Letter Report for the Internal Event Hazard Assessment for NGNP

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ABSTRACT

This report presents an overview of the literature search and NGNP design review for the identification of potential initiating events. The goal of this review is to compile an initial list of potential initiating events that would be eventually analyzed in a Probabilistic Risk Assessment (PRA). Three reactor designs were focused on for this report: 1) The AREVA Prismatic Modular High Temperature Gas Reactor, 2) the General Atomics Prismatic Modular High Temperature Gas Reactor and 3) the Westinghouse Pebble Bed Reactor. The report also presents a brief description of each of these designs.

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ACRONYMS

ATWS	anticipated transient without scram
BWR	boiling water reactor
FHSS	Fuel Handling and Storage System
FSAR	final safety analysis report
HICS	Helium Inventory Control System
HTGR	high-temperature gas reactor
HVAC	heating, ventilation, air conditioning
IE	initiating event
INEEL	Idaho National Engineering and Environmental Laboratory (became the Idaho National Laboratory)
INL	Idaho National Laboratory (formerly Idaho National Engineering and Environmental Laboratory)
LOCA	loss of coolant accident
LOFC	loss of forced circulation
LWR	light water reactor
MHTGR	modular high-temperature gas reactor
MLD	master logic diagram
NGNP	Next Generation Nuclear Plant
NRC	U.S. Nuclear Regulatory Commission
PRA	probabilistic risk assessment
PRS	Pressure Relief System
PSA	probabilistic safety assessment
PWR	pressurized water reactor
RCCS	Reactor Cavity Cooling System
RCSS	Reactivity Control and Shutdown System
RG	Regulation Guide
RUCS	Reactor Unit Conditioning System
SAPHIRE	Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (probabilistic risk and reliability analysis software)
SRP	Standard Review Plan
SSC	structures, systems, and components
TRISO	tristructural isotropic (fuel)

Internal Initiating Event Hazard Assessment for NGNP

1. Literature Search and Review

1.1 Background

A hazard is generally an event that results in some negative or "bad" outcome. Hazard Analysis or Hazard Assessments are the process of identifying energy sources and hazards associated with a task, process or system. The process of evaluating these hazards using a quantification approach is known as a probabilistic risk assessment (PRA). In general application, a PRA answers three questions:[1]

1. What can happen?
2. How likely is it to happen?
3. What are the consequences of it happening?

The starting point of a PRA is the identification of potential initiating event (IEs). An initiating event is the "event" that starts the accident that could result in an undesired consequence or "negative outcome." The current fleet of nuclear reactors in operation today has a very well established and accepted list of initiating events. The challenge involved with identifying initiating event for NGNP is in addition to determining if the current initiators for today's reactors are applicable to NGNP; any new potential initiators will need to be identified.

The primary objective of this report is to identify possible potential initiating events for the Next Generation Nuclear Plant (NGNP) design. Our secondary objective of this report is to generate a list of possible safety systems for the NGNP designs that could possibly mitigate the effects of the initiating events. Details for the NGNP design have not be finalized, so in order to proceed with initiating event identification, a number of current non-Light Water Reactor PRAs were reviewed along with a more general literature review for potential initiating events.

1.2 System Descriptions

The three primary NGNP reactor designs focused on for this report were the Westinghouse Pebble Bed, Areva Prismatic, and the General Atomics Prismatic designs. Though other designs were reviewed, these three designs are the ones currently under consideration by the NGNP program. The characteristics of these designs have several key differences when compared to the current fleet of reactors in the United States. These features include:

- Downsized into modules in order to keep heat generation down, inventory of radionuclides down, and the ability to mass produce the modules as a means for keeping costs down.
- Use of fewer welds and less linear footage of piping to reduce corrosion risks.
- Use of natural heat convection and less need for pumping hot fluid through heat exchangers.
- Implementing passive safety systems that do not rely on pumps or valves. Emergency cooling is accomplished by gravity or natural convection.

1.2.1 Westinghouse Pebble Bed

Pebble-bed reactors consist of thousands of small graphite spheres that contain tri-structural isotopic (TRISO) coated uranium dioxide (UO_2) particles, encased in 6 cm-diameter graphite fuel element. Power level is generally 500 MWt with helium gas used as the working fluid. Heat is transferred to the coolant gas (helium), and converted into electrical energy by means of a gas turbo-generator.[2]

During full power operations, helium enters the reactor at a temperature of approximately 500°C (930°F) and 1000 psia and moves downward through the fuel spheres in the reactor vessel. Heat is transferred to the spheres heated by the nuclear reaction. Helium leaves the reactor vessel at approximately 900°C (1600°F). The helium then is passed through a Turbine-Generator and drives compressors. It then passes through a power turbine that drives a generator. During reactor shutdown, residual heat is removed by active and/or passive cooling systems.[3]

Support and Safety Systems - The following is a summary list and brief description of the major support and safety system that are critical for the operation of the Pebble Bed Reactor.[3,4]

- Reactor Cavity Cooling System (RCCS). The RCCS consists of three independent sub-systems each having a closed loop pump-driven cooling system, with a water heat exchanger. Heat is transferred from these heat exchangers to the ultimate heat sink (large body of water or the atmosphere). In the event of the loss of pumps or heat exchangers, draught cooling towers on the roof of the module, automatically take over the cooling functions. If the towers are lost, reactor heat is absorbed by heating up and then boiling off chamber water. For even further backup, chamber water can be replenished by the Fire Protection System (FPS).

- Reactor Unit Conditioning System (RUCS). Maintains the RPV at a desired temperature and can be used to recover the core from very hot conditions (if the Power Conversion Unit trips).

- Helium Inventory Control System (HICS). Consists of three subsystems: 1) Inventory Control System (ICS); 2) Helium Purification System (HPS); and 3) Helium Make-up System. All three systems work in conjunction to control helium quality and inventory.

- Fuel Handling and Storage System (FHSS). The FHSS circulates the spherical fuel elements through the reactor core while at power. Movement of the fuel elements is achieved by gravitational flow and pneumatic transport flow.

- Reactivity Control and Shutdown System (RCSS). The RCSS consists of two independent and diverse systems: 1) The Reactivity Control System (RCS) that controls the Control Rod Drive Mechanism and rod position. 2) Reserve Shutdown System (RSS) operates very differently that the control rod system of the RCS. The RSS controls reactivity by storing, adding and removing absorber spheres.

- Pressure Relief System (PRS). This system protects the Primary Pressure Boundary by releasing helium through the Blowdown System.

- Auxiliary Systems. Auxiliary systems include the following systems:

    - Active Cooling System: This system removes waste heat from various auxiliary subsystems such as HVAC, Fuel Handling and Storage System, Helium Inventory Control System, Reserve Shutdown System, Core Conditioning System, and Reactor Pressure Vessel Conditioning System.

    - Heating Ventilating and Air Conditioning System (HVAC): This system provides fresh air and maintains specified environmental parameters, temperature, pressure and humidity

- Airlock System: The Airlock System is designed to act as a barrier between zones of different radiation levels, a barrier between zones of different pressures, and a protective barrier to contain over-pressure from a helium leak depressurization.

- Pressure Relief System: This system relieves overpressure in the building caused by a depressurization event by releasing helium to the atmosphere.

- Fire Protection System: This system provides detection, alerts and fire suppression functions to the plant.

1.2.2 AREVA Prismatic HTGR

The Areva High Temperature Gas Reactor contains Tri-isotopic coated uranium dioxide (UO_2) particles packed into compacts within hexagonal graphite prisms. The compacts are 12.5 mm in diameter and contain evenly dispersed spherical particles in the graphite matrix. Power level is generally 565 MWt with the secondary fluid being Helium-Nitrogen gas. Heat is transferred to the coolant gas, and converted into electrical energy by means of commercial turbine-generator equipment. Detailed system descriptions for the AREVA HTGR were limited in the literature search.

During full power operations, helium-Nitrogen gas enters the reactor at a temperature of approximately 500°C and 8 MPa and moves downward through the fuel prisms in the reactor vessel. Heat is transferred to the prisms heated by the nuclear reaction. The primary coolant gas leaves the reactor vessel at approximately 950°C. The helium-nitrogen gas then passes through a power turbine that drives a generator. During reactor shutdown, residual heat is removed by active and/or passive cooling systems.

Shutdown cooling is used for decay heat removal if the primary loop trips, during the refueling process, or during accident conditions. The Helium Clean-up System is a critical SSC in Areva design and is required to remove gaseous activities and chemical impurities from the primary coolant.[5]

1.2.3 General Atomics Prismatic HTGR

The General Atomics High Temperature Gas Reactor Contains Tri-isotropic coated uranium dioxide (UO_2) particles packed into compacts within hexagonal graphite fuel blocks (360 mm width; 794 mm length). With the General Atomics design, alternate fuels are possible (example: plutonium-based fuel, minor-actinide fuel. Power levels are generally 550/600 MWt with the secondary fluid being Helium. Heat is transferred to the coolant gas, and converted into electrical energy by means of direct gas turbine. Detailed system descriptions for the AREVA HTGR were limited in the literature search.

The Reactor Cavity Cooling System (RCCS) uses convection air-based cooling that removes heat from the reactor cavity during accident conditions when shutdown systems or PCS are not available. The RCCS is passive system that requires no electrical or mechanical pumps, valves or other active components. Shutdown cooling is a water-based system used for decay heat removal if the main heat removal system is offline or it can be used during accident conditions. This is an active system requiring pumps and other mechanical equipment.

2. Initiator Identification

2.1 Introduction

An initiating event in the nuclear industry is generally thought of as any potential occurrence that could disrupt plant operations. Or in general terms—initiating events are the events that force the activation of mitigating actions to prevent some "bad" outcome. Initiating events are quantified in terms of their frequency of occurrence (i.e. number of events per some time frame).[6]

Initiating events can occur while the reactor is at full power, low power or during shutdown. This report focus primarily on NGNP full power operations and most probabilistic risk assessments (PRAs) for the fleet of current reactors in use typically examine full power operations only. However, for a complete comprehensive safety evaluation, all modes of reactor operation should be considered.

There are two categories of initiating events: 1) Internal and 2) External. Internal initiating events involve either a breach of the primary coolant boundary (commonly referred to as a loss of coolant accident or LOCA) or a Transient, which does not involve a primary breach, but requires a reactor shutdown. External initiating events usually originate outside the plant systems and because of their dependent nature of potentially failing multiple systems, are treated separately. Examples of external events are fire, flood, and seismic.[7]

The initiating event identification process consists of identifying a comprehensive list of potential initiators that could upset normal plant operations, grouping the initiating events into categories based on their impact on plant accident response systems and then quantifying the initiating event into category frequencies. A screening process can be used for those events having very low frequencies of occurrence, and can be categorized as "incredible or extremely unlikely events."

Grouping of the NGNP initiating events was based on challenges to fundamental safety functions and resulted in the following groups:[8]

- Challenges to heat removal
- Challenges to reactivity control
- Challenges to confinement of radioactivity
- Challenges to control of chemical attack

Events may also be grouped according to dominant phenomena in the possible event sequence. Examples are:

- Primary system breach
- Loss of heat sink
- Air ingress
- Water ingress
- Reactivity transients
- Depressurized loss of forced cooling
- Pressurized loss of forced cooling
- Turbine or generator trip
- Station blackout
- Primary system pressure boundary breach (challenge to confinement of radioactivity)
- Primary system breach in the interface with a secondary cooling systems (i.e., heat exchanger tube breaks)

2.2 Identification Methods

Initiating events are a fundamental building block and indentifying the initiating events is one of the first steps in a probabilistic safety assessment. A combination of techniques was used for this analysis to generate a list of possible, credible initiating events. Some techniques are more involved and detailed than others. These techniques included: 1) a basic Preliminary Hazard Analysis using available system descriptions and diagrams, 2) a review of historical safety analysis reports on plant designs, 3) the development of a Master Logic Diagram using the software code SAPHIRE[9] listing events that could result in one of the four initiating event groups described above, and 4) a generic list of current initiating events used in today reactors and their applicability in NGNP reactors.

2.2.1 Preliminary Hazards Analysis

A preliminary analysis is a qualitative technique that attempts to identify energy sources, hazardous materials, and radioactivity in use or stored at a facility. A preliminary hazardous analysis identifies ways in which radioactive, toxic, and flammable material could be released.[10] This approach is best used for facilities having detailed system descriptions, technical specifications, material inventory, and plant layout and schematics. The details in the NGNP design and scope of this project were not sufficient to allow use this technique.

2.2.2 Historical Sources of Data

To identify possible initiating events, past safety studies on reactors similar to NGNP reactors were reviewed including: Final Safety Analysis Reports for Fort St. Vrain (1982),[11] AP600, ESBWR, LMFBR, and other reactor designs.

Fort St. Vrain Experience

Public Service Company of Colorado (PSC) began construction of the Fort St. Vrain HTGR in 1968, and received a full-power operating license in 1973. Extensive testing and design issues prevented commercial operation until 1979. Although designed for 330 MWe, it operated at 200 MWe during its 10-yr life, and shut down for decommissioning in 1989.

Fort St. Vrain experienced a number of operational issues which resulted in a capacity factor below 15%[12]:

- core thermal and neutron oscillations
- leakage from the water-cooled bearings of the helium circulator
- material issues with the helium circulator
- control rod drive failures, including failure to scram
- design analysis inadequacies
- cracks found in main steam outlet piping assemblies
- turbine building fire damage

These issues don't necessarily apply to the current designs under consideration. For example, materials issues will depend on details not yet available. Environmental issues will be evaluated and licensed using current environmental qualification programs. Similarly, the new design will be subjected to substantially revised fire protection standards. The following sections describe some of these issues, and highlight any initiating event considerations that arise.

Core Thermal and Neutron Oscillations

Fuel block oscillations at Fort St. Vrain were solved using mechanical restraints added to the original design. Current HTGR designs acknowledge and address this issue in the mechanical design of the upper-plenum elements and through flow testing.

Helium Circulator Water Ingress

Water intrusion into the coolant system was a problem for Fort St. Vrain, specifically from the bearing cooling subsystem of the coolant circulators. However, the NGNP HTGR main circulators have a single-stage, axial-flow compressor with magnetic bearings, and are located at the top of the steam generator vessel. The shutdown cooling system has a two-stage, axial-flow compressor with oil-lubricated bearings, and is located at the bottom of the reactor vessel.

A related issue involves missile shielding of the MHTGR circulators. The reactor and steam generator vessels are not classified as safety-related systems. Consequently, the MHTGR designs do not include missile shielding. This potential initiating event should be captured in the MLD.

Control Rod Drive Failures

In NUREG-1338, the NRC reviewed the similarities and differences between the Fort St. Vrain control rod system and the reserve shutdown control equipment (RSCE) of the MHTGR, including a discussion of the Fort St. Vrain lessons learned. The design similarities supplemented with improvements based on lessons learned from Fort St. Vrain led the NRC staff to conclude that “a satisfactory level of mechanical performance can be achieved.” However, calculations made by DOE and ORNL show that the rapid control rod ejection could cause a prompt criticality excursion, which must be considered as a potential initiator in the MLD.

Turbine Building Fire and Environmental Qualification Issues

The turbine building fire issue and implementation of the environmental qualification program are not relevant to the MHTGR, since current standards for fire protection and environmental qualification will be applied prior to design approval.

Materials Issues

The material problems listed above were also not considered because there was little information available on these operational problems and for the MHTGR design, unless the problems were discussed in draft NUREG-1338.

Inadequate Original Design Analyses

Analysts compiling NUREG-1338 could not determine the specific problems involving inadequacies in the original design analysis, and so no additional initiating events are indicated.

Fort St. Vrain Design Differences

NUREG-1338 includes an NRC review of the Fort St. Vrain design to identify potential licensing issues applicable to the MHTGR design. Some of those conclusions are relevant to the initiating event analysis.

Fuel Design and Containment

The current TRISO fuel pellet design is similar to the pellet design used at Fort St. Vrain. Whether through design or manufacturing improvements, estimates of the failed-fuel fraction for the MHTGR are more than an order of magnitude lower than that reported for Fort St. Vrain, including normal and accident conditions. However, the current designs do not include the same kind of containment structure as did Fort St. Vrain. For Fort St. Vrain, the containment leak rate was 0.2 percent building volumes per day, a value comparable to that for a conventional, leak-tight, and pressure-retaining light-water reactor (LWR) containment. By comparison, the containment for the MHTGR is designed to leak at less than one building volume per day.

These two issues together were deemed to compose a licensing issue. However, neither one raises a concern in terms of Level 1, internal events initiators.

Reactor Cavity Cooling System

The reactor cavity cooling system (RCCS) comprises a set of baffled heat exchanger panels that line the cavity in which the reactor vessel sits. It can work passively through natural circulation for some period of time, although some designs also include active components. For the MHTGR design, the reactor cavity cooling system (RCCS) is the only safety-grade cooling system for the design should there be a complete loss of forced flow or loss of the steam generator. The Fort St. Vrain plant did not have a similar safety system and relied on having at least one of four circulators and one of six steam generators operating.

Aside from any licensing issues for the MHTGR design; it does not appear that the RCCS presents any initiating event concerns.

2.2.3 Master Logic Diagram

Master Logic Diagrams offer a basic approach for initiating event identification[13]. MLDs resemble fault trees and have similar logical relationships, though they are not quantified or analyzed in the same way.

Beginning with a defined top event (some undesired event such as “Release of fission products” or “Loss of function”); the MLD deductively breaks down events into simpler contributing causes. Gates indicate the logical connection between causes and effect, just as with fault trees (i.e., the events of a certain level will in some logical combination cause the events of the level immediately above). The process continues by breaking down events into their causes until a certain desired level of detail is reached (e.g., direct challenges to plant safety functions).

The MLD process is a deductive one, starting from a defined top event and breaking it down to simpler events involving combinations of initiating events and failures of safety functions. A hazards analysis on the other hand is an inductive technique that starts from a detailed plant design, and that examines potential deviations from normal operation, based on component failures, procedural errors, historical experience, etc. The two techniques are, in essence, two sides of the same coin. MLD provides a deductive, top-down approach, and hazards analysis provides an inductive, bottom-up approach. As with event trees and fault trees, hazards analysis and MLD are in principle equivalent. One could generate the same results with either approach. The choice in this case is determined by the level of design information available. The MLD process is better suited to the current status of the NGNP designs.

A preliminary MLD for the NGNP is presented in Figure 1 through Figure 7 below. The top event has been divided into five separate categories, covering chemical attack, heat removal, primary boundary, reactivity addition, and long-term phenomena. There is some overlap between these categories, as indicated in the MLD. However, the process should be comprehensive, and some amount of overlap is not a problem.

The MLD presented here covers all three NGNP designs. At the level of system design detail available, there was insufficient differentiation available to make separate MLDs worthwhile. Further development of the system designs may make it necessary to revisit the MLD and create design-specific models in the future.

Table 1 shows a list of the basic events in the MLD, with expanded definitions. Note that some of the events listed are plant or system responses to initiating events, and are not initiators themselves.

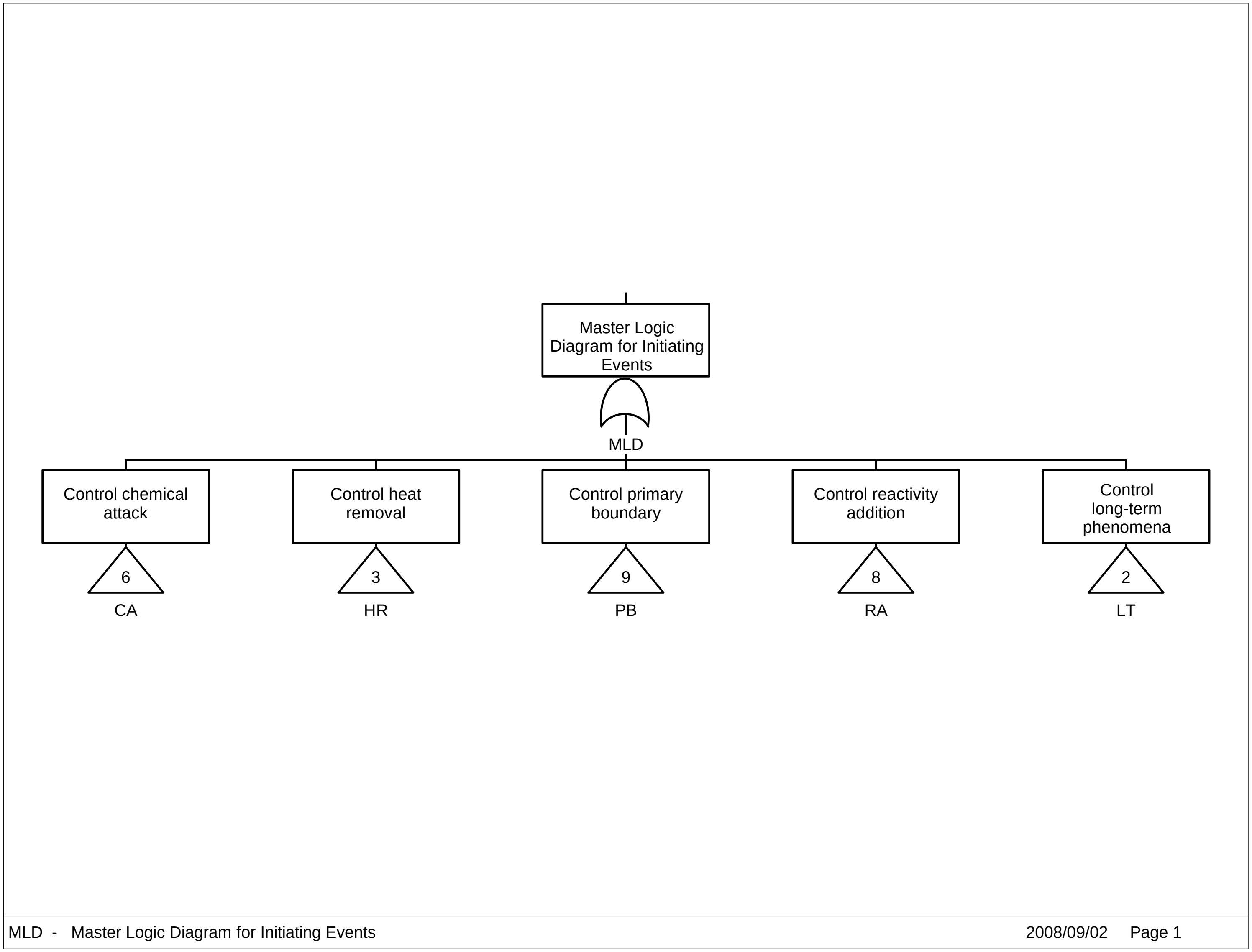

Figure 1. Master Logic Diagram for the NGNP

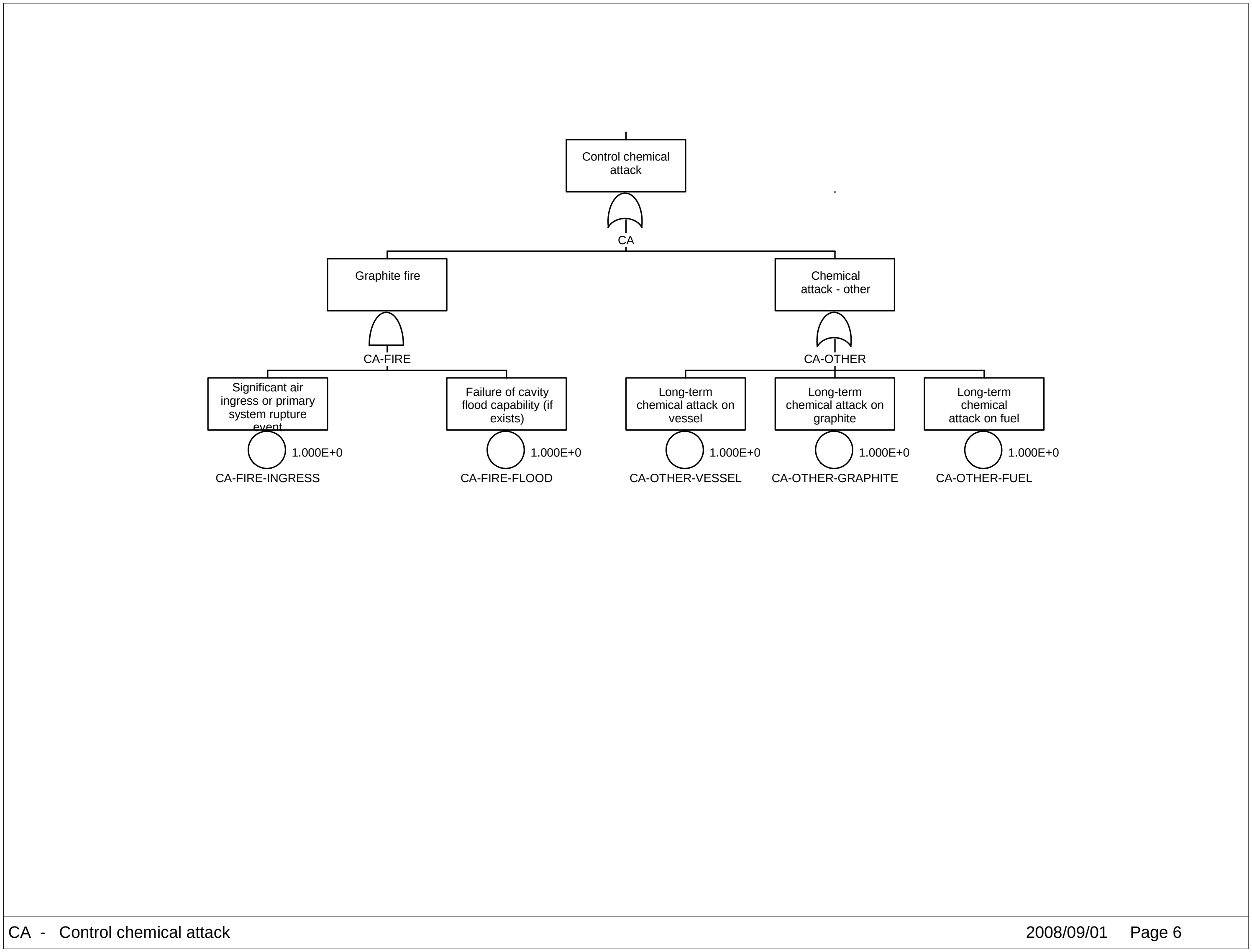

Figure 2. Master Logic Diagram for the NGNP – Control of chemical attack

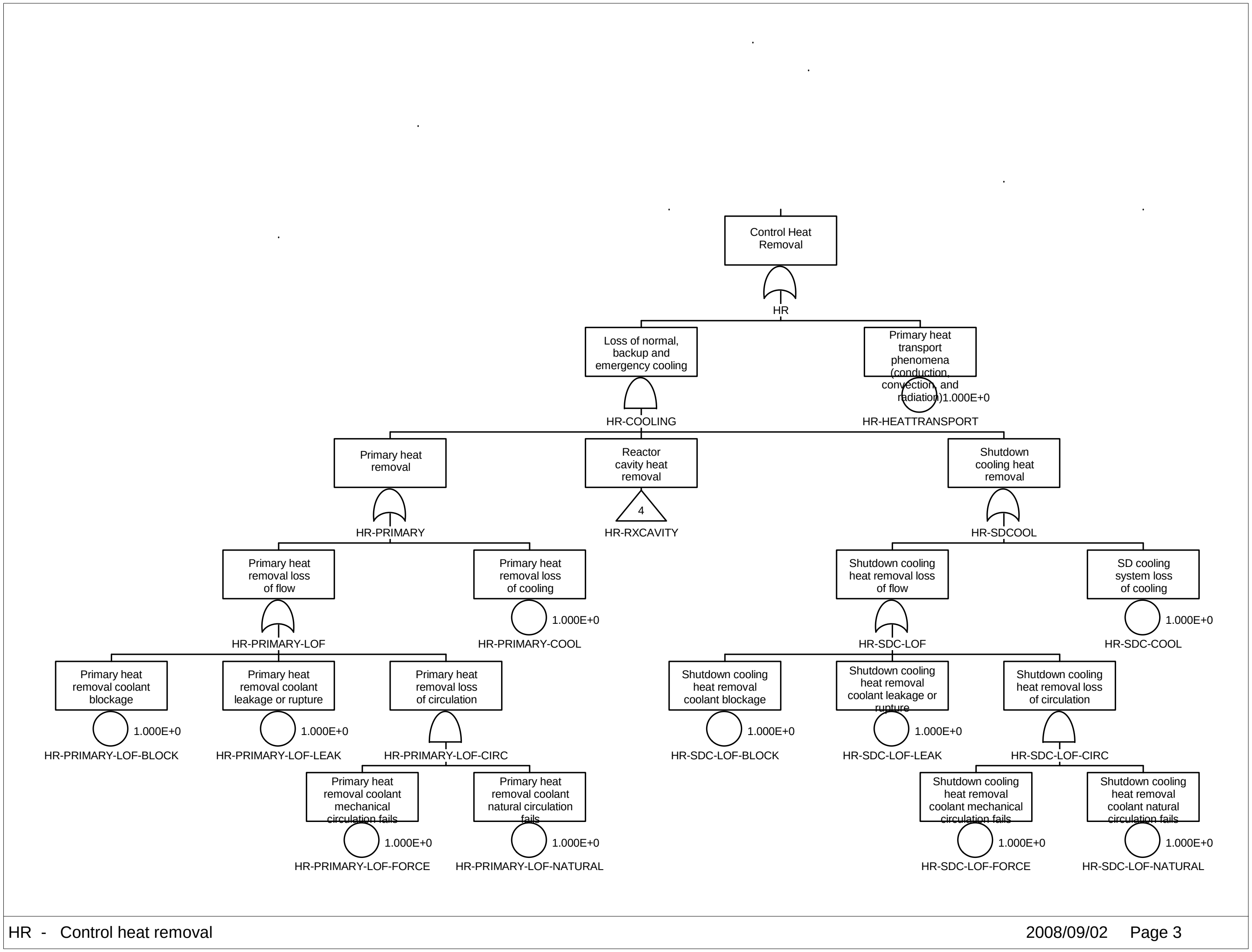

Figure 3. Master Logic Diagram for the NGNP – Control of heat removal

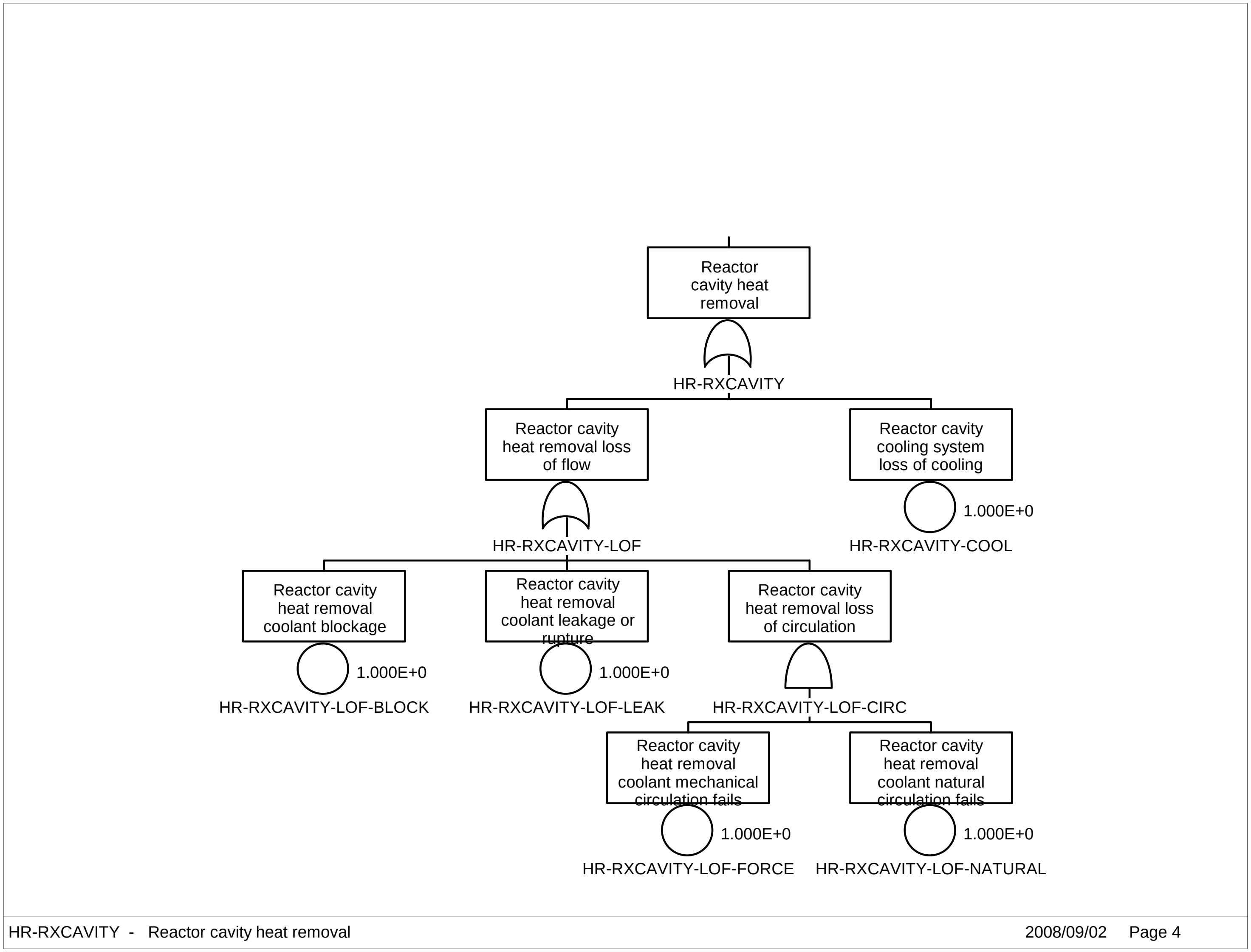

Figure 4. Master Logic Diagram for the NGNP – Control of heat removal (continued)

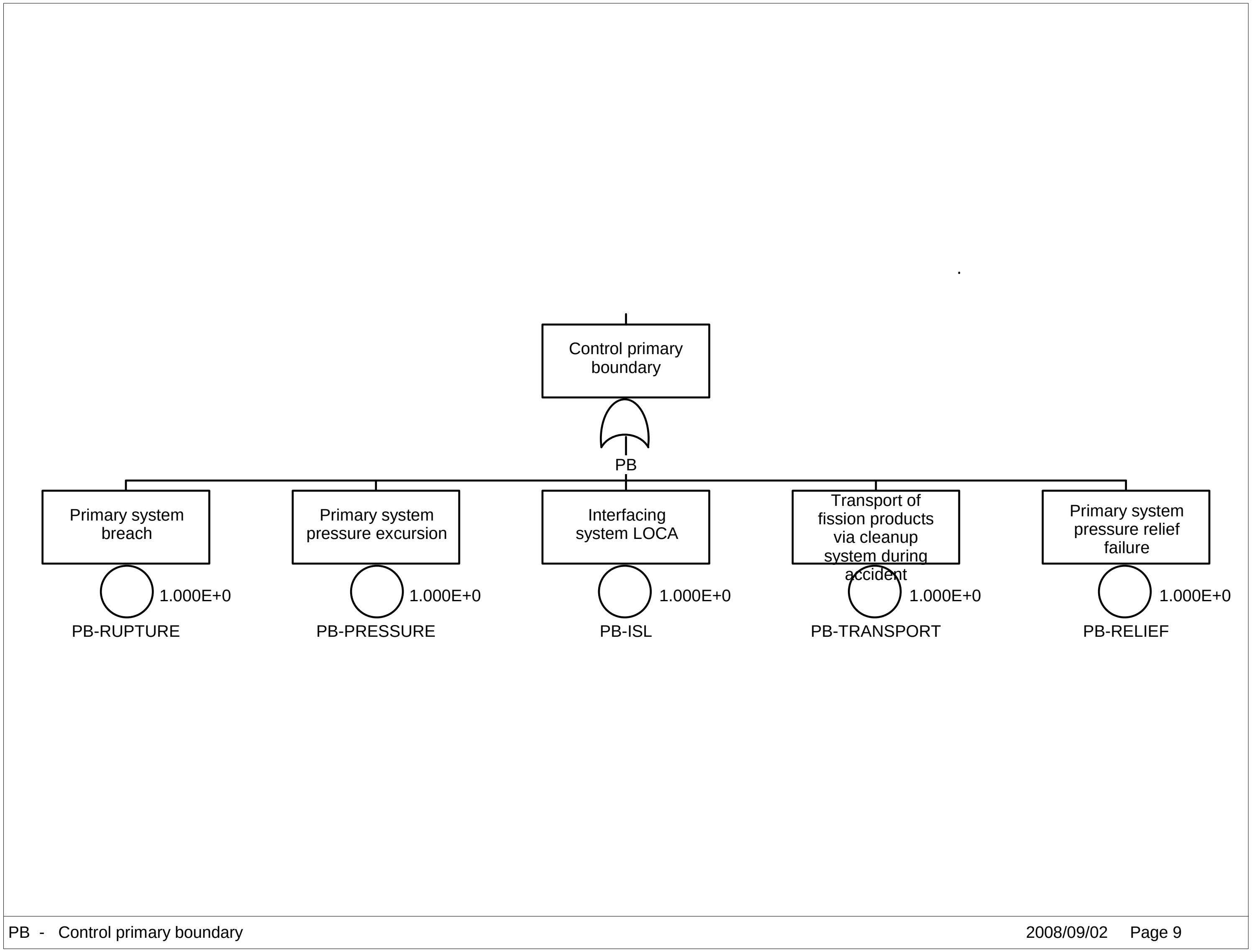

Figure 5. Master Logic Diagram for the NGNP – Control of primary boundary

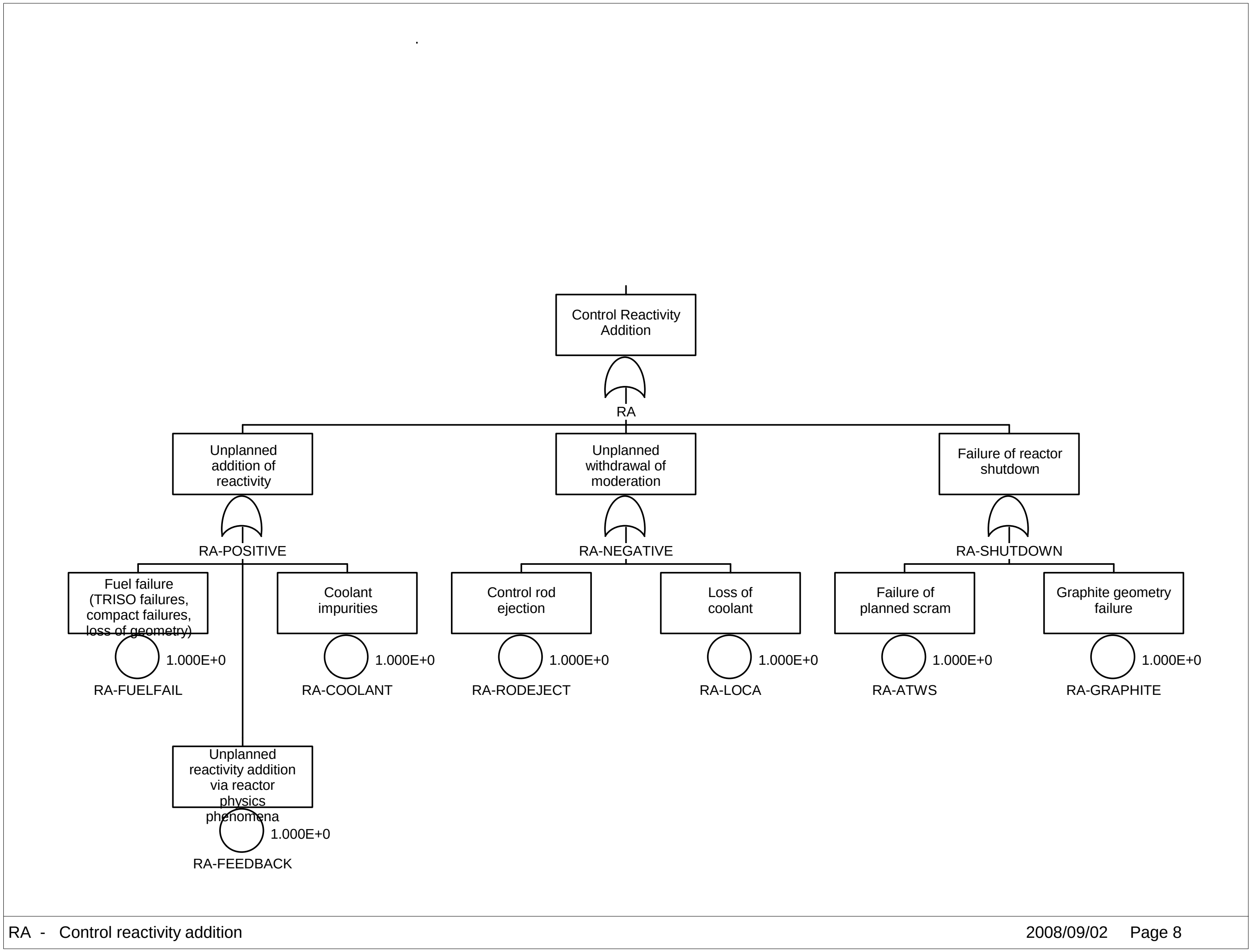

Figure 6. Master Logic Diagram for the NGNP – Control of reactivity addition

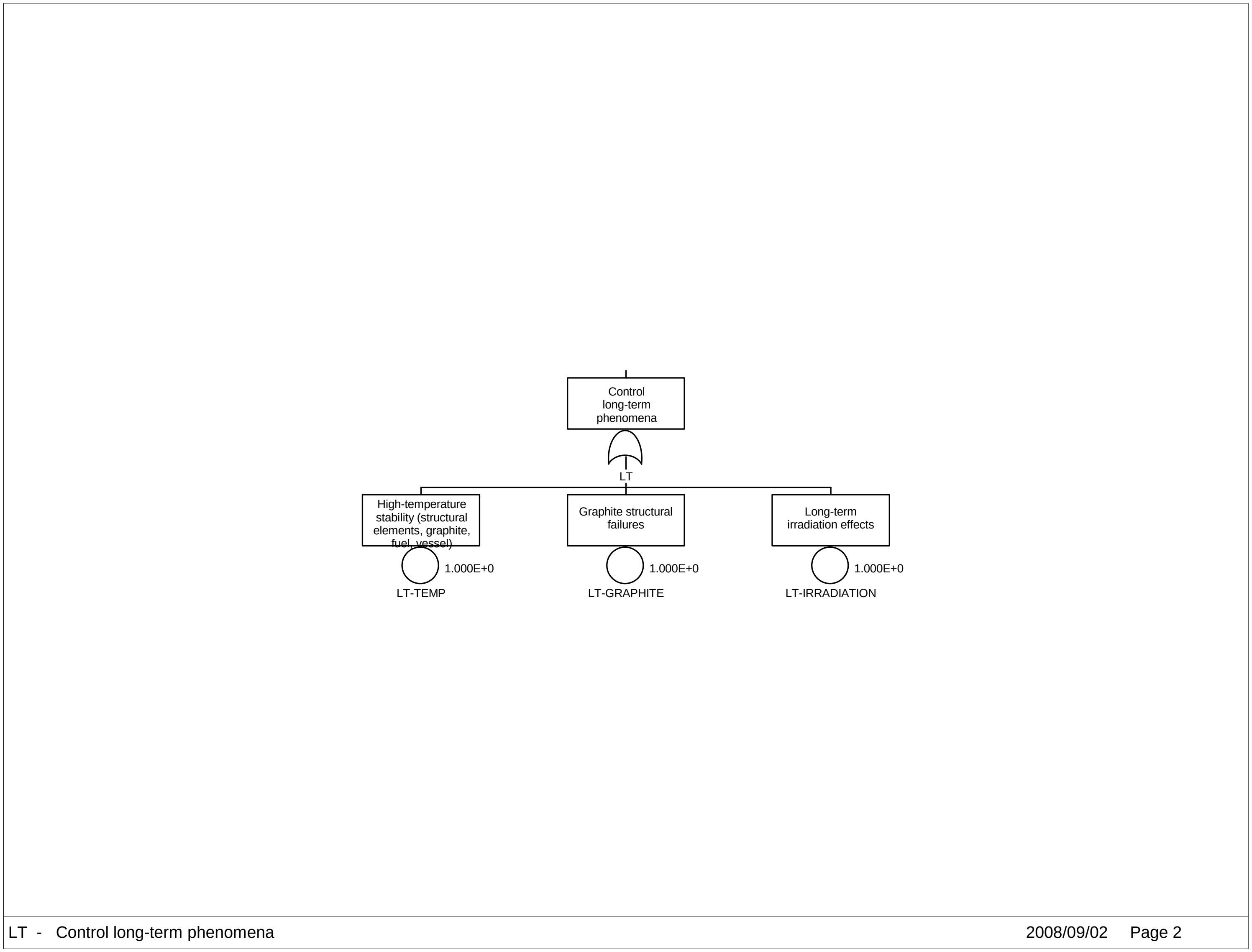

Figure 7. Master Logic Diagram for the NGNP – Control long-term phenomena

Table 1. Master Logic Diagram – List of basic events

CA-FIRE-FLOOD	Failure of cavity flood capability (if exists) – This event was included as a placeholder, although the NGNP designs don't seem to include this function. Fort St. Vrain did have a cavity flood function.
CA-FIRE-INGRESS	Significant air ingress or primary system rupture event – This event represents the existence of a sufficient primary system rupture such that air ingress and fire become significant threats. Note that a graphite fire might require a chimney effect, with multiple ruptures.
CA-OTHER-FUEL	Long-term chemical attack on fuel – This event covers any long-term chemical attack on the fuel pellets or compacts
CA-OTHER-GRAPHITE	Long-term chemical attack on graphite – This event covers any long-term chemical attack on the graphite blocks in the core
CA-OTHER-VESSEL	Long-term chemical attack on vessel – This event covers any long-term chemical attack on the primary system vessels (reactor, steam generator, or cross-duct)
HR-HEATTRANSPORT	Primary heat transport phenomena (conduction, convection, and radiation) – This event covers any unanticipated problems associated with heat transport parameters
HR-PRIMARY-COOL	Primary heat removal loss of cooling – This event covers events that cause loss of cooling to the primary cooling heat exchangers
HR-PRIMARY-LOF-BLOCK	Primary heat removal coolant blockage – This event covers mechanical blockages that prevent primary coolant circulation
HR-PRIMARY-LOF-FORCE	Primary heat removal coolant mechanical circulation fails – This event includes failures of the forced circulation capacity of the primary coolant system
HR-PRIMARY-LOF-LEAK	Primary heat removal coolant leakage or rupture – This event includes leakage or rupture of the primary coolant system such that the cooling capacity is significantly disrupted
HR-PRIMARY-LOF-NATURAL	Primary heat removal coolant natural circulation fails – This event includes failures of the natural circulation capacity of the primary coolant system
HR-RXCAVITY-COOL	Reactor cavity cooling system loss of cooling – This event covers events that cause loss of cooling to the reactor cavity cooling system heat exchangers
HR-RXCAVITY-LOF-BLOCK	Reactor cavity heat removal coolant blockage – This event covers mechanical blockages that prevent reactor cavity cooling system circulation
HR-RXCAVITY-LOF-FORCE	Reactor cavity heat removal coolant mechanical circulation fails – This event includes failures of the forced circulation capacity of the reactor cavity cooling system
HR-RXCAVITY-LOF-LEAK	Reactor cavity heat removal coolant leakage or rupture – This event includes leakage or rupture of the reactor cavity cooling system such that the cooling capacity is significantly disrupted
HR-RXCAVITY-LOF-NATURAL	Reactor cavity heat removal coolant natural circulation fails – This event includes failures of the natural circulation capacity of the reactor cavity cooling system
HR-SDC-COOL	SD cooling system loss of cooling – This event covers events that cause loss of cooling to the shutdown cooling heat exchangers
HR-SDC-LOF-BLOCK	Shutdown cooling heat removal coolant blockage – This event covers mechanical blockages that prevent shutdown cooling system circulation
HR-SDC-LOF-FORCE	Shutdown cooling heat removal coolant mechanical circulation fails – This event includes failures of the forced circulation capacity of the shutdown cooling system

Table 1. (continued)

HR-SDC-LOF-LEAK	Shutdown cooling heat removal coolant leakage or rupture– This event includes leakage or rupture of the shutdown cooling system such that the cooling capacity is significantly disrupted
HR-SDC-LOF-NATURAL	Shutdown cooling heat removal coolant natural circulation fails– This event includes failures of the natural circulation capacity of the shutdown cooling system
LT-GRAPHITE	Graphite structural failures – This event covers any unforeseen long-term structural issues associated with the graphite blocks due to aging (but not including irradiation or thermal phenomena)
LT-IRRADIATION	Long-term irradiation effects – This event includes any unanticipated long-term irradiation effects on structural components, graphite, vessel, or fuel
LT-TEMP	High-temperature stability (structural elements, graphite, fuel, vessel) – This event includes any unanticipated long-term temperature effects on structural components, graphite, vessel, or fuel
PB-ISL	Interfacing system LOCA – This event covers any events comparable to interfacing systems LOCAs in light water reactors
PB-PRESSURE	Primary system pressure excursion – This event covers failure of the primary system pressure boundary due to primary system pressure excursion beyond design capacity
PB-RELIEF	Primary system pressure relief failure – This event includes failures of the primary system pressure relief system
PB-RUPTURE	Primary system breach – This event comprises all significant primary pressure boundary rupture events
PB-TRANSPORT	Transport of fission products via cleanup system during accident – This event covers the accidental spread of fission products via normal cleanup system operation during accident conditions
RA-ATWS	Failure of planned scram – This event is comparable to the light water reactor ATWS
RA-COOLANT	Coolant impurities – This event includes any reactivity effects caused by coolant impurities
RA-FEEDBACK	Unplanned reactivity addition via reactor physics phenomena – This event covers any unanticipated reactivity feedback phenomena
RA-FUELFAIL	Fuel failure (TRISO failures, compact failures, loss of geometry) – This event covers any reactivity effects due to fuel failures, including fuel fabrication failures, aging issues, or accident phenomena
RA-GRAPHITE	Graphite geometry failure – This event covers any failures of graphite geometry that impact the ability of the reactor to shut down, including aging, irradiation, and thermal phenomena under anticipated and accident conditions
RA-LOCA	Loss of coolant – This event covers reactivity impacts of loss of primary coolant, including the effects of air ingress
RA-RODEJECT	Control rod ejection – This event covers control rod ejection events

The accident scenarios presented in NUREG/CR-6944[14] were reviewed during the construction of the MLD. The following scenarios were considered:

1. **P-LOFC** – the pressurized loss-of-forced circulation accident. This accident assumes a flow coast-down and scram with the RCCS operating continuously. The natural circulation of the pressurized helium coolant within the core makes core temperatures more uniform, lowering the peak temperatures more than in a depressurized core. The lack of buoyancy forces allows significant helium coolant recirculation flows. However, the chimney effect in P LOFC events increases the core (and vessel) temperatures near the top. The peak fuel temperature is not a concern, as it falls well within nominal temperature limits; the major concern is more likely to be the upper vessel and associated component temperatures.

2. **D-LOFC** – the depressurized loss-of-forced circulation accident. This scenario assumes a rapid depressurization of the primary coolant and scram, with the passive RCCS operational, and without air ingress. For the prismatic designs, this event is known as a "conduction heat-up" (or "cool-down") accident since the core effective thermal conductivity is the dominant mechanism for the transfer of afterheat from the fuel to the reactor vessel. For the pebble-bed design, radiation heat transfer is dominant in the core during the heat-up. Typically the maximum expected fuel temperature would peak slightly below the limiting value for the fuel (by design), and the peak would typically occur ~2 days into the accident.

3. **Augmented D-LOFC** – the depressurized loss-of-forced circulation accident followed by air ingress. A more extreme case of the D LOFC accident involves a significant and continuous inflow of air to the core following depressurization. This event may lead to graphite structure oxidation to the extent that the integrity of the core and its support is compromised. It could also involve oxidation of the graphite fuel elements, leading to exposure of the TRISO particles to oxygen, with a potential for subsequent fission product release. Finally, oxidation of graphite structures could lead to release of fission products previously absorbed.
4. **ATWS** – reactivity-induced transients, including events involving anticipated transients without scram. The most common postulated reactivity events assume a LOFC (either pressurized or depressurized) accompanied by a long-term failure to scram. These are extremely low-probability events, where the core heat-up transients are unaffected by the failure to scram until recriticality occurs upon the decay of the xenon poisoning (typically in ~2 days).

Other accident types are also discussed in NUREG/CR-6944[14], but these four were used in that analysis as a limiting set until further design information becomes available. In that sense, any evaluation of initiators will have to be revisited as well.

Several of the initiators generated in the MLD process have been eliminated from further consideration for various reasons. For example, the long-term effects were eliminated as they are too slow-acting to be viable as initiating events. Any impact from these events should be detected and managed during the life of the plant well before an accident could be initiated. Similarly, events based on undetermined effects of high temperatures, irradiation, etc., should be discovered during design testing and startup evaluations.

2.2.4 Initial List of Initiating Events

The initial list of IEs for the three NGNP reactor designs is shown in Table 2. This list is includes the susceptibility of the designs to the current traditional IEs effecting modern reactors.

Table 2. NGNP Susceptibility to Accident Initiators

INITIATOR			
Reactor design	**AREVA**	**General Atomics**	**Westinghouse**
Reactor design feature/variation	prismatic	prismatic	pebble bed
DESCRIPTION/PARAMETERS			
Power conversion	indirect	direct	indirect
Thermodynamic cycle	Brayton/Rankine	Brayton	Rankine
Safety design philosophy	passive	passive	passive
Primary coolant	helium-nitrogen	helium	helium
Primary coolant pressure (MPa)	7	7	9
Primary coolant outlet temperature (°C)	900	950	950
Secondary coolant	na	na	na
Secondary coolant pressure (MPa)	na	na	na
Reflector	graphite	graphite	graphite
Moderator	graphite	graphite	graphite
LOSS OF COOLANT ACCIDENTS			
Reactor Vessel Rupture	TBD	TBD	TBD
Large LOCA	✓	✓	✓
Medium LOCA	✓	✓	✓
Small LOCA	✓	✓	✓
Stuck Open Pressure Relief Valve	TBD	TBD	TBD
Primary Seal LOCA	✓	✓	✓
ISLOCA	TBD	TBD	TBD
Primary to Secondary HX LOCA	na	na	na
TRANSIENTS			
Pressure/Temperature Transient	HPCC, LPCC	HPCC, LPCC	HPCC, LPCC
Reactivity Transient	TBD	TBD	TBD
ATWS	✓	✓	✓
Loss of Heat Sink	✓	✓	✓
Loss of Secondary System Flow	na	na	na
High Energy Line Breaks/Leaks Outside Containment	✓ - H_2 plant	✓ - H_2 plant	na
High Energy Line Breaks/Leaks Inside Containment	✓	✓	✓
LOSS OF OFFSITE POWER			
LOOP	✓	✓	✓
LOSS OF RISK IMPORTANT BUS			
Loss of Medium Voltage AC Bus	TBD	TBD	TBD
Loss of Low Voltage AC Bus	TBD	TBD	TBD
Loss of DC Bus	TBD	TBD	TBD
LOSS OF SUPPORT SYSTEMS			
Loss of Service Water (Open)	TBD	TBD	TBD
Loss of Component (Closed) Cooling Water	TBD	TBD	TBD
Loss of Instrument Control Air	TBD	TBD	TBD
Loss of Other Support System(s)	TBD	TBD	TBD

EXTERNAL EVENTS			
Fire	✓	✓	✓
Flood	✓	✓	✓
Seismic	✓	✓	✓
Severe Weather (Tornado, Hurricane, etc.)	✓	✓	✓
NRC Design Basis Threat (ex: Terrorism)	✓	✓	✓
PLANT SPECIFIC			
Air Ingress	na	na	na
Water Ingress	✓	✓	✓
Loss of circulators	✓	✓	✓
Stuck Brayton Cycle Turbine	✓	✓	✓
Isolation valve closure	TBD	TBD	TBD
Fuel failure	✓	✓	✓
Online Refueling Fault	na	na	✓
Hydrogen Cogeneration Impact	✓	✓	✓

3. Conclusions

This letter report provided an overview of the process that was completed to develop an initial list of possible initiating events for NGNP reactors. The report focused on three NGNP designs: 1) Westinghouse Pebble Bed, 2) AREVA Prismatic HTGR, and 3) General Atomic Prismatic HTGR. To generate the initial list, a literature search on information currently available was gathered and reviewed. Currently, the three designs have not been fully developed and an extensive system description involving dependency requirements and success criteria is not available. From the available information, a general MLD was developed to aid in the identification of possible initiating events. More detailed studies of system and subsystem dependencies would be required to develop a more complete initiating event list for each reactor design.

Future work includes refinement of the initiating event list as more design information becomes available. Specifically, individual MLDs for each NGNP design may be desirable. Also, development of initiating event frequencies themselves must be generated (possibly after the design decision is finalized).

However, other possible follow-on options for initial PRA support for the NGNP project exist. The following are significant efforts that can and should be started in order to facilitate and expedite the development of the PRA, which will be an integral and necessary part of any NRC license application. These issues are described below in no particular order of importance.

External Hazards Assessment – This task would review the available information (primarily the ATR PRA) to identify (and possibly quantify) those hazards external to the NGNP that would need to be included in any PRA that is eventually performed. This work will be a necessary part of the PRA that would eventually be done and submitted to the NRC, and is largely independent of the design of the NGNP. The external hazards assessment is primarily dependent on the location, so could be done without detailed design information.

Site Demographics and Meteorology – The license application will require a full (all operating modes, all hazards) level-3 PRA. The level-3 portion of the PRA assesses the health consequences of any radioactive release into the environment. This assessment requires detailed information on demographics and meteorology for the site of the NGNP. Assembling this information (and possibly preparing test runs using the MACCS2 computer code) does not require detailed plant design information.

Simulation Capability for PRA - The new, advanced designs being considered for NGNP rely on passive systems and physical phenomena for maintaining reactor integrity and public health and safety. The successful operation of these passive systems will be determined by the presence and status of physical parameters (such as temperature, pressure, etc.). However, traditional PRA techniques were developed to model the reliability of active systems and components, and are not well suited for tracking physical parameters and modeling these passively operating design features. One approach that has been proposed to address this issue is the integration of simulation techniques into a traditional PRA framework. The INL has developed and maintained (under NRC sponsorship) a PRA code named SAPHIRE. The goal of this task would be to develop a simulation capability in SAPHIRE that would be used for modeling passive system reliability. (Note that simulation has also been proposed for modeling software reliability and severe accident phenomena in a PRA context.)

4. REFERENCES

1 Bedford, T. and R. Cooke, 2001, *Probabilistic Risk Analysis Foundations and Methods*, Cambridge University Press.

2 NGNP Pre-Conceptual Design Report, Revision 1, INL/EXT-07-12967, November, 2009

3 A.P. George, *Introduction to the Pebble Bed Modular Reactor (PBMR)*, Document Reference Number 009949-185. August 2001.

4 US Design Certification, Probabilistic Risk Assessment Approach for the Pebble Bed Modular Reactor, Document Number 039144, June 2006.

5 W. Shier, R. Fullwood, P. Rose, *Design Description and Preliminary Reliability Allocations for the Modular High Temperature Gas Cooled New Production Reactor*, BNL-NPR-MHTR-2, April, 1990.

6 NUREG/CR-2300, PRA Procedure Guide: A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants.

7 Bernreuter, D. et al, 1988, *Seismic Hazard Characterization of 69 Nuclear Plant Sites East of the Rocky Mountains*, NUREG/CR-5250.

8 IAEA Safety Reports Series #54, *Accident Analysis for Nuclear Power Plant with Modular High Temperature Gas Cooled Reactors*, 2008

9 Russell, K.D., et al., 1999, *Systems Analysis Programs for Hands-on Reliability Evaluations (SAPHIRE) Version 6.0 - System Overview Manual*, NUREG/CR-6532.

10 L. C. Cadwallader, J. L. Latkowski, *Preliminary Identification of Accident Initiating Events for IFE Power Plants*, 19th IEEE/NPSS Symposium on Fusion Engineering, October 2001.

11 Fort St. Vrain HTGR Final Safety Analysis Report (FSAR), 1982

12 NRC Office of Nuclear Reactor Regulation, *Preapplication Safety Evaluation Report for the Modular High-Temperature Gas-Cooled Reactor (MHTGR)*, NUREG-1338.

13 Papazoglou, I.A. and O.N. Aneziris, *Master Logic Diagram: Method for Hazard and Initiating Event Identification in Process Plants*, Journal of Hazardous Materials, 2002.

14 Oak Ridge National Laboratory (ORNL), *Next Generation Nuclear Plant Phenomena Identification and Ranking Tables (PIRTs)*, NUREG/CR-6944.

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