



Analysis Basis

SAFETY BASIS FOR ACR

ACR

108-03600-AB-003

Revision 0

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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
CNSC	Canadian Nuclear Safety Commission
CT	Calandria Tube
DE	Design Basis Event
DDT	Deflation-to-Detonation Transition
ECC	Emergency Core Cooling
ECCS	Emergency Core Cooling System
ECI	Emergency Coolant Injection
HTS	Heat Transport System
IAEA	International Atomic Energy Agency
IE	Initiating Event
LCDA	Limited Core Damage Accident
LOCA	Loss Of Coolant Accident
MSSV	Main Steam Safety Valve
P&IC	Pressure and Inventory Control
PIE	Postulated Initiating Event
PRA	Probabilistic Risk Assessment
PSA	Probabilistic Safety Analysis
PT	Pressure Tube
R/B	Reactor Building
RCPB	Reactor Coolant Pressure Boundary
SCDA	Severe Core Damage Accident
SG	Steam Generator
US	United States
US NRC	United States Nuclear Regulatory Commission

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

The Advanced CANDU Reactor (ACR), a next generation CANDU reactor, is designed for international markets. It is an internationally common practice to demonstrate the nuclear safety of a nuclear power plant by conducting a comprehensive safety analysis. It is also an international requirement that a complete spectrum of events be analysed and that the acceptance criteria be clearly defined [1]. The approach followed in the safety analysis of the ACR design should:

- a) comply with relevant Canadian nuclear safety requirements,
- b) be compatible with internationally accepted practices,
- c) be easily understood internationally, and
- d) result in one set of internationally acceptable safety analyses, which will eliminate or minimize duplication of effort for different customers in all jurisdictions.

This report provides the proposed approach for definition and classification of the events to be analysed in the safety analysis of the ACR design. The acceptance criteria for each class of events are provided. The general analysis input information and analysis methodologies are briefly discussed. This report is organized as follows:

- Section 2 provides the objectives of this report.
- Section 3 provides the proposed approach for definition and classification of the ACR events, their acceptance criteria, general analysis input information and general analysis methodologies.
- Section 4 summarizes the overall conclusions of this report.

2. OBJECTIVES

This report describes the bases for safety analysis of the ACR design in terms of classification of events to be analysed, the acceptance criteria, performance targets and general analysis methodologies for each class of events, and justifies the proposed safety analysis approach with respect to both Canadian and relevant international safety requirements.

This report covers the complete spectrum of the events to be analysed in the safety analysis of the ACR design.

This report is at the top of the hierarchy of a series of Analysis Basis (AB) reports. The series includes AB reports which are prepared for specific analysis areas consistent with the approach described in this report.

3. SAFETY ANALYSIS APPROACH FOR ACR

3.1 General

The overall objective of a nuclear safety analysis approach is to demonstrate that the radiological risk to the public, plant staff, and the environment from the nuclear power plant is acceptably low by national and international standards. To meet this objective, a complete spectrum of events needs to be analysed. Key to the safety analysis approach is the definition and classification of events to be analysed.

A well-balanced and acceptable safety analysis approach should be based on a widely accepted risk-informed concept; 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 safety analysis approach should exhibit a number of features that have been proven as good practices and/or can be found in national and international safety requirements. These features include:

- The scope should cover the entire spectrum of potential challenges to the safety of the plant from “design basis events (DEs)” to “severe core damage accidents (SCDAs)”;
- The safety requirements should be commensurate with the likelihood of the events so that more stringent requirements are applied to more probable events; and
- The overall approach should involve detailed analyses for more probable events and integral analyses for the less probable end of the accident spectrum. This ensures a proper focus on achieving a greater degree of confidence in the demonstration of compliance with requirements for more probable events.

The Canadian Nuclear Safety Commission’s consultative document, C-006 (Rev. 1) [2] provides a system of classification of events into five classes and associated acceptance criteria in terms of “Dose and Release Limits” and “Regulatory Requirements and Derived Acceptance Criteria”. This classification of events is mainly based on engineering judgement taking into account operating experience and relevant analyses. This consultative document also provides an alternative approach for event classification that is based on probabilistic techniques, and classifies events according to their estimated frequencies of occurrence. Other requirements for the classification of events and their application to safety analysis are provided in international standards and practices, such as the International Atomic Energy Agency’s (IAEA) safety requirements [1] and US NRC Regulatory Guide 1.70 [3].

To meet the intent of C-006 (Rev. 1) [2] and follow international standards and practices, a safety analysis approach has been proposed for the ACR. The proposed ACR approach:

- adopts five classes of events with different radiological dose limits presented in Table 1 and consistent with C-006 (Rev. 1) [2];
- follows the basic interpretation of the C-006 (Rev. 1) companion document, RABA 9703 [4], for classification and treatment of low frequency events;
- adopts acceptance criteria and performance targets based on safety margins that increase with the likelihood of the events in a class;

- uses assumptions and methods that provide a good balance between the need for conservatism at higher event frequencies and the reasonableness of a more design-centred, risk-informed assessment at the lower event frequencies; and
- extends the analysis scope to include SCDAAs that are beyond the scope defined by C-006 (Rev. 1) [2].

ACR events and their combinations are classified into classes 1 through 5 consistent with the intent of C-006 (Rev. 1) [2]. Since the events listed in C-006 (Rev. 1) [2] may not be exhaustive and are mostly applicable to traditional CANDU reactors rather than a new design, some classifications provided in C-006 (Rev. 1) [2] may not be appropriate and some ACR events may not be covered by C-006 (Rev. 1) [2]. Therefore, additional engineering judgement will be needed for the classification of certain ACR-specific events. C-006 (Rev. 1) [2] also provides an alternative frequency-oriented basis for event classification. This basis will also be consulted in making the engineering judgement for the classification of events.

The ACR safety analyses examine:

- design basis events (DEs),
- limited core damage accidents (LCDAs), and
- severe core damage accidents (SCDAAs).

The design basis events are events which must be accommodated with suitable margins to the breach of the physical barriers against the release of radioactivity to the environment. As such, analyses for these events must demonstrate only small consequences for the safety of the public. The design basis events set the design requirements for engineered safety features, in particular shutdown systems, Emergency Core Cooling System (ECCS) and containment. The design basis events cover Class 1 through 3 events.

The LCDAs 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. LCDAs include Class 4 and 5 events.

The SCDAAs are those events which result in widespread loss of the core and channel geometry, and are directly equivalent to severe accidents in other countries.

The safety analysis of design basis events will be based on conservative assumptions and will assume a single component failure in a mitigating system in addition to the initiating event. The single failure criterion as defined in the US requirements [5], [6], [7], has been applied to the design of the ACR mitigating systems credited in deterministic safety analysis beside other reliability requirements. This is an additional defence-in-depth built into ACR design because application of single failure criterion is considered to lead to more robust mitigating systems.

The safety analyses for the LCDAs and SCDAAs will use design-centred assumptions. Detailed deterministic analyses will be done for the LCDAs whereas a Level 2 Probabilistic Safety Analysis (PSA) will be performed for the SCDAAs.

3.2 Comparison with the IAEA Requirements

The proposed approach is consistent with the International Atomic Energy Agency's (IAEA) safety assessment requirements [8]. The IAEA requirements outline some principles of a modern safety assessment framework, such as:

- "4.19 In general, the deterministic analysis for design purposes should be conservative. The analysis of beyond design basis accidents is generally less conservative than that of design basis accidents.
- 4.20 The PSA should set out to determine all significant contributors to risk from the plant and should evaluate the extent to which the design of the overall system configuration is well balanced, there are no risk outliers and the design meets basic probabilistic targets. The PSA should preferably use a best estimate approach."

The IAEA framework for safety assessment follows the principle that a higher degree of assurance is required for the more probable postulated initiating events (PIEs), that might be expected to occur during the lifetime of the plant, so they will not result in any significant fuel damage and hence will not pose any undue risk to the health and safety of the public. It is also accepted that some less probable events may result in fuel damage but the safety systems will act to minimize any risks to the public or the environment. For events with a very low likelihood of occurrence, much less than would be expected to occur during the lifetime of a nuclear power plant, it is recognized that a greater degree of uncertainty is inevitable in the ability to demonstrate compliance with risk targets by analysis alone. Ultimately, for highly unlikely events, risk-based targets are set and are supplemented by additional practical measures such as formalized severe accident management and emergency response programs to provide additional safety assurance.

These principles found in the IAEA requirements are being followed in the proposed ACR safety analysis approach that includes conservative analyses of the more probable design basis events and design-centred, risk-informed analyses for the less probable events (LCDAs and SCDA).

The proposed ACR safety analysis approach includes both deterministic analysis and PSA.

To meet the IAEA requirement that severe accidents should be given adequate consideration [1], the safety analysis approach has extended the analysis scope to SCDA beyond the scope of C-006 (Rev. 1) [2]. Performance targets are imposed on the design of ACR in terms of total frequency of SCDA and total frequency of large releases.

3.3 Comparison with US Requirements

Both the US and Canadian regulations require safety analysis be performed in a graded manner. In US practice, the events analysed are classified as Design Basis Events and Severe Accidents, and the Design Basis Events are further grouped into three Plant Conditions as defined in the US NRC Regulatory Guide 1.70, Rev. 3 [3]. The event types and ranges for event classification are similar in US and Canadian practices.

The ACR design basis events are essentially equivalent to the Design Basis Events used in the US practice. These are events which are analysed to demonstrate the robustness of engineered safety features and to show that offsite dose limits are met. Conservative assumptions are used for safety analyses of design basis events in both the proposed approach and the US practice. As

already stated in Section 3.1, a single component failure of a mitigating system will be assumed in the safety analysis consistent with US requirements.

The adoption of design-centred assumptions for analysis of low frequency events, in the domain of LCDAs and SCDA, is also consistent with the US risk-informed approach, that allows less conservative assumptions to be used for analysis of severe accidents.

It is a US requirement that a Level 2 PRA be performed. This requirement is met by the conduct of a Level 2 PSA for SCDA of the ACR.

3.4 Identification and Classification of Initiating Events

The postulated initiating events will be identified by a systematic review of the ACR plant design. Potential failure modes of the plant that could lead to the release of fission products from the fuel will be identified. The systematic review uses, among other techniques, Probabilistic Safety Assessment and the review of operating experience from CANDU plants, and looks at multiple system failures as well as failures of single components. This review will follow the guidelines provided in C-006 (Rev. 1) [2]. Due consideration will be given to the fact that ACR is a new design with several innovative design and safety features. This means that the likelihood of an event occurring and its safety impact may be different for ACR than for traditional CANDU plants.

All identified events will be classified consistent with the intent of C-006 (Rev. 1) [2]. For cases in which C-006 (Rev. 1) [2] is insufficient or inappropriate, additional engineering judgement will be used based on relevant operating experience and analyses.

As a result of an initial review of the ACR, a preliminary list of major ACR events has been prepared and events classified (see Table 2). These events have been identified from C-006 (Rev. 1) [2], from previous experience on CANDU reactors and review of operations feedback. A small number of ACR events have been given a different classification from C-006 (Rev. 1) [2]. The justification for these exceptions, founded on probabilistic considerations, operating experience and international practices, will be provided in the relevant AB reports. A comprehensive list of ACR events will be available after the systematic review of the ACR is completed.

3.5 Design Basis Events

3.5.1 Definition

Design basis events are events for which the plant is designed to ensure that the specified radiological dose limits to the public are not exceeded and that the integrity of key barriers (including fuel, reactor coolant pressure boundary and containment) to the release of radioactivity to the environment are maintained with suitable margins. Design basis events include Class 1 through 3 events. A Class 1 event is an event of moderate frequency that may occur during a calendar year for a particular plant. A Class 2 event is an infrequent event that may occur during the lifetime of a particular plant. A Class 3 event is a limiting event that is not expected to occur but is postulated because of its potentially significant consequences.

3.5.2 Acceptance Criteria

3.5.2.1 Acceptance Criteria for Class 1 Events

Class 1 events shall meet the following acceptance criteria as presented in Table 3:

- The dose resulting from Class 1 events shall be no more than the limits presented in Table 1.
- Systematic fuel failures, i.e. failure of fuel that was not already defective prior to the accident occurring shall be prevented. This will be demonstrated by applying, for Class 1 events, the trip parameter acceptance criteria defined in Canadian Nuclear Safety Commission's Consultative Document, C-144 [9] or by performing detailed fuel analysis.
- Class 1 events shall not result in pressure tube failure.
- The level B Service Limit* shall not be exceeded assuming that the first shutdown system trips as intended, and the level C Service Limit shall not be exceeded assuming that the first shutdown system fails to act, but the second shutdown system trips. This criterion is in accordance with the Canadian Nuclear Safety Commission's Regulatory Policy Statement, R-77 [10].

3.5.2.2 Acceptance Criteria for Class 2 Events

Class 2 events shall meet the following acceptance criteria as presented in Table 3:

- The dose resulting from Class 2 events shall be no more than the limits presented in Table 1.
- Class 2 events shall not result in calculated failures of fuel in channels which are not affected by the initiating event.
- Class 2 events shall not result in pressure tube failure in channels which are not affected by the initiating event.
- The level C Service Limit shall not be exceeded assuming that the first shutdown system trips as intended, and the level D Service Limit shall not be exceeded assuming that the first shutdown system fails to act, but the second shutdown system trips. This criterion is in accordance with the Canadian Nuclear Safety Commission's Regulatory Policy Statement, R-77 [10].
- The containment peak pressure resulting from Class 2 heat transport system (HTS) events shall not exceed the design pressure of containment.
- The calandria shall remain intact.

3.5.2.3 Acceptance Criteria and Performance Targets for Class 3 Events

Class 3 events shall meet the following acceptance criteria as presented in Table 3:

- The dose resulting from Class 3 events shall be no more than the limits presented in Table 1.
- Fuel failures resulting from Class 3 events shall be limited.

* Level B, C, and D Service Limits are defined in the General Requirements under Section III of the American Society of Mechanical Engineer Boiler and Pressure Vessel Code (ASME Code)

- Class 3 events shall not result in PT failure in channels which are not affected by the initiating event.
- The level C Service Limit shall not be exceeded assuming that the first shutdown system trips as intended, and the level D Service Limit shall not be exceeded assuming that the first shutdown system fails to act, but the second shutdown system trips. This criterion is in accordance with the Canadian Nuclear Safety Commission's Regulatory Policy Statement, R-77 [10].
- The containment peak pressure resulting from any Class 3 event, with the potential to release radioactive material to the extent that the applicable dose limits might be exceeded if containment leaktightness were not maintained, shall not exceed the design pressure of containment. No Class 3 event shall result in damage to the containment structure. The hydrogen concentration in containment shall remain below the flammability limit.
- The calandria shall remain intact.

For Class 3 events, a performance target is that there be no significant plastic deformation of pressure tubes.

3.5.3 Analysis Methodology

Safety analysis will be performed for the design basis events to demonstrate compliance with the acceptance criteria given in Section 3.5.2.

The safety analysis will consider a single component failure in a mitigating system. The single failure criterion, as defined in the US requirements and used on advanced US reactors, has been applied to the design of the ACR mitigating systems that are credited in safety analysis.

Application of the single failure criterion includes consideration of active failures in the short and long term, and passive failures in the long term (typically after 24 hours from the initiating event). Some examples of single failures of mitigating systems are provided in Table 4; the single failures of mitigating systems to be used in safety analysis will be determined for each design basis event analysis.

Conservative assumptions will be used in the safety analysis of design basis events. In particular, each event will also use the assumptions of two shutoff rods stuck out of reactor for Shutdown System #1 or one liquid injection nozzle not available for Shutdown System #2. Systems and equipment credited in the safety analysis of design basis event will be qualified for the resulting event conditions. The safety analysis of the design basis events will be performed using detailed models and computer codes that have been adequately validated and verified.

Detailed analysis input information for design basis events will be prepared and documented in separate AB reports.

3.6 Limited Core Damage Accidents

3.6.1 Definition

Limited core damage accidents (LCDAs) are more improbable events, which must be accommodated within specified radiological dose limits to the public. Most LCDAs are

combinations of an initiating event and the total failure (or the worst failure) of a safety system. LCDAs include Class 4 and 5 events.

3.6.2 Acceptance Criteria and Performance Targets

Class 4 and 5 events shall meet the following acceptance criteria as presented in Table 3:

- The dose resulting from Class 4 and 5 events shall be no more than the limits presented in Table 1.
- The calandria shall remain intact.

For Class 4 and 5 events, the performance targets are:

- For all Class 4 and 5 events, there should be no fuel centreline or sheath melting in non-failed channels.
- Class 4 and 5 events should not result in channel failure in non-affected channels. If pressure tubes sag into calandria tube contact, sufficient moderator subcooling should be ensured to safeguard channel integrity.
- The peak pressure resulting from Class 4 and 5 events, with the potential to release radioactive material to the extent that the applicable dose limits might be exceeded if containment leaktightness were not maintained, should not exceed the design pressure of containment. The structural integrity of containment is ensured to a degree that consequential damage to reactor systems could not result from any Class 4 and 5 event, and the hydrogen concentration in the containment should remain below the deflagration-to-detonation transition (DDT) limit.

3.6.3 Analysis Methodology

Safety analysis will be performed for the LCDAs to demonstrate compliance with the acceptance criteria and performance targets given in Section 3.6.2.

The safety analysis for the LCDAs will use design-centred assumptions. Systems and equipment credited in the safety analysis but not qualified by design will be assessed for their adequacy of performance under the resulting event conditions. The safety analysis will be performed using detailed models and computer codes that have been adequately validated and verified.

Detailed analysis input information for LCDAs will be prepared and documented in separate AB reports.

3.7 Severe Core Damage Accidents

3.7.1 Definition

Severe Core Damage Accidents (SCDAs) are extremely improbable events, which lead to loss of core geometry.

3.7.2 Performance Targets

The following performance targets have been established for the SCDAs of the ACR:

- The total frequency of SCDAs should be less than 10^{-5} per reactor year.

- The total frequency for accident sequences leading to large releases of radioactivity should be less than 10^{-6} per reactor year.

The above total frequency targets also include external events except earthquakes. A seismic margin assessment will be performed separately for earthquakes.

A Level 2 Probabilistic Safety Analysis (PSA) will be performed to determine compliance with the above performance targets.

3.7.3 Analysis Methodology

The safety analysis (Level 2 PSA) of SCDA's will use design-centered assumptions.

A PSA using integral models will be employed in the safety analysis of SCDA's of the ACR to demonstrate that the performance targets given in Section 3.7.2 are met.

Detailed analysis input information for SCDA's will be prepared and documented in separate AB reports.

4. CONCLUSIONS

A safety analysis approach has been proposed for the ACR that includes

- an event classification system covering design basis events through severe core damage accidents;
- acceptance criteria and/or performance targets associated with each class of event; and
- general analysis methodologies including types of assumptions and models to be used in the analysis.

The proposed approach differs from the traditional CANDU practices in the following aspects:

- The proposed approach extends the safety analysis scope to severe core damage accidents that are beyond the scope of the traditional CANDU practices and C-006 (Rev. 1) [2];
- The safety analysis of design basis events will include consideration of a single component failure in a mitigating system in addition to other conservative assumptions; and
- The safety analyses for limited core damage accidents and severe core damage accidents will use design-centred assumptions.

The key features of the proposed approach are summarized in Table 5.

This approach is based on the widely accepted risk-informed concept, complies with the intent of Canadian regulatory requirements and is compatible with international requirements and practices, particularly those used in the US.

5. REFERENCES

- [1] IAEA Safety Requirements NS-R-1, Safety of Nuclear Power Plants: Design, Vienna 2000.
- [2] Draft Regulatory Guide, Safety Analysis of CANDU Nuclear Power Plants, C-006 (Rev. 1) (E), Canadian Nuclear Safety Commission (CNSC), September 1999.
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- [4] CNSC Report # RSP-51, Application of Event Tables, R.A. Brown and Associates Ltd., May 30, 1997. (RABA 9703).
- [5] US NRC, "Code of Federal Regulations, Title 10, Part 50, Appendix A: General Design Criteria for Nuclear Power Plants", January 1998.
- [6] IEEE, "Application of the Single Failure Criterion to Nuclear Power Generating Station Safety Systems", Standard No. 379-2000.
- [7] ANSI/ANS, "Single Failure Criteria for Light Water Reactor Safety-Related Fluid Systems", Standard No. 58.9-1981.
- [8] IAEA Safety Guide NS-G1.2, Safety Assessment and Verification for Nuclear Power Plants, Vienna 2001.
- [9] Consultative Document, Trip Parameter Acceptance Criteria for the Safety Analysis of CANDU Nuclear Power Plants, C-144, Canadian Nuclear Safety Commission (CNSC), October 1997.
- [10] Regulatory Policy Statement, Overpressure Protection Requirements for Primary Heat Transport Systems in CANDU Power Reactors Fitted with Two Shutdown Systems, R-77, Canadian Nuclear Safety Commission (CNSC), October 20, 1987.

Table 1
Dose and Release Limits**

Requirement	Event Class				
	1	2	3	4	5
effective dose (mSv)	0.5	5	30	100	250
lens of the eye (mSv)	5	50	300	1,000	1,500
skin (mSv, averaged over 1 cm ²)	20	200	1,200	4,000	5,000
30 day emissions of liquid effluent are within the derived annual emission limits for normal operation	✓	✓	N	N	N

✓ — the limit shall be met by the worst failure sequence in the event class.

N — not required.

** Taken from C-006 (Rev. 1) [2]

Table 2
Preliminary Listing of Major ACR Events

	Event Description	ACR Class	Notes
1.	Failure of control systems <ul style="list-style-type: none"> • Reactor power control • Steam generator pressure control • Steam generator inventory control • Primary coolant pressure and inventory control • Moderator temperature control 	1	
2.	Failure of normal electrical power	1	
3.	Failure of normal steam generator feedwater flow	1	
4.	Failure of moderator system (excluding piping failures)	1	
5.	Failure of reactor shield cooling system (excluding piping failures)	1	
6.	Failure of normal cooling system of fuelling machine	1	
7.	Failures resulting in inadvertent heat transport pump trip	1	
8.	Failure causing a loss of very small reactor primary coolant	1	Failure of reactor coolant pressure boundary (RCPB).
9.	Failure of a single steam generator tube	2	Nuclear standard passive component and based on international practice. Failure of RCPB.
10.	Failure of pressure tube of any channel assembly (calandria tube intact)	2	Nuclear standard passive component and improved ACR pressure tubes.
11.	Failure at any location of any pipe or header carrying steam from the steam generator to the turbine generator (outside R/B)	2	
12.	Feeder failure – Off-stagnation feeder break	2	Failure of RCPB.
13.	Failure of moderator system piping	2	
14.	Reactor shield cooling system piping failures	2	
15.	Partial single channel blockage	2	
16.	Failure at any location of any pipe or header carrying feedwater to the steam generators (outside R/B)	2	
17.	End fitting failure	2	
18.	Failure at any location of any pipe or header carrying feedwater to the steam generators (inside R/B)	3	
19.	Failure at any location of any pipe or header carrying steam from the steam generator to the turbine generator (inside R/B)	3	Nuclear standard passive component.
20.	Pressure tube/calandria tube failure	3	Improved PT and CT for ACR. Failure of RCPB.
21.	Seizure of a single reactor primary coolant pump	3	
22.	Reactor main coolant system large LOCA	3	Failure of RCPB.

	Event Description	ACR Class	Notes
23.	Fuelling machine backing off the reactor without the fuel channel assembly closure plug being replaced + failure of emergency coolant injection	4	Failure of RCPB.
24.	Feeder failure – Stagnation feeder break	5	Failure of RCPB.
25.	Severe channel flow blockage	5	Failure of RCPB.
26.	Failure of a large number of steam generator tubes	5	Failure of RCPB.
27.	Failure inside containment of any pipe or header carrying steam from the steam generators to the turbine-generator + failure of emergency coolant injection.	5	
28.	Failure inside containment of any pipe or header carrying feedwater to the steam generators + failure of emergency coolant injection	5	
29.	Feeder failure (off-stagnation feeder break)+ failure of emergency coolant injection	5	Failure of RCPB.
30.	End fitting failure + failure of emergency coolant injection.	5	Failure of RCPB.
31.	Reactor main coolant system large LOCA + failure of emergency coolant injection.	5	Failure of RCPB.

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 3
Acceptance Criteria for the ACR Events**

Events	Acceptance Criteria	Performance Targets
Class 1	<ul style="list-style-type: none"> - Dose: C-006 (Rev. 1) [2] Class 1 limits - Fuel: no systematic fuel failures - Fuel Channel: no PT failures - Overpressure/pressure boundary integrity: <ul style="list-style-type: none"> • Level B limit for first shutdown system to trip • Level C limit for second shutdown system to trip 	
Class 2	<ul style="list-style-type: none"> - Dose: C-006 (Rev. 1) [2] Class 2 limits - Fuel: no failures (in non-failed channels) - Fuel Channel: no PT failures (in non-affected channels) - Overpressure/pressure boundary integrity: <ul style="list-style-type: none"> • Level C limit for first shutdown system to trip • Level D limit for second shutdown system to trip - Containment: Peak pressure not to exceed design pressure (primary system events) - Other: Calandria remains intact 	
Class 3	<ul style="list-style-type: none"> - Dose: C-006 (Rev. 1) [2] Class 3 limits - Fuel: limit failures - Fuel Channel: no PT failures (in non-affected channels) - Overpressure/pressure boundary integrity: <ul style="list-style-type: none"> • Level C limit for first shutdown system to trip • Level D limit for second shutdown system to trip - Containment: <ul style="list-style-type: none"> • Peak pressure not to exceed design pressure for events causing release of significant amount of radioactive material • No damage to the containment structure • Hydrogen concentration to remain below flammability limit - Other: Calandria remains intact 	<ul style="list-style-type: none"> - No significant plastic deformation of PTs

Events	Acceptance Criteria	Performance Targets
<p>Class 4/5 (LCDAs)</p>	<ul style="list-style-type: none"> - Dose: C-006 (Rev. 1) [2] Class 4 and Class 5 limits respectively 	<ul style="list-style-type: none"> - Fuel: <ul style="list-style-type: none"> • no fuel centerline or sheath melting (in non-failed channels) - Fuel Channel: <ul style="list-style-type: none"> • no fuel channel failures (in non-affected channels) • ensure sufficient moderator subcooling if PT sags into CT contact to safeguard channel integrity - Containment: <ul style="list-style-type: none"> • Peak pressure not to exceed design pressure for events causing release of significant amount of radioactive material • Structural integrity of containment walls ensured to a degree that consequential damage to reactor systems could not result (for all events) • Hydrogen concentration to remain below DDT limit - Other: Calandria remains intact
<p>Severe Core Damage Accidents (SCDAs)</p>		<ul style="list-style-type: none"> - SCDA targets are in terms of frequency (from PSA) - Total frequency of SCDAs is $<10^{-5}$ per year and total frequency for accident sequences leading to large releases of radioactivity is $<10^{-6}$ per year - Total frequencies include external events except seismic (a seismic margin assessment will be performed for earthquakes)

Table 4
Examples of Single Failures Assumed in Analysis of Class 1 through 3 Events

Event Description	Single Failure
Large LOCA	One of ECI tank discharge valves or one large outlet header interconnection valve fails to open
Small LOCA	MSSVs in one channel fail to open
Pressure Tube/Calandria tube failure	MSSVs in one channel fail to open
End Fitting Failure	MSSVs in one channel fail to open
Feeder Break (off-stagnation)	MSSVs in one channel fail to open
Single SG Tube Failure	One faulted steam generator power-operated atmospheric steam discharge valve (ASDV) fails open
Inadvertent Increase of HTS Pressure due to failure of Pressure and Inventory Control	One HTS liquid relief valve fails to open
Partial Channel Flow Blockage	One HTS liquid relief valve fails to open
Loss of Normal Electrical Power	One HTS liquid relief valve fails to open
Seizure of A Reactor Primary Coolant Pump	One HTS liquid relief valve or one MSSV fails to open
Steam Main Breaks	One ECI tank discharge valve fails to open
Feedwater Line Break	One HTS liquid relief valve fails to open
Loss of Normal Feedwater Flow	One HTS liquid relief valve fails to open
Inadvertent Increase of Reactor Power due to failure of Reactor Power Control	One HTS liquid relief valve or one MSSV fails to open

**Table 5
Key Features of Safety Analysis Approach for ACR**

Event Category	Analysis Assumptions	Analysis Models	Acceptance Criteria	Targets
Design Basis Events (Classes 1,2 & 3)	Conservative ⁽¹⁾	Detailed ⁽³⁾	Performance Criteria – Radiological Doses	Performance Targets
Limited Core Damage Accidents (Classes 4 & 5)	Design-centred ⁽²⁾	Detailed	Radiological Doses	Performance Targets
Severe Core Damage Accidents	Design-centred	Integral ⁽⁴⁾	–	Frequencies of Severe Core Damage and Large Release

Notes for the Table:

1. An example of conservative assumptions is 102% power for circuit simulation.
2. An example of design-centred assumptions is 100% power for circuit simulation.
3. An example of detailed models is computer code CATHENA.
4. An example of integral models is computer code MAPP4 CANDU.