AECL EACL

Analysis Basis

INITIAL CONDITIONS AND STANDARD ASSUMPTIONS SAFETY ANALYSIS BASIS

ACR-700

10810-03510-AB-001

Revision 0

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1. INTRODUCTION

This Analysis Basis (AB) report presents the acceptance criteria and the major plant system assumptions to be used in the safety analysis of the ACR-700™* design. The assumptions pertain to the operating state before an event and to the plant response after a postulated event, but are not specific to any particular analysed event. The use of this document by safety analysts and designers ensures a consistent, well-supported approach to all the safety analysis in modelling the plant response to postulated accidents.

This report is part of a hierarchy of documents, which define the event classification for the ACR-700 design, as well as the methodology and assumptions to be used in the safety analysis. Within this hierarchy, the Safety Basis [1] is a high-level document, which describes and explains the event classification and overarching acceptance criteria and performance targets. This AB makes use of the Safety Basis, as well as the reference design [2] and existing safety analysis practices to establish the acceptance criteria and system assumptions. Analysis discipline-specific ABs will be prepared to define the methodology, models and codes to be used in the ACR-700 safety analyses.

The three main sections covered by this report are: Acceptance Criteria, Trips, and System Assumptions. The acceptance criteria section defines all of the criteria, which could apply to the analysis of a given event. The basic criteria are given in the Safety Basis [1] covering safety analysis. This section also describes more detailed criteria and analysis targets, which are used to demonstrate compliance with the acceptance criteria.

The trips section covers the main automatic plant responses, which may apply to an analysis event. The four safety systems, as well as the main safety related systems are covered. The treatment of uncertainties in trip setpoints and in trip delay times is explained and justified.

The system assumptions section presents key features and data about the main plant systems, which are modelled in safety analysis. The electrical and water supplies required by each system in order to operate are identified. The data is given in terms of initial conditions and standard assumptions. The initial conditions indicate pre-accident state of the system, and the standard assumptions include any assumptions on system performance during an accident. The approach to accounting for system data uncertainties is described, and specific design and analysis values are provided for the key system data.

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* ACR-700™ (Advanced CANDU Reactor™) is a trademark of Atomic Energy of Canada Limited (AECL).

2. ACCEPTANCE CRITERIA

The purpose of this section is to bring together all of the acceptance criteria that must be met when doing safety analysis for the ACR-700 design. The acceptance criteria are presented here, consolidated into this document, in order to ensure consistency as well as to provide clear documentation for the criteria.

- The overall requirements that must be met, in order that the safety analysis is considered acceptable, are derived from the Safety Basis for ACR [1]. The objective of this section of the AB is to provide analysis targets, which, if met by the safety analysis results, demonstrate that the acceptance criteria in Reference [1] are met. 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. The approach given in Reference [1] is based on a 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. Reference [1] meets this objective by providing a system of classification of events into five classes:
  - Classes 1,2 & 3 - Design Basis Events
  - Classes 4 & 5 – Limited Core Damage Accidents
  - Severe Core Damage Accidents

A key feature of the approach taken in Reference [1] is that the radiological dose limits associated with the five event classes are derived from and are the same as the radiological dose limits for the corresponding five event classes in CNSC Consultative Document C-6 Rev. 1 [3]. The acceptance criteria and targets adopted for the event classes are based on safety margins increasing with the likelihood of the events in a class. Analysis assumptions and methods are chosen to provide a good balance between the need for conservatism at the higher event likelihood end of the classification and the reasonableness of a more design centred assessment at the lower likelihood end.

Classes 4 and 5 generally include combinations of events, in particular combinations of initiating events and the total failure of a safety system. The Severe Core Damage Accident category is beyond the scope of the deterministic safety analysis, and will be treated in the level 2 PSA.

Design Basis Events (initiating events) are events, which must be accommodated by the plant design within specified limits of the 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 safety analysis of design basis events will include consideration of a single component failure in a mitigating system in addition to other conservative assumptions.

Limited Core Damage Accidents are more improbable events beyond the design basis, which must be accommodated within specified radiological dose limits to the public. Targets on the performance of the barriers against the release of radioactivity may be set to facilitate meeting the dose limits. The plant response to limited core damage accidents is analysed using design centred assumptions and detailed models.

Table 2-1 and Table 2-2 below tabulate a list of Design Basis Events and Limited Core Damage Accidents to be analysed during the pre-engineering phase in accordance with Reference [1]. These tables indicate reclassification, compared with Reference [3], of some design basis events such as heat transport pump seizure, pressure tube rupture, steam generator tube rupture and main steam line break inside containment. These reclassifications are based on the probabilistic considerations and operating experience (Canadian and international). In addition, a distinction has been made between partial flow blockage with no fuel melting and severe flow blockage leading to fuel melting, based on low probability of a large blockage size. A similar distinction was made regarding stagnation and off-stagnation feeder breaks. A Systematic Plant Review [4] will be performed to identify all of the initiating events, which need to be addressed in the ACR-700 design.

Table 2-1 gives the acceptance criteria and performance targets defined in Reference [1]. The application of these criteria to the safety analysis is discussed in more detail in the following subsections. The key features of the proposed approach to the safety analysis are summarized in Table 2-4.

Table 2-1
List of Design Basis Events

Design Basis Events	Class
Large LOCA	3
Large LOCA + LCIVP	3
Small LOCA	2
Small LOCA + LCIVP	3
Pressurizer Pipe Break	2
Pressurizer Pipe Break + LCIVP	3
Single SG Tube Rupture	2
Single SG Tube Rupture + LCIVP	3
Off-Stagnation Feeder Break	2
Off-Stagnation Feeder Break + LCIVP	3
Partial Single Channel Flow Blockage	2
Partial Single Channel Flow Blockage + LCIVP	3
PT Failure (CT intact)	2
PT/CT Failure	3
PT/CT Failure + LCIVP	3
Loss of Reactivity Control	1
Total LCIVP	1
Partial LCIVP	1
Single HTS Pump Trip	1
Pump Seizure	3
Main Steam Line Break (inside containment)	3
Main Steam Line Break (inside containment) + LCIVP	3
Feedwater Line Failure (inside containment)	3
Feedwater Line Failure (inside containment) + LCIVP	3

Table 2-2
List of Limited Core Damage Accidents

Limited Core Damage Accidents	Class
Large LOCA + LOECC	5
Small LOCA + LOECC	5
Pressurizer Pipe Break + LOECC	5
Single SG Tube Rupture + LOECC	5
Multiple SG Tube Failure	5
Off-Stagnation Feeder Break + LOECC	5
Stagnation Feeder Break	5
Partial Single Channel Flow Blockage + LOECC	5
Severe Flow Blockage	5
End Fitting Failure + LOECC	5
Main Steam Line Break (inside containment) + LOECC	5
Feedwater Line Failure (inside containment) + LOECC	5

Table 2-3 tabulates the acceptance criteria to be used to analyse the events listed in Table 2-1 and Table 2-2.

Table 2-3
Acceptance Criteria for Design Basis Events and Limited Core Damage Accidents

<table>
<tr><th>Event Class</th><th>Acceptance Criteria</th><th>Performance Targets</th></tr>
<tr><td colspan="3">CLASS 1</td></tr>
<tr><td>Dose</td><td>C-6 Rev. 1 Class 1 limits</td><td></td></tr>
<tr><td>Fuel</td><td>No calculated failures</td><td></td></tr>
<tr><td>Fuel Channel</td><td>No PT failures</td><td></td></tr>
<tr><td>Overpressure/pressure Boundary Integrity</td><td>Level B limit for SDS1 to trip, level C limit for SDS2 to trip</td><td></td></tr>
<tr><td colspan="3">CLASS 2</td></tr>
<tr><td>Dose</td><td>C-6 Rev. 1 Class 2 limits</td><td></td></tr>
<tr><td>Fuel</td><td>No calculated failures</td><td></td></tr>
<tr><td>Fuel Channel</td><td>No PT failures (in non-affected channels)</td><td></td></tr>
<tr><td>Overpressure/pressure Boundary Integrity</td><td>Level C limit for SDS1 to trip, level D limit for SDS2 to trip</td><td></td></tr>
<tr><td>Containment</td><td>Peak pressure not to exceed design pressure</td><td></td></tr>
<tr><td colspan="3">CLASS 3</td></tr>
<tr><td>Dose</td><td>C-6 Rev. 1 Class 3 limits</td><td rowspan="5">No significant plastic deformation of PTs [1]</td></tr>
<tr><td>Fuel</td><td>Limited failures</td></tr>
<tr><td>Fuel Channel</td><td>No PT failures (in non-affected channels)</td></tr>
<tr><td>Overpressure/pressure Boundary Integrity</td><td>Level C limit for SDS1 to trip, level D limit for SDS2 to trip</td></tr>
<tr><td>Containment</td><td>Peak pressure not to exceed design pressure (except for Main Steam Line Break-MSLB-accident)
For MSLB accident maintain containment integrity
Hydrogen concentration to remain below flammability limit</td></tr>
<tr><td colspan="3">CLASSES 4 and 5</td></tr>
<tr><td>Dose</td><td>C-6 Rev. 1 Class 4 and 5 limits respectively</td><td></td></tr>
<tr><td>Fuel</td><td></td><td>For all events no fuel centreline or sheath melting (in non-failed channels)</td></tr>
</table>

Event Class	Acceptance Criteria	Performance Targets
Fuel Channel		No fuel channel failures (in non-affected channels). Ensure sufficient moderator subcooling if PT sags into contact with CT
Containment		Peak pressure not to exceed design pressure (except for main steam line break accident) Structural integrity of containment walls ensured to a degree that consequential damage to reactor systems could not result Hydrogen concentration to remain below Deflagration-Detonation-Transition (DDT) limit

As an acceptance criterion, the calandria has to remain intact for all of the postulated events listed in Table 2-1 and Table 2-2.

Table 2-4
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	Detailed	Performance Criteria – Radiological Doses	Performance Targets
Limited Core Damage Accidents (Classes 4 & 5)	Design-centred	Detailed	Radiological Doses	Performance Targets
Severe Core Damage Accidents	Design-centred	Integral	–	Frequencies of Severe Core Damage and Large Release

See Reference [1] Safety Basis for ACR, 108-03600-AB-003, AECL July 2003

2.1 Public Dose Limits

The dose resulting from Class 1 to 5 events shall be no more than the limits presented in Table 2-5 [1]. According to CNSC document C-006 (Rev. 1) safety analysis should demonstrate that:

Para 4.2.27 – '…the dose and release limits of Table 4.8 [reprinted as Table 2-5 below] *are not exceeded for the period 30 days after the initiating event.'*

Table 2-5
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	1000	1500
Skin (mSv, averaged over 1 cm^2)	20	200	1200	4000	5000
30 days 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

Safety analysis is required to show that the predicted public dose remain below the limit for the appropriate event class.

2.2 Reactor Shutdown

It must be demonstrated (Reference [5] Section 3.2.1 (a)) that for all postulated event scenarios, each of the shutdown systems could independently shut down the reactor and maintain it subcritical for at least 15 minutes until operator action can be credited (Reference [3] Section 4.2.9 (d)) to further increase the reactor subcricality.

This means that the shutdown systems must have sufficient depth to compensate for any positive reactivity effects resulting from the postulated events, and the net system reactivity following initiation of SDS1 (or SDS2) must remain negative. A safety analysis margin of –5 mk is adopted to cover uncertainties.

2.3 Fuel Integrity

Class 1 to 5 events shall meet the fuel integrity acceptance criteria and performance targets from Reference [1]. For events where fuel failures must be prevented the following sufficient criteria are to be met:

- 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 [6], [7] 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 failure analysis will show the fuel sheath remains intact if all of the following subsidiary conditions are satisfied:
  a) No fuel centerline melting. A fuel element will not fail due to volume expansion causing excessive sheath strain if centerline temperature remains below melting [8].
  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% [9], [10].
  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% [11].
  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 [12].
  e) No sheath failure as a result of beryllium-braze penetration at bearing pad or spacer pad locations [13].
  f) No fuel sheath melting.

CNSC consultative document C-144 [14] is applied to the trip parameter performance targets of Class 1 events as follows:

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.

While C-144 is not yet a formal regulatory document and currently exists as a consultative document only, the targets will be applied to ACR trip coverage analysis of Class 1 events. If these targets are not met, then more detailed fuel analysis would be required to ensure that fuel failures are precluded.

2.4 Fuel Channel Integrity

Class 1 to 5 events shall meet the fuel channel integrity acceptance criteria and performance targets from Reference [1]. The fuel channels shall not fail, except for the affected channel during a single channel event. If a single channel event causes failure, then the failure shall not propagate to other channels. Pressure tube integrity is considered a sufficient, but not a necessary, condition to ensure fuel channel integrity. According to Reference [1] no pressure tube failure shall occur in non-affected channels.

The sufficient criteria and experimental evidence used to assess fuel channel integrity are as follows:

a) No fuel centreline melting, and

b) If the pressure tube temperature remains below 600ºC, pressure tube failure will not occur and no strain calculations are required. This is inferred from experimental work given in [15], [16], [17]. Thus, pressure tube temperatures below 600ºC will ensure that no fuel channel failures occur due to overheating. The results of high temperature creep tests for the ACR-700 pressure tube will be reviewed when they become available, to confirm that this temperature criterion is applicable to ACR-700 safety analysis,

c) If the pressure tube temperature were above 600ºC, then detailed fuel channel assessment would be required [16], [17].

2.5 Containment Integrity

Class 1 to 5 events shall meet the containment integrity acceptance criteria and performance targets from Reference [1]. In order for these requirements to be met, the following criteria must be satisfied:

a) For accidents in which releases may occur (i.e., Large LOCA) the peak pressure must not exceed the reactor building design pressure.

b) In cases where no releases (from the fuel) occur during the accident (i.e., Main Steam Line Break), the reactor building pressure is to remain below the threshold pressure such that loss of structural integrity would not occur.

c) The differential pressure across the reactor building internal walls shall not exceed the design differential pressure. The intent here is to maintain the integrity of support walls inside containment therefore ensuring the integrity of safety related systems. The design differential pressure may be different for each wall, and will be specified with the analysis.

A hydrogen deflagration or detonation could possibly raise internal containment pressures above the design limits. The lower flammability limits for a dry hydrogen-air mixture at atmospheric pressure and temperature are: 4% (vol.) hydrogen for upwards flame propagation, 6% (vol.) hydrogen for horizontal flame propagation, and 9% (vol.) hydrogen for downwards flame propagation [18]. For hydrogen concentrations between 4% and 6%, partial burning of hydrogen may occur. A detonation is not likely to occur until the hydrogen concentration is increased above 9%. The rate and extent of a hydrogen deflagration/detonation also depends on the concentration of steam, the initial temperature, and the degree of turbulence in the gas mixture and is dependent upon an ignition source being present.

The threshold for the Deflagration to Detonation Transition (DDT) not to occur is that the hydrogen concentration inside containment remains below 9.0% by volume.

2.6 Heat Transport System Overpressure Limits

Class 1 to 5 events shall meet the heat transport system overpressure limits from Reference [1]. These limits comply with CNSC documents R-10 and R-77.

Note that the “First Shutdown System” is the first that would normally trip for the event (it may be either SDS1 or SDS2).

For a level B (“upset”) transient, the limit is 110% of design pressure.

For level C (“emergency”) conditions, the analysis target for the peak pressure is to remain below 120% of design pressure [19]. For any emergency condition the allowable primary membrane stress intensity limit is the greatest of 120% of the allowable design stress intensity (tabulated S_m value) or 100% of the tabulated yield strength at temperature, whichever limit is more restrictive. However, a target of 12% of design pressure is used, since peak pressures in excess of 120% would necessitate detailed stress analysis to prove that the absolute limits quoted above are not exceeded.

For level D (“faulted”), the same analysis target as for level C is conservatively used to ensure HTS integrity.

2.7 Calandria Structure Integrity

Class 1 to 5 events shall meet the calandria structure integrity acceptance criteria and performance targets from Reference [1]. Calandria structure integrity is necessary to ensure that the moderator will remain available as a heat sink in the event of a loss of the ECCS. The acceptance criterion follows ASME level C service limits.

2.7.1 Pressure Transient due to Fuel and/or Coolant Discharge

The calandria structure shall remain intact. That is, during an in-core break, the increase in moderator pressure due to the discharge of hot fuel and coolant shall not cause failure of the calandria vessel. This requirement can be satisfied if the following criterion is satisfied.

a) The calandria vessel stresses and deformations will be limited to ensure that there is no gross loss of structural integrity (and thus no significant loss of moderator) and there is no interference with the operation of the shutdown systems. The local yielding of the structure components will be limited to the governing codes and standards, ASME Boiler and Pressure Vessel Code (Reference [19]) and CSA-N285.0 (Reference [20]).

2.7.2 Pressure Transient due to Deuterium Gas Explosion or Deflagration

Calandria integrity can also be challenged by the explosion or deflagration of deuterium gas in the moderator cover gas. A deflagration hazard may result if a combustible mixture of deuterium is generated coincident with the existence of an ignition source.

Gas mixtures will be considered to be combustible if:

a) The concentration of deuterium exceeds the lower limit for which a flame can propagate upwards in a dry mixture of deuterium and oxygen. If the maximum temperature of the gas

mixture could be 100°C higher than room temperature, then the lower limit corrected for temperature would be approximately 4.5 percent, References [21] and [22], and

b) The concentration of steam (D_2O vapour) is less than the inerting concentration, Reference [22].

Table 2-6
Inerting Concentration for Gas Mixture

Gas Mixture	Diluent	Inerting Concentration
Hydrogen-Oxygen	Steam	87%
Hydrogen-Air Steam	Steam	63%

3. TRIPS

The purpose of this section is to address all the trip parameters and describe how they are modelled in safety analysis. In addition to trip setpoints, descriptions of the uncertainties, time delays, time constants and any added allowances that are included to ensure margin between nominal and analysis trip setpoints are included.

This section includes the reactor trip parameters, ECCS initiating and conditioning signal parameters, containment isolation signals, the automatic heat transport pump trip signal and finally the reactor setback and stepback parameters.

3.1 Shutdown Systems

The shutdown systems act independently of one another and each of them is required to be sufficient for a safe reactor shutdown. No credit is taken for simultaneous operation of the two shutdown systems.

Shutdown System 1 (SDS1) contains 20 mechanical shutoff units and is designed to insert -50 mk in less than two seconds after actuation. In assessing the effectiveness of SDS1, the two 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. SDS2 is designed to inject enough gadolinium (concentrated gadolinium nitrate solution) to blanket the upper and lower reflector regions in less than 2 seconds after actuation. This is equivalent to -50 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.

Three measurement channels are used for each shutdown system. The reactor is tripped if two of the three channels are tripped. Table 3-1 summarizes the SDS1 and SDS2 trip parameters. The subsequent sections will discuss each trip parameter giving a brief description of the method of measurement.

3.1.1 High Neutron Power (Regional Overpower)

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

The ROP trip setpoint is confirmed primarily by the very slow loss of reactivity control analysis (otherwise known as critical channel power or CCP). This is also an important trip for the fast loss of reactivity control event. It also can be effective for some large LOCA cases. Compensation is required for both SDS1 and SDS2 detectors.

3.1.2 High Log Rate Power (HLR)

For SDS2, three compensated ion chambers are provided. For SDS1, fission chambers are used. The output current from each 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.

3.1.3 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, or 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.

3.1.4 Low Heat Transport System Pressure

The HTS low-pressure trips use the same instrumentation as for the HTS high-pressure trips (HTS pressure is measured in three widely separated locations on each of the reactor outlet headers, for each shutdown system).

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

3.1.5 Low Heat Transport System 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.

3.1.6 Pressurizer Low Level (PLL)

The pressurizer low level, on both SDS1 and SDS2, is a trip parameter effective for small loss-of-coolant accidents, and for loss of Pressure and Inventory Control.

3.1.7 Steam Generator Low Level (SGLL)

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.

3.1.8 Moderator High Level and Moderator Low Level

The high moderator level trip provides coverage for a loss of service water to the moderator heat exchangers as well as for a loss of moderator flow.

The low moderator level trip provides coverage for a loss of moderator event. Instrumentation for the high and low moderator level trips are shared since both trips are not required for the same event.

3.1.9 Reactor Building High Pressure

For each shutdown system a triplicated measurement of reactor building pressure is provided. The high building pressure trip provides coverage for a number of events, including large and small LOCA, as well as steam line breaks.

3.2 Emergency Core Cooling System (ECCS)

3.2.1 Emergency Core Cooling Injection

The following trip parameters and their setpoints initiate the Emergency Core Cooling injection System. The logic for an ECC injection trip to be initiated requires the HTS Low Pressure setpoint to be reached **and** a conditioning signal of either a Sustained HTS Low Pressure signal **or** a Moderator High Level **or** a Reactor Building High Pressure signal.

3.2.1.1 HTS Low Pressure

The header pressure transducers measure the pressure in three headers (ROH1, RIH1 and RIH2). Emergency Core Cooling injection is initiated when HTS Low Pressure setpoint is reached in 2 out of 3 headers (ROH1, RIH1 and RIH2).

3.2.1.2 Sustained HTS Low Pressure (Conditioning)

The measurement of the sustained low pressure is in three headers (ROH1, RIH1 and RIH2). The nominal setpoint is the same as the main low pressure signal. The low pressure must be sustained for a period of 2 minutes to generate the conditioning signal.

3.2.1.3 Moderator High Level (Conditioning)

Measurement instrumentations for these parameters are triplicated differential pressure transmitters in the moderator head tank. The setpoints for this parameter are to be determined.

3.2.1.4 Reactor Building High Pressure (Conditioning)

The pressure transducers are similar but separate from those used for shutdown system reactor building high-pressure trip.

The nominal trip setpoint and the analysis setpoint are to be determined.

3.2.2 Steam Generator Crash Cooldown

There are two steam generator crash cooldown systems. The trip parameters and their set points for crash cooldown number one is the same as those for ECC injection. The secondary crash cooldown system on ACR is configured similar to ECI transmitters configuration, which uses signals in ROH1, RIH1 and RIH2. The conditioning signal using signals in ROH1, RIH1 and RIH2 is sustained low ROH2 pressure for 2 minutes.

3.3 Containment System

The following parameters and their setpoints are used to initiate containment isolation.

3.3.1 Reactor Building (R/B) High Pressure

The pressure transducers are similar but separate from those used for shutdown system reactor building high-pressure trip and the ECC initiation.

3.3.2 High Radioactivity Levels in the Reactor Building

Narrow range radiation monitors in the ventilation system exhaust duct perform the activity measurements.

The detector's uncertainties encompass accuracy, drift and a measurement time constant.

3.4 Heat Transport System Pumps

The following pump trips are used for pump protection in potentially hazardous situations [23].

a) Pump trip due to very low pump suction pressure (e.g. Post LOCA Condition).
b) Pump trip due to high upthrust bearing temperature (e.g. Loss of bearing cooling water condition).
c) Pump trip due to loss of parallel pump at power.

On detection of loss of any one HT pump, a full reactor stepback is initiated immediately together with an automatic trip of the parallel pump.

3.5 Reactor Setback and Stepback Parameters

The setback and stepback parameters are not used in safety analysis; that is, reactor setbacks and stepbacks are not credited for decreasing reactor power in a transient even if the signals would be registered (unless perhaps it would make the event worse by masking a trip). They may be used as an indication of when the operator would be given a clear alarm to an event.

The setback routine monitors a number of plant parameters and reduces reactor power promptly in a ramp fashion if any parameter exceeds specified operating limits. The setback overrides other reactor power demands and is accompanied by alarm window annunciation. The stepback routine also reduces reactor power, but instead of doing it gradually like the setback routine, it drops the mechanical control absorbers either fully or partly into the reactor, causing a sudden power reduction.

3.6 Treatment of Trip Uncertainties

This section discusses the analysis approach to modelling the uncertainties associated with trip setpoints and the trip delays.

As is discussed in Section 2, the analysis of Class 1 to 3 events is performed using conservative assumptions. Therefore, trip uncertainties are conservatively evaluated and modelled in the analysis of these events. Class 4 and 5 events are analysed using design centre assumptions, and the design values of trip setpoints and trip delays are therefore used for these analyses.

Uncertainties in trip setpoints for Class 1 to 3 events will take into account instrumentation uncertainties as well as an allowance for uncertainty in the predicted trip parameter. Uncertainties in the instrumentation dead time, instrumentation time constant and logic delays will be accounted for in the trip models.

Table 3-1
Reactor Trip Parameters

PARAMETER	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. SYSTEM ASSUMPTIONS

This section presents key features and data about the main plant systems, which are modelled in safety analysis.

The systems to be credited in specific analyses are discussed in the relevant system specific analysis basis reports. If a system is to be credited in a given safety analysis, the modeling assumptions shall be consistent with this section. Cross-reference is also made to the trip section where automatic initiation may be a feature of a system to be credited.

4.1 Section Outline

This section is organized into sub-sections for each of the plant systems covered. The four safety systems are covered plus the key safety related systems. These are the systems which will be modelled or credited in the safety analysis. Each system is covered by further sub-sections that provide a system description, initial conditions, standard assumptions and equipment credits.

A brief description is provided of the main function and equipment of each system. As well, interconnections to other equipment or supply sources, such as cooling water or power supplies, are highlighted. The process for system initiation will be described, that is, whether it is started automatically or requires manual initiation by the operators. The level of equipment qualification is also noted.

The following sub-section on initial conditions presents the important starting points and pre-accident state of the systems required for analysis. Key design data that are used in analysis are presented here. Often the analysis values assumed are different than design data in order to be conservative. These conservative analysis values are used to perform the design basis event analysis. The design values are used to analyze the limited core damage accidents.

The following sub-section, standard assumptions, covers key assumptions and data on system performance modeled in the safety analysis during the event.

The final sub-section, equipment credits, also covers key assumptions on system performance during the event.

4.2 Requirements for System Credit in Analysis

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 [1] has been applied to the design of the ACR-700 mitigating systems credited in deterministic safety analysis beside other reliability requirements. This is an additional defence-in-depth built into ACR-700 design because application of single failure criterion is considered to lead to more robust mitigating systems.

The safety analyses for the LCDAs will use design-centred assumptions. Detailed deterministic analyses will be done for the LCDAs. This document can assist in deciding which systems should be credited, and will specify key values and assumptions if the system is modelled. In addition, there are some requirements, which can apply to any accident event, for crediting systems in safety analysis. These requirements are specified here.

4.2.1 Shutdown Systems

CNSC document R-8 [5] specifies the requirements for the shutdown systems. Clause 3.2.1 of R-8 clearly states that each shutdown system must meet performance criteria when acting alone. At most one shutdown system (the least effective shutdown system) is credited for any accident analysis.

Clause 3.6.1 of R-8 specifies the trip requirements for the shutdown systems. The second trip parameter shall be credited in the analysis. Exceptions are permitted to crediting the second trip, but must be justified for each case as per Clause 3.6.1 of R-8, or Clauses 3.4 through 3.6 of CNSC document R-77 [24].

Operator action is allowed as a backup trip parameter as per Clause 3.6.2 of R-8. Manual action can be credited if the requirements of Clause 3.6.2(c) are met in the analysis, and that the need for such a trip is made known to the ACR-700 designers such that Clauses 3.6.2 (a), (b) and (d) can be met in the future. The response time for the operator to most single failure events will be extended to 8 hours.

When a shutdown system is credited in an analysis, it will be as per the assumptions in this document. When a shutdown system is not credited, the system is assumed completely unavailable, but no credit will be taken for a reduced frequency of the event in the acceptance criteria. Partial or temporary failures will not be addressed, as it is impractical to address such failures.

4.2.2 Containment and ECC Systems

Accident analysis will be performed for events with both these systems credited and with impairments of the ECC system.

CNSC documents R-7 [25] and R-9 [26] cover the requirements for these safety systems. Clauses 3.8.4 and 3.5.4 (respectively) cover subsystem requirements. The total failure of ECC does not need to be considered in any analysis if subsystems are designed to be independent. Section 4.9 identifies any subsystems for the ECC system. Containment impairments are not analyzed as a Class 1 to 5 event because there are two independent means of containment isolation in the ACR-700 design.

The reliability of these systems will meet or exceed the requirements in Clauses 3.7.1 of R-7 and 3.4.1 of R-9. Any event with no action from both systems is beyond the design basis of ACR-700 and will not be analyzed.

When ECC or containment is credited in an analysis, it will be as per the assumptions in this document. When a system is not credited, credit will be taken for a reduced frequency of the event in the acceptance criteria. Partial or temporary failures may be addressed unless it is obvious that a total failure, notwithstanding subsystems, is the worst case.

4.2.3 Safety Related Systems

Safety related systems other than the four safety systems could mitigate the consequences of a postulated event in the safety analysis. Unless an accident event being addressed renders a given system unavailable (for example, an earthquake which damages an unqualified system), it is possible to credit a safety related system, subject to the conditions given below. If the action of a

safety related system is shown to be insignificant for a particular event, it is not necessary to model the system in the analysis.

Systems providing support to a special safety system (such as air, power, water, etc.) are credited when the special safety system is credited and must meet the requirements for these systems in R-7, R-8 and R-9.

Systems, which are not in operation before the accident, are only credited if there are automatic signals to provide initiation, or 15 minutes is allowed (as per current practice) for operator action (from the Main Control Room) from clear signals that operator action is required. When credited, the systems are modeled to respond as per assumptions in this document. It is ACR-700 design practice to automate the systems that are requested to respond to design basis events.

Systems, which are in operation or in standby mode before the accident, are credited to respond as per the assumptions in this document (including its qualification), unless the event specifically includes the failure of a safety related system and a reduced frequency of the event is recognized in the acceptance criteria. Exceptions to this include both the Pressure and Inventory Control System (for the HTS) and the RRS. These are dealt with in the following sub-sections.

4.2.4 Reactor Regulating System

The RRS is a key system for controlling reactor power during normal operation. CNSC document R-8, Clauses 3.5.4 and 3.5.5, states that the shutdown systems must perform their function without depending on any other system. In practice, the RRS is the main safety related system that would affect shutdown action. Therefore, analysis cases will be done with and without RRS action to show acceptable shutdown results. Only one such case needs to be analyzed if it is obvious which would yield the more conservative result.

When RRS is credited in an analysis, it will be as per the assumptions in this document. When the RRS is assumed to fail, it will be assumed to remain in its pre-accident state. All rods and mechanical zone controllers will remain “frozen”. Partial or temporary failures will not normally be addressed, as it is impractical to address such failures. An exception to this is when the RRS is assumed fail to setback or stepback, yet functions correctly by maintaining reactor power at a prescribed level (for example, during small LOCAs).

4.2.5 Pressure and Inventory Control System

The P&IC system is key for controlling the inventory and pressure of the HTS during normal operation. Analysis cases will be done with and without P&IC action. Only one such case needs to be analysed if it is obvious which would yield the more conservative result. An exception to this is where the pressure in the HTS increases and the acceptance criterion is for the HTS overpressure protection as specified in R-77 Clauses 4.1 and 4.2 [24].

When the P&IC system is credited in an analysis, it will be as per the assumptions in this document. When the P&IC system is assumed to fail, it will be assumed to fail in its pre-accident state. All valves will remain “frozen”. Failure of different P&IC components is only addressed in loss of pressure and inventory control (LOPIC) analyses, as it is impractical to analyse different P&IC components partial or temporary failures for other events. An exception to this is where the pressure in the HTS increases and the acceptance criterion is for the HTS overpressure protection as specified in R-77 Clauses 4.1 and 4.2 (Where power-actuated relief

valves are installed and are connected to the instrumentation associated with one of the shutdown systems, these relief valves should be considered as part of the shutdown system in question. Consequently, such relief valves should only be credited in analyses in which it is assumed that the shutdown system in question trips. Where power-actuated relief valves are installed but are not connected to the instrumentation associated with either shutdown system, these relief valves may be credited in all overpressure protection analyses).

4.2.6 HTS Liquid Relief Valves

R-77 states that where power-actuated relief valves are connected to the first shutdown system's instrumentation, it is possible that operation of the LRVs without a reactor trip could delay initiation of the second shutdown system. This may result in more severe pressurization or fuel dryout.

In analyses to demonstrate that the overpressure requirements are met, it is required that:

a) Process system protective actions (including regulating system action) are not credited.

b) Only the second trip parameter in each shutdown system is credited except in the special cases outlined in items 1, 2 and 3 below:

   - The first trip parameter may be credited in cases where this parameter is high pressure in the system under consideration.
   - In cases where only one trip parameter is installed in the first shutdown system and where this parameter is not high pressure in the system under consideration, the service limits given for the first shutdown system in Table 2-3 must be met by the first parameter of the second shutdown system.
   - In cases where only one trip parameter is installed in the second shutdown system, this trip parameter may be credited.
   - R-77 also states that in reactor plants where power actuated relief valves are installed and are connected to the instrumentation associated with one of the shutdown systems, these valves should be considered as part of the shutdown system in question. Consequently, such relief valves should only be credited in analyses in which it is assumed that the shutdown system in question trips.

When LRVs are credited for over pressurization transients (reactor trips on SDS1), the first LRV signalled to open is assumed unavailable. For overpressure protection analysis, the most severe over pressurization is obtained when the LRVs are not credited. SDS2 analysis is performed with and without crediting operation of the LRVs, since their action increases the amount of fuel sheath dryout.

When LRVs are credited in analyses of fuel dryout, crediting the LRVs may be the more severe case because of increased voiding. Therefore, all LRVs are assumed to open on overpressure signal from SDS1 for dryout considerations.

4.3 Heat Transport System (ASI 33100)

4.3.1 System Description

The heat transport system (HTS) circulates pressurized coolant (H_2O) through the reactor fuel channels to remove heat produced by the fuel and transports it to the steam generators (SGs). The major components of the HTS are the 284 fuel channels, two steam generators, four HTS pumps, two reactor inlet headers (RIHs), two reactor outlet headers (ROHs) and one 6 inch. ROH to ROH inter-connecting pipe. See Figure 4-1 for a simplified diagram.

4.3.1.1 Electrical Power Supply

The HTS pumps are supplied with Class IV power only.

4.3.1.2 Cooling Water Supply

The major cooling water systems involved are the light water coolant that transports heat from the fuel in the core to the SGs, and Main Feedwater and Auxiliary feedwater to the steam generators, on the secondary side of the SGs. In addition, Reserve Water System with a Reserve Water Tank provides emergency makeup water by gravity to the steam generators (emergency feedwater), moderator system, shield cooling system, heat transport system, dome and vault coolers, and the ECC sumps in the reactor building.

Water is supplied to the SGs by the main feedwater system, which is explained in detail in Section 4.16.

4.3.1.3 Seismic/Environmental Qualification

The heat transport system is designed for Design Basis Earthquake (DBE), qualification such that pressure boundary integrity is maintained during and following the earthquake. The heat transport pumps are designed such that they remain free wheeling following a DBE. The steam generators are qualified to DBE, Category A, such that SGs pressure boundary integrity is maintained following the event.

HT pumps are adequately qualified to maintain HTS integrity following a LOCA.

4.3.1.4 Operator Action

All required system functions are automatic. The response time for the operator to most single failure events will be extended to 8 hours.

4.3.2 Initial Conditions

	Design Value	Analysis Value	Comments
Heat Transport System Fission heat generated in:			The HTS is assumed full, at normal operating pressure and temperature before the start of the transient.
Fuel Sheaths plus bundle structure Coolant Pressure tube **Fission heat generated in fuel channel** Heat loss to moderator Heat loss to end shields **Net Fission heat to coolant** Heat loss to HT piping Heat loss to HT auxiliaries HT pump energy in coolant **Total heat transferred to Steam Generators** Heat loss from steam generators Net heat input to turbine cycle Overall Gross generator output Net generator output	1941.4 MW (th) 9.4 MW (th) 7.4 MW (th) 19.4 MW (th) 1977.6 MW (th) 2.8 MW (th) 2.4 MW (th) 1972.4 MW (th) 2.7 MW (th) 4.5 MW (th) 16.8 MW (th) 1982 MW (th) 2 MW (th) 1980 MW (th) 731 MWe 680 MWe		Design value from Reference [2]. The analysis value is 102% of the design value. Trip coverage analysis will be performed at initial powers between 0 and 102% full power.
Total Core Flow	6.9 Mg/s	6.9 Mg/s	Design value from [2]

	Design Value	Analysis Value	Comments
High Powered Channel			
Maximum Channel power	7.3 MW	8.0 MW	Design value from [2] The analysis value includes 2% power measurement uncertainty, 6% ripple allowance and 2% safety margin
HTS pumps Rated volumetric flow per pump	2250 L/s	2250 L/s	Design value from [2]
Full speed Rated head Inertia	1800 rpm 230 m 1264 kg-m^2	1800 rpm 230 m 1264 kg-m^2	Design value from [2] Design value from [2] Design value from [2]
ROH Temperature Pressure	325ºC 11.9 MPa (g)	325ºC 11.9 MPa (g)	Design value from [2] Design value from [2]
RIH Temperature Pressure	278.5ºC 13.1 MPa (g)	278.5ºC 13.1 MPa (g)	Design and analysis values are from [2] Design value and analysis value from [2]

4.3.3 Standard Assumptions

	Design Value	Analysis Value	Comments
Heat Transport System			The HTS remains intact and functional during the transient (except for events where the initiating event is a LOCA).

4.3.4 Equipment Credits

	Design Value	Analysis Value	Comments
HTS Pumps	4	4	All pumps continue to run until: • A loss of Class IV power or • A pump trip signal is generated; Refer to Section 3.4 for the trip parameters. The pump trip signal used in the analysis is sustained low pressure in either of the ROHs.

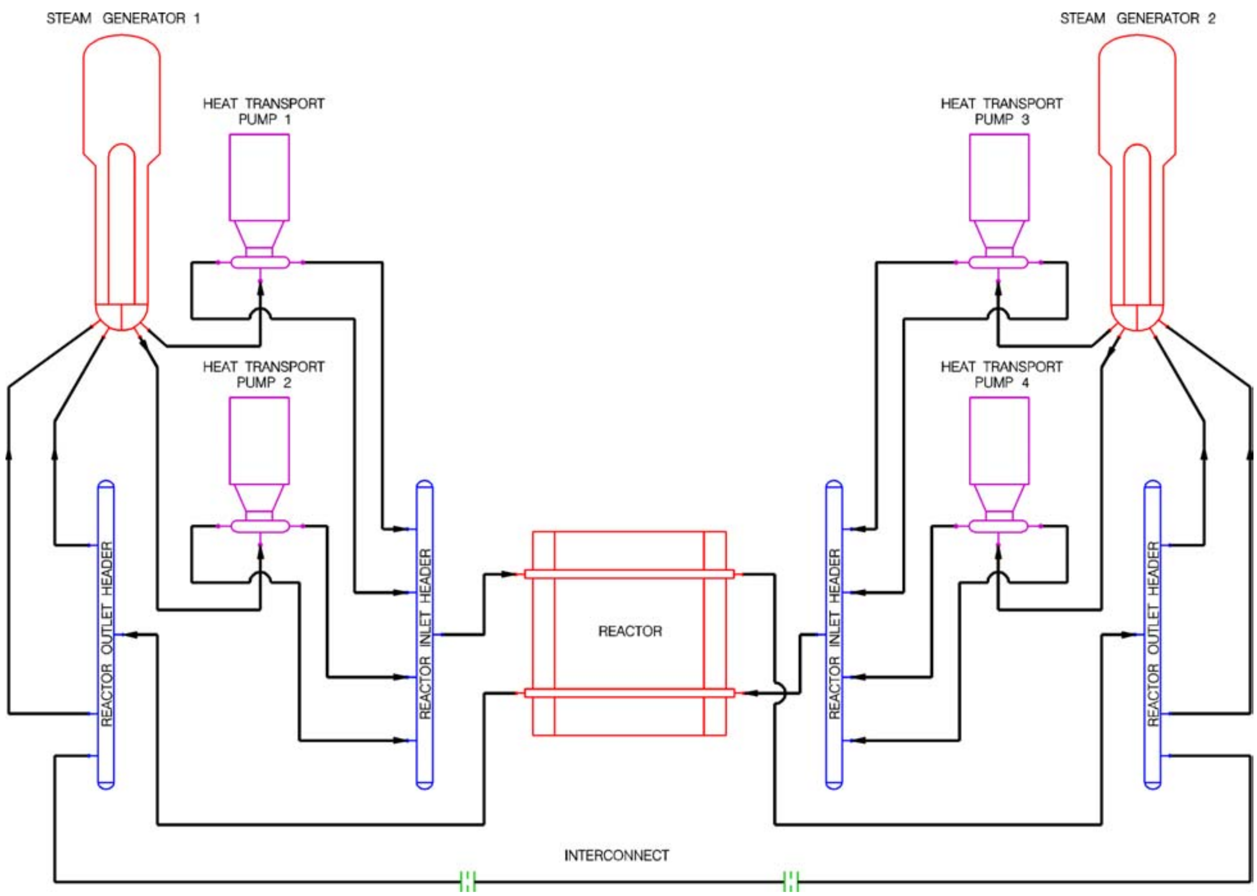


Figure 4-1 Heat Transport System Flow Diagram

4.4 Pressure and Inventory Control System (ASI 33300)

4.4.1 System Description

This system (Figure 4-2) provides pressure and inventory control and overpressure protection for the heat transport system [27]. The principal system functions are to:

- Control HTS pressure over the full range of heat transport system and reactor operating modes,
- Control HTS inventory over the full range of heat transport system and reactor operating modes,
- Limit HTS pressure increases or decreases caused by transients to acceptable values,
- Accommodate HTS coolant thermal expansion and contraction associated with warm-up, cool-down, start-up, shutdown, and power manoeuvring,
- Provide degassing of the HTS coolant.

The P&IC System shall control the Reactor Outlet Header (ROH) pressure at the desired value. If the pressure is above the setpoint, the spray will come first, pressurizer relief valves on the pressurizer open to reduce pressure and, if below the setpoint, pressurizer heater(s) is/are turned on to increase pressure.

The Feed and Bleed System control the pressurizer level. Level measurements are compared to a setpoint which is a function of RIH temperature, ROH pressure and reactor power. Under normal steady state conditions, the P&IC system both, bleeds and feeds constantly. The bleed flow (usually at the normal purification flow rate) is removed from the HTS, passes through the purification circuit and re-enters the HTS via the HT pressurizing pumps.

The pressurizer spray valves are sized to limit the HTS pressure increases to below the system allowable design targets during transient conditions. The spray valves control the pressurizer pressure as it rises above the calculated setpoint. Two Spray Valves are designed to control the Pressurizer pressure by directing water from RIH to the Pressurizer and are normally in automatic control. Each valve is sized to have a minimum capacity equivalent to 50% of the maximum spray flow during power manoeuvring.

Some of the P&IC system functions are only credited when their normal or expected action serves to make the event worse. In particular, the Feed and Bleed operation is often not involved in analysis, as their function does not have a significant impact on the accident progression. However, in some cases operation of either of these systems may adversely affect plant performance. The analyst must be aware of the functions of these systems to ensure that the analysis is conservative. The data supplied with the systems below are the values to be used if the system is to be credited.

Two instrumented liquid relief valves (LRVs) provide overpressure protection for the heat transport system. During reactor operation, these LRVs operate in conjunction with the safety systems to limit abnormal transient pressures in the HTS. The LRVs discharge into the bleed condenser. The actuation signal is obtained from the SDS1 high-pressure trip instrumentation (refer to Section 4.2.5 for rules on crediting the LRVs in safety analysis).

4.4.1.1 Electrical Power Supply

Equipment in the P & IC system are powered as follows:

a) Coolant pressurizing pump - Class III
b) Pressurizer heaters - Class IV
c) Class II supplies to control and instrumentation loads and motorized isolation valves at the HTS pressure boundary

4.4.1.2 Cooling Water Supply

This system is provided with coolant supply tanks that have adequate capacity to contain coolant from the HTS during maintenance mode. Piping connects the coolant supply system to the coolant storage tank in the reactor building. Division 1 of the Recirculated Cooling Water System (71340) provides cooling water to the heat exchangers in the P&IC system and the pressurizing pump motors.

4.4.1.3 Seismic/Environmental Qualification

The P&IC system is environmentally qualified per [28] and fulfills the following requirements:

a) The HTS liquid relief valves are environmentally qualified to ensure their operability following a Main Steam Line Break (MSLB) or LOCA.
b) The pressurizer is designed and environmentally qualified to accommodate coolant volume change in the HTS during MSLB or LOCA conditions.
c) The HT pressurizing pumps are required to be functional after an HTS leak. These pumps are environmentally qualified to ensure their operability during an HTS leak.
d) Valves and other associated instrumentation that must function to maintain the heat transport system pressure boundary integrity are environmentally qualified to LOCA and MSLB conditions. They shall also be designed to DBE category B. This ensures that these valves would not fail inadvertently to the open position under harsh environment caused either by a DBE or by other accidents.
e) All the subsystems directly connected to the HTS shall be seismically qualified to ensure pressure boundary integrity of the HTS, or be isolatable.
f) Instrumentation associated with Post Accident Monitoring shall be seismically qualified.
g) The pressure relief circuits shall be seismically qualified such that overpressure protection remains available.

The Nuclear Class 1 portion of the system shall be seismically qualified to DBE, and components within the boundary shall either be classified as Category A or Category B in compliance with SDG 108-03650-SDG-002 [29].

The basis of the above classification is to ensure the HTS integrity and to prevent the outflow of coolant from the heat transport system during and after a DBE event. In general, components located within the pressure boundary of the HTS such as the pressurizer and bleed condenser shall be designed for DBE Category A to retain the pressure integrity during and after a DBE.

Valves that are located at the boundary of the HTS shall be designed for DBE Category B so that pressure integrity and operability during and following a DBE are maintained. Control valves

must be capable of closing during and after a DBE. For normally closed manual valves, this requires the seal to be leak tight. All valves with an HTS inventory retention function shall be qualified to DBE category B.

The heat transport liquid relief valves and bleed condenser relief valves performing an essential overpressure protection function shall be designed and qualified to DBE Category B.

The HT liquid relief valves shall remain functional during and after the event with the position status being available in the MCR.

The pressure retaining portions of instrument systems connected to the Class 1 portion of the system shall be qualified to DBE category A. This will include instruments, valves, tubing and instrument racks.

4.4.1.4 Operator Action

All safety related system functions are automatic. No operator action related to safety analysis is required for this system in the first 8 hours of an accident.

4.4.2 Initial Conditions

	Design Value	Analysis Value	Comments
Pressure and Inventory Control System			The P&IC system is assumed to be functioning as designed before the transient.
Pressurizer Capacity	55 m^3	55 m^3	Design value from Reference [2] Pressurizer level is assumed to be at a maximum or minimum operating value for consevatism, depending in the analysis case.

4.4.3 Standard Assumptions

	Design Value	Analysis Value	Comments
Auxiliary Systems			On a loss of Class IV power, there will be no cooling water to the bleed cooler, so the bleed condenser may bottle up to contain a discharge from the LRVs. On restoration of Class III power the bleed cooler will receive service water.

4.4.4 Equipment Credits

	Design Value	Analysis Value	Comments
Pressure and Inventory Control System			The P&IC System can be assumed for analysis purposes to function correctly or be frozen in its state before the transient. Refer to Section 4.2.5 for rules on how the P&IC system is to be credited in analysis. If functioning correctly: on increasing pressure, pressurizer steam bleed valves function properly and open as demanded by the HTS pressure control program.
Pressurizer Heaters **Number of heaters** **Heat input**	 5 200 kW/each	 5 200 kW/each	 Design value from [2] Class IV power to all heaters. Automatic trip off on a low pressurizer level.
HT pressurizing pumps Number of pumps Flow rate (per pump) Liquid Relief Valves Number of LRVs Pressure setpoint	 2 24 L/s 2 12.25 MPa (g)	 1 24 L/s 2 12.25 MPa (g)	 Design value from [2], Analysis value assumes one of the 100% pumps is not available.

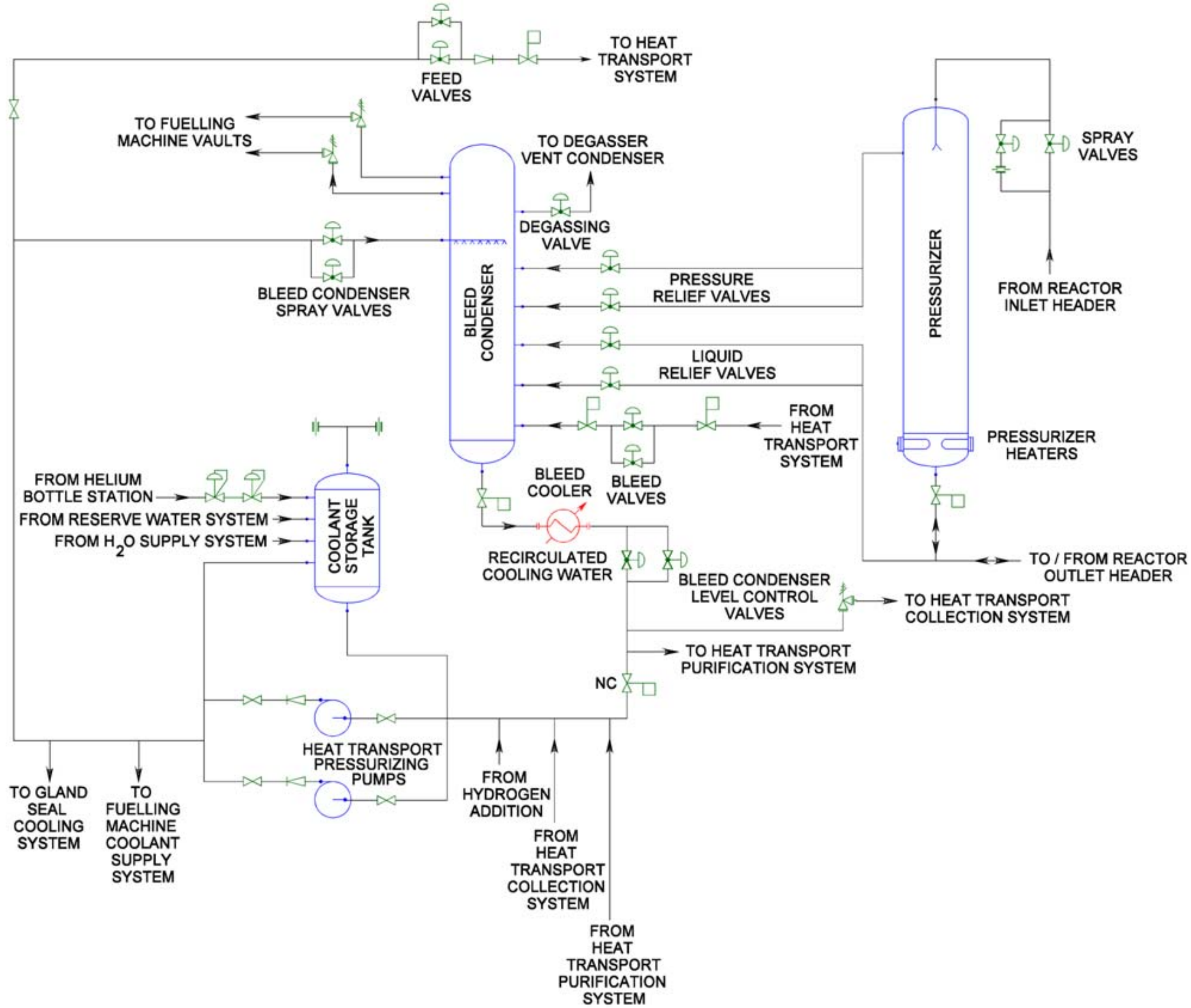


Figure 4-2 ACR-700 Pressure & Inventory Control Simplified Flow Diagram

4.5 Main Steam System (ASI 36000)

4.5.1 System Description

The principal function of the main steam system is to supply steam to the turbine-generator. The main components of this system are the steam main piping, the steam header, the eight Main Steam Safety Valves (MSSVs), the four Main Steam Isolation Valves (MSIVs), the four Atmospheric Steam Discharge Valves (ASDVs) and the eight Condenser Steam Discharge Valves (CSDVs). Refer to Figure 4-3.

The main feedwater system normally supplies full pressure water to the secondary side by three 50 percent main feedwater pumps (refer to Section 4.16). The auxiliary pumps (two 100 % auxiliary feedwater pumps) provide water to the steam generators at a rate sufficient to remove decay heat from the reactor core in the event of complete loss of the main feedwater pumps and consequential reactor shutdown.

The SGs are arranged such that each SG may be isolated separately. This is accomplished by closing the appropriate MSIV automatically or manually following a steam generator tube rupture (SGTR) or a main steam line break (MSLB).

<u>SG secondary side level control</u>

The steam generator level controller maintains the steam generator level at the set point during normal operation conditions and within limits under upset conditions such as turbine fast run-back or turbine trip.

<u>SG secondary side pressure control</u>

The program controls steam pressure to the desired set point by changing the reactor power set point or by adjusting the plant loads.

<u>‘Turbine leading’ vs. ‘Reactor leading’ mode</u>

The overall plant control can be in either ‘Turbine leading’ mode or ‘Reactor leading’ mode. In ‘Turbine leading’ mode, the demand power routine is sensitive to the demands from the steam generator pressure controller, resulting in steam pressure control by means of changes in reactor power. The reactor leading mode is used during plant upset conditions and normal operation when the unit is used as “base load” supply to the grid system. In this mode, the reactor power set point is adjusted to the desired value and the turbine output automatically seeks its highest value allowed by that reactor power.

<u>Overpressure protection</u>

The MSSVs, CSDVs and ASDVs function to protect the steam system and steam generators from overpressure. During normal operation, all are closed. The MSSVs are spring loaded with pneumatic control for over pressure protection and SG secondary side crash cooldown in LOCA. CSDVs are available to the steam generator pressure control program in case the turbine is unavailable. ASDVs are available to the steam generator pressure control program in case both the turbine and the CSDVs are unavailable. CSDV and ASDV can also be positioned by manual control.

4.5.1.1 Electrical Power Supply

The solenoid that controls the operation of the pneumatic actuator for each MSSV is on Class II power. The CSDVs are powered by Class II power. Class II power is also supplied to the electro/pneumatic control devices on the ASDVs.

4.5.1.2 Cooling Water Supply

The SG feedwater supplies are described in their relevant sections.

4.5.1.3 Seismic/Environmental Qualification

The main steam lines are seismically qualified to DBE Category A. The MSSVs are seismically qualified, respectively qualified to DBE Category B.

The MSSVs located outside the reactor building are environmentally qualified for MSLB.

4.5.1.4 Operator Action

The MSIVs, installed downstream of the main steam safety valves and upstream of the atmospheric steam discharge valves, are motorized and can be remotely, manually operated from the main control room. Atmospheric Steam Discharge Valves (ASDVs) are actuated in response to the steam generator pressure control program demands. Provision is made to allow the operator to open CSDVs via computer.

4.5.2 Initial Conditions

	Design Value	Analysis Value	Comments
Main Steam System			
Steam Generators			
Steam temperature (at drum nozzle)	281ºC	281ºC	Design value from [2]
Steam drum pressure	6.4 MPa (a)	6.4 MPa (a)	Design value from [2]
Feedwater temperature	218ºC	220ºC	Design value from [2]
Secondary Side Steam and Feedwater Flows			
Nominal full power rates			
Steam flow	1075 kg/s (total for 2 SGs)	1096.5 kg/s (total for 2 SGs)	Design value from [2], analysis value is assumed to be 102% of the design value.

4.5.3 Standard Assumptions

	Design Value	Analysis Value	Comments
Steam Generator Level Control			This system attempts to maintain a constant inventory in the SG secondary side (dependent on the reactor power) throughout the transient or until mitigation action occurs. Refer to Section 4.2.3 for rules on how the SG Level Control System is to be credited in analysis.
Steam Generator Pressure Control			This system attempts to maintain a constant pressure in the SG secondary side throughout the transient. Refer to Section 4.2.3 for rules on how the SG Pressure Control System is to be credited in analysis.
Steam Generators Fouling coefficient		0 to 0.0352 m^2°C/kW	The analysis range represents fouling corresponding to a new clean SG (coefficient equals 0) to the end of life.
Heat Transport System Chronic Leakage into the secondary system		20 kg/h	Leakage through a steam generator tube is assumed to occur for a period such that an equilibrium concentration of radionuclides exists on the secondary side.

4.5.4 Equipment Credits

	Design Value	Analysis Value	Comments
Main Steam Safety Valves (MSSVs) Number	8		Design value from Reference [2] Number of MSSV's credited in analysis consistent with single failure criterion.
Minimum capacity (each) Lift	179 kg/s Full lift at 3% above set pressure	179 kg/s Full lift at 3% above set pressure	Design value from Reference [2] Design value from Reference [2]
MSIV Stroking valve time	90-120 s	120 s	Design value from Reference [2]
Turbine			Assume to start to unload at 95% of nominal separator pressure and is completely offline at 90%.
Atmospheric Steam Discharge Valves (ASDVs) Number Flow rate each (maximum)	4 26.9 kg/s	4 26.9 kg/s	Design value from Reference [2] Design value from Reference [2]
Condenser Steam Discharge Valves (CSDVs) Quantity	8	8	Design value from Reference [2]

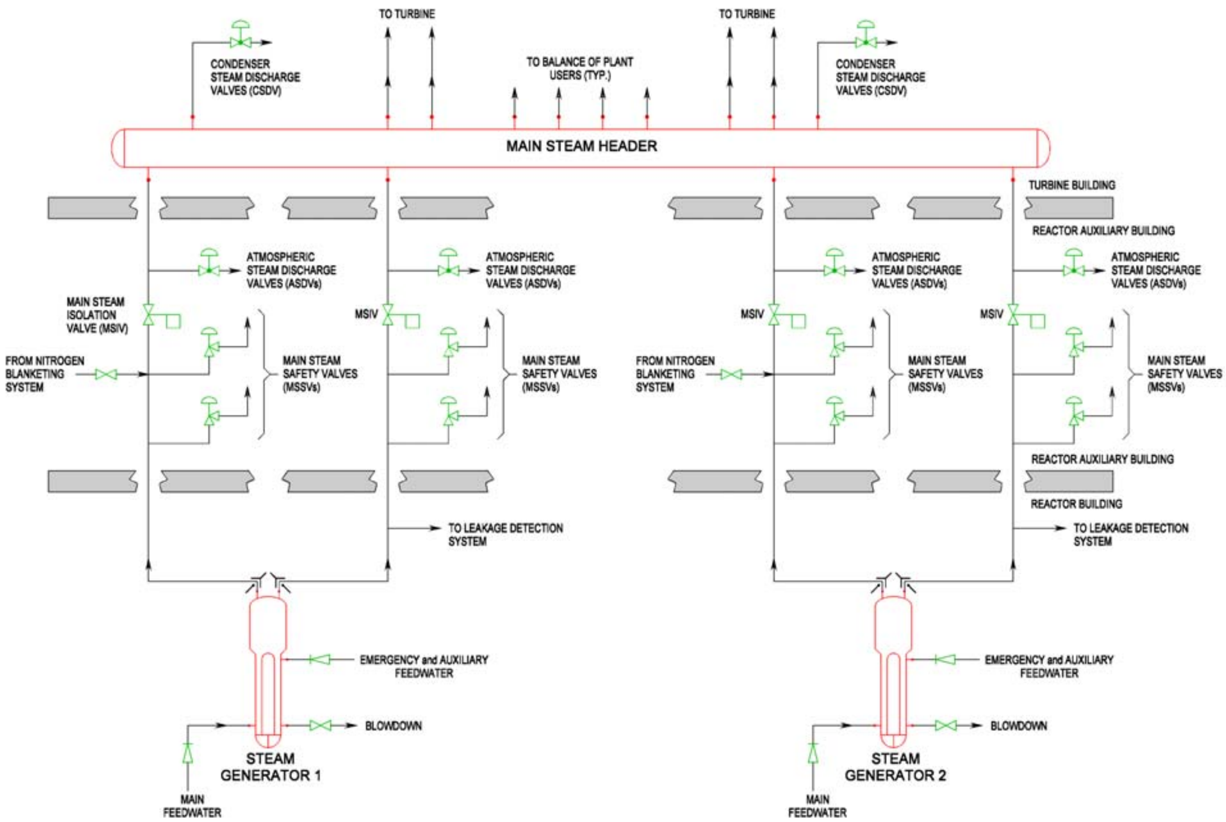


Figure 4-3 Main Steam System Flow Diagram

4.6 Reactor Regulating System (RRS) (ASI 63720)

4.6.1 System Description

RRS is an integrated system comprising reactor neutron flux and thermal power measurements, reactivity control devices, computer programs for control and monitoring of reactor power and flux tilts, and a control panel in the main control room [2].

RRS is part of the Overall Plant Control System, a set of related computer control functions that integrate the control of reactor power, steam generator pressure and turbine load to meet unit load demands.

Figure 4-4 shows a block diagram of RRS. The main functions performed by this system are:

- Monitor total reactor power; adjust reactivity devices to control power to a setpoint specified by the operator or dictated by grid commands.
- Monitor the reactor flux shape; adjust reactivity devices to oppose reactor flux tilts.
- Monitor important reactor and process system parameters; reduce reactor power at an appropriate rate if any parameter exceeds prescribed limits.

RRS is provided with the following types of reactor power measurements:

- Out-of core fission chambers
- In-core self-powered flux detectors
- Reactor thermal power measurements

RRS shall provide controls for the following reactivity control devices:

- Zone control units;
- Mechanical control absorbers;
- Shutoff units (out-drive only);
- Moderator poison (addition only).

<u>Zone Control Units</u>

The zone control units are the principal reactivity control devices, used both for control of total reactor power and for control of flux tilts (spatial control).

For control of overall reactivity, a sufficient number of zone control units are provided to allow adequate reactivity reserve to handle brief periods of refuelling incapability and xenon transients due to small reactor power changes.

For spatial control, a sufficient number of independently controllable zone control rods are provided to allow, as a minimum, side-to-side, top-to-bottom, and end-to-end flux tilts to be controlled.

<u>Mechanical Control Absorbers</u>

Control absorbers, normally withdrawn from the reactor, shall provide additional negative reactivity to back up the zone control units. They shall have the capability to shut down the reactor independently of the shutdown systems. They are capable of being dropped into the reactor as well as being driven in and out.

Shutoff Rods

RRS shall provide automatic control of shutoff rod withdrawal, after the SDS1 trip has been reset. Withdrawal is stopped if rate and/or power error are too high, indicating excessive positive reactivity or rate of change of reactivity.

Moderator Poison

Two moderator poison addition systems are provided:

- Boron addition, initiated manually, is used as a source of long-term negative reactivity when the reactor has excess fuel reactivity.
- Gadolinium addition, normally initiated manually, is used as a source of short-term negative reactivity, to compensate for a lack of xenon (gadolinium burns out at a rate similar to xenon production rate.) Under special conditions (positive flux rate and large power error) RRS should add gadolinium automatically.

RRS Control programs

The reactor control software must include the following functional program modules:

- RRS control program (including setbacks)
- Stepback program
- Flux mapping program

Each program may be further divided into sub-modules, as required by good software practices.

4.6.1.1 Electrical Power Supply

All RRS measurements (e.g., reactor flux and thermal power) and the DCS and PDS systems shall be supplied from uninterruptible electrical power, Class II. The 90 Vdc power supplies for the mechanical control absorber clutches shall also be powered from Class II.

The drive motors for the various reactivity devices (e.g., zone control rods, mechanical control absorbers, shutoff rods) shall be supplied with Class III power.

4.6.1.2 Cooling Water Supply

The moderator cools immersed reactivity devices when inserted in the core.

4.6.1.3 Seismic/Environmental Qualification

There are no seismic qualification requirements for RRS.

There are no environmental qualification requirements for RRS.

4.6.1.4 Operator Action

No operator action with respect to this system is required within the first 8 hours of an accident.

4.6.2 Initial Conditions

	Design Value	Analysis Value	Comments
Reactor Regulating System Reactor power setpoint	100%	102%	The RRS is assumed to be functioning normally before the transient. The RRS measures power in percent of full power. For analysis purposes, the reactor power is constant and steady at 102% full power before the initiating event unless specified by accident scenario.
Zone Control Rods (18)	50%	50%	50% out of core.
Mechanical Control Absorbers (4)	Out of core.	All out of core unless specified by accident scenario	
Shutoff Rods (20)	Out of core.	Out of core.	Poised
Moderator Poison (6 nozzles)			Analysis value is corresponding to operating condition at the time of accident.

4.6.3 Standard Assumptions

	Design Value	Analysis Value	Comments
Reactor Regulating System			After the event, the RRS is assumed in analysis to be frozen or to remain operational until a reactor trip dependent upon the accident scenario. Partially functional refers to cases where the RRS is credited to function to maintain the power constant but a setback/stepback signal is not credited to reduce reactor power.

4.6.4 Equipment Credits

	Design Value	Analysis Value	Comments
Zone Control Rods Static reactivity worth Insertion rate	 - 9 mk 0.1 to 0.2 mk/s		The reactivity worth calculation is dependent on the accident scenario. Design value from[2]
Mechanical Control Absorbers Static reactivity worth Reactivity-change rate	 - 11 mk 0.05 to 0.2 mk/s		Design value from[2]

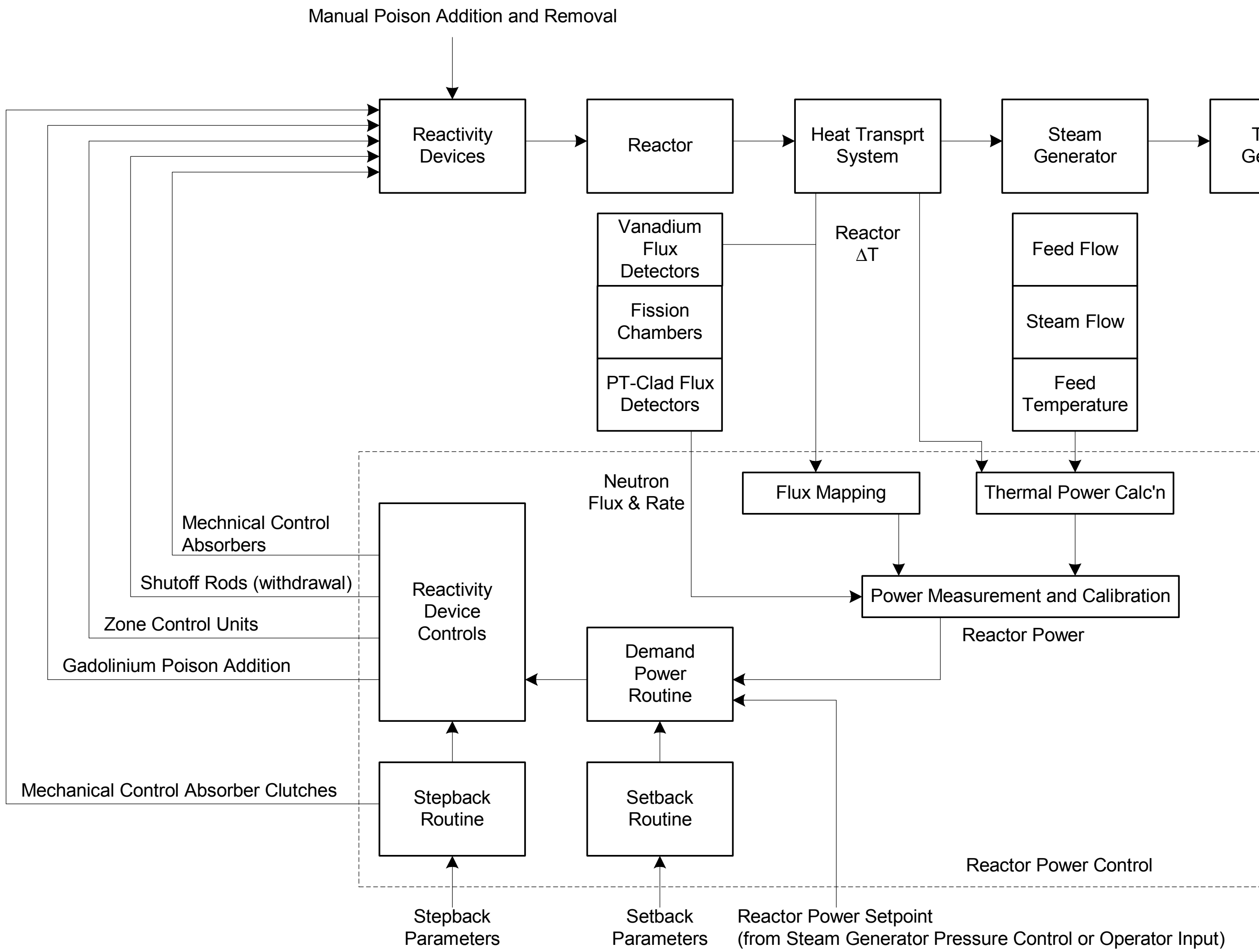


Figure 4-4 Reactor Regulating System Block Diagram

4.7 Shutdown System One (SDS1) (ASI 68200)

4.7.1 System Description

The main components of SDS1 are the 20 mechanical shutoff units, the trip computers, the trip logic and test circuitry, the flux detectors, and process instrumentation and startup instrumentation.

The SDS1 reactor trip criteria are described in Section 3.1.

During normal operation, SDS1 remains in a poised state. All shutoff rods will be fully withdrawn and are available. In assessing the effectiveness of SDS1, the 2 most effective rods are assumed to be unavailable. Following an in-core break more rods will be assumed to be unavailable.

4.7.1.1 Shutdown System Control Logic

Figure 4-5 shows a block diagram of the SDS1 trip logic. One trip computer per SDS is used.

There are three independent measurement channels, having completely independent and physically separated power supplies, trip parameter sensors, instrumentation, trip computers, and annunciation. SDS1 uses local coincidence voting logic; i.e., the shutoff rods are dropped when any two of the three channels trip, of the same trip parameter.

4.7.1.2 Shutoff Rod Withdrawal Logic

Dropping of the shutoff rods is controlled by SDS1. The reactor regulating system controls withdrawal of the rods. Withdrawal is inhibited until the shutdown signal is cleared. The design of this logic is similar to that for the adjuster and mechanical control absorber rods (see Section 4.6), except that constant speed drive is used. The logic counts the number of shutoff rods withdrawn to determine when SDS1 is poised.

For withdrawal, the shutoff rods are arranged in two banks and are withdrawn in separate banks. The operator may also select individual rods to be driven in or out under manual control. Analog rod position signals are provided from potentiometers on the mechanisms for all shutoff rods to the distributed control system.

4.7.1.3 Electrical Power Supply

SDS1 instrumentation and control functions are powered from seismically qualified uninterruptible Class I power. Separately channelled Class I power supplies are provided for each channel of SDS1. The DC clutches, operated from Class I power, release on loss of power.

Power loss to a channel results in an irrational signal to the trip computer, and a channel trip. A reactor trip occurs on a loss of power to two or more channels.

4.7.1.4 Seismic/Environmental Qualification

SDS1 is designed and qualified to DBE Category B, that is, to remain operable during an earthquake. This requires seismic qualification of all components required by SDS1 to fulfil its

function e.g., sensors, cabling, power supplies, supports, as well as active components such as shutoff rods.

The components of SDS1 are designed to maintain operation under the most adverse environmental conditions, LOCA and MSLB. This includes the instrumentation for the trip signals required for activation of SDS1 as well as parameters required for Post Accident Management.

4.7.1.5 Operator Action

No operator action is required for this system to function properly. The operator has the option of manually initiating the shutdown system. For accident analysis, manual activation may be credited for those initiating events where operator action is not required within 15 minutes of clear and unambiguous indication, as a backup trip. The response time for the operator to most single failure events for ACR–700 will be extended to 8 hours.

4.7.2 Initial Conditions

	Design Value	Analysis Value	Comments
SDS1	System poised	System poised	Except for one rod, which is assumed to be out of service for maintenance.

4.7.3 Standard Assumptions

	Design Value	Analysis Value	Comments
SDS1 **Trip signal logic** **Rod drop characteristics**	 2 out of 3 See Table 4-1 Shutoff Rods Performance Target	 2 out of 3	- The effectiveness of SDS1 is considered separately from that of SDS2, that is, only one shutdown system is credited at a time. - For analysis purposes, the first trip signal that is generated is not credited*. It is assumed that the second trip signal initiates the shutdown system. In reality the design would cause a reactor trip (SDS1 initiation) on the first trip signal (i.e. two out of three voting sends a signal to the trip computer). * There is an exception to this statement. For trip coverage analysis to determine heat transport system pressure during potential overpressure events, the high ROH pressure signal can be credited as the first trip signal for initiating SDS1.

	Design Value	Analysis Value	Comments
			- For analysis, all shutoff rods drop at the same velocity and are initially parked at the same elevation Table 4-1 from Reference [30]
ROP Detectors			- All ROP detectors are assume to have the same margin to trip, that is, they are assumed to have just been calibrated prior to the transient. Refer to Section 3.1.1 for details regarding ROP detectors for trip detection.

4.7.4 Equipment Credits

	Design Value	Analysis Value	Comments
Shutoff Rods	20	18	- Design value from Reference [2] for analysis purposes; the two most effective shutoff rods are assumed unavailable for SDS1. The missing rods are chosen to minimize the effectiveness of the remaining rods.
SDS1 Time Delays Time delay prior to 1st rod movement Additional time delay			- The time delay before first rod movement includes: instrumentation delay, hardware computer delay time and power relays delay. Additional analysis delays are included below - For analysis purposes the 15 ms time delay will be added for pressure tube sag due to creep. Additional time delay is included in all analysis except for those scenarios early in plant life before creep has occurred.

Table 4-1 Shutoff Rods Performance Target

Event	Time (s)	Rods Distance Travelled (mm)
Clutch circuit open	0	0
First movement	0.15	0
Gate 1 (1842 mm above calandria centre)	1.15	1220
Gate 2 (calandria centre)	1.55	3062
Gate 3 (1842 mm below calandria centre)	2.0	4904

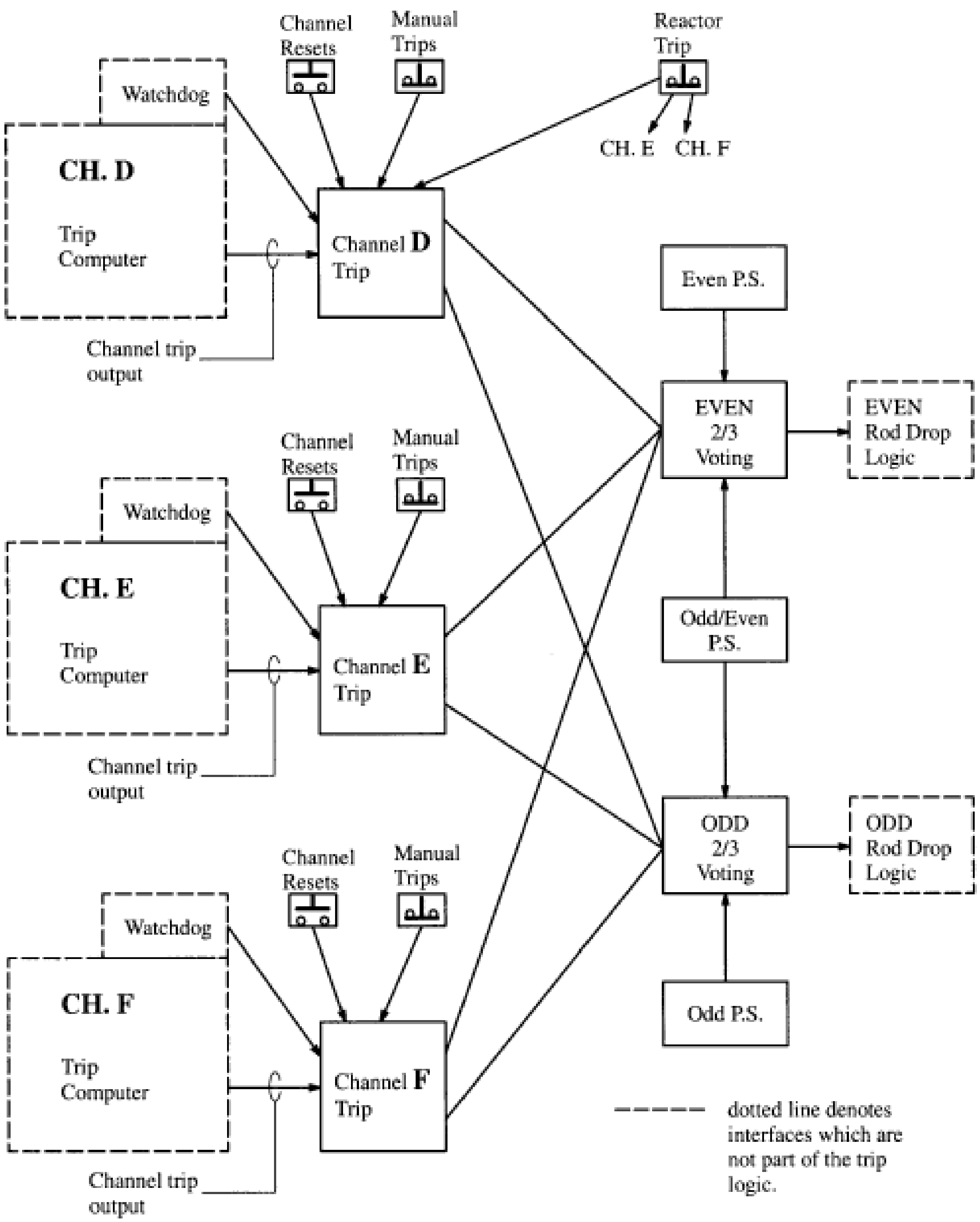


Figure 4-5 SDS1 Trip Logic Block Diagram

4.8 Shutdown System Two (SDS2) (ASI 68300)

4.8.1 System Description

SDS2 is comprised of several sub-systems, the SDS2 Trip Instrumentation (ASI 68320), the Liquid Injection Shutdown System (LISS) (ASI 34700), the Liquid Injection Shutdown Units (ASI 31820), the Computerized Trip, Display, Test, and Monitoring System, and the associated instrumentation. The main components of the LISS are, the poison tanks and nozzles, the helium supply tank, the array of quick opening valves, the mixing tanks and the two conductivity probes per tank. Refer to Figure 4-6.

Gadolinium nitrate in solution with D_2O is injected into the core to provide sufficient negative reactivity to rapidly shutdown the reactor upon SDS2 initiation. High-pressure helium supplies the energy for the rapid liquid injection. The helium tank is connected, through four quick-opening valves arranged in two successive pairs, to a helium header, which services the poison tanks. The quick-opening valves are air-to-close, spring-to-open, so that they fail safe on loss of air supply or electrical power.

Each poison tank is connected by a stainless steel pipe to a horizontal in-core injection tube nozzle, which spans the calandria and is immersed in the moderator [2]. The Zircaloy-2 nozzles penetrate the calandria horizontally and at right angles to the fuel channel tubes. Holes are drilled into the nozzle along its length to form four rows of jets, which facilitate complete dispersion of the poison into the moderator.

Two conductivity probes are installed in each poison injection line downstream of the poison tank. One is located close to the bottom of the poison tank to monitor the poison concentration and alarm on low poison concentration. The second probe is located close to the bellows assembly of the shield tank to detect when the poison solution reaches the downstream top of the U-section. Upon alarm from any of the latter probes, the associated poison injection line must be backflushed to pull the poison interface back to the ball valve or drain line inside the vault.

Each poison tank contains a floating polyethylene ball which seats at the top of the poison tank prior to injection to restrict the movement of poison upwards due to variations in moderator level. When an injection is initiated, the helium pressure transfers the poison to the calandria and the ball falls to the tank bottom. In the bottom position, the ball seats at the poison tank outlet and prevents the release of high-pressure helium to the calandria.

Manual isolation valves located in the gas and liquid legs to permit maintenance and testing on a poison tank without disabling the shutdown system can isolate each poison tank. A system of interlocks ensures that only one tank is out of service at any time. Alarms in the Main Control Room and Secondary Control Area warn the operator if valve closure occurs on more than one poison tank. Note that not all alarms will be provided in Secondary Control Building. At present, the alarms, which will be provided in Secondary Control Building (SCA), are not identified.

The SDS2 reactor trip criteria are described in Section 3.

4.8.1.1 Electrical Power Supply

Class II power is supplied to each of the SDS2 trip channels. Fuse failure or loss of power to a channel results in a channel trip, and is annunciated. A loss of power to two or more channels results in a reactor trip.

4.8.1.2 Seismic/Environmental Qualification

All active components required to initiate poison injection in the LISS and the horizontal RCUs, are qualified to an earthquake of DBE Category B. All other components, which have only to retain pressure boundary integrity during system operation, are qualified to DBE Category A.

Components whose failure during a DBE will not impair system operation are required to be seismically qualified to the National Building Code of Canada (NBCC).

All active components (required to initiate poison injection) in the LISS and the horizontal SDS2 RCUs are qualified to maintain operation under the most adverse environmental conditions, LOCA and MSLB.

4.8.1.3 Operator Action

No operator action required initiating system. A manual initiation of SDS2 is available for operator.

4.8.2 Initial Conditions

	Design Value	Analysis Value	Comments
SDS2	System Poised	System poised	
Poison tanks			
Poison front position	Between 2 conductivity probes	At conductivity probe at base of poison tank or at probe close to the shield tank	- The analysis assumption is dependent on the analysis objective (i.e., poison front at the base of the poison tank is conservative for speed of insertion calculations).
Poison tank level			- For analysis purposes, the level can range from the highest level for speed of insertion calculations to the lowest level for reactivity depth calculations.

4.8.3 Standard Assumptions

	Design Value	Analysis Value	Comments
SDS2 Number of injection nozzles	6	5	- Design value from [2]. Analysis assumes that one nozzle is unavailable (no poison comes out of the unavailable nozzles).
Poison tanks Number of available tanks Poison concentration Operating Pressure	6 8000 ppm of Gd in D_2O 8.3 MPa (g)	5	- Design value from [2]. Analysis assumes that one tank is unavailable.

4.8.4 Equipment Credits

	Design Value	Analysis Value	Comments
Helium injection valve opening times Total time from receipt of signal to fully open	110 ms	110 ms	Design value from [2]

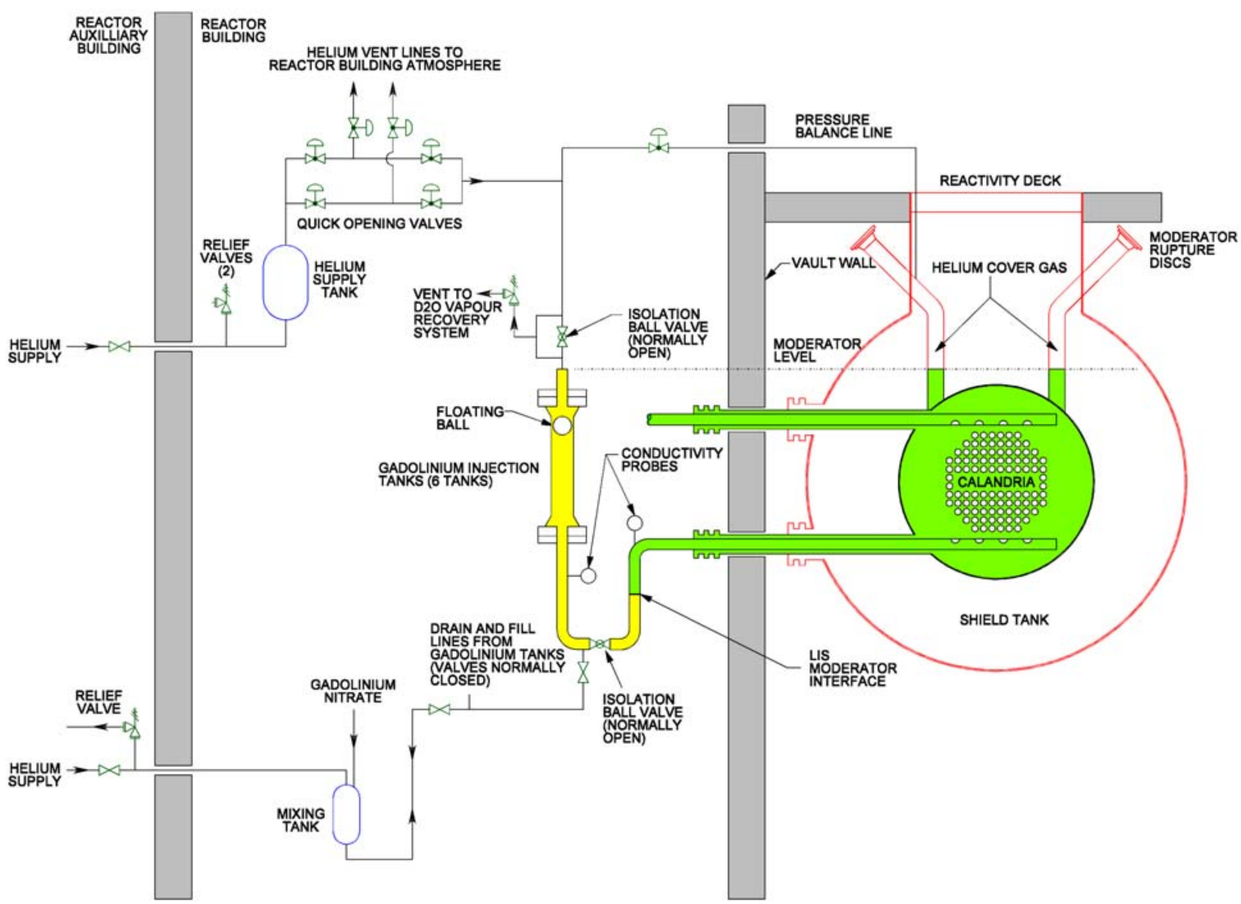


Figure 4-6 Schematic of ACR Liquid Injection Shutdown System

4.9 Emergency Core Cooling (ECC) System (ASI 34320)

4.9.1 System Description

The basic function of the Emergency Core Cooling (ECC) system is to provide a means of cooling the reactor fuel in the unlikely event of an accident which would deplete the normal coolant inventory in the heat transport system (HTS) to an extent that fuel cooling is not assured. The ECC System consists of two sub-systems: Emergency Coolant Injection (ECI) System for high-pressure coolant injection after LOCA and Long Term Cooling (LTC) System for long-term recirculation/recovery, after LOCA.

There are two separate ECI divisions, including high-pressure ECI water tank and piping. Each of these two separate divisions would inject water to both inlet headers. The driving force of the ECI System is provided by the two water tanks located inside the reactor building, with a compressed gas bubble introduced at the top portion of the ECI water tanks. See Figure 4-7 for a schematic representation of the ECI system.

Low pressure LTC consists of two independent divisions, each division including a sump, two pumps in parallel, one heat exchanger, and associated piping and valves. Each division is attached to the inlet header, outlet header and RB sump at the corresponding end of the core. The two divisions are independent of each other and are capable of handling 100% of the load during all modes of operation. In addition, the reserve water tank plays an important role during the Long Term Cooling stage. See Figure 4-8 for a schematic representation of the LTC system.

ECCS Logic

Emergency Coolant Injection (ECI) System

The Emergency Coolant Injection system supplies light water coolant to the HTS and refills the fuel channels in the short term of a LOCA.

The ECI system is poised during normal reactor operation and is automatically initiated on the 'LOCA Signal' (refer to Section 3.2). The 'LOCA Signal' causes the ECI valves to open and be ready to begin high-pressure injection to the reactor inlet headers. "One-way" rupture discs located in ECI piping isolate the ECCS tanks from the HTS. When the HTS pressure drops below the rupture pressure of the one-way rupture disc, the rupture discs burst, thereby enabling ECI coolant injection to the reactor inlet headers. The discs support a high pressure in the reverse flow direction but burst at a relatively low pressure in the forward flow direction.

In addition, valves on the ECI interconnect line between the reactor outlet headers, open up on LOCA Signal to assist in establishing a cooling flow path. Pressurizer isolation valves close on LOCA Signal to avoid coolant return to the pressurizer. On LOCA Signal, the water in the reserve water storage tanks located in the reactor building is dumped to the sump to be ready for LTC injection (see Section 4.18).

To enhance the effectiveness of the high pressure injection of water into the heat transport system, the main steam safety valves are also opened on an LOCA Signal to provide a rapid cool down (crash cool) of the steam generators and depressurisation of the heat transport system.

A steam generator second crash cooldown system is provided and is initiated to open the MSSVs under a LOCA condition. This is an independent system, which backs up the steam generator crash cooldown that is part of the ECI system.

Long Term Cooling (LTC) System

The safety functions of the Long term cooling (LTC) system are to provide fuel cooling in the long term (recovery stage) of a LOCA. The LTC system is also used for long term cooling of the reactor after shutdown following other accidents and transients.

For a LOCA, the LTC system is initiated following operation of the ECI system. A signal from the ECI system is provided to start the LTC pumps and open the sump isolation valves, and dump water from the RWT to the RB sump upon detection of a LOCA. This signal overrides any other operating mode of the LTC (i.e., standby or long term cooling mode) and begins operation in emergency cooling mode. When the water tanks are nearly empty (signalled by a low level), the ECI tank isolation valves close and the recovery stage begins by pumping water from the sumps into the HTS via the LTC heat exchangers and thus the LTC system is initiated. LTC delivers flow to the reactor inlet headers, thereby utilizing the cooling flow path already established by the high pressure ECI system. The water subsequently escapes from the break in the HTS, falls to the reactor building sump and is recirculated by the pumps to provide a long term heat sink.

The design is such that the LTC can provide shutdown cooling during normal reactor shutdown, and long term circulation/ recovery and cooling following a LOCA.

During normal shutdown cooling, only one division, with one pump in operation, is required to provide the adequate cooling. Flow is from the reactor outlet header to the inlet header via the LTC pumps, and LTC heat exchangers. For LOCA operation, both divisions operate with each division recovering water from the respective reactor building sump. During this mode, two of the four pumps are used and both heat exchangers are used to cool the recovered water before returning into the reactor inlet headers.

4.9.1.1 Electrical Power Supply

Control and instrumentation are powered from Class II power supplies. All components required to operate to allow emergency coolant injection (ECI) are supplied from Class II power. The LTC pumps and the valves required to operate for long-term recirculation phase are supplied from Class III power.

4.9.1.2 Cooling Water Supply

The two LTC heat exchangers and the four LTC pump motor coolers are supplied by the Recirculating Cooling Water (RCW) system.

4.9.1.3 Seismic/Environmental Qualification

The entire ECCS is qualified to DBE. An earthquake is not considered to occur within 24 hours after LOCA, for purposes of ECI and LTC analysis.

The systems and components of ECCS are qualified for a LOCA.

4.9.1.4 Operator Action

All actions required for this system related to safety analysis following initiation of a ‘LOCA Signal’ are fully automatic. All necessary actions of the ECCS equipment initiated by automatic

control can also be initiated manually from the main and secondary control rooms. The transition from ECI system to LTC system is also automatic.

4.9.2 Initial Conditions

	Design Value	Analysis Value	Comments
ECCS	Poised	Poised	
ECI Water Tanks Pressure Capacity High temperature limit for water in ECI tanks	 4.9 MPa (g) - 5.2 MPa (g) 2×170 m^3 water: 120 m^3, gas: 50 m^3 41°C	 4.9 MPa (a) 2×170 m^3 water: 120 m^3, gas: 50 m^3 41°C	 Design value from Reference [32]

4.9.3 Equipment Credits

	Design Value	Analysis Value	Comments
LTC Pumps Number of pumps Head Flow rate	 4 70 m 332 L/s	 2 70 m 332 L/s	 Design value from Reference [2]
LTC Heat Exchangers Number of heat exchangers Heat removal (per heat exchanger)	 2	 2	 Design value from Reference [33]
One-way rupture discs Number of discs Differential pressure to burst rupture discs	 2 0.52 MPa (d)	 2 0.52 MPa (d)	 All discs available to burst depending on burst pressure. Design value from Reference [2]

	Design Value	Analysis Value	Comments
ECI Water Tanks Number of water tanks	2	2	Design value from Reference [32]
Time for ECI Injection valves to fully open	< 20 seconds	20 seconds	
ECI tank level to signal completion of ECI injection	2.09 m	2.09 m	
Time for ECI Injection valves to fully close (following injection)	< 20 seconds	20 seconds	
LTC (level to signal) injection valve signal to open	2.09 m	2.09 m	
Time for ROH interconnect valves to fully open	20 seconds	20 seconds	

4.9.4 LTC Shutdown Cooling Mode

	Design Value	Analysis Value	Comments
Long Term Cooling Heat Exchangers			Cooled by Recirculated Cooling Water (RCW) system
Type	Shell & tube		Design values from Reference [34].
Quantity	2	2	
Capacity	100%	100%	
Operating Pressure at Inlet (RCW pressure)	~440 kPa (g)	~440 kPa (g)	
Inlet Temperature	30°C	30°C	
Long Term Cooling Pumps			Design values from Reference [34]
Type	Vertical, centrifugal		
Quantity 3435-P1 to -P4	4	4	

	Design Value	Analysis Value	Comments
Capacity	100%-Normal 50%-Post LOCA		
Intact HTS Cooling Lines Isolation Valves Quantity	8	8	The valves are normally closed and are opened to provide cooling after reactor shutdown
Post-LOCA Cooling Line Isolation Valves Quantity	2 check valves 4 motorized isolation valves		Each post-LOCA injection line is isolated from the HTS by the ECI rupture disc and a check valve inside the reactor building and two parallel motorized valves located outside the reactor building
Sump Isolation Valves Quantity	2 check valves 2 butterfly valves	2 + 2	Normally close butterfly valves with electrical actuation
Pump/Heat Exchanger By-Pass Lines Closure Valves	2 motorized valves	2	Normally closed motorized valves
Pump Recirculation Line Closure Valves	2 motorized valves	2	Normally closed motorized valves
Pump Discharge Isolation Valves	2 motorized valves	2	Normally open motorized valve
LTC Heat Exchanger By-Pass Line Closure Valves	2 motorized valves	2	Normally closed motorized valves

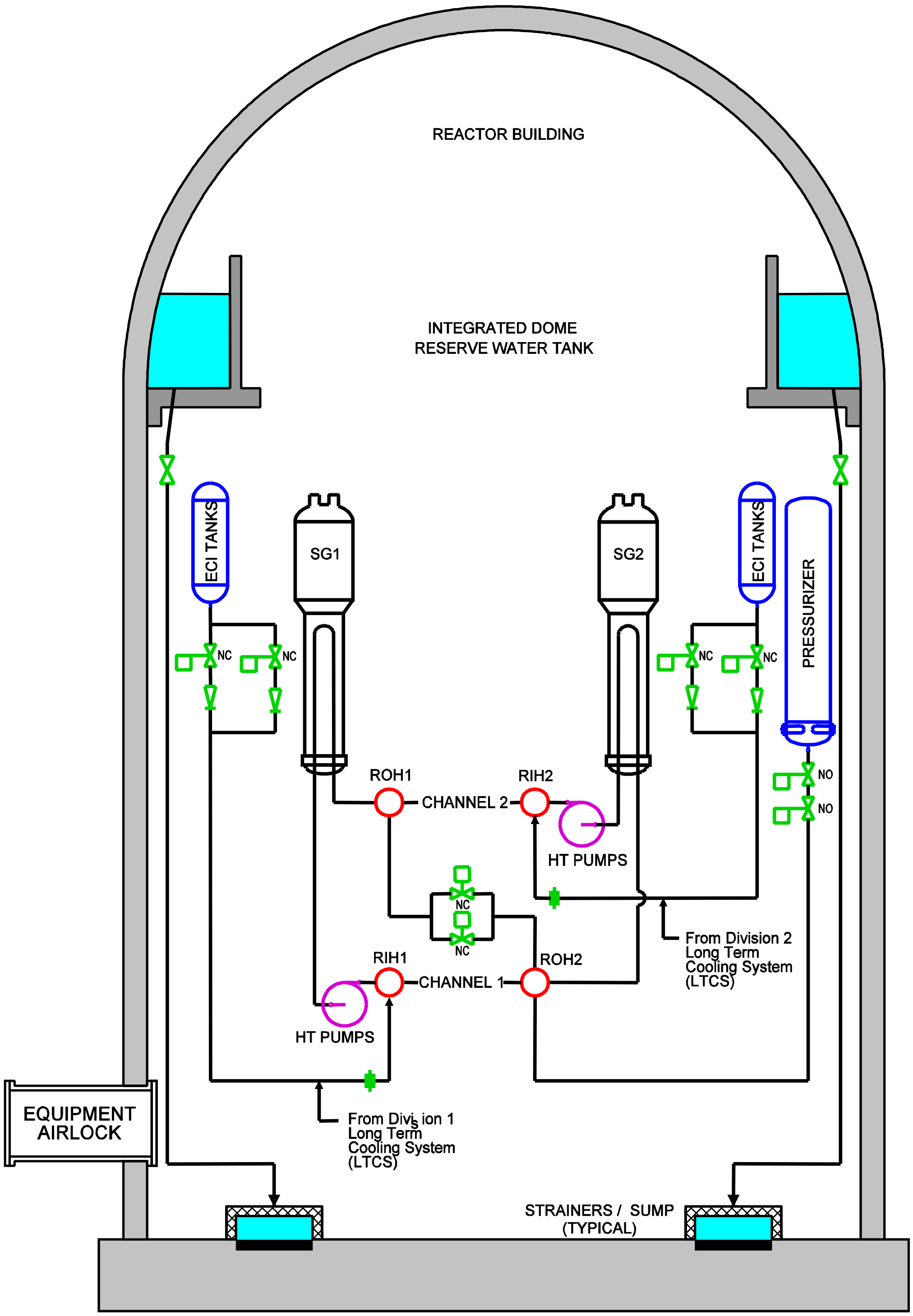


Figure 4-7 Emergency Coolant Injection System Flow Diagram

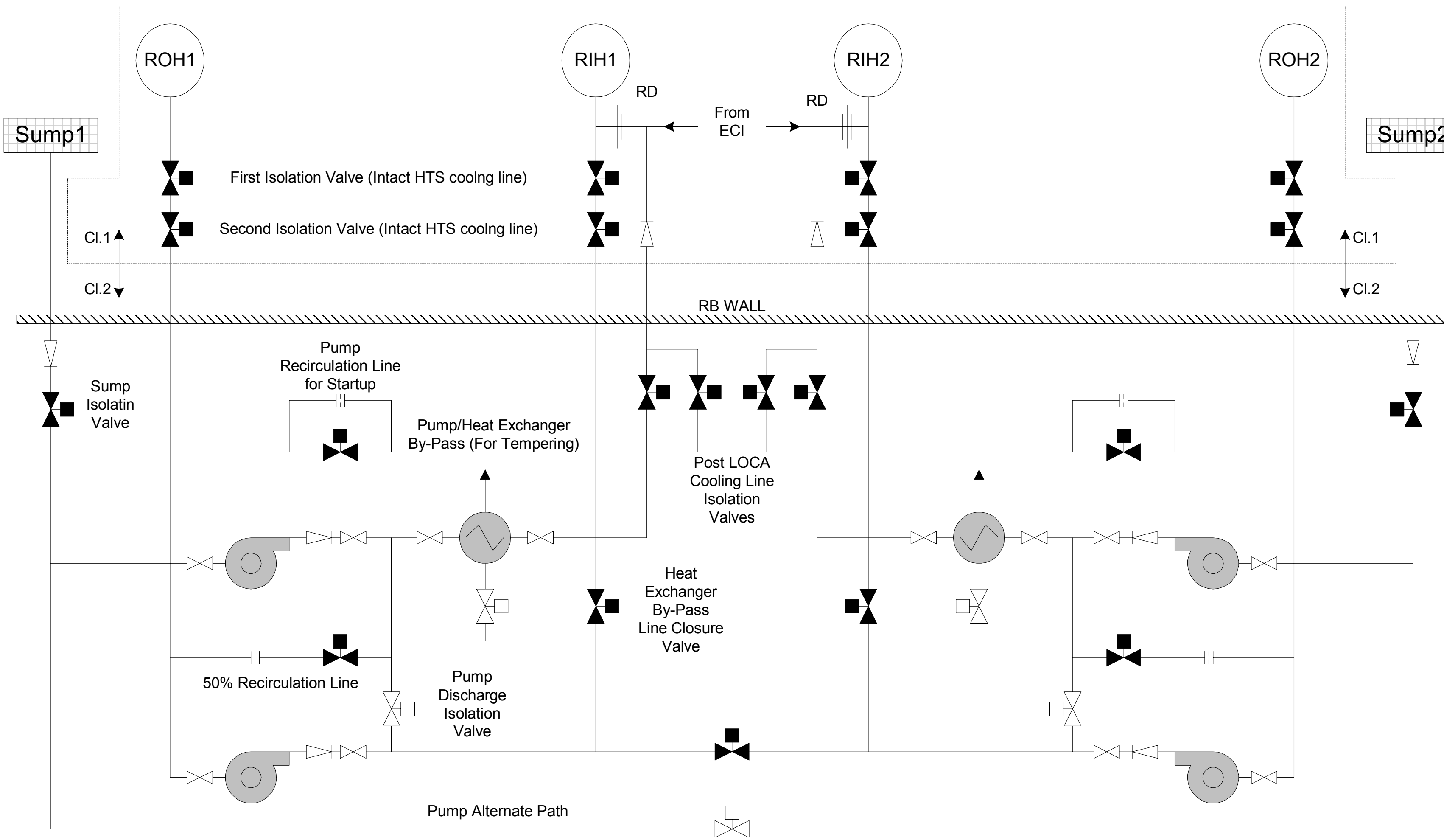


Figure 4-8 Long Term Cooling System Flow Diagram

4.10 Moderator System (ASI 32100)

4.10.1 System Description

The main purpose of the moderator is to provide a medium to slow down high-energy fission neutrons to the appropriate thermal energy level. The Moderator System functions to remove the heat generated within the moderator during reactor operation and maintain the moderator temperature at the appropriate value. The moderator system also acts as a medium for dispersion of reactivity control agents.

The main components of this system are the two 50% heat exchangers, two 100% capacity pumps (one on standby) and a head tank. Refer to Figure 4-9. The moderator temperature is controlled by instrumentation at the calandria outlet, which measures the temperature and feeds back to control the flow of recirculated cooling water to the heat exchangers. The moderator level is measured in the moderator head tank by differential pressure transducers. Sources of moderator heating are given in Table 4-2.

4.10.1.1 Electrical Power Supply

Both main moderator pumps are on the Class III power. Two pony motors are on class II power, which are not in use in normal operation condition.

4.10.1.2 Cooling Water Supply

The moderator heat exchangers are cooled by RCW.

4.10.1.3 Seismic/Environmental Qualification

The moderator system and its auxiliaries are qualified to DBE, Category A, to prevent draining in the event of an earthquake.

This system is environmentally qualified for both in-core and an out-of-core LOCA + LOECC event since the moderator performs as the heat sink when the ECCS is unavailable. The moderator system is also environmentally qualified such that it will not drain because of a MSLB.

4.10.1.4 Operator Action

No operator action related to safety analysis is required for the proper functioning of this system. The operator has the option to replenish the moderator by gravity feed from the reserve water tank in order to maintain moderator circulation and sufficient cooling.

4.10.2 Initial Conditions

	Design Value	Analysis Value	Comments
Moderator System			-The moderator system is operating as designed before the event.
Moderator Isotopic purity	99.90%(nominal)	99.80%	Design value from Reference [2]
Moderator Inlet temperature Outlet temperature Flow rate	 57ºC 80ºC 860 L/s	 57ºC 80ºC 860 L/s	 Design values from Reference [2]
Moderator Cover Gas Cover gas pressure	 71.5 kPa (g)	 71.5 kPa (g)	 Design value from [2]
Head Tank Capacity Operating Temperature	 4.4 m^3 80 ºC	 4.4 m^3 80 ºC	 Design value from [2]
Heat loads Total heat transfer to moderator heat exchangers	 78.9 MW	 80.5 MW	 Design value from [2] Analysis value at 102% full power
Minimum moderator system heat removal capacity	88.6 MW	88.6 MW	Design value from [35]

4.10.3 Equipment Credits

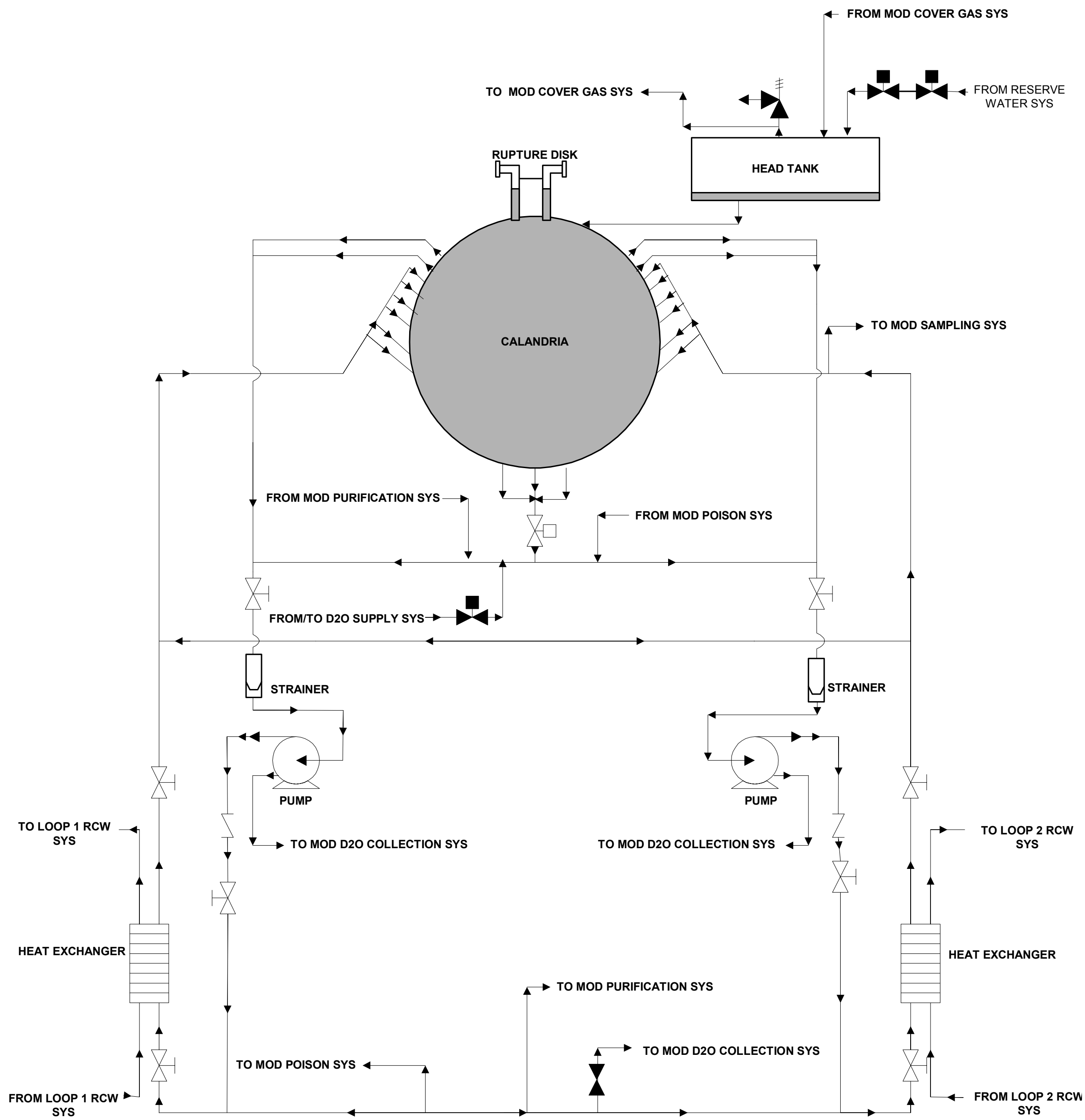
	Design Value	Analysis Value	Comments
Rupture Discs Number	 4	 4	 Design value from Reference [2]
Moderator Pumps Number Rated flow (per pump)	 2 860 L/s	 1 860 L/s	 Design value from Reference [2]
Rated head	55 m	55 m	Design value from Reference [2]

Moderator Heat Exchangers			
Number	2 x 50%	2 x 50%	
Heat transfer (per heat exchanger)	44.3 MW (th)	44.3 MW (th)	Design value from Reference [2]
Service water inlet temperature	30ºC	30ºC	
Service water outlet temperature	56ºC	56ºC	

Table 4-2
Nominal Moderator System Heat Load at Steady State 100% Full Power

Heat Generated in Moderator	54.3 MW
Heat Generated in Reflector	7.3 MW
Heat Generated in Calandria Tubes	9.5 MW
Heat Generated in Guide Tubes, Calandria Structures and Reactivity Mechanisms	4.1 MW
Heat Generated in Calandria Tubesheets	0.5 MW
Fission heat in moderator	75.7 MW
Heat Gained from Fuel Channels	2.8 MW
Total heat in Moderator	78.5 MW
Heat Loss to Moderator Piping	0.3 MW
Pump Energy Appearing in Moderator	0.7 MW
Total heat transfer to moderator heat exchangers	78.9 MW

Note: All data are from Reference [2]



* 4 valves (2 sets of 2 in parallel) from Reserve Water System.

Figure 4-9 Main Moderator System Flow Diagram

4.11 Class IV Power (ASI 53000)

4.11.1 System Description

The typical voltages of Class IV distribution system are shown in the key diagrams in

Figure 4-10 and Figure 4-11, [2]. The Class IV power supply system is capable of the following performance:

- Providing power for all station services in the nuclear generating station, including on-site lighting, pump house and administration building
- Starting the nuclear generating unit from an independent off-site power source
- Across-the-line starting of the last heat transport pump (11 kV), the largest 6.6 kV motor and the largest 415 V motor to be connected with at least 0.8 per unit (pu) voltage at the motor terminals
- Restoring the station services to normal operating conditions on a bus transfer at the 11 kV and 6.6 kV (typical) voltage level.
- Providing system grounding, protective relaying and short circuit capacity of the equipment such as to allow for a system designed for the least number of interruptions of service
- Continuously monitoring the voltage and frequency to ensure the distribution system is capable of performing its functions
- Being the preferred power source for the Class III system

Class IV - AC power, available from the grid/turbine-generator (via the UST/SST), serves as the primary power source to the station. Long-term interruption of Class IV can be tolerated without endangering equipment, personnel or plant safety. Class IV is an AC power system that has no back up from on-site sources and is fully dependent on the availability of the off-site network and the capabilities of the main turbine generator to supply the “house load” on loss of grid caused by events external to the station.

The Class IV distribution system supplies the following typical loads:

- Heat transport circulating pump
- Main feedwater pump
- Condensed circulating pumps
- Condensed extract pumps
- Low volt loads

4.11.1.1 Electrical Power Supply

As stated above, during normal operation the utility’s grid system and/or turbine–generator energize the Class IV buses.

4.11.1.2 Cooling Water Supply

The Class IV distribution system does not have Cooling Water Supply.

4.11.1.3 Seismic/Environmental Qualification

The Class IV power system is neither seismically qualified nor environmentally qualified.

In the event of an earthquake and a subsequent loss of normal plant power, the reactor must be capable of being shut down and the decay heat removed. On loss of Class IV power, the reactor will be automatically shut down.

4.11.1.4 Operator Action

Operator action is not applicable. In the event of a loss of normal Class IV supply, Class III power is automatically restored from the Class III diesel generators.

4.11.2 Initial Conditions

	Design Value	Analysis Value	Comments
Class IV Power Voltages			Normal loads energized

4.11.3 Standard Assumptions

	Design Value	Analysis Value	Comments
Class IV Power			Available before and throughout accident. For a random loss of Class IV supply following an accident (that is, loss of Class IV power is not an initiating event) the connection to the grid is assumed lost when the turbine begins to unload.

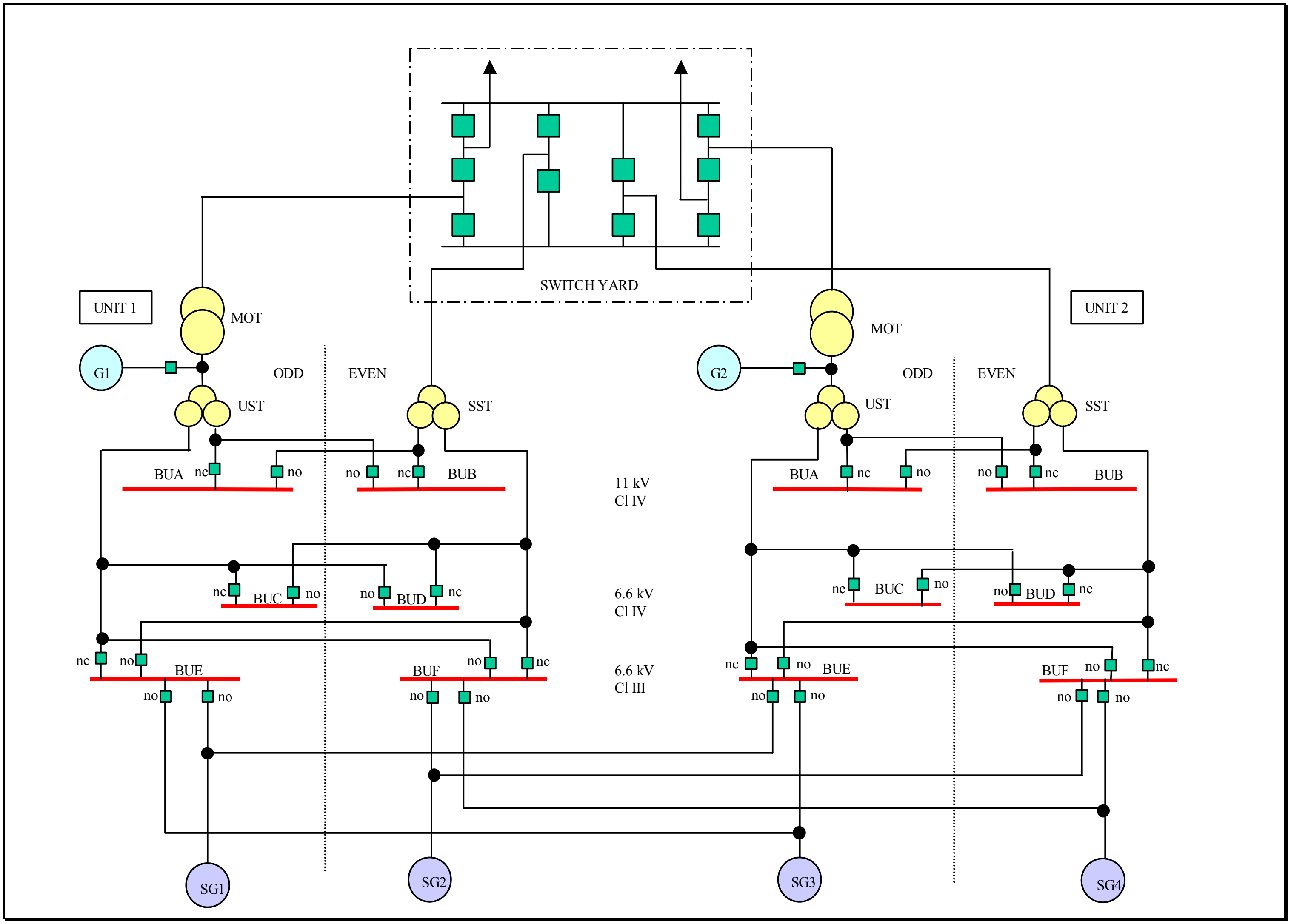


Figure 4-10 Simplified Single Line Diagram – Typical Main Station Connections – Two Unit Station

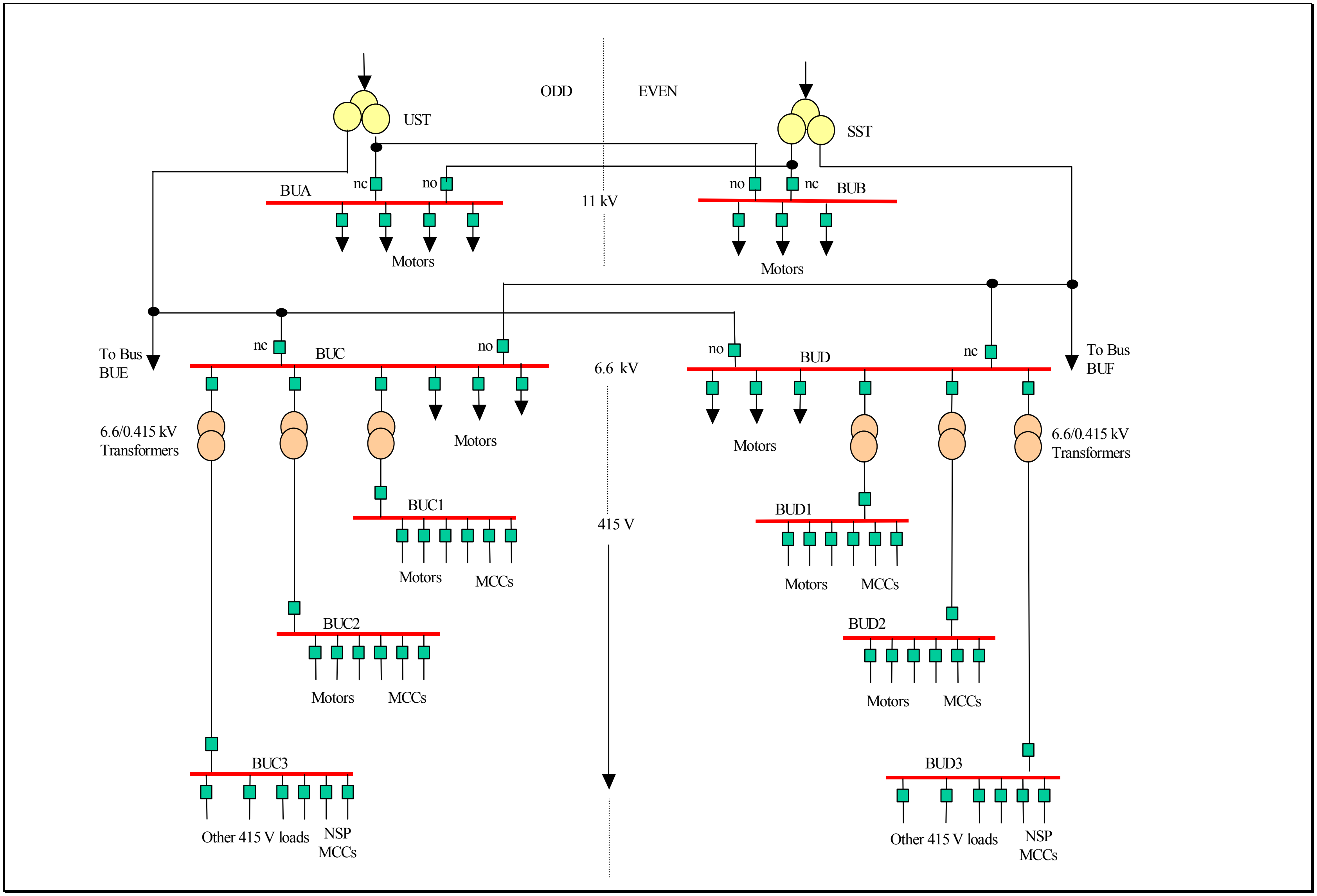


Figure 4-11 Simplified Single Line Diagram – Typical Class IV Electrical Distribution System – One Unit

4.12 Class III power (ASI 53000)

4.12.1 System Description

Class III power, although subject to minor interruptions, is sufficiently reliable to ensure safe plant shutdown and decay heat removal, and to prevent equipment damage, even though the plant may be disconnected from off-site power sources [2].

The Class III power distribution system shall normally obtain power from the Class IV system and the Class III power supply system shall meet the following requirements:

a) The system is capable of providing power to all Class III loads that are required to ensure a safe plant shutdown and continued fuel cooling.
b) The system is capable of performing satisfactorily under abnormal conditions ranging from total loss of Class IV supply with the reactor at full power to a LOCA and loss of Class IV power, when the reactor was producing full power.
c) Capacity to start groups of loads:
   - All sequenced loads on Class III

The standby AC power source(s) shall perform no function during normal station operation. If Class IV power is lost or there is a loss of coolant accident (LOCA) with subsequently loss of Class IV, during normal plant operation, all four DGs (for 2 units, for one unit- 2 DGs) shall start and run automatically and supply power to the affected unit's Class III buses. Only one out of the four DGs are required to supply power to the required loads for design basis events for the affected units.

The standby AC power source(s) would be required to function when all off-site AC power sources are unavailable. By definition, this would be classified as an abnormal operational state. The standby AC power source(s) fulfil the following functional requirements:

1. Each Standby Generator (DG) is capable of performing as a redundant unit.
2. The DG units are available as a source of power if there is a failure of the normal supply. The DGs will not usually be running or be connected to the network unless there has been a detected failure of the normal supply.
3. The standby power generation system is capable of starting and supplying the needs of all required loads during and after specified design basis events.
4. The DGs are located in the dedicated rooms of the DG building. The DGs and their auxiliary systems are designed to withstand or be protected from the applicable external and internal hazards/events.
5. The DGs are capable of performing reliably during the mission time consistent with the safety requirements of the station.

4.12.1.1 Electrical Power Supply

Class III power is supplied from the Class IV system until Class IV becomes unavailable. Subsequently the four backup diesel generators supply Class III power after they have started and have been connected to dead buses. In the event of failure of normal power sources, the diesel generators start automatically. They also start automatically on an LOCA Signal, but are

not connected unless a loss of Class IV power occurs. The diesel-generators come up to speed and accept key loads within 30 seconds and full load within 180 seconds of the start signal. Control power is required to signal the diesel generators to start automatically. They also have “black start” capability.

4.12.1.2 Cooling Water Supply

An independent cooling water system is provided for each standby diesel generator. This is a closed system with an expansion tank. The RCW system is not required to remove heat generated by the DGs.

4.12.1.3 Seismic/Environmental Qualification

Portions of the Class III systems located in RAB, DG building and RSW pump house are seismically qualified.

Qualified equipment is installed such that no damage to the qualified equipment shall result from non-qualified systems during or after a seismic event.

The components of the Class III electrical power system are environmentally qualified to ensure the equipment will successfully perform during and/or after the change of its local environment due to an initiating event.

The qualification process shall consist of:

- Qualification for operation under normal service conditions.
- Qualification for operation under accident conditions and post-accident conditions.
- Replacement of parts or complete components if their qualified life is shorter than the design life of 60 years.

4.12.1.4 Operator Action

The diesel generators will automatically start on loss of Class IV voltage, or an LOCA Signal. If the diesels have successfully run, diesels will feed Class III electrical systems. The standby sets are designed to accept key loads within 30 seconds and full load within three minutes, sequencing is automatic.

4.12.2 Initial Conditions

	Design Value	Analysis Value	Comments
Class III Power Voltages			Diesel generators are not operating prior to the event

4.12.3 Standard Assumptions

	Design Value	Analysis Value	Comments
Class III Power Time for key loads and full loads	30 to 180 s	180 s	Diesel generators start at the time Class IV power is lost or on the LOCA Signal. The standby generator sets are designed to accept key loads within 30 seconds and full load within three minutes. All loads are assumed to be reestablished at (and not before) three minutes.

4.12.4 Equipment Credits

	Design Value	Analysis Value	Comments
Class III Power Number of DGs (Standby Generators)	4 (for 2 units)	1	Only one out of the four DGs are required to supply power to the required loads for design basis events

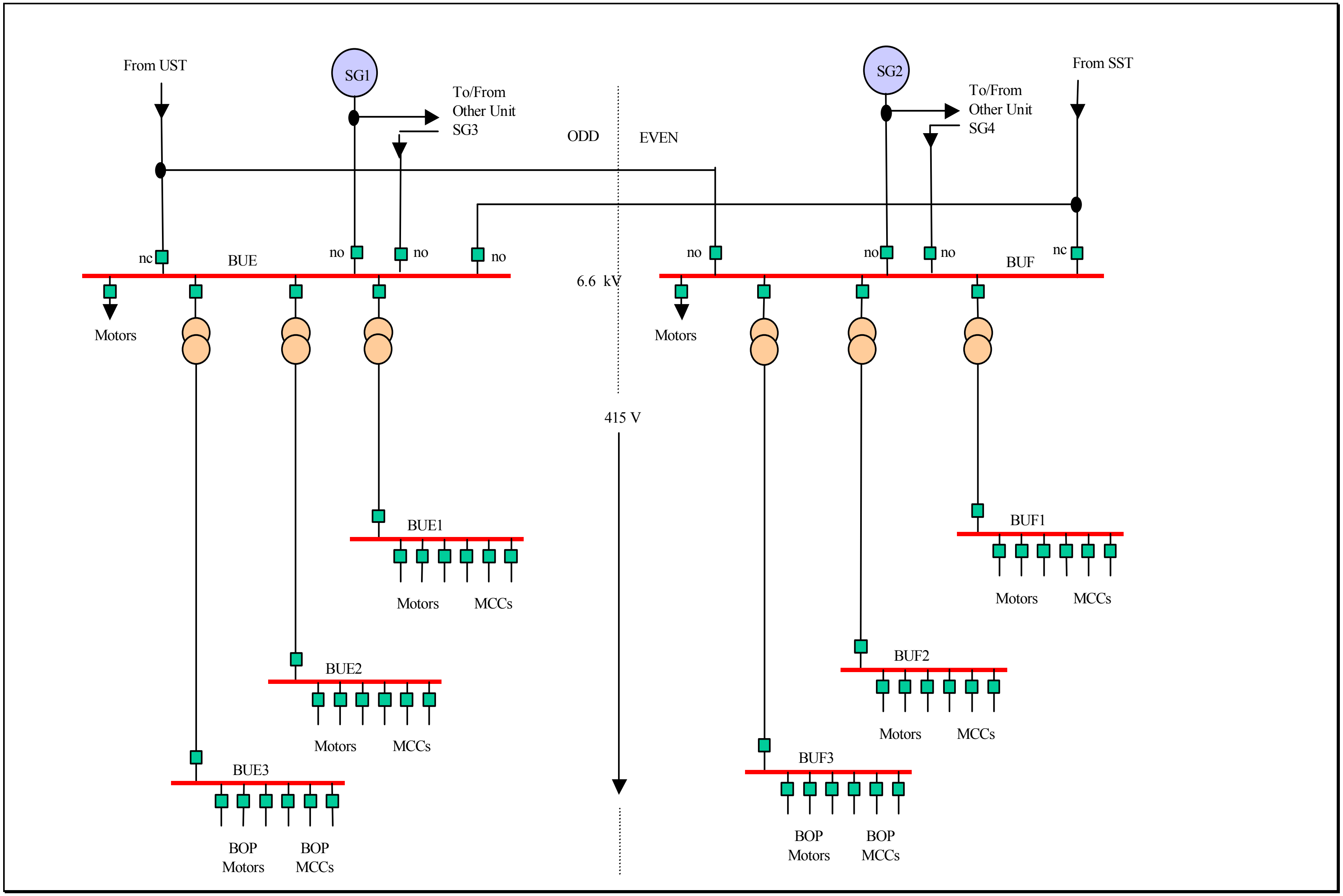


Figure 4-12 Simplified Single Line Diagram – Typical Class III Electrical Distribution System – One Unit

4.13 Class I and II Power (ASI 53000)

4.13.1 System Description

The main components of these systems are the batteries, inverters, distribution panels, rectifiers and associated cabling and accessories [2]. See Figure 4-13. Class I and II loads are not to experience interruption in supply at any time.

4.13.1.1 Electrical Power Supply

Each battery bank is sized to support all the required loads on its bus for a minimum of one hour at the end of the design life of the batteries following a loss of Class III power. In addition, the battery banks connected to post accident monitoring loads are sized to support those loads for a minimum of 8 hours at the end of the design life of the batteries following a loss of Class III power. The reliability of Class II power is directly related to the availability and reliability of class I power. To maintain this capability, a battery monitoring system is supplied for each battery bank that detects battery failure mechanisms and warns the operator or maintenance personnel of abnormal event(s) that may affect the battery availability. There are three battery rooms, one for each Class I power supply bus.

4.13.1.2 Cooling Water Supply

None required.

4.13.1.3 Seismic/Environmental Qualification

The batteries are DBE category B qualified. All batteries are mounted on seismically qualified racks located in battery rooms. All battery racks are seismically qualified to withstand a Design Basis Earthquake (DBE). The batteries, which are continuously charged by rectifiers, shall provide power without interruption for a specified time (typically 1 hour) when Class III is unavailable.

The Class II and I systems are seismically qualified to meet the safety requirements. All areas where these equipments are located shall be seismically qualified.

The components of the Class II and I systems are environmentally qualified, where necessary, to ensure that the equipment will successfully perform during and/or after the change of its local environment due to an initiating event such as LOCA, main steam line break (MSLB), flooding, earthquake or any other design basis accident.

4.13.1.4 Operator Action

Operation of Class II and I power is automatic (no operator action is required).

4.13.2 Initial Conditions

	Design Value	Analysis Value	Comments
Class I and II Power Voltages			System operating nominally before event.

4.13.3 Equipment Credits

	Design Value	Analysis Value	Comments
Class I and II Power			Analysis assumes these systems are always available, except for those loads, which are not seismically qualified following an earthquake.

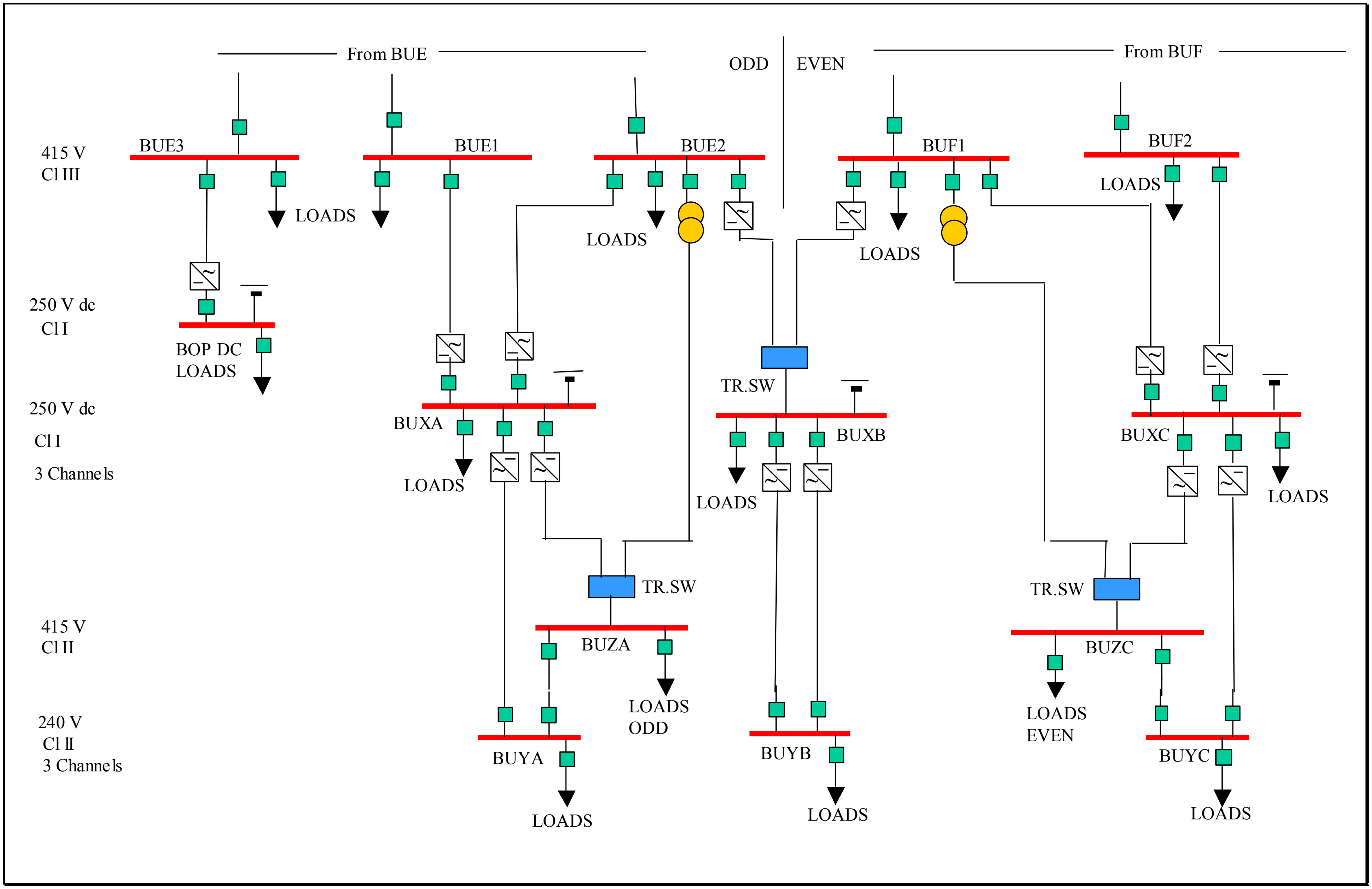


Figure 4-13 Simplified Single Line Diagram – Typical Electrical Distribution System – UPS – One Unit

4.14 Recirculated Cooling Water System (ASI 71340)

4.14.1 System Description

The RCW system is a closed loop system that provides dematerialized cooling water to all the Nuclear Steam Plant (NSP) heat loads [2]. It also supplies cooling water to the Balance of Plant (BOP) loads. This system is comprised of two redundant closed loop divisions. Both divisions have identical equipment and operate during normal operation. In the event that one division is not available, the other division is sufficient to cool the plant in a safe shutdown state. To improve overall reliability, the two RCW divisions are interconnected at the suction and discharge of the Division 1 and Division 2 RCW pumps as shown in Figure 4-14.

4.14.1.1 Electrical Power Supply

The four RCW pumps (two pumps for each division) are on Class III power.

4.14.1.2 Cooling Water Supply

Cooling water is supplied to various components as listed below:

- Moderator pump motors and heat exchangers
- Heat transport pump motors
- Heat transport pressurizing pumps
- Long term cooling pumps and heat exchangers
- Fuelling machine auxiliaries and heat exchanger
- Reactor building coolers
- Spent fuel storage bay coolers
- End shield cooling heat exchangers

4.14.1.3 Seismic/Environmental Qualification

This system is DBE qualified to provide cooling water to the long- term cooling system and to the containment cooling system.

This system is environmentally qualified to function after a MSLB and LOCA.

4.14.1.4 Operator Action

No operator actions, related to safety analysis, are required for this system.

4.14.2 Initial Conditions

	Design Value	Analysis Value	Comments
Supply temperature Return temperature Heat Transferred – (total)	41.8°C 30°C 168 MW	41.8°C 30°C 168 MW	From Reference [2]

4.14.3 Standard Assumptions

	Design Value	Analysis Value	Comments
RCW Pumps No of RCW Pumps Flow rate per pump Pump head	 4 2000 kg/s 48 m	 4 2000 kg/s 48 m	 From Reference [2]
RCW Heat Exchangers Quantity Flow rate Heat transferred (normal operation)	 8 (six in 100% power operation) 506 L/s 168 MW	 8 (six in 100% power operation) 506 L/s 168 MW	From Reference [2] The heat removal capability for accidents depends on the accident scenario.

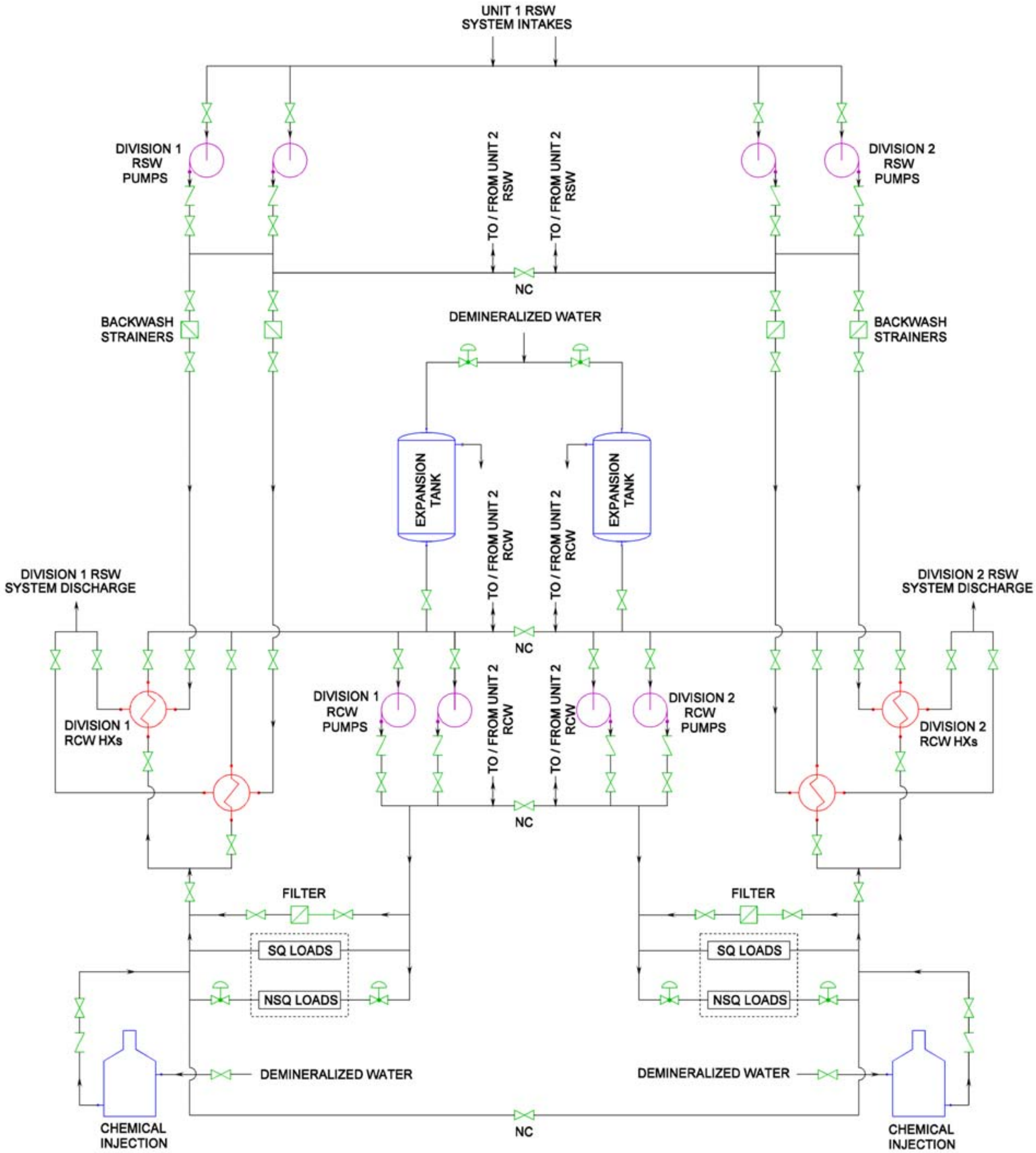


Figure 4-14 Service Water Systems Flow Diagram

4.15 Raw Service Water (RSW) System (ASI 71310)

4.15.1 System Description

The Raw Service Water (RSW) system disposes of the heat from the RCW system to the ultimate heat sink [2]. The RSW system is an once-through system, which is used for cooling the closed loop RCW system. The RSW system consists of two independent divisions, Division 1 and Division 2 with each division having identical equipment. During normal operation, both RSW divisions run. To improve overall reliability, the two RSW divisions are interconnected at the discharge of the Division 1 and Division 2 RSW pumps. Figure 4-1 presents a schematic of this system.

4.15.1.1 Electrical Power Supply

The four RSW pumps (two pumps for each division) are on Class III power.

4.15.1.2 Cooling Water Supply

The RSW pumps take cooling water from the environment (lake, river or ocean) and primarily cool the RCW system heat exchangers.

4.15.1.3 Seismic/Environmental Qualification

The Raw Service Water (RSW) system is seismically qualified to DBE Category B, to provide cooling to the recirculated cooling water system. The RSW pumphouse, with all contained systems, is seismically qualified to DBE Category A qualification.

No environmental qualification is required for Raw Service Water (RSW) system.

4.15.1.4 Operator Action

No operator action required related to safety analysis.

4.15.2 Initial Conditions

	Design Value	Analysis Value	Comments
Fluid	Sea/River/Lake water	Sea/River/Lake water	From Reference [2]
Operating temperature	25.5°C	25.5°C	
Operating pressure (nominal)	0.25 MPa (g)	0.25 MPa (g)	

4.15.3 Standard Assumptions

	Design Value	Analysis Value	Comments
RSW Pumps			
No of RSW Pumps	4	4	
Flow rate per pump	2200 kg/s	2200 kg/s	From Reference [2]
Pump head	23 m	23 m	

4.16 Main Feedwater System (ASI 43230)

4.16.1 System Description

The feedwater system provides controlled feedwater flow to maintain the required steam generator levels over the full operating power range [2]. The main feedwater system consists of three 50 percent main feedwater pumps and two auxiliary feedwater pumps. The auxiliary pumps provide water to the steam generators at a rate sufficient to remove decay heat from the reactor core in the event of complete loss of the main feedwater pumps and consequential reactor shutdown. A simplified flowsheet of the main feedwater system is shown in Figure 4-15.

4.16.1.1 Electrical Power Supply

Class IV power drives the three 50 percent main feedwater pumps. Class III electrical power drives the two auxiliary feedwater pumps.

4.16.1.2 Cooling Water Supply

The main feedwater pump motors are air cooled or cooled by RCW with a provision to allow direct air-cooling. No cooling water is required for the two auxiliary feedwater pumps.

4.16.1.3 Seismic/Environmental Qualification

The main feedwater system is located in the turbine building except for the portion downstream of the last heaters, which pass through the reactor auxiliary building and connect to the steam generators located in the reactor building. The portion of the feedwater piping within the reactor building is seismically qualified to ensure that the emergency feedwater, delivered by gravity from the reserve water tank, can function during and following a design basis earthquake.

This system is environmentally qualified to provide feedwater to the steam generators after a MSLB or a small LOCA.

4.16.1.4 Operator Action

No operator action is needed since the auxiliary feedwater pumps start automatically following a loss of main feedwater pumps.

4.16.2 Initial Conditions

	Design Value	Analysis Value	Comments
Main Feedwater Pumps			
Quantity	3	2	
Fluid	Demineralised Water	Demineralised Water	From Reference [2] Analysis value is based on 102% full power
Flow Rate	638 L/s	650 L/s	
Head	780 m	780 m	

	Design Value	Analysis Value	Comments
Operating Temperature (at Feed Water Pump suction)	159°C	159°C	
Final feedwater temperature [ºC]	218°C	220°C	

4.16.3 Equipment Credits

	Design Value	Analysis Value	Comments
Auxiliary Feedwater Pump Quantity	2	1	From Reference [2]
Head	780 m	780 m	
Flow rate	43 L/s	43 L/s	

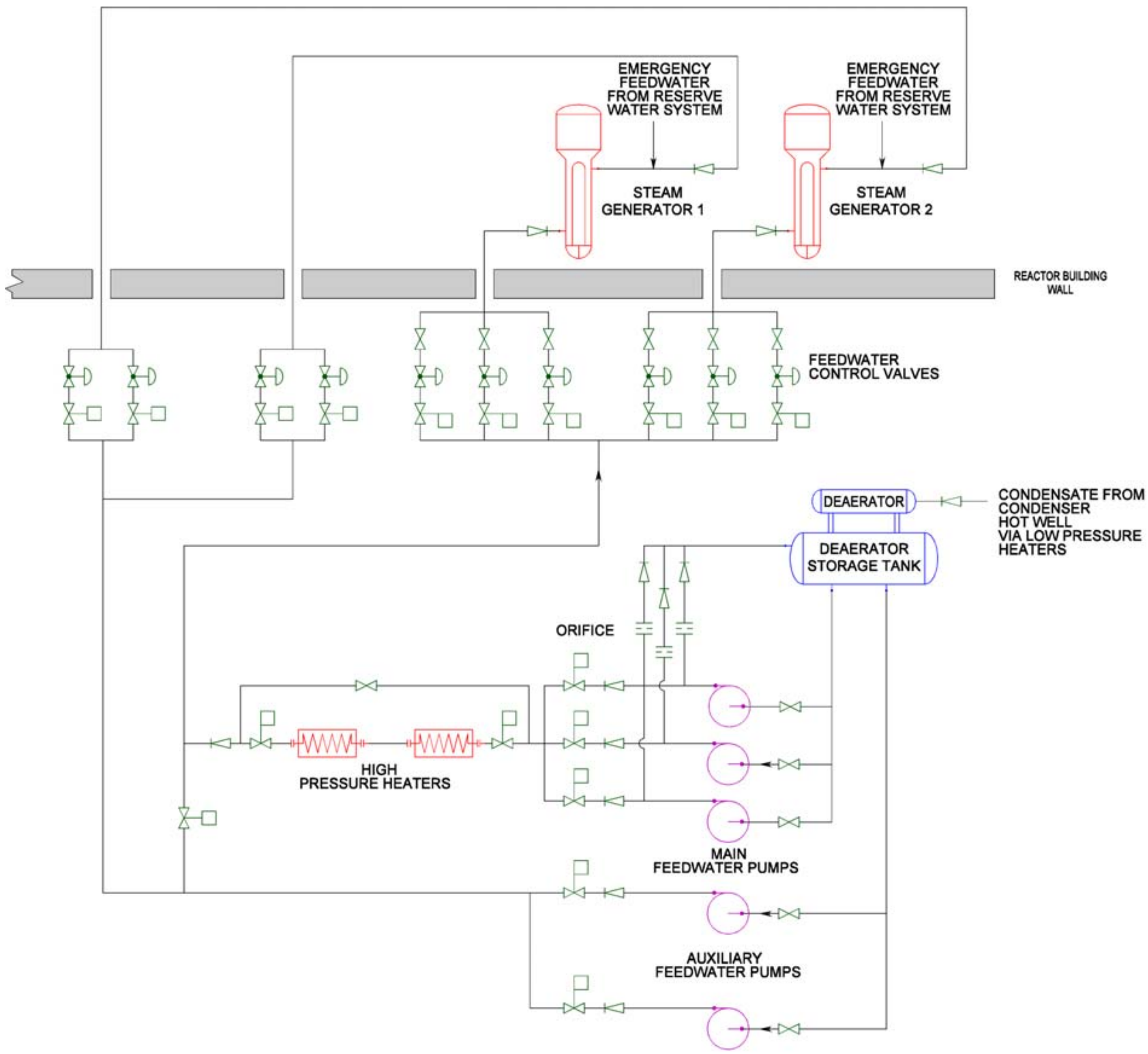


Figure 4-15 Main Feedwater System Flow Diagram

4.17 Instrument Air System (ASI 75100)

The compressed air system supplies instrument air, service air, and breathing air, which is filtered and dried as required [2]. The system provides backup receivers capable of supplying instrument air requirements during changeover from Class IV to Class III power.

Back-up local air tanks are capable of supplying instrument air during accident conditions or on the loss of power supply where required. To reduce or prevent post-accident pressurization of the reactor building, the system is capable of isolating instrument air, breathing air, and service air supplies to the reactor building.

4.17.1 System Description

The compressed air system, located in the reactor auxiliary building, provides compressed air for instrument, service and breathing air requirements throughout the plant.

Two main headers, identified as EVEN and ODD, supply the instrument air system, leading from the EVEN and ODD supply tanks in the reactor auxiliary building to the reactor building, and by two separate EVEN and ODD headers to areas outside the reactor building.

4.17.1.1 Electrical Power Supply

The compressors are on the Class III buses.

4.17.1.2 Cooling Water Supply

Recirculated Cooling Water System (RCW) will be used to cool compressors.

4.17.1.3 Seismic/Environmental Qualification

Qualified local air tanks to maintain air supplies, to all qualified systems to perform essential safety functions shall be qualified to DBE Category B.

The post LOCA instrument air tanks will function to provide back-up air supplies to qualified systems following a LOCA or a MSLB.

4.17.1.4 Operator Action

On loss of instrument air, there will be seismically qualified automatic backup to provide air to the following users (among others):

- MSSVs outside R/B
- Pneumatic Valves In ECC valve stations inside R/B
- Double pneumatic valves in the New Fuel Transfer inside R/B for post LOCA containment isolation
- Double pneumatic valves in Irradiated fuel transfer inside R/B for post LOCA containment isolation.
- Containment isolation dampers of the R/B for post LOCA containment isolation.
- Moderator HX TCVs are located in RAB.
- LTC HX TCVs (to be located in RAB)

4.17.2 Initial Conditions

	Design Value	Analysis Value	Comments
Instrument air Compressors			Common compressors for Service, Breathing, & Instrument Air Systems
Quantity	2	1	
Flow rate (STP)	4100 m^3/h	100 m^3/h	
Design Pressure	1.035 MPa (g)	1.035 MPa (g)	
Suction Pressure	Atmospheric	Atmospheric	
Discharge Pressure	1.035 MPa (g)	1.035 MPa (g)	
Operating Temperature	40-55°C	40-55°C	
Main Air Receivers			
Quantity	2	2	
Volume	15 m^3 (each)	15 m^3 (each)	

4.17.3 Standard Assumptions

	Design Value	Analysis Value	Comments
Instrument air			The capacity of an instrument air system will be based on the total requirements of all connected loads, assuming all instruments operate simultaneously. This produces a conservative estimate since it is extremely unlikely that all instrument air users will operate at the same time. At the present time the number of pneumatic valves and devices has not been fully established for ACR-700.

4.18 Reserve Water System (ASI 34340)

4.18.1 System Description

The ACR-700 design includes a Reserve Water System with Reserve Water Tank, shown in Figure 4-16. The tank, which is located at a high elevation in the reactor building, provides an emergency source of water to the containment sumps for recovery by the LTC system in the event of a LOCA to ensure net positive suction head for the LTC pumps. In addition, the tank provides emergency makeup water by gravity to the steam generators (emergency feedwater), moderator system, shield cooling system, SG and Dome Local Air Coolers, and the heat transport system if required [2].

The reserve water system is a backup water system and hence is non–operational under normal circumstances.

The destination of the reserve water is determined by valving. It can be used to recover the water from small leaks in either the moderator system or the heat transport system and deliver it back to the leaking system.

Specific functions of the reserve water system are [2]:

a) Make–up to the Shield Cooling System

   The reserve water system provides demineralized light water make–up for loss of shield cooling inventory events such as piping, end shield, and shield tank leaks. It may also be used for make–up during severe accidents after reactor shutdown if both the HTS and moderator coolants are unavailable as heat sink.

b) Make–up to the Heat Transport System

   The heat transport system contains light water and is provided with a dedicated pressure and inventory control system. The reserve water system also provides a backup supply of demineralized light water for heat transport system make–up if other sources are unavailable. During normal operation, the tank is isolated from the heat transport system by means of a valve station with two normally closed valves in series. These valves are opened only when all sources of light water in the station have been depleted.

c) Make-up to the Moderator System

   The reserve water system provides a means of moderator inventory make–up in the event of a leak exceeding normal D_2O make-up capability. The gravity fed moderator make–up from the reserve water tank is connected to the moderator header tank.

d) Feedwater to the Steam Generators

   In the event of a loss of Main Feedwater and Auxiliary feedwater to the steam generators, the emergency feedwater will provide a heat sink to Steam Generators after they are depressurized. The reserve water tank provides emergency feedwater. The steam generators are fed by gravity after they have been depressurized.

   A potential containment breach exists from the RB atmosphere to the RWT, to the secondary side of the SGs, to the atmosphere, if the RWT is valved into the SGs. To prevent this from happening, double isolation valves are provided.

e) Long Term Cooling System (LTC) Make-up

On Emergency Coolant Injection (ECI) initiation signal, the high pressure, gas-driven water injection phase begins and the isolation valves in the line from the reserve water tank are opened to discharge the demineralized water contents onto the floors of the fuelling machine vaults. This water, together with the lost coolant itself, provides sufficient head for the long term cooling system (LTC) pumps for the long-term recovery phase.

f) Local Air Coolers Make-up

On loss of RCW, SG vault and Dome local air coolers are supplied with gravity-fed cooling water from the reserve water tank. This is done in order to avoid water hammering in local air coolers when Class IV power is lost and then restored.

g) ECI Water Tanks Initial Fill-up

The ECI system does not operate during normal reactor operation. During normal reactor operation, the piping from the high-pressure ECI water tanks to the injection valves up to the rupture discs is filled with H_2O. The Reserve Water System provides this initial inventory.

4.18.1.1 Electrical Power Supply

Not applicable.

4.18.1.2 Cooling Water Supply

The RWS does not demand cooling water supply.

4.18.1.3 Seismic/Environmental Qualification

Not applicable.

4.18.1.4 Operator Action

The safety system responses are automated to the extent that no operator action is needed for a minimum of eight hours following most design basis accidents.

4.18.2 Initial Conditions

	Design Value	Analysis Value	Comments
RWS **Reserve Water Tank** Capacity	3000 (2500 H_2O + 500 vapour space) m^3	3000 (2500 H_2O + 500 vapour space) m^3	The reserve water system is a backup water system and hence is non–operational under normal circumstances. Design value from Reference [2]

4.18.3 Equipment Credits

	Design Value	Analysis Value	Comments
Time for RWT Valves to Sump to Fully Open	< 20 seconds	20 seconds	Design value from Reference [2]

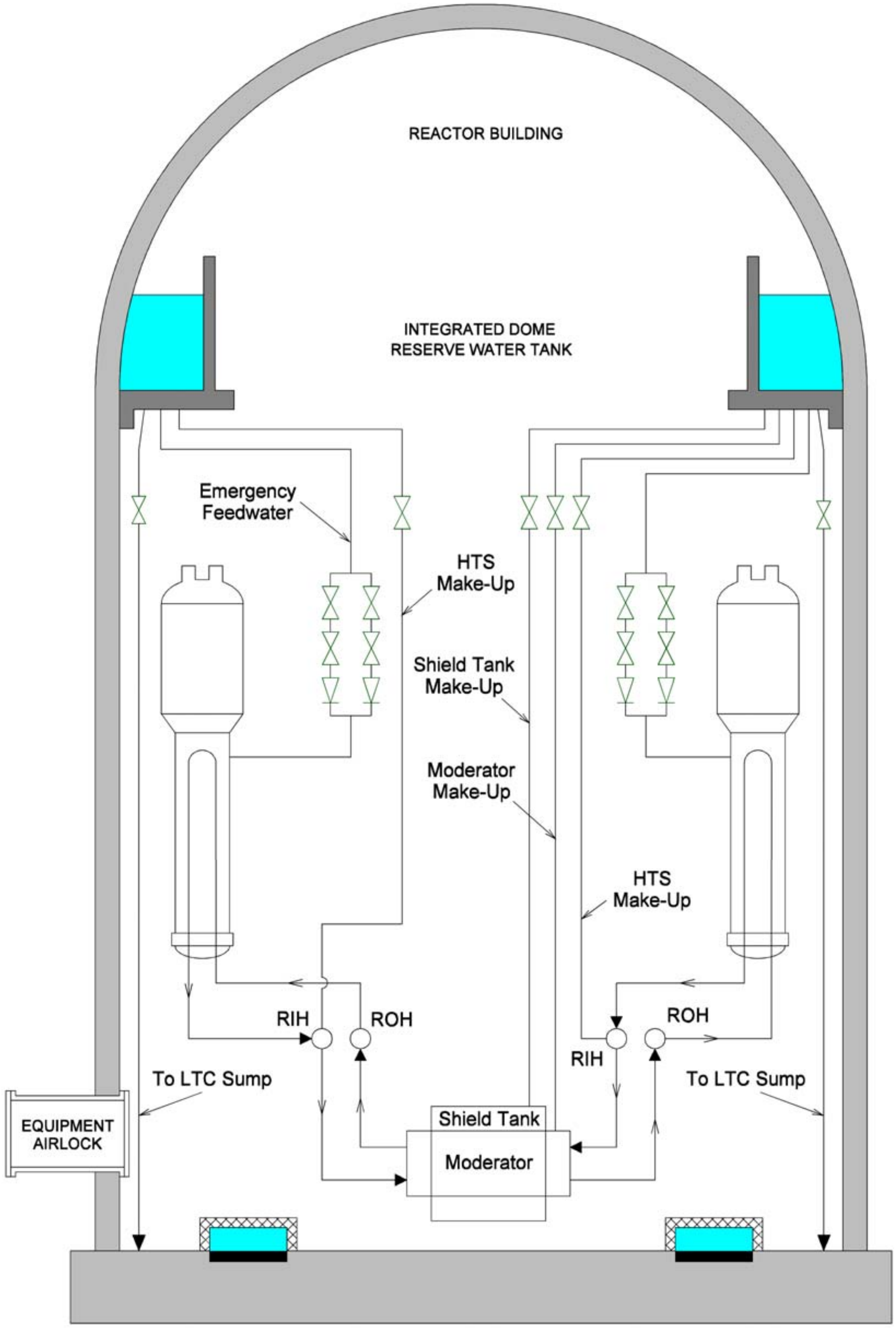


Figure 4-16 Reserve Water System Flow Diagram

4.19 Equilibrium Core State (ASI 03310)

4.19.1 System Description

Refuelling begins when the excess reactivity in the core, due to the fresh fuel used in the first loading, falls to a small value. After refuelling has taken place for approximately 400 to 500 full power days, the core reaches its equilibrium state. It is characterized by a relatively unchanging core configuration in which the global power and burn up distributions do not vary significantly over time. The burnup of discharged fuel, and the fuelling rate of new fuel, become essentially constant.

Operation of the reactor over an extended period with the zone control rods gradually withdrawn to the upper limit of the operating range can be done to compensate the loss of reactivity due to an unplanned interruption in fuelling.

4.19.1.1 Electrical Power Supply

Not applicable.

4.19.1.2 Cooling Water Supply

Not applicable.

4.19.1.3 Seismic/Environmental Qualification

Not applicable.

4.19.1.4 Operator Action

No operator actions related to safety analysis.

4.19.2 Initial Conditions

	Design Value	Analysis Value	Comments
Total Heat Transferred to SG **Fission Heat Generated In Fuel Channel**	1982 MW (th) 1977.6 MW (th)	2021 MW (th) 2017.1 MW (th)	The reactor is assumed to be at full power at the time of the accident. A 2% uncertainty in bulk reactor power measurement is allowed; hence, the initial reactor power used in the analysis is 102% full power. The channel powers are a calculated quantity and are tabulated in Appendix A. The reactor power will be derated from the maximum dependent upon the number of adjuster banks withdrawn, the analysis value will be calculated at a later date. The channel power map for shim operation will be calculated at a later date. Design values from Reference [2]

	Design Value	Analysis Value	Comments
Nominal Maximum Channel Power	7300 kW (th)	8000 kW (th)	Design values from Reference [2]
Nominal Maximum Bundle Power	851 kW (th)	940 kW (th)	Design values from Reference [2]
Maximum Instantaneous Linear Element Rating	52 kW/m	57 kW/m	Design values from Reference [2]
Core Configuration			- For the equilibrium core configuration, the shutoff rods and MCAs are out of core, the adjusters are in core, and the zone controllers are 50% inserted - Zone control rod positions and the resultant core state will be determined at a later stage in the analysis
Fuel Management			- The design fuel management scheme is 2-bundle shift over the whole core - Normal fuelling has been interrupted (unplanned)
Fission Products			For analysis, xenon is spatially distributed at its steady state concentration for the assumed initial reactor power: other saturating fission products at equilibrium concentrations are uniformly distributed
Flux Tilt	+/- 3% maximum	+/- 3%	Under normal equilibrium full power operation, side-to-side, top-to-bottom, and end-to-end flux tilts are controlled to within ± 3% [2].
Delayed Neutrons			For analysis, they are at steady state, equilibrium values.
Long Shutdown Condition			For analysis: All I completely decayed, All Xe completely decayed, All Rd^{105} set to zero, All Pm^{109} decayed to Sm^{149}, and all Np^{239} decayed to Pu^{239}

	Design Value	Analysis Value	Comments
Pressure Tube Creep	End-of-life creep	Up to End-of-life creep	Refer to fuel channel section.
Channel Sag	End-of-life	End-of-life	Refer to fuel channel section.
Moderator Poison			
Fuelling ahead	0 to 2 mk	2 mk	For analysis up to 2 mk of moderator boron may be added to compensate for excess reactivity due to future fuelling.
Long shutdown condition		Approx. 30 mk	Must add poison to compensate for excess reactivity due to the decay of fission products.
Moderator Purity	99.90 wt%	99.80 wt%	Design values from Reference [2]
Coolant Purity	100 wt%	100 wt%	Nominal design value for ACR.

4.20 Pre-Equilibrium Core State

4.20.1 System Description

The pre-equilibrium core states exist from initial criticality until the reactor reaches an equilibrium state some time after refuelling has begun. The reactor is initially loaded with fresh fuels having different enrichments in strategic locations in order to achieve a flattened power distribution similar to that in the equilibrium core. Negative reactivity is added in the form of moderator poison, boron, to make the reactor exactly critical. As the fuel is burned up, the boron is removed at a rate to maintain criticality.

4.20.1.1 Electrical Power Supply

Not applicable.

4.20.1.2 Cooling Water Supply

Not applicable.

4.20.1.3 Seismic/Environmental Qualification

Not applicable.

4.20.1.4 Operator Action

Not applicable.

4.20.1.5 Initial Conditions

	Design Value	Analysis Value	Comments
Initial Reactor Power	100%	102%	
Core Layout			Fuel loading pattern of the fresh core will be determined later. Refer to 4.22
Core Configuration			The assumed core configuration for the pre-equilibrium core state is: the shutoff rods and MCAs are out of core, and the zone controllers are 50% inserted.
Pressure Tube Creep			Refer to 4.22
Channel Sag			Refer to 4.22
Moderator Poison		2.716 ppm for fresh core. 3.76 ppm for plutonium peak.	Core excess reactivity is compensated with moderator poison. Maximum moderator poison concentration occurs at plutonium peak.

	Design Value	Analysis Value	Comments
Steam Generator Condition	Clean	Clean	The SGs are assumed to be unfouled to correspond with the assumption of initial criticality of the reactor.
Flux Tilt	+/-3%	3% side-to-side or 3% bottom-top	Design values from Ref. [2]
Moderator Purity	99.90 wt%	99.80 wt%	Design values from Ref. [2]
Coolant Purity	100 wt%	100 wt%	Nominal design value for ACR.

4.21 Fuel (ASI 37000)

4.21.1 System Description

The ACR uses the 43-element CANFLEX®* fuel bundles. The bundle includes two element sizes. The centre element and inner ring of seven elements have outer diameters of 13.5 mm. The outer two rings with 14 and 21 elements have a smaller outer diameter of 11.5 mm. The centre fuel element contains natural uranium while the remaining 42 fuel elements contain 2.1 wt % ^{235}U enriched UO_2 pellets. A small amount of dysprosium (7.5 wt%) is added to the fuel pellets of the centre fuel element to achieve the reference negative coolant void reactivity [2]. A very thin layer of graphite, CANLUB, covers the inside surface of all sheaths and protects them from fission-product damage. End caps are resistance welded to the sheath extremities to seal the elements. To facilitate leak testing and to improve pellet-to-sheath heat transfer, the void within the fuel elements is filled with unpressurized helium/air prior to end cap welding. Endplates are welded to the end caps to hold the elements in a bundle configuration. Inter-element spacers are brazed to the adjacent elements at their mid-planes to ensure inter-element separations. To enhance the Critical Heat Flux (CHF), special buttons are brazed to the elements at two planes, each one a quarter length from the end. Bearing pads are brazed to the outer elements, at the mid-plane and near the ends, to support the bundle in the fuel channel. The fuel sheaths, end caps, endplates and appendages are made of Zircaloy-4. Each fuel bundle weighs approximately 23 kg.

4.21.1.1 Electrical Power Supply

Not applicable.

4.21.1.2 Cooling Water Supply

The water in the heat transport system provides cooling to the fuel.

4.21.1.3 Seismic/Environmental Qualification

The fuel is qualified to DBE Category A. It must remain intact following a DBE. The fuel must also be designed to cope with ramps in power experienced during refuelling and the fuel sheath must be able to endure the radiation fields in core.

4.21.1.4 Operator Action

Not applicable.

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* CANFLEX® is a registered trademark of AECL and the Korea Atomic Energy Research Institute (KAERI).

4.21.2 Initial Conditions

The following table shows the pertinent fuel initial conditions used in safety analyses.

Description	Outer Elements	Inner Elements
Number of pellets in the fuel element	45	30
Number of fuel elements in the bundle	35	8
Outside diameter of fuel pellet [mm]	10.65	12.58
Density of UO_2 [Mg/m^3]	10.65	10.65
Length of fuel stack in the element [mm]	481.1	481.1
Axial clearance [mm]	2.6	2.6
Diametral clearance [mm]	0.04	0.04
Coolant pressure [MPa]	12.4	12.4
Fuel pellet enrichment [wt.%]	2.1/2.1	2.1/0.71

4.22 Fuel Channel (ASI 31100)

4.22.1 System Description

Each fuel channel assembly consists of a pressure tube, two end fittings, and associated hardware. The pressure tube is connected to an end fitting at each end by a roll-expansion mechanical joint. The zirconium alloy (Zr-2.5% Nb) pressure tube forms the in-core portion of the fuel channel and contains the fuel bundles. The end fittings are the out-core extensions of the pressure tube. They provide connections to the fuelling machine and feeder pipes. The outboard end of the end fitting is sealed by a removable channel closure. The pressure tube contains the fuel bundles and is positioned inside the calandria tube. Spacers in the annulus between the pressure tube and calandria tube separate the two tubes.

The annulus between the fuel channel components and the reactor components is sealed at both ends by bellows. The annulus is supplied with a recirculated dry gas from the Annulus Gas System through tubing connected to the bellows. This annulus gas serves to detect leaks by monitoring the moisture content of the gas, and it provides an insulating and protective atmosphere for the pressure tube.

Feeder pipes connect the inlet and outlet end fittings to the reactor inlet header and reactor outlet header, respectively, to complete the heat transport system loop. Each feeder pipe is connected to an end fitting by a welded connection. Coolant flows into the inlet end fitting, through holes in the liner tube, down through the inlet shield plug, through and around the fuel bundles, through the downstream shield plug, through the liner tube, and out into the downstream feeder. Each end fitting houses a shield plug. The shield plug locates the fuel in the channel and provides some of the required radiation shielding. The shield plug is removed and stored in the fuelling machine during fuel changing. The outboard end face of each end fitting makes a sealed connection with the fuelling machine so that the fuelling machine becomes an extension of the Heat Transport System during fuel changing. The channel closure is removed and stored by the fuelling machine during fuelling and is re-installed in the end fitting before the fuelling machine detaches from the channel.

The fuel channel is designed to achieve a 30-year operating life with a 90% capacity factor [2]. The fuel channel design will accommodate the predicted axial growth due to irradiation of the pressure tube over this period.

The inner surface of the calandria tubes is surface conditioned to promote radiant heat transfer and optimise contact conductance during potential loss-of-coolant accidents.

4.22.1.1 Electrical Power Supply

Not applicable.

4.22.1.2 Cooling Water Supply

The light water in the heat transport system and heavy water in the moderator provide cooling to the fuel channels.

4.22.1.3 Seismic/Environmental Qualification

The fuel channels are qualified to DBE Category A, to prevent flow blockage after an earthquake, this includes the case of a fuelling machine attached to a fuel channel. The fuel channel shall maintain all its structural integrity for the heat transport system coolant during and following a DBE so that fuel channels are not damaged. The effects of stress, irradiation, temperature, hydrogen absorption, corrosion, vibration and any other significant environmental factors on the fuel channels are accounted for.

4.22.1.4 Operator Action

Not applicable.

4.22.2 Initial Conditions

Following tables show the pertinent fuel channel initial conditions and standard assumptions used in safety analyses.

	Design Value	Analysis Value	Comments
Pressure Tube			
Inner diameter	103.38 mm (min. cold)	103.38 mm	Design value from reference [2]
Wall thickness	6.5 mm (min. cold)	6.5 mm	Design value from reference [2]
Length Creep	6270 mm (trimmed for installation) 4.5%	5943.6 mm Up to 4.5%	Analysis value of length for 12 bundles At the end of pressure tube design life [36]
Calandria Tube			
Inner diameter	151 mm	151 mm	Design value from reference [2]
Wall thickness	2.5 mm	2.5 mm	
Length	6040 mm (overall installed)	5943.6 mm	

	Design Value	Analysis Value	Comments
End Fitting			
Inner diameter Wall thickness Length	104 mm 10 mm (min.) 2760 mm (overall)	104 mm 10 mm 2335 mm	Design value from Reference [2] Analysis value of length between feeder and the fuel bundle [37].
Shield Plugs			
Length Estimated Weight	980.7 mm 25 kg	980.7 mm 25 kg	Design value from Reference [2] A shorter shield plug of 890 mm may be used

4.23 Containment (ASI 68400)

4.23.1 System Description

The containment system is designed to form a continuous pressure-confining envelope about the reactor core and the heat transport system in order to limit the release of radioactive material to the environment in the event of an accident. This is achieved through containment isolation, pressure/ activity reduction, hydrogen control and containment monitoring. The containment system includes a steel-lined, pre-stressed concrete perimeter wall and hemisphere dome. Figure 4-17 illustrates selected containment subsystems and components.

The containment barrier is established using a combination of structures, isolation devices, and metallic extensions of the containment envelope. In addition to the basic containment structure (reactor building), the containment system includes airlocks, process penetrations (with automatic isolation where appropriate, in the case of an accident) and electrical penetrations together with subsystems where needed for reducing containment internal pressure, controlling hydrogen concentrations, and limiting the release of radioactive material to the environment following an accident.

The following structures and systems make up the Containment System [2].

A. Containment Envelope

a. Reactor Building
b. Airlocks
c. Containment Isolation System
d. Extensions and Appurtenances
e. Spent and New Fuel Transfer Systems

A.1 Energy Suppression Systems

- Reactor Building Air Coolers

A.2 Atmospheric Control Systems

- Hydrogen Control System

4.23.1.1 Electrical Power Supply

The main containment, ducted and unducted, major SG and Dome Local Air Coolers (LACs), are supplied by Class III power. Seismically qualified power is provided via an interconnect line to the power supply system. Class IV power supplies all other small cooler fans only.

4.23.1.2 Cooling Water Supply

RCW is supplied to all LACs. Dome and SG LACs are supplied from separate divisions of RCW to ensure RCW availability for at least half of DOME and half of SG LACs in the event of accident. The water lines to the SG vault and dome LACs, which are required to operate under LOCA and MSLB, contain no temperature control valves (TCVs) for improved reliability. Other LACs have TCVs, which can fail closed on loss of Class IV power to conserve water so as not to starve the equipment requiring water on Class III power.

During peak summer months, chilled water may be also supplied to a number of Dome and SG vault LACs under normal plant operation. Table 4-3 summarises the LAC capacity.

4.23.1.3 Seismic/Environmental Qualification

The reactor building, containment penetrations and isolation systems, and reactor building ventilation system are qualified for the conditions that result from a MSLB. The main containment local air coolers and associated ducts and the hydrogen control system are qualified for LOCA conditions.

4.23.1.4 Operator Action

The operator may be required to act prior to 8 hours following a clear and unambiguous signal in the Main Control Room. These actions may include manually isolating containment.

4.23.2 Initial Conditions

	Design Value	Analysis Value	Comments
Containment Envelope	Intact	Intact	Unless impaired by assumptions of accident scenario
Ventilation System			
Flow rate	4.5 m^3/s	0-4.48 m^3/s	Design value from ref. [2]
Delay time to close dampers	0.5 - 1 s	1s	Design value from ref. [2]
Containment Atmosphere Conditions			Design value from ref. [2]
Nominal pressure	-0.62 kPa (g)		
Temperature	41-50ºC	25-55 ºC	Depending on analysis objective

4.23.3 Standard Assumptions

	Design Value	Analysis Value	Comments
Containment Envelope			
Building leakage rate	0.2%	0.5%	
Thickness of RB steel liner	6 mm	6 mm	0.2% of the containment volume per day [38]
Thickness of RB concrete wall	1.2 m	1.2 m	

4.23.4 Equipment Credits

	Design Value	Analysis Value	Comments
Containment Heat Loads			Reference [38]
7311-LAC 1 to LAC 8 SG Vault Coolers			Design value from Reference [2]
Type	Air/water heat exchanger	Air/water heat exchanger	
Quantity	8	8	
Fluid	Air	Air	
Total Cooling Capacity	366 kW	366 kW	
Air Flow (per unit)	23 m^3	23 m^3	
Air Inlet Temperature (Normal condition)	50°C	50°C	
Air Outlet Temperature (Normal condition)	36.2°C	36.2°C	
Cooling Fluid	Light water/chilled water	Light water/chilled water	
Cooling Water Inlet Temperature	30 (max)°C	30 (max)°C 32.6°C	
Cooling Water Outlet Temperature	32.6°C	32.6°C	
Cooling Water Flow Rate	34.4 L/s	34.4 L/s	
7311-LAC 9 to –LAC 12 Dome LACs			Design value from Reference [2]
Type	Air/water heat exchanger	Air/water heat exchanger	
Quantity	4	4	
Fluid	Air	Air	
Total Cooling Capacity	175 kW	175 kW	
Air Flow (per unit)	23.7 m^3	23.7 m^3	

	Design Value	Analysis Value	Comments
Air Inlet Temperature (Normal condition)	41°C	41°C	
Air Outlet Temperature (Normal condition)	34.6°C	34.6°C	
Cooling Fluid	Light water	Light water	
Cooling Water Inlet Temperature	30 (max)°C	30 (max)°C	
Cooling Water Outlet Temperature	33.4°C	33.4°C	
Cooling Water Flow Rate	12.3 L/s	12.3 L/s	
Local Air Coolers LAC 13 to LAC 18 Moderator Room LACs Type	Air/water heat exchanger	Air/water heat exchanger	Design value from Reference [2]
Quantity	6	6	
Fluid	Air	Air	
Total Cooling Capacity	35.5 kW	35.5 kW	
Air Flow (per unit)	4.65 m^3	4.65 m^3	
Air Inlet Temperature (Normal condition)	41 °C	41°C	
Air Outlet Temperature (Normal condition)	34.4°C	34.4°C	
Cooling Fluid	Light water	Light water	
Cooling Water Inlet Temperature	30 (max)°C	30 (max)°C	
Cooling Water Outlet Temperature	33.3°C	33.3°C	
Cooling Water Flow Rate	2.51 L/s	2.51 L/s	
7311-LAC 19 to LAC 34 Miscellaneous RB LACs			Design value from Reference [2]

	Design Value	Analysis Value	Comments
Type	Air/water heat exchanger	Air/water heat exchanger	
Quantity	16	16	
Fluid	Air	Air	
Air Inlet Temperature (Normal condition)	30°C	30°C	
Cooling Fluid	Chilled water	Chilled water	
Cooling Water Inlet Temperature	6 to 13 (max) °C	6 to 13 (max) °C	

Table 4-3
Sizing of Major LACs

	ACR-700	
Normal operation		
	Vault/SG Enclosure	**Dome**
Total number of units x Cooling load	8 x 366 kW	4 x 175 kW
Number of operating units	8	4
Heat removed	2928 kW	700 kW
Total heat removed	3628 kW	
Cooling water inlet temperature	30°C	30°C
Cooling water outlet temperature	32.6°C	33.4°C
Cooler inlet air temperature	50°C	41°C
Cooler outlet air temperature	36.2°C	34.6°C
Cooler inlet air flow	23.0 m^3/s	23.7 m^3/s
Cooling water flow per cooler	34.4 kg/s	12.3 kg/s
Total cooling water flow	324 kg/s	

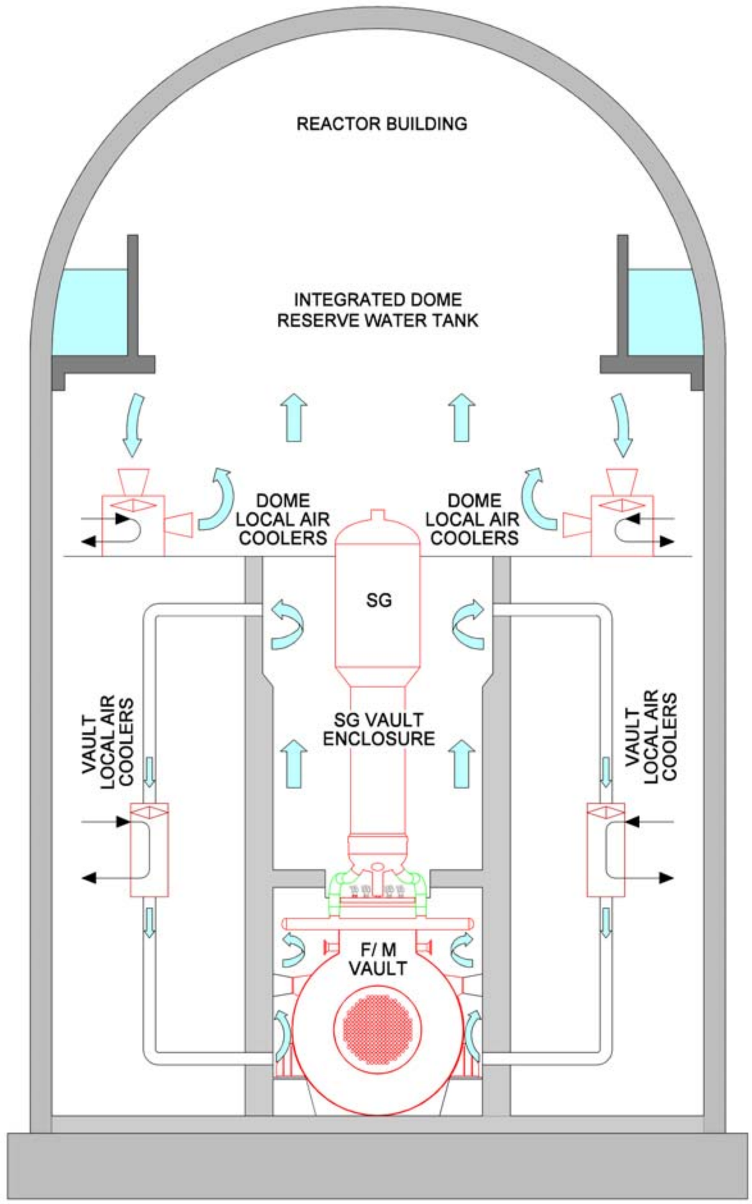

Figure 4-17 Components of Containment

4.24 Atmospheric Dispersion and Public Dose

4.24.1 Analysis Description

1. After an accident at nuclear power plant fission products may be released into containment and then escape to the environment. Prediction of atmospheric dispersion of the plume carrying these radionuclides and the subsequent dose to the public is outlined in CAN/CSA-N288.2 [39]. Based on the meteorological conditions as well as the characteristics of the site, the exposed individual and the plume release, the atmospheric dispersion or dilution factor of the plume is predicted. The time-integrated concentration and ground deposition of each radionuclide is determined using the predicted dilution factor, the activity released from containment and rainfall rate. Finally, the doses from ground deposition and cloud immersion are calculated for each radionuclide.

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6. ACRONYM LIST

ASDV	Atmospheric Steam Discharge Valve
ASI	AECL Subject Index
CHF	Critical Heat Flux
CPPF	Channel Power Peaking Factor
CSDV	Condenser Steam Discharge Valve
DBE	Design Basis Earthquake
ECCS	Emergency Core Cooling System
FSAR	Final Safety Analysis Report
HLR	High Log Rate
LCDA	Limited Core Damage Accidents
LCIVP	Loss of Class IV Power
LISS	Liquid Injection Shutdown System
LOCA	Loss of Coolant Accident
LOECC	Loss of Emergency Core Cooling
LOPIC	Loss of Pressure and Inventory Control
LRV	Liquid Relief Valves
MAPS	Minimum Allowable Performance Standards
MCA	Mechanical Control Absorber
MSLB	Main Steam Line Break
MSSV	Main Steam Safety Valve
NBCC	National Building Code of Canada
PHTS	Primary Heat Transport System
P&IC	Pressure and Inventory Control (System)
PLL	Pressurizer Low Level
PSA	Probabilistic Safety Analysis
RCU	Reactivity Control Unit
RCW	Recirculating Cooling Water
RIH	Reactor Inlet Header
ROH	Reactor Outlet Header
ROP	Regional Overpower
RRS	Reactor Regulating System
RSW	Raw Service Water

RWS	Reserve Water System
RWT	Reserve Water Tank
SCB	Secondary Control Building
SCDA	Severe Core Damage Accidents
SDC	Shutdown Cooling (System)
SDS1	Shutdown System One
SDS2	Shutdown System Two
SG	Steam Generator
TED	Technical Description