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Assessment Document

SYSTEMATIC REVIEW OF PLANT DESIGN FOR IDENTIFICATION OF INITIATING EVENTS

ACR

108-03660-ASD-001

Revision 1

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Analysis Report

Systematic Review of Plant Design for Identification of Initiating Events

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APPENDICES

1. INTRODUCTION

Nuclear regulatory and design organizations throughout the world have a tradition of looking at the response of nuclear power plants to a set of design basis accidents. The safety objective is to limit doses to members of the public from these events, thereby ensuring public protection. This objective is achieved through accident prevention and mitigation by: a) ensuring quality design, fabrication and construction, b) enforcing thorough inspections, and effective maintenance and testing of components, c) careful site selection, and d) designing appropriate operator interfaces and ensuring adequate operator training.

One of the tools used to estimate the potential risks of nuclear power plant safety is the Probabilistic Safety Assessment (PSA). This type of assessment starts by identifying initiating events, which in the absence of mitigating functions, may lead to radioactivity releases to the public and/or environment.

As a result of an initial review of the ACR design, a preliminary list of major ACR events has prepared and classified in the Safety Basis for ACR (Reference [6]). Furthermore, the safety basis report states that a systematic plant review of the ACR design for the identification of initiating events is to be conducted. In some cases, the ACR events classification is different than that in C-006 R1 (Reference [2]) based on the ACR design improvements.

To provide the necessary confidence that the set of initiating events chosen for analysis is complete and exhaustive, various systematic techniques can be applied. For the ACR™[*], master logic diagrams have been employed for the systematic review. This report describes the plant review process for initiating events identification, which generates a comprehensive set of scenarios for subsequent safety analysis. The methodology employed for carrying out present work is described in Reference [1]. This systematic review is presented in detail in Section 3 of this report. A minimum list of design basis initiating events to be addressed is given in the CNSC consultative document C-006 R1 (Reference [2]). After identification, the events are grouped and checked against the C-006 R1 event lists.

1.1 ACRONYMS

ASDV	Atmospheric Steam Discharge Valve
BOP	Balance of Plant
CANDU	Canadian Nuclear Deuterium
CCW(S)	Condenser Cooling Water (System)
CSDV	Condenser Steam Discharge Valves
CIGAR	Channel Inspection and Gauging Apparatus for Reactor
CNSC	Canadian Nuclear Safety Commission
CT	Calandria Tube
CV	Check Valve
DCS	Distributed Control System

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

ECC	Emergency Core Cooling (system)
ECI	Emergency Coolant Injection
FM	Fuelling Machine
FW	Feedwater
GSV	Governor Steam Valve
HTS	Heat Transport System
HVAC	Heating Ventilation and Air Conditioning
HX(s)	Heat Exchanger(s)
IA	Instrument Air
IE	Initiating Event
LCV(s)	Level Control Valve(s)
LOCA	Loss of Coolant Accident
LOR	Loss of Regulation
LRV	Liquid Relief Valve
LTC	Long Term Cooling
MLD	Master Logic Diagram
MSSV	Main Steam Safety Valve
MV(s)	Motorized Valve(s)
NPP	Nuclear Power Plant
NPSH	Net Positive Suction Head
P&IC	Pressure and Inventory Control
PSA	Probabilistic Safety Assessment
PT	Pressure Tube
PTR	Pressure Tube Rupture
RAB	Reactor Auxiliary Building
RB	Reactor Building
RCW	Recirculating Cooling Water
RD(s)	Rupture Disc(s)
RSW	Raw Service Water
RWS	Reserve Water System
SDS1	Shutdown System No.1
SDS2	Shutdown System No.2
SF	Spent Fuel
SFB	Spent Fuel Bay
SG	Steam Generator
SGPC	Steam Generator Pressure Control
SGTR	Steam Generator Tube Rupture

SLOCA	Small LOCA
SRPD	Systematic Review of Plant Design (for Initiating Events)
SST	Station Service Transformer
TB	Turbine Building
TBD	To Be Determined
TCV(s)	Temperature Control Valve(s)
TSV	Turbine Stop Valve

2. OBJECTIVES

2.1 Objectives

The systematic review of the ACR NPP design has two main objectives:

- To develop a list of postulated initiating events, which could lead to release of radioactivity to the public.
- To organize these events into groups and bounding categories in order to specify analysis requirements for them.

2.2 Scope Definition

The scope of this report is set forth by the SRPD Methodology document (Reference [1]):

- Identification of events that result in displacement of radioactive materials from their normal locations, which can have impact on public safety.
- Description of the event identification process via logic diagrams.
- Development of the initial list of events based on the logic diagrams.
- System-by-System review for the event identification process.
- Grouping of events into bounding events, based on the similarity of plant response, for convenience of subsequent deterministic and probabilistic assessment.
- Verification against CNSC's C-006 R1 document for completeness of events.

The SRPD assessment is based on the technical information of ACR-700 comprised in References [3], [4], [5], [7] and [8].

3. PROCESS OF INITIATING EVENTS IDENTIFICATION

An initiating event is a failure event that, in the absence of adequate mitigating functions, may start a mechanism leading to displacement of radioactive materials from their normal locations (plant systems and/or structures). During the normal operation of ACR, various amounts of radioactive material are present in the core, associated systems, and storage systems of the plant. Any potential accident situation must necessarily involve the displacement of that material from its normal location. If no radioactive material is released, no event is considered to have occurred.

The systematic review process is based on that assumption, and starts with the identification of all the sources of radioactive material in the plant. An example of one source of radioactive material is the reactor fuel.

For each distinct source of radioactive material, mechanisms are then identified which could lead to it being displaced from its normal location. An example of one type of displacement mechanism is damage to the fuel due to overheating.

For each displacement scenario, the physical nature of the plant systems and structures is then assessed to identify failures, which may result in that scenario. These failures are referred to as initiating events as described above. An initiating event may not necessarily result in a radioactivity release into the environment. An example of a failure, which may result in the overheating of the fuel, is a feeder break.

The list of failures identified using this approach may be very large. For subsequent stages of safety analysis, it is necessary to divide the failures into groups. Analysis of those groups will then result in a comprehensive analysis of the plant in an efficient manner, rather than analyzing each failure individually. The way in which failures are grouped is critical to the subsequent analysis and is discussed in Section 5.2.

The final stage of the systematic review is to determine frequencies to the grouped list of initiating events.

The systematic review of plant design for identification of initiating events employs the master logic diagram (MLD) as stated in Reference [1]. Flowsheets were reviewed as appropriate (References [4], [5], [7] and [8]).

The MLD method focuses on the potential causes that can generate releases of radionuclides from the plant and the potential causes that can generate it. Individual logic diagrams are constructed for each of the main (front-line) systems containing radionuclides, and their support systems. The steps that define this method are described further in Section 3.1.

A consistency check of those events identified by MLD against those specified by the CNSC consultative document C-006 R1(Reference [2]), was carried out and it is included in Tables 4, 5, and 6.

3.1 Master Logic Diagram Analysis

The starting point for the identification of initiating events using MLD is the construction of a high level fault tree or master logic diagram to identify the potential ways in which radioactive material can be displaced from its normal location. From the review of radioactive sources to the identification of initiating events, the following steps are carried out:

- Identification of distinct sources of radioactive materials within the plant;
- Identification of the mechanisms that could lead to radioactive materials displacement from their normal location;
- Identification of failures of systems and structures that can cause the displacement scenarios to occur, i.e. initiating events.

The level of detail of the logic diagrams is normally limited to the failure of the system function or failure of main equipment leading to the event under consideration. Some events, however, involve the loss of specific components rather than a system. For these events, e.g. a liquid relief valve or pressurizer relief valve failing open, the component failure could result in a reactor trip and/or radionuclide release, and the need for decay heat removal. These events are included in the logic diagram.

The output of the MLD Analysis consists of:

- Logic diagrams for HTS, moderator, fuel handling, BOP process systems and support systems (Figures A-1 to A-15).
- Tables presenting the methodological steps for identification, screening and grouping of the logic diagram basic events into bounding higher-level events for analysis.

The rare low frequency class 5 single events as described in C-006 R1 (Reference [2]) are not included in the master logic diagram.

3.2 Sources of Radioactive Materials

The normal locations of radioactive materials in the ACR plant are:

a) Inside the core
   - in the fuel bundles,
   - in the Heat Transport System coolant,
   - in the Moderator System,
   - in the structural components (pressure tubes, Calandria tubes, adjuster rods and shutdown rods)
   - in the Cover Gas System
   - in the Shield Cooling System and Shield Tank
   - in the Annulus Gas System.

b) Outside the core
   - in the fuel changing system,
   - in the spent fuel transfer system,
   - in the spent fuel storage system,
   - in the HT and auxiliary systems' water and in the Moderator D_2O management systems,
   - in the liquid, solid and gaseous radioactive waste management systems.

3.3 Radioactivity Displacement Mechanisms

Any one of the radioactive materials (e.g. HT light water coolant, moderator D_2O) can be displaced from the above-mentioned locations via a number of different mechanisms. These mechanisms are further verified against each source for identification of credible release scenarios and failures leading to those scenarios.

The mechanisms identified in this section may lead to displacement of radioactive materials from their locations alone or in various combinations. They are as follows:

- Loss of reactivity control;
- Mismatch between thermal power generated and thermal power removed from the core due to:
    - partial or total loss of coolant inventory,
    - partial or total loss of heat sink,
    - partial or total loss of coolant flow
    - pressure and inventory control system failures.
- Mechanical damage through direct interaction with adjacent equipment;
- Chemical damage to fuel and/or reactor control system due to inadequate coolant purification and corrosion control.

3.4 Identification of Initiating Events

The top-level master logic diagram shown in Figure A-1, simply presents the radioactivity sources identified in Section 3.2. Further down, the second level master logic diagrams associate to each of the sources a set of displacement mechanisms that are relevant for that particular case.

The analysis then identifies the specific failures that lead to each of the displacement mechanisms. These are the basic initiating events that are subsequently grouped to derive events for the purpose of deterministic and /or PSA development.

4. DISPLACEMENT MECHANISMS AND INITIATING EVENTS IDENTIFICATION

4.1 Release from the Heat Transport System

The top level master logic diagram that identifies the initiating events is presented in Figure A-1.

Chemical damage to fuel could only occur following a failure of the heat transport purification system. Small amounts of chemicals are added to the HTS coolant, to prevent corrosion of the core materials. As such, this event can only be due to maintenance practices and can only cause problems over long time intervals. Therefore, the event was screened out from the list of IEs.

4.1.1 Mechanical Damage to Fuel

This damage type is possible as a result of:

a) Damage to fuel by the fuelling machine while operating on the channel (Figure A-5 identifies the event under releases of radioactivity from the Fuel Handling System).
b) Fuel Damage due to FM Failures/Malfunctions (which includes damages caused by malfunctions of the FM ram and magazine, see Figures A-5 and A-13).

4.1.2 Mismatch Between Generation and Removal of Thermal Power in Normal Operation

This generic type of displacement mechanism is broken down into the following set of more specific mechanisms that are typical for any reactor at power:

a) Loss of Reactivity Control - this occurs when for various reasons the neutronic power level of the reactor cannot be maintained in the core;
b) Loss of HTS Coolant Inventory - this occurs when, for various reasons, HTS coolant leaves the core at a rate and for a duration that endangers system’s heat removal function;
c) Loss of HTS Heat Sink (other than Moderator as Heat Sink (see Section 4.2.2)) - this occurs when, for various reasons, heat cannot be transferred to the secondary side at the same rate it is generated;
d) Loss of Coolant Circulation in the HT system - this occurs due to reduced flow of the cooling agent resulting in a rate of heat transfer to the heat sink that is less than the rate of heat generated;
e) Failure of Pressure and Inventory Control System to match the requirements for HT System pressure and inventory adjustments during operational transient parameter excursions.

4.1.2.1 Loss of Reactivity Control

This mechanism is broken down into two sub-top events: a) Loss of Bulk Reactivity Control, b) Loss of Local / Spatial Reactivity Control (see Figure A-4).

4.1.2.2 The Loss of HTS Coolant Inventory

This mechanism of radioactivity release is developed in Figure A-6. Three major types are considered:

- Coolant is discharged within the core from the HT System into the Moderator, Annulus Gas and /or End Shield Cooling Systems;
- Coolant is discharged through the interfaces with the steam generator tubes, ECI or LTC Systems (i.e. containment bypass);
- Coolant is discharged outside the core into the containment environment.

Loss of HTS Coolant Inventory (LOCA) is classified by the magnitude of HT system boundary breach and by the proportional magnitude of the coolant discharge. Three categories of LOCA events have been employed in the logic diagram development:

a) Breaks causing coolant discharge rates smaller than the make-up rate provided by the Pressure and Inventory Control System pumps – HTS Leaks,
b) Breaks causing coolant discharge rates greater than the make-up capacity of the Pressure and Inventory Control System pumps and up to the equivalent of the largest feeder break – Small LOCAs,
c) Breaks causing coolant discharge rates greater than the equivalent of the largest feeder break – Large LOCAs.

The loss of cooling to end shields comprises events that may lead to a breach in the HTS boundary due to overstressing of the calandria tubesheets.

4.1.2.3 The Loss of HTS Heat Sink Events

This event is divided into four major failure groups (see Figure A-7):

- Feedwater system failures;
- Steam system failures;
- Loss of Condenser as a heat sink;
- Condensate system failures.

Feedwater failures are divided into loss of FW inventory and feedwater flow impairments. Feedwater flow impairments comprise either FW line breaks, FW pump/motor failures, control valves/check valves failures, Steam Generator level control failures and inadequate NPSH at the FW pumps suction due to loss of pegging steam to the deaerator.

Steam system failures are further categorised as: SG pressure high, de-pressurization failures of the steam generator, or reactor / turbine trip.

Loss of Condenser as a heat sink may occur due to failures leading to loss of vacuum or due to failures leading to loss of CCW system.

4.1.2.4 Loss of Coolant Circulation in the HT System

This event may occur locally on one channel or may affect the whole core either by fluid stagnation or by circulation at an inadequate rate (see Figure A-8).

4.1.2.5 Failures of the Pressure and Inventory Control System

These failure mechanisms are naturally split into loss of pressure, loss of inventory and failure of redundant DCS Group Controllers (see Figure A-8).

4.1.3 Mismatch between Generation and Removal of Thermal Power when Reactor is in Shutdown State

The failure mechanisms considered are HTS leaks, loss of LTC, and loss of support systems for various shutdown states (see Figures A-2, and A-9).

For ACR there are 3 shutdown states identified in this SRPD work:

1. Reactor shutdown, HTS full and pressurized;
2. Reactor shutdown, HTS full and de-pressurized;
3. Reactor shutdown, HTS full and de-pressurized and drained to the headers level.

4.1.4 Failures Associated with Reactor Power Manoeuvres

The ACR is used for base load electrical power generation. Power manoeuvres occur infrequently and the transition period is short. Therefore, failures associated with reactor power manoeuvres are bounded by full power operation and not considered further (see Figure A-2).

4.2 Releases from the Moderator System

The releases of radioactivity from the moderator system mechanisms are grouped into two event categories: a) Loss of Moderator Inventory and b) Loss of Moderator as a Heat Sink.

4.2.1 Loss of Moderator Inventory

This comprises those events that could lead to reduction in the quantity of moderator (see Figure A-3).

4.2.2 Loss of Moderator Heat Sink

This addresses those events that may lead to a loss of moderator flow or cooling (see Figure A-3).

4.3 Release from Moderator Cover Gas System

This addresses loss of D_2 concentration control, loss of moderator cover gas pressure and inventory (see Figure A-4).

4.4 Release from Fuel Handling System

This radioactive materials source has three groups of failures associated to it (see Figure A-5):

- Fuel Changing (Fuelling Machine) Failures,
- Spent Fuel Handling and Storage System Failures,
- New Fuel Handling and Storage System Failures.

4.4.1 Fuel Changing (Fuelling Machine) Failures

These events are sub-divided into two categories (see Figure A-5):

4.4.1.1 Fuel Damage Caused while FM is "Rotated to the Reactor"

The major groups of radioactivity displacement mechanisms are (see Figure A-5):

- FM Induced LOCA And/Or Fuel Damage While FM Is Clamped To Reactor End Fitting;
- FM Induced LOCA And/Or Fuel Damage While FM Is Unclamped To Reactor End Fitting

This set of failure events comprise:

- fuel is crushed in the channel due to incorrect manoeuvres or failure of ram "C" to stop when required,
- bridge brakes failures,
- inadvertent unclamping of the snout,
- inadvertent motion of the carriage or cradle, and
- loss of H_2O coolant through the machine (small LOCA).

4.4.1.2 Fuel Damage Caused while FM is "Off the Face of the Reactor"

These events are caused when FM is traversing the face of the reactor, or is heading towards Spent Fuel Port to discharge the bundles. The related failures are: loss of FM H_2O circulation, cooling and loss of FM H_2O inventory (see Figure A-5).

4.4.2 Spent Fuel Storage Failures

Two types of failures are identified (see Figure A-13):

- Loss of Spent Fuel (SF) Reception Bay Heat Sink,
- Mechanical Damage to Spent Fuel during Storage.

4.4.3 Spent Fuel Transfer System Failures

These failure events are grouped into two main categories (see Figure A-13).

4.4.3.1 Mechanical Damage to Fuel while in Transfer from the Spent Fuel Port to the Spent Fuel Receiving Bay

- Fuel Damage While Passing Through One of 4 Full-Bore Isolation/Containment Valves.
- Fuel Damage Due To Inadvertent Operation of SF Transfer Equipment.

4.4.3.2 Loss of Cooling to Fuel while Bundles are Transferred from the Spent Fuel Port to the Spent Fuel Receiving Bay

- Loss of Transfer Tube H_2O inventory,
- Loss of Transfer Tube cooling.

4.4.4 New Fuel Storage System

Certain mass of new fuel becomes critical due to flooding in the storage room (see Figure A-4).

4.5 Events Causing Failure of Support Systems

The failures of support systems during full power operation are shown in Figure A-14.

4.6 Failures of Support Systems while Reactor Shutdown

The failures of support systems during shutdown are shown in Figure A-15.

4.7 Release from Radioactive Waste Management System

These categories of release mechanisms are divided into three groups of events according to the aggregate status of the materials (see Figure A-4):

- Release from solid waste management system;
- Release from liquid waste management system;
- Release from gaseous waste management system.

4.8 Release from H_2O and D_2O Storage, Transfer and Recovery Systems

This category of release mechanisms comprises:

- Release from H_2O Leakage Collection System,
- Release from H_2O Supply System and P&IC H_2O Storage Tank, and
- Release from D_2O Leakage Collection and from D_2O Supply Systems.

4.9 Shield Cooling System

These failures are caused when there is a loss of shield cooling inventory, flow or cooling, which may lead to impairments of HTS geometry.

4.10 Release from Annulus Gas System

This category of release mechanism is not developed further, as it is considered insignificant from the viewpoint of radiological consequences.

4.11 External Events

The present section describes external events based on the C-006 R1 (Reference [2]).

- Fire (internal and external);
- Earthquake;
- Tornado / Hurricane;
- Tsunami waves;
- Extreme weather (wind, rain, hail, snow, ice, lightning, temperature, drought);
- Explosions;

- Turbine blades missiles;
- Release of toxic, explosive or corrosive chemicals due to transportation accidents;
- Internal Flooding (the source resides within NPP);
- External Flooding (the source resides outside NPP);
- Aircraft Crash;
- Electromagnetic Interference from Telecommunications equipment;
- Electric storm that may disable Class IV .

Some of these events are site specific. Therefore, they will be addressed selectively, by considering the specific site chosen for the ACR Plant and the customer requirements. Three events are included in the PSA analysis: 1) seismic, 2) internal fire and 3) internal flood.

The ACR is designed to ensure that structures, systems and components important to safety are appropriately protected against the effects of missiles that might result form turbine failure. The plant layout of ACR is such that the turbines are built perpendicular to the Reactor Auxiliary Building and the main control room.

5. SELECTION AND GROUPING OF MLD RESULTS

5.1 Selection / Screening of Logic Diagram Initiating Events

The internal initiating events resulting from the development of the logic diagrams are further assessed through a screening process. When MLD technique is employed for establishing the IEs, the focus is on deriving all credible accident scenarios. In this phase of the SRPD, work the emphasis was on completeness. As a result, it is possible that some failure events obtained are not meaningful for the PSA. Therefore, all events are examined against a set of criteria meant to eliminate unrealistically derived events:

- extremely low occurrence likelihood,
- insignificant safety consequences,
- out of the scope.

All events passing the above-mentioned screening criteria are selected in the list of events to be analysed in the ACR PSA. In the next paragraphs a brief description of the screening process is presented by examples.

Several events related to specific plant systems were removed from the lists due to low dose consequences (potential releases would meet safety acceptance criteria with respect to radioactivity releases into the environment). The fluids in these systems contain less than 100 times the radioactive material concentration in the HTS system (based on CANDU 6 plants information and estimates). Therefore, they would have limited radiological impact inside RB and RAB and negligible impact upon the environment and / or the public. As a result, the following sources of radioactivity releases are screened out from the list of events:

- Release from Water Management Systems:
    - Release From H_2O and D_2O Storage
    - Release From H_2O Supply Cleanup.
- Release from Radioactive Waste Management Systems:
    - Release From Solid, Liquid and Gaseous Wastes.
- Release from the Annulus Gas System.

The entire set of basic events screened out and the rationale for excluding them is presented in Table 1.

5.2 Grouping of Logic Diagram Initiating Events

Subsequent to the selection process, the initiating events are grouped by similarity of plant response and / or bounding consequences into a single, bounding, and higher-level event. The justification for the event grouping (e.g. same mitigating actions, bounding consequence) is described. For example, the loss of moderator circulation and the loss of moderator cooling events can be grouped into a single event, i.e. Loss of Moderator Heat Sink event, for the purpose of analysis. It is recognized that the dynamics of plant response for the two basic events will be different in that compared to loss of cooling, the loss of circulation results in a faster rate of rise in moderator temperature. However, the event tree analysis will assume the faster of the transients and thus bound the slower transient. The results will therefore be conservative.

The grouping process therefore yields a smaller yet conservative, more manageable number of initiating events for the purpose of analysis.

Both the selection and the grouping phases of the SRPD work are reflected in the tables produced. A frequency value was associated to each event in the set of grouped events (resulting after selection and grouping). The process of assigning frequency values to the grouped events is presented in the next section.

5.3 Initiating Event Frequencies

The initiating event frequencies were determined from CANDU or international operating experience or past CANDU 6 PSAs up to December 31, 2000. In some cases, where source frequencies were indicated as “ACR Updates of CANDU Data”, the values were derived based on engineering judgement. Example: based on the CANDU 6 experience, the dominant failure contributor to “Steam Generator De-Pressurization” grouped event is spurious opening of a Main Steam Safety Valve; ACR has only half of the number of the CANDU 6 valves, thus for this event a value equal to half of the CANDU 6 frequency is used. In a few cases, the initiating event frequencies will be determined during the PSA work.

5.4 Output of Master Logic Diagrams

The master logic diagrams are shown in Appendix A Figures A-1 to A-15.

a) logic diagrams for the events that are related to the failure of the heat transport, moderator, moderator cover gas, fuel handling, H_2O or D_2O storage, transfer and recovery, radioactive waste management, and annulus gas systems – Appendix A, Figure A-1 through to Figure A-13; and
b) a logic diagram for the events that are related to the failure of the support systems – Appendix A, Figure A-14 and A-15.

The initiating events screened out from PSA and /or deterministic analysis (see the screening rationale in Section 5.1) are presented in Table 1.

The rationale for grouping the logic diagram initiating events with similar plant responses into a single, bounding, higher-level event is shown in Table 2. The initiating events, the rationale and the grouped events are presented in this table.

In Table 3, the initiating event frequencies for the grouped events are shown.

In Tables 4, 5 and 6, the comparison of the ACR grouped events to the classification of C-006 R1 (Reference [2]) class 1, 2 and 3 is presented respectively. The purpose of comparison was to show completeness of the analysis and not for events classification. In some cases, the ACR events classification as identified in the Safety Basis for ACR (Reference [6]) is different than that in C-006 R1 (Reference [2]) based on the ACR features.

The class 4 event combinations and class 5 events as described in C-006 R1 (Reference [2]) are not part of these comparison tables.

6. CONCLUSIONS

This assessment report establishes a comprehensive list of initiating events for the ACR design by developing master logic diagrams. The MLD process of event identification consisted of a) identifying sources of radioactive material in the plant, b) examining mechanism for displacement of these materials from their normal locations, and c) identifying more specific causes for them (i.e., Initiating Events).

All the events in the logic diagrams were then examined to screen out events and then to group them by similarity of plant response and / or bounding consequences.

The total number of initiating events identified were 201, of which 34 were screened out and the remaining ones were assigned into 87 grouped events. Initiating event frequencies for grouped events were determined from CANDU operating experience to the extent practical and, where applicable, based on judgement, specific to ACR design. A few initiating event frequencies will be determined during the PSA work.

As well, the events from the master logic diagrams were checked against the events classified in C-006 R1.

The results of this assessment report will be used for carrying out the ACR deterministic and PSA work.

7. REFERENCES

[1] AECL – “Systematic Review of Plant Design - Methodology for Identification of Initiating Events”, Advanced CANDU Reactor, 108-03660-AB-002, Rev. 1, July 2003.

[2] CNSC – Draft Regulatory Guide C-006 Revision 1, September 1999.

[3] AECL – “ACR-700 Technical Description”, 10810-01371-TED-001, Rev 0, June 2003.

[4] AECL – “Heat Transport System Flowsheet, 10810-33100-0001-01-FS-E, Rev. P2, May 2003.

[5] AECL – “Pressure and Inventory Control System Flowsheet, 108-33310-0001-01-FS-E, Rev. P0, March 2003.

[6] AECL – “Safety Basis for ACR”, 108-03600-AB-003, Rev. 0, July 2003.

[7] AECL – “Emergency Coolant Injection System Flow Sheet”, 10810-34320-0001-01--FS-0, Rev. P1, July 2003.

[8] AECL – “Long Term Cooling System Flowsheet – 10810-34350-0001-01-FS-E, Rev. P1, November 2003.

Table 1
Events Screened Out from the Grouping Process

MLD Index	Event Description	Rationale for Screening Out the MLD Basic Events
IE-01	RELEASE FROM ANNULUS GAS SYSTEM	This event has insignificant radiological consequences due to extremely reduced quantities of radioactive material that may be present, at any time, within the system's boundaries. Therefore it is screened out.
IE-02	RELEASE FROM H_2O LEAKAGE SYSTEM	Same reasoning as for event IE-01 applies to this event. Accordingly, this event is screened out.
IE-03	RELEASE FROM D_2O LEAKAGE COLLECTION AND D_2O SUPPLY SYSTEM	Same reasoning as for event IE-01 applies to this event. Accordingly, this event is screened out.
IE-04	RELEASE FROM H_2O SUPPLY SYSTEM	Same reasoning as for event IE-01 applies to this event. Accordingly, this event is screened out.
A-IE-01	CHEMICAL DAMAGE TO FUEL (Failure of Chemistry Control in the HT System)	This event represents a very slow mechanism of fuel damage, which would only result from faulted plant maintenance practices, neglect of standard duties and/ or ignoring failures of HTS purification system components by the plant staff for extended time periods. This event is screened out on the basis of low frequency event in conjunction with a prolonged time for the event to develop.
B-IE-01	CALANDRIA TUBE LEAKS INTO ANNULUS GAS	This event leads to a gradual draining of the calandria and leads to an inherent shutdown. The leaks would be detected by the moisture monitoring and detection system in the annulus gas and eventually on low level in the moderator. This event is screened out on the basis of low frequency and a slow moving event.
B-IE-03	MODERATOR HX INTERNAL LEAKS	Moderator HX plate's leakage leads to poisoning of the moderator with light water. This event leads to an inherent shutdown. It is very unlikely that radioactivity would be released into RCW due to the fact that RCW water pressure is significantly higher than Moderator heavy water pressure. Therefore, this event is screened out on the basis of insignificant radiological consequences.
B-IE-04	MODERATOR HX INTERNAL RUPTURE	Moderator HX plate rupture would have consequences very similar to those of moderator HX internal leaks (B-IE-03), with the difference that moderator poisoning with light water would be considerably faster and would lead rapidly to shutdown. It is very unlikely that radioactivity would be released into RCW due to the fact that RCW water pressure is significantly higher than Moderator heavy water pressure. Therefore, this event is screened out on the basis of insignificant radiological consequences.
B-IE-06	MODERATOR PIPE BREAKS INSIDE SHIELD TANK	This event will lead to mixing of H_2O with D_2O leading to an inherent reactor shutdown. Furthermore, SDS1 and SDS2 will also trip on high moderator level caused by static head of the shield cooling system head tank. Moderator D_2O will discharge into the shield tank, due to the higher moderator pressure. With the reactor shutdown guaranteed, the consequence of this event is a small discharge of tritium through the moderator cover gas bleed valves. These valves open due to cover gas pressurization caused by the static head of the expansion tank. There are insignificant safety consequences and therefore, this event is screened out.

MLD Index	Event Description	Rationale for Screening Out the MLD Basic Events
C-IE-01	LOSS OF MODERATOR COVER GAS SYSTEM INVENTORY	De-pressurisation of cover gas system due to gas leakage outside system boundaries may affect moderator subcooling margin. If reduced, it may provide inadequate cooling following a LOCA and loss of ECC event. However, this event in itself does not lead to any radioactivity displacement mechanism. It is covered under LOCA plus loss of ECC analysis and need not be addressed as an initiating event.
C-IE-03	LOSS OF COVER GAS SYSTEM PRESSURE CONTROL	The cover gas pressure system is such that a failure of the control system itself does not create a low pressure condition beyond lowering the cover gas pressure to atmospheric. This constitutes a minor tritium release and a reduction in moderator subcooling margin. Contribution of the latter is covered under LOCA plus loss of ECC analysis. For high pressure conditions, there are two 2 x100% bleed valves. As a backup, the moderator rupture discs provide relief. This event is screened out based on the above rationale.
D-IE-03	FUEL CRUSHED / MECHANICALLY DAMAGED DUE TO FUELLING MACHINE FAILURES	This event is screened out based on the following considerations: • Design of the F/M ensures that it can not exert large force on the fuel bundle to crush it • In any case, if this event was to be postulated, the operational limits on HTS activity will guide the operator to shutdown the reactor and close any pathways outside the containment (e.g. in case of boiler tube leaks) as necessary such that radiological consequences from this event are not significant.
E-IE-01	RELEASE FROM SOLID WASTE MANAGEMENT SYSTEM	The radioactive material concentration of the Solid Waste Management System is at least 100 times less than that of the HTS. Therefore, this event is screened out.
E-IE-02	RELEASE FROM LIQUID WASTE MANAGEMENT SYSTEM	Same reasoning as for event E-IE-01 applies to this event. Accordingly, this event is screened out.
E-IE-03	RELEASE FROM GASEOUS WASTE MANAGEMENT SYSTEM	Same reasoning as for event E-IE-01 applies to this event. Accordingly, this event is screened out.
F-IE-01	CERTAIN MASS OF NEW FUEL BECOMES CRITICAL DUE TO FLOODING IN THE STORAGE ROOM	The design of the storage areas is to comply with two strict safety requirements: - geometry and spatial arrangement of the fuel bundles have to be established such that criticality cannot be achieved even if complete flooding occurs and, - flooding of the new fuel storage area(s) should be unlikely to occur. Consequently, this event is screened out.
H-IE-10	HEAT TRANSPORT SYSTEM PUMP CASING FAILURES	This event is considered to occur with extremely low probability and is screened out on that basis.
K-IE-06	LOSS OF COOLING TO ALL HTS PUMP BEARINGS	It is considered very unlikely that both the SW divisions will fail to cool HTS pumps bearings. Therefore, the event is screened out from the list.
M-IE-01	SMALL LOCA WHILE HTS FULL AND PRESSURIZED (REACTOR SHUTDOWN)	This event is unlikely to occur while reactor is shutdown because of the short time intervals involved in this state. This event is not considered necessary to analyse, as their consequences are bounded by the consequences of the same events at power. Therefore, the event is screened out.

MLD Index	Event Description	Rationale for Screening Out the MLD Basic Events
M-IE-10	LTCS HX TUBE BREAKS (LTCS FAILURES WHEN HTS IS FULL AND PRESSURIZED)	Same reasoning as for event M-IE-01 applies to this event. Accordingly, this event is screened out.
M-IE-11	LTCS PIPE BREAKS (LTCS FAILURES WHEN HTS IS FULL AND PRESSURIZED)	Same reasoning as for event M-IE-01 applies to this event. Accordingly, this event is screened out.
M-IE-12	LTCS PUMP SEALS FAILURES (LTCS FAILURES WHEN HTS IS FULL AND PRESSURIZED)	Same reasoning as for event M-IE-01 applies to this event. Accordingly, this event is screened out
M-IE-13	LOSS OF LTCS COOLING (LTCS FAILURES WHEN HTS IS FULL AND PRESSURIZED)	Same reasoning as for event M-IE-01 applies to this event. Accordingly, this event is screened out
M-IE-14	LOSS OF LTCS FLOW (LTCS FAILURES WHEN HTS IS FULL AND PRESSURIZED)	Same reasoning as for event M-IE-01 applies to this event. Accordingly, this event is screened out
M-IE-15	LRV/BLEED VALVE FAILS OPEN (LTCS FAILURES WHEN HTS IS FULL AND PRESSURIZED)	Same reasoning as for event M-IE-01 applies to this event. Accordingly, this event is screened out
M-IE-18	RUNNING FEED PUMP TRIPS & STANDBY PUMP FAILS TO START (WHEN HTS IS FULL AND PRESSURIZED)	Same reasoning as for event M-IE-01 applies to this event. Accordingly, this event is screened out
N-IE-01	P&IC SYSTEM FAILURES TO SOLID MODE DURING REACTOR POWER MANOEUVRES	The ACR is used for base load electrical power generation. Power manoeuvres occur infrequently and the transition period is short. Therefore, failures associated with reactor power manoeuvres are screened out.
Q-IE-10	END FITTING AND LATTICE TUBE FAILURES / LEAKAGES INSIDE THE ANNULUS GAS (HTS LEAKS NO CONTAINMENT BY-PASS)	This event is expected to occur with extremely low probability because lattice tubes are much stronger than the bellows. Following a postulated failure of end fitting, bellows will fail and lattice tube will remain intact. Therefore, this event is screened out.
R-IE-03	INTERFACE LOCA THROUGH EMERGENCY COOLANT INJECTION SYSTEM (SMALL LOCA WITH CONTAINMENT BY-PASS).	Due to the extremely high resistance of the rupture discs against thrust exerted by the HTS coolant plus a check valve and a closed motorized valve in the line, it is very unlikely that the HTS coolant blowback into ECI would occur. Furthermore, containment bypass would have to go through the nitrogen pressurizing lines which have normally closed valves and there is also the relief provision inside the RB. For these reasons the event is screened out.

MLD Index	**Event Description**	**Rationale for Screening Out the MLD Basic Events**
S-IE-03	CONDENSER COOLING WATER INTAKE / DISCHARGE TUNNEL FAILURE	This event is screened out on the basis of extremely low frequency
X-IE-01	TOTAL LOSS OF INSTRUMENT AIR (REACTOR SHUTDOWN, HTS FULL AND PRESSURIZED)	Same reasoning as for event M-IE-01 applies to this event. Accordingly, this event is screened out.
X-IE-02	LOSS OF ONE SW DIVISION (RCW AND/OR RSW) (REACTOR SHUTDOWN, HTS FULL AND PRESSURIZED)	Same reasoning as for event M-IE-01 applies to this event. Accordingly, this event is screened out.
X-IE-03	TOTAL LOSS OF CLASS IV POWER (REACTOR SHUTDOWN, HTS FULL AND PRESSURIZED)	Same reasoning as for event M-IE-01 applies to this event. Accordingly, this event is screened out.
X-IE-04	TOTAL LOSS OF HVAC IN THE PLANT (REACTOR SHUTDOWN, HTS FULL AND PRESSURIZED)	Same reasoning as for event M-IE-01 applies to this event. Accordingly, this event is screened out.

Table 2
Grouping of ACR Selected Events

Logic Diagram Index	Basic Event Description	Rationale for Events Grouping by Plant Response	Grouped Event ID	Description of the Grouped Event
C-IE-02	LOSS OF DEUTERIUM CONCENTRATION CONTROL IN THE MODERATOR COVER GAS	This event results in a loss of deuterium concentration in the moderator cover gas system. This event could be caused by loss of compressors or recombination units. This event is grouped under GE-01.	GE-01	LOSS OF DEUTERIUM CONCENTRATION CONTROL IN THE COVER GAS SYSTEM
H-IE-11	HEAT TRANSPORT PUMP SEALS FAILURES	This event represents a very small loss of coolant from the HT system and therefore it can be grouped in the HTS leaks category of events. In terms of consequences and plant response the analysis for this event will be covered by the grouped event GE-02, HTS Leaks.	GE-02	HTS LEAKS WITH NO CONTAINMENT BY-PASS
L-IE-07	ONE BC RELIEF VALVE OPENS SPURIOUSLY	This event leads to a loss of coolant accident (LOCA) from HTS via the P&IC system. This event is grouped under GE-02.	GE-02	HTS LEAKS WITH NO CONTAINMENT BY-PASS
Q-IE-03	PIPE FAILURES IN HTS AUXILIARY SYSTEMS WITH DISCHARGE IN THE REACTOR BUILDING (HT LEAKS NO CONTAINMENT BY-PASS)	This event has a similar plant response as L-IE-07 and is grouped under GE-02.	GE-02	HTS LEAKS WITH NO CONTAINMENT BY-PASS
Q-IE-06	TUBE FAILURES IN THE FM WATER SUPPLY SYSTEM HEAT EXCHANGER (LEAKS WITH CONTAINMENT BY-PASS)	This event has a similar plant response as L-IE-07 and is grouped under GE-02.".	GE-02	HTS LEAKS WITH NO CONTAINMENT BY-PASS
Q-IE-07	END FITTINGS / FEEDERS LEAKAGES INTO THE FM VAULT (HT LEAKS NO CONTAINMENT BY-PASS)	This event has a similar plant response as L-IE-07 and is grouped under GE-02.	GE-02	HTS LEAKS WITH NO CONTAINMENT BY-PASS

Logic Diagram Index	Basic Event Description	Rationale for Events Grouping by Plant Response	Grouped Event ID	Description of the Grouped Event
Q-IE-09	PT LEAKS INTO THE ANNULUS GAS WITH BURSTING OF CHANNEL BELLOWS (within pressurizing pumps capacity)	This event has a similar plant response as L-IE-07 and is grouped under GE-02.	GE-02	HTS LEAKS WITH NO CONTAINMENT BY-PASS
B-IE-05	MODERATOR PIPE BREAKS OUTSIDE SHIELD TANK (Calandria Inventory Preserved)	This event would cause a shutdown transient depending on the emptying rate of the Calandria D_2O inventory. As the moderator level in the tank decreases, fewer thermal neutrons are thermalized in the core, therefore the efficiency of the fission reactions is reduced progressively. The shutdown is guaranteed to happen by stepback due to moderator pumps trip, by SDS1 and SDS2 due to low level in the Head Tank and by intrinsic reduction of the moderator level. No other heat sinks are impaired and no effects upon support systems can occur. Further to that, this event leads to similar consequences with the event #B-IE-09 "Loss of Moderator Circulation" because it contributes to loss of Moderator System as a heat sink. This event is grouped under GE-03.	GE-03	TOTAL LOSS OF MODERATOR HEAT SINK
B-IE-07	MODERATOR AUXILIARY SYSTEMS LINE BREAKS	Consequences of this event are similar to those of the event B-IE-09 "Loss of Moderator Circulation", because Calandria inlet and outlet pipes are located near the top of the vessel, and as such, D_2O inventory covering the fuel channels will be preserved. Event tree analysis results for this event can be derived from the results of the analysis for the Grouped Event "Total Loss of Moderator as Heat Sink", GE-03.	GE-03	TOTAL LOSS OF MODERATOR HEAT SINK
B-IE-09	LOSS OF MODERATOR FLOW /CIRCULATION	Loss of Moderator Circulation leads to the same consequences as those for event B-IE-11 ("Total Loss of Moderator Heat Sink") except it leads to faster moderator overheating transient. This event will not cause a shutdown due to very slight negative reactivity insertion , which would be easily compensated by RRS. This event is grouped for analysis under the GE -03 grouped event.	GE-03	TOTAL LOSS OF MODERATOR HEAT SINK

Logic Diagram Index	Basic Event Description	Rationale for Events Grouping by Plant Response	Grouped Event ID	Description of the Grouped Event
B-IE-11	TOTAL LOSS OF MODERATOR HEAT SINK	This event may occur due to RCW TCVs failing closed or C&I failure that fails the valves closed. Loss of Moderator Cooling leads to the same consequences as those for event B-IE-09 ("Loss of Moderator Circulation"). It will also cause a slight negative reactivity insertion. This event is grouped for analysis under the GE -03 grouped event.	GE-03	TOTAL LOSS OF MODERATOR HEAT SINK
B-IE-10	PARTIAL LOSS OF MODERATOR COOLING	This event is grouped for analysis with B-IE-12 under GE-04 - "Partial Loss of Moderator as a Heat Sink".	GE-04	PARTIAL LOSS OF MODERATOR HEAT SINK
B-IE-12	PARTIAL LOSS OF MODERATOR FLOW	This event is grouped for analysis under GE-04 - "Partial Loss of Moderator as a Heat Sink".	GE-04	PARTIAL LOSS OF MODERATOR HEAT SINK
H-IE-08	FEEDER BREAK NO FLOW STAGNATION	All LOCA events resulting in loss of inventory, which is within the largest size feeder break discharge but beyond the make-up capacity of the D_2O pressurizing pumps are classified as small LOCA events. Feeder Break with no flow stagnation is an event similar to a break anywhere in the heat transport system resulting in a discharge into the containment. However, due to the fact that it does not lead to a PT/CT rupture, the plant response and the consequences of this event are quite different than those of a feeder break with flow stagnation. This event is grouped under GE-05.	GE-05	FEEDER BREAK NO FLOW STAGNATION
G-IE-01	FAILURE OF BULK FLUX MEASUREMENT (reactor operating at power)	Events G-IE-01, G-IE-02 and G-IE-04 are contributors to a bulk loss of regulation and can thus be grouped into a single event called "Bulk Core Power Excursion". It is therefore grouped under GE-06 event.	GE-06	BULK CORE POWER EXCURSION (REACTOR OPERATING AT FULL POWER)
G-IE-02	SPURIOUS WITHDRAWAL OF ALL ZONE CONTROL ABSORBERS	Events G-IE-01, G-IE-02 and G-IE-04 are contributors to a bulk loss of regulation and can thus be grouped into a single event called "Bulk Core Power Excursion". This event is grouped under GE-06 event.	GE-06	BULK CORE POWER EXCURSION (REACTOR OPERATING AT FULL POWER)
G-IE-04	FAILURE OF THERMAL POWER MEASUREMENT	Events G-IE-01, G-IE-02 and G-IE-04 are contributors to a bulk loss of regulation and can thus be grouped into a single event called "Bulk Core Power Excursion". This event is grouped under GE-06 event.	GE-06	BULK CORE POWER EXCURSION (REACTOR OPERATING AT FULL POWER)

Logic Diagram Index	Basic Event Description	Rationale for Events Grouping by Plant Response	Grouped Event ID	Description of the Grouped Event
G-IE-03	ERRONEOUS LOCAL NEUTRON OVERPOWER	This event may occur as a result of new fuel loading and inadequate local flux compensation /measurement . It leads to local loss of regulation, and is similar in this regard to G-IE-05. It is, therefore, under event GE-07, "Regional Core Power Excursion".	GE-07	REGIONAL CORE POWER EXCURSION (REACTOR OPERATING AT FULL POWER)
G-IE-05	SPURIOUS WITHDRAWAL OF A BANK OF ZONE CONTROL ABSORBERS	This event is representative for local loss of regulation events such as local flux measurement failures and spurious withdrawal of some zone control units. It is similar to the event G-IE-03 above and is, therefore, grouped under event GE-07.	GE-07	REGIONAL CORE POWER EXCURSION (REACTOR OPERATING AT FULL POWER)
D-IE-01	LOSS OF FM H_2O CIRCULATION (WHILE MACHINE OFF REACTOR FACE)	Events D-IE-01 and D-IE-02 are contributors to a loss of cooling to the fuel bundles that are in the machine while on transit to unload to the spent fuel port. It is identical to D-IE-02 in terms of plant response and consequence. Therefore, it is grouped with D-IE-02 under GE-08 event.	GE-08	LOSS OF COOLING TO FUEL WHILE IN THE FM HEAD (WHILE MACHINE OFF REACTOR FACE)
D-IE-02	LOSS OF FM H_2O INVENTORY (WHILE MACHINE OFF REACTOR FACE)	Events D-IE-01 and D-IE-02 are contributors to a loss of cooling to the fuel bundles that are in the machine while on transit to unload to the spent fuel port. It is identical to D-IE-01 in terms of plant response and consequence. Therefore, it is grouped with D-IE-01 under event GE-08.	GE-08	LOSS OF COOLING TO FUEL WHILE IN THE FM HEAD (WHILE MACHINE OFF REACTOR FACE)
D-IE-04	INADVERTENT UNCLAMPING OF THE FM SNOUT (SHIELD & CLOSURE PLUG REMOVED)	This event triggers the same plant response as a small LOCA event with very similar consequences because no fuel ejection occurs. It is grouped under GE-09.	GE-09	FM INDUCED LOCA (WHILE MACHINE CLAMPED)
D-IE-05	INADVERTENT UNCLAMPING OF THE FM SNOUT (SHIELD PLUG IN PLACE)	This event is not distinguishable from event D-IE-04 in terms of plant response. It is, therefore, grouped under GE-09.	GE-09	FM INDUCED LOCA (WHILE MACHINE CLAMPED)

Logic Diagram Index	Basic Event Description	Rationale for Events Grouping by Plant Response	Grouped Event ID	Description of the Grouped Event
D-IE-10	INADVERTENT "Z" MOTION OF THE SNOUT WHILE BRIDGE IS MOVING (WHILE MACHINE UNCLAMPED)	This event occurs when a fuelling machine is traversing the reactor face (FM off–reactor) and an inadvertent Z motion of the snout occurs. This could cause a Fitting Failure" event that results in a leak/LOCA without fuel ejection. Due to very little thrust that the "Z" drive can exert on the end fitting, it is not expected that a complete rupture of the end fitting would occur. This event is similar in plant response and consequence to the event D-IE-11. It is, therefore, grouped under the grouped event GE-10.	GE-10	FM INDUCED END FITTING FAILURE (WHILE MACHINE UNCLAMPED)
D-IE-11	INADVERTENT MOTION OF THE BRIDGE WHILE SNOUT NOT FULLY RETRACTED	This event occurs when a fuelling machine is unclamped after successful loading/unloading operation or the FM is homing on to an end fitting and bridge starts to move. as the snout is not retracted yet from the reactor face; an inadvertent Y motion of the bridge occurs while the machine is in this position. The result of this event would be a bending of the fitting on which FM was previously clamped and / or bending of one of the adjacent fittings. Due to the limited snout span on "Z" axis, the event is highly unlikely to result in a rupture of the fitting. This event is similar in plant response and consequence to the event D-IE-10. It is, therefore, grouped under the grouped event GE-10	GE-10	FM INDUCED END FITTING FAILURE (WHILE MACHINE UNCLAMPED)
H-IE-01	LARGE HEADER BREAKS IN THE HEAT TRANSPORT SYSTEM	This event comprises all breaks within the HT system boundaries that result in a discharge rate grater than that resulting from a feeder break. It is grouped together with other events under GE-11.	GE-11	LARGE LOSS OF COOLANT ACCIDENT
H-IE-02	LARGE BREAK ON THE ECI ROH INTERCONNECT LINE	Both in terms of plant response and in terms of consequences, this event is similar to a ROH break and is grouped under GE-11.	GE-11	LARGE LOSS OF COOLANT ACCIDENT
H-IE-16	LARGE BREAK ON THE ECI INJECTION LINE, DOWNSTREAM OF THE RUPTURE DISCS	This event is grouped as a contributor under the large loss of coolant grouped event. Both in terms of plant response and in terms of consequences, this event is grouped under GE-11.	GE-11	LARGE LOSS OF COOLANT ACCIDENT

Logic Diagram Index	Basic Event Description	Rationale for Events Grouping by Plant Response	Grouped Event ID	Description of the Grouped Event
H-IE-05	FEEDER BREAK WITH FLOW STAGNATION	The plant response following a break in the feeder event is very similar to that resulting from a small break in either the reactor inlet or outlet header. However, for a certain size of inlet feeder break sizes, the fuel in the downstream channel become overheated leading to a pressure tube rupture. Thus, plant response following a feeder stagnation break with flow stagnation is different, and this event is examined separately. This event is grouped under GE-12.	GE-12	FEEDER BREAK WITH FLOW STAGNATION
K-IE-01	FEEDER BREAK WITH FLOW STAGNATION	This event is a repeat of event H-IE-05 (feeder break with flow stagnation) and, therefore, it is already covered. The event was obtained in two instances of the MLDs development: once from the development of the sub-top category "Single Channel Flow Impairments" and the second time from the development of the sub-top category "In-Core Small LOCA Failures".	GE-12	FEEDER BREAK WITH FLOW STAGNATION
H-IE-07	PRESSURE TUBE AND CALANDRIA TUBE RUPTURE	Pressure tube rupture followed by consequential calandria tube rupture is a scenario results in draining of the HTS coolant into the moderator. However, due to its specific impact upon moderator as a heat sink, this event is treated as a unique grouped event, GE-13.	GE-13	PRESSURE TUBE AND CALANDRIA TUBE RUPTURE
H-IE-12	LOSS OF RCW SUPPLY TO PUMP SEALS COOLERS	This event is a small LOCA contributor, because it may lead to loss of HTS coolant through the damaged seals, after approximately 1 hour of loss of RCW cooling. Therefore, its derived frequency will be added to those of the Small LOCA events and its analysis will be covered by the analysis for grouped event GE-14, "Small LOCA Inside Reactor Building".	GE-14	SMALL LOSS OF COOLANT ACCIDENT (NO CONTAINMENT BY-PASS)

Logic Diagram Index	Basic Event Description	Rationale for Events Grouping by Plant Response	Grouped Event ID	Description of the Grouped Event
H-IE-14	SMALL HTS HEADER BREAKS (Less than 2.5% RIH equiv. Break)	A number of events, which may cause a coolant discharge from the HT system within approximately 2.5% RIH equivalent break discharge, were identified in this report. However, even though they were defined as small LOCAs mostly in terms of the timing of their progression and less in terms of consequences, only a few are grouped under the SLOCA group itself. Therefore, the grouping of many SLOCA events was carried out in the idea to optimize the trade-off between frequency versus consequence to obtain the best estimate results. In this table, the H-IE-14 event is grouped together with and with "End Fitting Breaks with Discharge Outside the Annulus Gas" events H-IE-12 and H-IE-15 respectively.	GE-14	SMALL LOSS OF COOLANT ACCIDENT (NO CONTAINMENT BY-PASS)
H-IE-15	END FITTING BREAKS (DISCHARGE OUTSIDE ANNULUS GAS)	This event triggers the same plant response as a small header break. Events H-IE-12, H-IE-14 and H-IE-15 are grouped under GE-14.	GE-14	SMALL LOSS OF COOLANT ACCIDENT (NO CONTAINMENT BY-PASS)
H-IE-09	PRESSURE TUBE RUPTURE (CALANDRIA TUBE INTACT)	Pressure tube rupture, event , is a heat transport leak through annulus gas bellows into containment. Plant response and consequences are quite different from the other LOCA events, as the discharge rate lies within the capacity of P&IC Feed pump. This event is grouped under event GE-15.	GE-15	PRESSURE TUBE RUPTURE (CALANDRIA TUBE INTACT)
J-IE-01	LOSS OF MAIN FW SUPPLY DUE TO PUMPS/VALVES FAILURE	This event belongs to a group of events that result in mechanical failures in the FW system. The end result of these failures is loss of main feedwater supply to the SGs. Event J-IE-01 triggers a plant response, with consequences that are similar to those of events J-IE-02, J-IE-06 and J-IE-08, all of which are grouped event under GE-16.	GE-16	LOSS OF FW FLOW DUE TO FAILURES OF ACTIVE MECHANICAL / C&I COMPONENTS
J-IE-02	LOSS OF PEGGING STEAM TO DEAERATOR	Event J-IE-02 results in inadequate NPSH for FW pumps due to de-pressurization of the Deaerator. There is a potential for loss of main feedwater water pumps. Therefore, it will be under grouped event GE-16.	GE-16	LOSS OF FW FLOW DUE TO FAILURES OF ACTIVE MECHANICAL / C&I COMPONENTS

Logic Diagram Index	Basic Event Description	Rationale for Events Grouping by Plant Response	Grouped Event ID	Description of the Grouped Event
J-IE-06	STEAM GENERATOR LEVEL CONTROL FAILS HIGH	This event will lead to a loss of main feedwater supply to the SGs, similar to that caused by events J-IE-01 and J-IE-02. (SG Level Control function is carried out by manoeuvring the Level Control Valves placed on the main FW lines). It is, therefore, grouped under event GE-16.	GE-16	LOSS OF FW SUPPLY DUE TO FAILURES OF ACTIVE MECHANICAL / C&I COMPONENTS
J-IE-08	FW CONTROL VALVES / CHECK VALVES FAILURE	This event will lead to a loss of main feedwater supply to the SGs, similar to that caused by events J-IE-01, J-IE-02 and J-IE-06. Therefore, the event is grouped under GE-16 event.	GE-16	LOSS OF FW SUPPLY DUE TO FAILURES OF ACTIVE MECHANICAL / C&I COMPONENTS
J-IE-03	SYMMETRIC STEAM GENERATOR BLOWDOWN LINE BREAK OUTSIDE REACTOR BUILDING	Even though classified as flow impairments, Steam Generator Blowdown Line Breaks are loss of inventory events which unfold very slowly in time. Due to the fact that there are no specific indications and/or alarms to reveal them, these events may cause FW inventory depletion to the point that reactor setback is triggered on low level in the Deaerator tank. The plant response to this event is similar to that for J-IE-03. Therefore, J-IE-03 and J-IE-04 are grouped under event GE-17.	GE-17	SYMMETRIC STEAM GENERATOR BLOWDOWN LINE BREAK OUTSIDE REACTOR BUILDING
J-IE-04	SYMMETRIC STEAM GENERATOR BLOWDOWN LINE BREAK INSIDE REACTOR BUILDING	For event J-IE-04 the plant response is similar with the one for event J-IE-03. For both events' progression there is a major mitigating function available -automatic isolation of the break via the SGLC program. Because J-IE-04 event consequences are covered by the analysis for event J-IE-03, these events are grouped under event GE-17.	GE-17	SYMMETRIC STEAM GENERATOR BLOWDOWN LINE BREAK OUTSIDE REACTOR BUILDING
J-IE-05	ASYMMETRIC STEAM GENERATOR BLOWDOWN LINE BREAK INSIDE REACTOR BUILDING	The plant response for this event is different from events J-IE-03 and J-IE-04. . The major difference between the two plant responses is that event J-IE-05 has an operator action to close the boiler blowdown isolation valves instead of automatic isolation via SGLC for event J-IE-03. Therefore, it is considered as a unique group event.	GE-18	ASYMMETRIC STEAM GENERATOR BLOWDOWN LINE BREAK INSIDE REACTOR BUILDING
J-IE-12	MAIN STEAM ISOLATION VALVE SPURIOUS CLOSURE	Closure of the Main Steam Isolation Valve leads to an overpressure transient in of one SG. Therefore, it is considered as a unique group event.	GE-19	MAIN STEAM ISOLATION VALVE SPURIOUS CLOSURE

Logic Diagram Index	Basic Event Description	Rationale for Events Grouping by Plant Response	Grouped Event ID	Description of the Grouped Event
J-IE-09	REHEAT STOP INTERCEPT VALVE CLOSURE	Reheat steam flow is much less than the flow through a single GSV, thus the overpressure transient following the spurious closure of a reheat stop valve is minor. The plant response and consequence of this event are bounded by those TSVs or GSVs closure (J-IE-10 and J-IE-11 respectively), all which of which are , therefore, grouped under GE-20.	GE-20	STEAM GENERATOR PRESSURE HIGH
J-IE-10	EMERGENCY STOP VALVE CLOSURE	This event is a spurious signal (e.g. spurious TG trip) resulting in closure of all TSVs and GSVs and cause main steam line over-pressurization. The MSSVs will open to relieve the pressure. This event is grouped under GE-20.	GE-20	STEAM GENERATOR PRESSURE HIGH
J-IE-11	SPURIOUS TSV/GOVERNOR CONTROL VALVE CLOSURE	For reasons given for events J-IE-09 and J-IE-10, this event is grouped under event GE-20.	GE-20	STEAM GENERATOR PRESSURE HIGH
K-IE-03	ONE HTS PUMP FAILS TO RUN	This event leads to a partial loss of HTS flow. The plant response and consequence of this event are the same as those for events K-IE-03 and K-IE-04. It is, therefore, grouped under event GE-21.	GE-21	PARTIAL LOSS OF HTS PUMPED FLOW
K-IE-04	TWO OPPOSITE HTS PUMPS FAIL TO RUN	This event is expected to occur due to combinations of mechanical failures of HTS pumps, followed by their trip. This event leads to a partial loss of HTS flow. The plant response and consequence of this event are the same as those for events K-IE-03 and K-IE-05. It is, therefore, grouped under event GE-21.	GE-21	PARTIAL LOSS OF HTS PUMPED FLOW
K-IE-05	ONE HTS PUMP BEARING OR SHAFT SEIZURE	This event has similar plant response and consequences K-IE-03 and K-IE-04 and, therefore, it is grouped under GE-21.	GE-21	PARTIAL LOSS OF HTS PUMPED FLOW

Logic Diagram Index	Basic Event Description	Rationale for Events Grouping by Plant Response	Grouped Event ID	Description of the Grouped Event
K-IE-07	LOSS OF CLASS 4 POWER SUPPLY TO TWO HTS PUMPS	This event may occur due to buses 5314-BUA or 5314-BUB failures, causing de-energization of the pump motors. Flow reduction would cause reactor setback and trip on low HTS flow (both SDS1 /SDS2), followed by turbine trip. This event will be analysed under grouped event GE-22.	GE-22	TOTAL LOSS OF HTS PUMPED FLOW
L-IE-04	GROUP CONTROLLERS ON P&IC *DCS* PARTITION FAIL "HIGH"	This event is grouped under GE-23 if the dual DCS segment fails high on P&IC system .	GE-23	HEAT TRANSPORT SYSTEM PRESSURE CONTROL FAILS HIGH
L-IE-06	FEED VALVES FAIL OPEN AND BLEED VALVES FAIL CLOSE	This event, together with event L-IE-04 and L-IE-13 are grouped under event GE-23.	GE-23	HEAT TRANSPORT SYSTEM PRESSURE CONTROL FAILS HIGH
L-IE-13	PRESSURIZER HEATERS FAIL ON	This event, together with event L-IE-04 and L-IE-06 are grouped event under GE-23.	GE-23	HEAT TRANSPORT SYSTEM PRESSURE CONTROL FAILS HIGH
L-IE-01	GROUP CONTROLLERS ON P&IC *DCS* PARTITION FAIL "LOW"	This event triggers the same plant response as events L-IE-02, L-IE-03, L-IE-05 and L-IE-11 with similar consequences. Therefore, all these events are grouped under event GE-24.	GE-24	HEAT TRANSPORT SYSTEM PRESSURE CONTROL FAILS LOW
L-IE-02	PRESSURIZER SPRAY VALVES FAIL OPEN	This event triggers the same plant response as events L-IE-01, L-IE-03, and L-IE-05 with similar consequences. Therefore, all these events are grouped under event GE-24.	GE-24	HEAT TRANSPORT SYSTEM PRESSURE CONTROL FAILS LOW
L-IE-03	BLEED VALVES FAIL OPEN	This event triggers the same plant response as events L-IE-01, L-IE-02, and L-IE-05 with similar consequences. Therefore, all these events are grouped under event GE-24.	GE-24	HEAT TRANSPORT SYSTEM PRESSURE CONTROL FAILS LOW
L-IE-05	PRESSURIZER HEATERS FAIL OFF	This event triggers the same plant response as events L-IE-01, L-IE-02, and L-IE-03 with similar consequences. Therefore, all these events are grouped under event GE-24.	GE-24	HEAT TRANSPORT SYSTEM PRESSURE CONTROL FAILS LOW
L-IE-11	FEED VALVES FAIL CLOSE	This event leads to the same plant response as with a consequence that is bounded by the consequences of event L-IE-03, which are more severe. Therefore, it is grouped under event GE-24	GE-24	HEAT TRANSPORT SYSTEM PRESSURE CONTROL FAILS LOW

Logic Diagram Index	**Basic Event Description**	**Rationale for Events Grouping by Plant Response**	**Grouped Event ID**	**Description of the Grouped Event**
L-IE-10	PIPE BREAK DOWNSTREAM OF THE PRESSURIZER SPRAY VALVES	This event is leads to a loss of coolant accident (LOCA) from HTS via the P&IC system. It has a similar plant response as L-IE-12. This event is grouped as event GE-25.	GE-25	LOCA INDUCED BY P&IC FAILURES (NO CONTAINMENT BY-PASS)
L-IE-12	PIPE BREAK UPSTREAM OF PRESSURIZER RELIEF VALVES	This event, similar to event L-IE-10, contributes to a loss of coolant accident (LOCA) from HTS via the P&IC system. Both events have a similar plant response and consequence. Therefore, they are grouped under event GE-25.	GE-25	LOCA INDUCED BY P&IC FAILURES (NO CONTAINMENT BY-PASS)
H-IE-03	LOSS OF RCW SUPPLY TO HTS PUMPS' BEARINGS (Spurious closure of any one valve on the RCW supply lines)	This event causes loss of cooling to the HTS pumps bearings, which in the absence of any mitigating action or due to failure of pump protective trips, leads to destruction of journal bearings high vibration and a potential for multiple point LOCA. This type of break may lead to a large LOCA accident that cannot be mitigated by ECC. This event is grouped under GE-26.	GE-26	LARGE LOSS OF COOLANT ACCIDENT DUE TO SEVERE PUMP DAMAGE
L-IE-08	ONE OR BOTH HTS LIQUID RELIEF VALVE(S) FAIL(S) OPEN	This event is under the grouped event GE-27. It has the potential to lead to a small LOCA if the automatic bottle up on the bleed condenser fails.	GE-27	HTS DE-PRESSURIZATION TRANSIENT
L-IE-09	PRESSURIZER RELIEF VALVE(S) FAIL(s) OPEN	This event leads to the same plant response as L-IE-08 and is grouped under event GE-27.	GE-27	HTS DE-PRESSURIZATION TRANSIENT
Q-IE-08	PT LEAKS INTO THE ANNULUS GAS WITHOUT BURSTING OF CHANNEL BELLOWS (within pressurizing pumps capacity)	This event is a HTS leaks produced by a crack in a Pressure Tube that has not reached the critical length. Due to the small size of the crack, the discharge flow accommodated by the annulus gas system and the channel bellows do not burst. The Plant response to this event is quite similar with other HTS leaks although at a slower rate. Several means of detecting the event by the operator are : a) Presence of liquid in the annulus gas; b) High pressure in the annulus gas; c) High dewpoint; d) High drain tank 34980-TK1 level. This event is grouped under GE-28.	GE-28	HTS LEAKS INTO ANNULUS GAS
D1-IE-04	MECHANICAL DAMAGE TO SPENT FUEL DURING STORAGE	This event is grouped under GE-29.	GE-29	MECHANICAL DAMAGE TO SPENT FUEL DURING STORAGE

Logic Diagram Index	Basic Event Description	Rationale for Events Grouping by Plant Response	Grouped Event ID	Description of the Grouped Event
M-IE-06	LOSS OF LTCS FLOW (LTCS FAILURES WHEN HTS IS PARTIALLY DRAINED)	This event is leads to loss of heat sink during shutdown. It is grouped under Grouped Event GE-30 .	GE-30	LOSS OF LTCS AS HEAT SINK - REACTOR SHUTDOWN AND DRAINED TO THE HEADERS
M-IE-08	LOSS OF LTCS COOLING (LTCS FAILURES WHEN HTS IS PARTIALLY DRAINED)	This event is similar in terms of plant response and consequences with the event M-IE-06. Therefore, it is grouped for analysis under Grouped Event GE-30.	GE-30	LOSS OF LTCS AS HEAT SINK - REACTOR SHUTDOWN AND DRAINED TO THE HEADERS
M-IE-16	LOSS OF LTCS FLOW (LTCS FAILURES WHEN HTS FULL AND DE-PRESSURIZED)	This event leads to loss of heat sink during shutdown. It is grouped under Grouped Event GE-31: "Loss of LTCS cooling to fuel while reactor shutdown and de-pressurized".	GE-31	LOSS OF LTCS COOLING TO FUEL, WHILE REACTOR SHUTDOWN AND DE-PRESSURIZED
M-IE-17	LOSS OF LTCS COOLING (WHEN HTS FULL AND DE-PRESSURIZED)	This event is similar in terms of plant response and consequences with the event M-IE-06. Therefore, it is grouped for analysis under Grouped Event GE-31.	GE-31	LOSS OF LTCS COOLING TO FUEL, WHILE REACTOR SHUTDOWN AND DE-PRESSURIZED
M-IE-02	BULK LOSS OF REGULATION (while not in GSS)	This event is going to be analysed together with M-IE-03 as grouped event GE-32 - Bulk core power excursion while reactor shutdown.	GE-32	BULK CORE POWER EXCURSION (REACTOR SHUTDOWN)
M-IE-03	INADVERTENT POISON REMOVAL (WHILE NOT IN GSS)	This event is going to be analysed together with M-IE-02 as grouped event GE-32 - Bulk core power excursion while reactor shutdown.	GE-32	BULK CORE POWER EXCURSION (REACTOR SHUTDOWN)
R-IE-01	INTERFACING LOCA THROUGH LTC SYSTEM (SMALL LOCA WITH CONTAINMENT BY-PASS)	This event results in blowback from the HTS system into the LTC system There is double or triple isolation on different lines into the LTC system from the HTS. The LTC system's ultimate strength is capable to handle the normal operating pressure of the HTS system. This event is grouped under GE-33.	GE-33	INTERFACING LOCA THROUGH LTC SYSTEM (SMALL LOCA WITH CONTAINMENT BY-PASS)

Logic Diagram Index	Basic Event Description	Rationale for Events Grouping by Plant Response	Grouped Event ID	Description of the Grouped Event
Q-IE-04	BLEED COOLER TUBE FAILURES (LEAKS WITH CONTAINMENT BY-PASS)	This event is similar to HTS leaks except that the leak occurs into the RCW system. This constitutes a containment bypass event. This event is grouped under event GE-34.	GE-34	HTS HX TUBE RUPTURE - HTS LEAKS WITH CONTAINMENT BY-PASS
Q-IE-05	HTS PUMPS SEAL COOLERS TUBE FAILURES (LEAKS WITH CONTAINMENT BY-PASS)	This event is similar in terms of plant response and consequences with the event Q-IE-04. Therefore, it is grouped for analysis under Grouped Event GE-34.	GE-34	HTS HX TUBE RUPTURE - HTS LEAKS WITH CONTAINMENT BY-PASS
Q-IE-02	SINGLE SG TUBE RUPTURE (LEAKS WITH CONTAINMENT BY-PASS)	This event leads to release of radioactivity through the secondary side i.e. a containment bypass event. This event is grouped under GE-35.	GE-35	SINGLE STEAM GENERATOR TUBE RUPTURE (LEAKS WITH CONTAINMENT BY-PASS)
U-IE-01	ASYMMETRIC FEEDWATER LINE BREAK OUTSIDE RB DOWNSTREAM OF FW REG. VALVES.	There are several FW line break locations defined in this document. They may have some similarities, but each one of them triggers in fact a somewhat different plant response. Each FW line break event determines a loss of FW inventory in a specific failure mode depending on the break location. Asymmetric feedwater line break outside RB, downstream of feedwater regulating valve station will cause loss of feedwater supply to a single boiler with the specificity that the existing inventory in the affected SG can still be used for cooldown. Also, this event does not trigger ECC signal in case a controlled cooldown is available. This event is grouped under GE-36.	GE-36	ASYMMETRIC FEEDWATER LINE BREAK OUTSIDE RB DOWNSTREAM OF FW REG. VALVES.
U-IE-02	ASYMMETRIC BREAK INSIDE REACTOR BUILDING BETWEEN STEAM GENERATOR CHECK VALVE AND CONTAINMENT WALL	This event is similar with in terms of plant response and consequences with U-IE-03, yet mitigating its consequences is less challenging due to the available inventory in the affected boiler. This event is grouped under event GE-37.	GE-37	ASYMMETRIC BREAK INSIDE REACTOR BUILDING BETWEEN STEAM GENERATOR CHECK VALVE AND CONTAINMENT WALL

Logic Diagram Index	Basic Event Description	Rationale for Events Grouping by Plant Response	Grouped Event ID	Description of the Grouped Event
U-IE-03	ASYMMETRIC BREAK INSIDE REACTOR BUILDING BETWEEN CHECK VALVE AND STEAM GENERATOR	Even though event U-IE-01 may seem similar with other asymmetric FW line break events in it terms of immediate consequences, it triggers a different plant response. The affected boiler loses its inventory. This event therefore will be analysed as GE-38.	GE-38	ASYMMETRIC BREAK INSIDE REACTOR BUILDING DOWNSTREAM STEAM GENERATOR CHECK VALVE
U-IE-04	SYMMETRIC FEEDWATER LINE BREAK OUTSIDE RB UPSTREAM OF FW REGULATING STATION	Event U-IE-04 would lead to a loss of feedwater to both SGs. This event is grouped under GE-39.	GE-39	SYMMETRIC FEEDWATER LINE BREAK OUTSIDE RB UPSTREAM OF FW REGULATING STATION
V-IE-01	ASDVs SPURIOUSLY OPEN (STEAM GENERATOR DE-PRESSURIZATION FAILURES)	Events V-IE-01, V-IE-02, V-IE-03 and V-IE-06 are all causing a de-pressurization of the SGs with loss of steam/FW inventory. In terms of consequences they are not significant. They are lead to a turbine unloading up to approximately 80%FP level, because the loss of steam and preservation of SG level. This event is grouped under the GE-40.	GE-40	STEAM GENERATOR DE-PRESSURIZATION
V-IE-02	SPURIOUS OPENING OF ONE MSSV (STEAM GENERATOR DE-PRESSURIZATION FAILURES)	This event has a similar plant response and consequence as V-IE-01 and is grouped under GE-40.	GE-40	STEAM GENERATOR DE-PRESSURIZATION
V-IE-03	DEAERATOR PRESSURE CONTROL FAILS HIGH - EXCESSIVE PEGGING STEAM DISCHARGE (STEAM GENERATOR DE-PRESSURIZATION FAILURES)	This event has a similar plant response and consequence as V-IE-01 and is grouped under GE-40.	GE-40	STEAM GENERATOR DE-PRESSURIZATION
V-IE-06	SMALL STEAM LINE BREAKS (STEAM GENERATOR DE-PRESSURIZATION FAILURES)	This event has a similar plant response and consequence as V-IE-01 and is grouped under GE-40.	GE-40	STEAM GENERATOR DE-PRESSURIZATION

Logic Diagram Index	Basic Event Description	Rationale for Events Grouping by Plant Response	Grouped Event ID	Description of the Grouped Event
J-IE-07	EXCESSIVE FEEDWATER FLOW TO ONE STEAM GENERATOR	The event is leads to a high level in the SG and a turbine trip on high SG level would be triggered shortly. Due to CSDVs inhibit, the steam pressure will be released by MSSVs lifting. In the worst consequences expected for this event are water carryover into the steam lines and potential for water-hammer effects. This event is bounded by main steam line break inside the RB. Therefore this event is grouped under GE-41 "Main Steam Line Breaks inside RB".	GE-41	MAIN STEAM LINE BREAKS INSIDE REACTOR BUILDING
J-IE-13	EXCESSIVE FEEDWATER FLOW TO BOTH STEAM GENERATORS	This event is similar to J-IE-07 and is grouped under GE-41.	GE-41	MAIN STEAM LINE BREAKS INSIDE REACTOR BUILDING
V-IE-05	MAIN STEAM LINE BREAKS INSIDE REACTOR BUILDING (STEAM GENERATOR DE-PRESSURIZATION FAILURES)	This event triggers ECC and creates harsh environment for the equipment in the turbine building. Heat transport system pumps may stop as a result. RB cooling may be affected to the point that damage to some internal structure walls may be created. The event is grouped separately as GE-41.	GE-41	MAIN STEAM LINE BREAKS INSIDE REACTOR BUILDING
V-IE-04	MAIN STEAM LINE BREAKS INSIDE TURBINE BUILDING (STEAM GENERATOR DE-PRESSURIZATION FAILURES)	This event triggers a specific plant response. It causes harsh environment in the TB. It grouped separately as GE-42.	GE-42	MAIN STEAM LINE BREAKS INSIDE TURBINE BUILDING
V-IE-07	MAIN STEAM BALANCE HEADER BREAKS (STEAM GENERATOR DE-PRESSURIZATION FAILURES)	This event's consequences are bounded by those of the Main Steam Line Break in the TB event. Therefore, it is grouped under GE-42 event.	GE-42	MAIN STEAM LINE BREAKS INSIDE TURBINE BUILDING
V-IE-08	SPURIOUS OPENING OF ALL MSSVs (STEAM GENERATOR DE-PRESSURIZATION FAILURES)	This event's consequences are bounded by those of the Main Steam Line Break in the TB event. Therefore, it is grouped under GE-42 event.	GE-42	MAIN STEAM LINE BREAKS INSIDE TURBINE BUILDING

Logic Diagram Index	Basic Event Description	Rationale for Events Grouping by Plant Response	Grouped Event ID	Description of the Grouped Event
D-IE-07	LOSS OF FM H_2O SYSTEM COOLING	This event is one of the contributors to failures to remove heat from the spent fuel by the FM water cooling circuit. This circuit has two alternate cooling systems serving it: "FM Water System" and "FM Emergency Water System". The event specifically addresses the loss of cooling to the water in the fuelling machine due to loss of RCW supply to the FM H_2O System cooler.	GE-43	LOSS OF HEAT SINK TO FUEL WHILE FM UNCLAMPED
D-IE-08	LOSS OF FM H_2O SYSTEM INVENTORY	This event is one of the contributors to failures to remove heat from the spent fuel by the FM water cooling circuit. It specifically addresses the loss of cooling to the spent fuel unloaded by the machine due to due to loss of water inventory from the FM water cooling circuit. This may be due to leakages from hoses or valves failures to remain closed during FM duties.	GE-43	LOSS OF HEAT SINK TO FUEL WHILE FM UNCLAMPED
D-IE-09	LOSS OF FM H_2O SYSTEM CIRCULATION	This event is one of the contributors to failures to remove heat from the spent fuel by the FM water cooling circuit. The event may occur due to failures of both P&IC water supply and "FM Emergency Water System" to provide coolant circulation past the spent fuel.	GE-43	LOSS OF HEAT SINK TO FUEL WHILE FM UNCLAMPED
H-IE-13	BLOWBACK OF HTS COOLANT THROUGH EMERGENCY COOLANT INJECTION SYSTEM (LOCA NO CONTAINMENT BY-PASS)	Due to the extremely high resistance of the rupture discs against thrust exerted by the HTS coolant plus a check valve and a closed motorized valve in the line, it is very unlikely that the HTS coolant blowback into ECI would occur. However, due to the potential consequence of this event it is grouped under GE-44.	GE-44	BLOWBACK OF HTS COOLANT THROUGH EMERGENCY COOLANT INJECTION SYSTEM (LOCA NO CONTAINMENT BY-PASS)
D1-IE-01	LOSS OF SPENT FUEL STORAGE BAY WATER INVENTORY	The only viable means of inventory loss is by evaporation, after a prolonged loss of make-up and cooling water. Piping coming in and out of the bay is provided with siphon breakers to prevent depletion of the water inventory. The event will be analysed as grouped event GE-45, "Loss of SF Storage Bay Heat Sink".	GE-45	LOSS OF SF STORAGE BAY HEAT SINK
D1-IE-02	LOSS OF SF STORAGE BAY WATER CIRCULATION	Event D1-IE-02 contributes together with events D1-IE-01 and D1-IE-03 to the loss of SF Bay heat sink. Therefore, they are grouped under GE-45.	GE-45	LOSS OF SF STORAGE BAY HEAT SINK

Logic Diagram Index	Basic Event Description	Rationale for Events Grouping by Plant Response	Grouped Event ID	Description of the Grouped Event
D1-IE-03	LOSS OF SPENT FUEL STORAGE BAY COOLING	See comment for events D1-IE-01, D1-IE-02. This event is grouped under GE-45, "Loss of SF Storage Bay Heat Sink".	GE-45	LOSS OF SF STORAGE BAY HEAT SINK
D2-IE-03	LOSS OF TRANSFER TUBE COOLING	This event is a contributor to grouped event GE-46 "Loss of cooling to fuel bundles while in transfer to SF Reception Bay".	GE-46	LOSS OF COOLING TO FUEL BUNDLES WHILE IN TRANSFER TO SF RECEPTION BAY
D2-IE-04	LOSS OF TRANSFER TUBE H_2O INVENTORY	This event is a contributor to grouped event GE-46 "Loss of cooling to fuel bundles while in transfer to SF Reception Bay".	GE-46	LOSS OF COOLING TO FUEL BUNDLES WHILE IN TRANSFER TO SF RECEPTION BAY
D2-IE-01	FUEL DAMAGE WHILE PASSING THROUGH ONE OF 4 FULL-BORE ISOLATION/CONTAINMENT VALVES	This event will be analysed as part of the grouped event "Mechanical Damage to Fuel while in Transfer to SF Reception Bay".	GE-47	MECHANICAL DAMAGE TO FUEL WHILE IN TRANSFER TO SF RECEPTION BAY
D2-IE-02	FUEL DAMAGE DUE TO INADVERTENT OPERATION OF SF TRANSFER EQUIPMENT	This event will be analysed as part of the grouped event GE-47 "Mechanical Damage to Fuel while in Transfer to SF Reception Bay".	GE-47	MECHANICAL DAMAGE TO FUEL WHILE IN TRANSFER TO SF RECEPTION BAY
D3-IE-01	FM BRIDGE DRIFTING DOWN EVENLY (FM CLAMPED)	This event represents an FM Induced LOCA caused by a drifting of the FM bridge. This event is grouped under GE-48.	GE-48	FM INDUCED LOCA (WHILE MACHINE CLAMPED)
D3-IE-02	INADVERTENT MOTION OF THE FM BRIDGE	This event represents an FM Induced LOCA due to bridge motion while FM clamped. This event is grouped under "FM Induced LOCA while machine clamped", GE-48.	GE-48	FM INDUCED LOCA (WHILE MACHINE CLAMPED)
D3-IE-03	FM BRIDGE DRIVING UP / DOWN UNEVENLY (FM CLAMPED)	This event's consequences are similar with those of events D3-IE-01, D3-IE-02, D3-IE-04. . The event is grouped under the grouped event FM Induced LOCA, GE-48.	GE-48	FM INDUCED LOCA (WHILE MACHINE CLAMPED).
D3-IE-04	FM BRIDGE DRIFTING DOWN UNEVENLY - BRAKES FAILURES (FM CLAMPED)	The comments made for events D3-IE-01, D3-IE-02, D3-IE-03-are also applicable to this event. The event is grouped under GE-48.	GE-48	FM INDUCED LOCA (WHILE MACHINE CLAMPED)

Logic Diagram Index	Basic Event Description	Rationale for Events Grouping by Plant Response	Grouped Event ID	Description of the Grouped Event
D4-IE-01	FM BRIDGE DRIVING UP / DOWN UNEVENLY	Both in terms of plant response and in terms of consequences, events D4-IE-01 and D4-IE-02 can be grouped for analysis as FM Induced LOCA events. This event is grouped under GE-49.	GE-49	FM INDUCED LOCA (WHILE MACHINE UN-CLAMPED)
D4-IE-02	FM BRIDGE DRIFTING DOWN UNEVENLY - BRAKES FAILURES (FM UNCLAMPED)	The same comment made for event D4-IE-01 applies to D4-IE-02. This event is grouped under GE-49.	GE-49	FM INDUCED LOCA (WHILE MACHINE UN-CLAMPED)
D-IE-06	LOSS OF HTS COOLANT DUE TO FM WATER SYSTEM FAILURES	This event is a result of various leakages from the HTS via the FM H_2O System while the machine is clamped and connected to the HTS coolant boundary. The consequences of and the plant response to this event would be similar to those of the HTS Leaks events but with one more mitigating function available -isolation of the leakages in the FM H_2O System. The event is grouped under grouped event GE-50 (Loss of HTS coolant due to FM H_2O System Failures).	GE-50	LOSS OF HTS COOLANT DUE TO FM WATER SYSTEM FAILURES
R-IE-02	MULTIPLE STEAM GENERATOR TUBE RUPTURES (SMALL LOCA WITH CONTAINMENT BY-PASS)	Even though event R-IE-02 may seem similar with Q-IE-02 in terms of consequences, it is different in terms of plant response. The considerably higher HTS coolant discharge rate that characterizes R-IE-02 places it in the Small LOCA category of events. This event is grouped under event GE-51.	GE-51	MULTIPLE STEAM GENERATOR TUBE RUPTURES (SMALL LOCA WITH CONTAINMENT BY-PASS)
M-IE-04	LTCS HX TUBE LEAKS (LTCS FAILURES WHEN HTS IS PARTIALLY DRAINED)	This event has a plant response and consequence similar to those of events M-IE-05 and M-IE-07. It leads to a HTS leak. Therefore, all these events are grouped under one event, GE-52.	GE-52	HTS LEAKS WHILE DRAINED TO THE HEADERS
M-IE-05	LTCS PUMP SEALS FAILURES (LTCS FAILURES WHEN HTS IS PARTIALLY DRAINED)	This event is similar in terms of plant response and consequences with the events M-IE-04 and M-IE-07. It is therefore grouped together with these events for analysis as GE-52	GE-52	HTS LEAKS WHILE DRAINED TO THE HEADERS

Logic Diagram Index	Basic Event Description	Rationale for Events Grouping by Plant Response	Grouped Event ID	Description of the Grouped Event
M-IE-07	HTS PIPE LEAKS (LTCS FAILURES WHEN HTS IS PARTIALLY DRAINED)	This event is similar in terms of plant response and consequences with the events M-IE-04 and M-IE-05. It is therefore grouped together with these events for analysis as GE-52.	GE-52	HTS LEAKS WHILE DRAINED TO THE HEADERS
S-IE-05	CONDENSER COOLING WATER LARGE BREAKS	CCWS large break may occur due to the catastrophic failure of a Condenser Inlet/Outlet Expansion Joint. The main safety impact of this event is the flooding of the Turbine Building basement. . Because of this consequence, this event leads to different plant response than a *Loss of Condenser As a Heat Sink* event. Therefore, it grouped under GE-53. Note: this event is addressed by the flooding PSA.	GE-53	LARGE CONDENSER COOLING WATER LINE BREAKS - LOSS OF CONDENSER AS A HEAT SINK
S-IE-06	CONDENSER COOLING WATER SMALL BREAKS	Failures of piping, fittings, welds, gaskets or expansion joints may cause a leak from the CCW system, leading to a flooding of the turbine building. This event is a much slower developing event then S-IE-05 and is grouped under GE-54. Note: this event is addressed by the flooding PSA.	GE-54	SMALL CONDENSER COOLING WATER LINE BREAKS - LOSS OF CONDENSER AS A HEAT SINK
M-IE-09	HTS PIPE LEAKS (LOSS OF LTCS INVENTORY HTS FULL, DEPRESSURIZED)	Events M-IE-09, M-IE-22, M-IE-23 are grouped for analysis under GE-55 - HTS Leaks while HTS Full and Depressurized.	GE-55	HTS LEAKS (LOSS OF LTCS INVENTORY WHILE HTS FULL, DEPRESSURIZED)
M-IE-22	LTCS HX TUBE (LOSS OF LTCS INVENTORY HTS FULL, DEPRESSURIZED)	Events M-IE-09, M-IE-22, M-IE-23 are grouped for analysis under GE-55 - HTS Leaks while HTS Full and Depressurized.	GE-55	HTS LEAKS (LOSS OF LTCS INVENTORY WHILE HTS FULL, DEPRESSURIZED)
M-IE-23	LTCS PUMP SEALS (LOSS OF LTCS INVENTORY HTS FULL, DEPRESSURIZED)	Events M-IE-09, M-IE-22, M-IE-23 are grouped for analysis under GE-55 - HTS Leaks while HTS Full and Depressurized.	GE-55	HTS LEAKS (LOSS OF LTCS INVENTORY WHILE HTS FULL, DEPRESSURIZED)
Y-IE-05	LOSS OF ONE RCW DIVISION	This partial loss of RCW cooling function is grouped under GE-56.	GE-56	TOTAL LOSS OF ONE RCW DIVISION
Y-IE-08	TOTAL LOSS OF RCW FLOW / CIRCULATION	This event has a plant response and consequences that are similar to those of events Y-IE-09 and Y-IE-11. Therefore, they are all analysed as one grouped event GE-57.	GE-57	TOTAL LOSS OF RCW COOLING - REACTOR OPERATING AT FULL POWER

Logic Diagram Index	Basic Event Description	Rationale for Events Grouping by Plant Response	Grouped Event ID	Description of the Grouped Event
Y-IE-09	TOTAL FAILURE OF RSW SCREENWASH SYSTEM	Same comment as the one for event Y-IE-08 applies for grouping of this event (Y-IE-09). It is therefore grouped under GE-57.	GE-57	TOTAL LOSS OF RCW COOLING - REACTOR OPERATING AT FULL POWER
Y-IE-11	FISH /DEBRIS CLOG RSW INTAKE CHANNEL	This event has a plant response and consequence that are similar to those of events Y-IE-08, and Y-IE-11. Therefore, they are all analysed as one grouped event GE-57.	GE-57	TOTAL LOSS OF RCW COOLING - REACTOR OPERATING AT FULL POWER
Y-IE-12	LOSS OF UNIT 1 CLASS IV POWER SUPPLY	This event has a plant response and consequence that are similar to those of events Y-IE-13, Y-IE-14, and Y-IE-15. Therefore, they are all analysed as one grouped event GE-58.	GE-58	TOTAL LOSS OF ONE UNIT CLASS IV POWER SUPPLY
Y-IE-13	SWITCHYARD FAILURE	This event has a plant response and consequence that are similar to those of events Y-IE-12, Y-IE-14, and Y-IE-15. Therefore, they are all analysed as one grouped event GE-58.	GE-58	TOTAL LOSS OF ONE UNIT CLASS IV POWER SUPPLY
Y-IE-14	DUAL FAILURE OF UST AND SST TRANSFORMERS	This event has a plant response and consequence that are similar to those of events Y-IE-12, Y-IE-13, and Y-IE-15. Therefore, they are all analysed as one grouped event GE-58.	GE-58	TOTAL LOSS OF ONE UNIT CLASS IV POWER SUPPLY
Y-IE-15	PARTIAL LOSS OF CLASS 4 POWER	This event has a plant response and consequence that are similar to those of events Y-IE-12, Y-IE-13, and Y-IE-14. Therefore, they are all analysed as one grouped event GE-58.	GE-58	TOTAL LOSS OF ONE UNIT CLASS IV POWER SUPPLY
M-IE-20	LEAKS DURING THROUGH -FLOW DE-FUELLING	This event may occur due to leaks from the FM Water System while the "Through Flow De-fuelling" operations are carried out. This event is analysed separately as GE-59.	GE-59	LEAKS DURING THROUGH - FLOW DE-FUELLING
Y-IE-18	DUAL FAILURE OF GROUP CONTROLLERS (ON ONE DCS PARTITION)	This event will be analysed separately as grouped event GE-60	GE-60	DUAL FAILURE OF GROUP CONTROLLERS (ON ONE DCS PARTITION)
Y-IE-19	DUAL FAILURE OF DCS DATA HIGHWAYS	This event will be analysed separately as grouped event GE-61.	GE-61	DUAL FAILURE OF DCS DATA HIGHWAYS

Logic Diagram Index	Basic Event Description	Rationale for Events Grouping by Plant Response	Grouped Event ID	Description of the Grouped Event
M-IE-21	LEAKS DURING CIGAR INSPECTIONS	This event may occur when channel inspections are carried out while reactor shutdown and de-pressurized. The water system of the Advanced Delivery Machine may leak during channel inspection, therefore causing a loss of primary coolant inventory. It is to be analysed separately because it involves a specific plant response.	GE-62	LEAKS DURING CIGAR INSPECTIONS
Y-IE-01	TOTAL LOSS OF I/A SUPPLY TO PLANT LOADS (REACTOR AT FULL POWER)	This event leads to a loss of instrument air. These failures include compressors failures, dryers failures, pipe breaks failures, etc. It is analysed as grouped event GE-63.	GE-63	TOTAL LOSS OF I/A SUPPLY TO PLANT LOADS (REACTOR AT FULL POWER)
Z-IE-01	TOTAL LOSS OF HVAC IN THE PLANT (REACTOR AT FULL POWER)	This event postulated in order to estimate potential impact of losing HVAC supply to safety loads, while reactor is at full power. This event is grouped under GE-64.	GE-64	TOTAL LOSS OF HVAC IN THE PLANT
Z-IE-02	PARTIAL LOSS OF HVAC IN THE PLANT (REACTOR AT FULL POWER)	This event postulated in order to estimate potential impact of a partial loss of HVAC supply to safety loads, while reactor is at full power. This event is grouped under GE-65.	GE-65	PARTIAL LOSS OF HVAC IN THE PLANT
Y-IE-04	PARTIAL LOSS OF CLASS 3 POWER (REACTOR AT FULL POWER)	This event is postulated in order to analyse the safety impact of a Class III bus failure, while the plant is operating at full power. This event is grouped under GE-66.	GE-66	PARTIAL LOSS OF CLASS 3 POWER (REACTOR AT FULL POWER)
S-IE-01	CCW SCREEN FLOW BLOCKAGE DUE TO FISH / DEBRIS	The event contributes to loss of condenser vacuum as it affects the CCW flow and consequently Condenser cooling (see the comments for S-IE-05). It is grouped event under GE-67: "Loss of Condenser Vacuum".	GE-67	LOSS OF CONDENSER VACUUM
S-IE-02	CCW SCREENWASH SYSTEM FAILURE	This event is part of a group of events resulting in a loss of condenser vacuum. This event is grouped under GE-67.	GE-67	LOSS OF CONDENSER VACUUM

Logic Diagram Index	Basic Event Description	Rationale for Events Grouping by Plant Response	Grouped Event ID	Description of the Grouped Event
S-IE-04	CONDENSER AIR EXTRACTION SYSTEM FAILURE	This event is part of a group of events resulting in a loss of condenser vacuum. This event is grouped under GE-67.	GE-67	LOSS OF CONDENSER VACUUM
V-IE-09	CSDVs SPURIOUSLY OPEN (STEAM GENERATOR DE-PRESSURIZATION FAILURES)	Event V-IE-09 contributes both to Steam Generators De-pressurization group of events and to Loss of Condenser Vacuum group of events. However, due to its more significant impact upon Loss of Condenser Vacuum, it was decided that this event will be grouped together with S-IE-01, and S-IE-04 into GE-67 "Loss of Condenser Vacuum". Because the event may lead to a fast pressurization of the condenser and to a turbine trip, serious damage to LP turbine stages may be inflicted. This potentially great economical loss does not necessarily lead to a great safety impact upon the ACR plant. However, if the CSDVs cannot be re-closed and in case turbine trip causes a loss of Class IV power supply, the fast de-pressurization of SGs would lead to rapid cooldown of the HTS beyond the zero power hot level. As a result of this sequence of events, HTS shrinkage may not be accommodated by the pressurizer inventory. Massive voiding may occur in the core with probably some fuel failures until ECC injection is triggered.	GE-67	LOSS OF CONDENSER VACUUM
T-IE-01	DEAERATOR LEVEL CONTROL VALVE FAILURES	This event may lead to loss of FW supply to the SGs. If level control valves overflood the Deaerator (valves control fails high), FW inventory would be lost at a rate dictated by the Deaerator overpressure protection relief valve. If the valves cause the Deaerator level to drop, the FW pumps would trip and again a loss of FW supply from the Deaerator would occur. Both scenarios can be grouped under Loss of Condensate Supply to the Deaerator, grouped event GE-68.	GE-68	LOSS OF CONDENSATE SUPPLY TO THE DEAERATOR
T-IE-02	CONDENSER LEVEL CONTROL VALVE FAILURES	This event can lead to loss of condensate water supply to the suction of Condensate System pumps, which in turn would cause Loss of Condensate Water Supply to Deaerator, GE-68.	GE-68	LOSS OF CONDENSATE SUPPLY TO THE DEAERATOR

Logic Diagram Index	Basic Event Description	Rationale for Events Grouping by Plant Response	Grouped Event ID	Description of the Grouped Event
T-IE-03	BOTH CONDENSATE PUMPS FAILED	This event leads to loss of Condensate water supply to Deaerator. It is grouped for analysis together with T-IE-01, T-IE-02 and T-IE-04 under grouped event GE-68, "Failures of Condensate Water Supply to Deaerator".	GE-68	LOSS OF CONDENSATE SUPPLY TO THE DEAERATOR
T-IE-04	LOSS OF CONDENSATE INVENTORY (LINE BREAKS)	See comment for T-IE-03.	GE-68	LOSS OF CONDENSATE SUPPLY TO THE DEAERATOR
Y-IE-02	LOSS OF OFFSITE POWER SUPPLY	This event needs to be analysed to estimate the safety impact of loss of power supply to both ACR plants, i.e. in a situation in which one unit cannot help the other by supplying Class IV power to it.	GE-69	LOSS OF OFFSITE POWER SUPPLY
Q-IE-01	HTS LEAKS THROUGH LTC SYSTEM PIPES/VALVES OUTSIDE RB	In terms of consequences and frequency, this event is bounded by HTS leaks event. There are two mechanisms which may lead to occurrence of these leaks: (A) Either double failure of any pair of isolation valves while reactor operates at full power, or (B) Leaks of LTC pump seals and /or external leaks through LTC valves. This event is grouped under GE-70.	GE-70	HTS COOLANT LEAKS THROUGH LTC SYSTEM OUTSIDE REACTOR BUILDING (HTS LEAKS WITH CONTAINMENT BY-PASS)
Y-IE-16	PARTIAL LOSS OF CLASS 2 POWER	A partial loss of Class I or II power (e.g. to one channel) will be addressed by event tree analysis. A representative channel will be selected at the time of analysis based on a review of loads on various Class I and II buses.	GE-71	PARTIAL LOSS OF CLASS 2 POWER
H-IE-04	PARTIAL CHANNEL FLOW BLOCKAGE	This event is similar in terms of plant response and safety consequences with, and it is bounded by, the H-IE-05 event, which is a pressure tube and calandria tube rupture due to feeder stagnation break This event is grouped under GE-72.	GE-72	PARTIAL CHANNEL FLOW BLOCKAGE
K-IE-02	PARTIAL CHANNEL FLOW BLOCKAGE	This event is a repeat of event H-IE-04 (partial channel flow blockage). The event was obtained in two instances of the MLDs development: once in the sub-top category called "Single Channel Flow Impairments" and the second time in the sub-top category "In-Core Small LOCA Failures".	GE-72	PARTIAL CHANNEL FLOW BLOCKAGE

Logic Diagram Index	Basic Event Description	Rationale for Events Grouping by Plant Response	Grouped Event ID	Description of the Grouped Event
Y-IE-17	PARTIAL LOSS OF CLASS 1 POWER	This event is included to be conservative. Based on past PSA experience, this is expected to be screened out based on its very low frequency. This is grouped under GE-73	GE-73	PARTIAL LOSS OF CLASS 1 POWER
J-IE-14	REACTOR OR TURBINE TRIP	This event covers basically unplanned reactor and turbine trips.	GE-74	GENERAL TRANSIENT EVENT
V-IE-09	CSDVs SPURIOUSLY OPEN (STEAM GENERATOR DE-PRESSURIZATION FAILURES)	Event V-IE-09 contributes to Steam Generators De-pressurization and is covered by a general transient (Grouped Event GE-74).	GE-74	GENERAL TRANSIENT
H-IE-06	SEVERE CHANNEL FLOW BLOCKAGE	This event is similar in terms of plant response and safety consequences with the H-IE-05 event, which is a pressure tube and calandria tube rupture due to feeder stagnation break (FBS). H-IE-05 event causes double impact on HTS (both internal channel rupture and severed feeder) whereas H-IE-06 (channel blockage) leads only to internal channel rupture.	GE-75	SEVERE CHANNEL FLOW BLOCKAGE
W-IE-01	LOSS OF SHIELD COOLING SYSTEM WATER INVENTORY	The Shield Cooling System is designed such that water inventory in the Shield Tank cannot be drained (siphon breakers provided at outlet headers). As a result, a pipe break anywhere in the system would result in depletion of the head tank inventory leading ultimately to loss of circulation. Consequences of this event are similar to a loss of water circulation event (see comment for event W-IE-02). Therefore, the events W-IE-02 and W-IE-03 are grouped into grouped event GE-76 "Loss of Shield Cooling System Heat Sink".	GE-76	LOSS OF SHIELD COOLING SYSTEM HEAT SINK
W-IE-02	LOSS OF SHIELD COOLING SYSTEM WATER CIRCULATION	This event causes a prompt drop of the flow rate in the system and therefore a setback on low shield cooling flow. Following the flow stagnation, temperature would rise to the second setback level on high temperature in the shields. The event is going to be analysed under grouped event GE-76.	GE-76	LOSS OF SHIELD COOLING SYSTEM HEAT SINK

Logic Diagram Index	Basic Event Description	Rationale for Events Grouping by Plant Response	Grouped Event ID	Description of the Grouped Event
W-IE-03	LOSS OF SHIELD COOLING SYSTEM HEAT SINK	This event causes a progressive overheating of the shield cooling system water, but the accident progression is slower than for the case of the loss of flow event due to the great thermal power absorption of the shield tank inventory (>100m^3). The shield coolant will overheat until it reaches the setback setpoint on high shield coolant temperature. Several indications /alarms are available to the operator to intervene in case the setback fails. Consequences of this event are bounded by the Grouped event GE-76 and it will be grouped in GE-76.	GE-76	LOSS OF SHIELD COOLING SYSTEM HEAT SINK
B-IE-08	CALANDRIA DRAIN LINE BREAKS OUTSIDE SHIELD TANK (Calandria Inventory is not Preserved)	This event can drain the entire calandria inventory. It will be analyzed as grouped event GE-77.	GE-77	CALANDRIA DRAIN LINE BREAKS OUTSIDE SHIELD TANK
B-IE-02	MODERATOR PIPE LEAKS (outside Calandria Vessel)	Consequences of this event are bounded by those of the event B-IE-05 "Moderator Pipe Breaks Outside Shield Tank". This event will lead to slower moderator drainage transient but the Calandria vessel inventory will be preserved. This event is grouped under GE-78.	GE-78	MODERATOR PIPE LEAKS (OUTSIDE CALANDRIA VESSEL)
X-IE-05	TOTAL LOSS OF INSTRUMENT AIR (REACTOR SHUTDOWN, HTS FULL AND DEPRESSURIZED)	This event is selected to analyse the safety consequences of loss of IA supply while the plant is shutdown, HTS full and depressurized. It is to be analyzed as grouped event GE-79.	GE-79	TOTAL LOSS OF INSTRUMENT AIR (REACTOR SHUTDOWN, HTS FULL AND DE-PRESSURIZED)
X-IE-06	LOSS OF ONE SW DIVISION (RCW AND/OR RSW) (REACTOR SHUTDOWN, HTS FULL AND DE-PRESSURIZED)	This event is postulated in order to estimate potential impact of losing SW supply to safety loads, while reactor is shutdown, HTS full and depressurized. This event is grouped under GE-80.	GE-80	LOSS OF ONE SW DIVISION (RCW AND/OR RSW) (REACTOR SHUTDOWN, HTS FULL AND DE-PRESSURIZED)
X-IE-07	TOTAL LOSS OF CLASS IV POWER (REACTOR SHUTDOWN, HTS FULL AND DE-PRESSURIZED)	This event is postulated in order to estimate potential impact of losing electric power supply to the safety loads, while reactor is shutdown, HTS full and depressurized. This event is grouped under GE-81.	GE-81	TOTAL LOSS OF CLASS IV POWER (REACTOR SHUTDOWN, HTS FULL AND DE-PRESSURIZED)

Logic Diagram Index	Basic Event Description	Rationale for Events Grouping by Plant Response	Grouped Event ID	Description of the Grouped Event
X-IE-08	TOTAL LOSS OF HVAC IN THE PLANT (REACTOR SHUTDOWN, HTS FULL AND DE-PRESSURIZED)	This event is postulated in order to estimate potential impact of losing HVAC to rooms throughout the plant in which safety loads are located, while reactor is shutdown, HTS full and depressurized. This event is grouped under GE-82.	GE-82	TOTAL LOSS OF HVAC IN THE PLANT (REACTOR SHUTDOWN, HTS FULL AND DE-PRESSURIZED)
X-IE-09	TOTAL LOSS OF INSTRUMENT AIR (REACTOR SHUTDOWN, HTS DE-PRESSURIZED AND DRAINED TO THE HEADERS)	This event is selected to analyse the safety consequences of loss of IA supply while the plant is shutdown and HTS drained to the headers. It is grouped event GE-83.	GE-83	TOTAL LOSS OF INSTRUMENT AIR (REACTOR SHUTDOWN, HTS DE-PRESSURIZED AND DRAINED TO THE HEADERS)
X-IE-10	LOSS OF ONE SW DIVISION (RCW AND/OR RSW) (REACTOR SHUTDOWN, HTS DE-PRESSURIZED AND DRAINED TO THE HEADERS)	This event postulated in order to estimate potential impact of losing SW supply to safety loads, while reactor is shutdown and HTS drained to the headers. This event is grouped under GE-84.	GE-84	LOSS OF ONE SW DIVISION (RCW AND/OR RSW) (REACTOR SHUTDOWN, HTS DE-PRESSURIZED AND DRAINED TO THE HEADERS)
X-IE-11	TOTAL LOSS OF CLASS IV POWER (REACTOR SHUTDOWN, HTS DE-PRESSURIZED AND DRAINED TO THE HEADERS)	This event is postulated in order to estimate potential impact of losing electric power supply to the safety loads, while reactor is shutdown and, HTS drained to the headers level. This event is grouped under GE-85	GE-85	TOTAL LOSS OF CLASS IV POWER (REACTOR SHUTDOWN, HTS DE-PRESSURIZED AND DRAINED TO THE HEADERS)
X-IE-12	TOTAL LOSS OF HVAC IN THE PLANT (REACTOR SHUTDOWN, HTS DE-PRESSURIZED AND DRAINED TO THE HEADERS).	This event is postulated in order to estimate potential impact of losing HVAC to rooms throughout the plant in which safety loads are located, while reactor is shutdown, and HTS drained to the headers. This event is grouped under GE-86.	GE-86	TOTAL LOSS OF HVAC IN THE PLANT (REACTOR SHUTDOWN, HTS DEPRESSURIZED AND DRAINED TO THE HEADERS)

Logic Diagram Index	Basic Event Description	Rationale for Events Grouping by Plant Response	Grouped Event ID	Description of the Grouped Event
M-IE-19	FAILURE OF FREEZE PLUGS DURING CHANNEL INSPECTION OR REPLACEMENT (FULL AND DEPRESSURIZED)	This event may occur during channel inspection of replacement. Reactor is shutdown, depressurized and cold and the subjected channel is isolated from the core by freeze plugs. Any failure of these plugs would lead to a small LOCA with complete drainage of the core up to the level of the affected channel. Due to its specific consequences this event is analysed as grouped event GE-87.	GE-87	SMALL LOCA WHILE HTS IS FULL AND DEPRESSURIZED
Y-IE-03	LOSS OF RCW INVENTORY DUE HX GASKET FAILURES	This event may occur due to failure of the gaskets on the RCW side of the plate type HXs. In terms of consequences, it is bounded by the flooding events (see Y-IE-06, Y-IE-07, Y-IE-10) .	N/A	FLOODING EVENT
Y-IE-06	RCW / RSW EXPANSION JOINTS FAILURES	This event leads to a Loss of One RCW/RSW Division. It is covered under flooding PSA.	N/A	FLOODING EVENT
Y-IE-07	LOSS OF RCW DUE TO LOSS OF INVENTORY / PIPE BREAKS	This event leads to a Loss of One RCW Division. It is covered under flooding PSA.	N/A	FLOODING EVENT
Y-IE-10	LOSS OF RSW INVENTORY / PIPE BREAKS	This event leads to a Loss of One RSW Division. It is covered under flooding PSA.	N/A	FLOODING EVENT

Table 3
Initiating Event Frequencies for Grouped Events

ACR Grouped Events List

Grouped Event ID	Description of the Grouped Event	Associated Frequency (event/year)	Source of Frequency Value
GE-01	LOSS OF DEUTERIUM CONCENTRATION CONTROL IN THE COVER GAS SYSTEM	1.07E-03	CANDU Operating Experience
GE-02	HTS LEAKS WITH NO CONTAINMENT BY-PASS	1.34E-01	CANDU Operating Experience
GE-03	MODERATOR PIPE BREAKS OUTSIDE SHIELD TANK	2.00E-04	CANDU Operating Experience
GE-04	PARTIAL LOSS OF MODERATOR HEAT SINK	3.28E-01	CANDU Operating Experience
GE-05	FEEDER BREAK NO FLOW STAGNATION	8.04E-04	CANDU Operating Experience
GE-06	BULK CORE POWER EXCURSION (reactor operating at full power	To Be Determined (TBD)	CANDU Operating Experience
GE-07	REGIONAL CORE POWER EXCURSION (reactor operating at full power)	TBD	CANDU Operating Experience
GE-08	LOSS OF COOLING TO FUEL WHILE IN THE FM HEAD (WHILE MACHINE OFF REACTOR FACE)	8.48E-02	CANDU Operating Experience
GE-09	FM INDUCED LOCA (WHILE MACHINE CLAMPED)	4.70E-02	CANDU Operating Experience
GE-10	FM INDUCED END FITTING FAILURE (WHILE MACHINE UNCLAMPED)	1.67E-03	CANDU Operating Experience
GE-11	LARGE LOSS OF COOLANT ACCIDENT	4.18E-04	CANDU Operating Experience
GE-12	FEEDER BREAK WITH FLOW STAGNATION	TBD	CANDU Operating Experience

Grouped Event ID	Description of the Grouped Event	Associated Frequency (event/year)	Source of Frequency Value
GE-13	PRESSURE TUBE AND CALANDRIA TUBE RUPTURE	4.18E-04	CANDU Operating Experience
GE-14	SMALL LOSS OF COOLANT ACCIDENT (NO CONTAINMENT BY-PASS)	7.07E-03	CANDU Operating Experience
GE-15	PRESSURE TUBE RUPTURE (CALANDRIA TUBE INTACT)	4.18E-03	CANDU Operating Experience
GE-16	LOSS OF FW FLOW DUE TO FAILURES OF ACTIVE MECHANICAL / C&I COMPONENTS	6.01E-02	CANDU Operating Experience
GE-17	SYMMETRIC STEAM GENERATOR BLOWDOWN LINE BREAK OUTSIDE REACTOR BUILDING	4.60E-06	CANDU Operating Experience
GE-18	ASYMMETRIC STEAM GENERATOR BLOWDOWN LINE BREAK INSIDE REACTOR BUILDING	1.94E-06	CANDU Operating Experience
GE-19	MAIN STEAM ISOLATION VALVE SPURIOUS CLOSURE	3.60E-03	CANDU Operating Experience
GE-20	STEAM GENERATOR PRESSURE HIGH	TBD	
GE-21	PARTIAL LOSS OF HTS PUMPED FLOW	2.89E-02	CANDU Operating Experience
GE-22	TOTAL LOSS OF HTS PUMPED FLOW	3.23E-04	CANDU Operating Experience
GE-23	HEAT TRANSPORT SYSTEM PRESSURE CONTROL FAILS HIGH	7.60E-03	CANDU Operating Experience
GE-24	HEAT TRANSPORT SYSTEM PRESSURE CONTROL FAILS LOW	1.06E-02	CANDU Operating Experience
GE-25	LOCA INDUCED BY P&IC FAILURES (NO CONTAINMENT BY-PASS)	TBD	

Grouped Event ID	Description of the Grouped Event	Associated Frequency (event/year)	Source of Frequency Value
GE-26	LARGE LOSS OF COOLANT ACCIDENT DUE TO SEVERE PUMP DAMAGE	TBD	
GE-27	HTS DE-PRESSURIZATION TRANSIENT	1.36E-02	CANDU Operating Experience
GE-28	HTS LEAKS INTO ANNULUS GAS	1.47E-02	CANDU Operating Experience
GE-29	MECHANICAL DAMAGE TO SPENT FUEL DURING STORAGE	9.54E-02	CANDU Operating Experience
GE-30	LOSS OF LTCS AS HEAT SINK - REACTOR SHUTDOWN AND DRAINED TO THE HEADERS	TBD	
GE-31	LOSS OF LTCS COOLING TO FUEL, WHILE REACTOR SHUTDOWN AND DE-PRESSURIZED	TBD	
GE-32	BULK CORE POWER EXCURSION (REACTOR SHUTDOWN)	8.04E-04	CANDU Operating Experience
GE-33	INTERFACING LOCA THROUGH LTC SYSTEM (SMALL LOCA WITH CONTAINMENT BY-PASS)	TBD	
GE-34	HTS HX TUBE RUPTURE - HTS LEAKS WITH CONTAINMENT BY-PASS	1.47E-02	CANDU Operating Experience
GE-35	SINGLE STEAM GENERATOR TUBE RUPTURE (LEAKS WITH CONTAINMENT BY-PASS)	8.04E-04	CANDU Operating Experience
GE-36	ASYMMETRIC FEEDWATER LINE BREAK OUTSIDE RB DOWNSTREAM OF FW REG. VALVES	8.61E-05	CANDU Operating Experience

Grouped Event ID	Description of the Grouped Event	Associated Frequency (event/year)	Source of Frequency Value
GE-37	ASYMMETRIC BREAK INSIDE REACTOR BUILDING BETWEEN STEAM GENERATOR CHECK VALVE AND CONTAINMENT WALL	1.70E-05	CANDU Operating Experience
GE-38	ASYMMETRIC BREAK INSIDE REACTOR BUILDING DOWNSTREAM STEAM GENERATOR CHECK VALVE	5.80E-05	ACR UPDATE OF CANDU DATA
GE-39	SYMMETRIC FEEDWATER LINE BREAK OUTSIDE RB UPSTREAM OF FW REGULATING STATION	2.17E-04	CANDU Operating Experience
GE-40	STEAM GENERATOR DEPRESSURIZATION	1.00E-01	ACR UPDATE OF CANDU DATA
GE-41	MAIN STEAM LINE BREAKS INSIDE REACTOR BUILDING	7.06E-05	CANDU Operating Experience
GE-42	MAIN STEAM LINE BREAKS INSIDE TURBINE BUILDING	2.82E-04	CANDU Operating Experience
GE-43	LOSS OF HEAT SINK TO FUEL WHILE FM UNCLAMPED	TBD	
GE-44	BLOWBACK OF HTS COOLANT THROUGH EMERGENCY COOLANT INJECTION SYSTEM (LOCA NO CONTAINMENT BY-PASS)	TBD	
GE-45	LOSS OF SF STORAGE BAY HEAT SINK	4.18E-02	CANDU Operating Experience
GE-46	LOSS OF COOLING TO FUEL BUNDLES WHILE IN TRANSFER TO SF RECEPTION BAY	2.53E-02	CANDU Operating Experience

Grouped Event ID	Description of the Grouped Event	Associated Frequency (event/year)	Source of Frequency Value
GE-47	MECHANICAL DAMAGE TO FUEL WHILE IN TRANSFER TO SF RECEPTION BAY	8.13E-02	CANDU Operating Experience
GE-48	FM INDUCED LOCA (WHILE MACHINE CLAMPED)	4.70E-02	CANDU Operating Experience
GE-49	FM INDUCED LOCA (WHILE MACHINE UN-CLAMPED)	TBD	
GE-50	LOSS OF HTS COOLANT DUE TO FM WATER SYSTEM FAILURES	TBD	CANDU Operating Experience
GE-51	MULTIPLE STEAM GENERATOR TUBE RUPTURES (SMALL LOCA WITH CONTAINMENT BY-PASS)	TBD	
GE-52	HTS LEAKS WHILE HTS DRAINED TO THE HEADERS	TBD	
GE-53	LARGE CONDENSER COOLING WATER LINE BREAKS - LOSS OF CONDENSER AS A HEAT SINK	8.04E-05	CANDU Operating Experience
GE-54	SMALL CONDENSER COOLING WATER LINE BREAKS - LOSS OF CONDENSER AS A HEAT SINK	8.04E-04	CANDU Operating Experience
GE-55	HTS LEAKS (LOSS OF LTCS INVENTORY WHILE HTS FULL, DEPRESSURIZED)	TBD	
GE-56	TOTAL LOSS OF ONE RCW DIVISION	TBD	
GE-57	TOTAL LOSS OF RCW COOLING - REACTOR OPERATING AT FULL POWER	TBD	
GE-58	TOTAL LOSS OF ONE UNIT CLASS IV POWER SUPPLY	2.17E-01	CANDU Operating Experience
GE-59	LEAKS DURING THROUGH - FLOW DE-FUELLING	TBD	

Grouped Event ID	Description of the Grouped Event	Associated Frequency (event/year)	Source of Frequency Value
GE-60	DUAL FAILURE OF GROUP CONTROLLERS (ON ONE DCS PARTITION)	TBD	
GE-61	DUAL FAILURE OF DCS DATA HIGHWAYS	TBD	
GE-62	LEAKS DURING CIGAR INSPECTIONS	TBD	
GE-63	TOTAL LOSS OF I/A SUPPLY TO PLANT LOADS (REACTOR AT FULL POWER)	TBD	
GE-64	TOTAL LOSS OF HVAC IN THE PLANT	TBD	
GE-65	PARTIAL LOSS OF HVAC IN THE PLANT	TBD	
GE-66	PARTIAL LOSS OF CLASS 3 POWER (REACTOR AT FULL POWER)	TBD	
GE-67	LOSS OF CONDENSER VACUUM	8.83E-02	CANDU Operating Experience
GE-68	LOSS OF CONDENSATE SUPPLY TO THE DEAERATOR	3.24E-02	CANDU Operating Experience
GE-69	LOSS OF OFFSITE POWER SUPPLY	TBD	
GE-70	HTS COOLANT LEAKS THROUGH LTC SYSTEM OUTSIDE REACTOR BUILDING (HTS LEAKS WITH CONTAINMENT BY-PASS)	TBD	
GE-71	PARTIAL LOSS OF CLASS 2 POWER	7.07E-05	CANDU Operating Experience
GE-72	PARTIAL CHANNEL FLOW BLOCKAGE	TBD	
GE-73	PARTIAL LOSS OF CLASS 1 POWER	3.42E-08	CANDU Operating Experience
GE-74	GENERAL TRANSIENT EVENT	TBD	

Grouped Event ID	Description of the Grouped Event	Associated Frequency (event/year)	Source of Frequency Value
GE-75	SEVERE CHANNEL FLOW BLOCKAGE	TBD	
GE-76	LOSS OF SHIELD COOLING SYSTEM HEAT SINK	TBD	
GE-77	CALANDRIA DRAIN LINE BREAKS OUTSIDE SHIELD TANK	2.00E-04	CANDU Operating Experience
GE-78	MODERATOR PIPE LEAKS (OUTSIDE CALANDRIA VESSEL)	TBD	
GE-79	TOTAL LOSS OF INSTRUMENT AIR (REACTOR SHUTDOWN, HTS FULL AND DE-PRESSURIZED)	TBD	
GE-80	LOSS OF ONE SW DIVISION (RCW AND/OR RSW) (REACTOR SHUTDOWN, HTS FULL AND DE-PRESSURIZED)	TBD	
GE-81	TOTAL LOSS OF CLASS IV POWER (REACTOR SHUTDOWN, HTS FULL AND DE-PRESSURIZED)	TBD	
GE-82	TOTAL LOSS OF HVAC IN THE PLANT (REACTOR SHUTDOWN, HTS FULL AND DE-PRESSURIZED)	TBD	
GE-83	TOTAL LOSS OF INSTRUMENT AIR (REACTOR SHUTDOWN, HTS PARTIALLY DRAINED)	TBD	
GE-84	LOSS OF ONE SW DIVISION (RCW AND/OR RSW) (REACTOR SHUTDOWN, HTS PARTIALLY DRAINED)	TBD	
GE-85	TOTAL LOSS OF CLASS IV POWER (REACTOR SHUTDOWN, HTS PARTIALLY DRAINED)	TBD	

Grouped Event ID	Description of the Grouped Event	Associated Frequency (event/year)	Source of Frequency Value
GE-86	TOTAL LOSS OF HVAC IN THE PLANT (REACTOR SHUTDOWN, HTS PARTIALLY DRAINED)	TBD	
GE-87	FAILURE OF FREEZE PLUGS DURING CHANNEL INSPECTION OR REPLACEMENT (FULL AND DEPRESSURIZED)	TBD	

Table 4
C-006 R1 Class 1 Events Compared with ACR Grouped Events

C-006 R1 Class 1 (failure of:)	ACR Grouped / MLD Basic Event Index	Description of ACR Grouped / MLD Basic Event
• primary coolant purification system	A-IE-01	CHEMICAL DAMAGE TO FUEL
• inadvertent ECCS actuation	GE-40	STEAM GENERATOR DE-PRESSURIZATION
• plant state beyond allowable limits	N/A	PLANT TECHNICAL SPECIFICATIONS ENSURE THAT ANY OPERATIONAL LIMITS AND CONDITIONS ARE NOT EXCEEDED.
• operator performs a single manipulation of a procedure when not appropriate	N/A	ERRORS OF COMMISSION ARE OUTSIDE THE SCOPE OF PRESENT WORK (SEE SECTION 2.2 - SCOPE DEFINITION).
• moderator cover gas system	GE-01 / C-IE-02	LOSS OF DEUTERIUM CONCENTRATION CONTROL IN THE COVER GAS
• fuel damage in the irradiated fuel bay	D1-IE-04	MECHANICAL DAMAGE TO SPENT FUEL DURING STORAGE
• radioactive waste (gas, liquid, and solid) management system(s)	E-IE-01 / 02 / 03	RADIOACTIVITY RELEASE FROM SOLID / LIQUID / GASEOUS WASTE MANAGEMENT SYSTEMS
• D_2O management	E-IE-02	RELEASE FROM LIQUID WASTE MANAGEMENT SYSTEM

C-006 R1 Class 1 (failure of:)	ACR Grouped / MLD Basic Event Index	Description of ACR Grouped / MLD Basic Event
• piping, causing a very small loss of reactor primary coolant	GE-02 / 34 /35	HTS LEAKS WITH NO CONTAINMENT BY-PASS / HTS HX TUBE RUPTURE - HTS LEAKS WITH CONTAINMENT BY-PASS / SINGLE STEAM GENERATOR TUBE RUPTURE
• moderator temperature control	GE-03 / B-IE-09	TOTAL LOSS OF MODERATOR HEAT SINK / LOSS OF MODERATOR FLOW (CIRCULATION)
• moderator system (excluding piping failures other than a heat exchanger tube)	GE-03	TOTAL LOSS OF MODERATOR HEAT SINK
• reactor power control	GE-06	BULK CORE POWER EXCURSION (REACTOR AT FULL POWER)
• fuelling machine to reinstall the fuel channel closure plug	GE-09	FM INDUCED LOCA (WHILE MACHINE CLAMPED)
• pressure tube of any fuel channel assembly	GE-15 / H-IE-09	PRESSURE TUBE RUPTURE (CALANDRIA TUBE INTACT)
• boiler inventory control	GE-16	LOSS OF FW SUPPLY DUE TO FAILURES OF ACTIVE MECHANICAL / C&I COMPONENTS
• normal boiler feedwater flow	GE-16	LOSS OF FW FLOW DUE TO FAILURES OF ACTIVE MECHANICAL / C&I COMPONENTS
• steam line isolation valve	GE-19 / GE-20	MAIN STEAM ISOLATION VALVE SPURIOUS CLOSURE
		STEAM GENERATOR PRESSURE HIGH
• primary coolant pressure control	GE-23/24	HEAT TRANSPORT SYSTEM PRESSURE CONTROL FAILS HIGH / LOW

C-006 R1 Class 1 (failure of:)	ACR Grouped / MLD Basic Event Index	Description of ACR Grouped / MLD Basic Event
• primary coolant inventory control	GE-25	LOCA INDUCED BY FAILURES OF P&IC SYSTEM (NO CONTAINMENT BY-PASS)
• primary pressure relief valve(s)	GE-27	HTS DE-PRESSURIZATION TRANSIENT
• residual heat removal system (excluding piping failures other than a heat exchanger tube)	GE-30 /31	LOSS OF LTCS AS HEAT SINK - REACTOR SHUTDOWN AND DRAINED TO THE HEADERS / LOSS OF LTCS COOLING TO FUEL, WHILE REACTOR SHUTDOWN AND DE-PRESSURIZED
• residual heat removal system temperature control	GE-31	LOSS OF LTCS COOLING TO FUEL, WHILE REACTOR SHUTDOWN AND DEPRESSURIZED
• boiler tube	GE-35	SINGLE STEAM GENERATOR TUBE RUPTURE (LEAKS WITH CONTAINMENT BY-PASS)
• seals or valve, causing a loss of reactor secondary coolant	GE-16 / J-IE-01	LOSS OF FW FLOW DUE TO FAILURES OF ACTIVE MECHANICAL / C&I COMPONENTS / LOSS OF MAIN FW SUPPLY DUE TO PUMPS/VALVES FAILURE
• boiler pressure control	GE-40	STEAM GENERATOR DE-PRESSURIZATION
• irradiated fuel bay cooling, purification, or ventilation systems	GE-45	LOSS OF SPENT FUEL STORAGE BAY HEAT SINK
• fuel damage during transfer of the fuel from the reactor core to the irradiated fuel bay	GE-46	LOSS OF COOLING TO FUEL BUNDLES WHILE IN TRANSFER TO SF RECEPTION BAY
• cooling system of fuelling machine	GE-50	LOSS OF HTS COOLANT DUE TO FM WATER SYSTEM FAILURES
• service water flow	GE-57	TOTAL LOSS OF RCW COOLING - REACTOR OPERATING AT FULL POWER

C-006 R1 Class 1 (failure of:)	ACR Grouped / MLD Basic Event Index	Description of ACR Grouped / MLD Basic Event
• seals or valves, causing a loss of service water	GE-57	TOTAL LOSS OF RCW COOLING - REACTOR OPERATING AT FULL POWER
• normal electrical power	GE-58	TOTAL LOSS OF ONE UNIT CLASS IV POWER SUPPLY
• dual computer control	GE-60	DUAL FAILURE OF GROUP CONTROLLERS (ON ONE DCS PARTITION)
• compressed air (instrument or service)	GE-63	TOTAL LOSS OF INSTRUMENT AIR SUPPLY TO PLANT LOADS - REACTOR OPERATING AT FULL POWER
• heating, ventilation, or air conditioning	GE-64	TOTAL LOSS OF HVAC IN THE PLANT
• condenser vacuum	GE-67	LOSS OF CONDENSER VACUUM
• de-aerator inventory control	GE-68	LOSS OF CONDENSATE SUPPLY TO THE DEAERATOR
• seals or valve, causing a loss of moderator water	GE-77 / 78	CALANDRIA DRAIN LINE BREAKS OUTSIDE SHIELD TANK / MODERATOR PIPE LEAKS OUTSIDE CALANDRIA VESSEL
• reactor shield cooling system (excluding piping failures other than a heat exchanger tube)	GE-76	LOSS OF SHIELD COOLING SYSTEM HEAT SINK
• turbine generator load rejection or control	GE-74	GENERAL TRANSIENT EVENT
• seals or valve, causing a loss of reactor primary coolant	GE-70 / GE-02 / H-IE-11	HTS COOLANT LEAKS THROUGH LTC SYSTEM OUTSIDE REACTOR BUILDING (HTS LEAKS WITH CONTAINMENT BY-PASS) / HTS LEAKS WITH NO CONTAINMENT BY-PASS / HEAT TRANSPORT PUMP SEALS FAILURES

C-006 R1 Class 1 (failure of:)	ACR Grouped / MLD Basic Event Index	Description of ACR Grouped / MLD Basic Event
• moderator inventory control	GE-03	TOTAL LOSS OF MODERATOR HEAT SINK
• pressure relief valve in a vacuum containment system	N/A	
• primary system loop interconnect valve or pressurizer connection valve	N/A	
• inadvertent containment dousing	N/A	

Table 5
C-006 R1 Class 2 Events Compared with ACR Grouped Events

C-006 R1 - Class 2 Events (failure of:)	ACR Grouped Event or MLD Basic Event Index	Description of ACR Grouped / MLD Basic Event
• piping, causing a loss of service water	GE-56 / Y-IE-05	TOTAL LOSS OF ONE RCW DIVISION / LOSS OF RCW INVENTORY DUE TO PIPE BREAKS
• piping, causing a loss of reactor secondary coolant	GE-36/37/38/39	FEEDWATER LINE BREAKS
• piping, causing a small loss of reactor primary coolant	GE-14	SMALL LOSS OF COOLANT ACCIDENT (NO CONTAINMENT BY-PASS)
• end fitting of any fuel channel assembly	GE-14 / H-IE-15	END FITTING BREAKS (DISCHARGE OUTSIDE ANNULUS GAS)
• residual heat removal system isolation valves	GE-33 / R-IE-01	SMALL LOCA WITH CONTAINMENT BY-PASS / INTERFACING LOCA THROUGH LTC SYSTEM
• ECCS isolation valves	GE-33	SMALL LOCA WITH CONTAINMENT BY-PASS / INTERFACING LOCA THROUGH LTC SYSTEM
• boiler primary head divider	GE-51	MULTIPLE STEAM GENERATOR TUBE RUPTURES (SMALL LOCA WITH CONTAINMENT BY-PASS)
• reactor primary coolant pump shaft	GE-21 / K-IE-05	PARTIAL LOSS OF HTS PUMPED FLOW / ONE HT PUMP BEARING OR SHAFT SEIZURE
• piping or calandria tube, causing a loss of moderator	GE-03 / GE-78	MODERATOR PIPE BREAKS / MODERATOR PIPE LEAKS (OUTSIDE CALANDRIA VESSEL)
• piping, causing a loss of reactor shield coolant	GE-76 / W-IE-01	LOSS OF SHIELD COOLING SYSTEM WATER INVENTORY
• fuelling machine pressure boundary	GE-09	FM INDUCED LOCA (WHILE MACHINE CLAMPED)
• flow blockage in a fuel channel	GE-72	PARTIAL CHANNEL FLOW BLOCKAGE
• seizure of a reactor primary coolant pump	GE-21 / K-IE-05	PARTIAL LOSS OF HTS PUMPED FLOW / ONE HT PUMP BEARING OR SHAFT SEIZURE

Table 6
C-006 R1 Class 3 Events Compared with ACR Grouped Events

C-006 R1 - Class 3 Events (failure of:)	ACR Grouped / MLD Basic Event Index	Description of ACR Grouped / MLD Basic Event
• piping, causing a large loss of reactor primary coolant	GE-11	LARGE LOSS OF COOLANT ACCIDENT
• large number of boiler tubes	GE-51	MULTIPLE STEAM GENERATOR TUBE RUPTURES (SMALL LOCA WITH CONTAINMENT BY-PASS)
• end fittings of many reactor-fuel-channel assemblies	GE-48 / D3-IE-01 - 04	FM INDUCED LOCA (WHILE MACHINE CLAMPED)
• end fitting of any fuel-channel assembly with consequential failure of its lattice-tube	Q-IE-10	END FITTING AND LATTICE TUBE FAILURES / LEAKAGES INSIDE THE ANNULUS GAS (HT LEAKS NO CONTAINMENT BY-PASS)
• components causing backflow to ECCS	H-IE-13	BLOWBACK OF HTS COOLANT THROUGH EMERGENCY COOLANT INJECTION SYSTEM (LOCA NO CONTAINMENT BY-PASS)
	GE-33 / R-IE-01	SMALL LOSS OF COOLANT ACCIDENT WITH CONTAINMENT BY-PASS / INTERFACING LOCA THROUGH LTC SYSTEM

Appendix A

Guide to Master Logic Diagrams

In order to ensure easier reading of the Logic Diagrams this introductory paragraph presents the sources of radioactive materials or displacement mechanisms and the letters that are indexing them in the Figures, as follows:

Item # – Description	Figure #
1. – Radioactive Materials Displaced from their normal location	Figure A-1
2. – Heat Transport System	Figure A-2
3. – Moderator System	Figures A-1 / A-3
4. – Moderator Cover Gas System	Figure A-4
5. – Spent Fuel Handling System	Figures A-5 / A-13
6. – Radioactive Waste Management System	Figures A-1 / A-4
7. – New Fuel Storage System	Figures A-1 / A-4
8. – Loss of Reactivity Control	Figures A-2 / A-4
9. – Loss of HTS Coolant Inventory	Figures A-2 / A-6
10. – Loss of HTS Heat Sinks	Figures A-2 / A-7
11. – Loss of Coolant Circulation in the HT System	Figure A-8
12. – Failures of Heat Transport Pressure and Inventory Control System	Figure A-8
13. – Power Cooling Mismatch in the Heat Transport System while Reactor Shutdown	Figure A-9
14. – Failures Associated with Reactor Power Manoeuvres	Figure A-11
15. – Fuel Damaged by Fuelling Machine while Refuelling	Figures A-2 / A-5
16. – Pressure and Inventory Control System Induced LOCA Events	Figures A-6 /A-8
17. – Heat Transport System Leaks	Figures A-6 / A-12
18. – LOCA with Containment By-Pass	Figures A-6 / A-12
19. – Loss of Condenser As A Heat Sink	Figures A-7 / A-10
20. – Condensate System Failures	Figures A-7 / A-10

21. – Loss of Feedwater Inventory	Figures A-7 / A-11
22. – Steam Generator De-Pressurization	Figures A-7 / A-10
23. – LOCA Induced by Overstressing the Calandria Tubesheets	Figures A-6 / A-12
24. – Events Causing Failure of Support Systems	Figure A-14
25. – Failures of Plant HVAC System	Figures A-14 / A-11
26. – Failures of Support Systems while Reactor Shutdown	Figure A-15

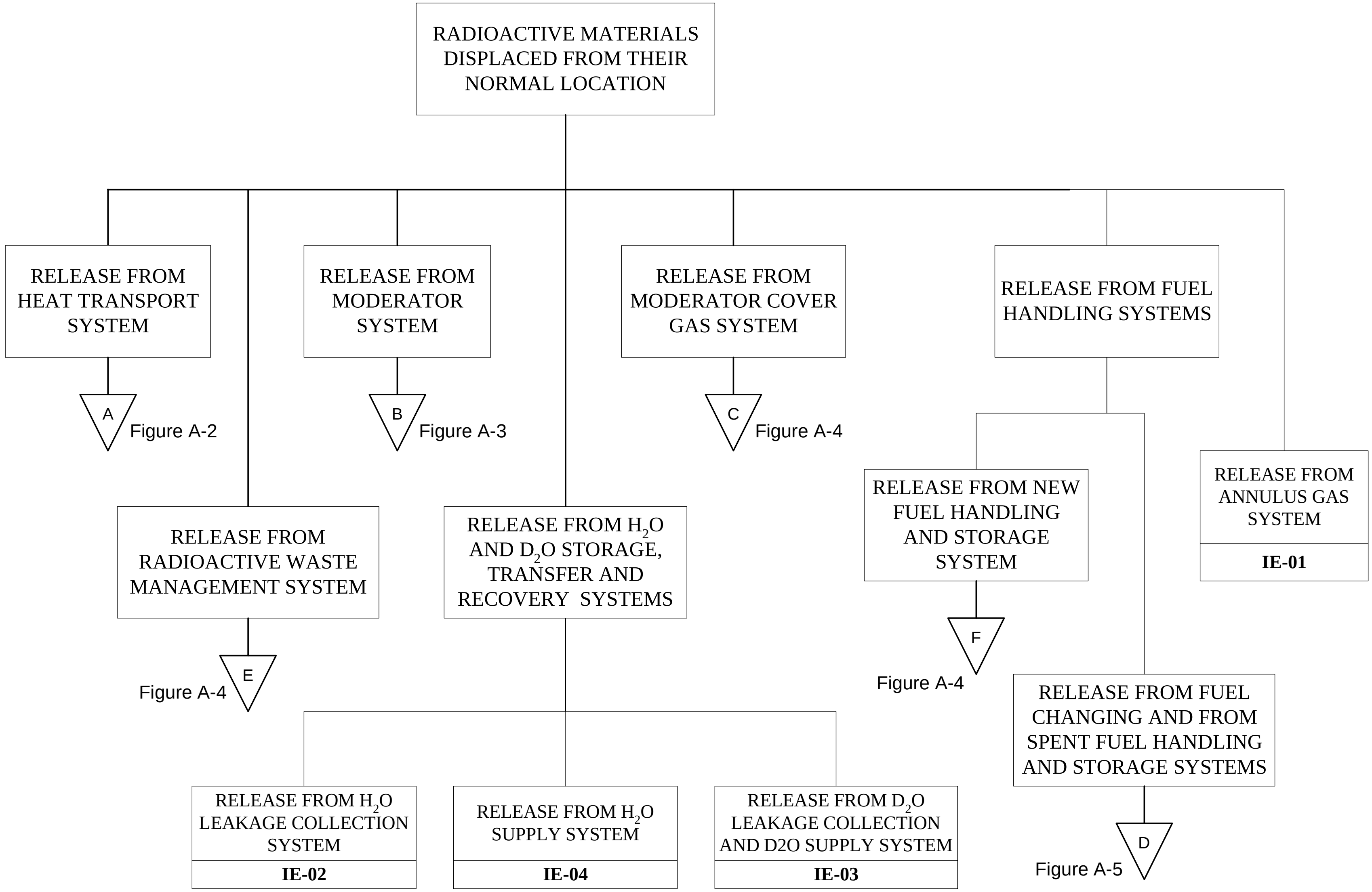


Figure A-1 Top Level Master Logic Diagram

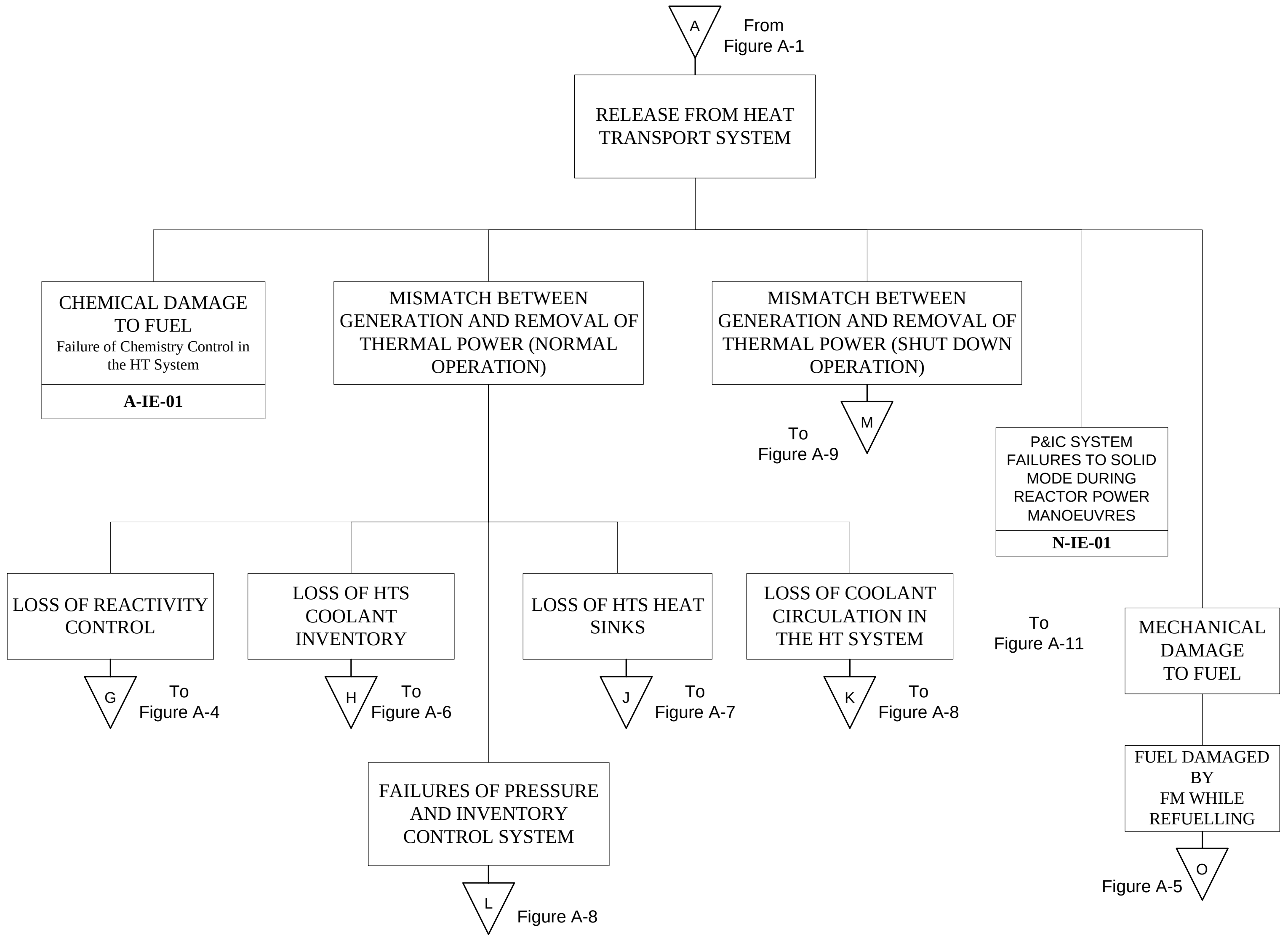


Figure A-2 Logic Diagram “A”/“N”: Release from HT System

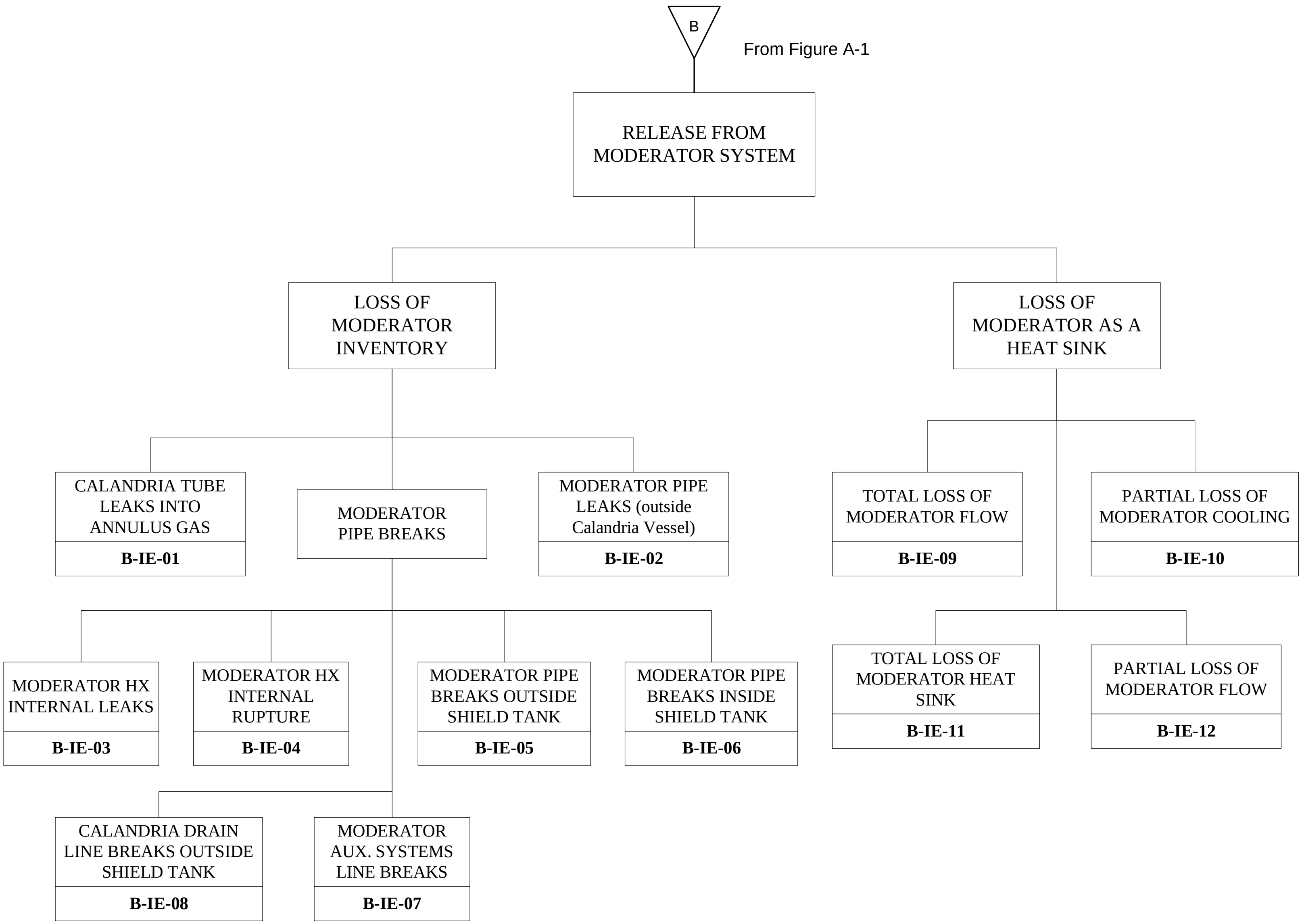


Figure A-3 Logic Diagram "B": Release from Moderator System

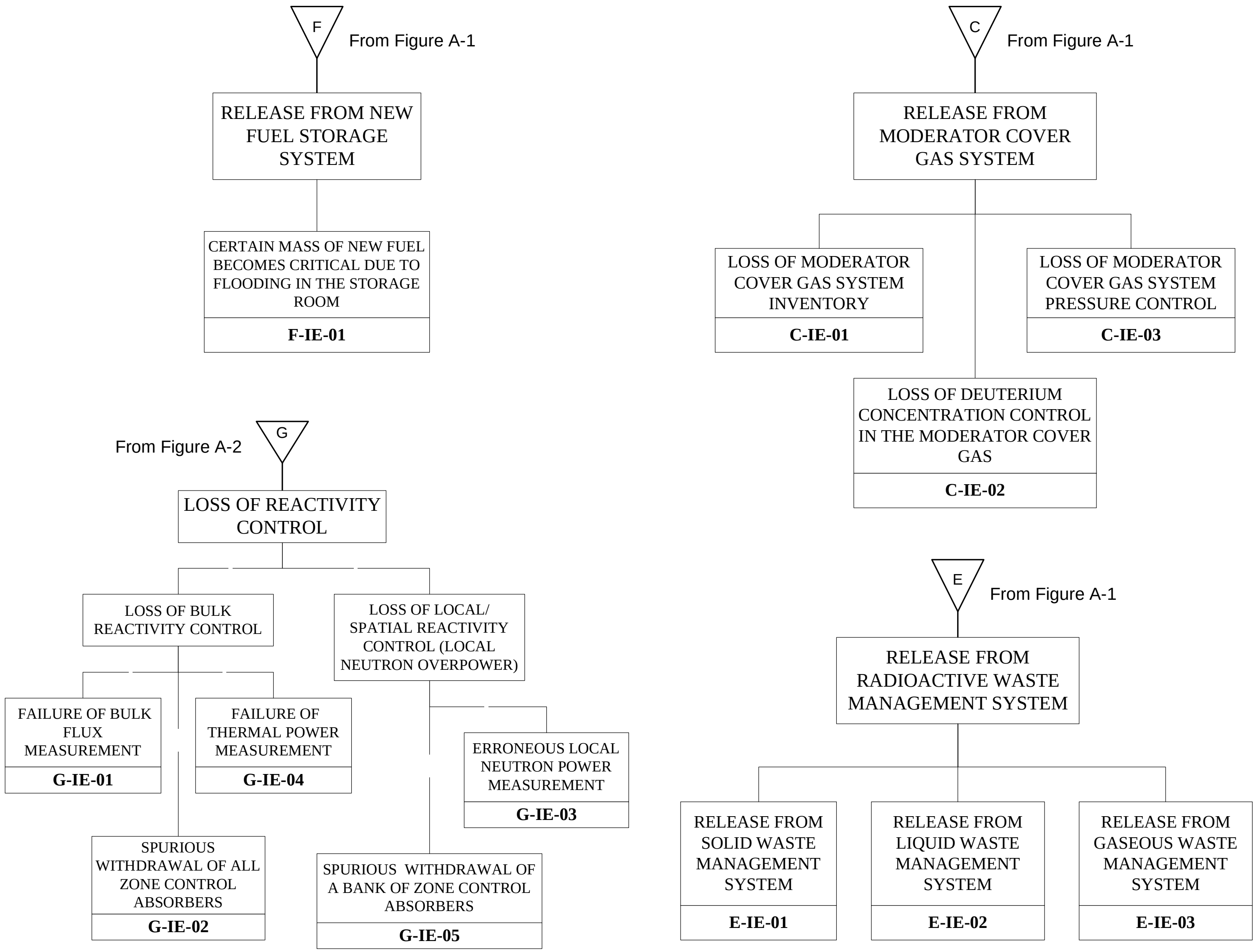


Figure A-4 Logic Diagram “C”/“E”/“F”/“G”

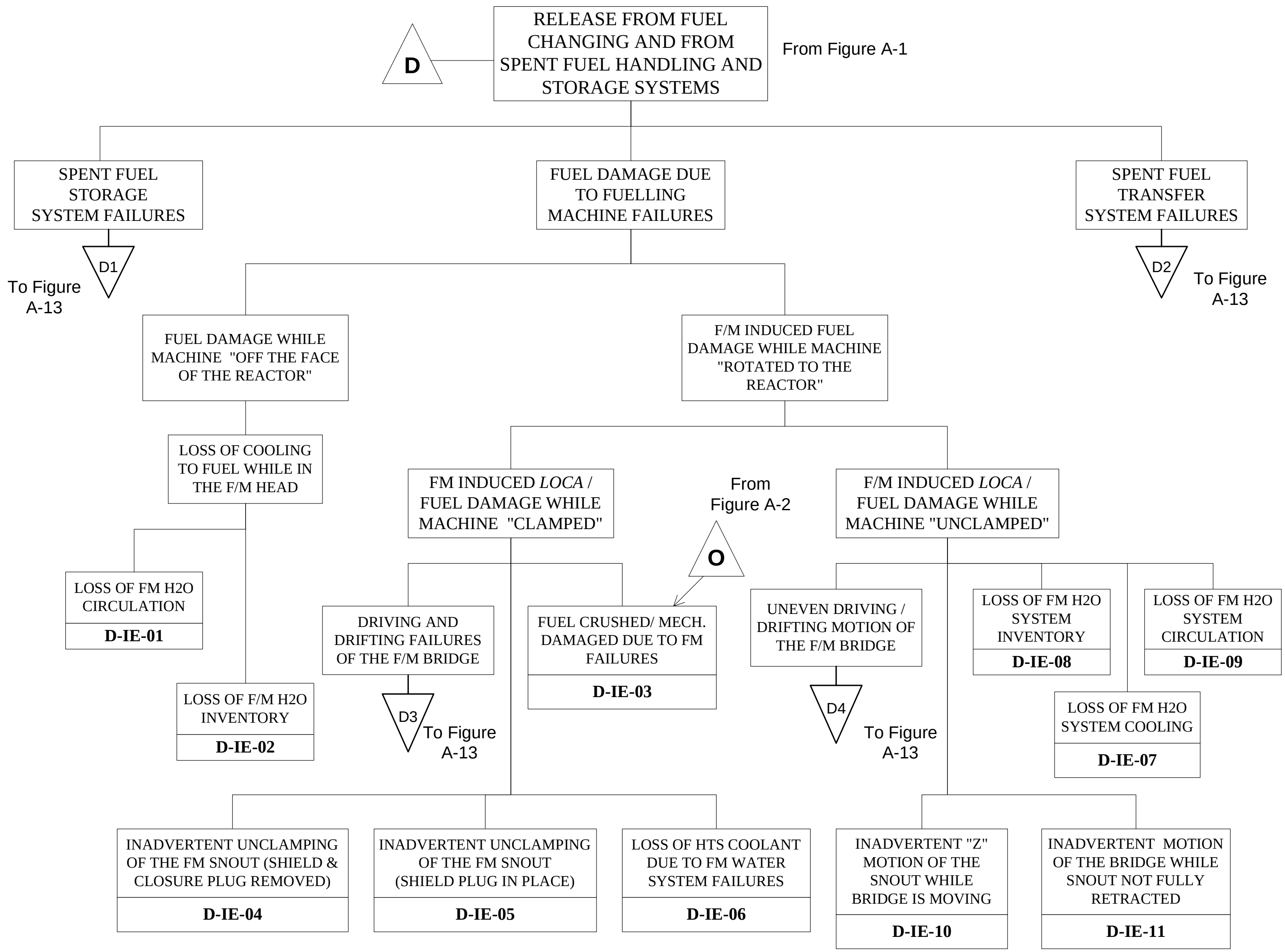


Figure A-5 Logic Diagram “D”: Release from Fuel Handling System

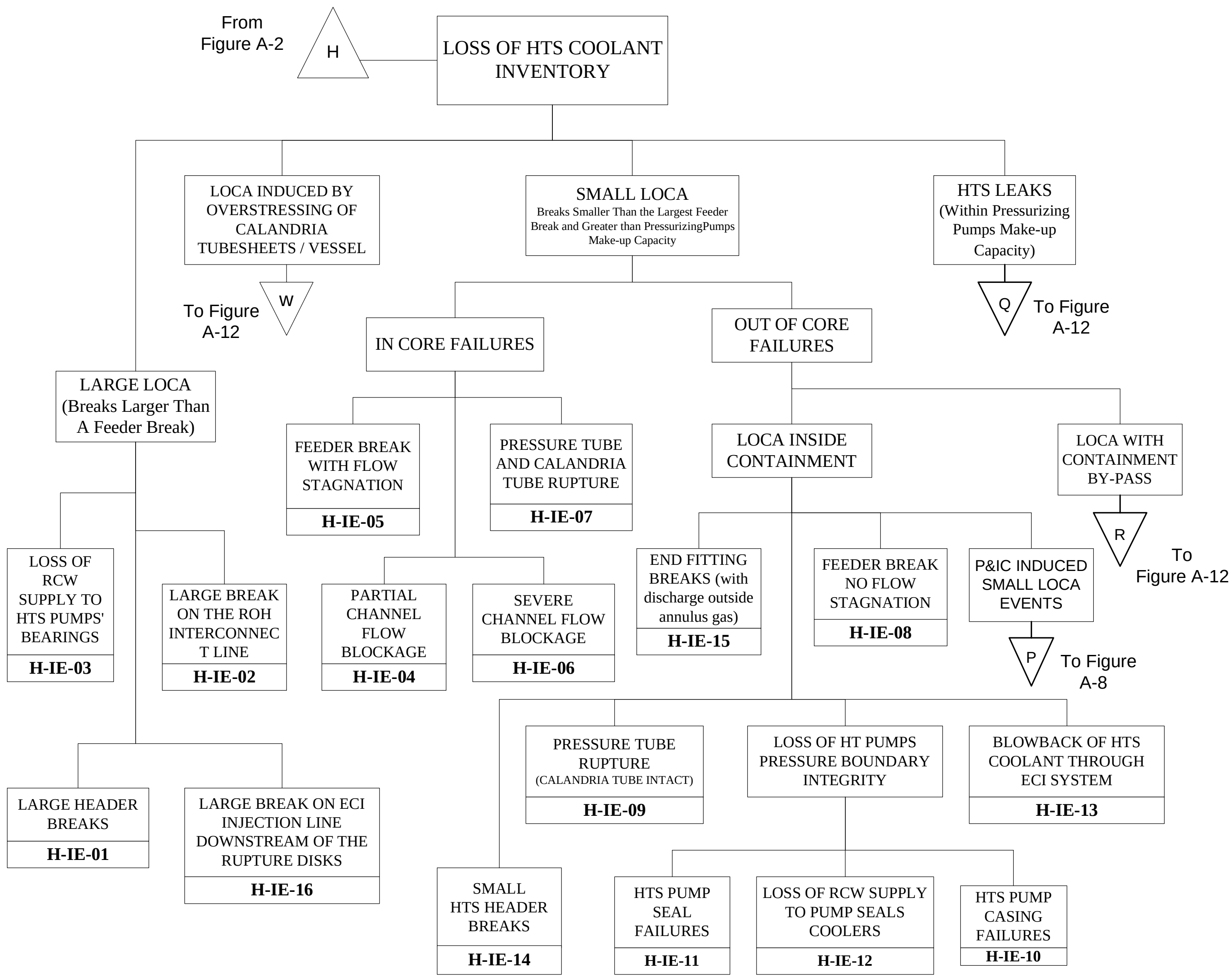


Figure A-6 Logic Diagram “H”: Loss of HTS Coolant Inventory

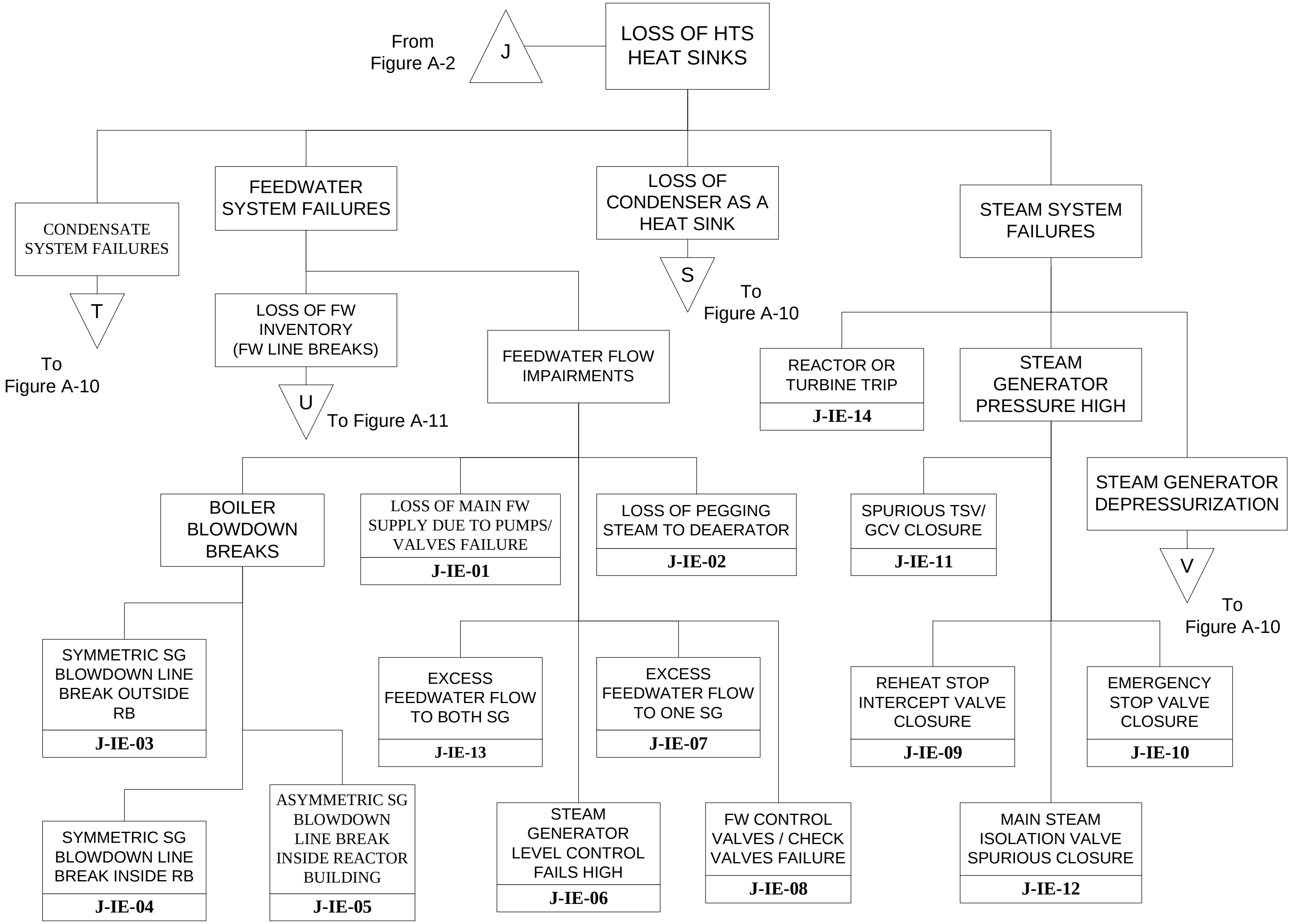


Figure A-7 Logic Diagram "J": Loss of HTS Heat Sink

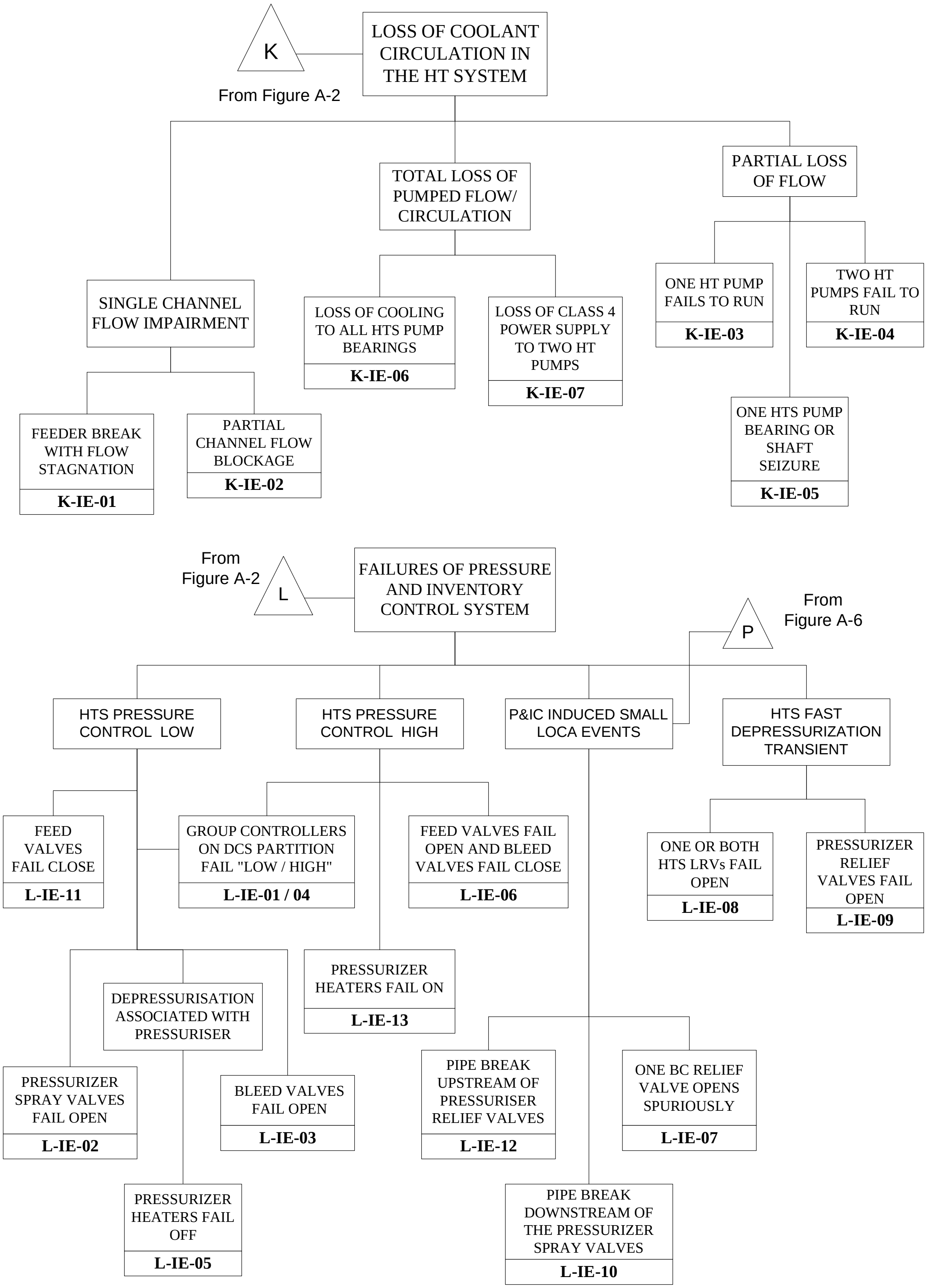


Figure A-8 Logic Diagram “K”/“L”/“P”

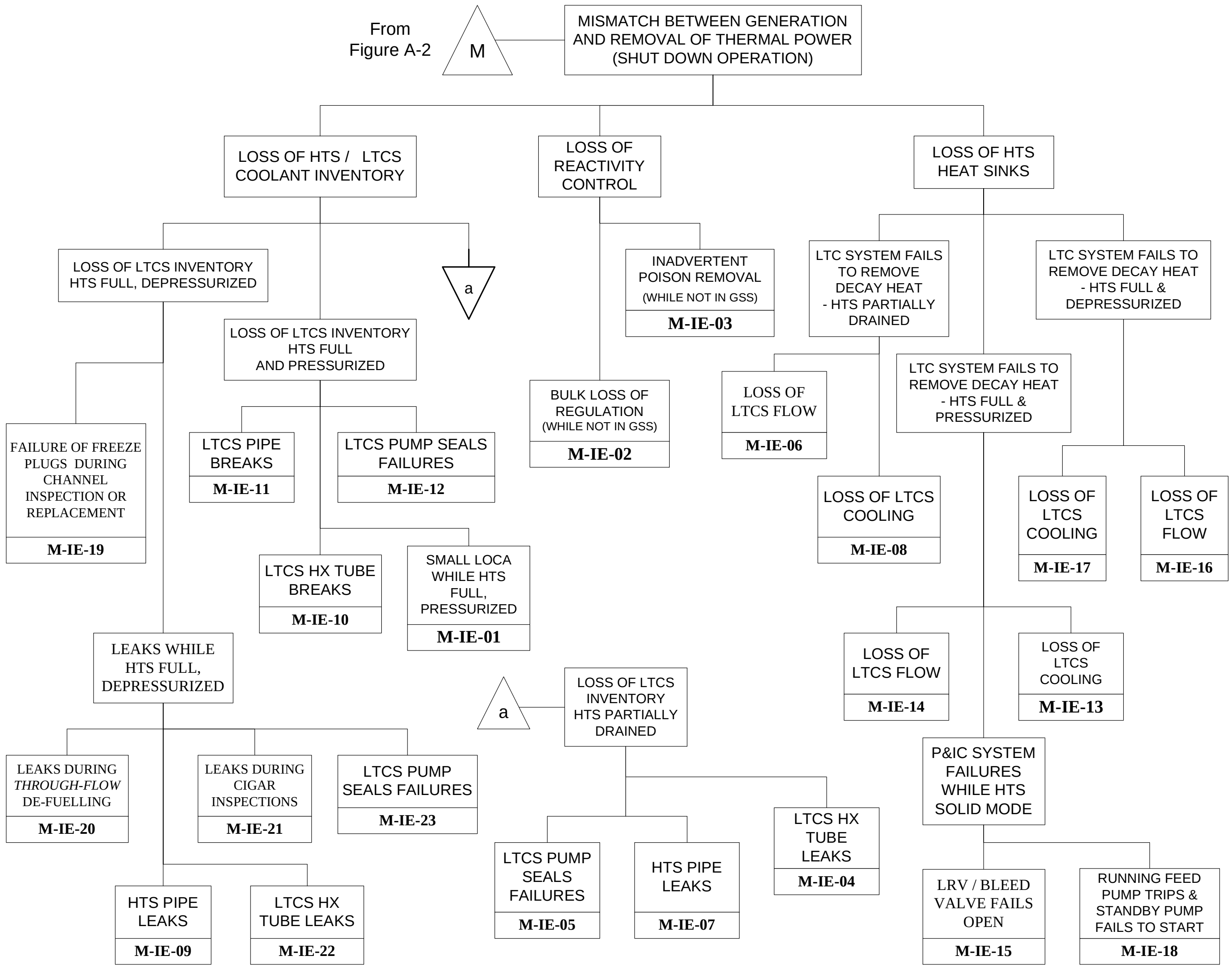


Figure A-9 Logic Diagram “M”: Power / Cooling Mismatch during Plant Shutdown

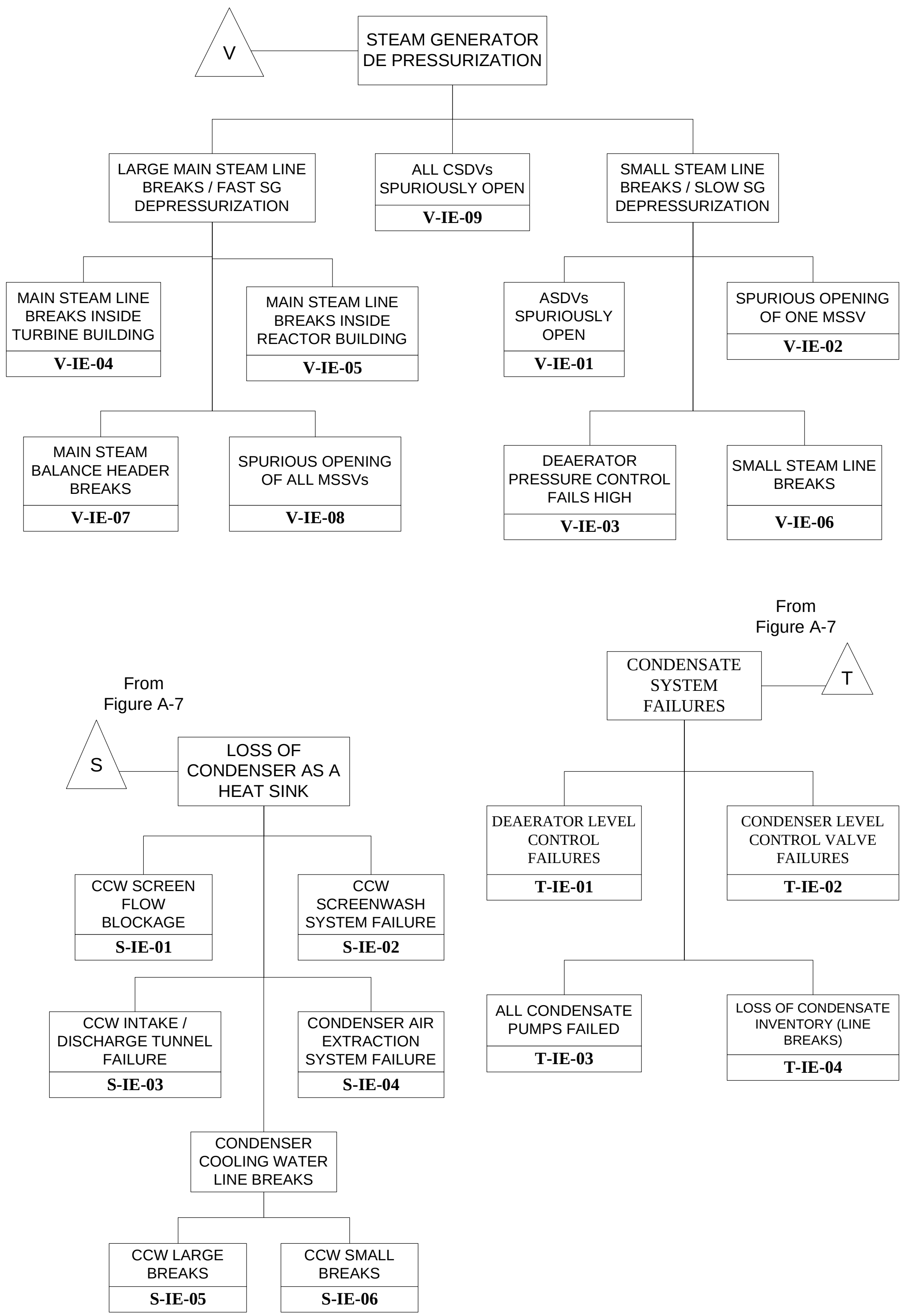
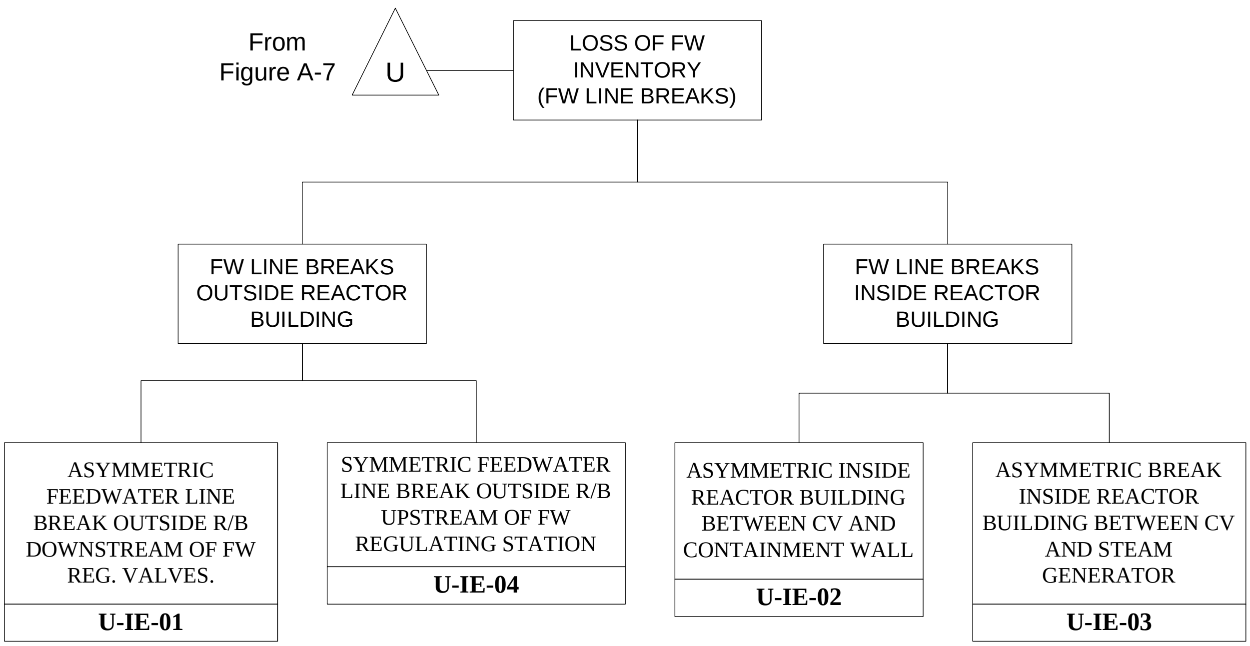


Figure A-10 Logic Diagram “S”/“T”/“V”



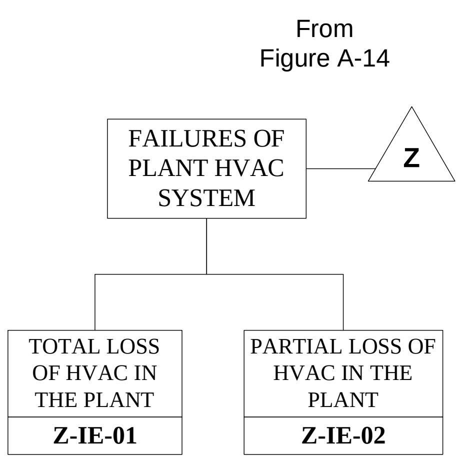


Figure A-11 Logic Diagram “U”/“Z”

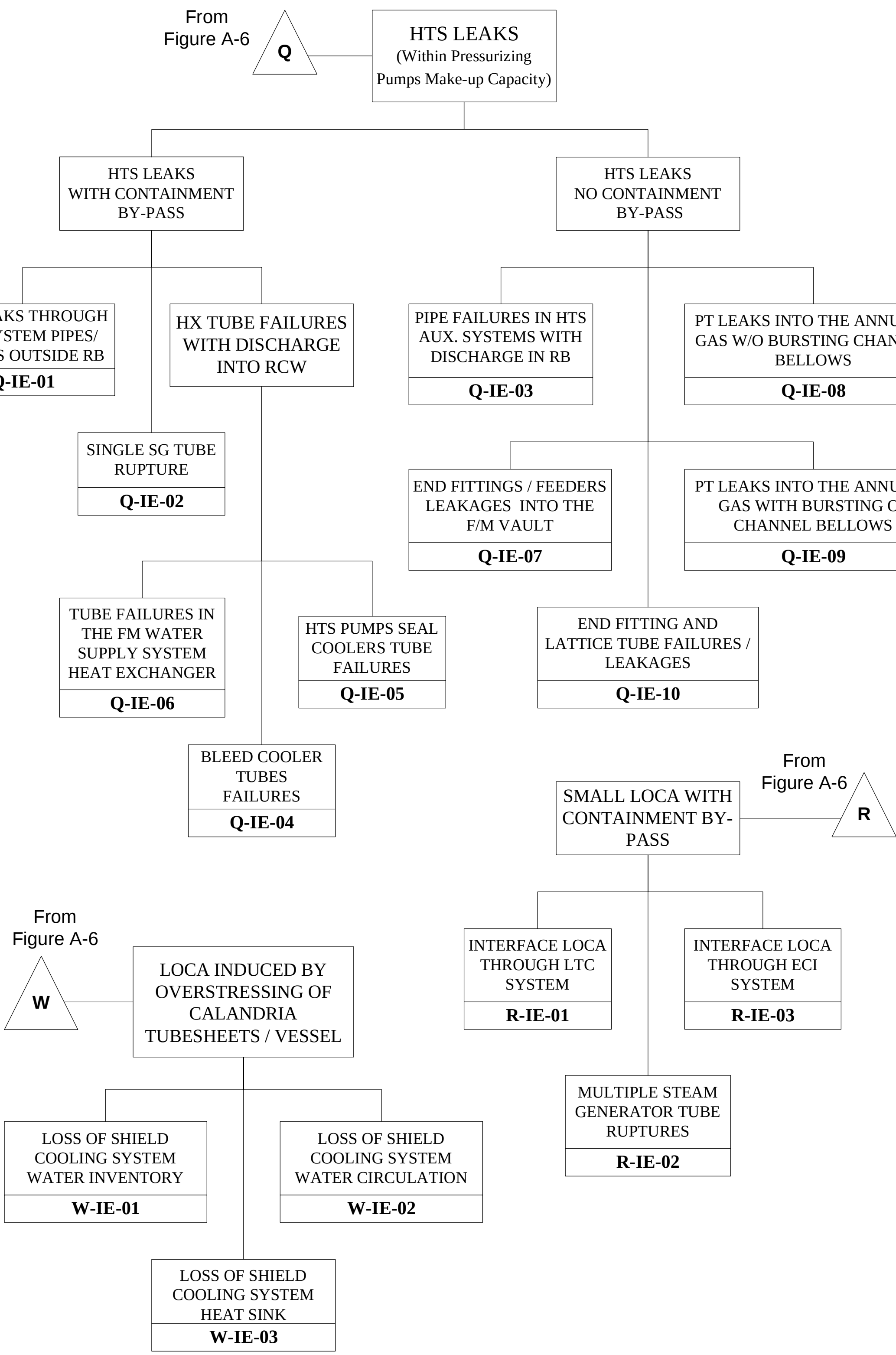


Figure A-12 Logic Diagram “Q”/“R”/“W”

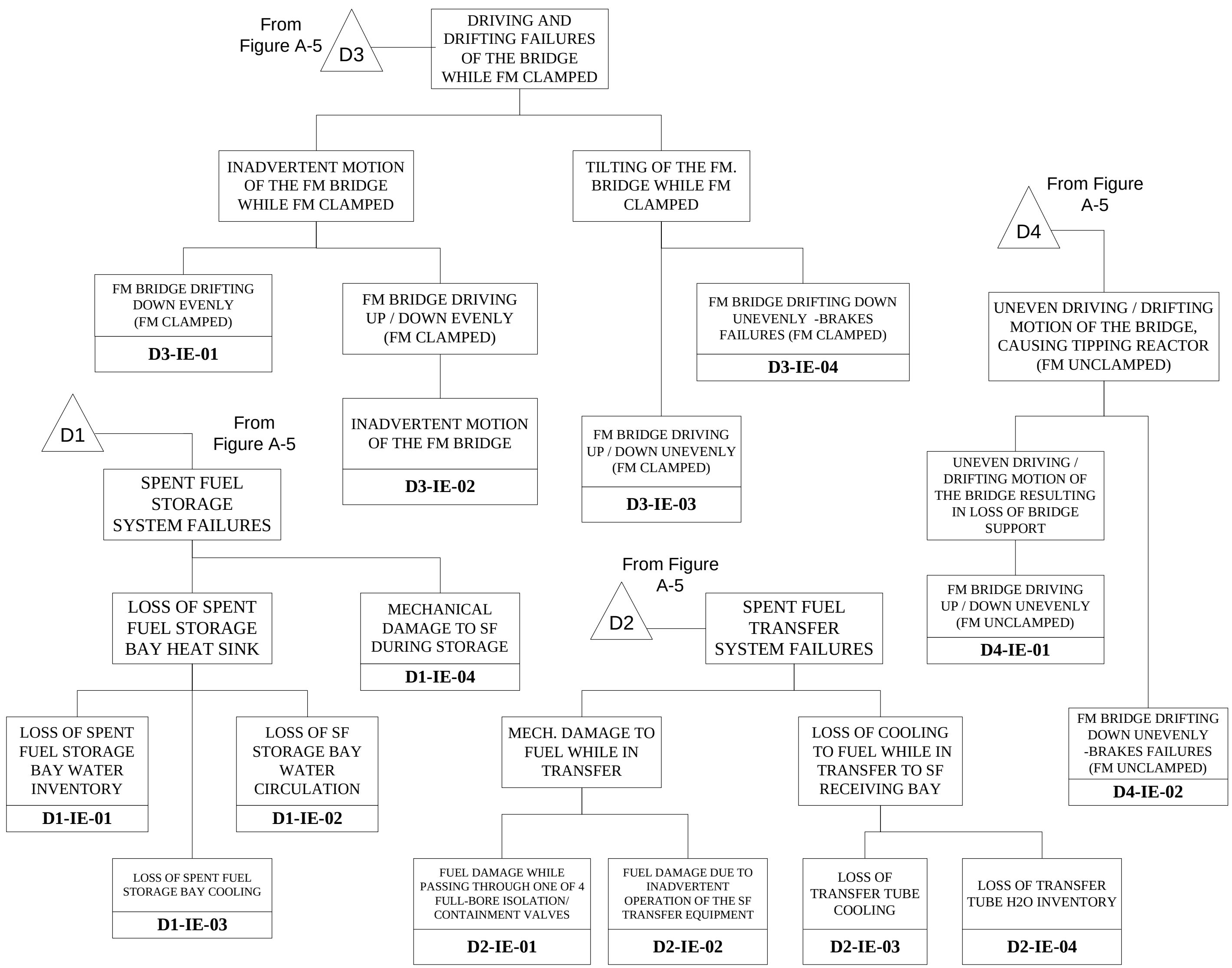


Figure A-13 Logic Diagram “D1”/“D2”/“D3”/“D4”: Fuel Handling System Failures

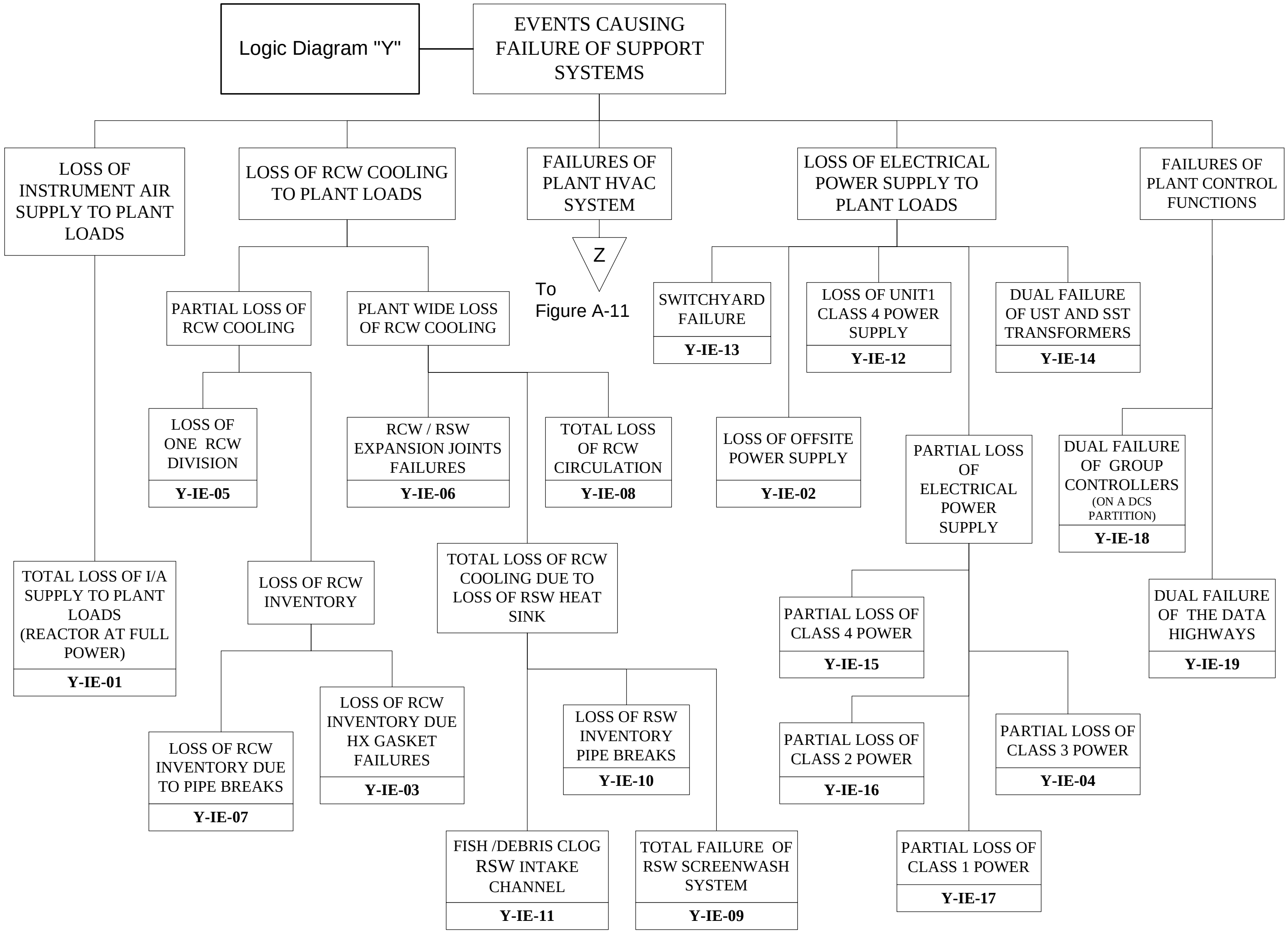


Figure A-14 Logic Diagram: Events Causing Failure of Support Systems

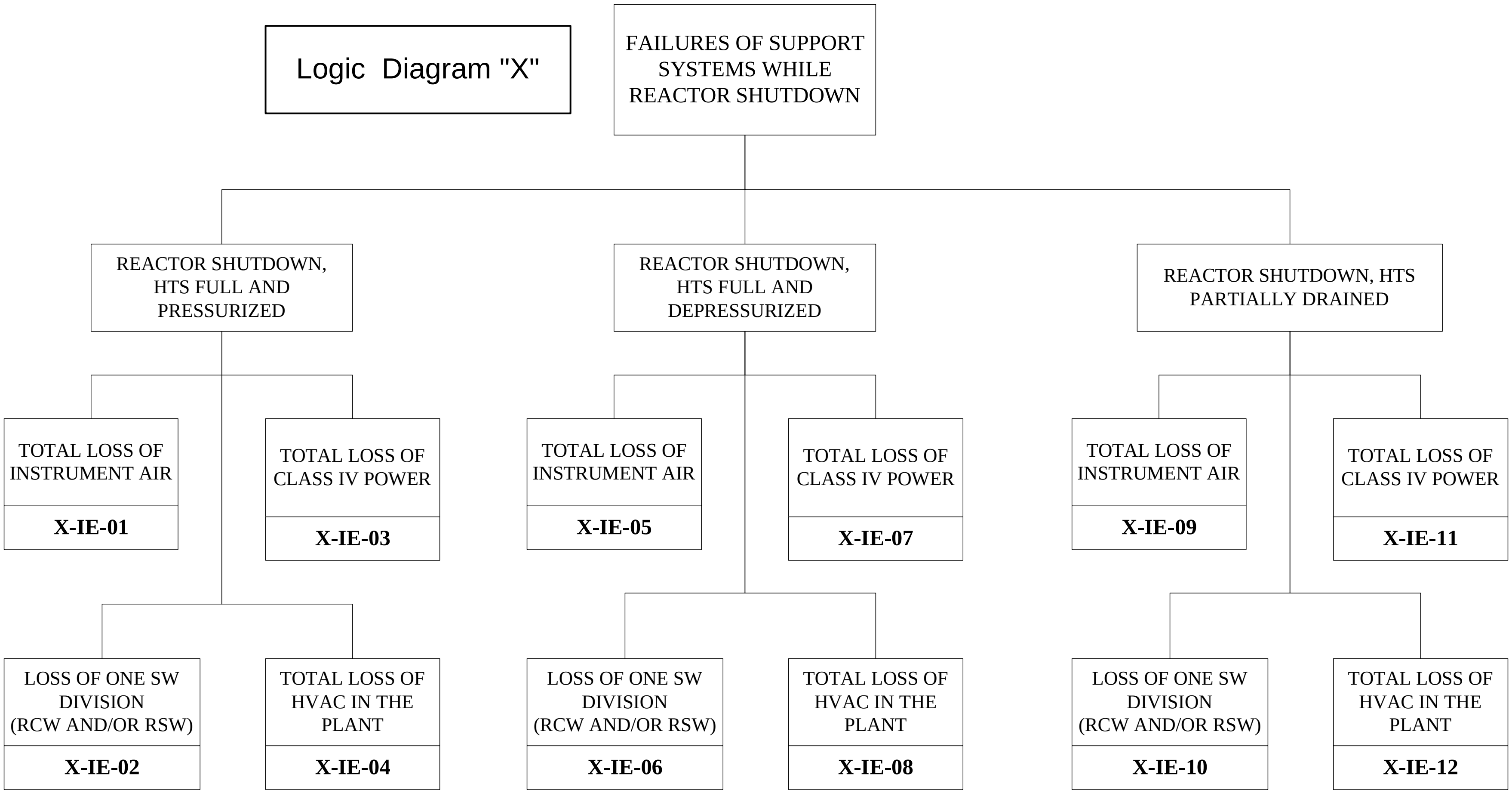


Figure A-15 Logic Diagram: Failures of Support Systems while Reactor Shutdown