AECL EACL

Analysis Basis

Containment Safety Analysis Methodology

ACR-700

108-03500-AB-002
Revision 1

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ACRONYMS

AB	Analysis Basis
ACR™	Advanced CANDU Reactor™[1]
AECL	Atomic Energy of Canada Limited
ASDV	Atmospheric Steam Discharge Valve
CANDU®[2]	CANada Deuterium Uranium
CIS	Containment Isolation System
CNSC	Canadian Nuclear Safety Commission
CT	Calandria Tube
DBA	Design Basis Accident
DDT	Deflagration-to-Detonation Transition
ECC	Emergency Core Cooling
ECCS	Emergency Core Cooling System
ECI	Emergency Coolant Injection
FM	Fuelling Machine
HTS	Heat Transport System
LAC	Local Air Cooler
LCDA	Limited Core Damage Accident
LOCA	Loss Of Coolant Accident
LOCIVP	Loss of Class IV Power
LTC	Long Term Cooling
MAPS	Minimum Allowable Performance Standards
MSLB	Main Steam Line Break
MSSV	Main Steam Safety Valve
P&IC	Pressure and Inventory Control
PT	Pressure Tube
R/B	Reactor Building
RCPB	Reactor Coolant Pressure Boundary
RCW	Recirculated Cooling Water
RWT	Reserve Water Tank
SCDA	Severe Core Damage Accident
SG	Steam Generator
SDG	Standby Diesel Generator

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[1] ACR™ (Advanced CANDU Reactor™) is a trademark of Atomic Energy of Canada Limited (AECL).

[2] CANDU® (CANada Deuterium Uranium) is a registered trademark of Atomic Energy of Canada Limited (AECL).

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1. INTRODUCTION

This analysis basis (AB) document is part of a set of documents that:

- Defines the classes of events;
- Categorizes the postulated events according to the established classes;
- Defines the acceptance criteria and performance targets for each class of events;
- Defines the overall safety analysis objective;
- Describes the analysis tools and methodologies that will be used to demonstrate how the safety analysis objectives, which include acceptance criteria and performance targets, will be met for the events in each particular class; and
- Reports the results of the analyses.

At the top of the hierarchy of reports is the Safety Basis for ACR (Reference [1]). This document sets the bases for safety analysis in terms of classification of events, acceptance criteria, performance targets, and basic analysis methodologies for each class of events, and justifies the proposed safety analysis approach with respect to both Canadian and relevant international safety requirements.

An additional supporting document within this hierarchy is the Initial Conditions and Standard Assumptions Safety Analysis Basis report (Reference [2]). This document outlines the major plant system assumptions that are to be used when performing the safety analysis. The assumptions pertain to the operating state of the reactor before a postulated event and to the plant response after the event, but are not necessarily specific to any particular analysed event. The purpose of this document is to ensure a consistent, well-supported approach to modelling the plant response to a postulated accident when performing design or safety analysis work.

Within the set of documents required to complete the safety analysis, the analysis basis is the penultimate document. This analysis basis is one of several documents that describes the acceptance criteria, system models, computer codes, and methodologies that will be used in the ACR-700 safety analyses to determine thermalhydraulics and radionuclide behavior within containment after a postulated accident.

For the events to be analyzed, the overall objectives of the containment safety analysis are to determine the following:

a) Pressure and temperature transients within containment to demonstrate that the maximum pressure for LOCA is below the containment design pressure and that the structural integrity of the reactor building is not impaired for main steam line break (MSLB) (Section 3.4 of Reference [3]).

   Note: For U.S. licensing, the maximum pressure in MSLB with an additional 10% allowance shall be within the containment design pressure.

b) The maximum differential pressure across the reactor building internal walls to assure the functioning of all parts of the containment system required to mitigate accident consequences (Section 3.6.1 of Reference [3]);

c) Timing of pressure-dependent signals such as the high-reactor building pressure signals for containment isolation* or reactor trip;

d) Hydrogen concentration and distribution inside containment to verify that the concentration of hydrogen is low enough to avoid deflagration or detonation and consequential damage to containment (Section 3.10.2 of Reference [3]); and

e) Radionuclide behavior inside containment and releases to the environment for input to the dose calculation to demonstrate that reference dose limits are not exceeded (Section 3.3 of Reference [3]).

* For events that result in containment isolation on high activity, a separate calculation to determine the timing will be performed

2. ACCEPTANCE CRITERIA

Containment analyses for postulated accidents are performed:

1. To demonstrate that containment integrity is maintained following an accident; and
2. In support of the overall consequence analysis to demonstrate that the reference public dose limits are met.

The requirements are derived from the following Canadian Nuclear Safety Commission (CNSC) documents:

- "Requirements for Containment Systems for CANDU Nuclear Power Plants", Regulatory Document R-7 (Reference [3]) and
- "Requirements for the Safety Analysis of CANDU Nuclear Power Plants", CNSC Consultative Document C–6, Rev. 1 (Reference [4]).

The objective is to clearly interpret the performance and safety requirements imposed by the above documents. The key to the safety design and analysis framework is the definition and classification of events in and beyond the design basis of the plant. All licensing approaches are based on the same risk-informed objective, that is, the most probable occurrences should yield the least radiological consequences, and situations having the potential for the greatest consequences should be least likely to occur. The CNSC Consultative document C-6 Revision 1 addresses this objective by providing a system of classification of events into five classes.

Compliance with the regulatory documents is achieved by meeting the intent of C-6 Revision 1 in the full respect of a risk-informed safety design and analysis framework. This consists of:

- Adoption of five classes of events with associated radiological dose limits;
- Following the basic interpretations of the C-6 Revision 1 Companion document (Reference [5]) for classification and treatment of rare events;
- Adoption of acceptance criteria and targets that are based on safety margins increasing with the increasing likelihood of the events; and
- Using assumptions and methods that provide a good balance between the need to be conservative at the higher-event-likelihood end of the classification, and the reasonable use of a design centred assessment at the lower-event-likelihood end.

In line with the above compliance, ACR considers three categories of events:

- Design basis events;
- Limited core damage events; and
- Severe core damage accidents.

Design basis events fall into classes 1, 2 and 3. The limited core damage category falls into classes 4 and 5. The severe core damage accident category will not be considered here, but will be treated in the level 2 PSA. Design basis events (initiating events) are events that must be accommodated by the plant design within specified limits of radiological dose to the public and of the key barriers (i.e. fuel, reactor coolant pressure boundary and containment) to the release of radioactivity to the environment. The plant response to design basis events is analyzed using conservative assumptions and detailed models.

The limited core damage events are of a lower probability than design basis events, and are: i) those high temperature accidents which are arrested at the channel boundary; and ii) severe single channel events which possibly result in small quantities of molten zircaloy being released into the moderator.

The safety analyses for the limited core damage events will use design-centred assumptions.

2.1 Containment Integrity for Design Basis Events and Limited Core Damage Accidents

For the demonstration of ACR-700 containment integrity the specific acceptance criteria for design basis events and limited core damage accidents are:

a) The maximum pressure in containment following a LOCA, including LOCA with LOECC shall be less than the containment design pressure;

b) The maximum pressure in containment following a main steam line break (MSLB) shall be low enough to ensure containment structural integrity;

   Note: For U.S. licensing, the maximum pressure in MSLB with an additional 10% allowance shall be within the containment design pressure.

c) The maximum differential pressure between containment floors and walls shall be low enough to ensure the integrity of systems which mitigate the accident or whose failure could exacerbate it;

d) The room-average concentrations of combustible hydrogen for design basis events shall remain below the flammability limit of 4.0% by volume. The performance target for limited core damage events is to maintain hydrogen concentrations below 9.0% by volume, i.e., deflagration-detonation-transition (DDT) is precluded.

For the design basis accidents, these criteria are to be met assuming a single component failure in a mitigating system.

For the external events such as earthquakes, the event identification methodology is addressed in the Systematic Plant Review (Reference [6]). The acceptance criteria listed above will apply to these events as well.

2.1.1 Containment Integrity for scenarios with loss of Class IV Electrical power

The classification of design basis events with loss of Class IV power is specified in Table 1 (Reference [1]). For loss of Class IV power, the following performance targets are to be met:

a) The maximum pressure in containment following a LOCA with loss of Class IV power shall be less than the containment design pressure;

b) The maximum pressure in containment following a MSLB with loss of Class IV power shall be low enough to ensure containment structural integrity;

   Note: For U.S. licensing, the maximum pressure with an additional 10% allowance shall be within the containment design pressure.

c) The maximum differential pressure between containment floors and walls shall be low enough to ensure the integrity of systems which mitigate the accident or whose failure could exacerbate it.

2.2 Containment Analysis for Input to Public Dose Analysis

The acceptance criteria for public dose vary depending on the event classification, and are given in Reference [1]. The analysis of radionuclide release from containment is input to the dose analysis. Dose analysis methodology is described in an Analysis Basis document for atmospheric dispersion and public dose.

For the analysis of radionuclide releases from containment for design basis accidents, a coincident single component failure in a mitigating system is considered. Some containment system single failures may increase the maximum pressure and temperature in containment (for example, single failures in the energy suppression system - LACs) and/or the leakage rate from containment (for example, single failure in the containment isolation system). The dose analysis shall demonstrate compliance with the reference dose limit for the accident as specified in Table 1.

The dose analysis for scenarios with loss of Class IV power shall demonstrate compliance with the reference dose limit corresponding to the classification of these events specified in Table 2 of Reference [1].

3. SYSTEMS, MODELS AND EVENT SEQUENCES

3.1 System Availability and Equipment Status

3.1.1 Containment Subsystem Categorization

Figure 1 shows the ACR-700 containment and some of the components. The containment subsystems may be divided into the following three categories:

a) The containment envelope that comprises the reactor building and extensions. Extensions of the containment envelope include the following:
   1) Open subsystems such as the ventilation subsystem (which closes on containment isolation);
   2) Airlocks (which are closed during operation);
   3) Closed subsystems (such as the recirculated cooling water system); and
   4) Sealed penetrations (such as electrical penetrations).
b) Energy suppression subsystems, primarily the reactor building local air coolers.
c) The atmosphere control subsystems, which include the ventilation subsystem, internal structures, internal barriers such as doors and panels, hydrogen recombiners and the heavy water vapour recovery subsystem.

Most of these subsystems perform a function during normal operation of the plant as well as controlling the releases of radionuclides following an accident. It is useful to differentiate between the various subsystems by dividing them into “passive” and “active” subsystems.

Passive subsystems are defined in this document as those systems whose functioning does not depend on an external input such as actuation, mechanical movement or supply of power. Passive subsystems include:

a) The containment structure (walls and internal structures), including water stored in RWT;
b) The closed and sealed extensions of the containment envelope;
c) The airlocks; and
d) The hydrogen recombiners.

Active subsystems are defined as those systems functioning depends on an external input such as actuation, mechanical movement or supply of powers. Active subsystems include:

a) The containment isolation system (CIS), which initiates closure of all open containment penetrations; and
b) The reactor building local air coolers (LACs).

The safety analysis will consider a single component failure in a mitigating system. Application of the single failure criterion includes consideration of a single active failure in the short and long term, and a single passive failure in the long term (typically after 24 hours from the initiating event).

3.1.2 Conservative Assumptions for Equipment Status and Performance

In general, the assumed status of containment equipment and components for a given scenario depends upon the specific objectives of the analysis.

A common set of conservative assumptions is used for scenarios in which the analysis objectives are the assessment of:

a) Maximum pressures inside containment;

b) Differential pressures across the internal walls; and

c) Hydrogen concentration and distribution;

The conservative assumptions are that containment mass sinks and heat sinks are underestimated, while mass sources and heat sources are overestimated. In addition, for objectives (b) and (c), assumptions are chosen to reduce mixing within containment(e.g., flow resistance terms and burst pressures of doors between rooms may be overestimated). These assumptions allow local pressure and/or hydrogen concentrations to build up. Detailed descriptions of the assumptions used in containment analysis for these three analysis objectives are given in Section 5.1.1. The ACR generic design includes two independent, redundant means of isolating the containment ventilation lines; therefore the failure of ventilation isolation system is not a concern in the accident analysis. For under-pressure containment analysis, the assumptions would be to maximize heat sinks, minimize long-term steaming etc.

For scenarios in which the analysis objective is to demonstrate timely initiation of pressure-dependent signals, mass and heat sources are underestimated, and mass and heat sinks are overestimated. This choice of assumptions results in delaying the time of the signal and increasing the size of the break that would be needed to bring in the signal. This is conservative from the point of view of trip effectiveness, radionuclide release and ECC effectiveness. Detailed descriptions of the assumptions used in containment analysis for these three analysis objectives are given in Section 5.1.2.

For scenarios or analysis objectives where early initiation of the signal is unfavourable, the assumptions will be reversed. An example of this is that early containment isolation would result in higher maximum pressure because less air/steam will be able to escape from containment through the open ventilation system before the dampers close.

For the assessment of radionuclide releases, the most limiting assumptions on heat sources and heat sinks may vary according to the size of the break, the magnitude and timing of radionuclide release or the rate of containment leakage. Detailed descriptions of the assumptions used in containment analysis for these three analysis objectives are given in Section 5.1.3.

Conservatism is applied in the specification of the performance for each system or component modelled. Assumptions are based upon two considerations: firstly, an allowance in the values of the design parameters to account for performance uncertainties (such as measurement uncertainties), and secondly, a margin to provide adequate operational flexibility (that is, deviations from design-centre values during operation). Assumptions made in the containment analysis define the minimum allowable performance standards (MAPS) for the various containment components or systems.

The parameters affected by the above requirement for conservatism are:

a) Initial conditions, (ventilation flow, containment heat loads, initial pressure and temperature);
b) Standard assumptions (containment leakage rate, mass of water in RWT, and instrument air); and
c) Equipment credit (state of doors, capacity and availability of air coolers and fans).

3.2 Event Sequences

During normal operation, the containment atmosphere is in a thermodynamic equilibrium state. This equilibrium state is maintained by a balance of the mass and energy added by the reactor, pumps, motors, leakage and instrument air with the mass and energy removed by the air coolers, heavy water vapour recovery system and the ventilation system. A break in the primary or secondary heat transport system inside containment upsets this equilibrium.

The breaks postulated encompass a broad spectrum of discharge rates ranging from very small breaks (of the order of a few kg/s) to very large breaks involving initial discharge rates as high as thousands of kg/s. A range of breaks that discharge high enthalpy steam or two-phase coolant into containment cause containment pressure and temperature to rise above normal operating values. If the break is large enough to increase the containment pressure above the reactor building high-pressure setpoint or if radioactivity is detected in the ventilation exhaust lines, the ventilation dampers and other isolation dampers and valves automatically close. Also, the reactor building high-pressure reactor trip signal and the conditioning signal for emergency core cooling injection are generated.

The heat sinks within containment, such as the air coolers, water stored in the reserve water tank (RWT), perimeter walls, and internal walls and structures absorb some of the heat transferred from the reactor core to the containment atmosphere and thus help control pressure buildup within containment. Eventually the heat input to containment may fall below the capacity of the heat sinks and the air temperature within containment will begin to drop. If the air temperature falls below the wall temperature only the air coolers will remain as effective heat sinks. For scenarios that result in emergency core coolant (ECC) injection, the recovery phase of ECC injection provides another method of heat removal from containment.

Breaks in the primary circuit may result in fuel damage and the release of radionuclides into the coolant. These radionuclides are transported through the primary circuit where they may chemically react with other species, or deposit on piping surfaces. Some of the radionuclides are carried through the break into the reactor building.

Iodines released into containment are highly soluble and are therefore assumed to be released into containment as iodide dissolved in the liquid phase. However, iodines are also reactive. That is, organic iodide (methyl 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). In general, the degree of iodine volatility is a function of the acidity (pH) of the water, the surface material in contact with the water and the presence (or absence) of radiation.

The noble gases (krypton and xenon) do not show significant attenuation. That is, noble gases are not easily removed from the containment atmosphere by any removal process (other than decay). Therefore, noble gases are assumed to be released directly to the containment atmosphere and to remain in the containment atmosphere without removal (other than decay). Also, noble gases are assumed to be released from the containment envelope, without attenuation, along with any air or steam discharged (including leakage).

Liquid aerosols are assumed to be created from the break discharge spray and are assumed to carry dissolved radionuclides released into the primary circuit fluid. Aerosol removal mechanisms are credited. Aerosols cannot leak through the containment wall; however, they are assumed to leak without attenuation through open pathways leading out of containment, such as open inlet and outlet ventilation ducting before the containment isolation system closes the dampers.

Detailed event sequences for the analysis scenarios are given below.

3.2.1 Containment Event Sequence for Large LOCA

Large LOCA is a Class 3 event (design basis accident). The initiating event is a large break in a heat transport system header pipe. The sequence of events relevant to the containment analysis is as follows.

a) Temperature and pressure in containment increase rapidly. The reactor trips on a process signal, either low pressure or low flow.

b) The reactor building high-pressure signals for reactor trip, containment isolation and emergency core cooling conditioning, are reached. If not already tripped, the high-pressure trip signal will trip the reactor. The normally-open containment isolation valves and dampers close on the isolation signal.

c) A LOCA signal occurs on the ECC initiation signal (low pressure in the headers) conditioned on reactor building high-pressure.

   The LOCA signal initiates the following events:

   - MSSVs are opened to cool and depressurize the primary circuit (crash cooldown) to allow emergency coolant injection;
   - The emergency coolant injection sequence is initiated;
   - Required number of safety LACs are put in operation;
   - The isolation valves in the reserve water system open to allow water to flow from the reserve water tank to the sump;
   - Class III standby diesel generators are started; and
   - Two of the long term cooling (LTC) pumps are started in recirculation mode.

d) If the reactor was operating with defective fuel elements, the coolant may contain some fission products released from the defective elements. These radionuclides are released to containment with the break discharge. Until containment isolation is complete (a few seconds after the containment isolation signal is received) the fission products may be released with the steam and air through the openings in containment.

e) For some sizes of break, some of the high-powered fuel elements in the reactor may fail, likely because of excessive sheath strain. Some fission products are released from the failed fuel into the heat transport system coolant and eventually to containment through the break.

f) If the reactor was operating with a leaking steam generator tube and defective fuel elements, then the secondary side water may contain some radionuclides. These radionuclides are released to the atmosphere through the MSSVs during crash cooldown. The maximum allowable leakage from the primary circuit to the secondary side is 20 kg/hr.

g) In the long-term, the reactor building pressure decreases due to heat removal by the LOCA-qualified air coolers.

3.2.2 Containment Event Sequence for Large LOCA and Loss of Class IV Power

Large LOCA with loss of Class IV power is a Class 3 event. A large LOCA causes reactor trip, which in turn results in turbine run-back. There is a small probability that this disruption to the grid will result in a loss of off site power. In the analysis, Class IV power is assumed to be lost when the turbine begins to unload (95% of nominal steam pressure). This is typically a few seconds after the reactor trip.

The event sequence for containment analysis is largely the same as for the large LOCA with normal power available (Section 3.2.1), with mainly the following exceptions.

a) For a given break size and location, the timing of containment signals is slightly different compared with the case with Class IV power available. This is because the heat transport pumps run down when Class IV power is lost, resulting in a different discharge rate and discharge enthalpy to containment.

b) When Class IV power is lost, the standby diesel generators are signalled to start. This occurs before the LOCA signal, which would also signal the SDGs to start.

c) The electrical load sequencer establishes the electrical connections within a few minutes, including some of the local air coolers.

d) The number of fuel element failures and therefore the fission product releases to containment may be different from the case with Class IV power available, because of the temporary absence of forced circulation by the heat transport pumps.

3.2.3 Containment Event Sequence for Large LOCA and Loss of ECC

Large LOCA with loss of ECC is a Class 5 event (limited core damage accident). In this scenario, the emergency coolant injection function of the ECC system is assumed to fail. However, the makeup from the reserve water tank (RWT) is available. Steam generator crash cooldown is also available because there are redundant independent sets of triplicated signals for this function.

The following event sequence describes the differences from the large LOCA sequence (Section 3.2.1):

a) The coolant inventory depletes without makeup. As the inventory decreases, the heat transport pumps become less and less effective until they trip automatically. Thus, flow decreases and fuel and pressure tube temperatures increase.

b) Fission products may start to be released from the fuel matrix and some may also be transported into containment.

c) If there is a sufficient supply of steam to the fuel channels, fuel sheath temperatures will get high enough for oxidation of Zircaloy components. Oxidation of the Zircaloy in the channel can produce additional heat and hydrogen. Steam and hydrogen escapes to the containment atmosphere through the break.

d) Hydrogen that may be released to the containment atmosphere is controlled by a combination of engineered mixing and recombination. Mixing is provided by the air cooler fans that are environmentally qualified and on Class III power. Standby coolers start automatically on a high reactor building pressure signal. Mixing is also enhanced by the containment geometry, which has been designed to maximize natural circulation. Hydrogen recombiners at various locations in the reactor building remove hydrogen from the containment atmosphere. This recombination is an exothermic reaction, adding heat into containment. As a result of all these provisions, localized pockets of hydrogen are unlikely to collect within containment.

3.2.4 Containment Event Sequence for Small LOCA

Loss of coolant accidents (LOCAs) are classified in terms of their break size. Small breaks are defined as breaks up to the size of the largest feeder, which roughly corresponds to 2.5% of the reactor inlet header break. Small LOCA is a Class 2 event (design basis event), except for a feeder break of the precise size to cause single channel flow stagnation, which is classified as Class 5 (limited core damage accident).

In-core ruptures, such as a pressure tube/calandria tube rupture (Class 3 event), are also small LOCAs. The containment event sequence is given in Section 3.2.5 for in-core LOCAs.

It is postulated that a small break occurs in a large diameter pipe of the heat transport system (HTS) or that a rupture occurs in a feeder pipe or pressure tube. As a result, coolant is discharged into containment. The containment event sequence for a small break LOCA is as follows:

a) The pressure and temperature of the containment atmosphere rise. Heat is absorbed at equipment/wall surfaces and by air coolers through condensation and sensible heat transfer.

b) The reactor trips on one or more of the process trips on either or both of the two shutdown systems.

c) Containment isolation is automatically initiated on a high containment pressure signal if the break discharge rate is large enough. The normally-open containment isolation valves and dampers close on the isolation signal.

   If the event is a single channel event such as a stagnation feeder break, the release of coolant will also be accompanied by significant fission product release. Hence, containment isolation may also be automatically initiated on a high activity signal.

   Otherwise, if automatic containment isolation does not occur, manual isolation can be performed if required.

   The high reactor building pressure signal also conditions the ECCS and steam generator crash cooldown signals.

d) A LOCA signal occurs on the ECC initiation signal (low pressure in the headers) conditioned on reactor building high-pressure or sustained low ROH pressure.

   The LOCA signal initiates the following events:

   - MSSVs are opened to cool and depressurize the heat transport system (crash cooldown)to allow emergency coolant injection;
   - The emergency core coolant injection sequence is initiated;
   - Required number of safety LACs are put in operation;
   - The isolation valves in the reserve water system open to allow water to flow from the reserve water tank to the sump;
   - Class III standby diesel generators are started; and
   - Two of the long term cooling (LTC) pumps are started in recirculation mode.

   For the very small breaks, ECCS injection and steam generator crash cooldown are conditioned by sustained low HTS pressure because the reactor building high-pressure conditioning setpoint is not reached.

e) If the reactor was operating with defective fuel elements, the coolant may contain some fission products released from the defective elements. These radionuclides are released to containment with the break discharge.

   Until containment isolation is complete (a few seconds after the containment isolation signal is received) the fission products may be released with the steam and air through the openings in containment.

f) Fuel failures will not occur for small LOCAs, with the exception of a very small range of break sizes in an inlet feeder (stagnation feeder breaks). For stagnation feeder breaks, some of the high-powered fuel elements in the affected channel may fail, likely because of excessive strain or oxidation. Some fission products are released from the failed fuel into the heat transport system coolant and eventually to containment through the break.

g) If the reactor was operating with a leaking steam generator tube and defective fuel elements, then the secondary side water may contain some radionuclides. These radionuclides are released to the atmosphere through the MSSVs during crash cooldown. The maximum allowable leakage from primary to secondary side is 20 kg/hr.

h) Long-term cooling of the fuel is provided by the long term cooling (LTC) system. The air coolers provide long-term containment cooling.

3.2.5 Containment Event Sequence for In-Core LOCA

For in-core LOCAs, the event sequence in containment is different from the small LOCA in a reactor header described above because the discharge is into the moderator rather than directly to containment. The initiating event is assumed to be a pressure tube/calandria tube rupture. The following event sequence applies for in-core LOCAs.

a) The coolant discharge is initially absorbed and cooled by the moderator unless it is at a high location in the core. Negative reactivity due to light/heavy water mixing in the moderator, and coolant voiding, reduces reactivity.

b) The moderator level rises to the high moderator level ECC conditioning signal setpoint.

c) The reactor trips on one of the process trips on either or both of the two shutdown systems.

d) Continued discharge into the moderator bursts the rupture disks at the top of the calandria relief pipes, spilling liquid into the reactor vault. The pressure and temperature of the containment atmosphere rise. Heat is absorbed at equipment/wall surfaces and by air coolers through condensation and sensible heat transfer.

e) Containment isolation is automatically initiated on a high containment pressure signal if the break discharge rate is large enough. The normally-open containment isolation valves and dampers close on the isolation signal. Containment isolation may also be automatically initiated on a high activity signal.

   Otherwise, if automatic containment isolation does not occur, manual isolation can be performed if required.

f) A LOCA signal occurs on the ECC initiation signal (low pressure in the headers) conditioned on high moderator level.

   The LOCA signal initiates the following events:

   - MSSVs are opened to cool and depressurize the heat transport system (crash cooldown) to allow emergency coolant injection;
   - The emergency coolant injection sequence is initiated;
   - Required number of safety LACs are put in operation;
   - The isolation valve in the reserve water system opens to allow water to flow from the reserve water tank to the sump;
   - Class III standby diesel generators are started; and
   - Two of the long term cooling (LTC) pumps are started in recirculation mode.

g) If the reactor was operating with defective fuel elements, the coolant may contain some fission products released from the failed elements. These radionuclides are released to containment with the break discharge.

   Coolant is mixed with heavy water moderator before escaping to containment through the moderator relief ducts. In addition to the radionuclides present in the coolant and diluted in the moderator, tritium from the moderator is released into containment.

   Until containment isolation is complete the fission products and tritium may be released with the steam and air through the openings in containment.

h) Fuel failures will not occur by overheating, but mechanical damage to some fuel elements may occur after the calandria tube rupture. Some fission products are released from the failed fuel into the moderator and eventually to containment through the moderator rupture discs.

i) If the reactor was operating with a leaking steam generator tube and defective fuel elements, then the secondary side water may contain some radionuclides. These radionuclides are released to the atmosphere through the MSSVs during crash cooldown. The maximum allowable leakage from primary to secondary side is 20 kg/hr.

j) Long-term cooling of the fuel is provided by the LTC system. The air coolers provide long-term containment cooling.

3.2.6 Containment Event Sequence for Small LOCA and Loss of Class IV Power

Small LOCA with loss of Class IV power is a Class 3 event. In the analysis, Class IV power is assumed to be lost after reactor trip when the turbine begins to unload (95% of nominal steam pressure).

The event sequence for containment is largely the same as for the small LOCA with normal power available (Section 3.2.4), with mainly the following exceptions.

a) For a given break size and location, the timing of containment signals is slightly different compared with the case with Class IV power available. This is because the heat transport pumps run down when Class IV power is lost, resulting in a different discharge rate and discharge enthalpy to containment.
b) When Class IV power is lost, the standby diesel generators are signalled to start.
c) Electrical load sequencer establishes the electrical connections within a few minutes, including some of the local air coolers.
d) Fuel element failures would not be expected to occur before the Class IV power is lost and the heat transport pumps begin to coast down.

3.2.7 Containment Event Sequence for Small LOCA and Loss of ECC

Small LOCA with loss of ECC is a Class 5 event (limited core damage accident). In this scenario, the emergency coolant injection function of the ECC system is assumed to fail. The makeup from the reserve water tank is available. Steam generator crash cooldown is also available by a separate signal.

The following event sequence describes the differences from the small LOCA event sequence (Section 3.2.4):

a) The coolant inventory depletes without makeup. As the inventory decreases, the heat transport pumps become less and less effective until they trip automatically. Thus, flow decreases, and, fuel and pressure tube temperatures increase.
b) Fission products may start to be released from the fuel matrix. Some fuel elements may fail and some of the fission products may be released from the failed fuel elements and released into containment.
c) If there is a sufficient supply of steam to the fuel channels, fuel sheath temperatures will get high enough for oxidation of Zircaloy components. Oxidation of the Zircaloy in the channel can produce additional heat and hydrogen. Steam and hydrogen escapes to the containment atmosphere through the break.
d) Hydrogen that may be released to the containment atmosphere is controlled by a combination of engineered mixing and recombination. Mixing is provided by the air cooler fans that are environmentally qualified and on Class III power. Standby coolers start automatically on a high reactor building pressure signal. Mixing is also enhanced by the containment geometry, which has been designed to maximize natural circulation. Hydrogen recombiners at various locations in the reactor building remove hydrogen from the containment atmosphere. This recombination is an exothermic reaction, adding heat to containment. As a result of all these provisions, pockets of hydrogen are unlikely to collect.

3.2.8 Containment Event Sequence for Main Steam Line Break in Containment

A main steam line break in containment is a Class 3 event (design basis event).

a) A large break occurs in one of the steam lines inside the reactor building.

b) The discharge of steam and water is choked due to the flow limiters in the steam generator nozzles, resulting in a smaller discharge.

c) The discharge of steam and water through the break fills the RB with steam, causing the pressure and temperature to increase rapidly. Containment isolation occurs almost immediately on reactor building high-pressure signal.

d) Immediately following the failure of the steam main, the flow rate through the corresponding steam generator increases. The steam separator efficiency decreases and as a result, a two-phase mixture is discharged into containment. Eventually, the steam generators depressurize, and the flow rate through them decreases to within the capability of the steam separators, whereupon the discharge becomes single-phase steam.

e) The increase in flow through the steam generators results in overcooling of the HTS coolant. The coolant temperature decreases and density increases. The net effect is the insertion of a very small amount of positive reactivity. The reactor regulating system is expected to be able to control the reactor power near the operating setpoint; however RRS will be assumed frozen in the thermalhydraulic analysis allowing reactor power to increase until the reactor trip occurs. It is expected that reactor building high pressure is the first reactor trip parameter to be received, and other trip parameters that may follow include heat transport system low pressure, steam generator low level and pressurizer low level (not necessarily in this order). The high discharge flow during the initial blowdown may result in a false steam generator level reading. Following reactor trip the primary circuit cools further and depressurizes.

f) As the steam generators depressurize, the total discharge rate starts to decrease. This causes the rate of rise of the reactor building pressure to slow down as well. The internal concrete walls, reserve water tank (RWT), steel structures and steel grating inside the reactor building absorb heat from the containment atmosphere. The air coolers, if they are available, also remove heat, but the combined heat removal rate does not totally compensate for the heat input rate during the early stages of the transient.

g) HTS pressure may decrease to the ECC initiation signal setpoint, whereupon the LOCA signal is generated, conditioned on reactor building high pressure. If so, crash cooldown occurs, MSSVs will open and containment can depressurize through the broken steam line to the atmosphere through the MSSVs. Part of the steam will come back from the steam headers.

h) The steam generators continue to be supplied by feedwater, removing stored heat and decay heat from the heat transport system and discharging steam into the reactor building. As the internal walls and steel inside the reactor building heat up, they become less effective in absorbing heat. The air coolers are capable of removing sufficient heat to prevent a further increase in reactor building pressure.

i) If the primary circuit pressure falls low enough, the heat transport pumps may trip. However, since the primary pumps are not environmentally qualified, pump stoppage could occur before the trip signal is received. Core heat removal is maintained by thermosyphoning of the heat transport system or ECC/LTC.

3.2.9 Containment Event Sequence for Main Steam Line Break and Loss of Class IV Power

The sequence of events for the steam main failure with a loss of Class IV power is the same as the sequence for the same break with Class IV power available up until the time that Class IV power is lost. Only the differences caused by the loss of Class IV power are discussed.

a) The heat transport pumps, the main feedwater pumps, 50% of SG and dome air coolers and all small coolers are lost.

b) The auxiliary steam generator feedwater pump and the SG and dome air coolers on Class III power begin operation within 3 minutes when Class III is established and connected via the load sequencer.

c) Following the loss of Class IV power, thermosyphoning will transfer decay heat to the steam generators.

4. COMPUTER CODES

The simulation of containment performance and radionuclide behavior requires the use of three computer codes, namely, GOTHIC, SMART, and SGDOSE. The GOTHIC code is used to simulate the transient containment thermalhydraulic behavior with the input from the HTS full circuit simulations. The results from GOTHIC analysis and fuel analysis are used in SMART to simulate the radionuclide behavior within containment in support of the public dose analysis. The SGDOSE code is used to analyze releases due to chronic leakage from steam generator tubes. Details of the overall analysis plan are described in the following section.

4.1 Analysis Plan and Interfaces

A generic overall analysis plan for public dose analysis is given below. Containment performance and radionuclide release analysis is a part of this overall plan.

- Thermalhydraulic analysis of the HTS is performed with CATHENA MOD 3.5d/Rev. 0 to obtain the mass (both coolant and hydrogen) and energy releases to containment for each accident case. The results generated include the transit time required for the fission products to travel from the location of the failed fuel to the break in the HTS.
- SGDOSE MOD 2 analysis computes the maximum radionuclide burden in the secondary circuit at the time of the accident.
- Fuel analysis (ELESTRES Version M14B.3.1, ELOCA-IST V2.1, SOURCE-IST V2.0, and ORIGEN-S) to determine the fuel heatup, if any, and the release of radionuclides from the failed fuel into the HTS.
- Containment thermalhydraulic analysis (GOTHIC-IST V6.1bp2) to obtain the transient conditions inside containment during the accident.
- Radionuclide behavior simulation (SMART-IST VER-0.300) and prediction of the species, types and quantities of radionuclides released to the outside atmosphere.
- Dose and dispersion calculations (ADDAM-IST V1.0).

The flow chart associated with this analysis plan is shown in Figure 2.

4.2 GOTHIC

GOTHIC-IST V6.1bp2 (References [7] and [8]) is used to analyze the thermalhydraulic behavior of the ACR-700 containment under accident conditions. It has been applied in the past to analyze active and passive CANDU containment behavior under design basis accident conditions, including containment pressurization, equipment qualification and hydrogen mixing analyses. It has been validated for use in CANDU containment safety and licensing analysis.

Two containment models suitable for running with GOTHIC version 6.1bp2 will be used for ACR-700. The first model is a one-dimensional (lumped parameter) model, and the second model is a hybrid of lumped parameter and distributed volumes, which refines the idealization of key nodes/volumes using the three-dimensional capability of GOTHIC. The second model is used for GOTHIC applications where three-dimensional effects are important, for example, hydrogen distribution analysis.

A brief description of the primary models used by GOTHIC is given as follows.

a) Control Volumes

The GOTHIC computer code is a control volume formulation, therefore, the discretization of containment consists of a number of non-overlapping control volumes with flow paths specified between them. Each control volume has a defined volume, height, elevation and hydraulic diameter. The basic node and link structure are the same in both the lumped parameter model and distributed model. Only the key control volumes are discretized further into more cells for the analysis of three-dimensional effects in the distributed model.

b) Flow Paths

Flow paths are used to model openings connecting control volumes. Momentum equations are solved for each flow path with each phase having unequal velocities. Flow paths are defined by upstream and downstream volumes they connect, the elevation of their connections, flow area, hydraulic diameter, inertia length, friction length, and forward and reverse loss friction coefficients. In the lumped parameter model, several flow paths may be grouped together since the volume properties are assumed to be perfectly mixed throughout the entire area represented by the volume. In the distributed model, properties are calculated for each cell. Flow paths that are grouped for the lumped parameter model are separated and therefore the specific cell to which each flow path connects must be specified.

c) Thermal Conductors

Thermal conductor models are used in GOTHIC to represent solid structures within containment (i.e. walls, structural, etc.). This information is needed for heat transfer calculation. In the lumped parameter model, one or more walls are grouped into a single model. In the distributed volume model, each wall is modelled with a unique thermal conductor model.

Thermal conductors for modelling actual walls of the room are spanned over the surface of each respective wall. That is, the thermal conductor properties are spread out over the entire wall surface. Thermal conductor models representing large structures other than walls, for example large pieces of equipment and stairs, are modelled as points.

d) Components

1) Valve Models (Dampers)

Valve models are used to represent dampers. Dampers are part of the reactor building cooling system. Following an accident some dampers are signalled to close and the rest are signalled to open on a high reactor building pressure signal. Therefore, dampers are modelled with ‘quick open’ and ‘quick close’ valves, depending on their functions.

2) Volumetric Fans

In the ACR-700 design, four ducted fans are located in each side of the ducted air cooling system. In both the lumped parameter model and distributed model, they are modelled as constant volumetric flows in the flow paths. In the design, there are also four large unducted fans in the dome area. In the distributed model, the unducted fans in the dome area are included. However, these fans are not included in the lumped parameter model since the lumped parameter control volumes are inherently assumed to be perfectly mixed.

3) Air Cooler Models

   The GOTHIC cooler model is utilized to represent the large qualified air coolers. Air coolers are a significant long-term heat sink. Half of the vault and dome LACs and associated fans supplied by Class III power are assumed to be available and functioning.

   When analysis objectives require overestimation of their capability, minimizing RCW temperatures and increasing the number of units credited overestimate the heat sink capability of the air coolers.

4) Heater Models (Additional Heat Loads)

   Additional heat loads due to piping, lighting, motors, reactivity mechanisms, etc. are included in the GOTHIC model by specifying heater models in the appropriate cells (distributed model) or control volumes (lumped parameter model).

   During normal reactor operation, the air coolers operating at their normal capacity balance these additional heat loads. In the event of an accident that results in a reactor trip, many of these heat loads will diminish and some will become negligible.

5) Hydrogen Recombiners

   GOTHIC has a built-in recombiner model that converts a specified fraction of flowing hydrogen to steam. The reaction is limited stoichiometrically by the amount of oxygen available. The recombiner model must be located in a flow path. In addition to the flow path where each recombiner is located, the following information is required:

   i) Efficiency: This is the fraction of the flowing hydrogen that is converted to steam. The AECL recombiners are expected to remove 50% to 60% of the incoming hydrogen. This value is conservatively assumed to be 25% in the model.

   ii) Momentum Height: This is the effective height of the recombiner. The AECL recombiners are 0.81 m tall (including the hood). This value is used as the momentum height.

4.2.1 Input

The input information used by GOTHIC-IST V6.1bp2 computer code includes:

a) Thermalhydraulic discharges from a pipe break in either the primary or secondary coolant systems, as calculated in the thermalhydraulic analyses,
b) A discretized model of the containment volume,
c) Data to define the miscellaneous mass and heat source/sink submodels in GOTHIC (including H_2),
d) Initial conditions.

4.3 SMART

Radionuclides in containment can be in the form of gases, particulates or dissolved species in liquid aerosols and bulk aqueous solutions. They can be either mobile (in the containment atmosphere) or non-mobile (in the sump or on wall and equipment surfaces).

Simulation of the location and behavior of these radionuclides is performed using the computer code SMART-IST VER-0.300. The SMART code consists of models for radionuclide decay and

buildup; behavior of liquid aerosols; iodine behavior (molecular iodine and organic iodides); and convective transport between nodes (or rooms) and through release paths to outside atmosphere. It also keeps track in each node of the transient inventory of all radionuclide species, their form, and their location [airborne, in the sump pool, removed via outleakage (for each leak path), or retained in the leak path (leak path retention)].

A brief description of the primary models used by SMART follows.

a) Radionuclides Decay and Build-up

Of the more than 800 nuclides that could be produced, 49 isotopes of 20 elements are identified as the most potential for contributing public dose under conditions. Of these, the most significant species released from fuel are 23 isotopes of 7 elements: iodine (^{131}I, ^{132}I, ^{133}I, ^{134}I, ^{135}I), noble gases (^{87}Kr, ^{88}Kr, ^{89}Kr, ^{133}Xe, ^{133m}Xe, ^{135}Xe, ^{135m}Xe, ^{137}Xe, ^{138}Xe), cesium (^{134}Cs, ^{137}Cs, ^{138}Cs), tellurium (^{131m}Te, ^{132}Te), ruthenium (^{103}Ru, ^{106}Ru) and strontium (^{89}Sr, ^{90}Sr). Thus, for the calculation of dose to public, the radioactive decay behavior of the 23 isotopes and tritium (^{3}H) is assessed to determine the amount of release to the outside atmosphere. Tritium is one of activation products that may contribute to the radiation dose if tritiated heavy water is released from the moderator.

Radionuclides undergo natural radioactive decay, disintegrating into other nuclides and eventually to stable nuclides. The decay chains of the 24 radionuclides fall into four types (Figure 3), based on the rate equations required to describe the decay and build-up behavior of the radionuclides. The exact analytical solutions for the differential rate equations applied to the four types of decay are known. The amounts of the 24 nuclides are followed as a function of time using these solutions.

b) Fission Products Transport and Transformation

While undergoing the natural radioactive decay, the fission products are also transported, transformed and/or removed from the containment atmosphere. For simulating the transport and transformation of the nuclides in containment, the nuclides are grouped essentially into three categories: noble gases, iodine and the other fission products.

1) Noble Gases

The inert and relatively insoluble noble gases (xenon and krypton) are only subjected to convective flow and release to outside atmosphere through leak paths. No other transport processes or physical and chemical transformation (other than the natural radioactive decay) are applied to the simulation of the noble gas behavior in containment.

2) Iodine

Iodine is released from the HTS into the containment atmosphere mainly as highly soluble non-volatile species, and, hence, mainly as dissolved species in liquid aerosols or in bulk water phase. Only a small fraction would be released as gaseous species. However, exposed to highly oxidizing and radiation environment, non-volatile iodine species dissolved in liquid aerosols and bulk water could be slowly but continuously transformed to volatile species. Whether initially released from the HTS or later formed in containment, the volatile iodine species also have intermediate solubility in water and/or are easily adsorbed on various surfaces. Because of their ability to undergo chemical transformation and the different transport and removal mechanisms for different

iodine species, iodine in various forms (gases, liquid aerosols, dissolved in aqueous solutions and adsorbed on surfaces) must be followed separately.

3) Other Fission Products

The other fission products released from the HTS would be non-volatile under post-accident containment conditions. Furthermore, they are not considered to transform to volatile species in containment. Thus, they are airborne only as dissolved species in liquid aerosols, and not as gases, and the transport and the removal of these fission products from the containment atmosphere follow the behavior of liquid aerosols.

c) Aerosol Behavior

Aerosols are airborne liquid droplets containing soluble or particulate radionuclides. Liquid aerosols are generated from the disintegration of liquid water as it is discharged into containment from a break in the primary heat transport system. The aerosol is transported within containment by carrier gas. Within a particular node or a room (up to 45 nodes can be used in SMART-IST VER-0.300), the liquid aerosol droplets undergo coagulation and also deposit on walls and floors due to various natural phenomena (for example, gravitational settling). To model the coagulation and removal processes mathematically, the continuous aerosol size spectrum is divided into a finite number of intervals, and within each interval, the aerosol size is assumed to be constant. The aerosol droplet sizes considered for analysis range from 0.1 μm to 10000 μm radius and a total of up to 30 particle size classes are modelled in SMART-IST VER-0.300. The initial mass distribution of the fission products in the aerosol size classes are provided by upstream analysis, while the changes in the fission product masses in the size classes due to radioactive decay and build-up, coagulation and deposition in containment are calculated in SMART.

The physical law governing the transport of aerosols within containment is the conservation of aerosol mass. Processes that can cause a change in the airborne aerosol mass in a node are:

1) Source from a break or other discharge into the room
2) Removal from the break discharge
3) Coagulation or agglomeration
4) Removal due to aerosol deposition mechanisms (gravitational settling, diffusiophoresis, thermophoresis, and turbulent deposition)
5) Convective flow from a room to next and flow through leak-paths to the external atmosphere

Aerosol droplets are typically larger than the microcracks in the containment perimeter wall and are therefore assumed to be unable to leak through the containment wall. However, they are assumed to leak through open pathways leading out of containment without attenuation (such as a pre-existing hole).

d) Iodine Behavior

Radioiodine is the most important nuclide with potential for public dose in both design basis accidents and beyond-design-basis accidents. This is because of the combination of its large inventory in irradiated fuel, potential for formation of volatile chemical species under post-accident reactor conditions, and its hazardous radiological effects. Phenomena and processes

governing iodine behavior in containment are also very complex. Iodine behavior in containment is simulated using IMOD-2.0 (Iodine Module) in SMART-IST VER-0.300. IMOD-2.0 is a mechanistic model developed based on fundamental understanding coupled with validation tests. IMOD-2.0 is the largest component of SMART, and consists of the rate equations for the processes:

1) The interconversion between non-volatile iodine species (e.g., I^-, HOI, I_3^-, IO_3^-, I_xO_y, etc) and volatile molecular iodine (I_2),
2) The formation and destruction of volatile organic iodides (High volatility organic iodides, HVRI and Low volatility organic iodides, LVRI),
3) The partitioning of volatile species (e.g., I_2 and RI) between aqueous and gas phases,
4) The transport of iodine species to (adsorption) and from (desorption) surfaces both in the gas and aqueous phases, and
5) The transport of gaseous iodine species to condensing films and to the bulk aqueous phase by condensation flows.

4.3.1 Input

The input information used by the SMART-IST VER-0.300 computer code includes:

a) Transient thermohydraulic conditions inside containment as predicted by GOTHIC above,
b) A list of the radionuclide species that are to be modelled, their primary chemical and radiological characteristics and their cumulative transient release. Fission product and hydrogen releases are provided by the fuel analysis. Tritium releases are estimated using thermalhydraulic results.

4.4 SGDOSE

a) Transient Radionuclide Concentrations

   SGDOSE is a simple explicit numerical integration of the concentration equation for the radionuclides in the nodes representing the secondary circuit. The calculation is quasi-steady, as the mass flow rates in the circuit are assumed constant over a time step. The radionuclide concentration is determined by the following factors:

   1) Flow rate of the HTS coolant escaping into the secondary circuit through the leaking steam generator tube,
   2) Radionuclide concentration in the HTS coolant escaping into the secondary circuit through the leaking steam generator tube,
   3) Addition rate of fresh feedwater,
   4) Recirculation rate of feedwater (from the condenser),
   5) Rate of steam loss to atmosphere through the MSSVs,
   6) Rate of steam loss to atmosphere through the ASDVs,
   7) Rate of water loss to atmosphere through steam generator blowdown flow,
   8) Chronic out-leakage rate from the secondary circuit either as liquid or steam,
   9) Changes in the water inventory of the steam generators and feed train,

10) Attenuation factor of radionuclides with limited volatility (expressed as the concentration of radionuclide in the steam phase as a fraction of that in the liquid phase,
11) Release rate of volatile radionuclides (noble gases such as krypton and xenon).

b) Steady-State Radionuclide Concentrations

For the determination of the consequences of chronic steam generator tube leakage, secondary circuit conditions are assumed to be at steady-state. Thus, the mass flows can be taken as constant and the inventory changes of item 9 above, are zero. Furthermore, as only the final equilibrium radionuclide concentrations are of interest, a set of steady-state equations can be written and solved to provide explicit expressions for the radionuclide concentration in each node of the secondary circuit. (These equations are further conservatively simplified by neglecting chronic out-leakage and ASDVs/MSSVs discharges, items 5, 6, and 8, above.).

4.4.1 Input

The input information used by the SGDOSE MOD2 computer code includes:

a) The transient discharge rate through the leaking steam generator tubes into the affected steam generator, the transient steam flow to the turbines, the transient feedwater flow to the steam generators, and the transient steam flow through the main steam safety valves. This data will be taken from thermalhydraulic predictions provided by the computer code CATHENA.
b) The concentrations of radionuclide species from the normal holdup in the heat transport system as well as the radionuclide species from the fuel if fuel failures occur.

The initial concentrations of radionuclide are determined based on the assumption that a chronic steam generator tube leak has been present sufficiently long such that equilibrium concentrations exist in the secondary system.

5. ANALYSIS METHODOLOGY

Methodologies for the various analysis objectives are described below, in separate sections for the design basis accidents, the limited core damage accidents and the containment system single failure scenarios.

5.1.1 Methodology for Analyses of Maximum Pressure in Containment, Maximum Differential Pressure Across Internal Walls and Hydrogen Concentration

A common set of assumptions is used in the containment analyses to determine the maximum pressures inside containment, differential pressures across the internal walls, and hydrogen concentration and distribution. In general, mass sinks and heat sinks are underestimated, while the mass sources and heat sources are overestimated.

One exception to this common set of assumptions is that instrument air discharge rate is underestimated for the hydrogen concentration assessment, because air would dilute the hydrogen. For the other objectives, instrument air discharge rate into containment is overestimated because it contributes to increasing the pressure in containment.

A detailed description of the assumptions for the mass and energy sources and sinks is given below.

a) Containment Envelope

The acceptable leakage rate for the containment structure during testing is 0.2 % of building volume per day at the design pressure. However, for these analyses, leakage from containment is assumed to be zero. This assumption removes leakage as a mass and heat sink.

b) Reactor Building Walls and Structures

The walls and structures considered for accident analysis are

- the perimeter wall (including the steel liner);
- the internal concrete walls;
- internal steel structures such as columns and beams; and
- doors.

The mass and surface areas of these walls and structures are underestimated for heat transfer calculations, to minimize the heat sink. Only those steel structures that are not insulated are included as heat sinks. Maximum ambient temperature is used to further minimize the heat sink.

c) Reserve Water Tank

Water stored in the RWT serves as heat sink in accidents. The water tank capacity is underestimated for heat transfer calculations to reduce its capacity to store heat.

d) Local Air Coolers

The SG and dome local air coolers are qualified for operation under accident conditions. Some of these are ducted units, arranged in two separate loops. Each loop handles the heat load in one of the two fuelling machine rooms and one of the two steam generator

enclosures. The remaining air coolers are unducted units and are located in the accessible area. In addition, there is a large number of smaller local air cooling units distributed throughout the remaining containment rooms.

The air coolers qualified for accident operation are supplied by Class IV power, backed up by Class III power. The remaining smaller air coolers are supplied only by Class IV power. During normal operation, half of the accident-qualified coolers operate to maintain and control the temperature inside containment, while the remaining units are in standby mode. However, when a LOCA signal is generated, the air coolers on standby mode start automatically.

For these design basis analyses, a minimum number of the large containment air coolers and fans that are supplied by Class III power are assumed to be available and functioning (that is, either working or on standby mode). Only 50% of these coolers (one Division of Class III power) are credited for operation.

No credit is taken for the remaining small local air coolers distributed throughout containment.

All air coolers receive recirculated cooling water (RCW) for heat removal. The heat sink represented by the coolers is limited by assuming the maximum (RCW) temperature. This maximum RCW temperature is assumed to be 30°C (MAPS value, summer conditions) with an additional allowance to account for instrumentation uncertainty (Reference [9]).

e) Additional Heat Sources

Models for the additional heat sources inside containment (motors, lights, the reactor face and reactor piping) are developed on the basis of estimates performed for the CANDU 6 reactor design. During the transient, the heat loads are calculated based on the difference between a user-defined pipe temperature and the volume atmospheric temperature.

f) Atmospheric Control

All doors are assumed to be initially closed and unavailable for atmosphere control.

g) Containment Isolation System

A containment isolation signal is generated when containment pressure reaches the initiation signal setpoint of 3.45 kPa (g) (Reference [9]). An isolation signal is also generated if high activity level is detected in the ventilation system outlet ducting.

Margin is added to the above setpoint values to account for instrument uncertainties and to ensure analysis conservatism. Thus the reactor building high-pressure setpoint for containment isolation and the high activity level setpoint are conservatively adjusted.

In practice, there is a delay between the time when the isolation signal is produced and the isolation valves are fully closed. However, since open containment penetrations act as an energy and mass sink before and during isolation, containment isolation is assumed to occur instantly at the start of the postulated event. This assumption eliminates the containment mass and energy sinks, thereby maximizing the predicted containment pressure.

h) Instrument Air System

Following an accident, the addition of instrument air to containment adds to the pressure increase of the reactor building. An estimate of the instrument air discharge into containment is based on a general event sequence, design values and assumptions.

The instrument air system is manually isolated 3 hrs after an accident occurs. The instrument air discharge transient to containment is divided into three time periods: 0 to 10 s, 10 s to 3 h, and 3 h until the air tanks are empty.

i) Hydrogen Recombiners

Hydrogen recombiners are installed within containment as a measure for hydrogen control. These units combine hydrogen with the oxygen in containment in a slow, continuous manner. The transfer of the resulting heat to the containment heat sinks occurs by natural convection. If a hydrogen burn as a result of electric spark or self-ignition by sufficiently high temperature of the discharged gas mixture itself is predicted, it will be considered in ensuring the integrity of nearby safety related equipment.

j) List of the Limiting Scenarios (Table 2)

The following analyses are performed for the assessment of maximum pressures within containment and maximum differential pressures across internal containment walls:

1) Large LOCA,
2) Large LOCA plus Loss of Emergency Core Cooling,
3) Main Steam Line Break.

For hydrogen distribution in containment, the Large LOCA plus Loss of Emergency Core Cooling scenario is analysed because it is the only scenario that is expected to result in hydrogen release in the short term. For long term hydrogen analysis, an additional scenario may be considered.

5.1.2 Methodology for Analysis of the Timing of Pressure Dependent Signals

Reactor building high pressure is an initiating signal for containment isolation, reactor trip and ECC conditioning. Depending on the objective of the analysis, it may be conservative to either overestimate or to underestimate the time to reach the setpoint.

Analysis is performed to determine the maximum time to reach the containment isolation setpoint because this is limiting for the objective of predicting the transient radionuclide release from containment as input to the public dose assessment. The same set of assumptions would be used for determining the maximum times to reach the trip setpoint and the ECC conditioning signal setpoint.

For the maximum containment pressure assessment described in Section 5.1.1, it is conservative to credit the earliest time for containment isolation because openings in containment act as mass and heat sinks, and therefore reduce the rate of pressure increase after a LOCA. Rather than determining the earliest credible time of containment isolation, it is assumed that containment is isolated immediately in the analysis. Therefore no containment analysis is needed for determining the minimum time to reach the setpoint.

An additional objective is to determine the threshold break size that will bring in the reactor building high-pressure signals for reactor trip and containment isolation. This threshold break size is input to the trip coverage analysis for defining the small LOCA trip coverage map. The assumptions for this threshold break size analysis are the same as for the assessment of the maximum time to reach the setpoint, discussed below.

For the analysis of the maximum timing to reach the setpoint, the containment mass sinks and heat sinks are overestimated, while the mass sources ad heat sources are underestimated. A detailed description of the assumptions for the heat and mass sources and sinks is given below.

a) Containment Envelope

The maximum acceptable leakage rate for the containment structure during testing is 0.2% of building volume per day at the design pressure. However, for pressure-dependent signals analysis purposes, a leakage rate of 0.5% is assumed because it will minimize the pressure increase and thus maximize the time required to reach the pressure signal setpoint.

b) Reactor Building Walls and Structures

For pressure-dependent signals analysis, the mass and surface areas are overestimated, to maximize the heat sink and potentially reduce the rate of pressure increase. Minimum ambient temperature is used to further minimize the heat sink.

c) Reserve Water Tank

For pressure-dependent signals analysis, the reserve water tank capacity is overestimated to maximize the heat sink.

d) Air Coolers

A description of the total number of air coolers available is given in Section 5.1.1 for the maximum pressure, pressure differential and hydrogen distribution assessments. In the case of pressure-dependent signals analysis, the heat sink capability of the air coolers must be maximized. Thus, all air coolers are assumed to be available and functioning. Furthermore, the minimum recirculated cooling water (RCW) temperature is assumed, which overestimates the heat sink. This minimum RCW temperature is assumed to be 10°C (MAPS value, winter conditions) with an additional allowance to account for instrumentation uncertainty.

e) Additional Heat Sources

For pressure-dependent signals analysis purposes, the additional heat sources such as motors, lights, the reactor face and reactor piping, are conservatively underestimated.

f) Doors

For pressure-dependent signals analysis, doors are assumed to remain open and unavailable for atmosphere control. This assumption connects all areas of the reactor building and may reduce the predicted rates of pressure increase at the locations of the pressure switches.

g) Containment Isolation System

One of the objectives of the pressure-dependent signals analysis is to determine the time to reach the containment isolation setpoint. Once the timing has been determined, it is input to the radionuclide release assessment, described in Section 5.1.3.

Containment isolation valve and damper closure is initiated upon receipt of an isolation signal on either reactor building high-pressure or high activity in the ventilation lines. An operator may also manually initiate containment isolation. For the assessment of the timing of pressure-dependent signals, isolation on activity or operator action is ignored.

h) Instrument Air System

For pressure-dependent signals analysis, the mass and energy source terms represented by instrument air are ignored.

i) Hydrogen Recombiners

For pressure-dependent signals analysis for large LOCA, the hydrogen recombiners are immaterial since no significant hydrogen production is expected during the short term. Nevertheless, the recombiners are included in the model. For large LOCA/LOECC, the signals will occur long before hydrogen production, therefore the presence of recombiners in the model is not relevant.

Table 2 lists the cases considered for pressure dependent signals.

5.1.3 Methodology for Analysis of Radioactive Releases from Containment

For the assessment of radionuclide releases, the most limiting assumptions regarding heat and mass sources and sinks may vary according to the size of the break, the magnitude of the radionuclide release or the containment leakage rate. This section outlines the considerations involved and the most likely assumptions based on past experience.

a) Containment Envelope

The maximum leakage rate is assumed to be 0.5% percent per day at the design pressure in order to overestimate the radionuclide releases to the environment. This is conservative because the maximum acceptable leakage rate for the containment structure during testing is 0.2% of building volume per day at the target design pressure.

b) Reactor Building Walls and Structures

To overestimate radioactive releases, the most limiting assumption with respect to the surface areas of walls and structures is used. The most limiting assumption may depend on the containment pressure transient, or the extent of radionuclide release through the open containment penetrations prior to containment isolation. This, in turn, may depend on the magnitude of the break discharge rate and enthalpy, or on the magnitude of radionuclide release into containment and its distribution within containment.

For low-pressure transients, most of the radionuclide releases occur through the open containment penetrations rather than by leakage. In these cases, if containment isolation occurs on a high-pressure signal or by operator action, then the mass and surface areas of the walls and structures must be overestimated to delay or prevent triggering the containment isolation signal. This isolation delay overestimates the radionuclide releases through the open penetrations of containment. Conversely, if containment isolation occurs on a high radionuclide activity signal, then the mass and surface areas of the walls and structures are underestimated to maximize the containment pressure, and hence the radionuclide releases due to leakage.

For high-pressure transients, containment isolation occurs very early so that most of the radionuclide releases occur due to leakage. In these cases, the mass and surface areas of the walls and structures are underestimated to maximize the containment pressure, and hence the radionuclide releases due to leakage. The maximum ambient temperature is used to further maximize the containment pressure.

For molecular iodine and aerosol plateout simulation, the surface areas are underestimated, to minimize the removal of radionuclides from the containment atmosphere.

c) Reserve Water Tank

To overestimate radioactive releases, the considerations regarding overestimation or underestimation of reserve water stored in RWT are the same as 5.1.3 item b).

d) Air Coolers

Air cooler performance and modelling is described in Section 5.1.1d). To overestimate radioactive releases, the considerations regarding overestimation or underestimation of cooler availability and performance are the same as for 5.1.3 item b).

e) Heat Sources

For low-pressure transients, if containment isolation occurs on a high-pressure signal or by operator action, then the minimum value for the additional heat sources is assumed. This overestimates the radionuclide releases through the open penetrations of containment. If containment isolation occurs on a high radionuclide activity signal, then the maximum value for the additional heat sources is assumed. This overestimates the containment pressure, and hence the radionuclide releases due to leakage.

For high-pressure transients, containment isolation occurs very early, so that most of the radionuclide releases occur due to leakage. For these cases, the maximum value for the additional heat sources is assumed, to overestimate the containment pressure, and hence the radionuclide releases due to leakage.

f) Doors

All doors are assumed to remain closed and unavailable for atmosphere control.

g) Containment Isolation System

Containment isolation is assumed to occur on the high-pressure signal, high activity signal, or operator action, whichever occurs first.

Valve closure exceeds 90 percent within 0.5 s after the isolation signal, and is completed in less than 2 to 3 s (Reference [10]). In the analysis, additional margin is credited. The valves and dampers are conservatively assumed to remain fully open for 5 s after receiving the containment isolation signal, followed by instantaneous closure.

h) Instrument Air System

To overestimate radioactive releases, the most limiting assumption is used. For low-pressure transients, if containment isolation occurs on a high-pressure signal or by operator action, then no instrument air source term is assumed. This overestimates the radionuclide releases through the open penetrations of containment.

If containment isolation occurs on a high radionuclide activity signal, then instrument air is assumed to be released into containment. This overestimates the containment pressure, and hence the radionuclide releases due to leakage.

For high-pressure transients, containment isolation occurs very early, so that most of the radionuclide releases occur due to leakage. In these cases, instrument air is assumed to be released into containment as described in Section 5.1.1 item h). This overestimates the containment pressure, and hence the radionuclide releases due to leakage.

i) Hydrogen Recombiners

Hydrogen recombiners, though available, are ignored for these scenarios.

j) List of Cases for Radioactive Releases (Table 2)

The cases anticipated to require analysis for the assessment of radioactive releases from containment are as follows:

1) Large LOCA
2) Small LOCA
3) Pressurizer Pipe Break,
4) Stagnation Feeder Break,
5) Flow Blockage,
6) PT/CT Failure,
7) End Fitting Failure,
8) Large LOCA plus Loss of Emergency Core Cooling,
9) Pressurizer Pipe Break plus Loss of Emergency Core Cooling,
10) Small LOCA plus Loss of Emergency Core Cooling,
11) Stagnation Feeder Break plus Loss of Emergency Core Cooling,
12) Flow Blockage plus Loss of Emergency Core Cooling,
13) PT/CT Failure plus Loss of Emergency Core Cooling,
14) End Fitting Failure plus Loss of Emergency Core Cooling.

5.2 Loss of Class IV Power (LOCIVP)

The only subsystem of containment that is affected by loss of Class IV power is the network of air coolers. For analysis of maximum pressure in containment and radionuclide release from containment, the LACs are modelled as described in Section 5.1.1.

The assumptions used in the analysis are the same as described in Sections 5.1.1 and 5.1.3. No analysis of the timing of pressure-dependent signals is required for the loss of Class IV power scenarios because the signals are expected to occur earlier than the scenario with Class IV power available; therefore the predictions for Class IV power available are bounding.

The scenarios that will be analysed to determine the maximum pressure in containment and maximum differential pressure are:

1. Large LOCA plus Loss of Class IV power,
2. Main Steam Line Break plus Loss of Class IV power.

The following scenarios will be analysed for radionuclide releases for input to the dose analysis:

1. Large LOCA plus Loss of Class IV power,
2. Small LOCA plus Loss of Class IV Power,
3. Pressurizer Pipe Break plus Loss of Class IV Power,
4. Stagnation Feeder Break plus Loss of Class IV Power,
5. Flow Blockage plus Loss of Class IV Power,

6. PT/CT Failure plus Loss of Class IV Power,
7. End Fitting Failure plus Loss of Class IV Power.

5.3 Single Failures in Containment Subsystems

For the design basis accidents, a containment single failure coincident with the initiating event is considered. A single failure may be postulated in any one of the containment subsystems: single active failure in the short term and single active or passive failure in the long term. For LCDAs such as large LOCA with LOECC, no single failure of containment equipment needs to be assumed.

Based upon CNSC regulatory documents (Reference [3]) and experience, the following subsystem failures have been considered:

- Containment ventilation system isolation failures: The ACR generic design includes two independent, redundant means of isolating the containment ventilation lines; therefore single failure will not fail nor reduce the effectiveness of containment isolation.
- Vapour recovery system isolation failures: The ACR vapour recovery system is entirely inside containment and does not require isolation after an accident. In previous designs, the vapour recovery system extended outside of containment, therefore impairment of isolation of the vapour recovery lines was included in the accident analysis.
- Deflated airlock door seals: The doors on the personnel and equipment airlocks in the ACR-700 containment design do not have inflatable seals; therefore impairments of airlock door seals are not relevant.
- Dousing system failures: The ACR-700 does not have a dousing system; therefore dousing impairment is irrelevant.
- Failures in the energy suppression systems: Single failure in the energy suppression system is considered for ACR-700. For this single failure, the LACs are assumed to be partially lost due to a limiting single failure such as loss of one division of the RCW system. This single failure could result in increased containment pressure after an accident and increased radioactivity release to the atmosphere. The analysis of the design basis events credits only half the LACs (Sections 5.1.1 d) and 5.1.3 d)).
- Failures in the hydrogen control systems: Single failure in any of the three engineered provisions that contribute to hydrogen control in the ACR-700 design is considered. The three systems are:

  a) The dampers associated with the reactor building cooling system (by which flow into ducted coolers is switched from the steam generator room to the accessible area),
  b) Fans of the air coolers on Class III power (which contribute to mixing),
  c) A network of hydrogen recombiners (which lower hydrogen concentrations by recombination).

  As the hydrogen recombiners in ACR are passive, they need not be considered in the single failure analysis for the short-term. In long-term analysis cases, a failure of a single recombiner would be postulated. Single failure of the power supply for the air coolers is already covered by crediting only half the LACs in the analysis. Given the fact that all LACs

in one vault may be unavailable, the cases with a loss of LACs in one vault will be analysed. If a single failure can prevent the functioning of the dampers to redistribute the flows, then the damper movement will not be credited in the analysis.

5.4 Steam Generator Tube Leak for Input to Dose Analysis

A chronic steam generator tube leakage is postulated to exist. That is, one or more steam generator tubes are assumed to be leaking long enough that an equilibrium radionuclide burden has accumulated in the secondary circuit. The primary circuit radionuclide burden is assumed to be at the maximum allowable concentration of iodine in the circulating coolant.

For chronic steam generator tube leakage, it is assumed that an accident occurs which allows the radionuclides in the secondary circuit to be released to atmosphere through the MSSVs during crash cooldown.

The results of this analysis are added to the releases from containment for the atmospheric dispersion and public dose analysis

6. REFERENCES

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[3] "Requirements for Containment Systems for CANDU Nuclear Power Plants", Regulatory Document R-7, Atomic Energy Control Board, 1991 February.

[4] "Requirements for the Safety Analysis of CANDU Nuclear Power Plants", CNSC Consultative Document C–6, Rev. 1, CNSC, Sept. 1999.

[5] CNSC Report RSP-51, "Application of Event Tables", R.A. Brown and Associates Ltd., May 30, 1997 (RABA 9703).

[6] P. Iliescu, "Systematic Review of Plant Design – Methodology for Identification of Initiating Events", 108-03660-AB-001, Rev. 1, July 2003.

[7] Numerical Applications Inc., "GOTHIC Containment Analysis Package Technical Manual", Version 6.1b, NAI 8907-06, Rev. 11, April 2001.

[8] Numerical Applications Inc., "GOTHIC Containment Analysis Package User Manual", Version 6.1b, NAI 8907-02, Rev. 12, 2001.

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Table 1
Preliminary Listing of Major ACR Events Relevant to Containment Analysis

	Event Description	ACR Class
1.	Failure of normal steam generator feedwater flow	1
2.	Failure causing a loss of very small reactor primary coolant	1
3.	Failure of pressure tube of any channel assembly (calandria tube intact)	2
4.	Failure at any location of any pipe or header carrying steam from the steam generator to the turbine generator (outside R/B)	2
5.	Feeder failure – Off-stagnation feeder break	2
6.	Partial single channel blockage	2
7.	End fitting failure	2
8.	Failure at any location of any pipe or header carrying feedwater to the steam generators (inside R/B)	3
9.	Failure at any location of any pipe or header carrying steam from the steam generator to the turbine generator (inside R/B)	3
10.	Pressure tube/calandria tube failure	3
11.	Reactor main coolant system large LOCA	3
12.	Feeder failure – Stagnation feeder break	5
13.	Severe channel flow blockage	5
14.	Failure inside containment of any pipe or header carrying steam from the steam generators to the turbine-generator + failure of emergency coolant injection.	5
15.	Failure inside containment of any pipe or header carrying feedwater to the steam generators + failure of emergency coolant injection	5
17.	Feeder failure (off-stagnation feeder break)+ failure of emergency coolant injection	5
18.	End fitting failure + failure of emergency coolant injection.	5
19.	Reactor main coolant system large LOCA + failure of emergency coolant injection.	5

Note for the Table:

(1) Event combinations with loss of Class IV are not listed in this table. However, the following deterministic approach will be used for classification of the events involving failure of the reactor coolant pressure boundary along with loss of Class IV power for the generic ACR design; final classification will depend on the ACR plant site and associated grid reliability.

- Class 1 failure of RCPB + Loss of Class IV = Class 2 event
- Class 2 failure of RCPB + Loss of Class IV = Class 3 event
- Class 3 failure of RCPB + Loss of Class IV = Class 3 event
- Class 4 failure of RCPB + Loss of Class IV = Class 5 event
- Class 5 failure of RCPB + Loss of Class IV = Class 5 event

Table 2
Summary of Analysis Cases

Containment	Cases to be Considered		
	Maximum pressure, ΔP and H_2 distribution	Pressure-dependent signals	Radionuclide releases
Intact with Class IV power available	1. Large LOCA 2. Large LOCA + LOECC 3. Large MSLB	1. Small LOCA 2. Large LOCA 3. MSLB	1. Large LOCA 2. Small LOCA 3. Pressurizer Pipe Break 4. Stagnation Feeder Break 5. Flow Blockage 6. PC/CT Failure 7. End Fitting Failure 8. Large LOCA + LOECC 9. Small LOCA + LOECC 10. Pressurizer Pipe Break + LOECC 11. Stagnation Feeder Break + LOECC 12. Flow Blockage + LOECC 13. PT/CT + LOECC 14. End Fitting Failure + LOECC
Intact with Loss of Class IV power	1. Large LOCA 2. Large MSLB		1. Large LOCA 2. Small LOCA 3. Pressurizer Pipe Break 4. Stagnation Feeder Break 5. Flow Blockage 6. PC/CT Failure 7. End Fitting Failure

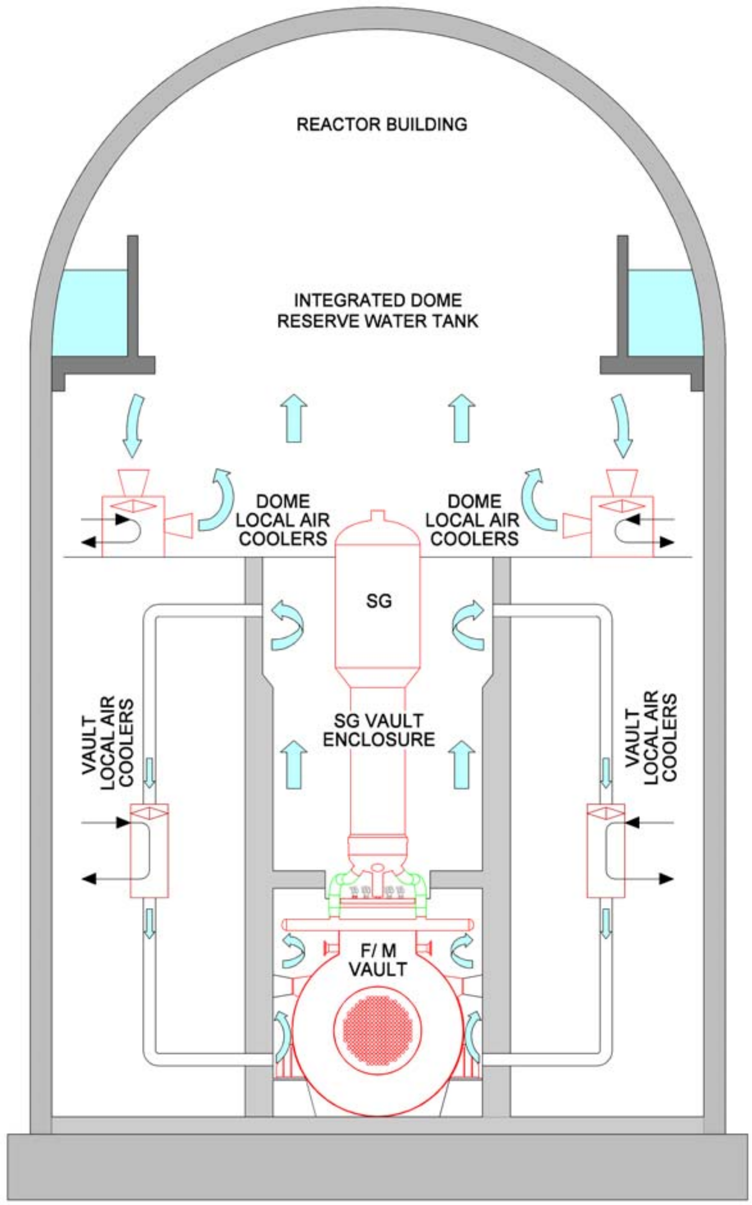

Figure 1 Components in Containment

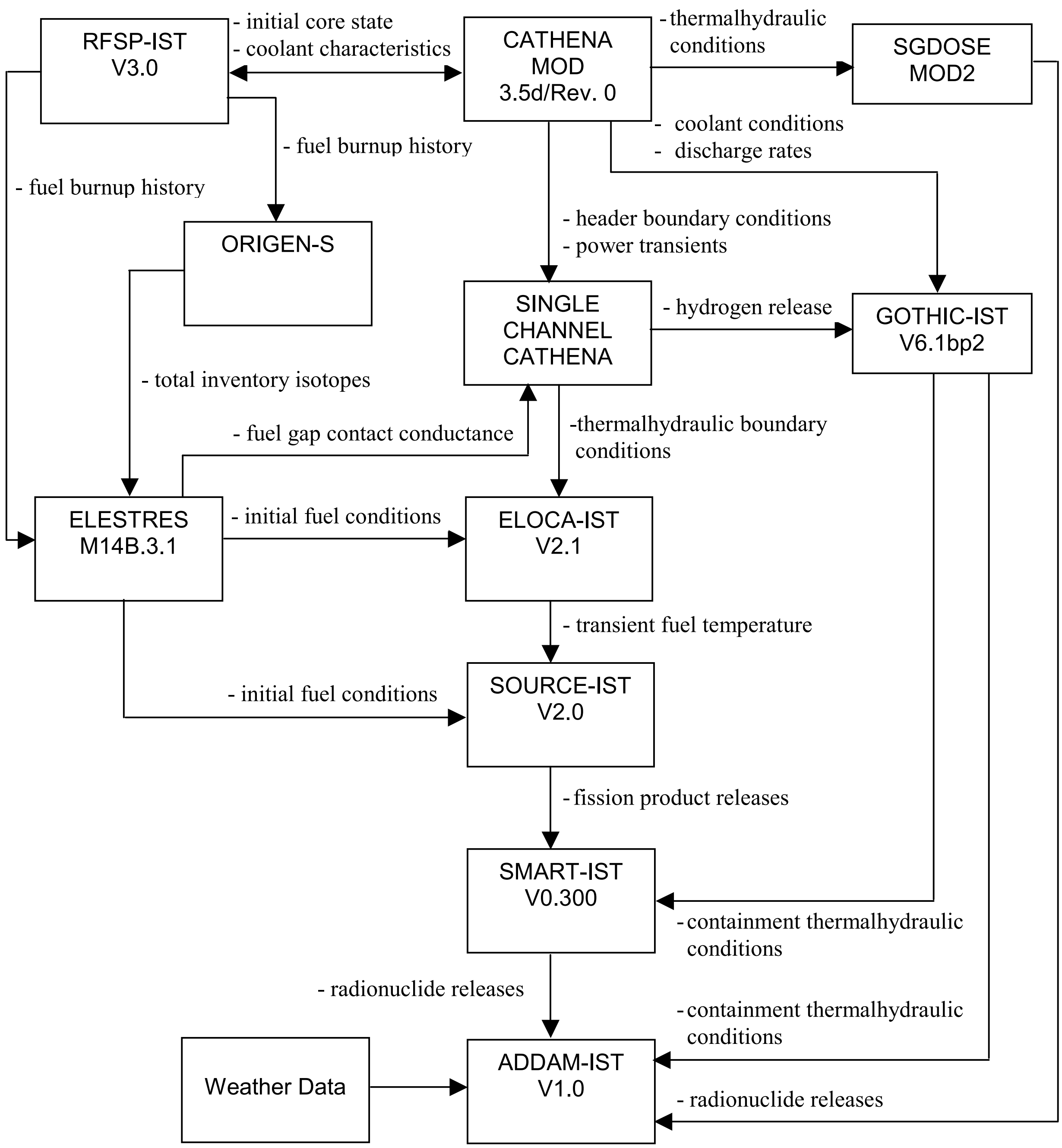

Figure 2 Schematic of Links to Suite of Codes

Type	Radioactive Decay and Build-up Scheme
1	$\underset{\chi_{amount}}{C} \xrightarrow{C_{half}} \text{Stable Isotope}$
2	$\underset{\beta_{amount}}{B} \xrightarrow{B_{half}} \underset{\chi_{amount}}{C} \xrightarrow{C_{half}} \text{Stable Isotope}$
3	$\underset{\alpha_{amount}}{A} \xrightarrow{A_{half}} \underset{\beta_{amount}}{B} \xrightarrow{B_{half}} \underset{\chi_{amount}}{C} \xrightarrow{C_{half}} \text{Stable Isotope}$
4	$\begin{tikzcd} & B\ \beta_{amount} \arrow[d, "B_{half}"] & \\ \underset{\alpha_{amount}}{A} \arrow[ur, "A_{half}\ a"] \arrow[r, "b", "A_{half}"'] & \underset{\chi_{amount}}{C} \arrow[r, "C_{half}"] & \text{Stable Isotope} \end{tikzcd}$

Figure 3 Radionuclide Decay Chain Types in SMART-IST