EUROPEAN COMMISSION

Community Research

RAPHAEL

Contract Number: 516508 (FI6O)

DELIVERABLE (D-ST4.3)

Application to RAPHAEL reference:
Determination of situations to be considered in the safety demonstration

Author(s): **Luigi MANSANI** **ANSALDO NUCLEARE**

Reporting period: 15/04/2008– 14/04/2009

Date of issue of this report : 16/06/2008

Start date of project : **15/04/2005** Duration : **48 Months**

Project co-funded by the European Commission under the Euratom Research and Training Programme on Nuclear Energy within the Sixth Framework Programme (2002-2006)

Dissemination Level		
PU	Public	
RE	Restricted to the partners of the RAPHAEL project	
CO	Confidential, only for specific distribution list defined on this document	X

RAPHAEL *(ReActor for Process heat, Hydrogen And ELectricity generation)*

Distribution list

Person and organisation name and/or group	Comments
Partners of the RAPHAEL SP ST	ON SINTER
P.Manolatos	EC DG RTD

ReActor for Process heat, Hydrogen And ELectricity generation (RAPHAEL)		
Sub-project: SP-ST **Work package:** 4	**RAPHAEL document no:** RAPHAEL-0806-D-ST4.3	**Document type: D** D=Deliverable / M=Minute / P=Progress report / S=Support
Issued by: ANSALDO NUCLEARE **Internal no.:** RAPHAEL SPST TRIX 002		**Document status:** Final

Document title

Deliverable D-ST4.3 - Application to RAPHAEL reference:

Determination of situations to be considered in the safety demonstration

Executive summary

The goal of this document is to identify, by means of a systematic approach, all the "internal" initiating events (i.e. events that originate inside the facility) that lead to abnormal or accident conditions in an indirect cycle High Temperature Reactor (HTR) with prismatic block type fuel elements. The indirect cycle HTR considered as reference, in this document, is the ANTARES plant design for both electricity production and process heat generation developed by AREVA NP. The document does not address "external" events (i.e. events that originate outside the facility).

After the identification, the initiating events are categorized. This will permit, subsequently (however out of the scope of this document), the selection of the limiting ones.

It is underlined that limited technical information is available for the HTR systems taken as a reference; the detail of the analysis presented in the document is thus necessarily commensurate to the amount of available information.

Revisions

Rev.	Date	Short description	Author	WP Leader	SP Leader	Coordinator
00	16/06/2008	First issue	L. Mansani, Ansaldo Nucleare	S. Ehster, AREVA NP	C. Angulo, Suez Tractebel	E. Bogusch, AREVA NP

AnsaldoNucleare

Una Società Finmeccanica

Progetto project	Identificativo document no.
RAPHAEL	RAPHAEL SPST TRIX 002

Cliente customer	Sistema sys.	Comm.-S.comm. job. no.	Emittente issued by	Pagina page	Di of
European Commission		SC2010	ETN/RSC	1	65

Rag. Soc. Fornitore suppliers	Sigla code	Ident. esterno / Rev. supplier ident. / rev.	Cl. ris. class	Tipo doc. doc type

Ordine attivo / Contratto order	Identificativo Componente equipment identif. code
516508 (FI6O-2005)	

Titolo title

Technical Report
Contractual Deliverable D ST4.3
Application to RAPHAEL reference: Determination of situations to be considered in the safety demonstration

Sostituisce substitutes

G.Q. q.a.	Cant. site	ITER Forn. supplier	ITER Ansaldo	ITER Cliente customer
B	N	-	E	-

Note notes

ELECTRONIC FILENAME: RAPHAEL SPST TRIX 002_0.DOC ELECTRONIC FORMAT: MSWORD2003 DESCRIPTION: TEXT

Rev. rev.	Rif. Approv. Ansaldo Ansaldo's appr. ref.	Rif. Approv. Cliente Customer's appr. ref.	Rev. rev.	Rif. Approv. Ansaldo Ansaldo's appr. ref.	Rif. Approv. Cliente Customer's appr. ref.

Rev. rev.	Data date	Descrizione description	Stato validità issue st.	Redazione prepared by	Controllo/ approvazione checked / approved by	Autorizzazione emissione issue authorization
0	16-06-2008	First Issue		L. MANSANI	A. ALEMBERTI	A. ALEMBERTI

Cronologia e storia delle revisioni
Chronology and history of revisions

Rev. / rev.	**Data** / date	**Descrizione** /description
0	16-06-2008	First Issue

CONTENTS

1. SCOPE

The goal of this document is to identify, by means of a systematic approach, all the "internal" initiating events (i.e. events that originate inside the facility) that lead to abnormal or accident conditions in an indirect cycle High Temperature Reactor (HTR) with prismatic block type fuel elements. The indirect cycle HTR considered as reference, in this document, is the ANTARES plant design for both electricity production and process heat generation developed by AREVA NP. The document does not address "external" events (i.e. events that originate outside the facility).

After the identification, the initiating events are categorized. This will permit, subsequently (however out of the scope of this document), the selection of the limiting ones.

It is underlined that limited technical information is available for the HTR systems taken as a reference; the detail of the analysis presented in the document is thus necessarily commensurate to the amount of available information.

The document is issued in the frame of the European Commission funded project RAPHAEL (6th Framework Programme, contract 516508 FI60) as contractual deliverable D ST4.3 within Work Package ST4 (Safety Approach and Licensing Issue).

2. DETERMINATION OF INITIATING EVENTS THROUGH TOP/BOTTOM APPROACH

2.1 GENERAL

The general objective of nuclear safety is to protect individuals, population and the environment by establishing and maintaining an effective defence against radiological hazard. The safety philosophy primarily addresses the prevention of accidents but also devotes attention to the protective features (i.e. those features designed to control and limit the consequences of a given accident) as well as the mitigation of the consequences of accidents which could give rise to major releases. The aim is to reduce both the probabilities of the events and their associated off-site consequences in order to avoid the need of extensive countermeasures and to offer the Authorities the possibility of simplifying or declaring not necessary the off-site emergency plan.

The basic approach is to systematically identify all the foreseeable initiating events and to group them either in Design Basis Conditions (DBC) or Design Extension Conditions (DEC). Design Basis Conditions are then analyzed to deterministically prove that their predicted consequences stay within prescribed limits (established according with the basic principle that the most probable occurrences should yield the least radiological risk to the public and those extreme situations having the potential for the greatest risk to the public shall be those least likely to occur). Design Extension Conditions, formerly called Beyond Design Conditions, include a specific set of accident sequences selected and analyzed with the basic aim to demonstrate that probabilistic safety objectives for release of radioactive products to the external atmosphere are met.

In the HTR the defence in depth concept is implemented to achieve effective defence against radiological hazards. Defence in depth consists in a hierarchical deployment of different levels of equipment and procedures in order to maintain the effectiveness of the physical barriers placed between radioactive materials and the workers, the public and the environment in normal operation, anticipated operational occurrences and, for some barriers, in accidents at the plant. Moreover, in ANTARES the safety characteristics are developed and emphasized in order to simplify or even to eliminate the need of safety system: a major pre-conceptual design phase objective is to be able to exclude early severe core damage.

Generally, several successive physical barriers for the confinement of radioactive material are put in place. Their specific design may vary depending on the amount of radioactivity of the contained material and on the possible deviations from normal operation that could result in the failure of some barriers. The public and the environment are protected primarily by means of these barriers, which may serve operational and safety purposes or safety purposes only. The defence in depth concept applies to the protection of their integrity against internal and external events that may jeopardize them. The physical barriers identified in the HTR are the following:

1. The fuel (TRISO-coated fuel particles). This is the first and the main confinement barrier that achieves this function during all of the events considered likely to occur. The remaining barriers (i.e., primary pressure boundary and reactor building) minimize the consequences. The radionuclide releases through intact fuel particles (e.g., diffusion) shall be limited during normal and accident conditions. The failed fuel ratio, which is low during normal operating conditions, shall not be significantly increase during any accident condition.
2. The pressure boundary. The pressure boundary comprises the vessel system, the heat exchangers located within the pressure boundary. Isolation valves implemented on the connected circuits and on the heat exchangers cooling circuits also achieve this function (e.g., in case of heat exchanger failure). This second barrier shall be leak tight during normal operation as well as during accident conditions.
3. The containment (confinement). A major objective, in design an HTR, is to make a pressure resistant containment unnecessary. It relies on the expected high performances of the coated fuel particle, complemented by a slow evolution of any accident transients resulting from the low power density and the large thermal inertia of the core. These characteristics allow to exclude the combination of large radiological releases in the primary circuit, with a high primary pressure. During normal operation, the confinement function is assured by the ventilation system of the reactor building monitoring and controlling the releases. The reactor building shall keep its structural integrity during any accident condition. In case of small primary helium leakage the radiological releases are monitored and minimized through adequate retention filters. In case of higher primary helium leaks, helium is first discharged into the reactor building and then to the stack. The radiological releases is controlled and minimized once the primary helium discharge is completed (by complete isolation of the reactor building or by maintaining the reactor building pressure below the external pressure and by adequate filtration of the monitored releases).

The objective of this document is to define a comprehensive set of events that encompasses both the anticipated occurrences and the conceivable malfunctions and failures that pose challenges to the integrity of the physical barriers (fuel, primary system, containment) and/or may result in radioactivity releases to the environment. However the initiating faults to focus in this phase of the project are mainly those occurring during normal operation of the plant and the consequence of which are likely to challenge the fuel particles confinement capability.

Two approaches can be taken in identifying the accident initiating events. One is a comprehensive engineering evaluation, taking into consideration information from previous risk assessments, documentation reflecting operating histories and plant-specific design data. The information is evaluated and a list of initiating events is compiled, based on the engineering judgment derived from the evaluation. Another approach is to more formally organize the search for initiating events by constructing a top-level logic model (called Master Logic Diagram) and then deducing the appropriate set of initiating events. Because of the peculiar characteristics of the HTR (and the resulting lack of plant specific data as well as, even more so, of operating data) the second approach has been adopted to define the initiating events set.

2.2 MAIN COMPONENTS/SYSTEMS OF THE ANTARES

RAPHAEL project is aimed at developing the HTR/VHTR technologies. ANTARES (AREVA New Technology based on Advanced gas cooled Reactor for Energy Supply) design has been selected as reference for this studies [1], [2] and [3]. The ANTARES main features are:

- Reactor core thermal power 600 MWth.
- Primary coolant Helium.
- Core inlet temperature 400°C.
- Core outlet temperature 850°C.
- Indirect cycle arrangement.
- Two options for the primary loop: i) one loop with plate type Intermediate Heat eXchanger (IHX) or ii) two loops with two tubular type IHXs.
- IHX secondary inlet temperature 350°C.
- IHX secondary outlet temperature 800°C.
- Primary loop(s) coupled to the process heat application through an intermediate heat transport loop that uses as coolant a mixture of 80% Nitrogen and 20% Helium.
- Power generating system combined cycle gas turbine with steam bottoming cycle.

The main parameters characterizing the normal operation of the ANTARES plant are summarized in Table 1.

The main components/systems of the ANTARES, relevant for the safety assessment, are listed below with a brief description of their functions. The description is provided to better understand the Master Logic Diagram constructed in the following Section 2.3. The ANTARES plant includes the following key components/systems:

- **Fuel Particles (TRISO) and Fuel Blocks**. Fuel takes the form (Fig. 2.2-1) of a particle containing a core of fissile material (kernel of UO_2) surrounded by a buffer layer of carbon, a layer of pyrolitic carbon, a layer of silicon carbide and an outer layer of pyrolitic carbon (the overall diameter is about 1 mm). The functions are differentiated: the inner porous layer of carbon serves as a buffer for the fission gases, the silicon carbide layer plays the role of a barrier to prevent the diffusion of solid fission products while the two dense pyrolytic carbon layers provide mechanical resistance to the internal pressure of the fission gases and help to retard the migration of solid fission products. The fuel particles are agglomerated in a graphite matrix in the form of cylindrical rods called compacts. The compacts are inserted into prismatic graphite blocks, to constitute an organized core structure.
- **Vessel System** (VS), which includes the Reactor Pressure Vessel (RPV, housing the reactor system, the reactor internals and the reactor support structures), the Intermediate Heat eXchanger (IHX) vessel (IHXV, housing the IHX and the Main Primary Gas Circulator (MPGC)), the cross vessel (housing the internally insulated hot duct and delimiting the surrounding annular cold duct), vessel supports, and lateral

restraints. The VS has the functions to confine the primary coolant and to maintain primary coolant boundary integrity (Fig. 2.2-2).

- **Reactor Core System** (RCS), which includes (Fig. 2.2-3 and Fig. 2.2-4) the reactor core (hexagonal graphite fuel blocks forming 102 columns with 10 block height), the reactivity control assemblies (Normal Shutdown System (NSS) control rods (split in two groups: 36 operating control rods and 12 start-up control rods) and Reserve Shutdown System (RSS)), the core supports, the internal structures (permanent side reflector, replaceable reflectors, metallic core support (barrel), upper core restraint, upper plenum shroud), and the hot gas duct assembly. The RCS has the functions to generate heat from the release of energy of nuclear fission, to transfer heat to the primary coolant Helium and to confine radioactive products.
- **Main Primary Gas Circulator** (MPGC), which includes an electric motor and a compressor immersed in the primary Helium inside the IHXV. It is used to control the primary Helium flow rate modulating the rotational speed. The MPGC rotor is supported by active magnetic bearings and by catcher bearings. The MPGC includes also a shutoff valve to isolate the heat transport system from the RPV when the MPGC not operate (Fig. 2.2-5).
- **Intermediate Heat eXchanger** (IHX), which transfers the heat from the primary coolant to the secondary. Two types of IHX are envisaged: plate type and helical tube type. The number of IHX and IHXV depend from the IHX type selected. The IHX can be isolated from the Secondary System by means of the Secondary Separation Valves (SSV-H on the hot gas side and SSV-C on the cold gas side) (Fig. 2.2-5 and Fig. 2.2-6).
- **Secondary System.** The nuclear heat source is coupled via the IHX to a secondary system. The heat transferred to the secondary system can then be used to generate electricity (Fig. 2.2-6) either in a Bryton cycle or in a steam (i.e. Rankin) cycle. The secondary system can also be used as process heat including hydrogen production (Fig. 2.2-7).
- **Core Heat Generation Control System** (CHGCS), which includes two reactivity control sub-systems, the Normal Shutdown System (NSS) control rods and the Reserve Shutdown System (RSS). NSS control rods are split into two groups: i) 36 operating control rods, ii) 12 start-up control rods. The NSS control rods have the same design and are made by boron carbide as neutron-absorbing material. NSS is used to maintain sub-criticality during cold shutdown conditions, to compensate for the xenon effect, to compensate reactivity effect in case of water ingress accident. The RSS is equipped by spherical neutron-absorbing elements that are dropped into the core fuel assembly channels by gravity. RSS provides reactor shutdown independently and diversely from NSS and it is designed for maintaining the core in sub-critical state if the NSS fails to operate.
- **Secondary Decay Heat Removal System** (SDHRS), a non safety-related loop implemented on the secondary system (Fig. 2.2-6). The operation of this system requests the availability of the forced helium circulation in the primary circuit and the IHX integrity. SDHRS could also be used for normal start-up and shutdown of the plant.

- **Shutdown Cooling System** (SCS), a non safety-related system which include three heat transport circuits in series (Fig. 2.2-8) designed to remove heat from the RCS and transfer that heat to the ambient air. The first circuit is in parallel with the plant Primary Heat Transport System (PHTS) across the RCS and consists of a helium-to-water heat exchanger, an electrically powered gas circulator and a shutoff valve. The second circuit is a closed pressurized water heat transport loop that runs from the helium-to-water heat exchanger to a water-to-air heat exchanger. The water is circulated by conventional electrically powered pumps, and the ultimate heat sink (third circuit) is an air-blast type heat exchanger with electric fans. SCS can operate even if the secondary circuit and the primary forced helium circulation are not available. SCS is designed for achieving this function in pressurized and depressurized conditions.
- **Reactor Cavity Cooling System** (RCCS), a safety-related passive water cooling system (Fig. 2.2-9) for decay heat removal during emergency cool-down, for cavity heat removal during normal plant operation and for confining of radioactivity released into the reactor cavity during normal operation. The RCCS consists of two independent and redundant trains operating in natural circulation. Each train consists of the following four major components, plus associated pipes, headers and valves, all located inside the reactor building and the reactor auxiliary building: i) a panel wall cavity cooler, consisting of alternating vertical pipes around the periphery of the RPV (a compact air-to-water heat exchanger that surrounds the RPV); ii) a water storage tank (a water-to-water heat exchanger is inside and integral to the pressure boundary of the water storage tank); iii) a water-to-air heat exchanger (closed circuit cooling tower); and iv) a circulating water pump.
- **Helium Processing System** (HPS)**.** System to transfer, to purify and to storage the Helium.
- **Fuel Handling and Storage System** (FHSS)**.** System to handle the fuel and reflector blocks, and to transport them between the receiving facility, the reactor core, and the fuel packaging and shipping facility.
- **Reactor Control and Protection System** (RCPS). System to provide the monitoring and control of the technological processes in all modes of plant operation, including emergencies.

TABLE 1: ANTARES main parameters for reference configuration

Parameter	Reference ANTARES Electric Plant
Reactor core power	600 MWt
Reactor inlet temperature	400°C
Reactor outlet temperature	850°C
Fuel type	TRISO, UO_2, prismatic block
Fuel cycle	LEU (< 20%), once-through, 18-month refueling interval, 36-month residence time (two batches)
Core geometry	102-column, annular, 10-block high
Primary pressure	6.0 MPa
Primary coolant	Helium
Number of primary loops	1
IHX type	Modular compact IHX
IHX secondary inlet temperature	350°C
IHX secondary outlet temperature	800°C
Secondary pressure	5.5 MPa
Secondary coolant	80% nitrogen and 20% helium
Pressure boundary material	Mod 9Cr 1Mo
Power generating system	Combined cycle: gas turbine with steam bottoming cycle
Net efficiency	> 46%
Number of modules per plant	4
Net plant output	~ 1130 MWe

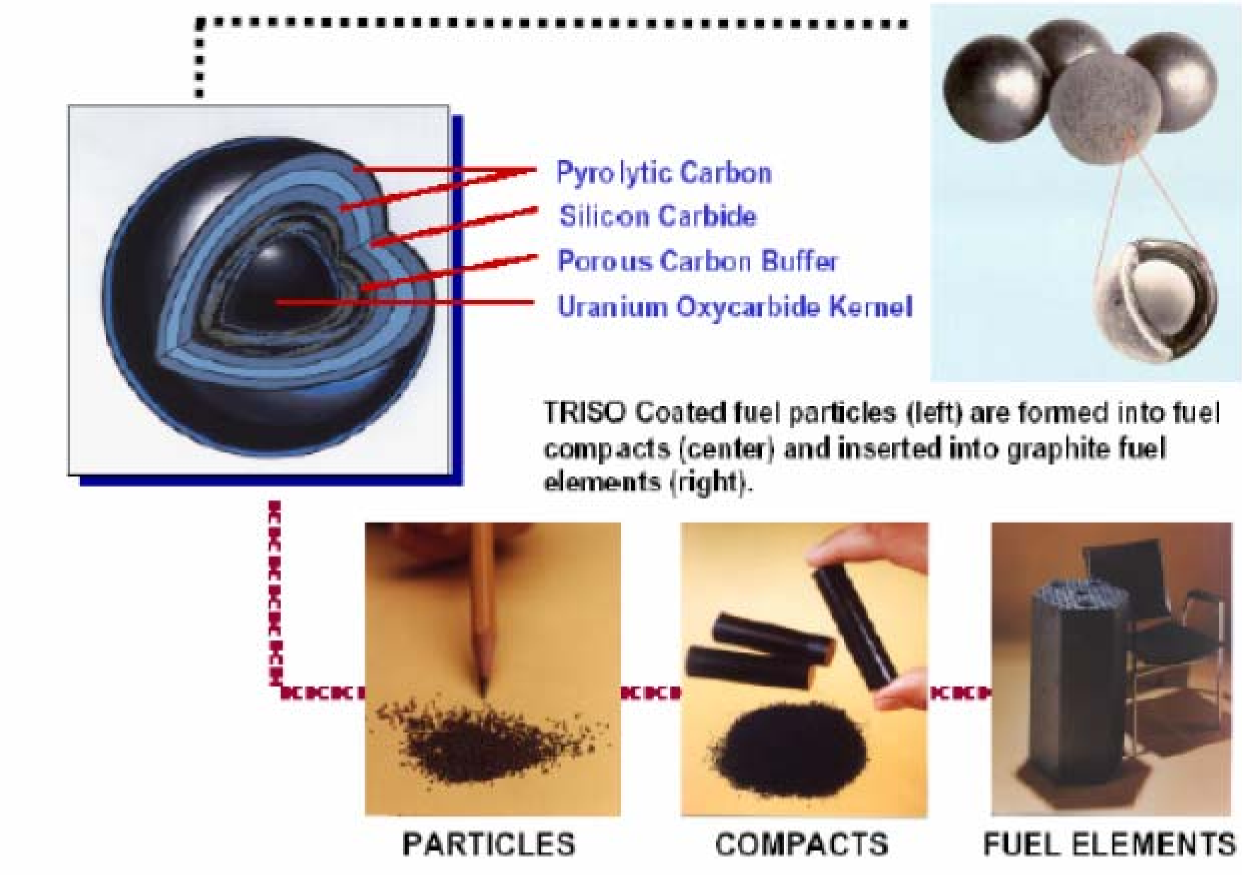

Figure 2.2-1 Fuel Particles (TRISO)

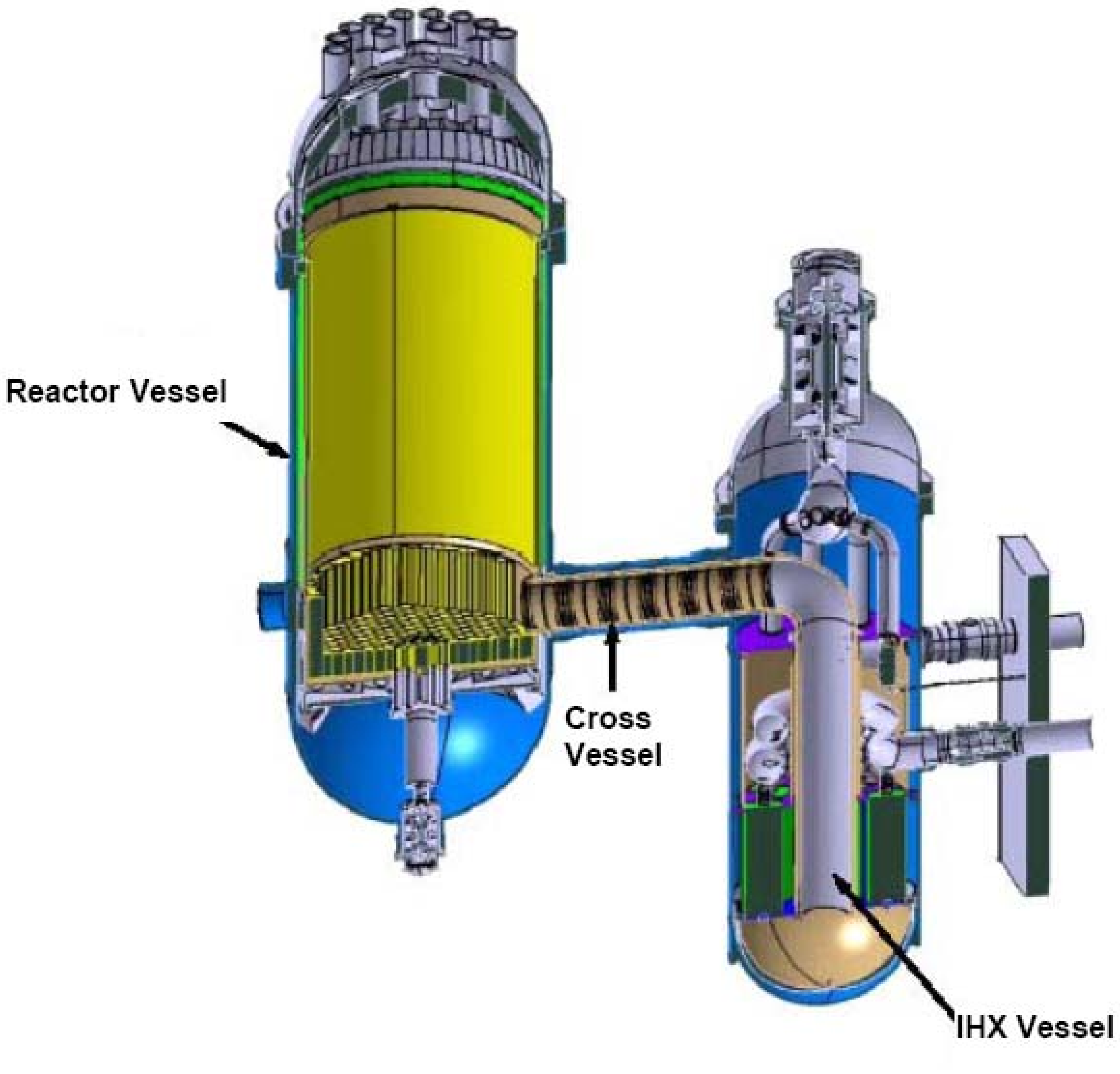

Figure 2.2-2 ANTARES Vessel System

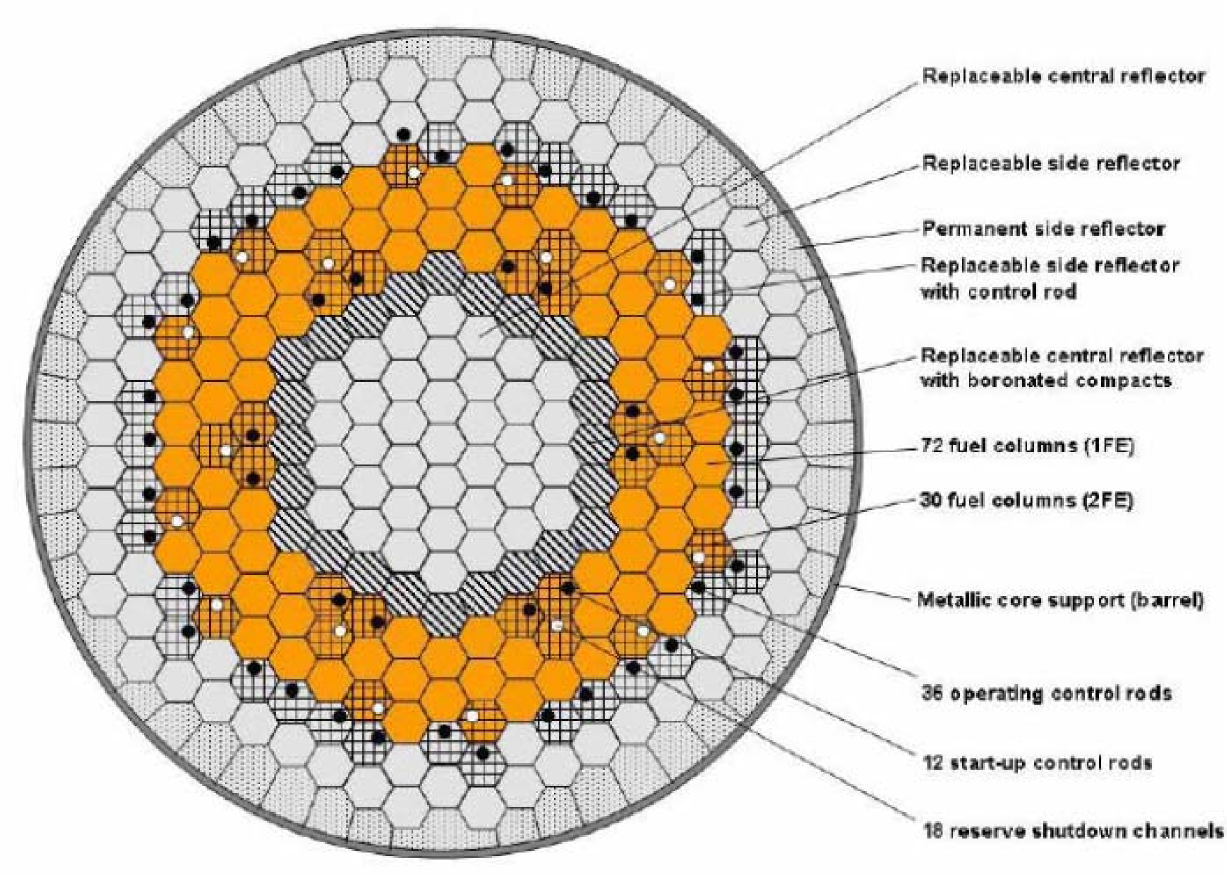

Figure 2.2-3 ANTARES Core Layout

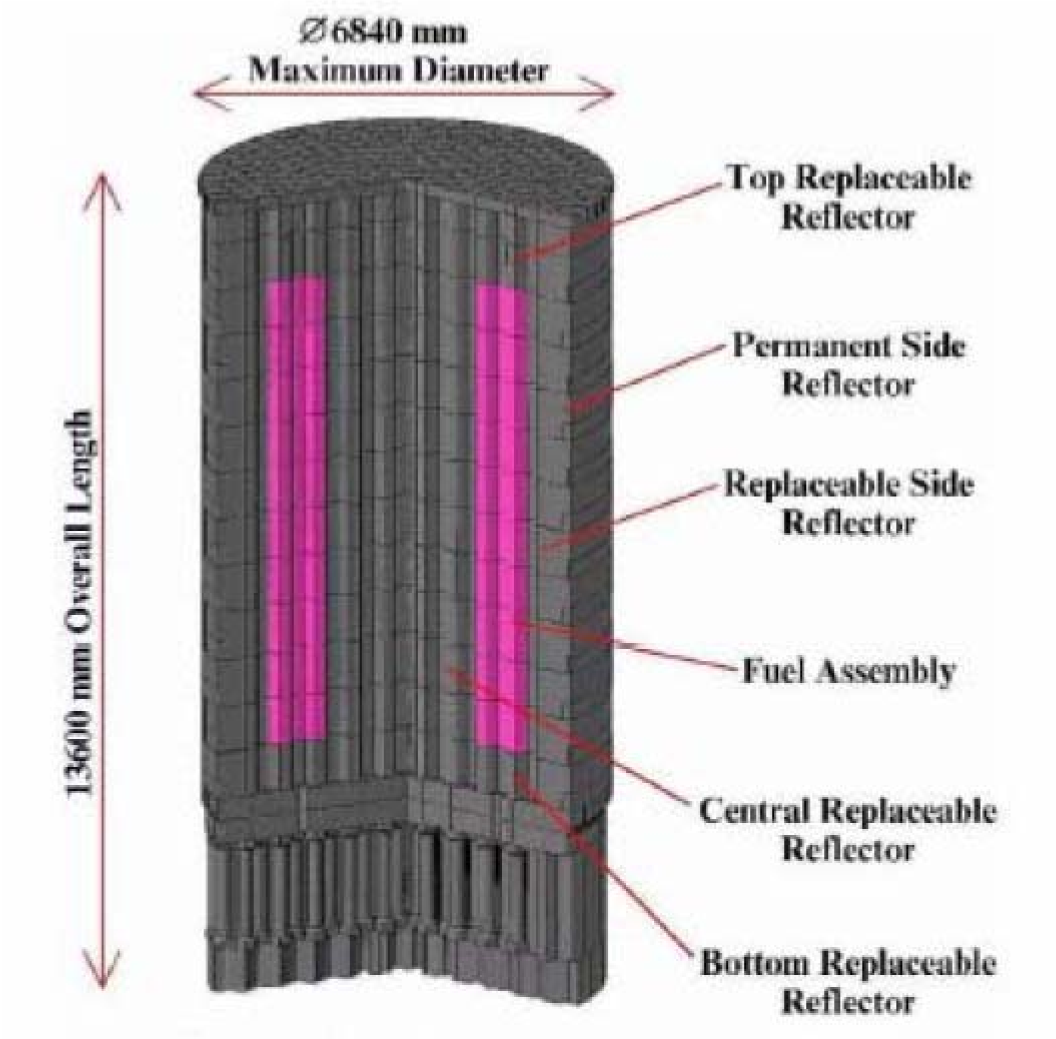

Figure 2.2-4 ANTARES Core Structure

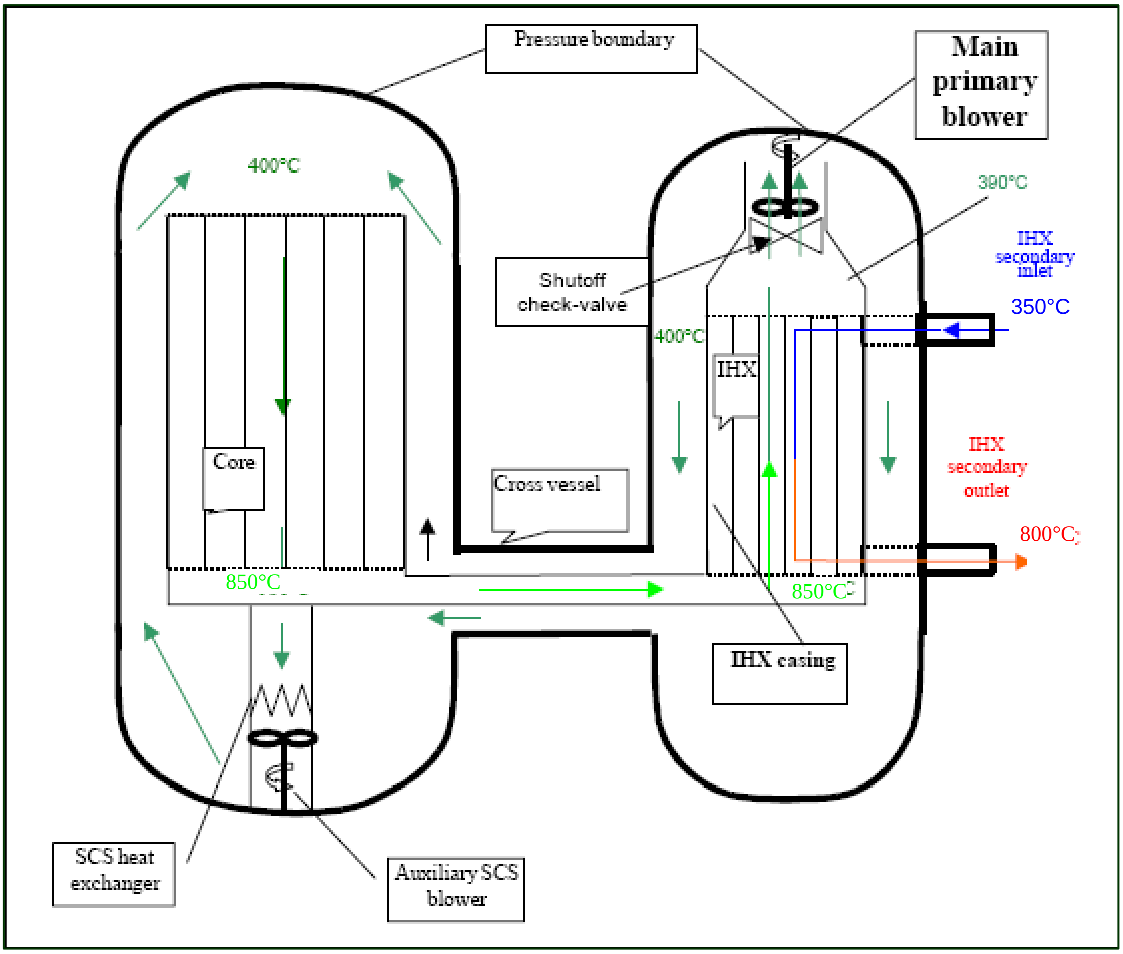

Figure 2.2-5 ANTARES Primary Pressure boundary

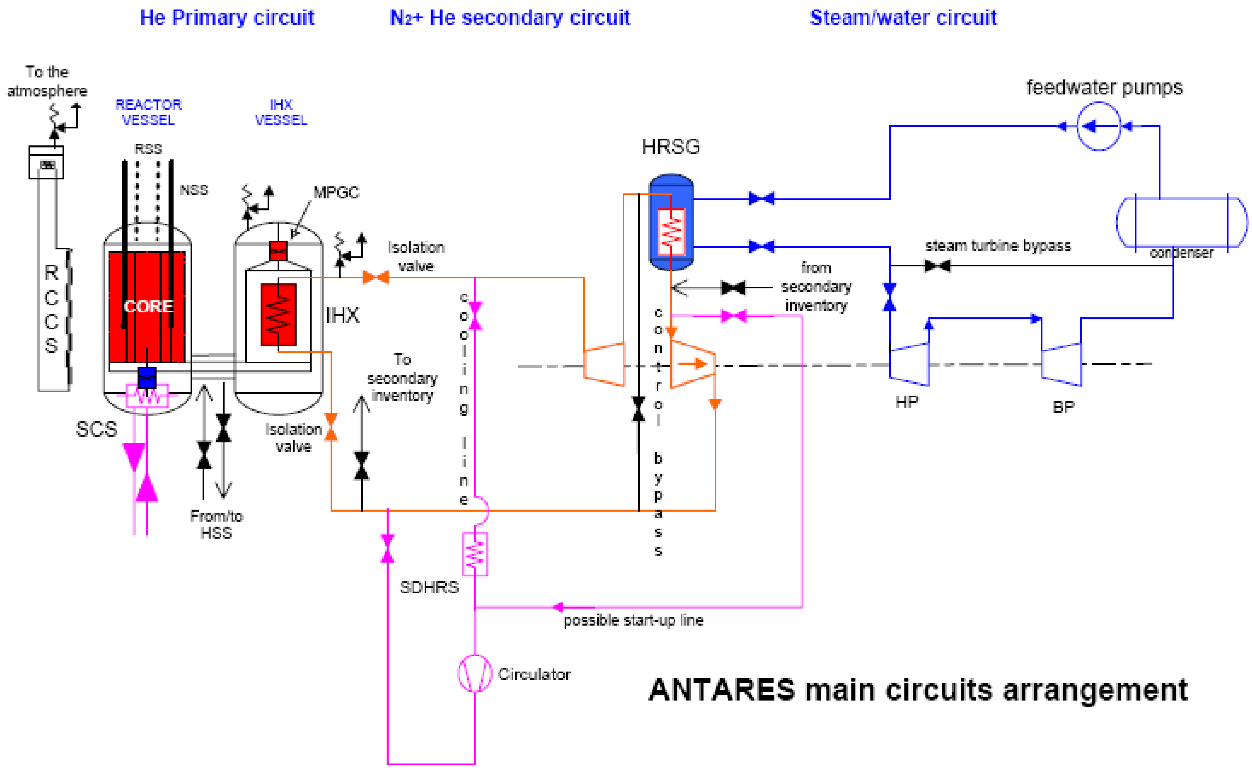

Figure 2.2-6 Architecture of main systems

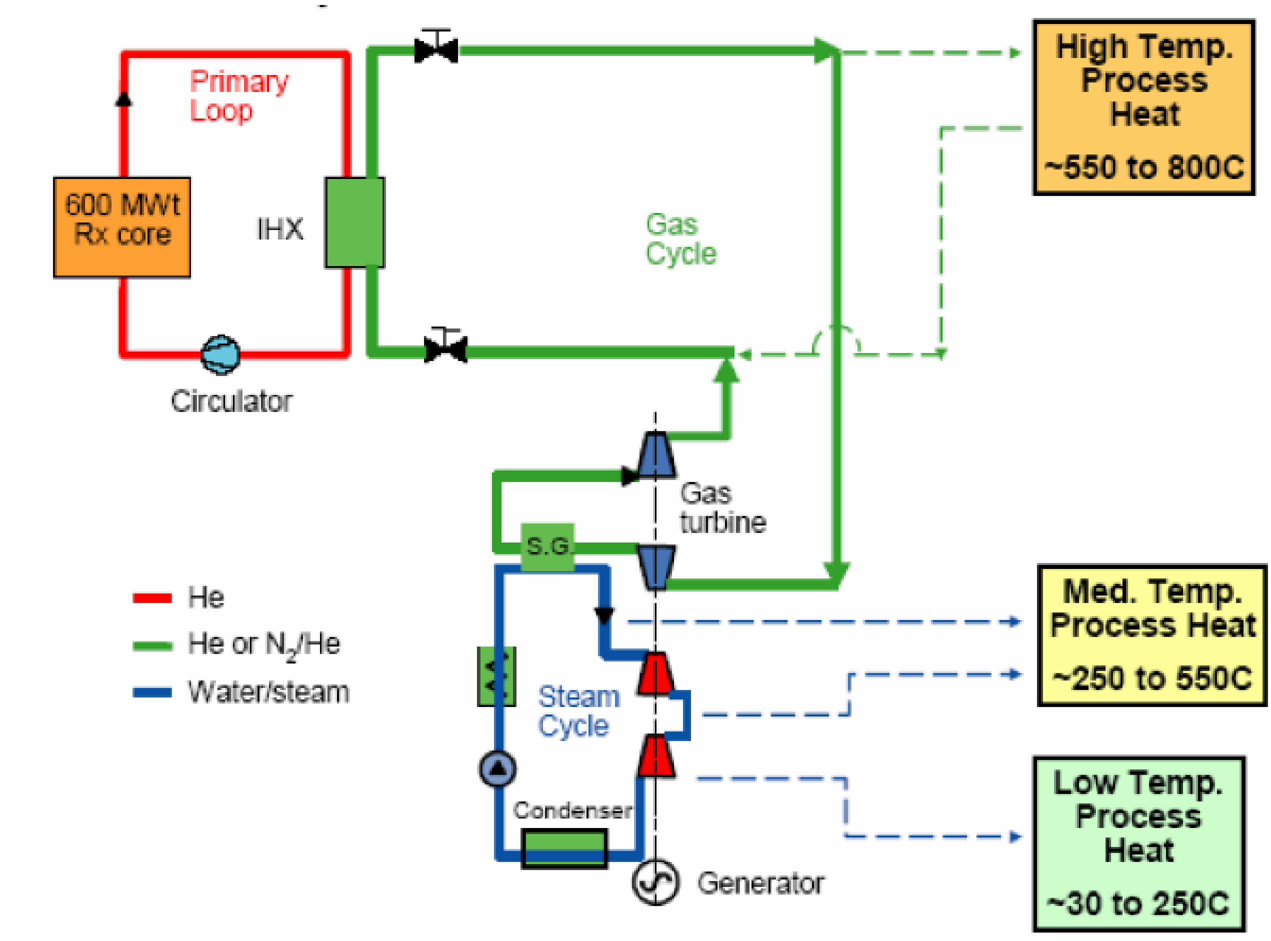

Figure 2.2-7 ANTARES Indirect Cycle Concept

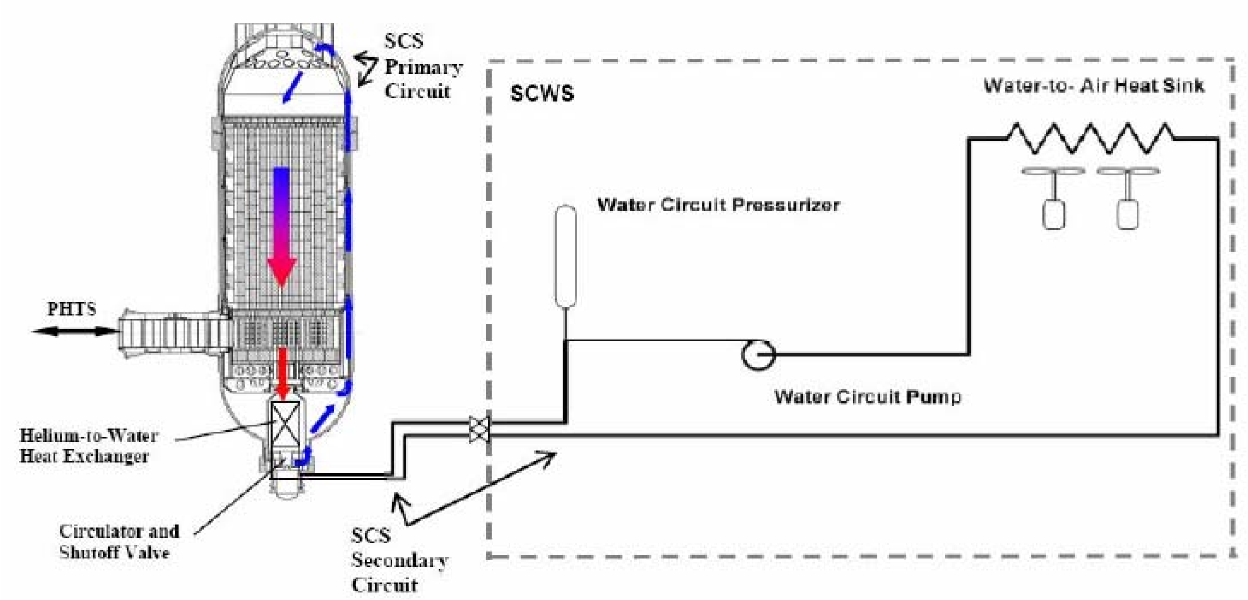

Figure 2.2-8 ANTARES Shutdown Cooling System

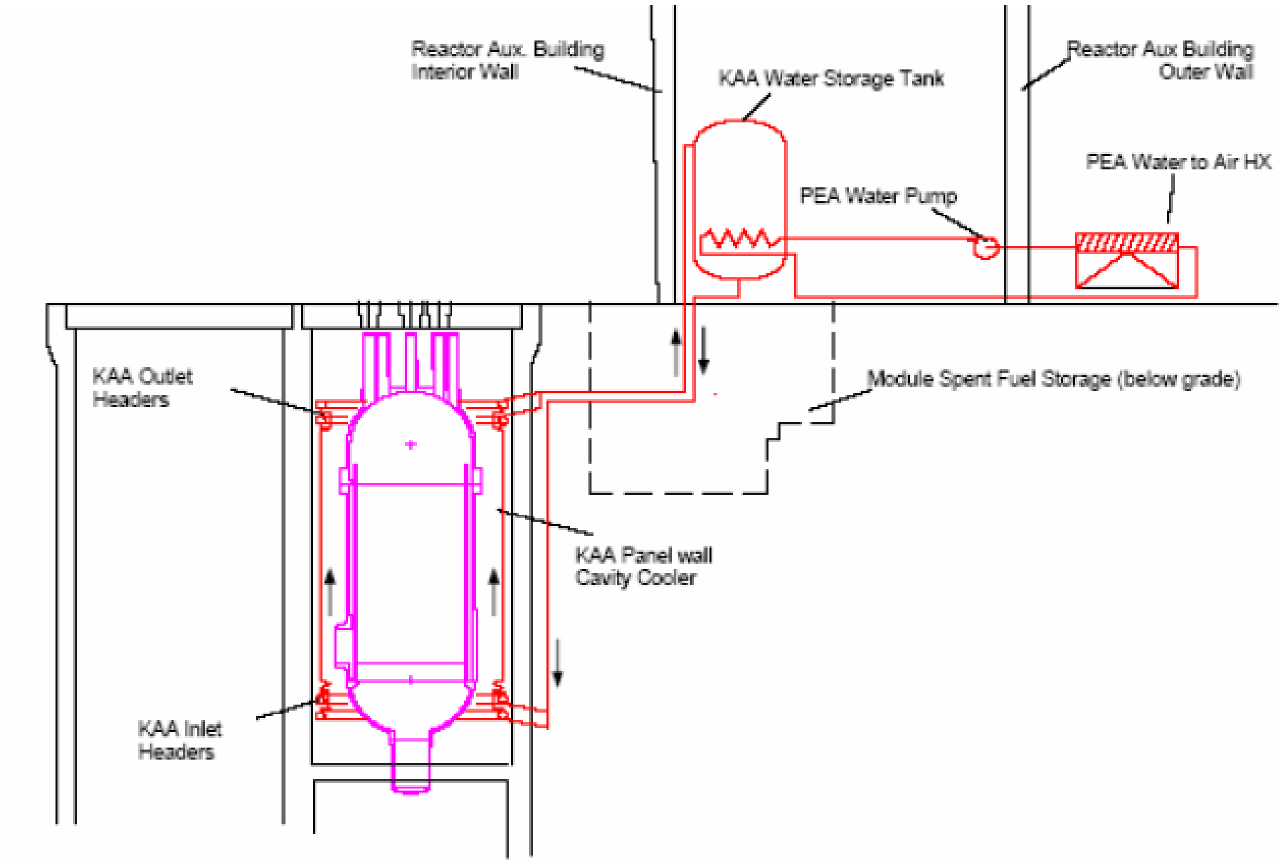

Figure 2.2-9 ANTARES Reactor Cavity Cooling System

2.3 MASTER LOGIC DIAGRAM

Two types of methods are possible to establish the list of the initiating faults: 1) a top down method (initiating faults are derived from the challenges on the sensitive confinement barriers), or 2) a bottom up method (initiating faults are derived from an exhaustive assessment of the potential plant system faults). A top down method is preferred at this first stage, because the main objective is to provide the convincing demonstration of severe core damage exclusion. The main sensitive barrier is the fuel particle. Besides, when a design is defined, a bottom up approach will confirm this list of initiating faults.

A master logic diagram (MLD) that is a top down method, is a helpful tool to guide the effort, aiming to identify the accident initiating events and to ensure completeness of the analysis.

The initiating faults to focus in this phase of the project are mainly those occurring during normal operation of the plant (including operation at power, reactor shutdown and refuelling) and the consequence of which are likely to challenge the fuel particles confinement capability. Of course the construction of the MLD is adapted to the level of information detail provided in the documents referenced in section 4.

The initiating faults can challenge the fuel particles directly or, indirectly, as a result of the evolution of any initiating event which challenges the primary system integrity first. The failure of the fuel particle confinement capability can be the result of several challenges, such as:

- a *mechanical challenge* - internal loading in the fuel particle, impact on the fuel element;
- a *chemical challenge* (corrosion) - Internal corrosion due to fission products, external corrosion (air, water), and
- a *thermal challenge* - the global (total core) or local (one or several fuel elements) thermal unbalance is likely to lead to challenge the fuel particle confinement capability (sub-function such as control of power (or reactivity), maintenance of cooling flow rate, availability of heat sink are required to control the fuel particle temperature).

The master logic diagram construction starts with the three fuel particles challenging mechanisms, as shown in Fig. 2.3-1, namely:

[1] MECHANICAL CHALLENGES [1.1];
[2] CHEMICAL CHALLENGES [1.2];
[3] THERMAL CHALLENGES[1.3].

In the following sections each main path will be developed through an MLD technique in order to find the elementary events that lead to the TOP EVENTS [1.1], [1.2] and [1.3].

The events which directly have the potential to challenge the second barrier (the Primary System) are here identified with their consequence on the fuel particles. The events which directly have the potential to challenge the third barrier (the Containment or Confinement) are not analyzed because it is out of the scope of this work. Also the family of events which directly release radioactive fluids outside containment is not included in the analysis (due to lack of information on the pertinent systems).

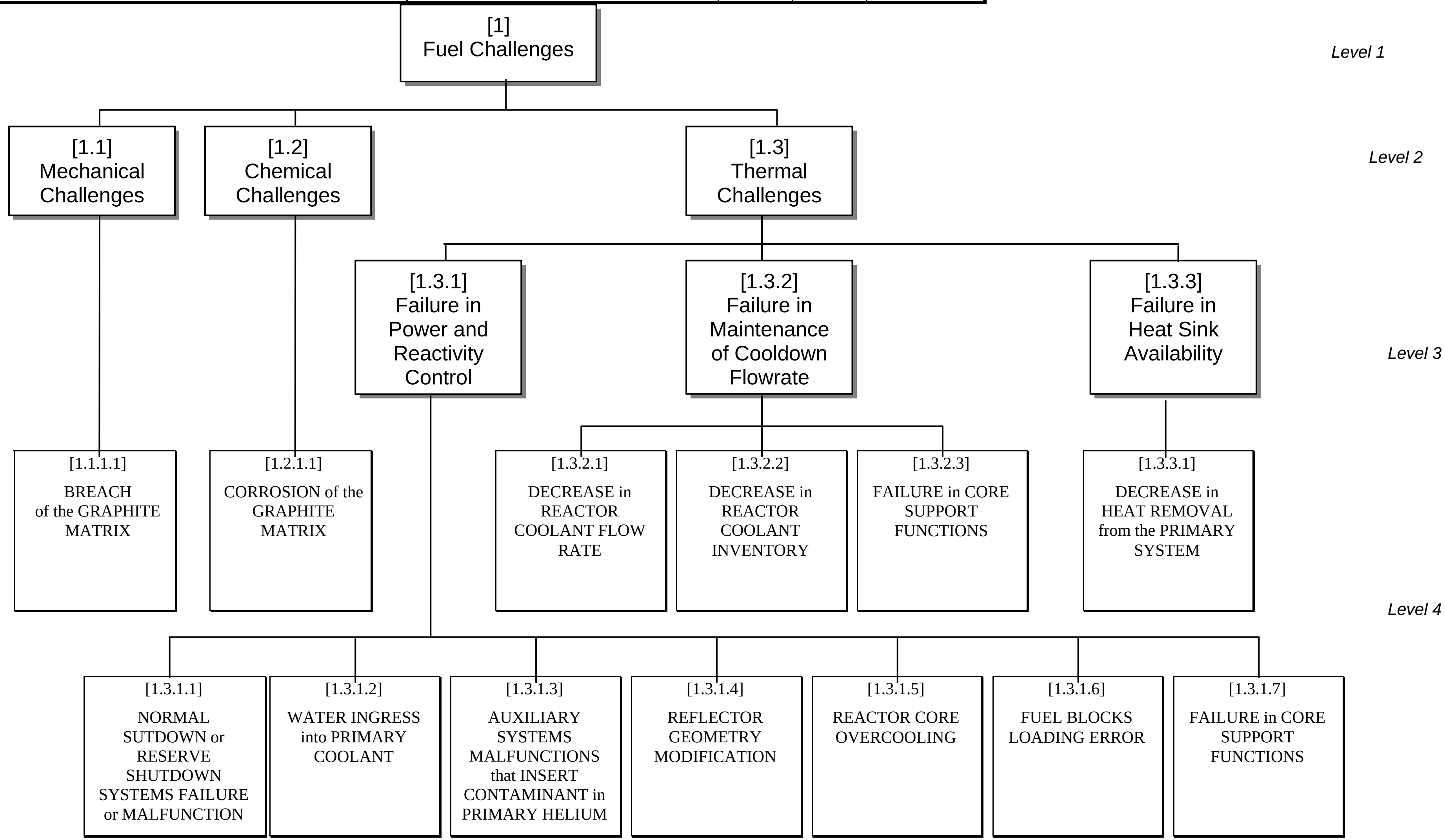

Figure 2.3-1 MLD development

2.3.1 Mechanical Challenges

The first path depicted in Fig. 2.3-1 aims to identify the initiating faults occurring during normal operation of the plant and the consequence of which are likely to mechanically challenge the fuel particles confinement capability like fuel impact or collision with objects and earthquake.

In particular the "MECHANICAL CHALLENGES [1.1]", which are a level 2 of the MLD, can be originated by the following event (level 4 of the MLD):

- BREACH of the GRAPHITE MATRIX [1.1.1.1].

level 4 events description

Event [1.1.1.1] “BREACH of the GRAPHITE MATRIX” refers to abnormal mechanical loads acting on the fuel blocks, such as from collision with objects during fuel handling (it has to be considered that also earthquake has the potential to breach the graphite, but since it is an external hazards it is not considered in this analysis).

The MLD branches detailing the event «BREACH of the GRAPHITE MATRIX», reported in Fig. 2.3.1-1, identify only one level 5 event that has the potential to challenge the fuel particles integrity.

level 5 events description

The event [1.1.1.1.1] “FUEL HANDLING ACCIDENT” can be postulated to occur inside the reactor vessel, inside the containment (confinement) or in the fuel handling area in the auxiliary building. The fuel handling accident is defined as the drop of a spent or fresh fuel block or the drop of a heavy load on it such that some or all compacts in the affected block can breach and release the contained radioactivity. If the spent or fresh fuel block drop over other fuel blocks (i.e fuel block drop inside the reactor vessel) also the hit blocks could be damaged.

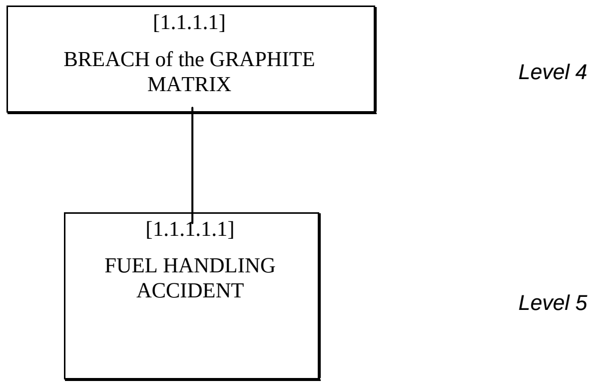

Figure 2.3.1-1 MLD development starting from the condition: Breach of the Graphite Matrix

2.3.2 Chemical Challenges

The second path depicted in Fig. 2.3-1 aims to identify the initiating faults occurring during normal operation of the plant and the consequence of which are likely to chemically challenge the fuel particles confinement capability like ingress or presence into the Helium of oxidant substances (i.e. oxygen or water).

In particular the "CHEMICAL CHALLENGES [1.2]", which are a level 2 of the MLD, can be originated by the following event (level 4 of the MLD):

- CORROSION of the GRAPHITE MATRIX [1.2.1.1].

level 4 events description

Event [1.2.1.1] “CORROSION of the GRAPHITE MATRIX” refers to contaminants, inadvertently present in Helium, or to foreign fluids, like water or air (oxygen) entered into the reactor coolant system, that when brought, by the primary coolant, in contact with the graphite matrix of the fuel or the reflector blocks can react with them. The MLD for these events, down to level 5, is shown in Figure 2.3.2-1.

The MLD branches detailing the event «CORROSION of the GRAPHITE MATRIX», reported in Fig. 2.3.2-1, identify three levels 5 events that have the potential to challenge the fuel particles integrity.

level 5 events description

The event [1.2.1.1.1] “ABNORMAL AMOUNT of CONTAMINANTS in PRIMARY HELIUM” can be attributed to malfunction or faults in the Helium Processing System, preventing it from providing the required degree of purification.

The event [1.2.1.1.2] “OXIGEN INGRESS into PRIMARY COOLANT” can be attributed to air ingress after the depressurization phase of any loss of coolant accident (see event [1.3.2.2.2]), to Helium Processing System malfunctions and auxiliary system failures or malfunctions (see event [1.3.1.3]).

The event [1.2.1.1.3] “WATER INGRESS into PRIMARY COOLANT” can be attributed to leakages or breaks occurring in the Helium-to-Water heat exchanger or connecting water piping of the SCS or of the MPGC cooling system. If the leakages or breaks occur with the primary system pressurized (according to the relative water/helium pressure evolution determined by the inflow of primary Helium) the event is the same as [1.3.2.2.2.3] considered in the primary system challenges. If the leakages or breaks occur during operation modes with primary coolant at near atmospheric pressure (refuelling, ISI or maintenance) the event is the same as [1.3.1.2].

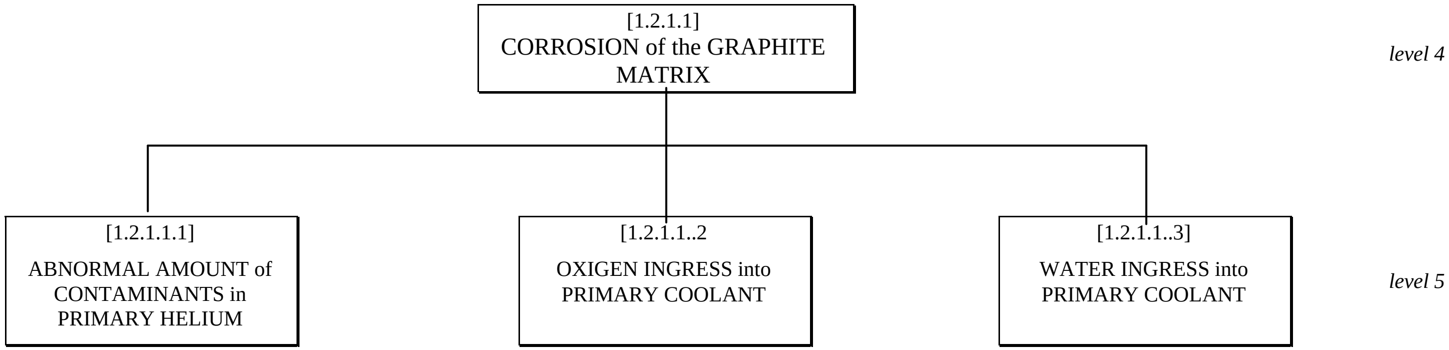

Figure 2.3.2-1 MLD development starting from the condition: Corrosion of the Graphite Matrix

2.3.3 Thermal Challenges

The third path depicted in Fig. 2.3-1 aims to identify the initiating faults occurring during normal operation of the plant and the consequence of which are likely to thermally challenge the fuel particles confinement capability like global (whole core) or local (one or several fuel blocks) thermal unbalance (abnormal fuel temperature). Failure to control the fuel particle temperature is due to the failure to control the power/reactivity and the failure to maintain the core flow rate or the heat sink.

In particular the "THERMAL CHALLENGES [1.3]", which are a level 2 of the MLD, can be originated by the following events (level 3 of the MLD):

- FAILURE in POWER and REACTIVITY CONTROL [1.3.1];
- FAILURE in MAINTENANCE of COOLDOWN FLOWRATE [1.3.2];
- FAILURE in HEAT SINK AVAILABILITY [1.3.3].

level 3 events description

Event [1.3.1] “FAILURE in POWER and REACTIVITY CONTROL“ refers to anomalies in the core reactivity or/and in the power generation leading, in particular, to an increase/decrease of the local or overall core power generation. The MLD for these events is shown in section 2.3.3.1.

Event [1.3.2] “FAILURE in MAINTENANCE of COOLDOWN FLOWRATE” refers to anomalies in the core flowrate leading, in particular, to a decrease of the local or overall core heat transfer. The MLD for these events is shown in section 2.3.3.2.

Event [1.3.3] “FAILURE in HEAT SINK AVAILABILITY” refers to anomalies in the core heat sink leading, in particular, to a decrease of the overall core heat removal. The MLD for these events is shown in section 2.3.3.3.

2.3.3.1 Failure in Power and Reactivity Control

A numbers of faults can be postulated that result in reactivity and power distribution anomalies. Control rod motion or ejection, absorber elements insertion, water or oxygen ingress, core geometry modification (up to the complete geometry loss) and core overcooling could cause reactivity changes. Power distribution changes could be caused by control rods motion or ejection, absorber elements insertion, core overcooling, core geometry loosing or by static causes such as fuel block misallocation.

The MLD branches detailing the event «FAILURE in POWER and REACTIVITY CONTROL», reported in Fig. 2.3-1, identify seven level 4 events that have the potential to challenge the fuel particles integrity:

- NORMAL SHUTDOWN or RESERVE SHUTDOWN SYSTEMS FAILURE or MALFUNCTION [1.3.1.1];
- WATER INGRESS into PRIMARY COOLANT [1.3.1.2];

- AUXILIARY SYSTEMS MALFUNCTIONS that INSERT CONTAMINANT in PRIMARY HELIUM [1.3.1.3];
- REFLECTOR GEOMETRY MODIFICATION [1.3.1.4];
- REACTOR CORE OVERCOOLING [1.3.1.5];
- FUEL BLOCKS LOADING ERROR [1.3.1.6];
- FAILURE in CORE SUPPORT FUNCTIONS [1.3.1.7].

level 4 events description

The event [1.3.1.1] “NORMAL SHUTDOWN or RESERVE SHUTDOWN SYSTEMS FAILURE OR MALFUNCTION” can be originated by the inadvertent withdrawal or ejection of one or several control rods, inadvertent insertion (drop) of one, several or all control rods and inadvertent actuation of one or more reserve shutdown units. The MLD for these events, down to level 5, is shown in section 2.3.3.1.1.

The event [1.3.1.2] “WATER INGRESS into PRIMARY COOLANT” can be attributed to failures occurring in the helium-water heat exchangers and connecting piping of the SCS, or of the MPGC cooling system, or failures or malfunctions occurring in the HPS auxiliary systems. During operation at power, the water pressure, in the SCS and in the MPGC cooling system is less than helium pressure while during refuelling, ISI and maintenance, helium pressure is close to atmospheric. Hence these operation modes certainly have the potential for water ingress into the primary coolant which, in turn, could cause positive reactivity insertion and graphite matrix corrosion. Water ingress, however, could also occur at power, depending on the relative water/Helium pressure evolution determined by the inflow of primary Helium in case of heat exchangers or connecting piping break; this event is considered among the event “Loss of Coolant Accident” (event [1.3.2.2.2.3]). The MLD for these events, down to level 5, will be shown in section 2.3.3.1.2.

The event [1.3.1.3] “AUXILIARY SYSTEMS MALFUNCTIONS that INSERT CONTAMINANT in PRIMARY HELIUM” can be attributed to failures or malfunctions in a auxiliary system causing the contamination of the helium with oxidant substances like oxygen. The MLD for these events will be shown in section 2.3.3.1.3.

The event [1.3.1.4] “REFLECTOR GEOMETRY MODIFICATION” can be attributed to overpressure transients, shock wave, drop of internals earthquake causing changes in reflector geometry. The event assumes the degradation or loss of the core support function that leads to changes in reflector geometry which have the potential to affect the reactivity and power control function (i. e. reactivity insertion in case of support internals failure due to withdrawal of inserted control rods; failure to insert control rods due to obstruction of control rods guide holes) and reflector cooling (i.e. degradation or loss of reflector coolability due to modified or destroyed coolant channels). An earthquake beyond the design basis is one of the possible causes of this event. The MLD for these events will be shown in section 2.3.3.1.4.

The event [1.3.1.5] “REACTOR CORE OVERCOOLING” can be attributed to faults that affect the heat removal process (such as malfunctions in the SCS or in the Secondary System that could result in a increase in heat removal from the reactor coolant system), to faults that affect the reactor coolant system flow rate (such as malfunctions in the MPGC primary flow rate control system that could result in an increase of the reactor coolant system flow rate) and to faults that affect the reactor coolant inventory (such as malfunctions in the HPS that could have the potential to increase the helium inventory in the reactor coolant system). The MLD for these events, down to level 7, will be shown in section 2.3.3.1.5.

The event [1.3.1.6] “FUEL BLOCKS LOADING ERROR” can be originated by loading errors arising either from the inadvertent loading of one or more fuel blocks in incorrect positions or loading of fuel blocks with incorrect compositions. The MLD for these events, down to level 5, will be shown in section 2.3.3.1.6.

The event [1.3.1.7] “FAILURE in CORE SUPPORT FUNCTIONS” can be attributed to mechanical failures in the reactor internals, to internal hazards (like missile due to rotating machines), drop of loads (drop of heavy objects during fuel and component handling) and reactor vessel failure (such as loss of the pressure boundary, reactor vessel support failure or failure to cool the reactor cavity). The MLD for these events, down to level 8, will be shown in section 2.3.3.1.7.

2.3.3.1.1 Normal Shutdown System or Reserve Shutdown System Malfunction or Failure

The MLD branches detailing the event «NORMAL SHUTDOWN or RESERVE SHUTDOWN SYSTEMS MALFUNCTION OR FAILURE», reported in Fig. 2.3.3-1, identify four level 5 events that have the potential to challenge the fuel particles integrity.

level 5 events description

The event [1.3.1.1.1] “INADVERTENT CONTROL ROD ASSEMBLY WITHDRAWAL” can be caused by a malfunction of the reactor control or rod control systems and can occur with the reactor subcritical, at hot zero power, or at power. It results in an addition of reactivity to the reactor core caused by the withdrawal of one or several control rod assemblies and generates a power excursion.

The event [1.3.1.1.2] “CONTROL ROD ASSEMBLY EJECTION” is originated by the mechanical failure of a control rod mechanism pressure housing, resulting in the ejection of a control rod assembly and drive shaft. The consequence of this mechanical failure is a rapid positive reactivity insertion generating an adverse local core power distribution

The event [1.3.1.1.3] “CONTROL ROD ASSEMBLY DROP” is originated by an electrical or mechanical failure in the plant control system that results in dropping one or more control rod assemblies.

The event [1.3.1.1.4] “INADVERTENT RESERVE SHUTDOWN UNIT ACTUATION” is originated by an electrical or mechanical failure in the plant control system that results in dropping one or more reserve shutdown units.

2.3.3.1.2 Water Ingress into Primary Coolant

The MLD branches detailing the event «WATER INGRESS into PRIMARY COOLANT», reported in Fig. 2.3.3-2, identify three level 5 events that have the potential to challenge the fuel particles integrity.

level 5 events description

The event [1.3.1.2.1] “SCS MECHANICAL FAILURES INSIDE THE REACTOR VESSEL” is defined as a postulated break that occurs, in one heat exchanger module or in the connecting piping inside the reactor system, during operation with primary system at low pressure.

The event [1.3.1.2.2] “AUXILIARY SYSTEM FAILURES or MALFUNCTIONS” is defined as a postulated break or malfunctions in a auxiliary system (mainly in the HPS auxiliary systems) causing the contamination of the primary helium with water.

The event [1.3.1.2.3] “MPGC COOLING SYSTEM MECHANICAL FAILURES INSIDE THE IHX VESSEL” is defined as a postulated break that occurs, in the MPGC heat exchanger module or in the connecting piping inside the reactor system. During operation at high pressure (operation at power) water ingress can occur after the helium and water pressure equalises. During operation at low pressure water ingress occurs immediately after the break.

2.3.3.1.3 Auxiliary Systems Malfunctions that Insert Contaminant in Primary Helium

The MLD branches detailing the event «AUXILIARY SYSTEMS MALFUNCTIONS that INSERT CONTAMINANT in PRIMARY HELIUM», is reported in Fig. 2.3.3-3. No level 5 events, which have the potential to challenge the fuel particles integrity, have been individuated for lack of detailed information on the auxiliary systems.

2.3.3.1.4 Reflector Geometry Modification

The MLD branches detailing the event «REFLECTOR GEOMETRY MODIFICATION», reported in Fig. 2.3.3-4, no level 5 events have been identified.

2.3.3.1.5 Reactor Core Overcooling

The MLD branches detailing the event «REACTOR CORE OVERCOOLING», reported in Fig. 2.3.3-5, identify three level 5 events that have the potential to challenge the fuel particles integrity.

- INCREASE in HEAT REMOVAL from the PRIMARY SYSTEM [1.3.1.5.1];

- INCREASE in REACTOR COOLANT FLOW RATE [1.3.1.5.2];
- INCREASE in REACTOR COOLANT INVENTORY[1.3.1.5.3].

level 5 events description

Event [1.3.1.5.1] "INCREASE in HEAT REMOVAL from the PRIMARY SYSTEM" refers to anomalies in the heat removal process that could result in an increase in heat removal from the reactor coolant system. Due to the negative temperature reactivity coefficient, such faults increase the core reactivity and core power.

Event [1.3.1.5.2] "INCREASE in REACTOR COOLANT FLOW RATE" refers to faults in the main primary gas circulator that could result in an increase of the reactor coolant system flow rate.

Event [1.3.1.5.3] "INCREASE in REACTOR COOLANT INVENTORY" refers to faults in the HPS that could have the potential to increase the helium inventory in the reactor coolant system.

level 6 events description

The event [1.3.1.5.1.1] "SECONDARY SYSTEM MALFUNCTION that RESULTS in an INCREASE of HEAT REMOVAL" refers to any malfunction in the secondary system that could increase the heat removed from the primary system (turbine overran, decrease in secondary system temperature, increase in process heat consumption, etc. The initiating events causing this malfunction can be more precisely individuated when data on the secondary system architecture is available).

The event [1.3.1.5.1.2] "SCS MALFUNCTION that RESULTS in an INCREASE of HEAT REMOVAL" refers to an inadvertent actuation of the SCS that could be caused by operator error, by false actuation signal or by malfunction.

The event [1.3.1.5.1.3] "SDRHS MALFUNCTION that RESULTS in an INCREASE of HEAT REMOVAL" refers to an inadvertent actuation of the SDRHS that could be caused by operator error, by false actuation signal or by malfunction.

The event [1.3.1.5.2.1] "MPGC MALFUNCTION that RESULTS in an INCREASE of REACTOR COOLANT FLOWRATE" can be attributed to malfunction in the MPGC flowrate control system causing the primary coolant flow rate increases up to its maximum value.

The event [1.3.1.5.3.1] "HELIUM PROCESSING SYSTEM MALFANCTION that INCREASE REACTOR COOLANT INVENTORY" can be attributed to malfunctions in the HPS that could have the potential to increase the helium inventory in the reactor coolant system. Malfunctions that cause a feeding and bleeding mismatch such as the spurious operation of the HPS compressors in the feeding path or the inadvertent closure (valve) of the letdown path. Spurious transfer compressor actuation (inadvertent helium filling operation) at power or primary helium pressure (mass) control system misoperation. An operator error, a false electrical actuation signal, or a valve malfunction would cause this event.

level 7 events description

The event [1.3.1.5.1.1.1] “LOSS of EXTERNAL ELECTRICAL LOAD” refers to a load loss on the plant that can result from loss of electrical load due to an electrical system disturbance. The AC power remains available to operate plant components such as pumps, motor operated valves, automated process control system, etc.; as a result, the standby onsite diesel generators do not function for this event.

The event [1.3.1.5.1.1.2] “GENERATOR FAILURE CAUSING the LOSS of the LOAD (load ejection)” refers to generator mechanical failures such as turbine-generator shaft seizure that causes the instantaneously turbo-machine disconnection from the generator.

2.3.3.1.6 Fuel Blocks Loading Error

The MLD branches detailing the event «FUEL BLOCKS LOADING ERROR», reported in Fig. 2.3.3-6, identify two level 5 events that have the potential to challenge the fuel particles integrity.

level 5 events description

The event [1.3.1.6.1] “FUEL BLOCK LOADED in an INCORRECT POSITION” is defined as a loading error arising from the inadvertent loading of one or more fuel blocks having fuel particles with wrong enrichment in wrong positions. This leads to locally increased heat fluxes if the error results in placing higher enrichment fuel in core positions calling for fuel of lower enrichment. Also included among possible core loading errors is the inadvertent loading of one or more fuel blocks without rods guide holes in a position designed for rods insertion or the inadvertent loading of one or more fuel blocks with burnable poison compacts in an incorrect core position.

The event [1.3.1.6.2] “FUEL BLOCK LOADED with INCORRECT COMPOSITION” is defined as a loading error arising from the inadvertent loading of one or more fuel blocks having one or more fuel compacts of the wrong enrichment, or having all the fuel particles of the wrong enrichment. This can cause power shapes more peaked than those calculated with the correct enrichments.

2.3.3.1.7 Failure in Core Support Function

The MLD branches detailing the event «FAILURE in CORE SUPPORT FUNCTION», reported in Figures 2.3.3-7 and 2.3.3-8, identify four level 5 events that have the potential to challenge the fuel particles integrity.

- INTERNALS MECHANICAL FAILURE [1.3.1.7.1];

- INTERNAL HAZARDS [1.3.1.7.2];
- LOAD DROP[1.3.1.7.3];
- REACTOR VESSEL FAILURE [1.3.1.7.4].

level 5 events description

The event [1.3.1.7.1] “INTERNALS MECHANICAL FAILURE” refers to breaks or ruptures of the core internals due to excessive loads (thermal or/and pressure loads, earthquake loads) or corrosion of structural components that can challenge the core support functions.

The event [1.3.1.7.2] “INTERNAL HAZARDS that can CHALLENGE the CORE SUPPORT FUNCTION” can be attributed to ruptures in the MPGC components (blades, vanes, compressor wheels and shaft) to rupture in rotating machine or pressurized tanks that generate missiles or projectiles that can challenge directly or indirectly (via reactor vessel or reactor vessel support challenge) the core support functions.

The event [1.3.1.7.3] “LOAD DROP that can CHALLENGE the CORE SUPPORT FUNCTION” can be attributed to drop of heavy object during fuel and component handling that can challenge directly or indirectly (via reactor vessel challenge) the core support functions.

The event [1.3.1.7.4] “REACTOR VESSEL FAILURE” can be attributed to mechanical failures in the reactor internals, in the reactor vessel support, in the cross duct and in general in the pressure boundary, to internal hazards (like MPGC missile), to drop of loads (drop of heavy objects during fuel and component handling) and to failures or malfunctions in the RCCS.

level 6 events description

The event [1.3.1.7.4.1] “REACTOR VESSEL MECHANICAL FAILURE” refers to breaks or ruptures of the reactor vessel due to excessive loads (thermal or/and pressure loads, earthquake) or corrosion. Excessive thermal loads can result from lack of MPGC trip and MPGC shutoff valve closure in case of loss of heat sink.

The event [1.3.1.7.4.2] “INTERNAL HAZARDS that can CHALLENGE the REACTOR VESSEL” can be attributed to ruptures in the MPGC components (blades, vanes, compressor wheels and shaft) to rupture in rotating machine or pressurized tanks that generate missiles or projectiles that can challenge directly or indirectly (via reactor vessel support challenge) the reactor vessel.

The event [1.3.1.7.4.3] “LOAD DROP that can CHALLENGE the REACTOR VESSEL” can be attributed to load drop during fuel and component handling that can challenge the reactor vessel.

The event [1.3.1.7.4.4] “CROSS DUCT FAILURE” refers to break or ruptures due to mechanical failure, internal hazards, load drop or fault in the pressure boundary like ruptures in the MPGC components, excessive loads (thermal or/and pressure loads, earthquake) or corrosion and load drop during fuel and component handling.

The event [1.3.1.7.4.5] “PRESSURE BOUNDARY CHALLENGE” refers to faults that have the potential to affect the integrity of the second physical barrier to the release of the radioactive fission products to the environmental, namely the primary system. The MLD detailing this events is reported in Fig. 2.3.3-8.

The event [1.3.1.7.4.6] “REACTOR VESSEL SUPPORT FAILURE” refers to break or ruptures due to mechanical failure or internal hazards.

The event [1.3.1.7.4.7] “RCCS FAILURE” refers to failure in one RCCS heat removal train or control system malfunctions that decrease the RCCS heat removal capacity. These events result in a reactor cavity heat-up and they have no a direct impact on the core but on structures essential to fulfill the core support function (they do not have as a direct consequence a reactor trip, but rather they require a forced reactor shutdown). Hence for these reasons these events are included in the list and their consequences have to be assessed.

level 7 events description

The event [1.3.1.7.4.5.1] “THERMAL CHALLENGE of the PRESSURE BOUNDARY” refers to faults that degrade the core heat removal process such as decreasing decrease in heat removal from primary system, in reactor coolant flow rate and in reactor coolant inventory.

The event [1.3.1.7.4.5.2] “PRESSURE CHALLENGE of the PRESSURE BOUNDARY” refers to faults that increase the primary system pressure or the differential pressure between primary system components.

The event [1.3.1.7.4.6.1] “REACTOR VESSEL SUPPORT MECHANICAL FAILURE” refers to breaks or ruptures of the reactor vessel support due to excessive loads (thermal loads, earthquake) or corrosion.

The event [1.3.1.7.4.6.2] “INTERNAL HAZARDS that can CHALLENGE the REACTOR VESSEL SUPPORT” can be attributed to ruptures in the MPGC components (blades, vanes, compressor wheels and shaft) to rupture in rotating machine or pressurized tanks that generate missiles or projectiles that can challenge the reactor vessel support.

The event [1.3.1.7.4.7.1] “RCCS PIPE BREAK” can be attributed to break or leakage on a pipe, on the surface cooler and on the heat exchanger/evaporator of one RCCS train.

The event [1.3.1.7.4.7.2] “RCCS RE-CIRCULATING WATER LOOP FAILURE” refers to failures in the RCCS re-circulating water loop such as pipe breaks, pump failures (mechanical or electrical), inadvertent isolation valves closure, heat exchanger flow blockage, water flow control system malfunction, etc.

level 8 events description

Event [1.3.1.7.4.5.1.1] “DECREASE in HEAT REMOVAL from the PRIMARY SYSTEM“ refers to the same [1.3.3.1] event. The MLD for these events is shown in section 2.3.3.3.1.

Event [1.3.1.7.4.5.1.2] “DECREASE in REACTOR COOLANT FLOW RATE“ refers to the same [1.3.2.1] event. The MLD for these events is shown in section 2.3.3.2.1.

Event [1.3.1.7.4.5.1.3] “DECREASE in REACTOR COOLANT INVENTORY“ refers to the same [1.3.2.2] event. The MLD for these events is shown in section 2.3.3.2.2.

Event [1.3.1.7.4.5.2.1] “INCREASE in REACTOR COOLANT INVENTORY“ refers to the same [1.3.1.5.3] event. The MLD for these events is shown in section 2.3.3.1.5.

Event [1.3.1.7.4.5.2.2] “BUCKLING RISK” refers to abnormal increase of the differential pressure across the components inside the pressure boundary or the vessels (Reactor Pressure Vessel, Cross Vessel and IHX Vessel).

2.3.3.2 Failure in Maintenance of Cooldown Flowrate

A numbers of faults can be postulated that result in a failure in maintenance of the cooldown flow rate (such as decreasing of the reactor coolant flow rate or inventory and loosing of core support functions) causing the core heat removal process degradation.

The MLD branches detailing the event «FAILURE in MAINTENANCE of COOLDOWN FLOWRATE», reported in Fig. 2.3-1, identify three level 4 events that have the potential to challenge the fuel particles integrity:

- DECREASE in REACTOR COOLANT FLOWRATE [1.3.2.1];
- DECREASE in REACTOR COOLANT INVENTORY [1.3.2.2];
- FAILURE in CORE SUPPORT FUNCTIONS [1.3.2.3].

level 4 events description

Event [1.3.2.1] “DECREASE in REACTOR COOLANT FLOW RATE“ refers to failures in the MPGC, to obstruction of the primary coolant cross sectional flow area or to malfunction in the MPGC shutoff valve that could result in a decrease of the reactor coolant system flow rate. The MLD for these events will be shown in section 2.3.3.2.1.

Event [1.3.2.2] “DECREASE in REACTOR COOLANT INVENTORY“ refers to a loss of coolant accident (LOCA) that is the result of a pipe rupture or vessel leakage of the reactor coolant system pressure boundary or to faults in the HSS that could have the potential to decrease the helium inventory in the reactor coolant system. The MLD for these events will be shown in section 2.3.3.2.2.

The event [1.3.2.3] “FAILURE in CORE SUPPORT FUNCTIONS” refers to the same event [1.3.1.7] described in section 2.3.3.1.7.

2.3.3.2.1 Decrease in Reactor Coolant Flowrate

The MLD branches detail the event «DECREASE in REACTOR COOLANT FLOW RATE», reported in Fig. 2.3.3-9, identify three level 5 scenarios that have the potential to challenge the primary system and the fuel particles integrity.

- PIPE BREAK INSIDE the PRESSURE BOUNDARY [1.3.2.1.1];
- PRIMARY COOLANT SYSTEM FLOW BLOCKAGE [1.3.2.1.2];
- MPGC MALFUNCTION or FAILURE [1.3.2.1.3];

level 5 events description

The event [1.3.2.1.1] “PIPE BREAK inside the PRESSURE BOUNDARY” refers to a break of a helium pipe or a component located inside the pressure boundary (including the hot duct assembly inside the cross-vessel) that cause an increase of the core bypass flowrate.

The event [1.3.2.1.2] “PRIMARY COOLANT SYSTEM FLOW BLOCKAGE” assumes an obstruction of the cross sectional flow area of ducts or components of the primary coolant system, the most likely location being the inlet regions of the IHX and the Fuel Blocks. An inadvertent MPGC shutoff valve closure can also cause this event. The flow blockage may results by foreign part inadvertent introduced into primary coolant during refueling, ISI or maintenance, by graphite, metallic or insulation material detached from damaged systems/components inside the vessel system.

The event [1.3.2.1.3] “MPGC MALFUNCTION or FAILURE” can be attributed to ruptures in the MPGC components. Major failure in the MPGC could involve blades, vanes, compressor wheels and shaft. These failures have also the potential to generate missiles or projectiles that can exited the MPGC casings. MPGC damage can be also initiate by metallurgical defects, by manufacturing defects and by foreign objects (external part inadvertent introduced into primary coolant during refueling, ISI or maintenance, or graphite, metallic and insulation material detached from damaged systems/components inside the vessel system).

level 6 events description

Event [1.3.2.1.2.1] “DECREASE OF HEAT REMOVAL FROM THE FUEL BLOCKS” refers to anomalies in the heat removal process from the fuel leading to a less effective heat extraction from the fuel. These events affect heat removal in a single coolant channel of a fuel blocks column, all coolant channels of one fuel blocks column or all the coolant channels of several fuel blocks column and, as such, may have a significant impact on the affected fuel blocks column(s) while the effect on the core and on the primary system is negligible. Also core geometry modifications that endanger the core cooling are ascribed to these events. The event assumes an obstruction of the cross sectional flow area of one or all coolant channels of one column of fuel blocks, the most likely location being the inlet region. The obstruction causes a coolant flow reduction and then an increase in fuel temperature. The flow blockage may result from foreign objects inadvertently introduced into the primary system during refuelling, ISI or maintenance, by loose graphite, metallic or insulation material resulting from damaged systems/components inside the vessel system, by drop of internals on the upper part of the core or by fuel blocks manufacturing defects. Moreover flow blockage may result also from the degradation or loss of the core support function that leads to changes in core geometry which affect the reactivity and power control function (i. e. reactivity insertion in case of support internals failure due to withdrawal of inserted control rods; failure to insert control rods due to obstruction of control rods guide holes) and fuel cooling (i.e. degradation or loss of core coolability due to modified or destroyed coolant channels; an earthquake beyond the design basis is one of the possible causes of this event).

The event [1.3.2.1.2.2] “FLOW BLOCKAGE in the IHX” refers to obstruction of part of the cross sectional flow area of the IHX due to foreign objects inadvertently introduced into the primary system during refuelling, ISI or maintenance, by loose graphite, metallic or insulation material resulting from damaged systems/components inside the vessel system.

The event [1.3.2.1.2.3] “INADVERTENT MPGC SHUTOFF VALVE CLOSURE” refers to an inadvertent partial or complete MPGC shutoff valve closure.

The event [1.3.2.1.3.1] “PRIMARY COOLANT FLOWRATE CONTROL SYSTEM MALFUNCTION that RESULT in a DECREASE of FLOWRATE” refers to a failure o malfunction in the control system of the MPGC electric motor drive that cause the primary coolant flowrate reduction.

The event [1.3.2.1.3.2] “SPURIOUS TRIP of MPGC” refers to a failure in the MPGC motor electric supply system or to a complete loss of electric power supplied to the MPGC motor.

The event [1.3.2.1.3.3] “LOSS of AC POWER to the PLANT AUXILIARIES” refers to a station blackout. The loss of power to the plant auxiliaries is caused by a complete loss of the offsite grid. The onsite standby AC power system (if assumed available) could remain available but is not credited to mitigate the accident.

The event [1.3.2.1.3.4] “FAILURE of a MPGC BLADE” refers to a failure in one or more compressor blades that shear off from the rotor. Failure of one blade has also the potential to damage other blades linked to the same wheel or even broken the wheel. Blades failure can also generates missiles or projectiles that can exit the compressor casing.

The event [1.3.2.1.3.5] “FAILURE of a MPGC WHEEL” refers to a rupture of one compressor wheel. Pieces of which can also exit the casing.

The event [1.3.2.1.3.6] “MPGC SHAFT BREAK” is postulated as an instantaneous rupture of the MPGC shaft. Depending on the break position, the compressor rotor would drop down and can be ejected. The blades can fracture the casing allowing fragments of both blades and casing to come out the MPGC casings. Flow through reactor coolant system is rapidly reduced through the compressor coastdown.

The event [1.3.2.1.3.7] “MPGC SHAFT SEIZURE” refers to a locked rotor. The accident postulated is an instantaneous seizure of the MPGC rotor. Flow through reactor coolant system is postulated to become immediately zero.

2.3.3.2.2 Decrease in Reactor Coolant Inventory

The MLD branches detail the event «DECREASE in REACTOR COOLANT INVENTORY», reported in Fig. 2.3.3-10, identify two level 5 scenarios that have the potential to challenge the primary system and the fuel particles integrity.

- HELIUM PROCESSING SYSTEM MALFUNCTION that DECREASE REACTOR COOLANT INVENTORY [1.3.2.2.1];
- LOSS of COOLANT ACCIDENT [1.3.2.2.2];

level 5 events description

The event [1.3.2.2.1] “HELIUM PROCESSING SYSTEM MALFUNCTION that DECREASE REACTOR COOLANT INVENTORY” can be attributed to malfunctions in the HPS that could have the potential to decrease the helium inventory in the reactor coolant system like: feeding and bleeding mismatch (due to malfunction of the extraction path or the inadvertent closure (valve) of the feeding path) or spurious helium draining operation, (inadvertent valve opening or transfer compressor actuation) at power or primary helium pressure (mass) control system malfunction. An operator error, a false electrical actuation signal, or a valve malfunction would cause this event..

The event [1.3.2.2.2] “LOSS of COOLANT ACCIDENT (LOCA)” can be attributed to break on a helium pipe within the reactor coolant pressure boundary, to leakage in the vessel system, to ruptures in the helium/water heat exchanger and connecting pipes of the SCS or of the MPGC cooling system, to rupture in the IHX (tube or plate) or their connecting pipes inside the pressure boundary, and to inadvertent safety valve opening. Air ingress into primary coolant can occur after the depressurization phase following a LOCA, air ingress causes fuel graphite matrix corrosion.

level 6 events description

The event [1.3.2.2.2.1] “LOSS of COOLANT ACCIDENT RESULTING of a SPECTRUM of POSTULATED PIPING BREAKS within the REACTOR COOLANT PRESSURE BOUNDARY” is the result of a pipe rupture of the reactor coolant system pressure boundary. A major pipe break (large break) is defined as a total cross-sectional area rupture of a large pipe (double end pipe break). This event is considered a limiting fault because it is not expected to occur during the lifetime of the plant but is postulated as a conservative design basis. A minor pipe break (small break) is defined as a rupture of small pipe of the reactor coolant pressure boundary in which the normally operating helium service system flow is not sufficient to sustain the primary coolant pressure. This is considered an infrequent fault that may occur during the life of the plant.

The event [1.3.2.2.2.2] “LOSS of COOLANT ACCIDENT RESULTING from VESSELS LEAKAGE” is the result of cracks in the vessels with the resulting leakage flow higher than the normally operating helium service system flow that is not sufficient to sustain the primary coolant pressure.

The event [1.3.2.2.2.3] "LOSS of COOLANT ACCIDENT RESULTING from HELIUM-WATER HEAT EXCHANGERS RUPTURES" is the result of rupture in the helium/water heat exchanger unit of the SCS and of the MPGC cooling system and in the water ducts inside the vessel system. Water ingress into primary coolant can occur during the consequent depressurization transient when the helium pressure equalizes the water pressure, water ingress causes positive reactivity insertion and fuel graphite matrix corrosion.

The event [1.3.2.2.2.4] "LOSS of COOLANT ACCIDENT RESULTING from IHX RUPTURES" is the result of rupture in the IHX (tubes for IHX helical tube option or plates for IHX plate type option) and in the IHX ducts inside the vessel system. As a consequence primary and secondary systems pressure equalizes.

The event [1.3.2.2.2.5] "INADVERTENT OPENING of a VESSEL SYSTEM SAFETY VALVE" is the result of a safety valve mechanical failure that causes the safety valve opening.

2.3.3.3 Failure in Heat Sink Availability

A numbers of faults can be postulated that result in a failure in heat sink availability (such as decrease in heat removal from the primary system and loss of core support functions) causing degradation of the core heat removal process.

The MLD branches detailing the event «FAILURE in HEAT SINK AVAILABILITY», reported in Fig. 2.3-1, identify one level 4 event that have the potential to challenge the fuel particles integrity:

- DECREASE in HEAT REMOVAL from the PRIMARY SYSTEM [1.3.3.1];

level 4 events description

Event [1.3.2.1] "DECREASE in HEAT REMOVAL from the PRIMARY SYSTEM" refers to anomalies in the heat removal process of the Secondary System or of the SCS, in cooldown modes, that could result in a reduction of its capacity to remove heat generated in the reactor coolant system. The MLD for these events will be shown in section 2.3.3.3.1.

2.3.3.3.1 Decrease in Heat Removal from the Primary System

The MLD branches detail the event «DECREASE in HEAT REMOVAL from the PRIMARY SYSTEM», reported in Fig. 2.3.3-11, identify two level 5 scenarios that have the potential to challenge the primary system and the fuel particles integrity.

- SECONDARY SYSTEM MALFUNCTION that RESULTS in a DECREASE of HEAT REMOVAL [1.3.3.1.1];
- SCS MALFUNCTION that RESULTS in a DECREASE of HEAT REMOVAL [1.3.3.1.2];

level 5 events description

The event [1.3.3.1.1] "SECONDARY SYSTEM MALFUNCTION that RESULTS in a DECREASE of HEAT REMOVAL" can be attributed to Secondary System failures or control system malfunctions that decrease the Secondary System heat removal capacity.

The event [1.3.3.1.2] "SCS MALFUNCTION that RESULTS in a DECREASE of HEAT REMOVAL" can be attributed to SCS failures or control system malfunctions that decrease the SCS heat removal capacity. This event applies to SCS cooling modes operation with reactor shutdown while the primary system is pressurized or while the primary system is near atmospheric pressure during refuelling or repair operation.

level 6 events description

The event [1.3.3.1.1.1] "TURBO-MACHINE MECHANICAL FAILURE" (the gas compressor - gas turbine machine) can be attributed to ruptures in the turbo-machine components. Major failure in the turbo-machine could involve blades, vanes, turbine disks, compressor wheels and shaft. These failures have also the potential to generate missiles or projectiles that can leaf the turbo-machine casings. Turbo-machine damage can also be initiated by metallurgical defects, by manufacturing defects and by foreign objects (external part inadvertent introduced into the coolant during ISI or maintenance, or metallic and insulation material detached from damaged systems/components). This event may lead to a large, rapid pressure change in the systems.

The event [1.3.3.1.1.2] "INADVERTENT SECONDARY SEPARATION VALVE CLOSURE" is the result of a secondary separation valve malfunction or mechanical failure that causes the secondary separation valve closing.

The event [1.3.3.1.1.3] "LOSS of SECONDARY COOLANT ACCIDENT RESULTING of a SPECTRUM of POSTULATED PIPING BREAKS within the SECONDARY SYSTEM" is the result of a pipe rupture of the secondary system boundary.

The event [1.3.3.1.1.4] "INADVERTENT OPENING of a SECONDARY SYSTEM SAFETY VALVE" is the result of a safety valve mechanical failure that causes the secondary system safety valve opening.

The event [1.3.3.1.1.5] "LOSS of STEAM CYCLE" can be attributed to steam/water circuit malfunction, connecting piping or HRSG tube break, inadvertent valve closure, feedwater pump fault, turbine trip, etc. that causes the loss of the water/steam circuit functionality. (It as to be noted that since no information is available for the steam cycle systems/components, no level 7 events description is provided but only generic causes are reported in level 6).

The event [1.3.3.1.2.1] “INADVERTENT SCS WATER FLOW RATE DECREASE” refers to failures in the cooling water loop such as pipe breaks, circulation pump failures (mechanical or electrical), inadvertent isolation valves closure, heat exchangers flow blockage, water flow rate control system malfunction, pressure control system malfunction, inadvertent safety valve opening, etc.

The event [1.3.3.1.2.2] “INADVERTENT SCS WATER TEMPERATURE INCREASE” refers to failure in the return water circuit such as pipe breaks, pump failures (mechanical or electrical), inadvertent isolation valve closure, heat exchanger flow blockage, water flow rate control system malfunction, etc.

level 7 events description

The event [1.3.3.1.1.1.1] “FAILURE of a TURBINE BLADE” refers to a failure in one or more turbine blades that shear off from the rotor. Failure of one blade has also the potential to damage other blades linked to the same disk or even broken the disk. Blades failure can also generates missiles or projectiles that can exit the turbine casing.

The event [1.3.3.1.1.1.2] “FAILURE of a COMPRESSOR BLADE” refers to a failure in one or more compressor blades that shear off from the rotor. Failure of one blade has also the potential to damage other blades linked to the same wheel or even broken the wheel. Blades failure can also generates missiles or projectiles that can exit the compressor casing.

The event [1.3.3.1.1.1.3] “FAILURE of a TURBINE DISK” refers to rupture of one turbine disk. Pieces of which can also exit the casing.

The event [1.3.3.1.1.1.4] “FAILURE of a COMPRESSOR WHEEL” refers to a rupture of one compressor wheel. Pieces of which can also exit the casing.

The event [1.3.3.1.1.1.5] “TURBO-MACHINE SHAFT BREAK” is postulated as an instantaneous rupture of the turbo-machine shaft. Depending on the break position, the turbine rotor and/or the compressor rotor would drop down and can be ejected. The blades can fracture the casing allowing fragments of both blades and casing to come out the turbo-machine casings. Flow through secondary coolant system is rapidly reduced through the compressor coastdown.

The event [1.3.3.1.1.1.6] “TURBO-MACHINE SHAFT SEIZURE” refers to a locked rotor. The accident postulated is an instantaneous seizure of the turbo-machine rotor. Flow through secondary coolant system is postulated to become immediately zero.

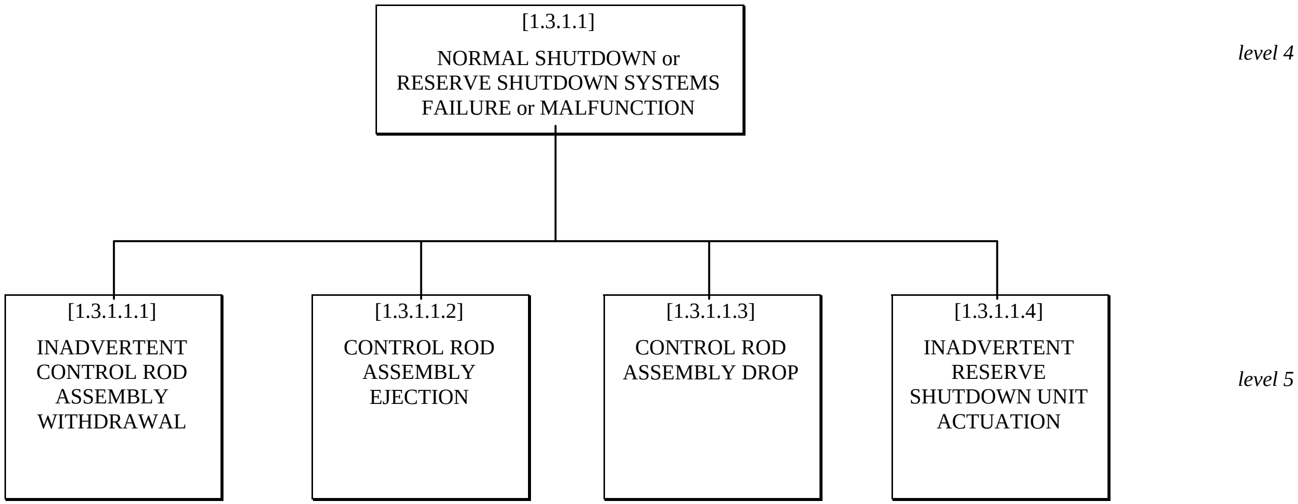

Figure 2.3.3-1 MLD development starting from the condition: Normal Shutdown or Reserve Shutdown System Failure or Malfunction

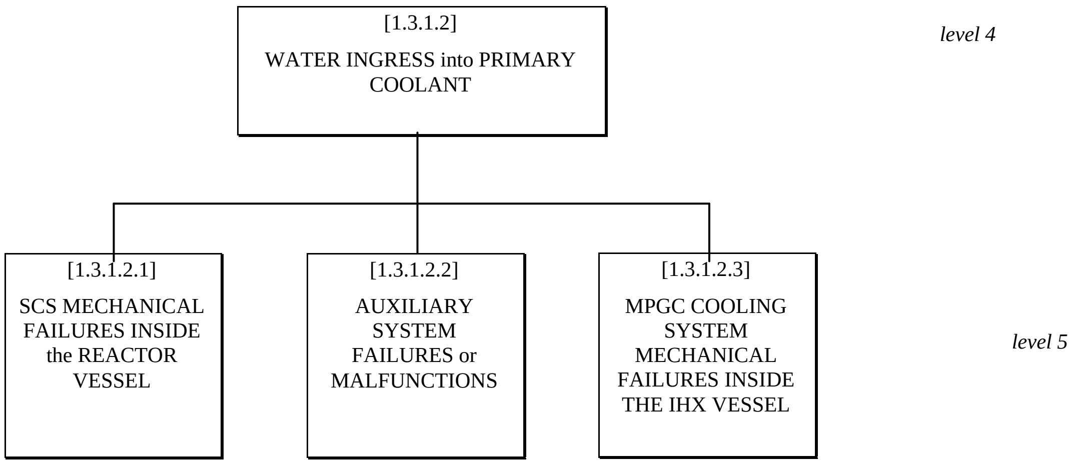

Figure 2.3.3-2 MLD development starting from the condition: Water Ingress into Primary Coolant

[1.3.1.3]

AUXILIARY SYSTEMS MALFUNCTIONS that INSERT CONTAMINANT in PRIMARY HELIUM

Figure 2.3.3-3
MLD development starting from the condition: Auxiliary Systems Malfunctions that Insert Contaminant in Primary Helium

[1.3.1.4]

REFLECTOR GEOMETRY MODIFICATION

Figure 2.3.3-4 MLD development starting from the condition: Reflector Geometry Modification

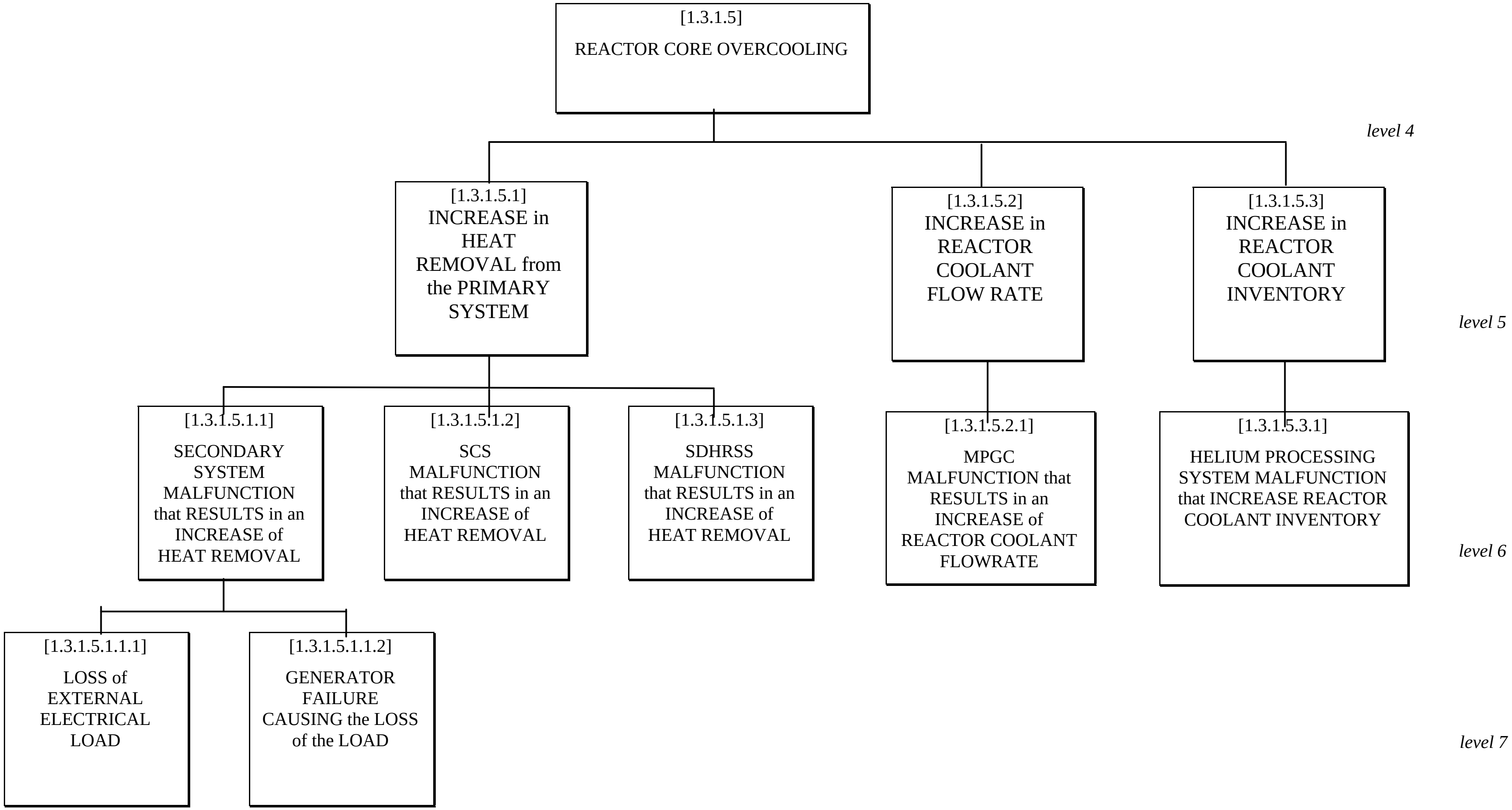

Figure 2.3.3-5 MLD development starting from the condition: Reactor Core Overcooling

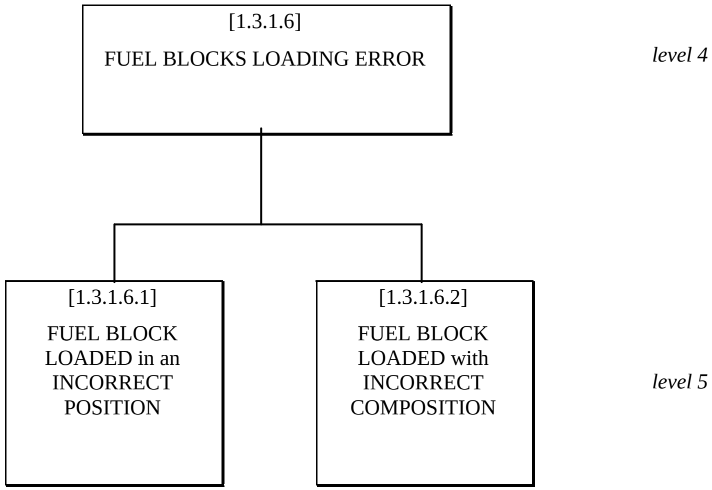

Figure 2.3.3-6 MLD development starting from the condition: Fuel Blocks Loading Error

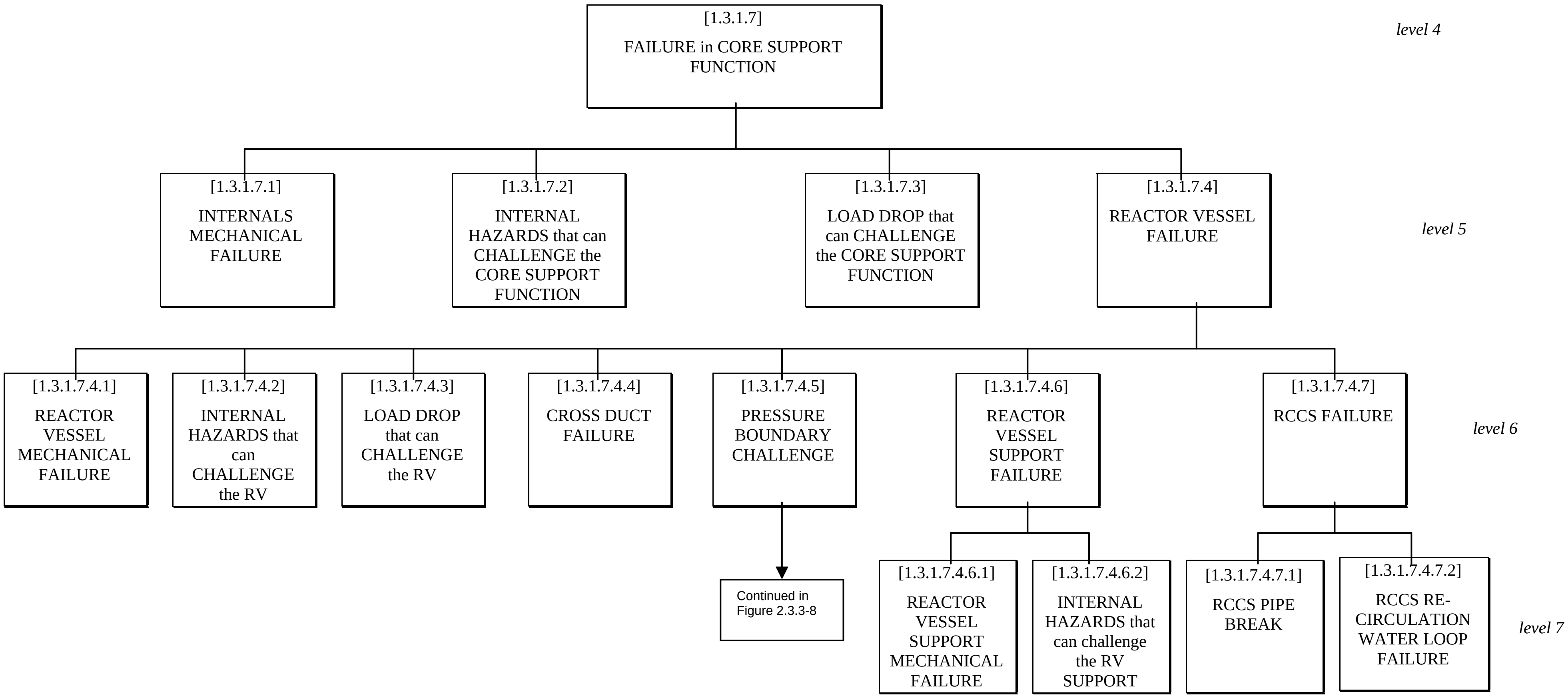

Figure 2.3.3-7 MLD development starting from the condition: Failure in Core Support Function

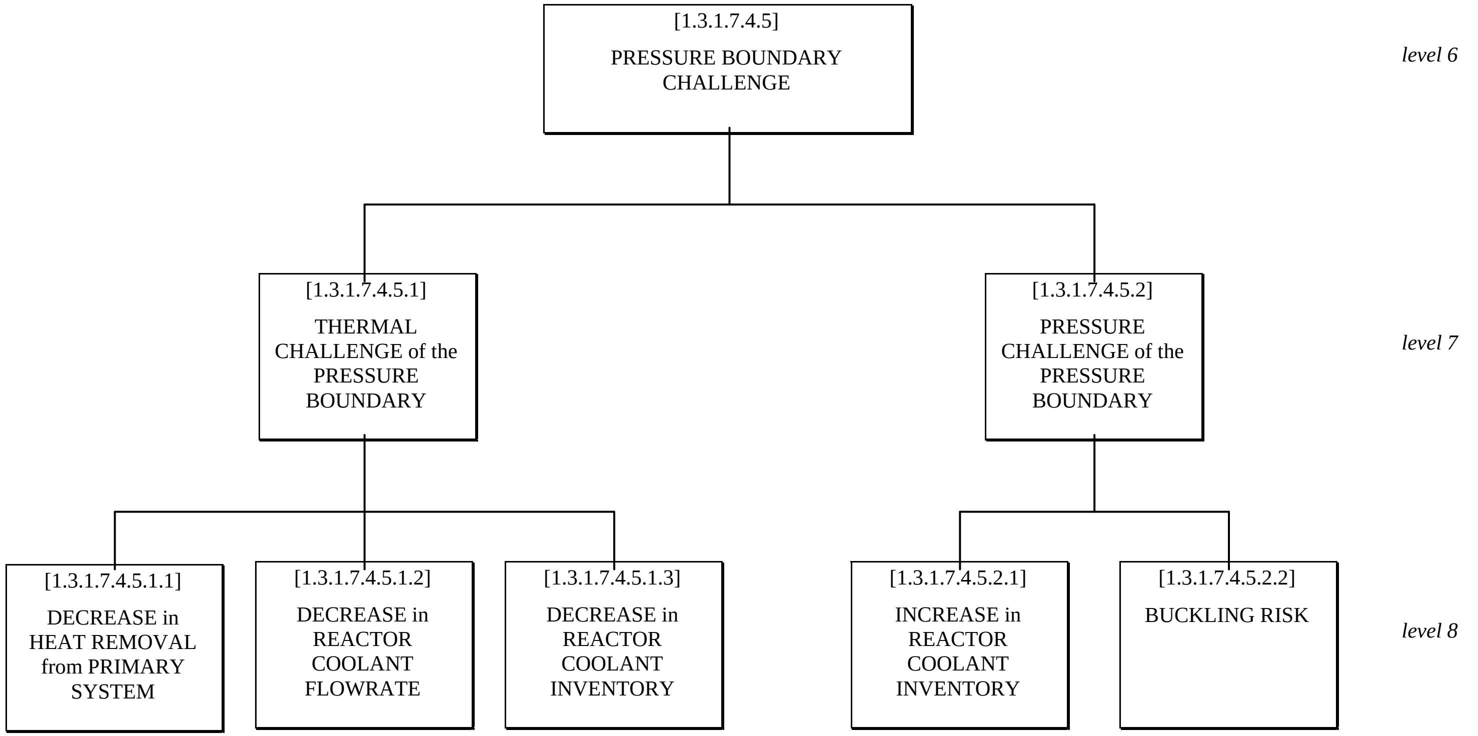

Figure 2.3.3-8 MLD development starting from the condition: Pressure Boundary Challenge

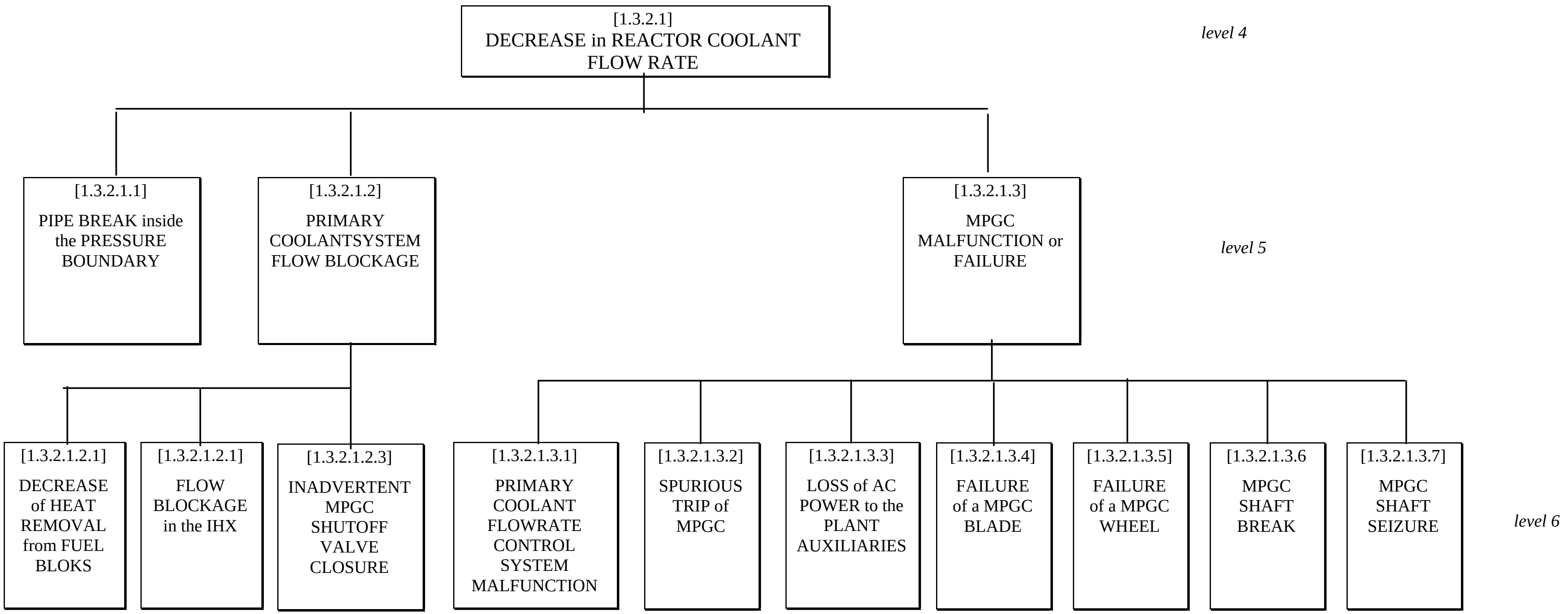

Figure 2.3.3-9 MLD development starting from the condition: Decrease in Reactor Coolant Flowrate

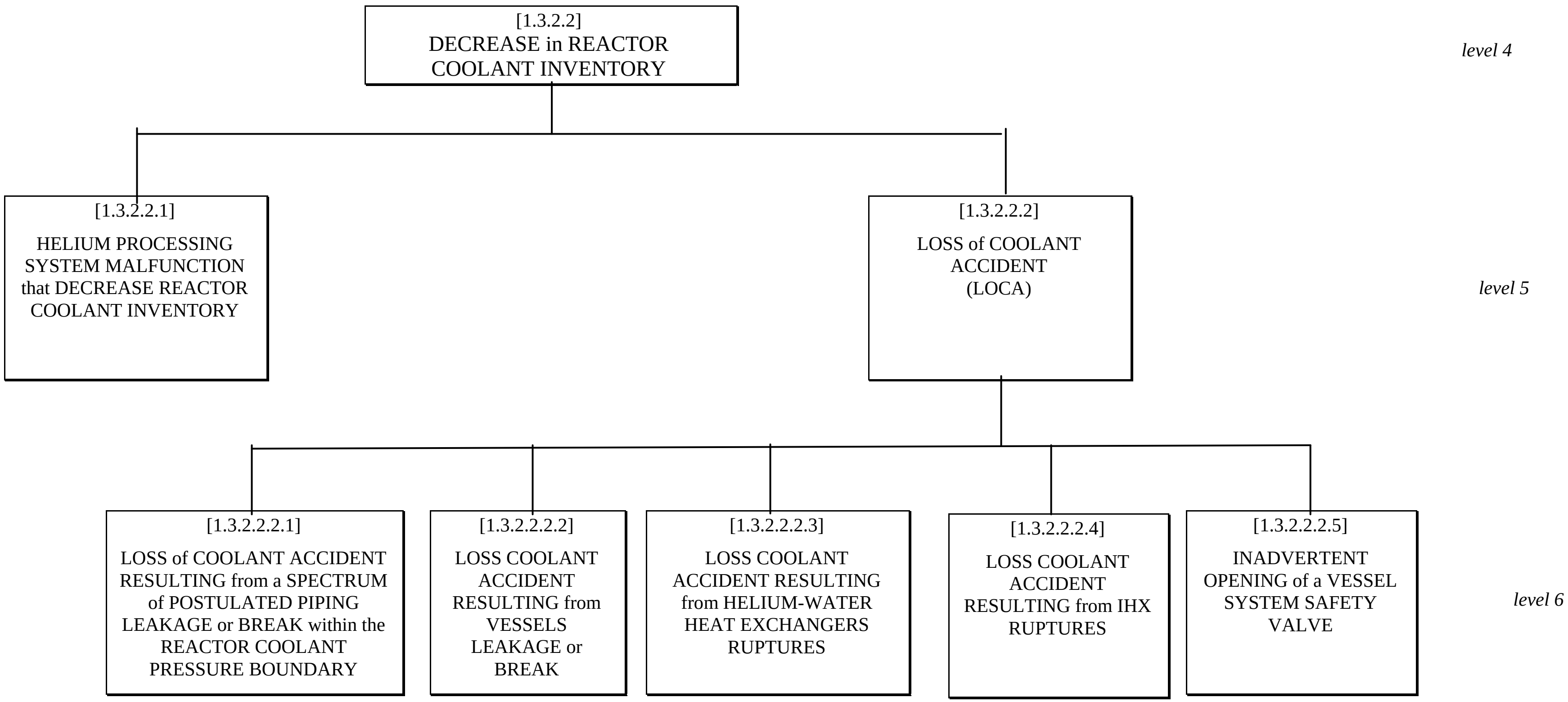

Figure 2.3.3-10 MLD development starting from the condition: Decrease in Reactor Coolant Inventory

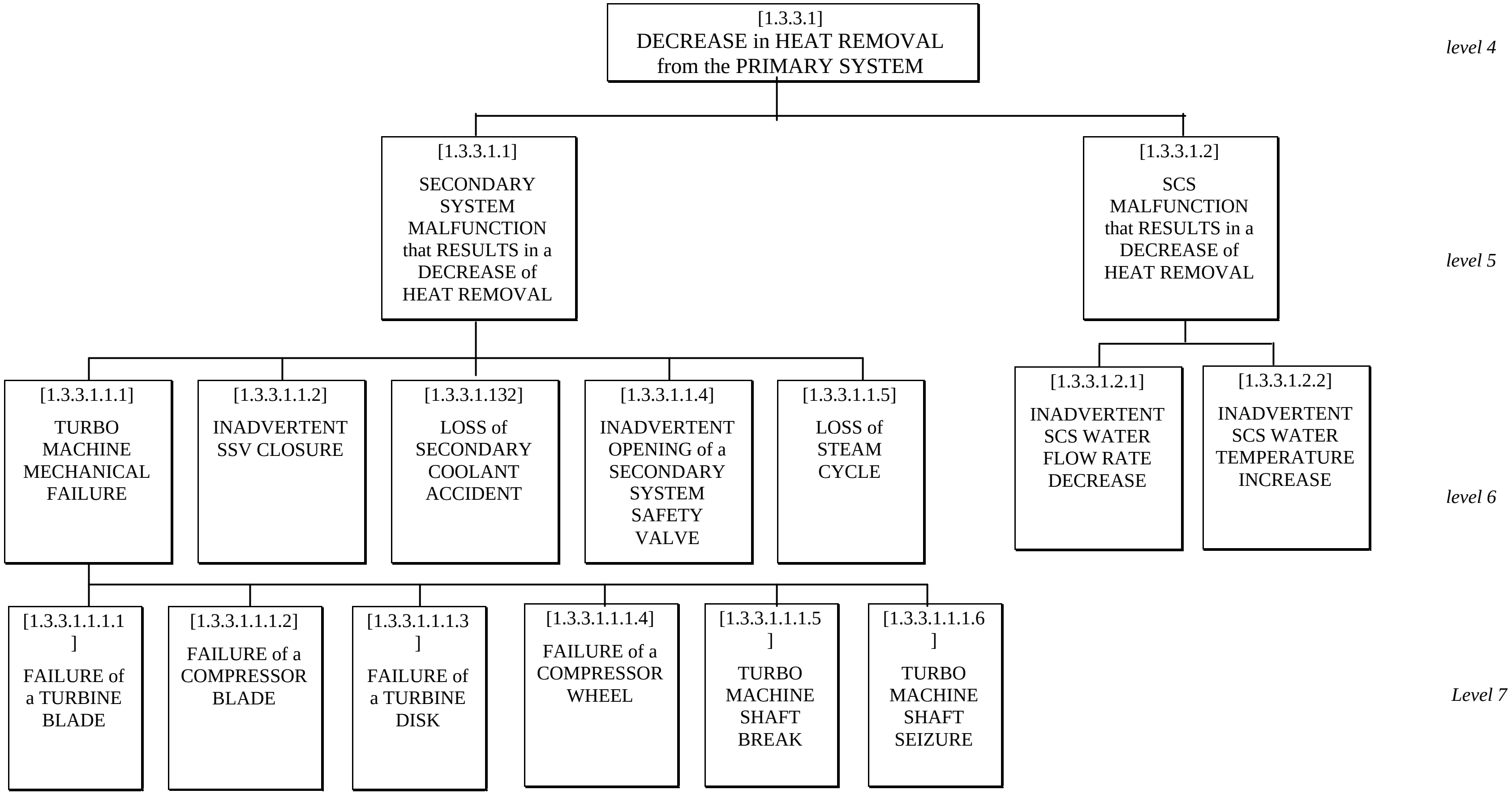

Figure 2.3.3-11 MLD development starting from the condition: Decrease in Heat Removal System

2.4 LIST OF ACCIDENT INITIATING EVENTS

All the accidents identified and scrutinized in previous section 2.3 are listed here.

Fuel Challenges

1. *Mechanical Challenges*

- Fuel Handling Accident

2. *Chemical Challenges*

- Abnormal amount of contaminants in primary helium
- Oxygen ingress into primary coolant
- Water ingress into primary coolant

3. *Thermal Challenges*

<u>Failure in Power and Reactivity Control</u>

- Inadvertent control rod assembly withdrawal
- Control rod assembly ejection
- Control rod assembly drop
- Inadvertent reserve shutdown unit actuation
- Shutdown cooling system mechanical failures inside the reactor vessel
- Auxiliary systems failures or malfunctions
- Main primary gas circulator cooling system mechanical failures inside the IHX vessel
- Auxiliary systems malfunctions that insert contaminant in primary helium
- Reflector geometry modification
- Loss of external electrical load
- Generator failure causing the loss of load (Load Rejection)
- Shutdown cooling system malfunction that results in an increase of heat removal
- Secondary decay heat removal system malfunction that results in an increase of heat removal
- Main primary gas circulator malfunction that results in an increase of reactor coolant flowrate
- Helium processing system malfunction that increase reactor coolant inventory
- Fuel block loaded in an incorrect position
- Fuel block loaded with incorrect composition
- Internals mechanical failure
- Internal hazards that can challenge the core support function
- Load drop that can challenge the core support function

- Reactor vessel mechanical failure
- Internal hazards that can challenge the reactor vessel
- Load drop that can challenge the reactor vessel
- Cross duct mechanical failure
- Buckling risk
- Reactor vessel support mechanical failure
- Internal hazards that can challenge the reactor vessel support
- Reactor cavity cooling system pipe break
- Reactor cavity cooling system re-circulation water loop failure

Failure in Maintenance of Cooldown Flowrate

- Pipe break inside the pressure boundary
- Decrease of heat removal from fuel blocks
- Flow blockage in the IHX
- Inadvertent main primary gas circulator shutoff valve closure
- Primary coolant flowrate control system malfunction
- Spurious trip of main primary gas circulator
- Loss of AC power to the plant auxiliaries
- Failure of a main primary gas circulator blade
- Failure of a main primary gas circulator wheel
- Main primary gas circulator shaft break
- Main primary gas circulator shaft seizure (locked rotor)
- Helium process system malfunction that decrease reactor coolant inventory
- Loss of coolant accident resulting of a spectrum of postulated piping breaks within the reactor coolant pressure boundary
- Loss of coolant accident resulting from vessels leakage or break
- Loss of coolant accident resulting from helium-water heat exchangers ruptures
- Loss of coolant accident resulting from IHX ruptures
- Inadvertent opening of a vessel system safety valve

Failure in Heat Sink Availability

- Failure of a turbine blade
- Failure of a compressor blade
- Failure of a turbine disk
- Failure of a compressor whell
- Turbo machine shaft break
- Turbo machine shaft seizure
- Inadvertent secondary separation valve closure
- Loss of secondary coolant accident
- Inadvertent opening of a secondary system safety valve
- Loss of steam cycle

- Inadvertent shutdown cooling system water flow rate decrease during cooling mode operation
- Inadvertent shutdown cooling system water temperature increase during cooling mode operation

3. EVENTS CATEGORIZATION

3.1 CATEGORIZATION RULES

The general objective of nuclear safety is to protect individuals, population and the environment by establishing and maintaining an effective defence against radiological hazards. The safety philosophy primarily addresses the prevention of accidents but also devotes attention to the protective features (i.e. those features designed to control and limit the consequences of a given accident) as well as the mitigation of the consequences of accidents which could give rise to major releases. The aim is to reduce both the probability of the events and their associated off-site consequences in order to avoid the need of extensive countermeasures and to offer the Authorities the possibility of simplifying or declaring not necessary the off-site emergency planning. The basic approach is to group all the identified initiating events either in Design Basis Conditions (DBC) or Design Extension Conditions (DEC). The Design Basis Conditions consist of the following 4 briefly explained categories.

- Design Basis Conditions I (DBC1, Normal Operations)
  This category contains the normal operation events, such as system start-up, design range operations, hot standby or shutdown, refuelling. Conditions associated with these events are accommodated with a margin between the value of any significant plant parameter and the value of that parameter which would require either passive, automatic or manual protective action.

- Design Basis Conditions II (DBC2, faults of moderate frequency)
  This category contains the events which have a frequency of occurrence higher than 10^{-2} per plant-year. This corresponds to the frequency of events expected to occur one or more times during the plant life (anticipated operational occurrences). These conditions, at worst, result in a reactor trip, with the plant being capable of returning to operation in the short term, if appropriate provisions are taken. These conditions do not propagate to cause a more serious fault (i.e. DBC3 or DBC4)

- Design Basis Conditions III (DBC3, infrequent faults)
  This category includes events not individually expected to occur during the plant lifetime; however, when integrated over all plant components, events in this category may be expected to occur with a frequency of 10^{-2} to 10^{-4} per plant-year. A DBC3 does not, by itself, generate a DBC4

- Design Basis Conditions IV (DBC4, limiting faults)
  This category includes events having a frequency of occurrence lower than 10^{-4} per plant-year. These events represent limiting cases of failures which are identified as Design Basis

In addition to the Design Basis Conditions a specific set of accident sequences, formerly called Beyond Design Conditions, shall be considered in the design of the plant. These are identified as the Design Extension Conditions (DEC), which shall be selected with the basic aim of meeting both probabilistic safety objectives for core damage and for release of radioactive products to the external atmosphere. DEC events include complex sequences (DBC3 and DBC4 combined with the complete failure of any active system, up to enveloping situations that are only mitigated by means of inherent behaviour and long term corrective actions), specific limiting events (defined in order to verify that no cliff edge effect occurs e.g., simultaneous failure of several pipes) and more severe hazards than the ones considered as DBC2, DBC3 and DBC4. The DEC are analysed using realistic assumptions; possible additional provisions are designed and implemented to cope with these specific sequences.

To categorise the initiating events a detailed probabilistic analysis should be performed. At this stage of the project no sufficient data are available to make a probabilistic analysis, hence an engineering judgment process is adopted and some rules are reported in the following.

Operator error

An operator error capable to initiate a fault is considered as a DBC2 event.

Active equipment failure

Pump, compressor

Control system failures are included.
The inadvertent reduction or acceleration of pump/compressor speed is considered as a DBC2 initiating fault.
A compressor single blade failure is considered as a DBC3 initiating fault. This initiating fault is likely to lead to the failure of all of the blades attached to the same disk.
A compressor single disc failure is considered as a DBC4 initiating fault.
The pump/compressor severe seizure is considered as a DBC4 initiating fault.
The pump/compressor shaft failure is considered at least as a DBC4 initiating fault.

Turbine

The control system failures are included.
The inadvertent reduction or acceleration of turbine speed is considered as a DBC2 initiating fault.
The inadvertent turbine trip is considered as a DBC2 initiating fault.
A turbine single blade failure is considered as a DBC3 initiating fault. This initiating fault is likely to lead to the failure of all of the blades linked to the same disk.
A turbine single disc failure is considered as a DBC4 initiating fault.
The turbine severe seizure is considered as a DBC4 initiating fault.
The turbine shaft failure is considered at least as a DBC4 initiating fault.

Valve

Control system failures are included.
The failure of the control system (spurious opening or closure) is considered as a DBC2 initiating fault.
Valve leakage is considered as a DBC2 initiating fault.
Failure to operation on demand is considered as a DBC2 initiating fault.

Control rod

The mechanisms and the control system failures are included.
The inadvertent drop of a single control rod is considered as a DBC2 initiating fault.
The inadvertent insertion/withdrawal of a single control rod is considered as a DBC2 or DBC3 initiating fault (depending on the design).
The inadvertent control rod group insertion/withdrawal is considered as a DBC2 or DBC3 initiating fault (depending on the design).
Spurious reactor trip is considered as a DBC2 initiating fault.
The single control rod ejection is considered as a DBC4 initiating fault.
Large absorber element ejection has to be rejected in the residual risk.

Passive equipment failure

Pipe, vessel, heat exchanger

Heat exchangers are included.
Leakage of a small pipe is considered as a DBC2 initiating fault.
Break of a small pipe is considered as a DBC3 initiating fault.
Leakage of a larger pipe or vessel:

- For high quality equipment (at least quality class 2), the small leakage is considered as a DBC4 initiating fault and double ended pipe or vessel break may be considered in the residual risk, if sufficient preventive measures (high design construction class, Leak Before Break, In service Inspection, exclusion of load drop…) are provided.
- For a lower quality equipment or if preventive measures are not sufficient, a small leakage is considered as a DBC2 initiating fault and a break is considered as a DBC3 initiating fault.

Pipe or vessel bursting or buckling is rejected in the residual risk (design provision and protection with safety valves).

Check valve and safety valve

Check valve, safety valve leakage is considered as a DBC2 initiating fault.
Safety valve spurious actuation is historically considered as a DBC 2 initiating fault, however if they are properly designed or passively actuated their reliability would increase up to consider their fault as DBC3 initiating fault.
No safety valve operation on demand is considered as a DBC4 initiating fault.

Vessel internals

Small failure is considered as a DBC3 initiating fault (plant availability reasons) .
Leakage between the cold and hot regions is considered as a DBC3 initiating fault.
Structural support failure is considered in the residual risk, if sufficient preventive measures (high design construction class, In service Inspection, exclusion of load drop…) are provided. If not, the failure is either considered as a DBC4 event or a limiting event.

System failure

The categorization of the failure will depend on the system design.
The failure of a system made up of two redundant trains (or more) permanently in operation is considered as a DBC4 initiating fault.
If the systems are diversified (using components with different design and/or from different manufacturers) and independent (the systems are designed to avoid the common mode failures), their simultaneous failures are not considered as an initiating fault.

Fuel failure

During normal operation, a failure rate of fuel particles is defined mainly based on the failure rate due to fabrication. Higher failure rates (not due to accidental conditions and expected to remain very low) have to be defined as DBC2, DBC3 and DBC4 events and Design Extension Conditions.

Loss Of Off-site Power (LOOP)

The total loss of offsite power with a small duration (to be defined) is considered as a DBC2 initiating fault.
The total loss of offsite power with a long duration (to be defined) is considered as a DBC3 initiating fault.

Fuel handling

Handling error at one core position is considered as a DBC2 initiating fault.
Loading error is considered as a DBC4 initiating fault.
Single fault in fuel handling is considered as a DBC2 initiating fault.
Multiple faults in fuel handling are considered as DBC3 initiating faults.
A fuel element drop/release during handling is considered as a DBC3 initiating fault.
The drop of the fuel handling device is a DBC4 initiating fault.

Load drop

The load drop during component handling is considered as a DBC4 initiating fault.
The drop of big upper internal structure during handling is at least a DBC4 initiating fault.

Local loss of fuel element cooling

The fuel element blockage due to the introduction of foreign parts (after maintenance, refuelling, manufacturing), the damage of internals (graphite, metallic or insulation material parts) or due to excessive oxidation or corrosion products deposition is considered as a DBC3 or DBC4 initiating fault depending on the type of blockage

3.2 LIST OF ACCIDENT INITIATING EVENTS FOR CATEGORIES

All the initiating events identified and listed in Section 2.4 have been categorized using the rule depicted in the previous section and reported as follows:

Condition II (DBC2, faults of moderate frequency)

- Fuel handling accident resulting in a single failure or breach of one fuel block
- Abnormal amount of contaminants in primary helium due to auxiliary systems malfunctions
- Fuel block loaded with incorrect composition (few pellets)
- Inadvertent control rod assembly withdrawal
- Control rod assembly drop
- Spurious reactor trip
- Auxiliary systems failures or malfunctions
- Loss of external electrical load for a time less than (tbd)
- Generator failure causing the loss of load (Load Rejection)
- Shutdown cooling system (including cooling circuits) malfunction that results in an increase of heat removal
- Secondary decay heat removal system malfunction that results in an increase of heat removal
- Main primary gas circulator malfunction that results in an increase of reactor coolant flowrate
- Helium processing system malfunction that increase reactor coolant inventory
- Primary coolant flowrate control system malfunction
- Spurious trip of main primary gas circulator
- Loss of AC power to the plant auxiliaries
- Flow blockage in the IHX resulting in a flow blockage of one pipe (or one flow path)
- Helium process system malfunction that decrease reactor coolant inventory
- Primary coolant leakage (including assessment of potential loss of coolant accident and air ingress)
- Loss of coolant accident resulting from IHX leakage

- Inadvertent secondary separation valve closure
- Loss of secondary coolant resulting from piping/components small break
- Inadvertent opening of a secondary system safety valve
- Loss of steam cycle
- Inadvertent shutdown cooling system water flow rate decrease during cooling mode operation (up to system trip)
- Inadvertent shutdown cooling system water temperature increase during cooling mode operation
- Reactor Cavity Cooling System leakage
- Failure of Reactor Cavity Cooling System cooling circuit (one train)
- Fuel block loaded in an incorrect position

Condition III (DBC3, infrequent faults)

- Fuel handling accident resulting in a drop of a single fuel block
- Fuel block loaded with incorrect composition (few pins)
- Inadvertent reserve shutdown unit actuation
- Main primary gas circulator cooling system mechanical failures inside the IHX vessel
- Shutdown cooling system mechanical failures inside the reactor vessel
- Loss of external electrical load for a time greater than (*tbd*)
- Reactor Cavity Cooling System pipe break
- Pipe leakage inside the pressure boundary
- Inadvertent main primary gas circulator shutoff valve closure
- Decrease in heat removal from fuel block resulting in a flow blockage in one channel
- Flow blockage in the IHX resulting in a flow blockage of several pipes (or several flow paths)
- Loss of a main primary gas circulator blade
- Inadvertent opening of a vessel system safety valve
- Loss of coolant accident resulting from a small break of a primary helium pipe (including air ingress assessment)
- Loss of coolant accident resulting from helium-water heat exchanger ruptures
- Loss of coolant accident resulting from IHX rupture
- Small amount of water ingress into primary coolant (up to *tbd*)
- Small amount of Oxygen ingress into primary coolant (up to *tbd*) following a primary helium leakage
- Failure of a turbine blade
- Failure of a compressor blade
- Loss of secondary coolant accident resulting from a large break in piping/components

Condition IV (DBC4, limiting faults)

- Fuel handling accident resulting in a drop of a fuel handling device or an heavy structure during handling
- Large amount of water ingress into primary coolant (from *tbd* to *tbd*)
- Single control rod assembly ejection
- Fuel block loaded with incorrect composition (all pins in a bundle)
- Internals mechanical failure
- Reactor Cavity Cooling System large pipe break
- Reactor Cavity Cooling System common mode failure
- Significant leak of primary helium pressure boundary (including assessment of air ingress)
- Decrease in heat removal from fuel block resulting in a flow blockage in all channels of one fuel block
- Loss of a main primary gas circulator wheel
- Main primary gas circulator shaft break
- Main primary gas circulator shaft seizure (locked rotor)
- Loss of coolant accident resulting from pressure boundary vessels or cross duct leakage (including air ingress assessment)
- Failure of a turbine disk
- Failure of a compressor wheel
- Turbo machine shaft break
- Turbo machine shaft seizure

Design Extension Conditions

- DBC with no credit to of active systems on short term
- Large amount of Oxygen ingress into primary coolant
- Large amount of water ingress into primary coolant

Residual Risk

- Multiple control rod assemblies ejection
- Large reflector geometry modification
- Internal hazards that can challenge the core support function
- Load drop that can challenge the core support function
- Reactor vessel mechanical failure
- Internal hazards that can challenge the reactor vessel
- Load drop that can challenge the reactor vessel
- Cross duct mechanical failure (break)
- Buckling risk
- Reactor vessel support mechanical failure
- Internal hazards that can challenge the reactor vessel support
- Loss of coolant accident resulting from vessels break

4. SUMMARY AND CONCLUSIONS

The indirect cycle modular High Temperature Reactor (modular HTR) with block type fuel will be designed to contain and limit radioactivity releases to the external atmosphere in order to guarantee that no undue risk will occur for the population surrounding the plant.

The indirect cycle HTR considered as reference, in this document, is the ANTARES plant [1] designed for both electricity production and process heat generation and developed by AREVA NP. The content of this document has been devised with the main intention to assist and organize the safety analyses o the HTR, that is the set of analyses intended to assess the plant response to abnormal and accident conditions. As such it primarily covers the identification of normal, abnormal and accident events.

All the "internal" initiating events (i.e. events that originate inside the HTR) that lead at abnormal or accident conditions have been identified with a top/bottom approach. They are defined as the ones that have the potential to challenge the first two physical barriers between the radioactive fission products and the external environment (namely the fuel particles and the primary system). However the initiating faults to focus in this phase of the project are mainly those occurring during normal operation of the plant and the consequence of which are likely to challenge the fuel particles confinement capability (the first barrier). Moreover containment) challenges have not been analyzed because out of scope of this work as well as the family of events which directly release radioactive materials outside the containment due to lack of system’s information.

The degree of detail of the contents of the document is necessarily commensurate to the amount of information available at the current stage of the HTR design. This is however inherent to the early design phase of a largely innovative concept.

Concerning the identification of normal, abnormal and accident events, the Master Logic Diagram (MLD) approach has been adopted. This approach, which is more formally organized, consists in identification of the initiating events by constructing a top-level logic model (called MLD), systematically describing all the abnormal and accident conditions resulting in potential challenges to the fuel particles and then deducing the appropriate set of initiating events. A list of all identified accident is provided.

Rules, based on engineering judgement, for initiating events classification are set-up. A preliminary events categorization is also provided.

5. REFERENCES

[1] AREVA NEPL-F DC 14 Rev. A, “RAPHAEL SP-SI, ANTARES data for SP-ST (answer to WP-ST4 request)”.

[2] Deliverable D-CT1.1 Rev. 00, “Component specifications for IHX, isolation valves and coolers”.

[3] Deliverable D-CT2.1 Rev. 00, “Specification for helium circulator”.