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Analysis Basis

Trip Coverage Analysis Methodology

ACR-700

10810-03550-AB-001
Revision 1

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1. INTRODUCTION

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

- Defines the classes of events;
- Categorize 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;
- Describe 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
- Report 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.

This analysis basis report presents the major system assumptions, acceptance criteria and analysis methodology for the trip coverage analysis.

The functions of the shutdown systems are to detect conditions in the reactor, heat transport system, moderator, feedwater and steam supply systems that could impair the ability to maintain adequate fuel cooling and heat removal, and to initiate a timely reactor shutdown. The general requirements for the shutdown systems are outlined in References 3 and 4.

In order to provide assurance that the trip parameters will perform these functions, an assessment of trip parameter effectiveness is performed for the events listed in CNSC regulatory document R-8 (References 3). This assessment involves verifying the adequacy of the trip parameters to detect potentially unsafe plant conditions, and to initiate an effective reactor shutdown.

The following sections describe the acceptance criteria, assumptions, and analysis methodology. The last section describes the event sequence and specific analysis for each event.

This analysis basis is limited to the thermalhydraulic analysis of the heat transport and steam and feedwater systems, including any specific single channel analysis that is required. It does not include containment, dose, or detailed fuel and fuel channel analysis, as they will be covered in separate analysis basis documents.

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

2. ACCEPTANCE CRITERIA

The following overall safety objectives are used for analysis of transients and accidents:

a) Systematic fuel failures shall be prevented for Class 1 and 2 events. This is achieved by preventing excessive fuel overheating. For Class 3 events, fuel failures shall be limited.
b) Pressure tube integrity shall be maintained (other than in the affected channel). This is achieved by preventing excessive fuel and fuel channel overheating.
c) Heat transport system (HTS) integrity shall be maintained after an overpressure accident. This is achieved by preventing excessive overpressure of the HTS system.
d) An adequate water inventory shall be provided in the steam and feedwater system after secondary circuit failure to provide a heat sink until a long-term heat sink can be established, when required.
e) Containment pressure following a large LOCA shall be limited to within the design pressure of the reactor building
f) Containment overpressure for steam main breaks inside containment shall be limited to ensure no damage to containment.
g) Public dose limits shall be met. The reference dose limits depend on the classification of the event.

The Safety Basis for ACR analysis (Reference 1) describes the events and event classification for ACR. A number of design basis events require demonstration of adequate trip coverage, as described in Reference 3. These are listed in Table 2–1, along with the event classification from Reference 1. Class 4 and 5 events are not included in Table 2-1 because the trip effectiveness analysis is already covered by the events in Class 1 to 3.

The detailed description of the acceptance criteria is contained in the initial conditions and assumptions analysis basis (Reference 2). The major criteria for trip coverage, including system overpressurization and prevention of fuel failures are described briefly in the following sections.

Table 2–1
Preliminary Listing of Major ACR Design Basis Events

	Event Description	ACR Class
1.	Failure of control systems • 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 very small loss of reactor primary coolant	1
9.	Failure of a single steam generator tube	2
10.	Failure of pressure tube of any channel assembly (calandria tube intact)	2
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
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
20.	Pressure tube/calandria tube failure	3
21.	Seizure of a single reactor primary coolant pump	3
22.	Reactor main coolant system large LOCA	3

2.1 Fuel Integrity

For Class 1 events such as loss of flow and loss of reactivity control, the acceptance criterion is to prevent systematic fuel failures; that is, to prevent failure of fuel that was not already defective prior to the accident occurring. If dryout is precluded prior to trip, sheath temperatures remain low and fuel failures will not occur. If pre-trip dryout does occur and sheath temperatures increase significantly, then fuel failures could occur if the dryout is prolonged. For these cases, detailed fuel analysis may be required to determine if fuel failures are precluded.

CNSC Consultative document C-144 (Reference 5) states the trip parameter acceptance criteria as the following:

a) The primary trip parameter on each shutdown system shall prevent the onset of intermittent fuel sheath dryout.

b) The secondary or backup trip parameters on each shutdown system shall prevent:

    1) Fuel sheath temperature from exceeding 600°C

    2) The duration of post-dryout operation from exceeding 60 seconds.

The C-144 criteria are sufficient, but not necessary to preclude fuel failures. The criteria will be used as trip parameter performance targets for Class 1 events; however detailed fuel analysis may be performed, if necessary, to demonstrate fuel integrity for Class 1 events. The criteria that must be met to ensure fuel integrity are given in Section 2.1.1.

For Class 2 events such as small LOCA and secondary pipe break outside containment, fuel failure shall be prevented. The criteria that must be met to ensure fuel integrity are given in Section 2.1.1.

For Class 3 events, fuel failures shall be limited.

2.1.1 Fuel Integrity Acceptance Criteria

The acceptance criteria for fuel integrity, as described in Reference 2, are:

- If fuel sheath dryout or flow stratification in the channel can be shown not to occur, fuel failures will not occur. In such a case, fuel sheath temperatures would remain near normal operating conditions (< 400ºC).
- If dryout or flow stratification does occur, but fuel sheath temperatures remain below 800ºC, then fuel failures are precluded and no detailed fuel failure analysis is required. Experiments have shown that at temperatures lower than 800ºC the fuel will not fail before 10 hours in dryout conditions.
- If fuel sheath temperatures are predicted to be above 800ºC then detailed fuel analysis is required. This detailed fuel analysis will show the fuel sheath remains intact if all of the following conditions are satisfied:

a) No fuel centreline melting. A fuel element will not fail due to volume expansion causing excessive sheath strain if centreline temperature remains below melting.

b) No excessive strain. A fuel element will not fail due to excessive sheath strain if, for sheath temperatures less than 1000ºC, the uniform sheath strain remains less than 5%.

c) No significant cracks in the surface oxide. A fuel element will not fail due to significant cracks in the surface oxide if, for sheath temperatures greater than 1000ºC, the uniform sheath strain remains less than 2%.

d) No oxygen embrittlement. A fuel element will not fail due to oxygen embrittlement if oxygen concentration remains less than 0.7 wt% over half the sheath thickness.

e) No sheath failure as a result of beryllium-braze penetration at bearing pad or spacer pad locations.

f) No fuel sheath melting.

2.2 Heat Transport System Overpressurization

The heat transport system pressure must remain within the ASME pressure limits for each event. The events are classified according to their frequency and the shutdown system credited. This is documented in the Safety Basis document for ACR (Reference 1).

A moderately frequent event has service limits of level B when the first shutdown system is credited, and level C when the second shutdown system is credited. A low frequency event has a service limit of level C for the first shutdown system and level D for the second.

For a level B or "upset" transient, the limit is 110% of design pressure.

For level C or "emergency" conditions, the analysis target for peak pressure is to remain below 120% of design pressure. For any emergency condition, the allowable primary membrane stress intensity limit is the greater of 120% of the allowable design stress intensity (tabulated S_m value) or 100% of the tabulated yield strength at temperature. However, an analysis target of 120% of design pressure is used to ensure that the above ASME limits are not exceeded.

For level D or "faulted" conditions, the same analysis target as for level C is conservatively used to ensure heat transport system integrity.

Table 2–2 is a list of the trip coverage events that are analysed for overpressure protection, including the service limit for each event.

Table 2–2
Overpressure Protection Requirements for Major ACR Events

Event	Service level for first shutdown system	Service level for second shutdown system
Loss of Class IV power	Level B "upset"	Level C "emergency"
Single pump trip	Level B	Level C
Single HTS pump seizure	Level C	Level D "faulted"
Loss of reactivity Control	Level B	Level C
Loss of pressure and inventory control	Level B	Level C
Loss of Feedwater Flow	Level B	Level C
Feedwater Line break	Level C	Level D

3. SYSTEM ASSUMPTIONS

This section describes the basic assumptions regarding the operation of the process and safety systems following the postulated trip coverage event. Details of the system component behaviour are included in the discussion if they have an impact on the results. The assumptions discussed here are generic to most trip coverage events. System assumptions for individual events, where they deviate from the generic, are discussed with the event sequence in a later section. Unless otherwise stated, these system components are assumed to continue to function normally. More detailed descriptions of the systems and assumptions are contained in Reference 2 and are not repeated in this document.

Operational parameters are chosen to be conservative within the range allowed by design. An allowance will be made for operational flexibility (e.g. parameters in operation deviating from design values). These values or initial conditions are documented in Reference 2.

Section 3.1 describes the basic, common assumptions regarding the operation of the process systems, including the heat transport system, the pressure and inventory control system, and steam and feedwater system; it also includes the reactor regulating system.

Section 3.2 describes the assumptions regarding the operation of the shutdown systems.

3.1 Process and Control Systems

All the HTS as well as pressure relief functions are considered in the trip coverage assessment. When evaluating the overpressure protection provided by the shutdown systems, the availability or unavailability of the HTS relief valves is considered.

The trip coverage assessment, in general, assumes that process and control systems function normally. However, where the initiating event can have a direct effect on the process or control system, failure of the relevant system is considered. In addition, failures of normally inactive components to respond when called upon to mitigate the consequences of the event are also considered, when such failures can have a significant adverse effect on the outcome for the initiating event.

Overall unit control is affected by the interaction of the major control programs. The function is to match the reactor power level and steam load requirements while maintaining steam generator drum pressure at its setpoint value, for both constant power level operation and during load manoeuvring. The overall unit control system controls the unit load, steam generator pressure and reactor power by adjusting:

a) Demand reactor power
b) Turbine load reference setpoint, and
c) Opening of the ASDVs and CSDVs.

Four distinct modes of reactor power setpoint control are employed by the unit control system:

a) Normal Mode

   The turbine load is set or changed as desired by the unit control and reactor power is controlled by the Reactor Regulating System (RRS) to a power setpoint determined to maintain the steam pressure constant.

b) Alternate Mode

The reactor power is controlled by the RRS to a specified setpoint. The plant loads are adjusted by the steam generator pressure control to maintain steam generator drum pressure, i.e. the turbine follows the reactor. This is the preferred mode for low reactor powers, or for upset conditions in which the reactor power cannot follow the variations in the load.

c) Setback

Reactor power is ramped down at a controlled rate to bring an out-of-limit process parameter back within limits.

d) Stepback

Reactor power is promptly reduced when certain variables are outside their acceptable range.

3.1.1 Reactor Regulating System

Trip coverage analysis is normally performed with the reactor in alternate mode, i.e. controlled to a fixed setpoint. In addition, the setback and stepback functions are normally not credited in the analysis, unless the action can potentially delay or mask a trip.

The reactor regulating system is normally modelled to perform its function of maintaining reactor power at the setpoint. If warranted, RRS frozen cases are also considered. In these cases, reactivity changes resulting from the accident or event result in changes to reactor power without compensation by the RRS system.

3.1.2 Heat Transport Pressure and Inventory Control System

The HTS pressure and inventory control system (P&IC) consists of a pressurizer, bleed condenser, and the feed and bleed circuit. The P&IC system is designed to provide a means of pressure and inventory control for the HTS system, as well as to provide adequate overpressure protection. The pressurizer is connected to the HTS at one of the reactor outlet headers. Feed and bleed flow provides inventory control to the pressurizer. A connecting line from the reactor inlet header provides spray flow at the top of the pressurizer to assist in pressure control. Heaters are provided at the bottom of the vessel to compensate the heat losses and to maintain a steady temperature in the pressurizer. The bleed condenser receives a bleed flow from the HTS, which is mixed with the continuous bleed condenser spray flow from the discharge of the HT pressurizing pump, to maintain a steady fluid temperature in the bleed condenser. The P&IC system is modelled in the analysis and is credited or failed depending on the specific scenario.

Two instrumented Liquid Relief Valves (LRVs) provide overpressure protection for the heat transport system main circuit. The instrumentation is a part of shutdown system one. During reactor operation, these LRVs open on a high-pressure signal and discharge into the bleed condenser. Since instrumentation is shared with SDS1, the LRVs are credited in the SDS1 trip coverage analysis; however only one of the two valves is credited. When assessing the effectiveness of SDS2, both cases with and without the LRVs are considered in the analysis.

3.1.3 Steam and Feedwater Systems

The steam produced by the two steam generators is fed, by four steam lines, to the main steam header. Each line has a main steam isolation valve (MSIV), an atmospheric steam discharge valve, and two main steam safety valves (MSSVs). The two condenser steam discharge valves

(CSDVs), located on the main steam header, are sized to permit continued reactor operation following a loss of line or turbine trip.

Most of the trip coverage analysis does not credit the CSDVs or ASDVs unless their operation can mask a trip. The MSSVs are normally credited; the exact number credited is determined from reliability considerations. Normally 4 of the 8 will conservatively be credited for most overpressure events.

The main feedwater system consists of three 50% main feedwater pumps providing feedwater through the feed heating system and to the steam generators through the main feedwater lines. There are two auxiliary feed pumps powered by Class III power. The steam generator level control program normally controls the position of the feedwater valves. The feedwater train is modelled in the thermalhydraulic analysis. Depending on the scenario, the feedwater system is normally credited.

3.2 Shutdown Systems

The shutdown systems act independently of one another. However, credit for simultaneous operation of the two shutdown systems is not taken; a reactor shutdown by only one of the shutdown systems is assumed.

Shutdown System 1 (SDS1) contains 20 mechanical shutoff units and is designed to insert a minimum of -25 mk in less than two seconds after actuation. In assessing the effectiveness of SDS1, the 2 most effective rods are assumed to be unavailable.

Shutdown System 2 (SDS2) consists of liquid gadolinium injection tubes, traversing the calandria in the upper and lower reflector regions. The SDS2 system is designed to inject enough gadolinium (concentrated gadolinium nitrate solution) to blanket the upper and lower reflector region in less than 2 seconds after actuation. This is equivalent to a minimum of -25 mk in less than 2 seconds. SDS2 contains enough gadolinium to guarantee a system reactivity that is lower than –150 mk after thorough mixing of the gadolinium within the entire moderator system. This is more than sufficient to keep the reactor shutdown after an accident. SDS2 trip effectiveness is assessed assuming that the most effective nozzle is out of service.

3.2.1 SDS1 and SDS2 Trip Setpoints

The SDS1 and SDS2 trip parameters are listed in Table 3–1.

Uncertainties are also discussed in the following sections, along with timing issues. These are also subject to confirmation and are to be considered preliminary.

3.2.1.1 High Neutron Power

Flux detectors are provided for each shutdown system for the high neutron power (or Regional Overpower, ROP) trip. The detector units are separate for each shutdown system and separate from the reactor regulating system.

Under normal conditions, the reactor trip signal will be initiated when the design trip setpoint is reached and detected by the flux detectors. However, because the detectors are calibrated such that they indicate the product of current reactor power and channel power peaking factor (CPPF),

the effective setpoint is lower than design trip setpoint. The effective setpoint will be used in the analyses. The CPPF is not yet finalized for the ACR, but will likely be 5%.

The ROP trip setpoint is designed to maintain channel powers below their critical channel powers (CCP) and is determined primarily by the very slow loss of reactivity control analysis. This is also an important trip for the fast loss of reactivity control event.

3.2.1.2 High Rate Log Neutron Power

Three fission chambers and ion chambers are provided for SDS1 and SDS2 respectively. The output current from each fission chamber/ion chamber goes to an amplifier which produces logarithmic neutron power, linear neutron power, and rate logarithmic signals, of which only the last is used as a direct trip parameter. The log neutron power is used as a conditioning parameter for various process trips.

The log rate trip can provide trip coverage for fast loss of reactivity control events.

An allowance will be added to the nominal trip setpoint to cover measurement loop uncertainties, including drift, and noise as well.

3.2.1.3 High Heat Transport System Pressure

HTS pressure is measured in three widely separated locations on each of the reactor outlet headers, for each shutdown system. This instrumentation provides the high-pressure trip signals and, for SDS1, it also provides the signal for the HTS relief valves (LRVs). The same pressure transmitters are also used for HTS low-pressure trips.

The HTS high-pressure trip provides coverage for the loss of flow events as well as for the fast loss of reactivity control events.

There are two HTS high pressure trip signals. The first one is the prompt trip with a higher trip setpoint. The second trip signal has a relatively lower trip setpoint with a few seconds delay. This trip signal is implemented to provide trip coverage in the event of a single HTS pump trip. An allowance of 0.1 MPa will be added to the nominal trip setpoint to account for measurement errors, transducer accuracy and drift, and current-to voltage conversion.

3.2.1.4 Heat Transport System Low Pressure

The HTS low pressure trips use the same instrumentation as the HTS high-pressure trips. The nominal trip setpoints are the same for both SDS1 and SDS2. Again, an allowance of 0.1 MPa is used in the analyses to cover errors and uncertainties. There is also a 2% allowance for power measurement. The trips are conditioned out when the fission/ion chamber log power measures below 0.1%FP for SDS1 and 0.3%FP for SDS2. The setpoint is ramped dependent on the fission/ion chamber linear power.

The HTS low-pressure trip provides coverage for LOCAs as well as for pressure and inventory control failures.

3.2.1.5 Heat Transport Low Flow

Heat transport system flow is measured in three channels per core pass per shutdown system, for a total of six instrumented channels for each shutdown system for the ACR-700.

The low flow trip provides coverage for loss of flow events, as well as for large and small LOCAs.

An uncertainty of approximately –0.5 kg/s will be applied, which includes 0.3 kg/s for the flow variation due to power ripple and 0.2 kg/s for instrumentation uncertainty.

The trip is conditioned out when the fission/ion chamber log power measures below 0.1%FP.

3.2.1.6 Pressurizer Low Level

The low pressurizer level (on both SDS1 and SDS2) provides protection for LOCAs as well as for some loss of pressure control scenarios resulting in a decrease in pressure in the HTS.

The trip setpoint is the same for SDS1 and SDS2. It is a function of power (dependent on the in-core flux detectors). The analysis setpoint is reduced by 0.6 m compared to the nominal setpoint to account for uncertainties in the ROH pressure measurement, RIH temperature measurement, and pressurizer level and temperature measurements. For breaks inside containment, the pressurizer level measurement may be affected by the increase in ambient temperature after the break. The resulting bias to the trip setpoint will be calculated and accounted for in the analysis. The analysis setpoint will also include a 2% error allowance for the power measurement.

The trip is conditioned out when the fission/ion chamber log power measures below 0.1%FP for SDS1 and 0.3%FP for SDS2.

3.2.1.7 Steam Generator Low Level

The steam generator low level trip, on both shutdown systems, provides trip coverage for secondary circuit failures, including breaks in the steam and feedwater system, and loss of feedwater flow. The same trip setpoint is used for both shutdown systems. All measurements will be in the steam drum. The trip setpoint is a function of power, dependent on the fission/ion chamber log power and is conditioned out below 1%FP for SDS1 and 2%FP for SDS2.

The analysis trip setpoint will include an uncertainty in level measurement to include the effect of transmitter drift and calibration accuracy. For breaks inside containment, the steam generator level measurement may be affected by the increase in ambient temperature after the break. The resulting bias to the trip setpoint will be calculated and accounted for in the analysis.

3.2.1.8 High Moderator Level

The high moderator level trip provides coverage for a loss of service water to the moderator heat exchangers, loss of moderator flow and in-core breaks. Without cooling, the moderator would heat up, boil, and burst the relief duct rupture disks. The trip setpoint is reached prior to the loss of moderator through the rupture disks.

3.2.1.9 Low Moderator Level

The low moderator level trip provides coverage for a loss of moderator event. The nominal trip setpoint is chosen to prevent the uncovering of any channels. Instrumentation for the high and low moderator level trips is shared as both trips are not required for the same event.

3.2.1.10 High Reactor Building Pressure

A triplicated measurement of reactor building pressure is provided for each shutdown system. The high building pressure trip provides coverage for a number of events, including large and small LOCA, as well as steam line breaks. An uncertainty allowance of 0.4 kPa for both shutdown systems will be included in the analysis.

3.2.1.11 Manual Trip

There is provision for a manual trip which can be used as a backup trip for some of the slower transients. In cases where a manual trip is credited, there must be a clear indication of the event and sufficient time to credit the operator action. Normally a triplicated alarm would be required as a sufficient indication and at least 15 minutes following the alarm is required before this backup trip can be credited.

3.3 Timing of Trip Initiation

In addition to uncertainties in the trip setpoint, timing delays are also accounted for in the trip coverage analysis. For most trips, there is a measurement time constant to account for the response time of the instrumentation (e.g. pressure transmitters), as well as a fixed delay, to account for trip logic processing, including relays.

For most trip parameters, a measurement time constant of 300 ms will be allowed for pressure measurements. The fixed delays will include an allowance for the trip computers, sonic delays in the instrument tubing, and delay in the relays. An additional allowance will be added, bringing the total fixed delay to 300 ms for all process trips.

Table 3–1
ACR Shutdown System Trip Parameters

Trip Parameters
High Neutron Power (ROP)
High Log Rate
HTS High Pressure
HTS Low Pressure
HTS Low Flow
Pressurizer Low Level
Steam Generator Low Level
Moderator High Level
Moderator Low Level
Reactor Building High Pressure

4. ANALYSIS METHODOLOGY

The analysis of trip parameter effectiveness is performed using a number of computer codes which simulate various aspects of the neutronics, heat transport system, steam and feedwater system, moderator system, and containment. The main computer code used for trip coverage analysis is CATHENA. However, POINTSIM may also be used for loss of reactivity control from shutdown. GOTHIC is used for all containment analysis, which is covered in a separate Analysis Basis document.

Reactor physics calculations are performed using the point kinetics model in CATHENA, which is coupled with the thermalhydraulics. For some loss of reactivity control transients, from shutdown conditions, POINTSIM may also be used. Both the main circuit as well as slave channel analysis will be performed with CATHENA.

4.1 Reactor Physics

The neutron power calculation is performed using the point kinetics model included in CATHENA. Six delayed neutron groups are used. Reactivity feedback from coolant density and fuel temperature changes are modelled for the relevant scenarios.

4.2 Thermalhydraulics

Thermalhydraulic analysis determines the HTS and the secondary side responses to the trip coverage event, modelling and predicting the important process parameters (e.g. HTS flow, temperature, pressure, etc.) This, in turn, determines the timing of the reactor trip.

The thermalhydraulic analysis is conducted with CATHENA. CATHENA is a non-equilibrium, two-fluid code where mass, momentum, and energy conservation equations are solved individually for each phase. CATHENA version 3.5d rev 0 is used in the thermalhydraulic analysis.

The reference ACR-700 model is used, with the addition of the point kinetics data. A simplified pressure and inventory control system model has also been added.

To assess fuel performance in more detail, a single channel analysis is performed. This channel is based on a high-powered channel, with the licensing limit for the bundle power and channel power. It is used to predict the onset of dryout for the various scenarios and to evaluate the fuel behaviour. If excessive fuel sheath temperatures are predicted, a more detailed fuel analysis will be performed.

A single channel model of the instrumented channels will also be developed to determine the effectiveness of the HTS low flow trip.

5. TRIP COVERAGE EVENTS

The following sections summarize the trip coverage issues for each event. A brief description of the event sequence is given, along with specific systems assumptions and analysis methodology, if it deviates from the previous section. The classification of the events is shown in Reference 1 and summarized in Table 2–1.

5.1 Loss of Flow

This category of events includes a total and partial loss of Class IV power, a single HTS pump trip, and a single pump seizure.

5.1.1 Loss of Class IV Power

Both a total loss of Class IV power as well as loss of a single bus are considered in the analysis. Both events are classified as Class 1. All equipment powered by the Class IV buses are lost, but the immediate consequence is the loss of power to the HTS pumps, which leads to pump rundown and a reduction in the effectiveness of fuel cooling.

This maximum pressure during this event must be below the appropriate overpressure limits (level B service limits). In addition, fuel failures must be precluded.

Feedwater pumps are also tripped, as they are powered by Class IV power.

The Liquid Relief Valves (LRVs) are credited for SDS1 analysis, as they have shared instrumentation. For SDS2, analysis is performed with and without crediting the LRVs.

Analysis will be performed both crediting the RRS system to maintain reactor power and with RRS assumed frozen.

The important trip parameters for this event are:

- High HTS pressure
- Low coolant flow

5.1.2 Single Pump Trip

For a single pump trip, power is lost only to the affected pump. This is classified as a Class 1 event. All other HTS pumps continue to operate and the tripped pump freewheels due to the flow produced by the other pumps.

This maximum pressure during this event must be below the appropriate overpressure limits (level B service limits). In addition, fuel failures must be precluded.

Analysis will be performed both crediting the RRS system to maintain reactor power and with RRS assumed frozen.

The important trip parameters for this event are:

- High HTS pressure
- Low coolant flow

5.1.3 Single Pump Seizure

For a single pump seizure, a mechanical failure is postulated to occur within the pump. This is a Class 3 event. This results in a rapid loss of pump head (assumed over 3 seconds). The other HTS pumps continue to operate but the seized pump provides a significant flow resistance.

This maximum pressure during this event must be below the appropriate overpressure limits (level C service limits). The target is to prevent fuel failures for this Class 3 event.

Analysis will be performed both crediting the RRS system to maintain reactor power and with RRS assumed frozen.

The important trip parameters for this event are:

- High HTS pressure
- Low coolant flow

5.2 Loss of Reactivity Control

In the loss of reactivity control scenario reactivity insertion is presumed to be caused by a malfunction of the reactor regulating system. This is classified as a Class 1 event. A range of linear insertion rates, up to the physical limits of the system are considered, from the full range of initial reactor powers. The faster reactivity insertion rates will result in an increase in pressure.

This maximum pressure during this event must be below the appropriate overpressure limits (level B service limits). In addition, fuel failures must be precluded.

The important trip parameters for this event are:

- High HTS pressure
- High neutron power
- High log rate

5.3 Loss of HTS Pressure and Inventory Control

Both pressurization and depressurization events are considered for loss of inventory or loss of pressure control. The specific failures include failures of the feed and bleed valves, failures of the LRVs, steam bleed, and pressurizer heaters. This is classified as a Class 1 event.

For many of the scenarios, trips are not required because the system reaches a new steady state, without exceeding overpressure limits or jeopardizing fuel cooling. For the pressurization events, the level B limits must be met. Prevention of fuel failures is also a criterion.

Analysis will be performed both crediting the RRS system to maintain reactor power and with RRS assumed frozen.

The important trip parameters for this event are:

- High HTS pressure
- Low HTS pressure
- Low coolant flow

- Low pressurizer level

5.4 Small LOCA

To provide a composite trip coverage assessment for small breaks, the effectiveness of individual trip parameters is assessed for a range of break sizes over the entire range of operating powers. Break sizes up to the discharge rate equivalent to a guillotine break of the largest feeder are considered. The break is normally modelled in the reactor inlet header, because this location proves to be the most challenging for fuel cooling.

Other specific breaks are also analysed, including breaks at the top of the pressurizer (which effectively disables the low pressurizer level trip), in-core breaks into the moderator system, and steam generator tube ruptures.

The classification of different small LOCA events is shown in Table 2–1.

Analysis will be performed both crediting the RRS system to maintain reactor power and with RRS assumed frozen.

The important trip parameters for this event are:

- High reactor building pressure
- Low HTS pressure
- Low coolant flow
- Low pressurizer level
- Moderator high level (for in-core LOCAs only)

5.5 Large LOCA

To provide a composite trip coverage assessment for large breaks, the effectiveness of individual trip parameters is assessed for the entire range of break sizes, from the largest of the small LOCAs to a 100% guillotine break of the largest pipe in the heat transport system, over the entire range of reactor powers.

Trip coverage analysis for large LOCA is performed for the initial period of the transient and supports the refill analysis, which is described in the thermalhydraulics analysis basis document.

The assessment of the effectiveness of the reactor building high-pressure trip is covered in the containment analysis. The analysis is performed to demonstrate that the maximum containment pressure is below the reactor building design pressure.

The effect of RRS operation on trip parameters and timing will be assessed.

The important trip parameters for this event are:

- High reactor building pressure
- Low HTS pressure
- Low coolant flow
- Low pressurizer level

5.6 Feedwater Failures

Feedwater failures include breaks, at various locations both inside and outside containment, and loss of feedwater flow. The classification of various feedwater system failure events is shown in Table 2–1.

Three types of feedwater line breaks are considered:

- Symmetric breaks, outside containment, affecting both steam generators equally
- Asymmetric break between the control valve and the check valve in the feedwater line (outside containment)
- Asymmetric breaks downstream of the check valve in the feedwater line (inside containment)

In addition, feedwater line failures are also considered. These include:

- Loss of feedwater pumps
- Feedwater valve closure

The main objective for these events is to ensure that an adequate heat sink is provided throughout the event. After a feedwater line failure, an alternate heat sink must be established. For ACR, in order to reduce the reliance on the operator, the target is to bring in the alternate heat sink automatically.

For feedwater line breaks downstream of the check valve, the reactor power will increase slightly before reactor trip, due to the negative void feedback for ACR-700. The effect of RRS operation will be assessed.

The trips considered for these events are:

- Low steam generator level
- High reactor building pressure
- HTS high pressure

5.7 Steam Line Breaks

Trip coverage assessment is performed for breaks in the main steam lines, both inside and outside the reactor building. A break in the piping between the boilers and turbine can lead to a rapid loss of secondary circuit inventory. If the discharge is within the reactor building, it would cause a rise in the containment pressure. A range of break sizes is assessed up to a complete guillotine break. The classification of various steam line break events is shown in Table 2–1.

The timing of the reactor building high-pressure trip and the assessment of the maximum pressure within containment are covered in the containment analysis basis document.

Three break locations are considered:

- Symmetric, outside containment, affecting both steam generators equally
- Asymmetric, outside containment, affecting one steam generator
- Breaks inside containment

The major criteria for these cases are to maintain containment integrity by preventing excessive overpressure and to provide an automatic backup heat sink. For steam line breaks, the reactor

power will increase slightly before reactor trip, due to the negative void feedback for ACR-700. The effect of RRS operation will be assessed.

The trips considered for this event include:

- Low steam generator level
- High HTS pressure
- High reactor building pressure

5.8 Loss of Secondary Circuit Pressure Control

Both pressurization and depressurization events are considered. A loss of condenser vacuum can result in a system pressurization which is relieved by the atmospheric discharge valves and main steam safety valves. For this event, fuel cooling is normally not in jeopardy and trips are not always required to terminate the event. Instead, the relief capacity of the system is tested and confirmed. This is classified as a Class 1 event.

For depressurization events, such as inadvertent opening of the ASDVs or MSSVs, releases must remain within the dose limits for the event.

These scenarios are analysed to show that the events are controlled and terminated with little effect on the heat transport system conditions. The effect of RRS operation will be assessed.

5.9 Moderator System Failures

Moderator failures include the following events:

- Loss of service water to the moderator heat exchanger
- Loss of moderator flow (loss of pumps)
- Moderator system pipe break

The classification of various moderator system failure events is shown in Table 2–1. The immediate consequence of a loss of service water to the moderator heat exchangers is the loss of moderator heat sink, which leads to heat up of the moderator fluid. A similar response occurs after a loss of moderator pumps. The moderator level increases and boils, and, if unterminated, will result in a bursting of the rupture discs in the relief ducts. The moderator level then collapses. The moderator high-level trip provides coverage for this event. ROP or moderator low level then becomes the backup trip.

For the pipe break in the moderator system, the moderator level drops and eventually uncovers channels in the reactor. Eventually the reactor goes into self-shutdown after the loss of moderator. The low moderator level trip provides coverage for this event, along with ROP.

The trip parameters for these events are:

- High moderator level
- Low moderator level
- High neutron power

5.10 Shield Cooling Failures

Shield cooling failures include loss of flow (Class 1) and breaks (Class 2) in the shield cooling system. This loss of heat sink can result in thermal transients in the calandria, particularly across the tubesheet, which could in turn result in deformation in the calandria assembly. The analysis is performed to ensure that there is sufficient time for operator intervention before the integrity of the fuel channels is jeopardized.

6. REFERENCES

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