break scenario of Chapter 4. Assessment of the impact of design changes in the fuel handling system as compared to previous CANDU designs is in progress. This section will be revised when the design of the fuel handling system and the assessment of the design modifications will be completed.

7. LOSS OF REGULATION

7.1 Description of Accident

Loss of regulation includes two major types of scenarios: loss of reactivity control and loss of HTS pressure and inventory control failures.

For a loss of reactivity control, reactor power will increase following an uncompensated reactivity insertion. This reactivity insertion is presumed to be caused by a malfunction in the reactor regulation system and the rate of insertion is governed by the reactivity control devices (i.e., mechanical control absorbers and mechanical zone control units) that may be involved. The setback and stepback routines are designed to reduce power for specific failures in the regulation system. However, should both these control functions become unavailable, reactor power would be rapidly reduced by shutdown system action. Loss of reactivity control from low power (with HTS pumps running as well as stopped) is also considered, but the limiting events are the loss of reactivity control from higher power. Depending on the cause of loss of reactivity control, the reactor core power may increase in spatially uniform or non-uniform patterns.

A loss of HTS pressure and inventory control can lead either to system pressurization or depressurization. Some of these events would require either shutdown system or operator intervention to mitigate the consequences of the failures.

Loss of HTS inventory control events involve malfunctions of the feed and bleed valves and include both pressurization as well as depressurization events.

Loss of HTS pressure control failures also include both pressurization and depressurization events.

7.2 Key Safety Concerns

The safety objective for loss of regulation scenarios is to keep doses to the public within the specified guidelines. This is achieved by demonstrating that the event does not lead to release of radioactivity from the primary circuit into the containment, e.g., by preventing consequential fuel failures and by limiting the HTS overpressurization. The trip parameters and setpoints for the two shutdown systems are selected to meet this criterion. To ensure these conditions are met, for gradual increases in power (i.e., critical channel power analysis), it is demonstrable that fuel sheath dryout does not occur. For a more rapid power increase, fuel integrity must be demonstrated. The key safety criteria are:

- Prevention of fuel and sheath failures, fuel centreline melting, and fuel breakup.
- Effectiveness of reactor trips in preventing fuel failures.
- Limit system overpressurization to within required service limits

The integrity of the HTS is maintained by meeting the above criteria.

7.3 Accident Behaviour

7.3.1 Introduction

For a loss of reactivity control, reactor power increases until a trip occurs. Trip effectiveness would ensure fuel integrity. If the reactivity increase is slow, the pressure and inventory control system compensates for any system pressurization due to the change in coolant density. If the reactivity increase is fast, the subsequent heat transport system overpressurization is a concern and the HTS overpressure limits must be met.

For a loss of pressure and inventory control, power generally remains constant, unless the reactor regulating system fails. The heat transport system is either pressurized or depressurized (depending on the nature of the failure) until either a reactor trip occurs or the reactor reaches a new steady state after which the operator can be credited to terminate the event.

7.3.2 Phases of the Accident

Important behaviour for a loss of reactivity control is classified with respect to the speed of the transient, i.e., whether the LOR is fast, resulting in a pressure increase, or slow, resulting in a constant pressure.

For a loss of pressure and inventory control transient, system behaviour is determined by control system actions and responses.

7.3.2.1 Reactor Physics

For a loss of reactivity control, power increases as reactivity is inserted. If the reactivity increase is fast, there is an increase in coolant voiding. An increase in reactivity from high power levels increases coolant voiding. The increase in void will oppose power increase due to the negative coolant-density-change induced reactivity. For a small reactivity insertion rate, the negative coolant-density-change induced reactivity would be enough to control the power increase. For certain malfunctions of the RRS and resulting loss of reactivity control, loss of reactivity control may occur from initially distorted flux shapes or from spatially distorted power shape and high local power may occur.

For a loss of pressure or inventory control, power generally remains constant, unless the reactor regulating system fails.

7.3.2.2 Trip Coverage/Fuel Channel and System Thermalhydraulics/Fuel and Fuel Channel Thermal-Mechanical Effects/Fission Product Release and Transport

Loss of regulation is primarily a trip coverage event. For loss of reactivity control, the relevant trips include high neutron power, high HTS pressure, regional overpower and to a lesser extent high log rate. For the loss of pressure and inventory control failures, the relevant trips include high neutron power, low HTS pressure, high HTS pressure, and the manual trip. For both events, the criteria are to prevent fuel failure, and to limit the HTS overpressure to within the required service limits. The same criteria apply for the loss of pressure and inventory control failures. For some control failures, trips might not be required as the system reaches a new quasi-steady-state and the fuel remains well cooled.

For a loss of reactivity control, the thermalhydraulic behaviour is dominated by the speed of the transient, i.e., by the rate of reactivity insertion. For very slow reactivity insertion rates, the HTS remains in a quasi-steady-state condition as the pressure and inventory control system is able to maintain HTS pressure at the control setpoint. Generally, loss of reactivity control and resulting power increases may cause fuel and fuel sheath temperature and coolant pressure increase until the reactor is tripped or a new steady state is reached. For loss of reactivity control resulting in distorted flux shapes or starting from an initially distorted flux shape, sheath dryout may occur.

For fast reactivity insertion rates, the increase in voiding, caused by the increasing power, results in an increase in the heat transport system pressure. For these rates, the overpressure limits of the HTS must be met, and in some cases, the overpressure limit may become the main criterion for trip effectiveness assessment.

Loss of reactivity control from low power also includes long term cooling modes of operation as well as normal operation from decay powers (i.e., before reactor cooldown is initiated). Again, the phenomena are the same, as power increases until a reactor trip setpoint or a quasi-steady-state is reached. In the case of the long term cooling mode of operation (i.e., HTS pumps stopped and Long Term Cooling pumps running), the process trips (e.g., low pressure, low flow, etc) are conditioned on power.

For a loss of pressure and inventory control, there are two main types of transients. One type results in a heat transport system overpressurization, while the other results in a system depressurization.

A pressurization scenario can be initiated either by the feed valves failing to the open position with the bleed valves closed, or by failing the pressurizer heaters on with the steam bleed valves closed. In the first case, the pressurization is slow. The pressurizer would eventually fill and discharge through the steam bleed valves, balancing the excess feed flow, reaching a quasi-steady-state. No automatic trips are required, nor is operator action required. Eventually, the operator will return the plant to its normal operating state.

If all the pressurizer heaters are failed in the “on” condition, the pressure increase is more rapid, as the liquid in the pressurizer begins to boil and the level then decreases. The liquid relief valves open and the event is terminated on a high-pressure trip.

A depressurization can be caused by several types of failures, including a failure of the pressurizer level control (e.g., bleed valve fail open with feed closed), or by a failure of the pressure control (e.g., steam bleed valves or liquid relief valves failed open with the pressurizer heaters failed off). For these cases, there is a net drain from the heat transport system, resulting in an increase in channel exit quality, and in an increase in fuel and sheath temperatures until a reactor trip occurs. For the case with the liquid relief valves failed open the bleed condenser will bottle up and the bleed condenser relief valves will open. This event becomes a small LOCA inside containment.

During a loss of regulation, no failure of the coolant pressure boundary is expected. Therefore, any activity present in the heat transport system before the transient and any additional activity released by any previously defected fuel will be confined in the heat transport system.

There are no consequential fuel failures or releases to the containment for a loss of regulation.

7.3.2.3 Moderator and Shield System Thermalhydraulics

The state of the moderator plays no role in loss of regulation accident scenarios.

7.3.2.4 Containment

Containment plays little role in loss of regulation accident scenarios.

7.4 Governing Physical Phenomena

7.4.1 Reactor Physics/Fuel and Fuel Channel Thermal-Mechanical Effects

The primary reactor physics phenomena influencing the neutronic overpower transients for loss of regulation events are the device-movement induced reactivity *(PH11)*, and the flux and power distribution (prompt/decay heat) in space and time *(PH14)*.

The coolant-density-change induced reactivity *(PH1)* under initial conditions is determined by the amount of initial boiling. The coolant-void reactivity is driven by the coolant voiding *(TH2)* during the power transient following the loss of regulation event.

7.4.2 Trip Coverage/Fuel Channel and System Thermalhydraulics

A sudden increase in reactivity increases reactor power. Therefore, heat transfer to the coolant increases, causing the primary coolant temperature and pressure to rise. Depending on the initial power, void *(TH2)* may be generated, balancing the reactivity insertion. The increasing fuel and fluid temperature is terminated by a reactor trip. After reactor trip, fuel surface heat flux will decrease. At this stage convective heat transfer (*TH7*), CHF dryout and post dryout heat transfer (*TH9*) and quench rewet characteristics (*TH12*) are phenomenon of primary importance.

Not all cases require a reactor trip, e.g., Loss of Pressure and Inventory Control (LOPIC) or events with small reactivity insertion rates. In some cases, a quasi-steady-state can be reached.

The phenomena of relevance during a loss of regulation are similar to those discussed for other events, although some of the phenomena which are of primary importance for other events, may be of lesser importance for LOR events.

7.4.3 Moderator and Shield System Thermalhydraulics

Moderator system phenomena are not important to loss of regulation accident scenarios.

7.4.4 Fission Product Release and Transport to Containment

At the beginning of the accident, the inventory and distribution of fission products in the core are defined by the irradiation history of each fuel element (Section 2.2.6). During the reactivity increase phase the fuel element temperature may rapidly increase and cracking of the peripheral regions of the pellet may take place. However fuel failures are precluded for the events considered here.

The rapid power reduction following the reactor trip, coupled with continued availability of forced convective cooling, quickly reduces fuel temperatures.

During LOR transients, the HTS is considered to be full of liquid water, perhaps containing some amount of void. As no failure of the pressure boundary is expected, activity present in the HTS before the transient and any additional activity released during the transient will be confined to the HTS. Possible release paths are chronic failures of steam generator tubes and any other minor leaks in the HTS.

7.4.5 Containment (Containment Thermalhydraulics, FP Chemistry, Hydrogen, Aerosols)

There is no impact on containment behaviour from loss of regulation accident scenarios since fuel failures are precluded and no releases are expected into containment.

For the case with the liquid relief valves failed open, the discharge from the bleed condenser relief valves into containment results in consequences similar to a small LOCA.

7.4.6 Radiation Physics

This accident scenario introduces no additional radiation physics phenomena of significance as all radioactive materials remain in the core or HTS.

7.4.7 Atmospheric Dispersion

Any releases from this accident category are within the normal operating conditions, (Section 2.2.9).

7.5 Summary

For a fast loss of reactivity control, the resultant system pressurization dominates the reactor behaviour. For smaller reactivity insertion rates, there is no significant system overpressurization, as the pressure and inventory control system can control the pressure.

For a loss of pressure and inventory control failure, the resultant transient can either be pressurization (like the loss of reactivity control) or a depressurization (similar to a small LOCA). Some transients require shutdown system action, while others reach a quasi-steady-state which can continue until the operator terminates the event.

8. LOSS OF FLOW

8.1 Description of Accident

Loss of flow scenarios can be grouped into two categories, based on the initiating cause:

1. Total or partial loss of Class IV electrical power resulting in loss of all HTS pumps, and loss of electrical power to a single HTS pump. This scenario includes a total loss of Class IV power, loss of one bus, and a single pump trip.
2. Mechanical failure of a single HTS pump (including pump seizure).

For the first type, the HTS coolant flow is reduced gradually as the affected pumps rundown, and for the second type, the HTS coolant flow is reduced abruptly for the failed pump.

In both cases, the reduced flow causes voiding in the core, which produces small negative reactivity feedback. The consequent decrease in reactor power is likely controlled by the regulating system. The event is terminated by the action of the shutdown systems on HTS high pressure or low flow.

For a complete loss of Class IV power, all HTS pumps run down and the turbine trips due to a loss of condenser vacuum. Before the condenser vacuum is lost, the turbine may trip on various signals including an electrical protection signal. The primary coolant flow decreases due to the rundown of the pumps. The flow-power mismatch raises the primary coolant temperature and pressure, which initiates the opening of the HTS liquid relief valves. The steam generator feedwater pumps run down, as a result of a total loss of power, to cause a temporary loss of makeup to the steam generators prior to the automatic operation of the auxiliary feedwater pumps. The reduced flow in the HTS causes void in the core which produces a negative reactivity feedback. The decrease in power is likely controlled by the regulating system. Shutdown systems (i.e., trip) are activated on HTS high pressure or low flow. The condenser circulating water pumps, and hence the steam condenser, becomes unavailable. This prevents the CSDV’s from opening. The MSSV’s can open, if required, to control the steam generator pressure. Coolant feed to the HTS becomes temporarily unavailable as the respective pumps are operated on Class IV power.

For a partial loss of Class IV power, the behaviour is similar, except that only the pumps powered by the failed bus are tripped.

For a single pump trip, the pump rundown creates asymmetric flow and pressure distributions in the HTS. Coolant flow in the core decreases, and, as a consequence, the system temperature and pressure increase. The reduced flow in the core can cause voiding in the core which results in induced negative reactivity and accordingly, power reduction. The reactor power is likely controlled until reactor trip on high HTS pressure or low flow.

Mechanical failures within a HT pump may result in a substantial flow reduction in the associated HTS pass. The second pump in parallel continues to provide flow circulation for the core. The two major failure modes include a failure of the pump shaft and a seizure of the pump. The immediate consequence of either a pump shaft failure or a pump seizure is the rapid loss of head in the affected pump. The distinguishing difference between the failure modes is whether or not the pump impeller is free to rotate. This factor determines the effective resistance to flow of the failed pump.

8.2 Key Safety Concerns

There are two major safety concerns for the loss of flow scenarios. The first is to limit the system overpressurization to within the required service limits. Consequently, the integrity of the HTS is maintained. This is achieved by providing timely and effective trips on each shutdown system.

The second concern is to maintain adequate cooling to the fuel, as the pumps run down, to prevent fuel failures. This is achieved at higher initial powers by providing timely and effective trips. At sufficiently low initial powers the fuel is cooled by thermosyphoning flows.

8.3 Accident Behaviour

8.3.1 Introduction

Significant degradation in fuel cooling can occur during a loss of flow scenario. At higher initial powers, degradation in fuel cooling is terminated by the shutdown system action due to the reduced power. For total loss of Class IV power and from low power levels, the fuel is cooled by thermosyphoning flows.

8.3.2 Phases of the Accident

In the pre-trip phase of the accident, the flow is reduced due to the pump(s) failure and pressure increases. At higher powers, the system overpressure is limited by regulating or shutdown system action.

After trip, or at low initial powers, the fuel is cooled by thermosyphoning, with the steam generators acting as the heat sink.

8.3.2.1 Reactor Physics

For higher initial powers, and when RRS is unavailable, the reduced flow causes voiding in the core, which produces negative reactivity feedback and a decrease in reactor power. The reactor is shut down by process trips.

8.3.2.2 Trip Coverage/Fuel Channel and System Thermalhydraulics/Fuel and Fuel Channel Thermal-Mechanical Effects

Loss of flow is primarily a trip coverage event. The relevant trips include high HTS pressure, and low core flow trips.

For a loss of flow accident, the thermalhydraulic behaviour is dominated by the rate of reduction in the flow and by the initial reactor power. All transients result in a flow rundown and a HTS overpressurization. For a total loss of Class IV power or for a pump seizure event, the overpressurization is more pronounced. For higher initial powers, void generation results in a subsequent decrease in power. Reactor power is terminated by a reactor trip. For a pump seizure event, the flow is reduced quickly due to increased resistance of the seized impeller. For a single pump trip, the flow rundown is more gradual, as the failed pump freewheels, and the other parallel pump maintains flow.

For lower initial powers, the pressure increase is within the required limits, and a trip is not required. Adequate fuel cooling is maintained by thermosyphoning.

During a loss of flow event, the integrity of the coolant pressure boundary is maintained. Therefore, any activity present in the HTS before the transient and any additional activity released by any previously defected fuel will be confined in the HTS and bleed condenser.

8.3.2.3 Moderator and Shield System Thermalhydraulics

There are no relevant moderator and shield system thermalhydraulics phenomena for this event.

8.3.2.4 Fission Product Release and Transport to Containment

In the event of a loss of class IV power, the primary circuit remains intact and consequential fuel failures are precluded, thus preventing the release of radioactive material to containment. The only release path would be through the opening of the MSSV's after a loss of Class IV power, if the plant had been operating with a leaking steam generator tube.

8.3.2.5 Containment (Containment Thermalhydraulics, FP Chemistry, Hydrogen, Aerosols)

There are no relevant containment phenomena for this event.

8.3.2.6 Radiation Physics

There are no relevant radiation physics phenomena for this event.

8.4 Governing Physical Phenomena

8.4.1 Reactor Physics

The flux and power distribution in space and time (*PH14*) and the device movement induced reactivity (*PH11*), which affect the reactor trips and response to the reactor regulating system are of primary importance. Another phenomena, of secondary importance, influencing the neutronic power transient for a loss of flow event is the coolant-density-change induced reactivity *(PH1)*, which is driven by the coolant voiding *(TH2)* during the transient.

8.4.2 Trip Coverage, Fuel Channel and System Thermalhydraulics

The coolant flow rundown is affected by the nature of the event and by the HTS pump characteristics *(TH5)*, including the pump inertia which dominates the loss of flow. If all pumps are failed (i.e., a total loss of Class IV power), or if one pump fails due to a seized impeller, the flow rundown is more rapid. This flow rundown can result in coolant voiding *(TH2)* and phase separation *(TH3)* which results in a HTS pressure increase.

CHF dryout and post dryout heat transfer (*TH9*) is of particular importance for trip effectiveness. Other important heat transfer related phenomena which occur due to loss of flow are convective heat transfer (*TH7*), nucleate boiling (*TH8*), condensation heat transfer (*TH10*), quench/rewet characteristics (*TH12*). For loss of flow combined with other failures, reflux condensation (*TH14*) and counter current flow (*TH15*) are important.

At a sufficiently higher power, reactor trip occurs which is affected by the flow oscillations (*TH16*), and at decay power levels, fuel cooling is provided by thermosyphoning *(TH17)*.

8.4.3 Fuel and Fuel Channel Thermal-Mechanical Effects

The important phenomena are those associated with maintaining sheath integrity and as a result fuel integrity, e.g., containing radioactivity within the fuel matrix or the sheath. These phenomena include fission and decay heating (*FC1*), diffusion of heat in fuel (*FC2*), fuel-to-sheath heat transfer *(FC3)*, fission gas releases to gap and internal pressurisation (*FC5*), sheath to coolant heat transfer (*FC13*) including, CHF (dryout) and post-dryout *(TH9)* and channel and subchannel flow effects (*FC14*).

8.4.4 Moderator and Shield System Thermalhydraulics

Other than moderator phenomena related to shut-down systems (see Section 5.4.4), the moderator system has no role in accidents involving loss of flow.

8.4.5 Fission Product Release and Transport to Containment

At the beginning of the accident, the inventory and distribution of fission products in the core are defined by the irradiation history of each fuel element (Section 2.2.6). Fuel failures are precluded for these events.

8.4.6 Containment (Containment Thermalhydraulics, FP Chemistry, Hydrogen, Aerosols)

There are no containment phenomena of importance to loss of flow accident scenarios since fuel failures are precluded and no releases are expected into containment.

8.4.7 Radiation Physics

This accident scenario introduces no additional radiation physics phenomena of significance, as any radioactive materials remain in the core and HTS.

8.4.8 Atmospheric Dispersion

Any releases from this accident category would be within the normal operating range (Section 2.2.9) excluding potential releases from MSSV's with boiler tube leaks.

8.5 Summary

For a loss of flow event, the reactor behaviour is governed by the pump rundown or failure. Any excessive pressure increase is terminated by shutdown system action. CHF/dryout and post dryout are important phenomena in determining trip effectiveness. Asymmetric events such as single pump trip or pump seizure result in degradation in fuel cooling due to flow reduction. At lower powers, trips are not required and the fuel is cooled by thermosyphoning.

9. AUXILIARY SYSTEM FAILURES

A number of auxiliary systems are important to the function of main process systems and safety systems and/or supplement their function in certain scenarios. Failures of these systems introduce a few additional phenomena not addressed in the previous accident categories. In the context of this document, auxiliary systems of importance include the long term cooling system, moderator system, shield cooling system, service water systems, and off-site power systems. Failures in some of these systems may lead to failures of process systems and safety systems. Failures of the off-site power system and service water systems will not be discussed in this document since they will not lead to any new phenomena not mentioned in previous sections.

9.1 Description of Accidents

A brief description of auxiliary system accidents in each category follows.

9.1.1 Long Term Cooling System

The long term cooling system is designed to provide sufficient cooling capacity for the removal of fuel decay heat following reactor shutdown, and to keep the HTS coolant at design temperature for an indefinite period of time. The HTS system may be closed and pressurized or it may be depressurized, opened and drained to the header level when maintenance is undertaken. Potential accidents include failure of relatively small piping in the system, and loss of cooling flow. Analysis is undertaken to demonstrate that alternate heat sinks may be re-established while doses to the public are maintained within limits.

9.1.2 Moderator System

The moderator system removes heat generated in the reactor core, and transferred to the moderator fluid. Deuterium gas is formed in the moderator by the radiolysis of heavy water. The moderator system includes a sub-system (moderator cover gas system), which controls deuterium gas concentration and provides an inert cover gas to prevent ignition. It also controls deuterium gas levels to avoid any possibility of a deflagration within the cover gas system. Analysis of potential accidents considers the possibility that piping failures may occur in the moderator system or that cooling capacity of the moderator is either significantly reduced due to loss of heat sink, or significantly enhanced due to loss of temperature control.

Moderator piping failures can result in decreasing moderator level and a corresponding change in the spatial reactor power distribution. Should the moderator system fail due to loss of cooling to the moderator heat exchangers or loss of circulation pumps, the moderator temperature increases, and bursting of the protective rupture discs in the calandria relief ducts could occur, discharging tritium-containing moderator to the containment.

9.1.3 Shield Cooling System

The shield cooling system removes heat generated in the shield tank and the end shields due to nuclear radiation from the reactor core and heat transferred from the fuel channels, HTS feeders, and the moderator. This system also maintains the shield tank and end shields full of water to provide biological shielding against radiation during normal operation and shutdown conditions. Failures of the shield cooling system that are assessed include loss of coolant inventory, loss of service water to the shield cooling system and loss of shield cooling flow. The goal of the analysis is to show that the operator will have sufficient time to act and prevent unacceptable deformation of the calandria assembly.

9.2 Key Safety Concerns

The ultimate safety concern of relevance to the accidents described in the preceding section is public dose related to fission product releases from the fuel and/or moderator. The concerns related to the failure of the long term cooling system are similar to those of small LOCA accidents. For loss of moderator and end shield inventory or loss of moderator and end shield cooling events, the key safety concerns are (in addition to the public dose):

- The possibility of developing thermal deformation and stress in the reactor structures sufficient to jeopardize the integrity of the HTS and/or the operability of the shutdown systems.
- For accidents where the moderator drains, the reactivity mechanism devices become uncovered and heat up as they lose cooling provided by the moderator. If the concentration of deuterium and oxygen in the moderator cover gas reaches the flammable limit and the uncovered reactivity-mechanism-device temperature becomes sufficiently high, the cover gas could ignite and deflagration in the cover gas may occur. The resulting pressure and temperature increases in the cover gas load the calandria and the containment boundaries.

9.3 Accident Behaviour

9.3.1 Introduction

The response of the reactor systems following failure of the long term cooling system, the moderator system, or the shield cooling system, is discussed in Section 9.1.

9.3.1.1 Long Term Cooling System

Failures of the long term cooling system do not result in new behaviour and their consequences can be similar to those discussed for large and small break LOCA in Chapters 3, and 4, except that the reactor is at decay power levels and therefore phenomena such as reactivity transients do not occur.

9.3.1.2 Moderator System

The general sequence of events following a loss of moderator cooling is described below:

- Loss of moderator cooling results in gradual heat-up of the moderator, moderator boiling, moderator pressure increase, and moderator level swell.
- The change of moderator temperature and level triggers a series of alarms. The change in moderator temperature changes core reactivity. Since this is a relatively slow transient, the RRS can compensate, if credited, for the reactivity change in order to maintain the reactor power constant.
- Solubility of deuterium decreases with an increase in temperature so that the gas is driven from solution into the cover gas.
- Should the event continue, the high level moderator level trip would be reached. If this trip is not credited, the moderator overpressure rupture discs might burst to relieve pressure by discharging moderator into containment.

- After the rupture discs burst, the moderator fluid will flash due to the sudden drop in pressure and the level may drop to the moderator low level trip, if not already tripped.
- The discharge of flashing moderator may cause the pressure in the containment to rise and trip the reactor on high reactor building pressure signal if moderator trip not credited.

The general sequence of events following pipe ruptures at various locations in the moderator system is described below:

- Following a pipe break, the moderator level decreases as a result of loss of moderator inventory.
- Moderator low level alarms and indications are annunciated.
- Due to the reactivity effects of low moderator level, the reactor power would be reduced. However, for some critical loss rates, the RRS can compensate for the reactivity change and maintain reactor power.
- Should the draining continue, eventually rows of fuel channels will be uncovered resulting in re-distribution of power in the remaining fuel channels. The reactor trips on high local power by the ROP system or on low moderator level. This is applicable to both shutdown systems.
- If the moderator loss rate is fast, the reactor shuts itself down, as the RRS can no longer compensate for the reactivity decrease due to falling of the moderator level.

9.3.1.3 Shield Cooling System

The consequence of loss of shield cooling due to a loss of flow, loss of recirculated cooling water or loss of shield coolant events is the heating of the end-shield components. This heating causes thermal distortion of the end shield components and has a potential for affecting core geometry and threatening core integrity. In particular,

- Loss of cooling to the end shields results in a gradual heat-up of the end shield and there is a series of alarms associated with the event. To prevent boiling in the end shields, the operator is required to shut down the reactor in a timely manner and cool down the HTS. Prior to reactor shut down, the overpressurization of the end shields is prevented by demonstrating that the level swell rate is less than the relief capacity of the relief line.
- Loss of end shield water due to a pipe break causes the water level to fall and parts of tube sheets to uncover and heat up. This heating generates a temperature difference between the end shield and moderator side of the tube sheets causing thermal distortion and potentially jeopardizing the integrity of the HTS and operability of the shutoff rods.

9.3.2 Phases of the Accidents

9.3.2.1 Reactor Physics/Trip Coverage

Interruption and breaks of the long term cooling system and interruption and breaks of the shield cooling system do not result in reactor physics behaviour significantly different than that during shutdown and normal reactor operation.

In the case of loss of recirculated cooling water to the moderator heat exchangers, the moderator eventually boils, decreasing the total core reactivity.

A loss of moderator inventory has the following phases:

1. Initial Conditions - the steady state conditions of the reactor just prior to the loss of moderator inventory.
2. Drain Phase - the period during which the moderator drains.

A loss of moderator heat sink has the following phases:

1. Initial Conditions - the steady state conditions of the reactor just prior to the loss of moderator heat sink.
2. Pre-Rupture-Disk-Burst Phase - the period during which the loss of moderator heat sink occurs and the moderator fluid heats up. The reactor remains at full power during this phase due to RRS reactivity compensation, and eventually trips on a high moderator level.
3. Post-Rupture-Disk-Burst Phase - the period following the bursting of the rupture disks during which moderator inventory is lost, if reactor trip on moderator high level is not credited.

9.3.2.2 Fuel Channel and System Thermalhydraulics/Fuel and Fuel Channel Thermal-Mechanical Effects/Fission Product Release and Transport

Failures of auxiliary systems other than the LTC system have no significant impact on the fuel channel and system thermalhydraulics behaviour, fuel and fuel channel thermal-mechanical behaviour, and fission product release and transport. The effects of failure of the long term cooling system are similar to those covered in the section describing failures in the HTS (Section 4).

9.3.2.3 Moderator and Shield System Thermalhydraulics

The moderator thermalhydraulics phases are covered in Section 9.3.2.1. The phases for loss of shield cooling or a loss of shield coolant are,

A loss of shield system inventory has the following phases:

1. Initial Conditions - the steady state conditions of the reactor just prior to the loss of shield system inventory.
2. Drain Phase - the period during which the end shields drain and heat up.

A loss of shield system coolant has the following phases:

1. Initial Conditions - the steady state conditions of the reactor just prior to the loss of shield system cooling.
2. Heat up phase - the period during which the end shields heat up due to the loss of cooling.

9.3.2.4 Containment

Failure of the long term cooling system piping results in containment behaviour similar to that discussed in connection with LOCA events (Sections 3.3.2.5, and 4.3.2.5). Failure of moderator cooling leads to heat up of the moderator and eventually discharge of moderator heavy water into containment (via the relief ducts), if the reactor trip on moderator high level is not credited. The containment behaviour associated with this event is similar to the behaviour covered in Chapters 3, and 4. Loss of shield cooling can lead to a modest discharge of light water coolant into containment. Steam and energy discharge rates are relatively low.

9.4 Governing Physical Phenomena

The physical phenomena governing accident behaviour involving failures of the long term cooling system, the moderator system and the shield cooling system are identified below.

9.4.1 Reactor Physics/Trip Coverage

In this category of events, loss of moderator cooling and pipe break are the accidents which involve reactor physics phenomena. The moderator density-change induced reactivity *(PH3)* and core physics response to moderator level *(PH19)* are dominant during the moderator-level collapse in the post-rupture-disk-burst phase as discussed in Section 9.3.2.1. During this phase, the moderator may remain as a homogeneous mixture. This corresponds to a significant decrease in reactivity.

The device-movement induced reactivity *(PH11)* is a primary phenomenon during both the loss of moderator-inventory and loss of moderator cooling events. In the case of a loss of moderator inventory, the reactor regulating system responds to the distorted flux distribution created by the loss of moderator during the drain phase. The reactor is either tripped on high or low moderator level, depending on the failure.

The prompt/delayed neutron kinetics *(PH12)* are relevant because of the change in reactor configuration during loss of moderator.

The flux-detector response *(PH13)* is a primary phenomenon during the post-rupture-disk-burst phase of the loss of moderator-heat-sink case. The shutdown system detectors respond to the large bottom-to-top flux tilt and determine the timing of shutdown-system actuation on an ROP signal, if not already tripped on moderator level.

The flux and power distribution (prompt/decay heat) in space and time *(PH14)* is an important phenomenon for the initial and final phases of the accidents under consideration. The initial flux and power distributions characterize the reactor configuration and can have an influence on the subsequent transient if a reactor trip on moderator level is not credited.

9.4.2 Fuel Channel and System Thermalhydraulics

Failure of piping in the long term cooling system may invoke the phenomena identified in Sections 3.4.3, 4.4.3. The moderator system and shield cooling system failures introduce no new phenomena relevant to fuel channel or system thermalhydraulics.

9.4.3 Fuel and Fuel Channel Thermal-Mechanical Effects

As in Section 9.4.2, the only accident of relevance to fuel channels is the failure of the long term cooling system piping, which can invoke the phenomena discussed in Sections 3.4.4, 4.4.4.

9.4.4 Moderator and Shield System Thermalhydraulics/Trip Coverage

9.4.4.1 Moderator Thermalhydraulics

For the loss of moderator heat sink scenario due to either loss of recirculated cooling water flow to the moderator heat exchangers or loss of moderator circulation (loss of pumps), the moderator fluid will heat up. For this scenario there is a possibility that the calandria relief rupture discs may burst, causing heavy water to be discharged to containment if a moderator high level trip is not credited.

The relevant phenomena related to the discharge are moderator swell *(MH42)*, and liquid, vapour and two-phase discharge *(MH41)* from the relief ducts since these phenomena will determine the timing of the high level trip and the rate of heavy water discharge into containment. Moderator heat-up will result in degassing thereby increasing the deuterium concentration in the cover gas space. This could result in a flammable mixture of the cover gas. Prior to bursting of the rupture discs, the important phenomena relating to the distribution of deuterium in the cover gas space are: moderator degassing and transfer processes in the moderator cover gas *(MH46)*, and deuterium deflagration *(MH34)*. Since SDS2 is called upon for reactor shutdown, injection of poison *(MH15)*, and moderator/coolant poison mixing *(MH19)* are important phenomena.

The phenomena governing circulation behaviour in the moderator are also important from a mixing perspective, i.e., interaction of moderator flow with the calandria tubes *(MH10)*, moderator flow turbulence *(MH11)*, moderator buoyancy *(MH12)*, moderator inlet jet development *(MH13)*, and moderator pump cavitation *(MH9)*. Phenomena related to energy deposition/distribution in the moderator, such as thermal conduction *(MH43)*, convective heat transfer *(MH44)*, radiative heat transfer *(MH45)*, and moderator cover gas pressure *(MH48)* are also important.

The phenomena that occur during the loss of moderator inventory event are similar to the phenomena relevant for the loss of moderator heat sink event. In addition, decrease in cover gas pressure occurs, thereby increasing the deuterium concentration in the cover gas space. The important phenomena relating to the distribution of deuterium in the cover gas space are moderator degassing and mass transfer processes in moderator cover gas *(MH46)* and deuterium deflagration *(MH34)*.

Loss of moderator temperature control due to failures of the moderator heat exchangers recirculated cooling water control valves can lead to overcooling of the moderator fluid, particularly during the winter season. However, this event does not introduce new phenomena beyond those already discussed.

9.4.4.2 Shield System Thermalhydraulics

Loss of shield cooling (due to loss of recirculated cooling water to heat exchangers or loss of circulation) and loss of shield cooling inventory events result in end-shield heat up. The important phenomena determining the rate of heat up and temperature distribution within the end-shield are: thermal conduction *(MH43)*, convective heat transfer *(MH44)*, radiative heat transfer *(MH45)*, and interaction of the end-shield flow with the end-shield solid matrix *(MH47)* (lattice tubes and steel balls).

Loss of shield temperature control due to failures of the shield heat exchangers service water control valves can lead to overcooling of the end shield coolant, particularly during the winter season. However, similarly to the loss of moderator temperature control low event, this event does not introduce new phenomena beyond those that have already been discussed.

9.4.5 Fission Product Release and Transport to Containment

Loss of shield cooling events results in no significant fission product releases or transport. Long term cooling pipe breaks involve the phenomena discussed in Section 7.4.4 for Loss of HTS Coolant Pressure or Inventory Control (LOPIC) events.

Loss of moderator cooling leads to some moderator boiling and, once the rupture discs burst due to increased pressure, tritium is released from the moderator into containment. The phenomena of relevance to such accidents are discussed in Section 9.4.7 on radiation physics.

9.4.6 Containment (Containment Thermalhydraulics, FP Chemistry, Hydrogen, Aerosols)

Failure of moderator cooling, or moderator flow leads to heat-up of the moderator and, eventually a possible discharge of heavy water through the moderator relief ducts into containment. Since the moderator contains tritium, the discharge is associated with release of an activation product into containment and potential release to the outside atmosphere. The primary phenomena relating to these events are similar to those for a small LOCA event, as described in Section 4.4.7, although some of the phenomena which may be ranked as being of primary importance for the small LOCA scenario, are of secondary importance for auxiliary system failures.

Shutdown and shield cooling interruptions do not significantly affect containment.

9.4.7 Radiation Physics

The main radiation physics safety concerns associated with these events include minimizing public dose due to tritium releases from the moderator, and maintaining the integrity of the calandria, heat transport system and shutdown systems. Excessive thermal stresses and/or pressure transients due to deuterium deflagration in the calandria could jeopardize reactor shutdown and core integrity.

The production of tritium *(RAD2)* in the moderator following neutron emission *(RAD1)* and transport *(RAD3)* result in the release of tritium in the moderator discharges. For scenarios which involve moderator heat up and discharges through relief valves or calandria rupture discs, tritium is released into containment.

The energy deposition in the calandria has two components. One is proportional to the neutronic power of the reactor core *(RAD6)*; the other is the decay component *(RAD4),* which is dominated by the inventory of fission product build-up during full power operation.

The production of deuterium in the moderator through radiolysis *(RAD8)* allows for the possibility of deflagration if a flammable mixture is generated in the cover gas, and an ignition source is present.

9.4.8 Atmospheric Dispersion

Atmospheric diffusion is primarily governed by external atmospheric conditions as discussed in Appendix A. A special consideration for this accident is possibly the high level of tritium in the discharge. However, tritiated heavy water is assumed to behave as any other airborne radioactive contaminant and the concentrations may be reduced by the continued operation of the D_2O vapour recovery systems that collect heavy water vapour.

9.5 Summary

Auxiliary system failures are mainly characterized by behaviour associated with long term cooling system, moderator system and shield cooling system. A trip on high or low moderator level occurs depending on the transient. A neutronic trip is expected to occur on flux tilt as the moderator drains because of the bottom-to-top flux tilt.

10. LIMITED CORE DAMAGE ACCIDENTS

10.1 Description of Accidents

This chapter discusses limited core damage events that are identified as,

- design basis accidents combined with the failure of the emergency core cooling system, and
- limiting single channel events that include severe channel flow blockage and stagnation feeder break accidents.

For small and large break LOCA, the HTS behaviour is the same until the time of ECC initiation, whether ECC occurs or not. The large and small break LOCA events, described in Chapters 3 and 4, may proceed along different paths should emergency coolant fail to be injected as designed. In this accident scenario, the primary coolant continues to be discharged uncompensated. Ultimately, all liquid coolant will be lost in the core, severely reducing fuel cooling. The fuel decay heat continues to heat up the fuel channel components. As the fuel continues to heat up, water present as steam in the core may react with the metallic core components to generate additional energy and hydrogen. The hydrogen is discharged into containment where the hydrogen could potentially ignite, releasing considerable energy. Pressure tubes eventually sag into contact with their associated calandria tubes. When the metal-water reaction has consumed most of the zirconium in the fuel bundles, a quasi-steady-state is achieved in which the PT/CT contact allows the fuel decay heat to be transferred through the PT/CT to the moderator and end shields.

A channel flow blockage increases flow resistance which reduces the coolant mass flow in a single channel. For small blockages, the fuel bundles in the affected channel do not dry out and hence channel conditions are similar to normal operating conditions. Flow blockages that are severe (>95% flow reduction) result in steam being formed in the channel. This causes the fuel bundles and pressure tube to heat up rapidly, likely leading to pressure tube failure. If a pressure tube were to rupture and the calandria tube remain intact, then the bellows may fail due to large stresses caused by the coolant pressure. The general behaviour following a bellows failure is similar to a small out of core break. However, the progression of this type of event is such that the likelihood of consequential calandria tube failure is very high. Severe flow blockages leading to pressure tube and calandria tube rupture involve high fuel temperatures in the affected channel and are considered in this section.

A stagnation feeder break event is similar to a severe flow blockage event. For a small range of break sizes in an inlet feeder there continues to be flow in the downstream channel in the forward direction sufficient to provide adequate fuel cooling. On the other hand, a large break, such as a complete severance of an inlet feeder, causes the channel flow to quickly reverse in the channel also maintaining adequate fuel cooling. These two events are covered under the small LOCA event. Between these two break sizes, there exists a range of break sizes in which progressively lower flow in the affected channel is attained. Over a very narrow range of break sizes, the pressure drop across the channel degrades to near zero and accordingly the channel flow reaches near stagnant condition. If a critical inlet feeder break size persists, the stagnant flow condition can result in channel dryout and this can lead to rapid fuel heat up, fuel damage and failure of the fuel channel similar to that associated with a severe channel flow blockage. Such an inlet feeder break scenario is called a stagnation feeder break. An inlet feeder break where channel flow is low enough to result in fuel failure but high enough that the pressure tube remains intact is identified as the off-stagnation break (see Chapter 4). A stagnation feeder break leading to pressure tube and calandria tube rupture is considered in this section. The ECC system is assumed to be available for the severe flow blockage and stagnation feeder break accidents considered in this section.

10.2 Key Safety Concerns

The relevant safety concerns are:

- Public dose due to the release of fission and activation products;
- Core coolable geometry and fuel channel integrity;
- Effect of jet discharges and pipe whip on safety systems; and
- Containment integrity related to pressurization and possible hydrogen combustion.

10.3 Accident Behaviour

10.3.1 Introduction

For the LOCA events with loss of ECC, the system behaviour during the first phase of the transient, until the time for ECC injection initiation, is identical to the behaviour described in Chapters 3 and 4. At the time of ECC initiation, the reactor has been shut down and depletion of the primary coolant inventory is well underway.

Overall HTS system behaviour is governed by the blowdown and phenomena involving heat generation form the fuel, including oxidation of metals as they reach high temperatures, with the consequent releases of fission products and hydrogen into containment. This behaviour evolves over limited time periods and proceeds in a specific order determined by heating of fuel sheaths and other components within the core.

Severe flow blockage and stagnation feeder break events are characterized by very high fuel temperatures and rupture of the affected channel prior to reactor trip. The overall HTS behaviour is similar to that of a small LOCA with ECC available (Chapter 4).

10.3.2 Phases of the Accident

The early phases of LOCA/LOECC up to the point of ECC injection initiation are identical to those described for large LOCA in Chapters 3 and 4. Subsequent to the failure of emergency core cooling, the phases are defined according to the primary phenomena or time periods during the accident progression in which characteristic system behaviours are exhibited. For severe flow blockage and stagnation feeder break, accident phases are governed by the timing of the rupture of the affected channel and the consequential break discharge into the moderator. For each of the major technical/analysis disciplines involved, in the following sections, these phases are defined and the primary phenomena during these phases are identified.

10.3.2.1 Reactor Physics

Accident phases are similar to those discussed in Sections 3.3.2.1 and 4.3.2.1.

10.3.2.2 Trip Coverage

The trip coverage for LOCA/LOECC is the same as that for LOCA with ECC available.

In general, trip coverage for limiting single channel events is similar to that discussed in Section 4.3.2.2.

10.3.2.3 Fuel Channel and System Thermalhydraulics/Fuel and Fuel Channel Thermal-Mechanical Effects/Fission Product Release and Transport

The phases of LOCA/LOECC are,

1. *Extended Blowdown Phase* - The period of ongoing heat transport system blowdown without injection of emergency coolant. In this phase, overall behaviour is characterised by depressurization, blowdown cooling, fuel sheath failure, fuel deformation, pressure tube deformation (sagging), heat transfer to the moderator, fuel sheath and other metallic component oxidation, oxidation energy release, and hydrogen generation. Fission product release is determined by fuel temperature transients and oxidation. Fission products are transported to the containment via the break. The extent of metal oxidation (zirconium-steam reaction) will depend on the nature of the ECC impairment and the steam flow in the channels. Conditions may occur that will result in substantial oxidation with consequential impact on fuel behaviour, fission-product release and hydrogen loading of containment.
2. *Steam Cooling/Heat Rejection to Moderator* – In this period, core component deformation has ceased and fuel heat transfer to the moderator is established by conduction through the fuel channels. Overall behaviour is dominated by moderator heat transfer phenomena and moderator circulation.

For limiting single channel events the phases are identified in the relative subsections of Section 10.4.

10.3.2.4 Moderator and Shield System Thermalhydraulics

Phases important for LOCA/LOECC accident are:

1. *Extended Blowdown Phase* - The reactor is shutdown and gamma heating of the moderator continues due to fission product decay and radiation heat transfer from the fuel channels. As the blowdown phase progresses, decay heat is transferred to the moderator by conduction through contacting pressure tube and calandria tube walls.
2. *Quasi-Steady-State/Heat Rejection to Moderator* – In this period, core component deformation has ceased and heat transfer to the moderator has stabilized, fuel/channel temperatures have peaked and decay heat from the fuel is removed by the moderator system.

For single channel events that result in channel rupture, the event phases for moderator thermalhydraulics are similar to those discussed in Section 4.3.2.4.

10.3.2.5 Containment

The containment initial behaviour for LOCA/LOECC is identical to that of large and small LOCA events when ECC is available (Chapters 3 and 4). Increased heat load to containment, hydrogen release into containment and the higher possibility of hydrogen combustion are additional phenomena.

1. *Extended Blowdown, Pressurization and Mixing Phase* - Containment will achieve a high pressure when ECC is unavailable due to additional energy resulting from fuel oxidation and the unavailability of the ECC heat sink. In addition, substantial quantities of hydrogen may

be released into containment and mixed with the air-steam atmosphere. Forced convection in the containment atmosphere is the dominant mixing mechanism, driven by the local air coolers. Natural convection circulation paths within containment also have a significant impact on mixing behaviour.

2. *De-pressurization and Condensation Phase* - Containment pressure will decrease once heat transfer from the fuel channels to the moderator is established and energy discharge to containment decreases due to HTS depressurization and voiding. Long-term heat removal via local air cooler units is the dominant phenomenon. Steam condensation in containment will tend to increase the hydrogen concentration, while at the same time hydrogen recombiners will remove hydrogen from the containment atmosphere.
3. *Combustion Pressurization* - If the local hydrogen concentration exceeds the flammability threshold and an ignition source is present, hydrogen ignition can occur, leading to an increase of the global pressure and local temperature. The extent of overpressure depends on the amount and concentration of hydrogen.

The accident phases of limiting single channel events included in this section are similar to that of design basis single channel events discussed in Section 4.3.2.5.

10.4 Governing Physical Phenomena

The primary physical phenomena governing systems behaviour for a LOCA with the failure of emergency core cooling and limiting single channel accidents are discussed below. The discussion is developed with reference to the phases of the LOCA and single channel accidents, as described in Chapters 3, 4 and 10.3.

10.4.1 Reactor Physics

The flux and power distribution (prompt/decay heat) in space and time *(PH14)* is the primary phenomenon during and after blowdown in a large LOCA transient with loss of emergency core cooling (as discussed in Section 3.4.1). Loss of emergency core cooling has a negligible effect on the fuel decay heat transients. The rankings of the phenomena for LOCA/LOECC are similar to those in large LOCA with ECC available. The important physics phenomena in the single channel events of this section are identical to those discussed in Section 4.4.1.

10.4.2 Trip Coverage

Trip Coverage for LOCA with the failure of emergency core cooling is the same as the trip coverage discussed in Sections 3.4.2 and 4.4.2.

In general, trip coverage for limiting single channel events is similar to that discussed in Section 4.4.2.

10.4.3 Fuel Channel and System Thermalhydraulics

Sections 3.4.3 and 4.4.3 discuss the primary phenomena which influence the physical processes and determine the HTS behaviour following large and small break LOCA when the ECC injection is available. These phenomena also govern the early HTS behaviour following large and small break LOCA when ECC injection is unavailable. However, in case of LOECC, the HTS coolant inventory continues to decrease during the extended blowdown. The HTS coolant density continues to decrease, and steam convective and radiative cooling of the fuel dominates (since there is no ECC to refill the HTS). Ultimately fuel channel deformation *(TH18)* leading to heat transfer to the pressure tube and cooling of the fuel channels by the moderator system is the dominant phenomenon. Non-condensable gas effect (*TH23*) is considered important for LOCA with LOECC because of the hydrogen generated from the metal-water reaction.

Eventually, quasi-steady-state cooling is achieved where all of the channels are cooled by the moderator via heat transfer through the pressure tube and calandria tube walls.

The system thermalhydraulic phenomena for the limiting single channel events are similar to those discussed in Section 4.4.3.

10.4.4 Fuel and Fuel Channel Thermal-Mechanical Effects

10.4.4.1 LOCA/LOECC

The LOCA/LOECC scenario differs from a LOCA event primarily in the following aspects:

a) maximum fuel and sheath temperatures are higher,
b) fuel sheath oxidation is more extensive,
c) there is a potential for large quantities of fission products and hydrogen to be generated and released,
d) heat transfer from the fuel channels to the moderator provides the heat sink at later stages, and
e) substantial fuel bundle deformation occurs.

The phenomena discussed in Sections 3.4.4 and 4.4.4 are also relevant to LOCA combined with loss of emergency coolant, except that, for the LOECC event, the fuel and fuel channel temperatures are higher. In addition, pressure tube sag is an important phenomenon in this scenario. Pressure tubes will continue to heat up until their strength decreases sufficiently that they will sag into contact with calandria tubes and cool by heat transfer to the moderator *(FC16, FC17).* Primary phenomena include those related to heat up and oxidation *(FC1, FC10, FC20)* of fuel and channel materials. This oxidation generates additional localized heating *(FC10, FC20)*, and a significant amount of hydrogen which is released into containment.

Certain phenomena, like flashing coolant hydrodynamic transient in the moderator (*FC23*), high temperature channel debris interaction with water (*FC24*) and ruptured channel projectile formation and impact (*FC25*) are of importance in limiting single channel events, but not in LOCA with LOECC.

10.4.4.2 Severe Flow Blockage and Stagnation Feeder Break

For both severe flow blockage and stagnation feeder break events, it is likely that the affected pressure tube ruptures prior to contact with the calandria tube. However, the possibility of ballooning contact prior to rupture cannot be ruled out entirely. If such contact occurs, the pressure loading is transferred to the calandria tube. Subsequently the calandria tube could dryout and fail *(FC19).* If the calandria tube does not fail the fuel would continue to heat up. The fuel eventually operates dry at full power, and ultimately sheath melting, fuel melting, and Zircaloy/UO_2 dissolution could occur. Molten material contacting the pressure tube would rapidly fail both the pressure tube and the calandria tube.

Alternatively there is the potential for formation of local hot spots *(FC22)* on the pressure tube by molten material contact *(FC11, FC12 and FC15)* or forced element contact prior to pressure tube ballooning or failure. If such a hot spot were to develop on the pressure tube prior to contact with the calandria tube, it is likely that local strain-to-rupture could occur. The governing fuel and fuel channel parameters to assess the potential for local pressure tube rupture due to hotspot formation are sheath/end cap/end plate temperatures, axial expansion of the fuel string, axial expansion/contraction of the pressure tube, pressure tube temperatures, bundle mechanical deformation *(FC12)*, pressure tube deformation *(FC18)*, and heat transfer between the bearing pad or sheath and the pressure tube *(FC22)*.

The sequence of events for a severe single channel event is takes place rapidly. The total duration from the initiation of a severe flow blockage to calandria tube failure is on the order of 10 seconds. There are several alternative pathways that all lead to rapid failure of the calandria tube.

The relative and absolute heat-up rates of the fuel and sheath versus the pressure tube, and the pressure tube strain rate, are important during flow blockage and stagnation feeder break events. The absolute value of sheath temperature and degree of sheath oxidation determine the margin to melt and, if sheath melting occurs, how mobile any melt would be. Sheath heat-up rates compared to the rate of pressure tube heat-up (and straining) indicate if fuel element-to-pressure tube contact would occur and/or if the formation of any molten material can occur prior to pressure tube contact with the calandria tube or pressure tube failure.

The key parameter used to determine the likelihood of pressure tube contact with the calandria tube is the pressure tube strain rate. The pressure tube strain rate is a derived parameter depending upon the pressure tube temperature, the system pressure and the pressure tube material properties.

10.4.4.2.1 Fuel Channel Failure

For limiting single channel events reaction forces including moderator hydrodynamics *(FC23)* from the failed channel have the potential to lead to failure of additional channels or to interfere with shutdown *(FC25)*.

High-temperature channel components (specifically fuel elements and fuel bundle components) can potentially be expelled into the moderator following channel rupture. Fuel-to-moderator interaction (FMI) between the hot (or possibly molten) channel components and the subcooled moderator could produce a significant volume of steam and generate hydrodynamic transients within the liquid moderator *(FC24)*. The intensity of the hydrodynamic transient is primarily determined by the rate at which the channel debris is delivered to the moderator water, and the rate of heat transfer from the debris to the moderator. Debris fragmentation influences the surface area available for interaction between the debris and moderator and is also a key factor in the amount of steam generation and the subsequent magnitude of the hydrodynamic transients.

10.4.5 Moderator and Shield System Thermalhydraulics

Phenomena relevant to large LOCA/LOECC determine the quasi-steady-state temperature and pressure distributions in the moderator. The pressure distribution is the result of the combined cover gas and static pressure head at various locations in the core prior to this event. The quasi-steady-state and transient moderator temperature distributions are governed by the phenomena identified in Section 2.2.5. These phenomena include the interaction of moderator flow with calandria tubes *(MH10)*, moderator flow turbulence *(MH11)*, moderator buoyancy

(MH12), and moderator inlet jet development *(MH13)*. The primary phenomena also include calandria tube/moderator heat transfer *(MH22)* (which is particularly important after pressure tube/calandria tube contact occurs, and the moderator becomes the main heat sink for the channels), and moderator heat exchanger response *(MH36)*. The moderator heat exchanger response determines the rate of energy removal from the moderator.

Phenomena related to SDS2 performance, such as injection of poison *(MH15)*, and moderator/coolant poison mixing *(MH19)*, become important during the transient. Moderator mixing is a component in demonstrating effective shutdown system action in the long term. For a small out-of-core LOCA/LOECC event, the relevant phenomena are the same as those described above for a large break LOCA/LOECC event.

For in-core LOCA events, the coolant and possibly the fuel from the broken fuel channels discharge into the moderator, the moderator level increases and fluid discharges through the moderator rupture discs. For this event, the important phenomena are: failed channel interaction with core components *(MH30)*, liquid, vapour and two phase discharge *(MH41)*, moderator swell *(MH42)* and moderator heat exchanger response *(MH36)*, in addition to those associated with out-of-core LOCA/LOECC.

10.4.6 Fission Product Release and Transport to Containment

For a LOCA/LOECC event, the fuel behaviour and fission product releases are identical to those discussed in Sections 3.4.6 and 4.4.6 for the LOCA scenario up to the point at which emergency core coolant is injected into the primary heat transport system. The failure of the ECC injection results in continued reduction of the primary heat transport system coolant inventory. Channel flows decrease and fission product decay causes fuel temperatures to increase. The resulting steam and elevated fuel temperatures can cause sheath oxidation. This exothermic reaction produces additional heat and contributes to increased fuel temperatures. Rising fuel temperatures increase the probability of sheath failures and enhance diffusion of fission products out of the UO_2 fuel matrix. If either the sheath becomes completely oxidized or the sheath ruptures, the UO_2 may be exposed to a steam environment resulting in fuel oxidation, and enhancing fission product release. The resulting fuel and sheath temperatures are lower than the UO_2 and ZrO_2 melting temperatures, respectively.

The detailed characterization of the dominant phenomena for fission product release and transport to containment for large LOCA/LOECC is discussed in Section 10.4.6.1 and that for stagnation feeder break and severe flow blockage in Section 10.4.6.2.

10.4.6.1 Large LOCA/LOECC

10.4.6.1.1 Extended Blowdown Phase

The fission product release and transport phenomena occurring during the early blowdown phase are identified in Section 3.4.6.2.

10.4.6.1.2 Steam Cooling/Heat Rejection to the Moderator

Fission Product Release

During the late heat up phase, the temperatures of high-powered fuel elements may be high. Therefore, the diffusion *(FPR2)* and grain boundary sweeping/grain growth *(FPR3),* occurring at these severe temperatures, will play a significant role in the overall releases.

The failure of a sheath may allow steam to enter through the rupture site, or oxygen may diffuse through an intact, completely oxidized section of sheath. In these cases, some of the fuel may be exposed to an oxidizing environment. The fragility of completely oxidized sheath makes it virtually certain that the main mechanism of oxygen transmission to the fuel will be via cracks in the sheath. Therefore, the fuel may be susceptible to enhanced releases due to UO_{2+x} formation *(FPR9)* during a LOCA/LOECC transient. Under these conditions fuel oxidation may play a primary role in fission product release.

The channel will contain hydrogen produced as a product of the chemical reaction between the steam and the hot zirconium sheaths. Since sheath failures are expected, hydrogen may have access to the fuel. Therefore, enhanced fission product releases due to UO_{2-x} formation *(FPR11)* may occur. In some locations, UO_2 liquefaction due to UO_2-Zircaloy interaction *(FPR12)* will also occur; the duration of the high-temperature transient is sufficient to allow the diffusion of oxygen out of other sections of the fuel matrix, providing a further mechanism for UO_{2-x} formation *(FPR11).*

The sheaths may become nearly completely oxidized for the high-powered elements, thereby reducing the amount of zirconium available to sustain the zirconium-steam exothermic reaction. With a reduced heat production associated with the chemical reaction, the fuel temperatures are reduced to values that correspond to a heat balance between the decay heat generated and the heat removal by radiation to the pressure tubes (through the calandria tubes to the moderator) and convective cooling. Following this cool-down phase, the ruptured elements may still be exposed to a steam or steam/hydrogen environment and the fuel temperatures for the high-powered elements may remain elevated.

Other primary fission product release phenomena include: grain boundary coalescence/tunnel interlinkage *(FPR4)* for high-volatility fission products, fission product vaporization/volatilisation *(FPR15),* and matrix stripping *(FPR16)* for lower-volatility fission products, and enhanced releases due to temperature transients *(FPR17).*

Since the fuel temperature during a LOCA/LOECC scenario is lower than the threshold temperature (~2400K) for onset of grain boundary separation *(FPR18),* grain boundary separation should not contribute to the overall releases.

Sheath failure does not necessarily result in the immediate release of all fission products contained in the gap; that is, the gap inventory of some fission products (e.g., Iodine) may chemically combine with other fission products (e.g., Cesium) and be retained on the fuel and sheath surfaces. However, noble gases such as krypton and xenon in the gap are mostly released at the time of sheath failure, since these isotopes do not react with other elements. During a LOCA/LOECC scenario, some sheaths will be fully oxidized by the steam flow present in the channel, and, as a result, any fission products residing or adhering to these sheaths may be released (e.g., tellurium can chemically react with the sheath under certain conditions and will be released once the sheath is fully oxidized). Retention *(FPR8)* and gap transport *(FPR7)* of fission products in the fuel-sheath gap are classified as secondary contributors to the overall releases.

Fission Product Transport

Various fission product phenomena are important for different fission products and for different locations in the HTS. Fission products and structural materials *(FPT24)* will be released in vapour form from the fuel elements with defected/failed sheaths. The noble gases will not be affected by any transport phenomena other than simple gas transport considerations. The chemical speciation *(FPT23)* of some fission products and structural materials will allow their nucleation as aerosols *(FPT4)* in regions of lower temperatures (in the coolant stream near the pressure tube or when the coolant stream passes into a cooler region of the channel or feeder). Aerosols may also nucleate if the coolant stream changes in chemical nature (e.g., if more of the steam reacts to form hydrogen). The aerosols will grow by condensation of other fission product compounds around these nuclei *(FPT10)*. Aerosol agglomeration will occur in the turbulent flow *(FPT7)* and by gravitational settling *(FPT5)*. Aerosol deposition in the channel will be dominated by thermophoretic deposition *(FPT11)*, gravitational deposition *(FPT13)*, turbulent deposition *(FPT15)*, and resuspension *(FPT20)* in the turbulent flow. Resuspension may become important during any flow transients that occur in the scenario.

Fission product compounds may also deposit directly on surfaces from the vapour phase *(FPT2)*, once the appropriate condensation temperature is reached. Revaporization *(FPT2)* may occur if the surface temperature increases or the gas composition changes significantly.

In the end-fittings and feeder pipes, the temperatures and materials will probably be suitable for interaction of fission products in the vapour phase with structural materials, as discussed in Section 10.4.6.2.

Many of the deposited fission products may be released from the HTS if liquid water flows over the pipe surface *(FPT22)*. Liquid flows may occur during flow transients, or if delayed triggering of the ECC injection occurs. The dissolved or resuspended fission products will be transported out of the HTS in the break discharge flow.

Other phenomena of secondary importance can occur. Brownian motion agglomeration *(FPT6)* and deposition *(FPT14)* will also occur, though the primary phenomena are expected to dominate. Fuel particulate material may be formed from fuel elements that exhibit extensive sheath degradation, and can be transported *(FPT1)* by very rapid gas flows or by liquid flow. If condensing conditions occur at any point along the flow path, diffusiophoretic deposition *(FPT12)* may also become important, but condensing conditions downstream of the fission product releases are unlikely during this scenario. Similarly, pool scrubbing *(FPT21)* of fission product aerosols and vapours may become important during flow through a partly-voided header or other section of the HTS, but the header and feeders are considered to be completely voided during the stages of high fission product release in a LOCA/LOECC scenario.

10.4.6.2 Stagnation Feeder Break and Severe Flow Blockage

In a stagnation feeder break, a break occurs in an inlet feeder pipe. The coolant flow in the affected channel is so small (whether in the forward or reverse direction) that the channel rapidly becomes filled with superheated steam. If the flow is in the forward direction, the radionuclides released from failed fuel elements are initially transported to the reactor outlet header, where they become well mixed and diluted with the coolant prior to being discharged through the break. If the flow is in the reverse direction, they will be discharged through the break, into the reactor vault and without significant dilution. Depending on the void fraction in the piping between the fuel and the break, the fission products may be entrained in water or steam.

A flow blockage to a single channel will also cause degraded fuel cooling and formation of superheated steam. Prior to channel failure in the severe flow blockage scenario, fission products are released into, and transported by, coolant flow within the HTS, but no paths are available for release from the HTS other than possible pre-existing leaks in steam generator tubes.

In stagnation feeder break and severe flow blockage scenarios, the fuel continues to operate at full power during the event and the internal pressure in the channel remains near the normal operating level. Both the fuel and the pressure tube heat up rapidly, and the sheath may reach its melting temperature and dissolve UO_2 from the fuel. Following sheath failure, the fuel may experience oxidizing conditions in steam or reducing conditions in hydrogen. In the most severe fuel temperature transients, the pressure tube and calandria tube both fail due to molten material contact. Once the pressure and calandria tubes fail, most of the released radionuclides pass into the moderator. Some of these radionuclides are discharged with the moderator through the calandria relief ducts. Many radionuclides, including Iodine are washed out, and noble gases become dispersed in the liquid. Noble gases and some radioiodine-bearing liquid aerosols become airborne within containment.

The fission product release and transport phenomena that contribute to the quantity of radionuclides released into the containment during a stagnation feeder break or severe flow blockage event are presented by accident behaviour phase.

10.4.6.2.1 Heat-up (Prior to Sheath Melting)

Fission Product Release

The fuel temperatures increase rapidly during stagnation feeder break and flow blockage events, as a result of the power/cooling mismatch in the affected channel (reactor at full power, and low flow in the affected channel). The high fuel temperatures result in a substantial redistribution of the fission product inventory (from the UO_2 matrix to the grain boundaries, to the grain boundary tunnels, and finally to the free gap volume). The primary phenomena applicable for this redistribution are diffusion *(FPR2)*, grain boundary sweeping/grain growth *(FPR3)*, grain boundary coalescence/tunnel interlinkage *(FPR4)*, temperature transients *(FPR17)*, and grain boundary separation *(FPR18)*. For highly degraded cooling conditions and high power fuel, fuel melting *(FPR14)* and vapour transport/columnar grain growth *(FPR5)* also occur. The composition of fission products released during this phase is affected by fission product vaporization/volatilization *(FPR15)*. If fuel elements with failed sheaths are present in the affected channel during the pre-transient phase, gap transport *(FPR7)*, gap retention *(FPR8)*, and UO_{2+x} formation *(FPR9)* will be primary release phenomena for the failed fuel.

Athermal release *(FPR1)* will also occur as a secondary phenomenon, because the reactor is still at full power. Depending on the sheath temperature, UO_2-Zircaloy interaction *(FPR12)* will be of secondary importance in causing release from some fuel elements.

Fission Product Transport

The primary fission product transport phenomenon during this phase is transport of deposits by water *(FPT22)*.

Prior to sheath failure by melting in the stagnation feeder break and flow blockage accident scenarios, the main fission products present in the HTS are dissolved fission products from fuel with previously defected sheaths. An additional amount of fission products will be released if the defected bundle is in the channel with impaired flow. Because of the limited steam supply during this period, transport of structural materials *(FPT24)*, such as tin from fuel sheathing will be of secondary importance.

10.4.6.2.2 Heat-up (Following Sheath Melting)

Since the fuel and sheath temperatures are expected to increase at a high rate, any steam residing in the channel will have a short period of time to oxidize the sheath. Therefore, sheath melting is expected to occur at temperatures close to the melting point of as-received zirconium.

Fission Product Release

In addition to the phenomena identified during the first stage of the accident, melting of the sheath and bundle end plates will cause UO_2 dissolution by molten Zircaloy *(FPR13)*.

In some locations, coolant access to the fuel will cause UO_{2+x} formation *(FPR9)*. Matrix stripping *(FPR16)* may also occur as a secondary phenomenon if sufficient oxygen is present after sheath failure to oxidize the UO_2.

Fission Product Transport

Many of the fission product transport phenomena will be important for different fission products and for different locations in the HTS. Fission products and structural materials *(FPT24)* will be released in vapour form from the fuel elements with defected/failed sheaths. The noble gases will not be affected by any transport phenomena other than simple gas transport considerations. The chemical speciation *(FPT23)* of some fission products and structural materials will allow aerosol nucleation *(FPT4)* when their temperature drops (in the coolant stream near the pressure tube or when the coolant stream passes into a cooler region of the channel or feeder). Aerosols may also nucleate if the coolant stream changes in chemical nature (e.g. if more steam reacts to form hydrogen). Aerosol growth/revaporization *(FPT10)* will cause growth by condensation of other fission product compounds around these nuclei. Aerosol agglomeration will also occur in the turbulent flow *(FPT7)* and by gravitational settling *(FPT5)*. Aerosol deposition in the channel will be dominated by thermophoretic *(FPT11)*, gravitational *(FPT13)*, and turbulent-flow *(FPT15)* phenomena, and by resuspension in the turbulent flow *(FPT20)*. Resuspension may become important during any flow transients that occur in the scenario, particularly following channel failure (see Section 10.4.6.2.3).

Fission product compounds may also deposit directly on surfaces from the vapour phase *(FPT2)*, once the condensation temperature is reached. Revaporization may occur if the surface temperature increases or the gas composition changes significantly.

In the channel, end-fitting and feeder pipe, the temperatures and materials will probably be suitable for interaction of fission products in the vapour phase with structural material *(FPT3)*; most notably, Cesium, Iodine and tellurium compounds will react with stainless steel, carbon steel, Inconel alloys and their oxides. The rate of these interactions will depend on the temperature, the chemical condition of the surface, and chemical speciation *(FPT23)* of the fission products. Aerosol deposition will also occur at these locations, by inertial deposition *(FPT18)* due to the many changes in direction as the flow passes through the complex component geometry, as well as by the mechanisms mentioned above.

Condensing conditions are likely to occur in the end-fitting and feeder pipe during a stagnation feeder break or flow blockage, causing diffusiophoretic deposition *(FPT12)* to become important. Similarly, pool scrubbing *(FPT21)* of fission product aerosols and vapours will be important during passage through a partly-voided end-fitting or feeder pipe. Many of the deposited fission products may be re-dissolved or resuspended if liquid water flows across the pipe surface *(FPT22)*. Liquid flows may occur during flow transients, or when emergency core cooling is triggered. In the stagnation feeder break scenario, the dissolved or resuspended fission products will be transported out of the HTS in the fluid flow, while they will remain within the HTS during the flow blockage scenario.

Secondary phenomena in this phase include: Brownian motion (diffusional) agglomeration in HTS *(FPT6)*, electrostatic agglomeration *(FPT9)*, Brownian motion deposition *(FPT14)*, electrostatic deposition *(FPT17)*, photophoretic deposition *(FPT19)* and transport of structural materials *(FPT24)*.

10.4.6.2.3 After Channel Failure

Fission Product Release

Since the channel remains at full pressure, the pressure tube and calandria tube may strain and fail. Following channel failure, some or all of the fuel bundles (as well as the molten material, if any) may be ejected into the moderator. Fuel cracking *(FPR6)* may be induced by the rapid quenching from high temperature.

In a severe flow blockage, fuel elements, which remain in the ruptured channel between the channel rupture location and the unblocked side, will be cooled by the flow to the break location. Fuel elements that remain in the ruptured channel between the channel rupture location and the blocked side will experience minimal flow originating from the downstream end of the blockage. Depending on the exact location of the fuel, the blockage and the rupture in the fuel channel, some of the fuel may not be efficiently cooled by the increased primary circuit flow in the affected channel and may remain hot until the reactor is shut down.

Fuel elements that fail will release some or their entire gap inventory of fission products. For fuel in which the sheath failure is small, gap transport *(FPR7)* and gap retention *(FPR8)* are of primary importance. Some of the fuel elements may be severely damaged to the extent that the UO_2 fuel is ejected from the fuel sheath. Ejected fuel elements and fragments of fuel are well cooled by the moderator, so their temperatures (even in failed elements) remain low. The primary phenomenon applicable to fuel outside its sheath is fission product leaching *(FPR19)*.

Fission Product Transport

The transport phenomena of primary importance during this period of high flow following channel failure are fuel particulate suspension *(FPT1)*, aerosol resuspension *(FPT20)*, pool scrubbing *(FPT21)*, and transport of deposits by water *(FPT22)*. Cracked fuel fragments and previously molten U-Zr-O particles will be entrained *(FPT1)* in the water flows and rapid steam flows associated with channel rupture. Aerosols will be resuspended *(FPT20)* by the turbulent steam in advance of the quench front and deposited fission products will be transported by water *(FPT22)* after the quench front has passed by. Fission products will be retained in the moderator liquid by pool scrubbing *(FPT21)*. Chemical speciation *(FPT23)* is considered to be of secondary importance, as it will only affect whether the fission products are transported in solution or as suspended particles. If the calandria tube does not fail following pressure tube failure, but the annulus bellows do fail, some coolant will discharge directly to the reactor vault and radionuclides released to the coolant will wash out to some extent with this coolant flow through the failed bellows. The primary phenomena will be the same as above except that pool scrubbing does not occur.

In this phase of the severe flow blockage scenario, a fission product discharge path is opened into containment through the calandria vessel and its relief ducts, or through the calandria tube and its annulus bellows.

10.4.7 Containment (Containment Thermalhydraulics, FP Chemistry, Hydrogen, Aerosols)

The primary safety concern related to containment in the LOCA/LOECC event is containment integrity associated with pressurization during the discharge of fluid from the primary circuit and from potential hydrogen ignition. The primary phenomena determining elevated containment pressure during the blowdown phase are related to the total inventory and energy content of steam and water discharged into containment during the event *(C1 to C6, C8)*. This inventory depends on the behaviour of the HTS. Since energy discharge from the primary circuit continues practically indefinitely, heat transfer to local air coolers and building structures is of primary importance in limiting containment pressure and establishing the transient pressurization in containment. Heat transfer to the containment walls, in turn, develops stresses in the structure over a long time period. These stresses develop slowly as heat transfer through concrete is relatively slow compared with the discharge from the cooling circuit.

Containment pressure and temperature conditions are strongly influenced by phenomena such as break discharge flashing *(C1)*, heat removal by local air coolers *(C6)*, and heat transfer to the containment walls *(C3-C5)*. Containment pressure/temperature conditions may also depend on phenomena related to hydrogen behaviour, such as buoyancy and momentum induced mixing *(C11, C12)*, hydrogen vented deflagration *(C13)* and turbulent combustion *(C14)*, standing flames *(C15)*, and deflagration to detonation transition *(C16)*. The hydrogen behaviour will also be influenced by hydrogen recombiners *(C17)*. The chemical reaction between zirconium sheath and steam produces significant amounts of hydrogen, which enters containment through the HTS break. If the concentration of hydrogen in some parts of the containment reaches the flammable limit and an ignition source exists, hydrogen combustion may occur. However, local hydrogen concentrations are not high enough to result in deflagration to detonation transition (DDT).

The concentration of hydrogen in any part of containment depends on the extent of mixing of hydrogen with the containment environment, which depends on the hydrogen generation rate, break discharge rate, containment design, etc. The hydrogen distribution is needed to assess the possibility and effect of a hydrogen burn and to determine appropriate locations for recombiners. Recombiners are placed in certain containment regions to avoid a slow build up of accumulating hydrogen, such as in the containment dome and fuelling machine vaults.

Fission product attenuation inside containment is governed by processes related to water and solid (particulate) aerosol growth, retention and removal phenomena and includes aerosol removal due to jet impingement *(C25)*, gravitational settling *(C26)*, removal in HEPA filters *(C30)*, and removal due to leak path *(C32)*. Fission product release from containment is also governed by these attenuation processes, as well as secondary phenomena related to Iodine behaviour (partitioning of the gas phase and water borne Iodine, *C18, C19, C21, C22, C23 and C24*), leakage from containment and filter performance.

For events involving in-core breaks, phenomena relating to fission product retention (e.g. *FPR15*, *FPR19*) in the moderator are also important, as they influence the amount and form of fission products entering containment.

10.4.8 Radiation Physics

The main radiation physics concerns relate to habitability analyses which ascertain whether station personnel can perform required safety functions following an accident. The primary areas of concern in a plant for personnel are the main control room and secondary control building. Habitability analyses are used to establish shielding and procedural recommendations for post-accident access to the plant.

Radionuclides released to containment through the breach of the HTS and airborne pathways constitute sources of radiation from which doses to personnel are assessed. Doses in the control room and other areas of importance, including passageways to and from rooms, are considered. The radioactivity inventory in the fuel prior to the accident establishes the radionuclide release to containment. Calculations of isotope generation and depletion *(RAD2)* also consider the core power transient *(RAD6)* following shutdown.

The releases to containment result in gamma and beta radiation sources *(RAD1)* which are used to calculate radiation fields *(RAD4)* in critical plant areas. Airborne concentrations are also assessed which yield inhalation doses *(AD16)* dominated by Iodine-131 and tritium. Dose conversion factors are then used to assess worker doses *(RAD7)* from fields and airborne radionuclides following the accident.

For single channel events some phenomena such as external exposure (*RAD7*) and radiolysis (*RAD8*) are a primary concern.

10.4.9 Atmospheric Dispersion

Atmospheric diffusion is primarily governed by external atmospheric conditions as discussed in Appendix A. LOCA and limiting single channel accident conditions relevant to dispersion include phenomena related to aerosol settling and the location of leakage paths through the containment structure as these input factors from containment analysis affect dispersion through natural convection and settling (Section 3.4.9).

10.5 Summary

LOCA events with failure of ECC generally involve the same phenomena as LOCA events with ECC available (Chapters 3 and 4) up to the time of ECC injection. Subsequent to this, reactor physics considerations involve only the flux and power distribution due to decay heat, while fuel channel thermal-mechanical effects are more severe than those discussed in Chapters 3 and 4. Fuel channel and system thermalhydraulics concerns are also the same as those for a large

LOCA with ECC available until heat transfer to the moderator begins to dominate in the quasi-steady-state.

Moderator thermalhydraulics plays a primary role in the LOCA with LOECC accident scenario with the moderator acting as the primary heat sink via increased heat transfer from fuel through the calandria tubes. Containment pressurization from the LOCA discharge or from potential hydrogen addition is of primary importance, as are radiation physics effects mainly related to the habitability of control areas for reactor operators.

Fission product release and transport considerations are dominated by temperature and chemical effects due to decay heat and the reaction of the zirconium sheath with steam to produce hydrogen, and by oxidation of UO_2 by steam. Transport phenomena lead to the retention of some of these fission products in the HTS, reducing the amount discharged into containment.

Single channel accidents are analyzed in phases established by containment pressurization, reactor shutdown, ECC injection, pressure tube deformation for the affected fuel channel and long-term containment pressure suppression.

Fuel channel and system thermalhydraulic considerations are dependent on the specific single channel event, but heat transfer to the secondary coolant, break discharge rates, and degraded cooling to the affected channel (as the reactor continues at high power), are the important considerations. Fuel melt, pressure tube failure, and pressure tube/calandria tube contact are dominant thermal-mechanical effects for limiting single channel events, as the continued full power operation lasts for some time during the transient. Fission product release and transport phenomena are most important in severe flow blockage and stagnation feeder break accidents.

11. SUMMARY

This Technical Basis Document for Computer Program Validation relates ACR safety analysis scenarios to the fundamental physical phenomena that form the basis for computer modelling of these accident scenarios. The document describes the categorization of accident scenarios and physical phenomena of importance for computer code validation. It provides an organized approach to the management of detailed data for computer program validation.

The identification and ranking of phenomena is not a new process but an evolutionary one. A comprehensive review of the phenomena has been performed and all relevant phenomena have been identified and ranked according to their impact on the event key safety concerns.

The Technical Basis Document addresses design basis events and limited core damage accidents and then identifies system behaviour that results from complex interactions of important phenomena. The governing phenomena are then identified for use by the validation matrix teams.

12. GLOSSARY AND ACRONYMS

Actinide	Any element of atomic number 89 (actinium) through 103 (lawrencium) inclusive.
AECB	See **CNSC**.
ASDV	Atmospheric Steam Discharge Valve.
Aerosol	Suspension of solid or liquid particles in a gas.
Bearing Pad	Small metal pads on the outer elements of a fuel bundle preventing direct fuel-element-to-pressure-tube contact.
Behaviour	The way a machine, element, etc., acts or functions. (Webster) The manner in which a system, a component of a system, or a group of phenomena functions. This is usually described qualitatively in scientific terms, and often can be described by one or more mathematical or symbolic equations, or by a graphical representation of relationships between dependent and independent variables.
Blowdown	Process of depressurization and coolant discharge resulting from a piping failure.
CANDU	CANada Deuterium Uranium.
CHF	Critical Heat Flux.
CNSC	Canadian Nuclear Safety Commission (formerly the Atomic Energy Control Board (AECB)).
CSDV	Condenser Steam Discharge Valve.
Computer Program	Sequence of instructions which are readable and suitable for processing by a computer.
Critical Break	Hypothetical piping failure of a flow area which minimizes flow in some section of the coolant circuit containing fuel.
CT	Calandria Tube.
Data Set	Collection of operational data, experimental measurements, analytic solutions, single effect tests, integral tests, etc.
DBA	Design Basis Accident, Accident conditions against which a nuclear power plant is designed according to established design criteria, and for which the damage to the fuel and the release of radioactive material are kept within authorized limits.
DDT	Deflagration to Detonation Transition.
Deflagration	Combustion with a flame velocity substantially sub-sonic relative to the fluid.
Discipline	Technical specialty or sub-section of a reactor system providing a logical framework for safety analysis within the Canadian Nuclear Industry.
ECC	Emergency Core Coolant.

End-Fitting	Component installed at the end of pressure tubes allowing for connection to feeder pipes and a fuelling machine.
Feedback	The return to the input of a part of the output of a machine, system, or process. (Webster) Neutronic, thermalhydraulic or other reactor behaviour in which changes impart an influence on the system itself and invoke further changes.
Feeder	Piping that connects end fittings to headers.
Fission Product	Residual nuclear fragment of a nuclei having undergone fission; activation products such as Co-60 and H-3 are often included within this general term as well.
Flow Blockage	Unintended obstruction of the flow in, or to, a channel.
Fuel Bundle	Fuel assembly of several fuel elements bound together, as inserted into the reactor.
Fuel Element	Single closed Zircaloy tube fuel bundle component filled with fuel pellets.
Fuel Failure	Rupture of the fuel sheath, possibly leading to the release of fission products and/or other radionuclides.
Fuel Pellet	Smallest cylindrical uranium dioxide fuel component.
Gas Gap	Free volume in a fuel element intended to contain gases released from fuel pellets.
Header	Piping manifold.
HTS	Heat Transport System.
Large Break	Failures in large piping (headers, lines to pumps and steam generators), up to a full pipe severance (guillotine failure).
LOCA	Loss of Coolant Accident.
LOECC	Loss of Emergency Core Coolant.
LOPIC	Loss of Pressure and Inventory Control.
LTC	Long Term Cooling.
MSSV	Main Steam Safety Valve.
Phenomenology	A descriptive or classificatory account of the phenomena of a given body of knowledge, without any further attempt at explanation. (Webster)
Phenomenon/phenomena	Any event, circumstance, or experience that is apparent to the senses and that can be scientifically described or appraised. (Webster) An event or circumstance that: affects the process of changing the physical state of the system, is either directly apparent to the senses or is indirectly apparent by means of measurements of the physical state of the system, and can be represented quantitatively by a model or correlation.

PRV	Pressure Relief Valve.
Pressure Suppression	System intended to keep containment pressure substantially below that which would ensue if coolant were simply discharged into containment.
Process	A continuing development involving many changes. (Webster) The evolving transitions of state of the system which are characterized by the occurrence of, and interaction between, phenomena.
Process System	An assemblage of components comprising a system which performs a major function of normal reactor operations.
Property	A characteristic quality; any of the principal characteristics of the substance. (Webster) A property is a function of the state and depends only on the state and not on the method of change between two states. Properties can be either intensive or extensive. Intensive properties are those that do not depend on the extent or magnitude of the system, e.g., pressure, temperature. Extensive properties depend on the extent of the system, e.g., mass, volume, energy. Specific values, or normalized per unit values of extensive properties are considered to be intensive, e.g., density (mass per unit volume), enthalpy etc.
PT	Pressure Tube.
Puff Release	Postulated instantaneous release of radionuclides.
Radionuclide	Any radioactive atom (e.g. fission products and actinides produced during reactor operation).
Reactivity	Relative deviation of neutron multiplication from a value of 1.0 in a fission reactor core.
Refill	Process of replacement of discharged primary coolant with emergency coolant.
ROP	Regional Over-Power
RRS	Reactor Regulating System. Performs the normal operational control of reactor reactivity and power.
RWS	Reserve Water System.
SG	Steam Generator.
Sheath	Sealed Zircaloy tube encasing the uranium dioxide pellets in a fuel element.
Shutdown	Cessation of reactor fission power production, leaving only radioactive decay power as a source of heat generation in the core.
Sievert	Unit of effective radiation dose to living tissue.
Small Break	Used to describe piping failures of a size equivalent to guillotine failure of a single channel or its feeder piping or smaller.

Stagnation Feeder Break Hypothetical piping failure of a flow area which results in minimal flow in some section of the coolant circuit containing fuel.

Standing Start Temporary flow stagnation in some channels following core refill and HTS pumps trip, which may occur in breaks located at the inlet side of the HTS (e.g., inlet header break).

State A condition as regards physical structure, constitution, internal form, stage or phase of existence. (Webster)

The values assumed at a given instant by the variables (properties) that define the characteristics of a system. The passage from one set of values to another is termed a state transition.

Steady State Condition or period in which the characteristics of a system do not change with time.

Subcooling The difference between the local liquid temperature and saturation temperature at local conditions.

TBD Technical Basis Document.

Transient Period or condition when characteristics of a system change with time.

Trip Actuation of rapid reactor shutdown, often when one or more parameters fall outside of pre-determined acceptable ranges during a transient.

Validation Process of supporting or corroborating on a sound or authoritative basis. (Webster)

Zircaloy Name for an alloy of zirconium used for sheathing, bearing pads and calandria tube components of a reactor core.

13. REFERENCES

[1] "Separate Effects Test Matrix for Thermal-Hydraulic Code Validation", Vol. 1. "Phenomena Characterizations and Selection of Facilities and Tests", Vol. 2 "Facility and Experiment Characteristics", Report OECD/GD(94)82 and NEA/CNSI/R(93)14/Part 1/Rev., Paris, 1993.

[2] "Quality Assurance of Analytical, Scientific, and Design Computer Programs for Nuclear Power Plants General Instructions No. 1", N286.7-94, CSA, Toronto, 1994.

[3] Regulatory Guide, "Computer Programs Used in Design and Safety Analyses of Nuclear Power Plants and Research Reactors", G-149 (E), CNSC, October 2000.

[4] Moeck, E.O., J.C. Luxat, L.A. Simpson, M.A. Petrilli and P.D. Thompson. "Validation of Computer Codes Used in Safety Analyses of CANDU Power Plants", Proceedings of a Technical Committee Meeting, Mumbai, India, Jan. 29 - 1 Feb. 1996. IAEA-TECDOC-984, Advances in Heavy Water Reactor Technology, November 1997.

Table 3
Impacts of Phenomena on Accident Scenarios

Legend

Primary: A phenomenon is listed as having a ‘primary’ importance if it has a significant impact on one or more of the event key safety concerns.

Secondary: A ‘secondary’ designation is used for phenomenon which are not primary but have some impact on one or more of the event key safety concerns.

Blank: Phenomena which are determined to be neither primary nor of secondary significance during any phase of an accident, are left blank.

	PHENOMENON	Large LOCA	Small LOCA, Single Channel Accident	Secondary Coolant Failures	Fuel Handling Accidents	Loss of Regulation	Loss of Flow	Auxiliary System Failures	Limited Core Damage Accidents
	Reactor Physics								
PH1	Coolant-Density-Change Induced Reactivity	secondary	secondary	secondary	secondary	secondary	secondary		secondary
PH2	Coolant-Temperature-Change Induced Reactivity	secondary	secondary		secondary		secondary	secondary	secondary
PH3	Moderator-Density-Change Induced Reactivity		secondary					primary	secondary
PH4	Moderator-Temperature-Change Induced Reactivity		secondary[*]				secondary	secondary	secondary[+]
PH5	Moderator-Poison-Concentration-Change Induced Reactivity	secondary	secondary				secondary		secondary
PH6	Moderator-Purity-Change Induced Reactivity		secondary[*]						secondary[+]
PH7	Fuel-Temperature-Change Induced Reactivity	secondary	secondary		secondary		secondary	secondary	secondary[+]
PH8	Fuel-Isotopic-Composition-Change Induced Reactivity	secondary	secondary	secondary	secondary			secondary	secondary
PH9	Refuelling-Induced Reactivity	secondary							secondary
PH11	Device-Movement Induced Reactivity	primary	primary	primary	primary	primary	primary	primary	primary

+ Primary for limiting single channel accidents.

* Primary for single channel accidents.

	PHENOMENON	Large LOCA	Small LOCA, Single Channel Accident	Secondary Coolant Failures	Fuel Handling Accidents	Loss of Regulation	Loss of Flow	Auxiliary System Failures	Limited Core Damage Accidents
PH12	Prompt/Delayed Neutron Kinetics	secondary	secondary		secondary	secondary	secondary	primary	primary
PH13	Flux-Detector Response	secondary	secondary	secondary	secondary	secondary	secondary	primary	secondary
PH14	Flux and Power Distribution (Prompt/Decay Heat) in Space and Time	primary	primary	secondary	primary	primary	primary	primary	primary
PH15	Lattice-Geometry-Distortion Reactivity Effects	secondary	secondary		secondary				secondary
PH17	Core Physics Response to Moderator Level Change							primary	
	Fuel Channel and System Thermalhydraulics								
TH1	Break Discharge Characteristics and Critical Flow	primary	primary	primary	primary				primary
TH2	Coolant Voiding	primary	primary		primary	primary	primary	secondary	primary
TH3	Phase Separation	primary	primary	primary	primary		primary	primary	primary
TH4	Level Swell and Void Holdup	secondary	secondary	primary	secondary			secondary	secondary
TH5	HTS Pump Characteristics (Single and Two Phase)	primary	primary	primary	primary		primary	primary	primary
TH6	Thermal Conduction	primary	primary	primary	primary	secondary	secondary	secondary	primary
TH7	Convective Heat Transfer	primary	primary	primary	primary	primary	primary	primary	primary
TH8	Nucleate Boiling	secondary	primary	primary	primary	secondary	primary	primary	primary
TH9	CHF/Dryout and Post Dryout Heat Transfer	primary	primary		primary	primary	primary	primary	primary
TH10	Condensation Heat Transfer	primary	primary	primary	primary		primary	primary	++
TH11	Radiative Heat Transfer	primary	primary		primary	secondary	secondary	primary	primary
TH12	Quench/Rewet Characteristics	primary	primary		primary	primary	primary	primary	++
TH13	Zircaloy/Water Thermal-Chemical Reaction	primary	secondary		secondary				primary
TH14	Reflux Condensation		secondary	secondary	secondary		primary	primary	+++
TH15	Counter Current Flow	primary	primary	secondary	primary		primary	primary	+++
TH16	Flow Oscillations	secondary	secondary	secondary	secondary	secondary	primary	primary	+++
TH17	Density Driven Flows (Natural Circulation)	secondary	primary	primary	primary		primary	primary	secondary

++ Primary for limiting single channel events (with ECC available).

+++ Secondary for limiting single channel events (with ECC available).

	PHENOMENON	Large LOCA	Small LOCA, Single Channel Accident	Secondary Coolant Failures	Fuel Handling Accidents	Loss of Regulation	Loss of Flow	Auxiliary System Failures	Limited Core Damage Accidents
TH18	Fuel Channel Deformation	secondary	secondary		secondary				primary
TH19	Fuel String Mechanical-Hydraulic Interaction	secondary	secondary						primary
TH20	Waterhammer		secondary	secondary	secondary				
TH21	Waterhammer (Steam Condensation Induced)	secondary		secondary					
TH22	Pipe Thrust and Jet Impingement	secondary	secondary	secondary					secondary
TH23	Non-Condensable Gas Effect	secondary	secondary		secondary			secondary	primary
	Fuel and Fuel Channel Thermal-Mechanical Effects								
FC1	Fission and Decay Heating	primary	primary	primary	primary	primary	primary	primary	primary
FC2	Diffusion of Heat in Fuel	primary	primary	primary	primary	primary	primary	primary	primary
FC3	Fuel-to-Sheath Heat Transfer	primary	primary	primary	primary	primary	primary	primary	primary
FC4	Fuel-to-End Cap Heat Transfer	secondary							secondary
FC5	Fission Gas Release to Gap and Internal Pressurization	primary	primary	primary	primary	primary	primary	primary	primary
FC6	Sheath Deformation	primary	primary	secondary	primary	secondary	secondary	primary	primary
FC7	Sheath Failure	primary	primary	secondary	primary	secondary	secondary	primary	primary
FC8	Fuel Deformation	primary	primary	secondary	primary	secondary	secondary	primary	primary
FC9	Sheath Oxidation or Hydriding	secondary	primary	secondary	primary	secondary	secondary	primary	primary
FC10	Fuel Oxidation or Reduction	primary	secondary†	secondary	secondary	secondary	secondary	secondary	primary
FC11	Fuel or Sheath Melting and Relocation								secondary+
FC12	Bundle Mechanical Deformation	primary	secondary*		primary	secondary			primary
FC13	Sheath-to-Coolant and Coolant-to-Pressure Tube	primary	primary	primary	primary	primary	primary	primary	primary

† Primary for feeder off-stagnation break.

+ Primary for limiting single channel accidents.

* Primary for single channel accidents.

	PHENOMENON	Large LOCA	Small LOCA, Single Channel Accident	Secondary Coolant Failures	Fuel Handling Accidents	Loss of Regulation	Loss of Flow	Auxiliary System Failures	Limited Core Damage Accidents
	Heat Transfer								
FC14	Channel and Subchannel Flow Effects	primary	primary	secondary	primary	primary	primary	primary	primary
FC15	Local Melt Heat Transfer to Pressure Tube								secondary+
FC16	Pressure Tube-to-Calandria Tube Heat Transfer	secondary	secondary*	secondary	secondary	secondary	secondary	secondary	primary
FC17	Calandria Tube-to-Moderator Heat Transfer	secondary	secondary*	secondary	secondary	secondary	secondary	secondary	primary
FC18	Pressure Tube Deformation or Failure	secondary	secondary*		secondary	secondary			primary
FC19	Calandria Tube Deformation or Failure		secondary*		secondary	secondary			secondary+
FC20	Pressure Tube Oxidation or Hydriding	secondary	secondary		secondary				primary
FC21	Element-to-Pressure Tube Radiative Heat Transfer	primary	secondary*		secondary				primary
FC22	Element or Bearing Pad-to-Pressure Tube Contact Heat Transfer	primary	secondary*		secondary				primary
FC23	Flashing Coolant Hydrodynamic Transient in Moderator		secondary*		secondary				secondary+
FC24	High Temperature Channel Debris Interaction with Water		secondary*		secondary				secondary+
FC25	Ruptured Channel Projectile Formation and Impact		secondary*		secondary				secondary+
	Moderator and Shield System Thermal-hydraulics								
MH9	Moderator Pump Cavitation							primary	
MH10	Interaction of Moderator Flow with Calandria Tubes	secondary	secondary		secondary			primary	primary
MH11	Turbulence	secondary	secondary		secondary			primary	primary
MH12	Moderator Buoyancy	secondary	secondary		secondary			primary	primary
MH13	Moderator Inlet Jet Development	secondary	secondary		secondary			primary	primary
MH15	Injection of Poison	primary	primary	primary	primary	primary	primary	primary	primary

\+ Primary for limiting single channel accidents.

* Primary for single channel events.

	PHENOMENON	Large LOCA	Small LOCA, Single Channel Accident	Secondary Coolant Failures	Fuel Handling Accidents	Loss of Regulation	Loss of Flow	Auxiliary System Failures	Limited Core Damage Accidents
MH19	Moderator/ Coolant/Poison Mixing	primary	primary	primary	primary	primary	primary	primary	primary
MH22	Calandria Tube/Moderator Heat Transfer	secondary	secondary		secondary				primary
MH30	Failed Channel Interaction With Core Components		secondary[*]						secondary[+]
MH34	Hydrogen Deflagration							primary	
MH36	Moderator Heat Exchanger Response	secondary	secondary					primary	primary
MH41	Liquid, Vapour and Two-Phase Discharge		secondary[*]					primary	secondary[+]
MH42	Moderator Swell		secondary[*]					primary	secondary[+]
MH43	Thermal Conduction							secondary	
MH44	Convective Heat Transfer							secondary	
MH45	Radiative Heat Transfer							primary	
MH46	Moderator Degassing and Transfer Processes in Moderator Cover Gas							primary	
MH47	Interaction of End-Shield Flow with End-Shield Solid							primary	
MH48	Moderator cover gas pressure		secondary[*]					secondary	secondary
	Fission Product Release								
FPR1	Athermal Release	secondary	secondary	secondary	secondary	secondary	secondary	secondary	secondary
FPR2	Diffusion	primary	secondary[*]	secondary	secondary	secondary	secondary	secondary	primary
FPR3	Grain Boundary Sweeping and Grain Growth	primary	secondary[*]	secondary	secondary	secondary	secondary	secondary	primary
FPR4	Grain Boundary Coalescence and Tunnel Interlinkage	primary	secondary[*]	secondary	secondary	secondary	secondary	secondary	primary
FPR5	Vapour Transport and Columnar Grains	secondary	secondary[*]	secondary	secondary	secondary	secondary	secondary	secondary[+]
FPR6	Fuel Cracking (thermal)	primary	secondary[*]	secondary	secondary	secondary	secondary	secondary	secondary[+]
FPR7	Gap Transport (failed elements)	primary	secondary[*]	secondary	secondary	secondary	secondary	secondary	primary
FPR8	Gap Retention	primary	secondary[*]	secondary	secondary	secondary	secondary	secondary	primary

[*] Primary for single channel accidents.

[+] Primary for limiting single channel accidents.

	PHENOMENON	Large LOCA	Small LOCA, Single Channel Accident	Secondary Coolant Failures	Fuel Handling Accidents	Loss of Regulation	Loss of Flow	Auxiliary System Failures	Limited Core Damage Accidents
FPR9	UO_{2+x} Formation	primary	secondary[*]	secondary	secondary	secondary	secondary	secondary	primary
FPR10	U_4O_9 and U_3O_8 Formation		secondary[‡]		secondary				
FPR11	UO_{2-x} Formation								primary
FPR12	UO_2-Zircaloy Interaction		secondary		secondary				primary
FPR13	UO_2 Dissolution by Molten Zircaloy								secondary[+]
FPR14	Fuel Melting								secondary[+]
FPR15	Fission Product Vaporization/ Volatilization	secondary	secondary[*]		secondary				primary
FPR16	Matrix Stripping		secondary[†]		secondary				primary
FPR17	Temperature Transients	secondary	secondary[*]		secondary				primary
FPR18	Grain Boundary Separation	secondary	secondary[*]						secondary[+]
FPR19	Fission Product Leaching	primary	secondary[*]		secondary				secondary[+]
	Fission Product Transport								
FPT1	Fuel Particulate Suspension	secondary	secondary[†]	secondary	primary				secondary[+]
FPT2	Vapour Deposition and Revaporization of Deposits	primary	secondary[†]		primary				primary
FPT3	Vapour Structure/ Interaction	primary	secondary[†]		secondary				primary
FPT20	Aerosol Resuspension	secondary	secondary[†]		secondary				primary
FPT21	Pool Scrubbing	primary	secondary[†]						primary
FPT22	Transport of Deposits by Water	primary	secondary[†]		secondary				primary
FPT23	Chemical Speciation	primary	secondary[†]		secondary				primary
FPT24	Release of Structural Materials (Aerosol Nucleation and Growth)	secondary	secondary[†]		secondary				primary

‡ Primary for end fitting failure.

† Primary for feeder off-stagnation break.

\+ Primary for limiting single channel accidents.

	PHENOMENON	Large LOCA	Small LOCA, Single Channel Accident	Secondary Coolant Failures	Fuel Handling Accidents	Loss of Regulation	Loss of Flow	Auxiliary System Failures	Limited Core Damage Accidents
FPT4	Aerosol Nucleation	secondary	secondary[†]		secondary				primary
FPT10	Aerosol Growth/ Revaporization	secondary	secondary[†]		secondary				primary
	(Aerosol Agglomeration)								
FPT5	Gravitational Agglomeration in the HTS	secondary	secondary[†]						primary
FPT6	Brownian Motion in the HTS	secondary	secondary		secondary				secondary
FPT7	Turbulent Agglomeration in the HTS	secondary	secondary[†]						primary
FPT8	Laminar Agglomeration	secondary			secondary				
FPT9	Electrostatic Agglomeration	secondary	secondary		secondary				secondary
	(Aerosol Deposition)								
FPT11	Thermophoretic Deposition in the HTS	secondary	secondary[†]		secondary				primary
FPT12	Diffusiophoretic Deposition	secondary	secondary[†]		secondary				primary
FPT13	Gravitational Deposition	secondary	secondary[†]						primary
FPT14	Brownian Motion Deposition	secondary	secondary		secondary				secondary
FPT15	Turbulent Deposition in the HTS	secondary	secondary[†]						primary
FPT16	Laminar Deposition	secondary			secondary				
FPT17	Electrostatic Deposition	secondary	secondary		secondary				secondary
FPT18	Inertial Deposition	secondary	secondary[†]		secondary				primary
FPT19	Photophoretic Deposition	secondary	secondary		secondary				secondary

[†] Primary for feeder off-stagnation break.

	PHENOMENON	Large LOCA	Small LOCA, Single Channel Accident	Secondary Coolant Failures	Fuel Handling Accidents	Loss of Regulation	Loss of Flow	Auxiliary System Failures	Limited Core Damage Accidents
	Containment (Thermalhydraulics)								
C1	Flashing Discharge	primary	primary	primary	secondary			primary	primary
C2	Evaporation from Pools	primary	primary	primary	secondary				primary
C3	Convection Heat Transfer	primary	primary	primary	secondary			secondary	primary
C4	Conduction Heat Transfer	primary	primary	primary	secondary			secondary	primary
C5	Condensation Heat/Mass Transfer	primary	primary	primary	primary			secondary	primary
C6	Air Cooler Heat Transfer	primary	primary	primary	primary				primary
C8	Laminar/ Turbulent Leakage Flow	primary	primary	secondary	primary				primary
C9	Choked Flow	primary	primary	primary	secondary			secondary	secondary
C10	Liquid Re-Entrainment	secondary	secondary		secondary				secondary
	(Hydrogen Behaviour)								
C11	Buoyancy Induced Mixing	secondary	secondary		secondary				primary
C12	Momentum Induced Mixing	secondary	secondary		secondary				primary
C13	Hydrogen Vented Deflagration	secondary	secondary		secondary				primary
C14	Turbulent Combustion	secondary	secondary		secondary				primary
C15	Standing Flame								primary
C16	Deflagration/Detonation Transition								primary
C17	Removal by Recombiners	primary	secondary		secondary				primary
	(Iodine Chemistry)								
C18	Interfacial Mass Transfer	primary	primary		primary			primary	secondary
C19	Partition Coefficient		primary		primary			primary	secondary
C20	Adsorption	primary	secondary		secondary			primary	secondary
C21	Carbon Filter Removal Efficiency	primary	primary		primary			secondary	secondary
C22	Total Waterborne Iodine	primary	primary		primary			primary	secondary

	PHENOMENON	Large LOCA	Small LOCA, Single Channel Accident	Secondary Coolant Failures	Fuel Handling Accidents	Loss of Regulation	Loss of Flow	Auxiliary System Failures	Limited Core Damage Accidents
C23	Fraction Airborne Organic Iodine		primary		primary			primary	secondary
C24	Total Airborne Iodine		primary		primary			primary	secondary
	(Aerosol Behaviour)								
C25	Jet Impingement	primary	primary		secondary			primary	primary
C26	Gravitational Settling	primary	primary		primary			primary	primary
C27	Thermophoresis	secondary	secondary		secondary			secondary	secondary
C28	Diffusiophoresis	secondary	secondary		secondary			secondary	secondary
C29	Diffusional Agglomeration	secondary	secondary		secondary			primary	secondary
C30	Removal in HEPA Filters	secondary							primary
C31	Removal in Demisters	secondary							secondary
C32	Removal in Leakage Paths	secondary	primary		primary				primary
C33	Condensation	secondary	secondary		secondary			primary	secondary
C34	Evaporation	secondary	secondary		secondary			primary	secondary
C35	Turbulent Agglomeration	secondary	secondary		secondary			primary	primary
C36	Turbulent Deposition	secondary	secondary		secondary			primary	secondary
C37	Formation in a Flashing Jet	secondary	secondary		secondary			primary	secondary
C38	Formation in a Steam Jet	secondary	secondary		secondary				primary
C39	Gravitational Agglomeration	secondary	secondary		secondary			primary	secondary
C40	Inertial Deposition	secondary	secondary		secondary			primary	secondary
C41	Diffusional Deposition				secondary				secondary
	Radiation Physics								
RAD1	Radiation Emission	primary	primary		primary			primary	primary
RAD2	Isotopes Generation and Depletion	primary	primary	primary	primary			primary	primary
RAD3	Neutron Transport and Streaming	secondary	secondary	primary	secondary			primary	primary
RAD4	Photon Transport, Streaming and Skyshine	secondary	secondary		secondary			primary	primary

	PHENOMENON	Large LOCA	Small LOCA, Single Channel Accident	Secondary Coolant Failures	Fuel Handling Accidents	Loss of Regulation	Loss of Flow	Auxiliary System Failures	Limited Core Damage Accidents
RAD5	Electron Transport		secondary		secondary				
RAD6	Heating (Energy Deposition)	primary	primary		primary			primary	primary
RAD7	External Exposure	secondary	secondary*		secondary				primary
RAD8	Radiolysis	secondary	secondary*		secondary			primary	primary

* Primary for single channel accidents.

Table 4
Atmospheric Dispersion Ranking

	Atmospheric Dispersion	**Positive Pressure Containment**
AD1	Plume Rise	primary
AD3	Downwash	secondary
AD4	Modification of Effective Release Height Due to Building Entrainment	primary
AD5	Plume Broadening Due to Building Entrainment	primary
AD6	Fumigation	primary
AD7	Height of the Thermal Internal Boundary Layer	secondary
AD8	Reflection at an Elevated Inversion	secondary
AD9	Plume Transport	primary
AD10	Plume Diffusion	primary
AD11	Wet Deposition	primary
AD12	Dry Deposition	primary
AD13	Plume Depletion	Secondary
AD14	Exposure to Cloudshine	primary
AD15	Exposure to Groundshine	primary
AD16	Internal Exposure due to Inhalation	primary

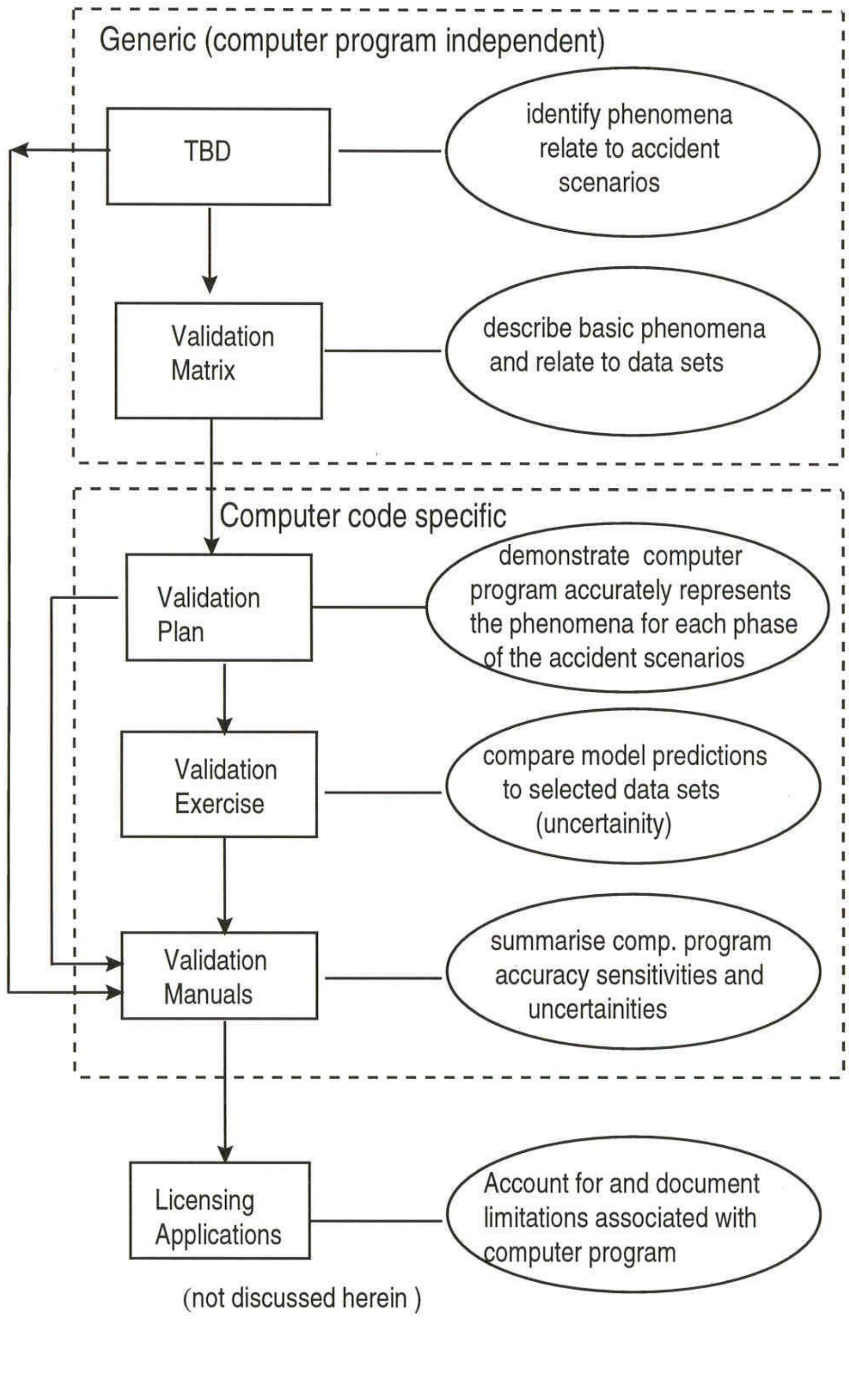

Figure 1 Relationship Between Documents

Appendix A
Atmospheric Dispersion

A.1 Introduction

Safety analyses of a nuclear power plant are undertaken to show that the consequences of any hypothetical event or accident are acceptable. The consequences are traditionally expressed as radiation doses to a member of the public or radiation doses to a population, and are assessed against regulatory dose limits. The radiation doses are calculated in the final step of safety analysis through simulations that model the transport of radioactive material released from containment and the resultant doses to the public.

Following release from containment, the transport of radioactive contaminants through the atmosphere is dictated by the existing atmospheric conditions. These atmospheric conditions are unaffected by physical conditions inside containment, that is, the air flow-fields outside containment are independent of the accident. However atmospheric dispersion is affected by the nature of release (i.e., puff or continuous release) and by the location of radioactivity release (i.e., through stack or MSSV's etc.).

A.2 Safety Concerns

The safety concern with the atmospheric dispersion of radioactive contaminants arises from the radiation doses to members of the public due to an airborne plume of radioactive material and from any layer of radioactive material deposited on the ground from that plume. Normally, airborne radioactivity released from containment only poses a hazard to the public when it is transported beyond the exclusion area boundary of the site. However, the threat to workers inside the exclusion area boundary may also need to be considered in some circumstances.

Radiation doses are due to external exposures from immersion in the radioactive plume and through the inhalation of airborne radioactive contaminants in the plume. In addition, depletion of the plume by wet and dry deposition creates a layer of surface contamination. That layer of surface contamination becomes a source of external exposure for an individual residing in the contaminated area.

The passage of a plume of radioactive contaminants can also contaminate food supplies either directly as deposits on vegetation or indirectly through uptake in the food chain. The ingestion of contaminated food will also lead to public exposure. However, no exposure from ingestion is considered in safety analyses because the interdiction of food sources has been credited.

A.3 Accident Behaviour

Several general conditions must be satisfied for a release of radioactive material from containment or the HTS following an accident:

1. Airborne radioactive contaminants must be present in the containment atmosphere;
2. A pathway must exist through the containment envelope; and
3. A driving force inside containment must exist to expel the contaminated air from containment.

However, not all items in this list need to be satisfied. For example, a containment by-pass event can arise in which an intended path through containment leads to a release of activity to the environment without the activity reaching containment. The classic example is a steam generator tube leak or tube rupture that allows activity from the primary heat transport system to reach the secondary side. Thus fission products could be released particularly in the presence of fuel defects. Opening of the Main Steam Safety Valves (MSSVs) or a secondary side break will release airborne activity that will undergo atmospheric dispersion and the ground level concentration of activity will expose members of the public to radiation.

Once the radioactive material has escaped from containment, the hazard from the plume of radioactive contaminants depends on:

- The characteristics of the release (dictated by the type of accident and the containment design);
- The meteorological conditions persisting at the time of the release and following the release (dictated by the site location);
- The characteristics of the receptor; and
- The characteristics of the site (dictated by the site location).

A.3.1 Characteristics of Release

The characteristics of the release refers to the quantity of radioactive material released, the specific radionuclides contained in the release, the chemical speciation *(FPT23)* of the radionuclides and details of the physical conditions associated with the release. These physical conditions include the physical height of release, the heat content of the release, the temperature of the release, and the duration and exit velocity of any release.

The hypothetical events postulated in safety analyses that involve fuel damage give rise to a range of release characteristics. These hypothetical events range from small to large loss of coolant accidents in which the safety systems are available or unavailable. This range of accident scenarios results in a variety of reactor physics, thermalhydraulic and fuel temperature transients. These different transients release different amounts of radioactivity into containment. The subsequent behaviour of the radioactivity released to containment will depend on the detailed design of containment. The fuel and subsequent containment behaviour would dictate the quantity, type and species of radionuclides that are released from the containment building as well as the initial thermalhydraulic conditions in the contaminant plume.

A.3.2 Containment Design Implications

In the ACR positive-pressure containment system, large diameter dampers close to stop the ventilation flow and heavy water vapour recovery system flows. Even though the large diameter ventilation dampers close promptly, airborne radioactivity release during the time taken for the dampers to close is considered. The release pathways terminate in the station stack or the ventilation system inlet. Thus these releases from containment can take place at very different elevations. Once the containment building is isolated airborne release of radioactivity to the environment is due to gross leakage from the containment building. Consequently, the release from a containment system can have a range of heat contents, temperatures, efflux velocities, pathways etc.

A.3.3 Meteorological Conditions

The dispersion of airborne radioactive material in the atmosphere is determined largely by the prevailing meteorological conditions during the release. These meteorological conditions are not phenomena in the context of validation matrices but their representation is an essential component of atmospheric dispersion modelling.

The simulation of meteorology in modelling atmospheric dispersion requires a representation of several continuously varying parameters, i.e., wind speed, wind direction, precipitation, atmospheric turbulence, inversions etc. The period of post-accident release, which may last up to several weeks or months, can conveniently be divided into four phases. This division allows the treatment of atmospheric dispersion to take some account of the variation in meteorological conditions. The phases are:

1. *Prompt phase* (duration up to two minutes); release takes on the three-dimensional characteristics of a "puff" release;
2. *Short-term phase* (duration of up to about one hour); release takes the form of a two-dimensional plume and meteorological parameters are generally considered to be constant;
3. *Intermediate* or *prolonged-term phase* (duration of one hour up to about twenty-four hours); some allowance for variation in meteorological parameters can be made; and
4. *Long-term phase* (duration greater than twenty-four hours); meteorological parameters can be assumed to approach the long-term average values.

The importance of each phenomenon may vary from phase to phase, and models and assumptions used to represent the phenomena may vary to reflect this.

A.3.4 Receptor Characteristics

The exposure of the public to radiation from airborne radioactivity is a function of the physical and radiological properties of the plume, the "footprint" of the plume after it has passed and the location of the receptor relative to the plume over the duration of the release and thereafter. It is conservatively assumed that the receptor is located on the plume centerline, at the start of the release period and remains at that location indefinitely.

Although it is acknowledged that countermeasures such as sheltering and evacuation could greatly reduce calculated doses, these countermeasures are not generally credited in safety analyses. However, longer-term countermeasures are usually credited to limit public exposure from ingestion pathways such that this contribution to exposures may be neglected.

A.3.5 Site Characteristics

The transport of the plume and the deposition of contaminants are sensitive to the surface roughness of the terrain between the source and receptor.

A.4 Governing Phenomena

The phenomena governing atmospheric dispersion and dose received can be grouped into four categories:

1. those governing the initial dispersion at the point of release (buoyant plume rise, momentum driven plume rise *(AD1)*, downwash *(AD3)*, building entrainment *(AD4)*, and plume broadening *(AD5)*);
2. those governing the distribution and depletion of radioactivity as it is transported downwind (plume advection, plume diffusion *(AD10)*, wet deposition *(AD11)*, dry deposition *(AD12)*, and plume depletion *(AD13)*);
3. those governing the intersection of a dispersing plume with an atmospheric boundary layer (fumigation *(AD6)*, marine inversion, and reflection at boundary layers *(AD8)*); and
4. those governing the degree of exposure of members of the public (cloudshine *(AD14)*, groundshine *(AD15)*, and internal exposure due to inhalation *(AD16)*).

For the initial dispersion, one of the relevant phenomena will usually dominate. Either the plume has sufficient momentum or buoyancy that it can escape the effects of the nearby buildings or it is drawn by the flow fields around the structures into the building wake. These phenomena influence the dose calculation at or close to the site boundary and within a few kilometres of the source.

Plume advection and diffusion *(AD10)* are governed by prevailing meteorological conditions and are the primary phenomena governing dispersion and radionuclide concentrations at, and beyond the exclusion area boundary. For elevated releases or at large distances where the plume can become very broad, boundary layer effects become important. Usually the vertical height of the plume is assumed to be limited by the presence of an atmospheric layer, reflecting any intersecting material towards the ground. A special case is that of fumigation *(AD6)*, which causes a plume, released above the boundary layer, to be quickly brought back to ground level when the plume intersects the boundary layer. Under these conditions fumigation may become a dominant phenomenon.

Cloudshine *(AD14)*, the external exposure from immersion in a cloud of contaminants, is likely to be the governing phenomenon for exposure if the release is made up largely of noble gases. Inhalation *(AD16)* becomes important if Iodine is present, and groundshine *(AD15)* from deposited activity is important if Iodine and Cesium are released. Phenomena related to dose uptake from deposited activity other than by groundshine (e.g., resuspension, food-chain pathways, waterborne releases) are omitted because safety analyses assume that in the longer-term these exposures are negligible or can be controlled by suitable counter-measures.