ENEA
Agenzia nazionale per le nuove tecnologie,
l'energia e lo sviluppo economico sostenibile

MINISTERO DELLO SVILUPPO ECONOMICO

Ricerca di Sistema elettrico

Confronto e valutazione della risposta di sistemi attivi e passivi in reattori innovativi a fronte di sequenze incidentali significative ai fini della sicurezza

L. Burgazzi, G. Bianchini, A. Cervone, M. Polidori, F. Giannetti, D. Vitale Di Maio, T. Murgia, L. Ferroni, A. Naviglio

CIRTEN - CONSORZIO INTERUNIVERSITARIO PER LA RICERCA TECNOLOGICA NUCLEARE

Report RdS/2013/068

CONFRONTO E VALUTAZIONI DELLA RISPOSTA DI SISTEMI ATTIVI E PASSIVI IN REATTORI INNOVATIVI A FRONTE DI SEQUENZE INCIDENTALI SIGNIFICATIVE AI FINI DELLA SICUREZZA

L. Burgazzi, G. Bianchini, A. Cervone, M. Polidori (ENEA), F. Giannetti, D. Vitale Di Maio, T. Murgia, L. Ferroni, A. Naviglio (Università di Roma "La Sapienza")

Settembre 2013

Report Ricerca di Sistema Elettrico

Accordo di Programma Ministero dello Sviluppo Economico - ENEA
Piano Annuale di Realizzazione 2012
Area: Produzione di energia elettrica e protezione dell'ambiente
Progetto: Sviluppo competenze scientifiche nel campo della sicurezza nucleare e collaborazione ai programmi internazionali per il nucleare di IV Generazione
Obiettivo: Sviluppo competenze scientifiche nel campo della sicurezza nucleare
Responsabile del Progetto: Felice De Rosa, ENEA

Il presente documento descrive le attività di ricerca svolte all'interno dell'Accordo di collaborazione "Sviluppo competenze scientifiche nel campo della sicurezza nucleare e collaborazione ai programmi internazionali per il nucleare di IV generazione"
Responsabile scientifico ENEA: Felice De Rosa.
Responsabile scientifico CIRTEN: Giuseppe Forasassi

Titolo

Confronto e valutazione della risposta di sistemi attivi e passivi in reattori innovativi a fronte di sequenze incidentali significative ai fini della sicurezza

Descrittori

Tipologia del documento:	**Rapporto Tecnico**
Collocazione contrattuale:	Accordo di programma ENEA-MSE: Piano Annuale di Realizzazione 2012, Linea Progettuale 1, Obiettivo B: Metodologie avanzate per la valutazione delle conseguenze incidentali, Task B.3
Argomenti trattati:	Sicurezza nucleare Analisi incidentale Analisi di sicurezza probabilistica

Sommario

Il presente documento riporta le attività svolte nell’ambito della Linea Progettuale 1 (LP1), obiettivo B (Metodologie avanzate per la valutazione delle conseguenze incidentali), task B.3, del PAR 2012, ADP ENEA-MSE.

Lo studio illustra il confronto della risposta dei sistemi di sicurezza attivi e passivi a fronte di sequenze incidentali, relativamente a reattori avanzati di prossima generazione.

In particolare sistemi di sicurezza attivi e passivi che svolgono la stessa funzione di sicurezza, come lo smaltimento del calore residuo e lo spegnimento del reattore, vengono analizzati e comparati a livello di sistema in termini di prestazioni ed affidabilità; il medesimo confronto viene effettuato a livello di sequenze incidentali, in termini di adeguatezza relativamente alle azioni in risposta a determinati eventi iniziatori.

Note:

Questo documento è stato preparato col contributo congiunto del seguente personale di ricerca ENEA e CIRTEN:

- L. Burgazzi, G. Bianchini, A. Cervone, M. Polidori (ENEA)
- F. Giannetti, D. Vitale Di Maio, T. Murgia, , L. Ferroni, A. Naviglio (Università di Roma “La Sapienza”)

Sigla doc. rif.: CIRTEN-Università di Roma “La Sapienza”: CERSE-UNIRM RL 1185/2013

2			NOME			
			FIRMA			
1			NOME			
			FIRMA			
0	EMISSIONE	02/09/13	NOME	L. Burgazzi	F. De Rosa	F. De Rosa
			FIRMA			
REV.	DESCRIZIONE	DATA		REDAZIONE	CONVALIDA	APPROVAZIONE

Sommario

L’aumento del grado di sicurezza dei reattori innovativi, come i reattori di quarta generazione, avviene attraverso l'implementazione di caratteristiche di sicurezza intrinseca e passiva nei rispettivi progetti. Una motivazione per l'uso di sistemi passivi per realizzare funzioni di sicurezza, come lo spegnimento del reattore e la rimozione del calore di decadimento, consiste in una maggiore affidabilità per i vantaggi dichiarati di semplicità, riduzione della necessità dell’intervento umano nonché della alimentazione elettrica esterna. Tuttavia recenti studi sollevano preoccupazioni e cautela circa la rivendicata superiorità dei sistemi passivi in termini di prestazioni e relativa disponibilità e affidabilità.. Quindi, come risposta a questo problema, è stata effettuata una valutazione comparativa dei sistemi attivi e passivi in termini, principalmente, dei dati di performance e di affidabilità. A tale scopo la analisi a livello di sistema viene integrata con l'analisi a livello di sequenza incidentale, a seguito della forte interazione tra le prestazioni del sistema e lo scenario incidentale. Per il miglioramento della sicurezza e affidabilità diversi concetti sono presi in esame come il raffreddamento del nucleo con circolazione naturale in caso di station blackout e l'uso di sistemi passivi per lo spegnimento del reattore.

L'analisi evidenzia la rilevanza del valore dell'affidabilità come il più importante fattore nel processo di scelta tra le due alternative: la relativa valutazione è riconosciuta essere ancora un problema aperto, nonostante negli ultimi anni un importante sforzo sia stato fatto dalla comunità internazionale, sia per lo sviluppo che per la valutazione dei sistemi di sicurezza passiva. L'inclusione di potenziali guasti e le stime di affidabilità dei sistemi passivi è pertanto raccomandato in studi probabilistici di valutazione di sicurezza.

In particolare, per quanto riguarda i sistemi a circolazione naturale, i risultati mostrano che la probabilità di guasto della funzione di sicurezza passiva non è da trascurare. Tuttavia con i modelli qui presentati, le ipotesi semplificative e gli scenari limitati considerati, non è ragionevole stabilire che l'affidabilità funzionale per questi sistemi sia tale da impedire l'espletamento della funzione di sicurezza. Ma si può dedurre che attenzione deve essere posta agli aspetti funzionali del sistema passivo, (quelli non appartenenti alla "hardware" del sistema), che può mettere in discussione la loro superiore "accreditata" affidabilità rispetto a quelli attivi.

Sono stati inoltre analizzati i sistemi passivi di spegnimento del reattore con il loro potenziale di migliorare le prestazioni del sistema di protezione del reattore con l'aggiunta di una ulteriore funzione di sicurezza, aumentando l'affidabilità dello “scram” e contribuendo in modo significativo alla affidabilità dell'impianto complessivo. Tuttavia anche in questo caso la mancanza di prove sperimentali e il loro stadio di sviluppo precoce non consente di verificare e validare il prefissato obiettivo di affidabilità, che li rende interessanti per la loro inclusione nel progetto dei reattori innovativi.

A tal proposito è stata effettuata una analisi delle prestazioni dei sistemi di asportazione del calore, sia attivi che passivi, considerando le relative probabilità di fallimento degli stessi, in relazione alle sequenze funzionali che li caratterizzano. Suddette valutazioni sono state effettuate utilizzando la metodologia basata sugli alberi dei guasti in cui si analizzano le possibili modalità di fallimento dei sistemi e/o componenti, che sono stati tenuti in considerazione sia per il funzionamento in modalità attiva, sia per il funzionamento in modalità passiva che in modalità mista. I risultati ottenuti attraverso questa analisi sono stati riportati all’interno del documento al fine di poter confrontare preliminarmente le prestazioni del sistema nelle diverse condizioni.

Alla fine si può concludere che l’affidabilità del sistema passivo non è meglio o peggio di quelli attivi: l’ affidabilità dipenderà dal progetto complessivo e dal funzionamento del sistema, indipendentemente dal fatto che il sistema sia attivo o passivo. Un progetto di buon livello dell'impianto può comprendere sistemi attivi, sistemi passivi, o una combinazione di entrambi i tipi di sistemi per soddisfare gli obiettivi di sicurezza e di prestazioni.

Anche se questi sistemi vengono accreditati di una maggiore affidabilità rispetto a quelli "tradizionali" - a causa della minore indisponibilità dovuta ad un guasto hardware - o perfino sono ritenuti “esenti” da guasti, essi pongono tuttavia alcuni problemi per quanto riguarda la la valutazione delle relative prestazioni, in quanto esiste sempre un rischio non nullo del verificarsi di fenomeni fisici che portano a modi di guasto pertinenti.

Table of contents

Executive summary

Safety of innovative reactors, like Gen IV reactors, is expected to be enhanced through the implementation of passive safety features within their designs. A motivation for the use of passive systems to accomplish safety functions, as reactor scram and decay heat removal, is their potential for enhanced safety through increased safety system reliability, because of the claimed advantages of simplicity, reduction of the need for human interaction, reduction or avoidance of external electrical power. However recent studies raise concerns and caution about the claimed superior performance of passive systems and relative higher availability and reliability. Hence, as a response to this concern, a comparative assessment of active and passive systems has been performed in terms, principally, of the expected performance and reliability figures of merit. To this aim the system-based analysis is complemented with the sequence-based analysis, since the strong interaction between the system performance and the accident scenario. For safety and reliability improvement several ideas are included such as the core cooling by natural circulation in case of Station Blackout and the use of passive reactor shutdown systems.

The analysis points out the relevance of the reliability figure of merit as the most important factor in the process of opting out of one system in favor of the other alternative: in fact the relative assessment is recognized as being still an open issue, despite in the recent years an important effort has been made by suppliers, industries, utilities and research organizations on passive safety systems both for their development and assessment. Inclusion of potential failure modes and reliability estimates of passive components for all systems is recommended in Probabilistic Safety Assessment (PSA) studies.

In particular, as regards natural circulation systems, results show that the probability of failure of the passive safety function is not to be neglected. However with the models presented here, the simplifying assumptions and the limited scenarios considered, it is not reasonable to confidentially establish that the functional reliability for these systems is such that it constitutes a challenge for the accomplishment of the safety function. But one can deduce that attention has to be paid to the functional aspects of the passive system, (i.e. the ones not pertaining to the "hardware" of the system), that can challenge their "credited" higher reliability with respect to active ones.

SASS (Self Actuated Shutdown Systems) with the potential to improve reactor protection system performance by adding an important safety feature to the defence-in-depth case, while increasing scram reliability and contributing significantly to the reliability of the overall plant, system have been analyzed. However also in this case the lack of experimental evidence and their premature stage of development doesn't allow to verify and validate the required reliability target, that makes them attractive for their inclusion in the design of innovative reactors.

Concerning this, an analysis has been performed in order to evaluate the heat removal system performance, under all active, passive and hybrid operating modes, considering the corresponding failure rate probabilities, in relation to their functional sequences. These analyses have been carried out through the FTA methodology, in which possible failure modes of components and/or systems, under active, passive or mixed modes have been taken into account. The results obtained through this analysis are reported in the present document to preliminarily compare the performance of the system operating under different conditions.

Lastly one can conclude that passive system reliability is not better or worse than the active ones: reliability will depend on the overall design and operation of the system, regardless of whether the system is active or passive. A good overall plant design may include active systems, passive systems or combination of both types of systems to meet performance and safety objectives. Although these systems are credited a higher reliability with respect to the "conventional" ones - because of the smaller unavailability due to hardware failure - or even they are claimed to be immune from faults, they pose however some challenges as regards the availability/ reliability issues and more in general their performance assessment, because there is always a nonzero likelihood of the occurrence of physical phenomena leading to pertinent failure modes.

1. Introduction and motivation

This study is to be considered as the follow-up of the previous work done throughout the previous year (2012) which emphasized the comparison between active and passive safety systems, to be implemented in advanced nuclear plants, as an important topic to be investigated (ref.1), after the identification of the key issues related to the reliability of passive systems.

This issue has been raised as well by a recent work on the state-of-the-art relative to the passive systems (ref.2), with an attempt to identify the experience lessons, merits, demerits and challenges of the two possible alternatives.

Hence in the present study innovative reactors safety assessment is considered under the focus of a comprehensive comparative assessment between active and passive systems, to accomplish fundamental safety functions, as DHR (Decay Heat Removal) and reactivity control. To this aim some illustrative examples both at the system level and in the frame of PSA (Probabilistic Safety Assessment) context, i.e. sequence-based, are provided and Gen IV reactor are considered mostly, since the implementation in their designs with passive features to cope with safety concerns. In addition to safety systems devoted to DHR, safety systems for reactivity control are being investigated.

2. Background on the topic

Ref.1 lays the foundations for the passive vs active designs assessment, by detailing the relative features and issues with respect to safety requirements, performance and reliability. Henceforth these concepts are briefly recalled, with main focus on passive systems, and some cornerstones for the comparative assessment are briefly delineated.

The design and development of innovative reactors address the use of passive safety systems, i.e. those characterized by no or very limited reliance on external input (forces, power or signal, or human action) and whose operation takes advantage of natural forces, such as free convection and gravity, to fulfil the required safety function and to provide confidence in the plant’s ability to handle transients and accidents. Therefore, they are required to accomplish their mission with a sufficient reliability margin that makes them attractive as an important means of achieving both simplification and cost reduction for future plants while assuring safety requirements with lesser dependence of the safety function on active components like pumps and diesel generators.

On the other hand the concern arising from the factors impairing their performance leads to the consideration that, despite the fact that passive systems “should be” or, at least, are considered, more reliable than active ones - because of the smaller unavailability due to hardware failure and human error - there is always a nonzero likelihood of the occurrence of physical phenomena leading to pertinent failure modes, once the system enters into operation.

These characteristics of a high level of uncertainty and low driving forces for heat removal purposes justify the comparative evaluation between passive and active options, with respect to the accomplishment of a defined safety function (e.g. decay heat removal) and the generally accepted viewpoint that passive system design is more reliable and more economical than active system design has to be discussed.

Here are some of the benefits and disadvantages of the passive systems that should be evaluated vs. the correspondent active system.

– Advantages

- No external power supply: no loss of power accident has to be considered.
- The passive nature of the safety systems reduces the reliance on operator action, which could imply no inclusion of the operator error in the analysis. In fact the minimization of the intrinsic complexity of the system results in improved human reliability. The natural circulation core heat removal without, e.g., the incorporation of mechanical pumps results in reduction of operating and maintenance staff requirements, generation of low-level waste, dose rates, and improvement of operational reliability and plant safety and security.
- Passive systems must be designed with consideration for ease of ISI (In Service Inspections), testing and maintenance so that the dose to the worker is much less.
- The freedom from external sources of power, instrumentation and control reduces the risk of dependent failures such as the common cause failures
- Better impact on public acceptance, due to the presence of “natural forces”.
- Less complex system than active and therefore economic competitiveness.

– Drawbacks

- Reliance on “low driving forces”, as a source of uncertainty, and therefore need for T-H uncertainties modelling.
- Licensing requirement (open issue), since the reliability has to be incorporated within the licensing process of the reactor. For instance the PRA’s should be reviewed to determine the level of uncertainty included in the models and their potential impact. In fact some accident sequences, with frequencies high enough to impact risk but not predicted to lead to core damage by a best estimate t-h analysis, may actually lead to a core damage when t-h uncertainities are considred in the PRA model.
- Need for operational tests, so that dependence upon human factor can not be completely neglected.
- Time response: the promptness of the system intervention is relevant to the safety function accomplishment. It appears that the inception of the passive system operation, as the natural circulation, is conditional upon the actuation of some active components (as the return valve opening) and the onset of the conditions/mechanisms for natural circulation start-up
- Notwithstanding the fact that passive safety systems are claimed to have higher reliability compared with active safety systems, reliability and performance assessment in any case and their incorporation in the reactor concepts needs to be tested adequately, due to several technical issues as formerly pointed out. Quantification of their functional reliability from normal power operation to

transients including accidental conditions needs to be evaluated. Functional failure can happen if the boundary conditions deviate from the specified value on which the performance of the system depends.

- Ageing of passive systems must be considered for longer plant life; for example corrosion and deposits on heat exchanger surfaces could impair their function.

- Economics of advanced reactors with passive systems, although claimed to be cheaper, must be estimated especially for construction and decommissioning.

The question whether it is favourable to adopt passive systems in the design of a new reactor to accomplish safety functions is still to be debated and a common consensus has not yet been reached, about the quantification of safety and cost benefits which make nuclear power more competitive, from potential annual maintenance cost reductions to safety system response.
Ref.1 considers both active and passive systems designed to accomplish the required safety functions, as the decay heat removal, for their investigation mainly in terms of the expected safety performance and reliability. An illustrative example of comparative analysis between two different (passive and active) thermal-hydraulic safety systems fulfilling the same heat removal safety function is given and the results of the Station Black Out events' probabilistic safety assessment for EPR and AP1000 reactors are shown: the analysis revealed various important insights while some significant conclusions have been drawn. removal
With reference to passive systems, it is recognized that their reliability assessment is still an open issue, mainly due to the amount of concerned uncertainties, to be resolved among the community of researchers in the nuclear safety. Moreover a comparative analysis shows that their safety achievement is comparable to or even less than the active systems' one, since the claimed higher reliability and availability are challenged by some important functional aspects, impairing their performance.
On the other hand the same safety level achieved for active safety system reactors through a higher level of redundancy causes a higher level of complexity of the plant, that is a risk factor itself, and makes the plant vulnerable to common cause of failures.

3. System-based comparison active vs passive

As primary and foremost distinction between the two concepts, passive systems are less susceptible to component failures. Instead they are subject to "inherent failures", caused by unexpected change in either the internal physical state, such as due to stratification, or in the system environment, such as plugging of heat exchanger tubes.
Conversely active systems are not subject to inherent failures because they can essentially "power" through these phenomena.
It is important to recognize that the reliability of passive safety systems may vary among different accident scenarios. While this can be true of active systems as well, their reliability more often depends on the operation of the equipment, rather than the physical characteristics and thermal-hydraulic details of each accident scenario.
For example an active safety injection system with a centrifugal pump may have uncertainties regarding the ability to model the system and the specific scenario characteristics. However the system's pumping capabilities are well above the required capabilities. This results in a relatively large functional margin and a negligible contribution by these uncertainties to the likelihood of failure of the system.

On the contrary a passive safety system with relatively smaller capabilities will significantly reduce the available functional margin: as a consequence, the uncertainties in these elements may dominate the reliability of such a passive system.

Consequently the system-based approach should be complemented with the sequence-based approach, since ultimately the accident scenarios are the focus of the analysis rather than the individual passive systems, for safety and licensing purposes.

This research focuses on a different set of reactor technologies examining thermal-hydraulic uncertainties in passive cooling systems, as well as passive shutdown systems behaviour for Generation IV reactors, with regard to the safety level achievement estimation as opposed to active systems. In particular cooled fast reactors have received much of the attention of Generation IV plants.

For Generation IV plant designs, the regulatory concerns are slightly different in the following.

First, a new regulatory regime is under development that may introduce a more risk-informed or even risk-based, licensing and regulatory process. Due to the novelty of the design, especially as regards the utilization of gas or liquid metals as main coolants, as compared to water-cooled reactors, passive system reliability methods may require a more detailed approach than for the current generation of advanced light water reactor designs.

Some of these more advanced plant designs may include even more reliance on passive safety systems or features then the near-term advanced light water reactor designs.

The analysis will be performed consistently with the fact that the Generation IV concepts are at different stages of development and their designs are still evolving.

3.1 Decay heat removal safety systems

This case study refers to a specific Gas-Cooled Fast Reactor Demonstrator (GFRD) system implementing the natural circulation to accomplish the DHR safety function and specifically the ALLEGRO design, under development within the European research program GoFastR (European Gas Cooled Fast Reactor, 2010 to 2013), whose description is offered in references 3 and 4.

The GFRD design promotes the adoption of natural circulation as much as possible to cope with the DHR (Decay Heat Removal) specific challenge, since the reliance on conduction cool-down and radiation, as for HTR for instance, will not work, owing to the high power density, low thermal inertia, poor conduction path and small surface area.

Thus the overall DHR strategy is based on the following principles (ref. 5, 6):

- A natural convection flow is preferred following shutdown
    - This is possible when the circuit is pressurized.
- A forced flow is required immediately after shutdown when depressurized
    - This is because the gas density is too low to achieve enough natural circulation and heavy gas injection is helpful; in addition power requirements for the blower are very large at low pressure.
- The primary circuit must be reconfigured to allow DHR
    - This implies DHR system activation after a reactor trip by a valve sequence, entailing the isolation of the main loops and the connection across the core of the DHR dedicated loops.

This explains some design criteria and options:

- an upward core cooling flow and a location of the exchangers (DHR and IHX) above the core level;
- a core design criterion based on a low pressure drop (which eases the He circulation);
- the use of a guard containment in order to limit the loss of pressure (sufficient backup pressure) after a primary circuit depressurization;
- the use of dedicated DHR loops, allowing to increase the difference of elevation between the DHR heat exchanger and the core, possibly supported by nitrogen injection in order to enhance the core cooling in natural convection.
- If needed, the primary DHR circuit can operate with a blower, generating a forced circulation in addition to the natural circulation;
- the use of three specific loops (3x100% redundancy) directly connected to the primary vessel, assuming that one could be lost due to the accident initiating event (break for example) and another is assumed unavailable (single failure criterion). The 3 x 100% DHR loop systems are designed to remove 3% of the nominal power.

Figure 1 shows a schematic of one out of three systems. The decay heat removal strategy is as follows:

- DHR with natural circulation is proposed for the pressurized case (case of loss of flow combined with a station blackout, where the DHR blowers – which would be launched at first intent - would not be available).
- For the depressurized case, the most attractive option is a combination of active/passive approach to the decay heat removal. In fact the purely passive decay heat analysis suggests that passive heat removal is possible if the system back pressure is high enough to sustain sufficient natural circulation flow through the core and the emergency heat exchanger. If some other mechanism(s) can maintain core cooling in the early phase of a shutdown transient then the requirement can be relaxed for natural circulation cooling (passive cooling) in the later phase of the transient.

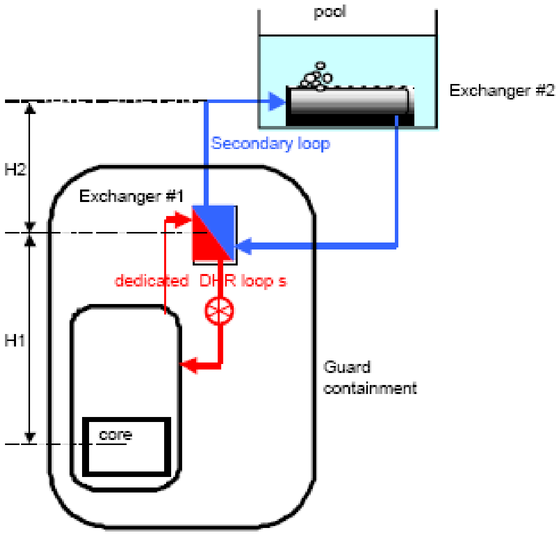


Figure 1. Schematic of the DHR system

Thus, for the first 24-hours after shutdown when natural circulation alone may not be sufficient to cool the core, the system will operate in the active mode with the blower running. With this regard, being the pumping power almost inversely proportional to the square of the gas pressure, the pumping power decreases very significantly when the pressure increases. Therefore an alternative option – while keeping the same forced convection strategy - could be to pre-pressurize the guard containment in order to limit the pumping power requirements, especially as regards scenarios where emergency power supply (such as batteries) is involved. Since the natural circulation mass flow rate through the primary circuit and the corresponding heat removal rate both increase with system pressure, a guard containment structure surrounding the primary system is designed to support an elevated back pressure condition in a depressurization accident.
Like for the main circuit, the DHR secondary loop is a pressurized water loop. The heat is removed from the secondary loop through a horizontal heat exchanger located in a water pool. The water pool is able to act as final heat sink during 24 to 48 hours, if necessary, but would be cooled by an external source after that time.
The intermediate heat exchanger of the decay heat removal loop are calculated according to the two following conservative situations:

- Loss of flow (LOF), with all forced convection unavailable and nominal pressure use (70 bars)
- Loss of coolant gas accident (LOCA), with different back-up pressure levels.

In conclusion the DHR strategy relies on:

- natural convection to ensure a diversified functional redundancy for heat removal in case of high pressure scenarios;
- DHR circulation for scenarios at low pressure, with emergency power supply to be able to start and/or to maintain the forced convection. For the long term, due to the close containment, pressure is set to the adequate pressure to ensure natural convection.

Probabilistic analysis

The expected reliability of each system providing the DHR function should be such that the frequency target for the total loss of DHR function should be less than 10^{-7} per year, consistent with international recommendations for fast reactors (ref.7). The basic DHR function could be jeopardized by several types of fault depending on the operating conditions at the time, including loss of primary coolant inventory, reduction or loss of primary coolant circulation, a loss of coolable core geometry and reduction or loss of heat removal from the primary circuit.
Core coolability and a lack of thermal inertia is a main issue for a helium-cooled fast reactor and this must be taken into account both in the design process and in the safety analysis. In the case of depressurization of the primary circuit it is essential that either forced circulation is maintained within the primary circuit or a minimum pressure adequate for sufficient natural circulation is maintained.
Within a facility, failure of a number of safety devices/components to perform their functions may occur as a result of a single specific cause. Consideration and prevention of such CCF, by provision of DHR systems with adequate diversity and independence, is essential to ensure

adequate reliability levels are achieved. Where redundant systems/components are at potential risk from common cause failures, one means of reducing the susceptibility of the design to such effects is to employ diverse provisions in separate redundant trains or systems. To achieve the required levels of reliability for the DHR function (frequency of total loss of DHR function < 10^{-7} per year), there should be an adequate diversity strategy, which should ensure that the GFR design will be able to maintain the DHR safety function when required, for all operational and accidental states. The general goal should be to achieve an appropriate level of diversity to enable the exclusion of common cause failures from design basis considerations. In particular, the issue of system/component unavailability during maintenance and repair should be considered.

The analysis refers to the plant configuration following an initiator event requiring the reactor shutdown, as for instance following a loss of off-site power supply accident or a depressurization event due to LOCA in the primary circuit.

Therefore the value of the shutdown frequency (1/ry), should be combined with the probability of the DHR system failure, in order to achieve a final reliability figure. It’s worth noticing that the analysis is performed according to the level of definition of the design (due to the early stage of the design many systems are not yet established); in addition the general lack of statistically reliable data makes somehow difficult the assignment of qualified numerical probabilities to events.

The reliability analysis concerns the DHR system performance with regard to both the passive and active operations envisaged for DHR function with reference to the configuration previously set out:

- natural circulation for both pressurized situations with intact helium pressure boundary and in the long term depressurized situations,
- forced circulation in the depressurized situations in the short (24 hours) term decay heat removal in a loss of coolant accident.

In order to limit the loss of pressure in case of primary depressurization and provide the back pressure to sustain natural circulation through the reactor core and the DHR system, a guard containment has been designed to enclose the whole primary system and maintain a pressure at 10 bar.

The findings are expressed in terms of probability of failure of the systems, to be combined with the corresponding number of system intervention demands per year.

Structural failure of the components (e.g., pipes) leading to loss of coolant and loss of heat exchange process, such as loss of coolable geometries (e.g. alteration of the surface area of heat exchange) are the most relevant causes of system failures. More specifically, the modes of failures of the system operating in natural circulation mode are:

- Piping rupture
- DHX1 failure to operate (“lump” failure mode) to include
    - DHX1 Shell rupture
    - DHX1 multiple tube rupture
    - DHX1 multiple tube plugging
- DHX2 failure to operate (“lump” failure mode) to include
    - DHX2 fouling

  - DHX2 broken tubes

- Main loop isolation valve failure to close (required to isolate the main primary loop)
- DHR loop check valve failure to open (required to activate the circuit operation)
- Water tank unavailability (unease to use and reduced effectiveness, due for instance to ice

  formation, high temperature, water make-up failure)
- Loss of leak tightness guard vessel (leak tight guard vessel)

Table below shows the failure rates corresponding to the various modes of failure, as regards the system operation in the natural circulation mode, which includes

- a “U” tubes helium/water heat exchanger (DHX1)
- a water/water heat exchanger (DHX2)
- isolation valves
- check valves
- water tank
- leak tight guard vessel

The DHR loop is exclusively relying on natural convection heat transfer, being the isolation valves the only active components, required to isolate the main circuits while a check valve is supposed to open and activate the loop itself.
The values are taken from relevant databases including data for fission plants, previous works, where applicable (ref. 8, 9) and, if not available, inferred by some kind of “expert” or “engineering judgment” procedure. In any case conservative assumption are adopted. The values in this table are valid for the evaluation of the failure probability of the system, being every failure probability is calculated as λT (T mission time), which is valid for λT<0,1.

Table I. Failure probabilities for passive DHR loop (1 out of 3)

Component	Failure Mode	Failure rate/h-probability/d	Failure probability	Comment	Source reference [a]
Piping	Rupture	3,0E-9/hm	6,5E-6	(3,0E-9/hm)*30m*72h	E.A.
DHX1	Failure to operate	3,0E-5/h	2,16E-3	(3,0E-5/h)*72h	I
DHX2	Failure to operate	8,30E-7/h	5,9E-5	(8,30E-7/h) *72h	I
Isolation valve	Failure to close	1,3E-3/d	2,60E-3	(1,30E-3/d)*1demand*2(one inlet and one outlet valve)	I
Isolation valve	Failure to close CCF	1,3E-3*0,1	1,30E-4	(1,30E-3/d)*1demand *0,1(beta factor to take into account CCF)	E.A.
Check valve	Failure to open	1,0E-4/d	1,0E-4	(1,0E-4/d) *1 demand	I
Water tank	Unavailability	9,3E-7/d	9,3E-7	(9,3E-7/d)*1 demand	I
Guard vessel	Failure	1,0E-7/h	7,2E-6	(1,0E-7/h)*72h	E.A.
	Total system failure probability		6,96E-3		

[a] Reliability data sources: (I) GFR reliability analysis methodology (ref. 6); E.A. Engineering Assessment,

The failure probability for one single loop is accomplished by summing up all the failure probabilities pertaining to the single modes of failure. Mission time of 72 hours is supposed, according to the grace period for the passive systems. The total failure probability for all the three redundant “passive” systems, including the CCFs among the loops, is 1,3E-4.
Conversely, as regards the system operation in the “active” mode, the relative analysis will include also the failure modes associated with the blower required to assure the forced circulation, as shown in Table II. The total failure probability for all the three redundant “active” systems, including the CCFs among the loops, is 1,33E-4: as can be seen the two reliability figures in connection with the two conditions are quite comparable, meaning that the influence of the additional components (that is the blower) to assure the operation in the active mode is unimportant.
Finally, as pointed out in section 2, it would be appropriate to consider additionally the “hybrid” active/passive operation mode, since the system performance in the depressurized condition is driven by the combination of both active (battery-powered blower) and passive means (natural circulation, sustained by backup pressure) respectively in the short (24 hours) and long terms: in this case the failure probability of the system is obtained by the combination of both active and passive circumstances, by summing up the corresponding probability figures. Table III summarizes the distribution of the failure probabilities by operation mode and corresponding event.

Table II. Failure probabilities for active DHR loop (1 out of 3)

Component	Failure Mode	Failure rate/h-probability/d	Failure probability	Comment	Source reference [a]
Piping	Rupture	3,0E-9/hm	6,5E-6	(3,0E-9/hm)*30m*72h	E.A.
DHX1	Failure to operate	3.0E-5/h	2,16E-3	(3,0E-5/h)*72h	I
DHX2	Failure to operate	8,30E-7/h	5,9E-5	(8,30E-7/h) *72h	I
Isolation valve	Failure to close	1,3E-3/d	2,60E-3	(1,30E-3/d)*1demand*2(one inlet and one outlet valve)	I
Isolation valve	Failure to close CCF	1,3E-4/d	1,30E-4	(1,30E-3/d)*1demand *0,1(beta factor to take into account CCF)	E.A.
Check valve	Failure to open	1,0E-4/d	1,0E-4	(1,0E-4/d) *1 demand	I
Water tank	Unavailability	9,3E-7/d	9,3E-7	(9,3E-7/d)*1 demand	I
Blower	Failure to operate	1,0E-3/d	1,0E-3	(1,0E-3/d)*1 demand	I
Blower	Failure to run	1,0E-4/h	7,2E-3	(1,0E-4/h)*72h	I
Guard vessel	Failure	1,0E-7/h	7,2E-6	(1,0E-7/h)*72h	E.A.
Total system failure probability			1,52E-2		

[a] Reliability data sources: (I) GFR reliability analysis methodology (ref. 6); E.A. Engineering Assessment,

Table III. Summary of reliability results

Mode of operation	Event	Single circuit failure probability	System failure probability
Passive	Pressurized condition[1]; Long term depressurized condition [2]	6,96E-3	1,3E-4
Active	Short term depressurized condition[2]	1,52E-2	1,33E-4
Hybrid	Depressurized condition[2]	2,63E-4	

[1] as a consequence of LOF (Loss of Flow)

[2] as a consequence of LOCA (Loss of Coolant Accident)

At the end the total failure frequency for the loss of the DHR function is achieved by combining the unreliability of the envisaged DHR system (either via passive or active units arrangement or hybrid) with the number of demands per year of the system, which can be assumed corresponding to the frequency of any initiating event resulting in system intervention, assumed conservatively equal to 1,0E-1/ry: this results in a failure frequency value which doesn't meet the sought reliability target of 1,0E-7/ry.

This result clearly exhibits the need of implementing additional provisions to achieve the reliability goal to cope with events in order to keep the CDF below 10^{-6}/year at full operating power. This implies that, because of a high reliability target, an appropriate degree of redundancy and diversity has to be introduced into the DHR systems to exclude common mode failures: in fact the analysis stresses the relevance of the CCFs among the different loops of each DHR system on the total reliability.

Additionally the combination of three loops working either in the passive or active mode leads to an operational diversity of the systems for all the operational and accidental states.

Passive system reliability

Up to now the passive system has not been seen from the point of view of the physical process failure. As it is recognized the unreliability of passive systems has mainly two types of contribution. One is from the failure of hardware like coolant boundary failure, referred to as hardware failure. The second contribution, which is usually neglected for active systems, is from uncertainty in achieving the intended design function, referred as functional reliability (ref. 10). This arises from the onset of physical phenomena occurring during the system operation, and by variation of critical parameters which may affect the performance of the system and ultimately lead to its failure. Thus probability of failure of the system due to parameter variations without failure of a component, i.e. functional failure, is a distinguishing characteristic of passive systems, to be addressed within safety and availability analysis.

Most methodologies for passive system reliability assessment prompted so far suggest propagation of important system uncertainties through Monte-Carlo simulation of a detailed best estimate mechanistic system (ref.11). Monte-Carlo Simulation (MCS) is well known to be robust to the type and dimension of the problem. Its main drawback, however, is that it is not suitable for finding small probabilities (e.g., $PF \leq 10^{-3}$), because the number of samples required to achieve a given accuracy, is proportional to 1/PF. Passive safety systems of innovative nuclear designs are usually designed to meet very high reliability values. As a result, Monte–Carlo simulations incorporated in the functional reliability analysis methodologies should require a large number of system analyses using best estimate system code. Typically mechanistic thermal hydraulic codes of complex nuclear safety systems are computationally expensive and MCS using such models require considerable and often prohibitive computational effort to achieve acceptable accuracy.

One common and straightforward approach to reduce computational effort is adopting approximate solutions which require less computational effort for evaluating system response. The simplest and most direct way to tackle this issue is modeling the passive system by relating to the modeling of the unreliability of the hardware components of the system: this is achieved by identifying the hardware failures that degrade the natural mechanisms upon which the passive system relies and associating the unreliability of the components designed to assure the preeminent conditions for passive function performance (ref. 12), e.g. heat exchanger plugging is assumed to impact the degradation of heat transfer coefficient. Thus, the probabilities of degraded physical mechanisms are reduced to unreliability figures of the components whose failures challenge the successful passive system operation. If, on the one hand, this approach may in theory represent a viable way to address the matter, on the other hand, some critical issues arise with respect to the effectiveness and completeness of the performance assessment over the entire range of possible failure modes that the system may potentially undergo and their association to corresponding hardware failures. In this simplified methodology, degradation of the natural circulation process is always related to failures of active and passive components, not acknowledging, for instance, any possibility of failure just because of unfavorable initial or boundary conditions. The adoption of this approach in the present case entails the resorting to the previous analysis, with regard to the system operation in the passive configuration.

Recently, a promising method based on the concept of functional reliability, within the reliability physics framework of load-capacity exceedance (ref.10) has been proposed. The functional reliability concept is defined as the probability of the passive system failing to achieve its safety function as specified in terms of a given safety variable crossing a fixed safety threshold, leading the load imposed on the system to overcome its capacity. In this framework, probability distributions are assigned to both safety functional requirement on a safety physical parameter (for example, a minimum threshold value of coolant mass flow required to be circulating through the system for its successful performance) and system state (i.e., the actual value of coolant mass flow circulating), to reflect the uncertainties in both the safety thresholds for failure and the actual conditions of the system state. Thus the mission of the passive system defines which parameter values are considered a failure by comparing the corresponding pdfs according to defined safety criteria. Within this method, devised by ENEA, the selection and definition of the probability distributions, that describe the characteristic parameters, are based mainly on subjective/engineering judgment.

According to the functional reliability approach, illustrated in the previous section, the system is represented with a resistance/stress (*R*-*S*) model or load/capacity model, borrowed from reliability physics, with probabilistic density distributions chosen accordingly (see Figure 2), to define the probability of failing to successively carry out a given safety function (e.g. decay heat removal).

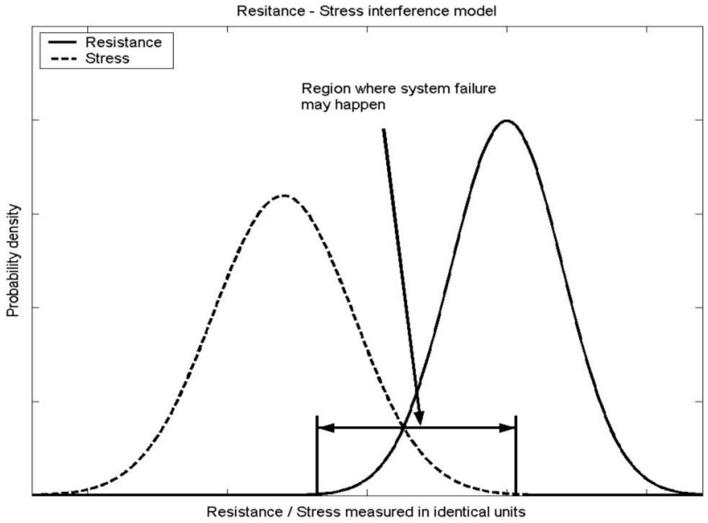


Figure 2. Resistance-Stress interference model

In the framework of T-H passive systems reliability assessment, *R* and *S* express respectively the safety functional Requirement (*R*) on a safety physical parameter (for example, a minimum threshold value of He mass flow required to be circulating through the system for its successful performance) and system State (*S*) (i.e., the actual value of He mass flow circulating). Probability distributions are assigned to both *R* and *S* to reflect the uncertainties in both the safety thresholds for failure and the actual conditions of the system state. The function of the passive system defines the safety parameter values that characterize system failure, whose probability is obtained by comparing the state probability density function with that of the defined safety functional requirement. Thus the probability that the achieved He flow rate is less than the required He flow rate to assure the natural circulation is the convolution product of the probability distributions.

In our analysis mass flow rate is assumed as characteristic parameter of the system performance and failure criterion is chosen according to relation 1:

$$Z_s / Z_n \leq 0{,}8 \qquad (1)$$

Z_n is the mission requested nominal value for natural circulation

Z_s is the actual value

Assuming *S* and *R* respectively as the actual mass flow rate and its critical value below which free convection is stopped the probability of failure of the system is given by:

$$P_f = P(W_s - W_r < 0) = \iint\limits_{W_s - W_r \leq 0} f(W_s) f(W_r) dW_s dW_r \qquad (2)$$

Where: W_s actual flow rate value

W_r minimal flow rate value required for natural circulation (i.e. $W_r = Z_n$ *0,8 from eq.1)

$f(W_s)f(W_r)$ is equivalent to the joint probability distribution of the parameters W_r,W_s (independent individual distributions)

As already stressed the difficulties lie in the definition of the probability distributions $f(W_s)$and $f(W_r)$ of *S* and *R* respectively, from which the failure probability is derived. In lack of reliable data, engineering judgment must again be used to obtain such distributions.

In order to duly characterize the representative parameter on the probabilistic standpoint (i.e. ranges and distributions), as a general rule, a central pivot has to be identified, and then the range are to be extended to higher and lower values. The pivot value represents the nominal condition for the parameter, and the limits are chosen in order to exclude unrealistic values or those representing a limit zone for the operation demand of the passive system. Probability values are peaked to the nominal value and decrease gradually towards the minimum and maximum allowed values: thus the lower the probability values, the wider the "distance" from the nominal value. On one hand a uniform probability would be suitable to describe the parameter if one considers the design range of the parameter (that is the pdf would be uniform); on the other hand the values of the flow rate could correspond to the end status of a system transient or accident. No finalized study has been carried out to propose the relevant probability distributions. For instance, very simply, such a situation could derive from the occurrence of a failure mode impairing the natural circulation performance, like e.g. thermal stratification: therefore normal distributions are considered, the mean value being the nominal one and confidence limits the ones corresponding to two standard deviations of the mean. It's worth noticing that the ranges defined by two standard deviations roughly cover the 95% confidence interval, considering that the two-sided 95% confidence interval lies at ± 1,96 standard deviations from the mean value.

In this treatment one evaluates the failure probability/unreliability of the free convection by accounting for relationship 1 and 2, considering directly the distribution of the parameter *W*: this, besides simplifying the problem, allows to go straightforward to the problem.

Table below shows the parameters of interest of the normal distributions, with reference to the flow-rate *W*, across the system. The value of the expected natural circulation flow rate corresponds to the 3% of the flow rate of the reactor operation at full power.

Table IV. Normal pdf characteristics

Parameter	Range(a-b, kg/sec)	Characteristics (kg/sec)

Wr	1,2 – 1,6	μ = 1,4
		σ = 0,1

W_s	1,5 - 1,9	μ = 1,7
		σ = 0,1

As ultimate step the probability value is assessed by solving equation 2: the estimated failure probability of the natural circulation is equal to 1,8E-02. This value is quite dependent on the assumptions taken in the probabilistic model: for instance normal distributions and associated characteristics are postulated to characterize the uncertainties related to the performance parameters. Nevertheless the analysis highlights the natural circulation defiance as a risk factor for the system availability and safety. This value should be integrated with the previous outcome of the reliability analysis concerning passive system reliability, leading in any case to a small deterioration of the whole system reliability, as shown in Table V. The reason for the single loop failure probability significant increase due to functional failure vs the slight growth in system total failure probability, as compared to Table III, lies in the CCF as dominant contributor in the final reliability figure.

Table V. Summary of reliability results (functional reliability included)

Mode of operation	Event	Functional reliability	Single circuit failure probability	System failure probability
Passive	Pressurized condition[1]; Long term depressurized condition [2]	1,8E-2	2,5E-2	1,45E-4
Active	Short term depressurized condition[2]		1,52E-2	1,33E-4
Hybrid	Depressurized condition[2]	2,78E-4		

[1] as a consequence of LOF (Loss of Flow)

[2] as a consequence of LOCA (Loss of Coolant Accident)

Main findings

Failure probabilities are calculated on various system configurations by integrating the probabilities of occurrence of corresponding hardware components and natural circulation reliability. These results show as well that passive system configuration is more vulnerable to failure than active system configuration, because of their proneness to functional failure, which therefore constitutes a risk factor for the system performance.

3.2 Reactivity control safety systems

This section looks at different special shutdown systems specifically engineered for prevention of severe accidents, to be implemented on Fast Reactors, with main focus on the investigation of the characteristics and performance of passively actuated shutdown systems in Sodium Fast Reactors.

Some innovative approaches suited to improve the overall plant safety have been analyzed, in particular as regards the design options and solutions for passively activated safety devices and/or reactivity feedbacks that would allow the fuel and core design to meet the safety objectives.

The unprotected transients, also denoted as “Anticipated Transients Without Scram” (ATWS) are a group of beyond design basis events that can significantly challenge sodium fast reactors (SFR) safety and are used to categorize the higher probability core disruptive accident (CDA) initiators. This category results from the observation that a small group of high probability events combined with a reactor protection system (RPS) failure (no reactor scram) would lead to coolant boiling and a core melting scenario. The general probability of occurrence for the

initiator events is typically in the range of 0.1 to 0.01 per reactor year and when combined with the probability of RPS failure, 10^{-6} per reactor year, the probability for an ATWS event is in the range 10^{-7} to 10^{-8} per reactor year (ref.13).
These events include: 1) the unprotected transient overpower (UTOP); 2) unprotected loss of primary coolant flow (ULOF); 3) unprotected loss of heat sink (ULOHS); 4) unprotected loss of flow and heat sink; and 5) unprotected safe shutdown earthquake (ref. 14).
Consequently the undesirability of the consequences arising from an unprotected transient, as an energetic core disruptive accident (CDA), vessel failure, and a large early release, has led some SFR designers to consider appropriate measures to avoid the occurrence of this kind of event. One way to protect the plant from such an accident is to include a highly reliable reactor protection system (RPS) by incorporating self-actuated shutdown systems (SASS).
For this purpose the conceptual design of candidate passively activated or "self-actuated" reactor shutdown systems, which have been devised during SFR development programs are reviewed. The SASS' s, pursued during this programs, are, with the exception of their associated hardware, entirely self-contained within the reactor core structure. They incorporate devices which respond inherently to abnormal local process conditions (neutron flux, coolant flow) by passive reactivity feedback devices, thus shutting down the core independently of the RPS.
Current SFR plant protection systems employ two reactor shutdown systems having redundancy of control rod worth and diversity of design. Some of the diversity occurs naturally due to the differing operational requirements of the two systems: the primary system controls reactivity during normal operation, whereas the secondary system is used only to shut down the reactor. Reactor scrams are initiated in response to certain combinations of off-normal signals from sensors which detect neutron flux, sodium temperature, pump speed, etc.. This involves a chain of sub-systems from the sensors, through the logic circuits, amplifiers, electro-magnetic actuators and control rod release mechanisms, culminating in the insertion of the control rods. However, despite proven component reliability, good inspection access and preventive maintenance, there may be a very low probability that an unacceptable change in the fuel element power level or cooling conditions does not result in reactor shutdown. It is argued that the failure of commonly affected links in both of the above-mentioned chains which could possibly occur, for example, during a severe earthquake, requires consideration of events involving inability to scram the reactor. With this possibility, combinations of occurrences could be postulated which might result in a CDA (Core Disruptive Accident).
As previously pointed out, particular faulted condition occurrences, of interest could be for instance:

a. Pump loss with failure to scram, referred to as loss of flow (LOF) events,

b. Uncontrolled rod withdrawal with failure to scram, producing a transient overpower (TOP) event.

Under all these circumstances, intensive efforts have led to the introduction of some innovative control systems, to be investigated, with main focus on their performance analysis in order to assess their suitability to meet the safety requirements.
At first the concept of inherent safety vs the concept of passive safety are illustrated and the fundamental functional requirements of the safety systems devoted to the safety shutdown are recalled.

Inherent safety vs passive safety

It is important to underline the difference between inherent safety and passive safety. In fact IAEA, as in ref.15, cautions about the misuse of safety related terms such as passive and inherent safety particularly with respect to advanced nuclear plants, generally without definition and sometimes with definitions inconsistent with each other.

Inherent or Intrinsic Safety refers to the achievement of safety through the elimination or exclusion of specific hazards through the fundamental conceptual design choices made for the nuclear plant. Elimination of all these hazards is required to make a nuclear power plant inherently safe. Since this appears to be impossible, it is recommended to avoid the unqualified use of "inherently safe" for an entire nuclear power plant or its reactor.

On the other hand, a reactor design in which one of the inherent hazards is eliminated is inherently safe with respect to the eliminated hazard. An inherent safety characteristic is a fundamental property of a design concept that results from the basic choices in the materials used or in other aspects of the design which assures that a particular potential hazard can not become a safety concern in any way. In this case no changes of any kind, such as internally or externally caused changes of physical configuration can possibly lead to an unsafe condition.

When an inherent hazard has not been eliminated, engineered safety systems, structures or components are provided in a design to make its use acceptable without undue risk. Such provisions generally aim to prevent, mitigate, or contain potential accidents.

On the other hand the concepts of active and passive safety describe the manner in which engineered safety systems, structures, or components function and are distinguished from each other by determining whether there exists any reliance on external mechanical and/or electrical power, signals or forces. The absence of such reliance in passive safety means that the reliance is instead placed on natural laws, properties of materials and internally stored energy.

Thus inherent safety characteristic refers to the safety achieved by the elimination of a specified hazard by means of the choice of material and design concept, while passive safety feature refers to the safety achieved by the operation of a passive system, i.e. which does not need any external input to operate.

The adoption of these self actuated safety systems obeys more to the inherent safety principle, which is the pursuit of designing hazards out of a process, as opposed to using engineering or procedural controls to mitigate risk. Therefore the plant design safety is improved inherently because of its essential characteristics, those which belong to the process by its very nature, making possible to avoid and remove hazards rather than to control them by relying on “add-on” safety features, in particular by eliminating or reducing their likelihood of occurrence.

Unfortunately the lack of information complicates the evaluation of the inherent safety and the judgment of a reliability figure. As initial step, it is important to analyse the factors that may affect the inherent safety of the plant, by systematically evaluating the inherent safety characteristics. This could be achieved by identifying the basic principles of inherent safety to be described at first as parameters and finally as inherent safety indexes. For instance, from the various system description temperature and pressure are parameters suitable to illustrate the inherent safety as direct measures of the specific intrinsic property.

In general passive and inherent safety devices may, between other measures, contribute to reduce the probability of a CDA and, to a large extent, increase the overall safety level of nuclear reactors. For this reason such systems are considered while designing the fourth generation of advanced nuclear reactors.

In the present case of the SASS safety of a reactor refers to the inherent safety of a reactor that can be achieved through additional systems (to the main safety system), referred to as inherent safety systems. Safety achievement in the emergency situations, in case of unavailability of the devoted safety systems, implies that the reactor can be shut down and the decay heat can be removed off the reactor automatically by means of the natural processes process (the gravity, the coolant flow, the thermal principles, etc.). In this way the reactor is brought into a safe permanently sub-critical state and temperatures are kept well below the boiling point of the coolant.

The main designing principle of inherent safe shut-down system is: under an accident condition, even if reactor protection system RPS does not work, reactor can be shut down only by inherent passive reactivity feedback mechanism. Therefore it is a functional requirement for the system to be able to handle all the unprotected transient events, as previously listed.

As a result, the principal objective is to provide an independent safety functional shutdown system which will: inherently respond to the off-normal conditions accompanying these events; be entirely self-contained within the reactor core control assemblies; and, be completely unobstructed by action of the RPS or any other external effect. It would thus be immune to power failures, structural dislocations between the head and the core, or wrong inferences of fuel conditions from combinations of indirect sensor measurements.

In addition the system shall be designed in such a way as to maximize the fail-safe characteristic. These systems are required as well to withstand the impact of internal and external hazards, such as severe earthquake, in such a way that no risk will be posed to their operation.

Incorporation of such a shutdown system, having the same response time and negative reactivity insertion as the secondary RPS, will substantially reduce the probability of failure to scram in a timely manner and, consequently, will help avoid excessive licensing conservatism with respect to CDAs. These factors may represent a substantial benefit to the reactor operator in the form of reduced uncertainty of licensing requirements, which will in turn result in reduced plant costs.

Currently there are not relevant availability/reliability studies or significant databases which would fit for reliability estimates or performance evaluation, being available only some experiments on test devices limited to some of them. In this context probabilistic studies will be required to assess the fulfilment of the safety criteria, once specific reliability/availability targets, in terms of probability per demand, will be set.

Finally it’s noteworthy that the introduction of these kind of systems implement the criteria for redundancy, diversity and independence improving the safety and the reliability of the whole shutdown system.

Passive shut-down systems in sodium fast reactors development

SASS are simple, inexpensive design modifications that induce scram when core temperatures and/or coolant flow rates reach certain design limits. These devices have the potential to diversify the RPS, add to defense-in depth, and increase scram reliability.

Over a dozen designs for these systems have been proposed, and the twelve such designs considered are listed below (ref.16):

1. Lithium Expansion Module
2. Lithium Injection Module

3. Curie Point Latches
4. Thermostatic Switches
5. Fusible Link Latches
6. Thermal Volumetric Expansion Drives
7. Flow Levitated Absorbers
8. Cartesian Diver
9. Sodium Injection
10. Enhanced Thermal Elongation of Control Rod Drive Line
11. Gas Expansion Module
12. Periphery Channels for Na Voiding

In the following only some of them will be considered, for which there are enough information to provide a sufficiently detailed description, since unfortunately for other concepts the information is quite poor or they are at a very beginning development phase and thus they are not included in the present analysis (ref. 17).

Lithium expansion module

The concept of the Lithium Expansion Module (LEM) for inherent reactivity feedback is illustrated in Figure.3. The LEM consists of an envelope of refractory metal in which liquid poison of 95 per cent enriched ^{6}Li is enclosed. ^{6}Li placed in the positive void region of the core would result in negative reactivity insertion because neutron absorption by ^{6}Li dominates over scattering.

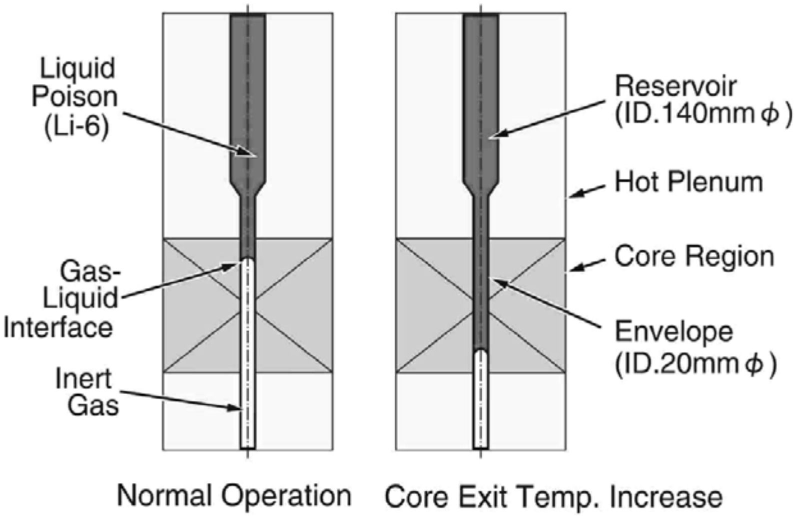


Figure 3. LEM concept

Lithium-6 is suspended in the upper part of the envelope by surface tension exerted on the gas–liquid interface. The LEM is actuated by the volume expansion of ^{6}Li itself. If the core outlet temperature increases, the gas–liquid interface goes down and negative reactivity insertion can be achieved.

Accordingly, it is effective to mitigate the anticipated transient without scram. The gas–liquid interface in the nominal operation is placed at the active core top. If the core outlet temperature decreases, the gas–liquid interface increases and no positive reactivity insertion is expected. Drawbacks related to this system concern the untested reliability and safety concerns as the boundary failures and the lithium hazards, including the chemical reactions. Boundary failures could result in gas release within the core with insertion of positive reactivity, depending on the position of LEM. Another issue pertains to the stability of the gas-liquid interface: in fact if the buoyancy force exceeds the surface tension, the gas–liquid interface will be broken. The critical diameter of the LEM envelope is determined by the balance between the surface tension, which is dependent on the temperature, and buoyancy force.

Lithium reacts not only with oxygen, like sodium, but also with nitrogen, so that to avoid lithium fires it is necessary to utilize other gases, as helium. In addition there is the risk of beryllium formation due to lithium absorptions of neutrons.

Lithium injection module

In the recent years LIM innovative system has been proposed for the RAPID fast reactor concept in Japan. The concept of the Lithium Injection Module (LIM) for inherent ultimate shutdown is illustrated in Figure 4. The LIM also comprises an envelope in which 95 per cent enriched ^{6}Li is enclosed. If the core outlet temperature exceeds the melting point of the freeze seal, ^{6}Li is injected by a pneumatic mechanism from the upper into the lower region to achieve negative reactivity insertion. In this way the reactor is automatically brought into a permanently subcritical state and temperatures are kept well below the boiling point of lithium (1330 °C).

The time required for reactivity insertion of the LIM is 0.24 s, which is shorter than the time required for free drop of conventional scram rods (i.e. as much as 2 s).

Similarly to LEMs, LIMs assure sufficient negative reactivity feedback in unprotected transients, like UTOP and ULOF. The role of the LIM is to provide variety and redundancy of inherent safety in unprotected transients. Either LEMs or LIMs can meet such transients independently.

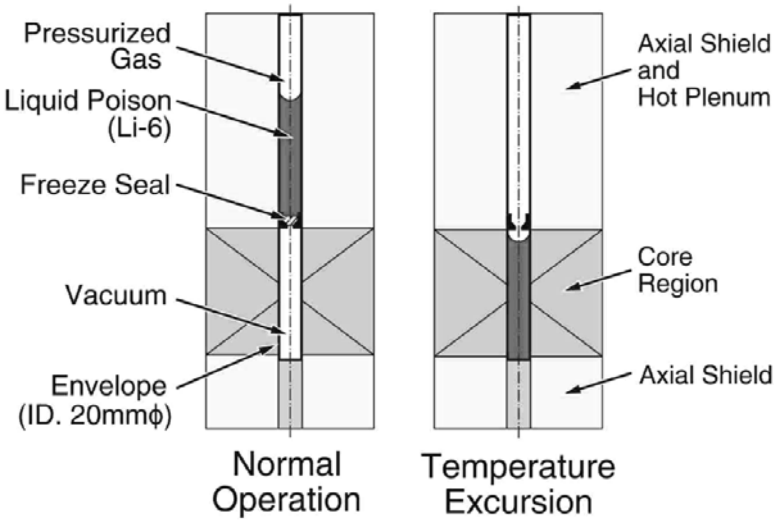


Figure 4. LIM concept

The injection temperature, which depends on the requirements of the core design, can be selected from several candidate materials of the freeze seal support. Thus freeze seal design is the key issue to ensuring accurate injection temperature over the design lifetime. The freeze seal segment consisting of CuNi alloy (trade name L-30) is to assure an injection temperature of 1240 °C. When adopting Al for the freeze seal, LIM injection would be performed at 660 °C. This innovative concept has undergone some experimental verifications of its performance, as injection tests on the LIM specimen at quasi-steady-state heat up, to demonstrate the freeze seal function. LIM freeze seal function has been confirmed by experiments including long life behaviour, as reported.

Main issues concern firstly the integrity of the LIM envelope that must be assured during reactor operation; in addition the irradiation behaviour of the freeze seal must be assessed, especially as regards the time span of fuel design life or core design life. Also the freeze seal performance is to be assessed in terms of prompt response to abnormal events leading to excessive core outlet temperatures. The use of lithium poses the same challenges with respect to the LEM concept.

Curie point latches

The Curie point electromagnet SASS consists of an electromagnet and an armature that are parts of its magnetic circuit containing a temperature-sensing alloy as shown in Figure 5. The magnetic force is abruptly lost when the alloy is heated up to its Curie point by the heated coolant from the core. Then the armature de-latches at the detach surface and drop together with the control rod into the reactor core. The Curie point SASS is a simple structure and has flexibility of the detaching position.

The representative of this system is SASS designed for the commercial fast reactor Demonstration Fast Breeder Reactor (DFBR) design study in Japan.

As in the previous case the system performance is to be assessed in terms of system prompt response to abnormal events leading to excessive core outlet temperatures,

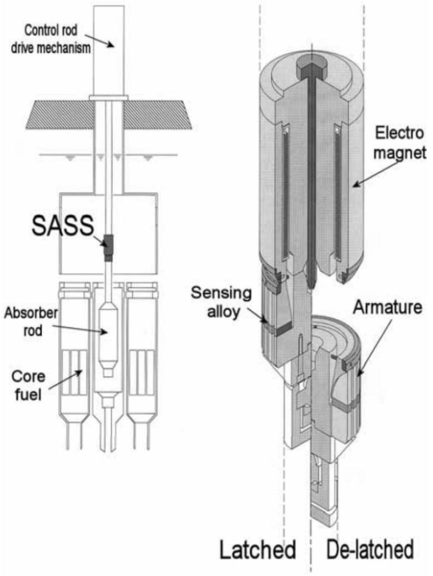


Figure 5. Curie point self-actuated Shutdown System

Gas expansion module

Gas Expansion Module (GEM) is essentially a passive shut-down device to insert negative reactivity during a primary system unprotected loss of coolant flow (ULOF). The device is basically a hollow removable sub-assembly sealed at the top and open at the bottom. The gas trapped inside the sub-assembly expands when core inlet pressure decreases due to flow reduction and expels sodium from the sub-assembly. Neutron leakage increase and negative reactivity is inserted, as shown in Figure 6. By using GEM, an additional negative reactivity feedback is induced in the core by an increase of the neutron leakage, caused by the lowering of the coolant level due to the decrease of the coolant pressure at the core inlet under the loss-of-flow conditions. However, GEM is not sufficient in large cores and produces negative reactivity on loss of hydraulic pressure only.

GEM is conceived for Advanced Liquid Metal Reactor (ALMR), USA, and KALIMER-150 (Republic of Korea) demonstration reactors.

The integrity of the envelope has to be assured in order to avoid the gas ingress into the core and the consequent positive reactivity insertion.

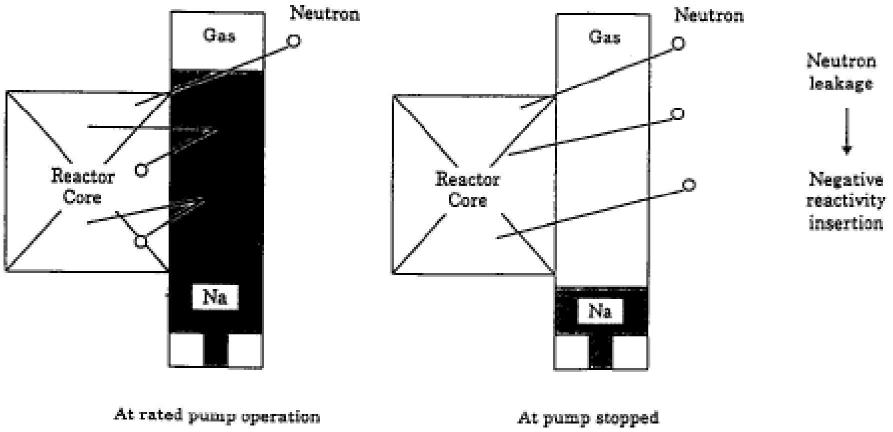


Figure 6. Design concept of GEM

Main findings

High reliability for the reactor shut-down system (RSS) based on two independent active RSS and one additional innovative RSS is envisaged. For their specific characteristics (safety implementation through inherent features with respect to passive safety) SASS should be deemed more reliable than active shutdown systems, although their inclusion in the designs poses some challenges to be addressed.

In fact reliability for most of them is as yet unconfirmed, being their performance only tested on some specific experimental apparatuses: therefore it is not possible to rank them upon the reliability and availability figures of merit: as an initial evaluation one can assert that the more reliable device is the Curie Point Latch, since it can be considered as the closest to a more conventional shutdown system. This applies also as regards the state of development of the system.

Potential for inadvertent actuation is not suitable to be considered a concern, since the system performance is initiated by physical principles triggered by some accident conditions, implying abnormal temperature increase at the rector core outlet.
Another final issue to be considered in future fast reactors is the simplicity of operation: considering the premature nature and the lack of experimental validation, further work is necessary in order to determine the practicability of the present design concepts.

4. Sequence-based comparison active vs passive

4.1 DHR strategy for GFRs

The main requirement of each DHR strategy is to remove the residual heat from the core after the reactor shutdown after operational state and during fault conditions.
As previously said, the key safety issue for Gen IV GFR is the decay heat removal, due to the little thermal inertia of the gas, which results in rapid overheating of the core consequently to the loss of cooling functions. DHR systems must provide proper and reliable cooling of the core, both under pressurized and (especially) depressurized conditions.

Decay Heat Removal Strategy
The following consideration are based on CEA 2006 DHR for GFR Strategy (see ref.18). The following conditions are considered as reference: after all accidental cases the reactor is tripped, the power conversion system is detached and dedicated DHR loops are connected to the vessel.
2 different cases related to the primary side pressure during the accident sequence can be discriminated:

- Accident during which the pressure stays at its nominal value (LOFA)
  During this accident the decay heat is removed by blowing the helium through the DHR-HXs using the DHR blowers.
- Accident during which primary side is depressurized (LOCA)
  In case of a LOCA, the helium pressure within the primary loop would decrease up to a minimum level which would avoid both the possibility of natural circulation (He density is too low at low pressure) and the possibility of forced circulation through the DHR blowers(due to the He low density, an excessive pumping power would be required). For depressurizing accidents (i.e. LOCA) the strategy relies on a small containment, called Guard Containment (GC), surrounding the primary side and the DHR loop, which limits the depressurization of the system, guaranteeing ≈ 10 bar(back-up condition) inside the primary loop. DHR blowers are designed to work both under nominal and back-up pressures, so that it can be used to circulate He within the DHR loop (and hence cooling the core) even in case of an accident that causes the loss of primary pressure (under back-up pressure condition).

In case of loss of DHR blowers, the strategy is different for the two situations stated above.

- Pressure at its nominal value (LOFA)
  If the DHR blowers are not available and the pressure of the primary side is at its nominal value, the decay heat is removed by natural circulation in the DHR loops.

- Depressurized events with back-up pressure condition (LOCA)
  Depressurization of the primary side (i.e. LOCA) leads to low He density. If the He density is too small, it will results impossible to circulate the gas into the DHR loops both by the action of the blowers (the energy required to pump the gas, which is almost inversely proportional to the square of the gas pressure, would exceed the maximum power of the blowers) or by natural circulation (at low density the mass flow rate ensured by natural circulation is too small to allow the decay heat removal of the heat decay generated inside the core). Since in the back-up condition, the pressure is too low to ensure a proper decay heat removal by natural circulation, during the first 24h after the accident (when the decay heat power is still large), the cooling function is ensured by the action of the blowers, which are powered by the emergency diesel generators (EDGs) in case of Loss Of On-site Power (LOOP), or by dedicated batteries, able to supply the blowers for at least 24h, in case of SBO. After 24 hours, the decay heat power is sufficiently low (< 1% of its nominal value) to allow a proper cooling only by natural circulation.

Decay Heat Removal Strategy for depressurized events, during which the back up pressure condition is maintained, can be summarized as follow:

- Depressurization event (i.e. LOCA) begins;
- Reactor trip;
- The GC maintains the primary side pressure to an intermediate level (Back Up Pressure), high enough to ensure proper decay heat removal by forced circulation.
- The decay heat power is removed by forced circulation through the DHR blowers for the first 24 hours following the accident. In case of LOOP the blowers are powered by the EDGs, while under SBO condition the blowers are powered by dedicated batteries ensuring the power supply for at least 24h.
- After 24hours from the beginning of the accident, the decay heat power is sufficiently low (<1% of the nominal power) to allow the proper cooling of the reactor only by natural circulation through DHR loops.

The DHR strategy for depressurized events, during which the back up pressure condition is assured, is a combination of active and passive approaches, since a complete passive DHR would not able to guarantee a proper cooling of the core, especially during the earlier accidental stages.

- Depressurized events with back up pressure condition not assured
- In case of the GC failure, which would lead to a pressure lower than the back up pressure condition, the primary pressure would not allow to achieve a core cooling condition; hence this accidental scenario leads to a Severe Accident.

Figure 7 summarizes the DHR Strategy for the 3 reference conditions (pressure at its nominal value, depressurized event with Back Up Pressure andLoss of Back Up Pressure - LOBP) stated above. The Back-Up Strategy refers to the case of loss of DHR blowers power supply (LOOP or SBO).

DHR Strategy		
Accident	Reference	Back-Up (Accident + SBO)
LOFA (pressure at nominal value) 70 bar	Forced Convection DHR Blowers (power supplied by grid)	Natural Convection (power supply not necessary)
LOCA (back-up pressure) 10 bar		Combined active/passive (power supply by batteries in case of SBO)
LOCA + loss of GC integrity 2 bar	Severe Accident	

Figure 7. DHR strategy

In conclusion, the DHR Strategy, in case of loss of normal power supply to DHR Blowers (Back-Up Strategy), relies on:

- natural convection to ensure adequate heat removal in case of high pressure scenario;
- forced DHR circulation in case of low pressure scenario (back up pressure maintained), with emergency power supply (EDGs and batteries connected to the DHR Blowers) able to start and/or maintain forced circulation. For the long term, the back up pressure condition is sufficient to ensure adequate cooling by natural convection.

4.2 Demonstrator DHR system requirements and design

As stated in CEA DHR Strategy, the demonstrator DHR System, which has to start to operate just after the reactor shutdown, must provide the following functions:

- natural convection flow after shutdown, for pressurized events;
- back up pressure condition (10 bar), for depressurized events;
- forced convection flow for the first 24 hours after the shutdown, for depressurized events.

All those functions are provided by dedicated DHR loops, properly isolated from the primary loop by isolation valves, which are activated after the reactor trip.
In addition, the Gen IV safety requirements state that the DHR System expected failure frequency has to be lower than 10^{-7}eventsper year.

$$f \leq 10^{-7} \text{ [y-1]}$$

In order to achieve this frequency target, the system design has to be designed in accordance with the following safety principles:

- redundancy;

- diversity;
- independence.

Here below, a general overview of the demonstrator DHR system design is reported.

Demonstrator DHR System Design

The demonstrator DHR System is composed by 3 independent loops, each one able to provide a 100% of the heat removal capacity (rated at 3% of nominal core power). The DHR system is therefore characterized by a 3x100% redundancy, which enables the system to ensure a proper coolability of the core even if 2 of the 3 independent loops are out of order.

Each loop is composed by the following components:

- Isolation valve, which closes at the beginning of the reactor trip. The valve closure entails the isolation of the primary circuit and the connection across the core of the dedicated DHR loops.
- Check valve, which opens at the beginning of the reactor trip entailing the He to flow across the dedicated DHR loop.
- DHR piping, which connects the core to the DHR Heat Exchanger.
- DHR blower, which ensures forced convection flow rate through the DHR loop when natural circulation is not sufficient to properly remove decay heat power from the core (depressurized conditions).
- DHX1 (DHR Heat Exchanger 1), which removes the residual heat from the primary DHR helium loop, and delivers it to the secondary DHR pressurized water loop [5].
- DHX2 (DHR Heat Exchanger 2), which removes the residual heat from the secondary DHR pressurized water loop. The DHX2 is an horizontal heat exchanger located inside a water pool, which is able to act as final heat sink for 24 to 48 hours.
- Water Tank, which refills the water pool in order to ensure a proper heat removal for at least 72 hours.
- GC, surrounding the primary circuit and the DHR loops; its purpose is to establish an intermediate back up pressure in case of depressurized events. The back-up pressure condition foresees about 10 bar while the nominal condition has a pressure of 70 bar.

DHX1 and DHX2 are located above the core in order to ensure natural circulation of the He and water inside the DHR loop in case of both high pressure scenario (accident with pressure at its nominal value) and long term heat removal (24h after reactor shutdown) in case of depressurized events.

Figure 8 gives a schematic layout of one of the three loops of the Demonstrator DHR System. The different parts that compose the system are drawn in the following colors:

- Black: core, pressure vessel and primary circuit.
- Red: helium DHR System’s piping.
- Blue: pressurized water DHR System’s piping.
- Green: DHX2 pool and water tank.
- Purple:back-up containment.

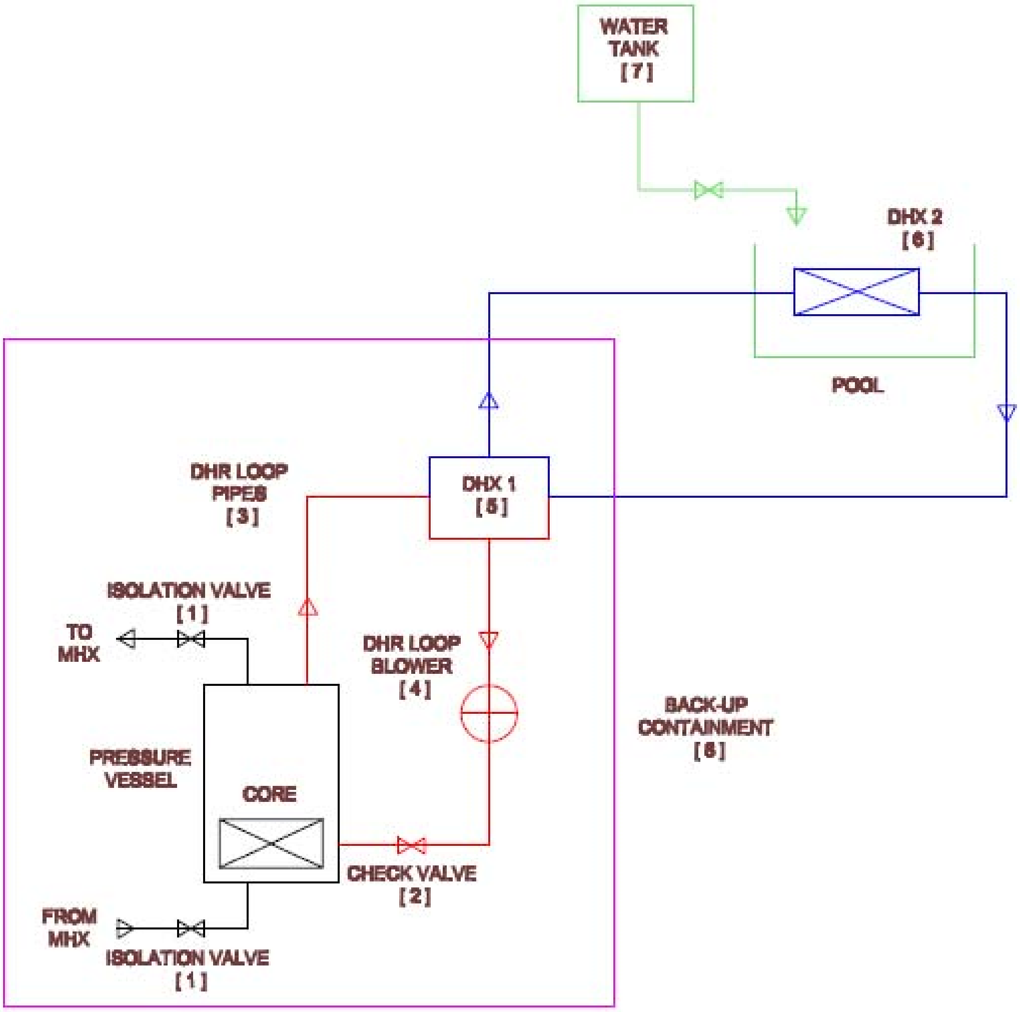


Figure 8. Scheme of a Demonstrator DHR loop

DHX1 is located about 15 m over the core in order to enable natural circulation of the helium inside the DHR loop in both pressurized and long term back-up pressure conditions. Otherwise, the DHX2 is located about 10 m over the DHX1 to assure natural circulation of the pressurized water inside the loop in any condition. Other important parameters of the DHR system are the nominal and the back-up pressure of the primary coolant.

- Nominal pressure: 70 bar;
- back-up pressure: about 10 bar.

Passive and Active modes of DHR system

As previously said the DHR system can operate in both passive and active modes. The functions needed for the DHR system to operate in passive mode are stated below:

1. reactor trip;
2. isolation valves [1] closure (to ensure the isolation of the core from the primary side of the reactor);

3. check valves [2] opening (allowing He to flow through the DHR loops);
4. natural convection inside DHR loops if the following conditions would be verified:
   - integrity of GC [8] in order to assure the back-up pressure, which is needed to establish natural convection flow rate in case of depressurized long term heat removal conditions (natural convection at nominal pressure requires the integrity of the primary side of the reactor);
   - integrity of DHR piping [3];
   - integrity and availability of both DHX1 [5] and DHX2 [6].
5. long term (72h) heat removal is assured by the availability of the Water Tank [7] to refill the DHX2 pool.

The only active component needed to operate the DHR system in passive mode is the isolation valves, which close to isolate the core from the main circuit.
The functions needed for the DHR system to operate in active mode are stated below:

1. reactor trip;
2. isolation valves [1] closure (to ensure the isolation of the core from the primary side of the reactor);
3. check valve [2] opening (allowing helium to flow through the DHR loops);
4. forced convection inside DHR loops if the following conditions would be verified:
   - integrity of GC [8] in order to assure the back-up pressure needed to establish forced convection flow rate by the action of DHR blowers;
   - operability of the DHR blowers [4];
   - integrity of DHR piping [3];
   - integrity and availability of both DHX1 [5] and DHX2 [6].
5. long term (72h) heat removal is assured by the availability of the Water Tank [7] to refill the DHX2 pool.

In order to establish forced convection flow condition, two active components for each loop (isolation valve and DHR blower) must be operating.

4.3 Probabilistic safety analysis

In this section the failure frequency value of DHR system (defined as the probability of total loss of DHR functions per reactor-year), by using the Fault Tree Analysis (FTA), will be evaluated in order to verify that this value is lower than the GEN IV reactors target:

$$f \leq 10^{-7} \text{ [y-1]}$$

The first step of the FTA is to define the Failure Rate $\lambda(t)$ of all the components of the DHR system. The Failure Rate is defined as the probability that a generic component X fails during the time step [t, t + dt], given that it was working at the time t.

If $N_w(t)$ is the number of X components working at the time t, the Failure Rate results:

$$\lambda(t) = \frac{1}{N_w(t)} \frac{N_w(t) - N_w(t+dt)}{dt} = -\frac{1}{N_w(t)} \frac{dN_w(t)}{dt} \quad \left[h^{-1}\right]$$

Integrating the previous equation between t and t+T and considering the Failure Rate as a constant (same failure rate at any given time of working), it results:

$$-\lambda \int_t^{t+T} dt = \int_t^{t+T} \frac{dN_w(t)}{N_w(t)}$$

$$-\lambda T = ln\left(\frac{N_w(t+T)}{N_w(t)}\right)$$

$$\frac{N_w(t+T)}{N_w(t)} = e^{-\lambda T}$$

Defining the failure probability of the component X during the mission time T as:

$$P_f(T) = \frac{N_w(t) - N_w(t+T)}{N_w(t)} = 1 - \frac{N_w(t+T)}{N_w(t)}$$

it is possible to define the failure probability of a generic component X, which has to work for a mission time T, known its Failure Rate:

$$P_f(T) = 1 - e^{-\lambda T} \approx \lambda T \qquad if \;\; \lambda T \;\; < 0{,}1$$

It is possible to define the failure probability of each DHR system component, for a time mission of 72h, given the Failure Rate associated to each component failure mode.
The possible component failure modes are:

- isolation valves [1]: failure to close;
- check valve [2]: failure to open;
- DHR piping [3]: rupture;
- DHR blower [4]: failure to start;
- DHR blower [4']: failure to operate;
- DHX1 [5]: failure to operate;
- DHX2 [6]: failure to operate;
- water tank [7]: unavailability;
- GC [8]: leaking.

The Failure Rate and failure probability for a mission time of 72h for each component and failure mode are given in Tab II:

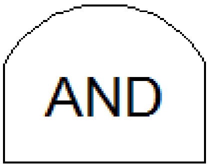
- Failure Rate of the piping is given for unit length; it has to be multiplied for the total length of the DHR loop (about 30 m). In the Table II the Failure Rate value is referred to the total length of the piping.
- Valves (isolations and check),Water Tank and blowers start to operate on demand. Their Failure Rate is given as the failure probability for single demand. This value has to be multiplied for the number of demands required to assure the DHR system functionality for the whole mission time. The number of demands for each of the four components (component mode of failure for the blower) listed above is 1.

The second step of the FTA is to assess the path that leads from the failure mode of the single component to the failure of the DHR system. Components failure modes are connected through Boolean operators:

- AND: system Y fails if fail component X1 and component X2 (both of them have to fail). If P(X1) and P(X2) are the failure probabilities of X1 and X2, the failure probability of the Y system is:

$$P(Y)=P(X_1)\cdot P(X_2)$$

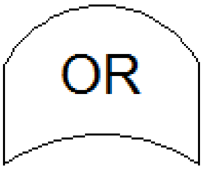
and it will represent by the following symbol:



- OR: system Y fails if fail component X1or component X2 (failure of one component leads to the failure of the whole system). If P(X1) and P(X2) are the failure probabilities of X1 and X2, the failure probability of the system Y is:

$$P(Y)=P(X_1)+P(X_2)$$

and it will represent by the following symbol:

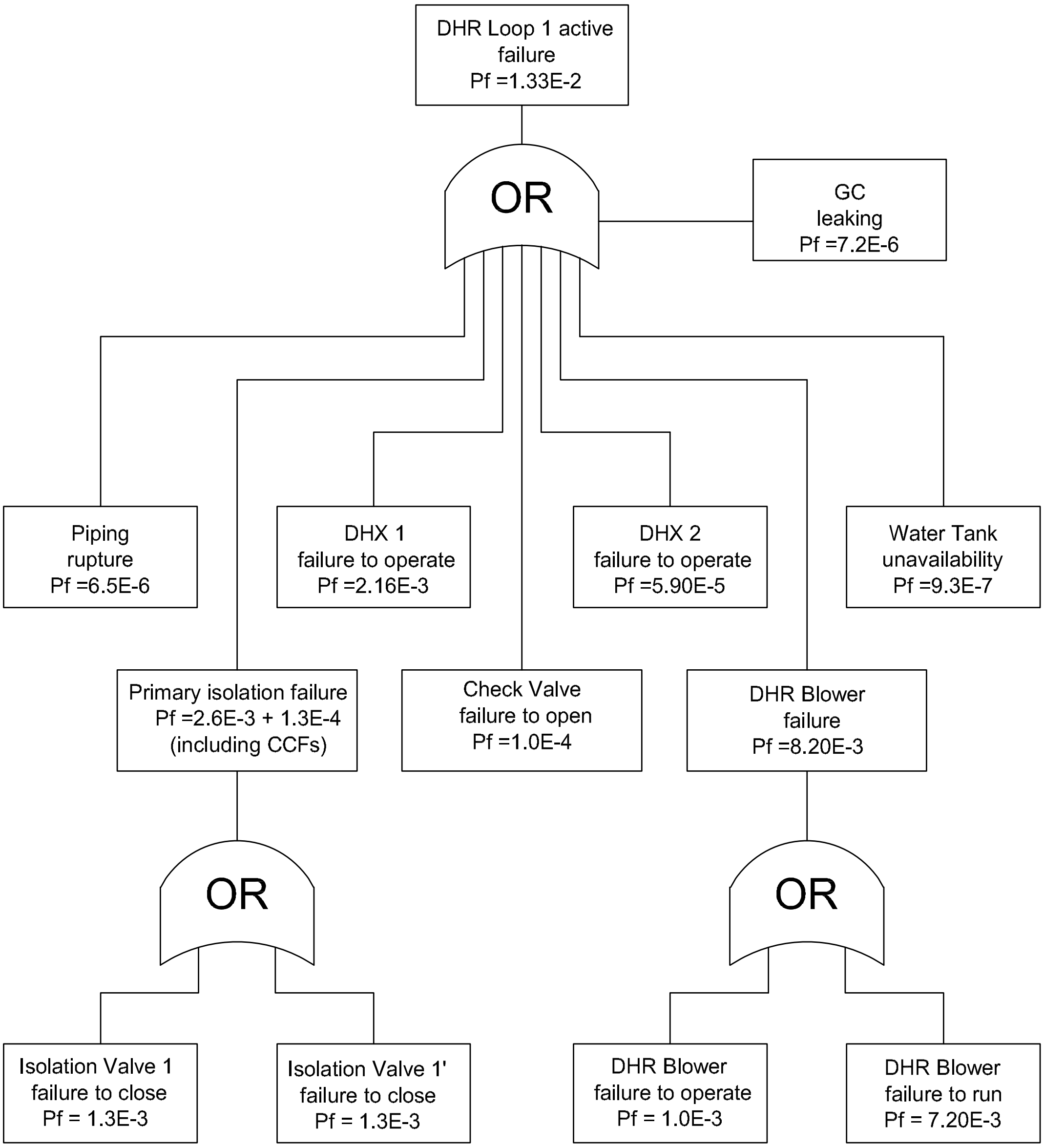


FTA of Demonstrator DHR system

DHR Operating in Active Mode

The failure of only one of the components (or components mode of failure) listed in Table II would lead to the active mode DHR system operating failure. Components failure modes are then logically connected by OR Boolean operator. The path that leads from the single component failure to the system failure is shown in the following Fault Tree.

Figure 9 shows the fault tree for possible incidental paths, which lead to the failure of one of the three DHR loops, operating in active mode (Pf is the failure probability for each component and mode of failure).



DHR Loop 1 active failure Pf =1.33E-2
OR
GC leaking Pf =7.2E-6
Piping rupture Pf =6.5E-6
DHX 1 failure to operate Pf =2.16E-3
DHX 2 failure to operate Pf =5.90E-5
Water Tank unavailability Pf =9.3E-7
Primary isolation failure Pf =2.6E-3 + 1.3E-4 (including CCFs)
Check Valve failure to open Pf =1.0E-4
DHR Blower failure Pf =8.20E-3
OR
OR
Isolation Valve 1 failure to close Pf = 1.3E-3
Isolation Valve 1' failure to close Pf = 1.3E-3
DHR Blower failure to operate Pf = 1.0E-3
DHR Blower failure to run Pf = 7.20E-3

Figure 9. FT of DHR active mode

Since all the component modes of failure are connected through OR operators, the failure probability of DHR loop operating in active mode is the sum of all the failure probability of the components. Finally, the total failure probability of each DHR loop operating in active mode is equal to:

$$P_{f,i}^{A} = 1.33 \cdot 10^{-2}$$

The failure of DHR system results by the failure of all the three DHR redundant loops (operating in active mode), meaning that DHR loops are logically connected by an AND operator (failure of the system when all the three loops fail). Since the loops are composed by the same components, they are characterized by the same probability failure. Therefore, the failure probability of DHR system operating in active mode is given by the following equation:

$$P^A_{f,TOTAL} = P^A_{f,1} \cdot P^A_{f,2} \cdot P^A_{f,3} = \left(P^A_{f,i}\right)^3 = \left(1.33 \cdot 10^{-2}\right)^3 = 2.33 \cdot 10^{-6}$$

In order to assess a more conservative estimation of system failure probability, also the Common Cause of Failures (CCFs), which would contemporary affect all the three loops, have to be taken into account. FTA methodology says that the failure probability of a system composed by n (in this case 3 loops) redundant subsystems, characterized by the same failure probability and subject to CCFs, is given by the following equation:

$$P^A_{f,TOTAL(CCFs)} = P^A_{f,TOTAL} + P^A_{f,CCFs} = \left[\left(1-\beta\right) \cdot \lambda t\right]^n + \beta \cdot \lambda t$$

where:

- n: redundancy of the system (3 in this case)
- β: beta factor, which takes into account for the failure of the system due to CCFs of redundant loops. In this case, a conservative Beta Factor has been considered:

$$\beta = 0.01$$

- λt: failure probability of the single loop of the n-redundant system.

Hence,the failure probability of DHR system operating in active mode, calculated by using the equation above, is:

$$P^A_{f,TOTAL(CCFs)} = 1.33 \cdot 10^{-4}$$

which is the unreliability of DHR system operating in active mode. The total failure frequency for the loss of DHR system is given by the combination of unreliability of the system with the number of demands per year of the system, which are conservatively assumed equal to:

$$f^A_{demand} = 1.0 \cdot 10^{-1} \left[y^{-1}\right]$$

Finally, the failure frequency for total loss of DHR system (operating in active mode) can be expressed as:

$$f^A_{TOTAL} = f^A_{demand} \cdot P^A_{f,TOTAL(CCFs)} = 1.33 \cdot 10^{-5} \left[y^{-1}\right]$$

The loss of active DHR system total frequency results to be larger than the target for Gen IV reactors of 10-7 [y-1].

DHR Operating in Passive Mode

DHR system operates in passive mode in case of pressurized transient (e.g. LOFA: primary circuit integrity preserved) or long term heat removal for depressurized transient (stars 24h after the beginning of the accident, when the residual heat to be removed is lower than the critical limit of 1% of the nominal power).

The case of depressurized long term heat removal, for which the integrity of GC is to be maintained to assure the Back-up Pressure, is conservatively analyzed.

DHR system operating in passive mode does not require the action of DHR blower to guarantee the gas circulation. No active components, except for the Isolation Valves, are required to allow DHR natural circulation operation.

According to what stated above, the DHR System's components whose integrity is to be assured in order to operate in passive mode are listed below.

• isolation valves [1] and [9]:	failure to close;
• check valve [2]:	failure to open;
• DHR piping [3]:	rupture;
• DHX1 [5]:	failure to operate;
• DHX2 [6]:	failure to operate;
• water tank [7]:	unavailability;
• GC [8]:	leaking.

The failure of only one of those components leads to the failure of the entire DHR loop operating in passive mode (logically connected by Boolean operator OR). The failure probabilities of the components, computed for 72 hours of mission time (long term passive heat removal mode) are shown in Table I.

Figure 10 shows the fault tree for possible incidental paths, which lead to the failure of one of the three DHR loops, operating in long term passive mode (Pf is the failure probability for each component and mode of failure).

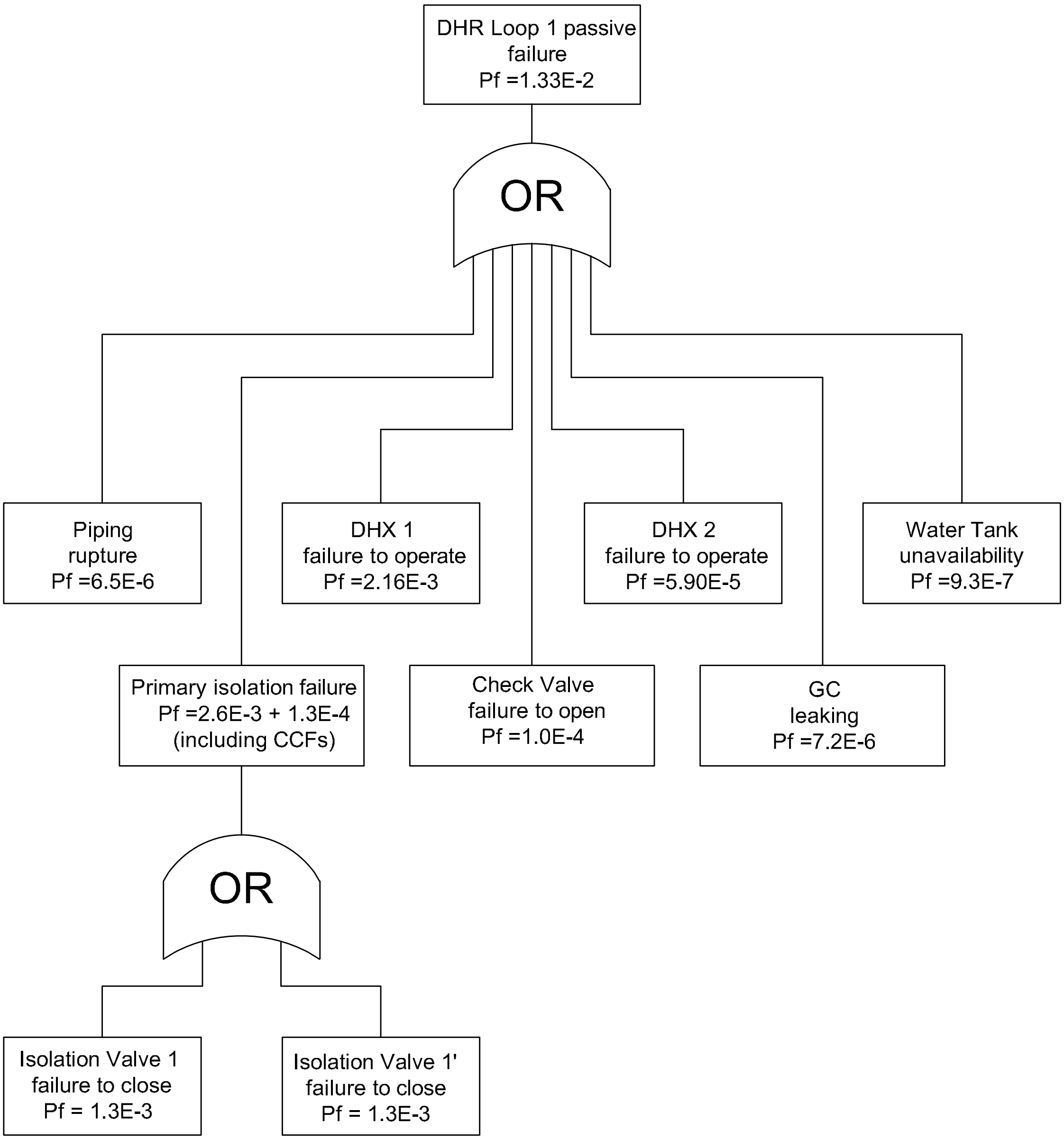


Figure 10. FT of DHR passive mode

The logical connections of the DHR system failure modes, operating in passive mode, are OR Boolean operators, as in active mode operation. The difference between active and passive modes is that the failure of DHR blower is not considered in the fault tree (since they are not required in passive operating mode). The failure probability of DHR i-loop operating in active mode is the sum of all the failure probabilities related to each component, equal to:

$$P_{f,i}^{P} = 5.06 \cdot 10^{-3}$$

System failure results from the failure of all the three redundant loops. Taking into account the CCFs that would affect all the three loops, the total system failure, as for the active mode, and considering a conservative Beta Factor equal to:

$$\beta = 0.0257$$

leads to a failure total probability of:

$$P^P_{f,TOTAL(CCFs)} = 1.30 \cdot 10^{-4}$$

which is the unreliability of DHR system operating in passive mode. The total failure frequency for the loss of DHR system is given by the combination of unreliability of the system with the number of demands per year of the system, which are conservatively assumed equal to:

$$f^P_{demand} = 1.0 \cdot 10^{-1} \left[y^{-1} \right]$$

Finally we are able to state he failure frequency for total loss of DHR System, if operating in passive mode.

$$f^P_{TOTAL} = f^P_{demand} \cdot P^P_{f,TOTAL(CCFs)} = 1.30 \cdot 10^{-5} \left[y^{-1} \right]$$

The loss of active DHR system total frequency results to be larger than the target for Gen IV reactors of 10-7 [y-1].

Beta Factor

The Beta Factor method states that the failure rate of a loop (and hence its failure probability) belonging to an n-redundant system is composed by two components:

- λi: failure rate related to the independent failure occurred to the single loop;
- λCCFs: failure rate related to the CCFs occurred to the all n redundant loops.

$$\lambda = \lambda_i + \lambda_{CCFs}$$

Beta Factor is defined as:

$$\beta = \frac{\lambda_{CCFs}}{\lambda}$$

Its value ranges between 0 (if the redundant loops are not affected by CCFs) and 0.25. The total failure rate of the loop can be written as follow:

$$\lambda = (1-\beta)\lambda + \beta\lambda$$

Hence, the failure probability of the loop, considering a mission time t, is:

$$P_{f,loop} = \lambda t = (1-\beta)\lambda t + \beta\lambda t$$

For a system composed by n-redundant loops, characterized by the same failure rate and logically connected by the AND Boolean operator, the failure probability result to be:

$$P_{f,CCFs} = \left[P_{f,loop} \right]^n \cong \left[(1-\beta)\lambda t \right]^n + \beta\lambda t$$

which is the equation used to calculate the failure probability of DHR active/passive system composed by n=3 redundant loops with equal failure probabilities. The system fault tree is:

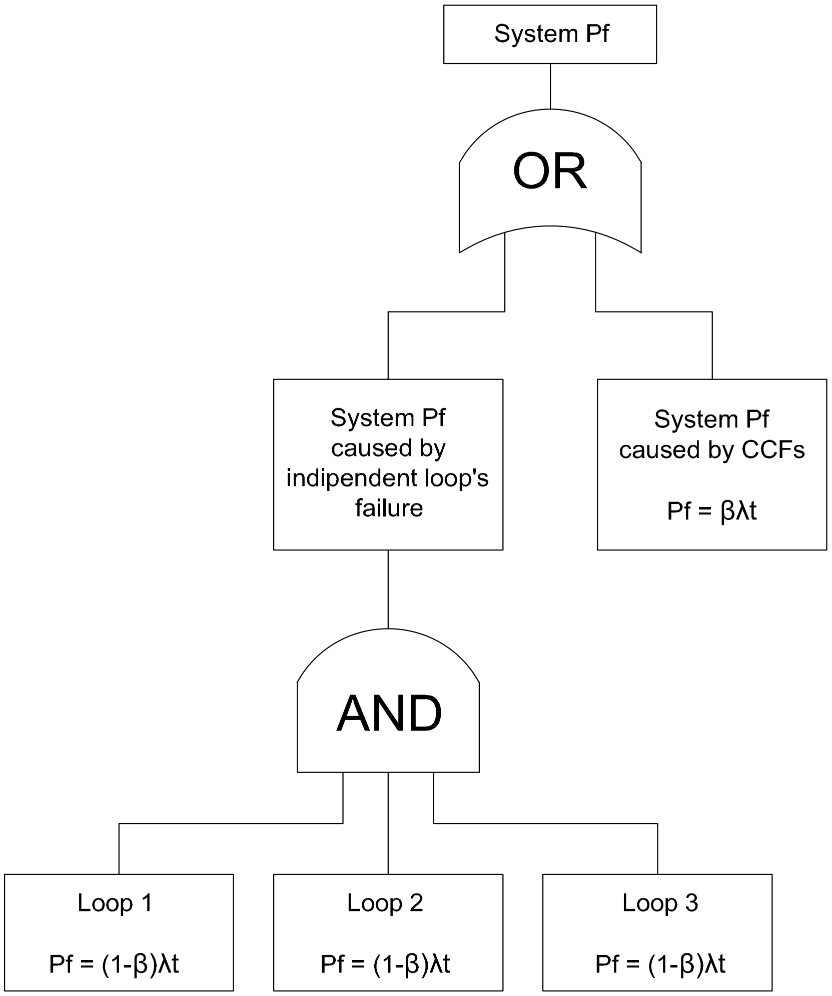


Figure 11. FT of DHR

System operating in Combined (active-passive) Mode

In case of LOCA, the pressure reduction (10 bar when integrity of GC is assured) would not allow the decay heat to be removed by natural circulation. Forced circulation must be assured at least for the first 24 hours after the beginning of the incidental sequence. In case LOCA is combined with SBO accident, the forced circulation during the first 24 hours is assured by the plant bank of batteries, which assures sufficient energy supply to the DHR blowers. Hence, the DHR strategy in case of LOCA+SBO events relies on combined active-passive operational mode of DHR system, as follow:

- Active Mode (during the first 24 hours):energy supply to the blowers is assured by the plant bank of batteries.
- Passive Mode (after the first 24 hours): DHR operates in passive mode for 72 hours.

Failure of DHR System operating in combined mode (active/passive) results from the failure of DHR System operating in active mode for the first 24 hours and the failure of DHR System operating in passive mode for the following 72 hours.

Hence, the failure probability of DHR system operating in combined mode is given by AND logical connections (sum) between the failure probability of the DHR system in active mode (mission time of 24 hours) and the failure probability of the DHR system in passive mode (mission time of 72 hours).

- Failure probability of DHR System operating in active mode.

The failure probability (equals to the failure rate times the mission time λT) has been calculated using the same failure rates of Table II, with a mission time of 24 hours (instead of 72 previously used); CCFs of the 3 loops have been considered by using the Beta Factor Method, with a conservative beta factor equal to: β = 0.01. The resulting failure probabilities of each component used in the computation are reported in Table VI.

Table V I. Failure modes, rates and probabilities of DHR active/passive mode

Component	**Failure Mode**	**Failure Rate**	**Mission Time/Demand**	**Failure Probability**
1-Isolation Valve (hot leg)	failure to close	1,3*E-3 [1/demand]	1 [demand]	1.3E-3
1'-Isolation Valve (cold leg)	failure to close	1,3*E-3 [1/demand]	1 [demand]	1.3E-3
2-Check Valve	failure to open	1,0*E-4 [1/demand]	1 [demand]	1.0E-4
3-Piping	rupture	3,0 *E-9[1/(h*metro)]	24 [h]	2.17E-6
4-DHR Blower	failure to start	1,0E-3 [1/demnd]	1 [demand]	1.0E-3
4'-DHR Blower	failure to operate	1,0E-4 [1/h]	24 [h]	2.4E-3
5-DHX 1	failure to operate	3,0E-5 [1/h]	24 [h]	7.2E-4
6-DHX 2	failure to operate	8,3E-7 [1/h]	24 [h]	1.97E-5
7-Water Tank	unavailability	9,3E-7 [1/demand]	1 [demand]	9.3E-7
8-BC	leaking	1,0E-7 [1/h]	24 [h]	2.4E-6
9-Isolation Valve CCF	CCF	1,3*E-3 [1/demand]*0.1[beta]	1 [demand]	1.3E-4

The Table shows the failure rate that, multiplied by the mission time (time interval for which must sessere guaranteed the operation of the system DHR) provide the probability of failure of the same-component.

As previously said, the DHR failure probability is:

$$P^{A}_{f,TOTAL(CCFs)} = 7.01 \cdot 10^{-5}$$

- Failure Probability of DHR System operating in passive mode.

$$P^{P}_{f,TOTAL(CCFs)} = 1.30 \cdot 10^{-4}$$

Since the two operating modes are logically connected by AND Boolean operator, the total system failure probability, operating in active/passive combined mode is given by the sum of the previous failure probabilities:

$$P^{A/P}_{f,TOTAL(CCFs)} = P^{A}_{f,TOTAL(CCFs)} + P^{P}_{f,TOTAL(CCFs)} = 7.01 \cdot 10^{-5} + 1.30 \cdot 10^{-4} = 2.00 \cdot 10^{-4}$$

Considering a system demand frequency equal to:

$$f^{A/P}_{demand} = 1.0 \cdot 10^{-1} \left[y^{-1} \right]$$

The total failure frequency of DHR System operating in combined mode results to be:

$$f^{A/P}_{TOTAL} = f^{A/P}_{demand} \cdot P^{A/P}_{f,TOTAL(CCFs)} = 2.00 \cdot 10^{-5} \left[y^{-1} \right]$$

Failure Frequency of DHR System

Referring to DHR strategy (Figure 7), different operating modes of DHR system for different reactor accidental sequences can be identified:

- LOCA and LOFA with energy supply available: Active Mode.
- LOFA + SBO (pressurized condition with no energy supply available): Passive Mode.
- LOCA + SBO (depressurized condition with no energy supply available): Combined Mode.

Table VII gives the total failure frequency of DHR.

Table VII. DHR total failure frequency

AccidentalSequence	Mode of Operation	Failure Frequency[1/y]
LOCA	Active	1.33E-5
LOFA	Active	1.33E-5
LOFA+SBO	Passive	1.30E-5
LOCA+SBO	Combined	2.00E-5

5. Conclusions and recommendations

Safety of innovative reactors, like Gen IV reactors, is expected to be enhanced through the implementation of passive safety features within their designs. A motivation for the use of passive systems to accomplish safety functions, as reactor scram and decay heat removal, is their potential for enhanced safety through increased safety system reliability, because of the claimed advantages of simplicity, reduction of the need for human interaction, reduction or avoidance of external electrical power. However recent studies raise concerns and caution against the claimed superior performance of passive systems and relative higher availability and reliability. Hence, as a response to this concern, a comparative assessment of active and passive systems has been performed in terms, principally, of the expected performance and reliability figures of merit. To this aim the system-based analysis is complemented with the sequence-based analysis, since the strong interaction between the system performance and the accident scenario. For safety and reliability improvement several ideas are included such as the core cooling by natural circulation in case of Station Black Out and the use of passive reactor shutdown systems.

The analysis points out the relevance of the reliability figure of merit as the most important factor in the process of opting out of one system in favor of the other alternative: in fact the relative assessment is recognized as being still an open issue, despite in the recent years an important effort has been made by suppliers, industries, utilities and research organizations on passive safety systems both for their development and assessment. Inclusion of potential failure modes and reliability estimates of passive components for all systems is recommended in probabilistic safety assessment studies.

In particular, as regards natural circulation systems, results show that the probability of failure of the passive safety function is not to be neglected. However with the models presented here, the simplifying assumptions and the limited scenarios considered, it is not reasonable to confidentially conclude that the functional reliability for these systems is such that it constitutes a challenge for the accomplishment of the safety function. But one can conclude that attention has to be paid to the functional aspects of the passive system, (i.e. the ones not

pertaining to the "hardware" of the system), that can challenge their "credited" higher reliability with respect to active ones.
SASS with the potential to improve RPS performance by adding an important safety feature to the defence-in-depth case while increasing scram reliability and contributing significantly to the reliability of the overall plant system have been analyzed. However also in this case the lack of experimental evidence and their premature stage of development doesn't allow to verify and validate the required reliability target, that makes them attractive for their inclusion in the design of innovative reactors.
As shown in Table VII, the DHR failure frequencies calculated (through conservative assumptions) for the different accidental sequences (LOCA, LOFA, LOCA + SBO, LOFA + SBO) fail to meet the failure frequency target stated for Gen IV Reactors:

$f \leq 10^{-7}$ [y^{-1}]

The high values of failure frequencies calculated in this report are mainly related to the CCFs which would simultaneously affect the 3 redundant loops of DHR system. Improvement on redundancy, separation and diversification of DHR system components and loops would lead to an increase of the safety level of the entire system.
It is important to point out that this is a preliminary study and further investigations, especially on the influence of CCFs, should be performed as soon as a more detailed design is available. However, general considerations about failure frequencies can be identified:

1. Passive operational mode assures a lower failure frequency than active operational mode since natural circulation does not need the operation of DHR blowers. However, passive mode failure frequency is still significant and further investigations on component failure rates have to be performed.
2. Further investigation on batteries bank (that has not been considered here) is needed for the system combined operational mode.
3. SBO frequency in case of DHR system passive operational mode has to be considered in further analysis.

The simplified models and the limited accidental scenarios considered, do not allow to state conclusive assumptions on the safety performance of the DHR system operating in active, passive and combined modes. However, the relevant values of failure frequencies, even for passive operational mode, suggest that further safety analysis have to be performed. Particular attention has to be addressed on functional aspects and configurations of DHR system operating in passive mode.
Passive system reliability is not better or worse than the active ones: reliability will depend on the overall design and operation of the system, regardless of whether the system is active or passive. A good overall plant design may include active systems, passive systems or combination of both types of systems to meet performance and safety objectives.
Although these systems are credited a higher reliability with respect to the "conventional" ones - because of the smaller unavailability due to hardware failure - or even they are claimed to be immune from faults, they pose however some challenges as regards the availability/ reliability issues and more in general their performance assessment, because there is always a nonzero likelihood of the occurrence of physical phenomena leading to pertinent failure modes.

References

1. L. Burgazzi, F. Giannetti et al.,Valutazione della risposta di sistemi attivi e passivi a fronte di sequenze incidentali rilevanti ai fini della sicurezza, ENEA report NNFISS-LP2-066, August 2012
2. L. Burgazzi, Addressing the Challenges posed by Advanced Reactor Passive Safety System Performance Assessment, Nuclear Engineering and Design 241, 1834-1841, 2011
3. R. Stainsby et al., "Gas-Cooled Fast Reactor research and Development in the European Union", Science and Technology of Nuclear Installations, Volume 2009, article ID 238624, 7 pages
4. C. Poette et al., "Allegro: The European Gas Fast Reactor Demonstrator Project" 17th International Conference on Nuclear Engineering (ICONE 17), July 12-16, 2009, Brussels, paper ICONE17-75326, pp. 815-822
5. C. Bassi, M. Marques, "Reliability Assessment of 2400 MWth Gas-Cooled Fast Reactor Natural Circulation Decay Heat Removal in Pressurized Situations", Science and Technology of Nuclear Installations, Volume 2008, article ID 287376, 16 pages
6. L. Cheng, T. Y.C. Wei, "Decay Heat Removal in Gen IV Gas-Cooled Fast Reactors ", Science and Technology of Nuclear Installations, Volume 2009, article ID 797461, 13 pages
7. IAEA TECDOC-1083."Status of Liquid Metal Cooled Fast Reactor Technology", International Atomic Energy Agency (Sep. 1999)
8. P. Saignes, " Reliability Database for PSA in support to the design of the innovative CEA 2400 MWth gas fats reactor". Proceedings of PSAM 9, Ninth International Probabilistic Safety Assessment and Management Conference, 18-23 May 2008, Hong Kong, China
9. C. Bassi et al., "Level 1 probabilistic safety assessment to support the design of the CEA 2400 MWth gas-cooled fast reactor", Nuclear Engineering and Design 240, 2010, pp. 3758-3780
10. L. Burgazzi, "Reliability evaluation of passive systems through functional reliability assessment", Nuclear Technology 144, 2003, pp. 145-151
11. M. Marques, L. Burgazzi, et al., "Methodology for the reliability evaluation of a passive system and its integration into a probabilistic safety assessment", Nuclear Engineering and Design 235, 2005, pp. 2612–2631
12. L. Burgazzi, 2002. Passive System Reliability Analysis: a Study on the Isolation Condenser, Nuclear Technology 139, 2002, pp. 3-9
13. A.E. Dubberley, "Super Prism Metal core Margins to Severe Core Damage", ICONE 8, Baltimore MD, paper 8001 (2000)
14. A.E. Dubberley, "Transient Performance of SPRISM", ICONE 9, Nice, France (2001)
15. IAEA-TECDOC-626 " Safety Related Terms for Advanced Nuclear Plants", IAEA, Vienna, September 1991
16. J. Dewitte et al. "Self-actuated shutdown system performance in sodium fast reactors", Transactions of 2010 American Nuclear Society Winter Meeting, Las Vegas , (NV), Vol. 103, pp. 609-610
17. L. Burgazzi, "Analysis of solutions for passively activated safety devices and/or reactivity feedbacks", ENEA Report UTFISSM-P9P4-004, December 2011

18. Bertrand, F., et al., Preliminary transient analysis and approach of hypothetical scenarios for prevention and understanding of severe accidents of the 2400 MWth Gas-cooled Fast Reactor. Proc. of NURETH-13. 2009. Kanazawa City, Japan.