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PASSIVE SHUTDOWN SYSTEMS FOR FAST NEUTRON REACTORS

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FOREWORD

One of the IAEA's statutory objectives is to "seek to accelerate and enlarge the contribution of atomic energy to peace, health and prosperity throughout the world." One way this objective is achieved is through the publication of a range of technical series. Two of these are the IAEA Nuclear Energy Series and the IAEA Safety Standards Series.

According to Article III.A.6 of the IAEA Statute, the safety standards establish "standards of safety for protection of health and minimization of danger to life and property". The safety standards include the Safety Fundamentals, Safety Requirements and Safety Guides. These standards are written primarily in a regulatory style, and are binding on the IAEA for its own programmes. The principal users are the regulatory bodies in Member States and other national authorities.

The IAEA Nuclear Energy Series comprises reports designed to encourage and assist R&D on, and application of, nuclear energy for peaceful uses. This includes practical examples to be used by owners and operators of utilities in Member States, implementing organizations, academia, and government officials, among others. This information is presented in guides, reports on technology status and advances, and best practices for peaceful uses of nuclear energy based on inputs from international experts. The IAEA Nuclear Energy Series complements the IAEA Safety Standards Series.

The designs of nuclear power plants in general and innovative nuclear systems in particular increasingly include passive features. The availability of these inherent and passive safety features becomes important when active systems such as emergency shutdown systems for reactor shutdown are not functioning properly. Some experience has been gathered in this technical area, and the innovative fast reactor concepts under development worldwide are presumed to include substantial passive safety features.

This publication details the findings of a study of passive shutdown systems for fast neutron reactors, and aims to provide comprehensive information about such systems. It discusses experience in developing these systems, along with the research that is being undertaken. It provides information on the basic design principles for passive shutdown systems and the related operational experience gathered so far, and it reviews the innovative concepts under development as well as the areas for further R&D and qualification tests.

The IAEA is assisting Member States in the area of advanced fast reactor technology development by providing a means for information exchange and collaborative R&D to pool resources and expertise. The IAEA's fast reactor technology development activities are pursued within the framework of the Technical Working Group on Fast Reactors (TWG-FR). This study was conducted on the recommendation of the TWG-FR. The IAEA acknowledges the support provided by the TWG-FR and expresses its appreciation to all participants in the study for their dedicated efforts leading to this publication.

The IAEA also expresses its appreciation for the generous contributions of all the contributors to drafting and review in the preparation and review of this study. Their participation, and the generosity of their organizations in providing permission to use information, including figures and tables, in this publication are acknowledged. Information and data included in the publication were delivered during the IAEA Technical Meetings on Passive Shutdown Systems for Liquid Metal Cooled Fast Reactors in October 2015 and in February 2017.

The IAEA would like to express special thanks to S. Qvist (Sweden) and E. Bubelis (Germany) for coordinating this study. The IAEA officers responsible for this report were V. Kriventsev, S. Monti and C. Batra of the Division of Nuclear Power.

CONTENTS

1. INTRODUCTION

1.1. BACKGROUND

A major focus of the design of modern fast reactor systems is on inherent and passive safety. Inherent and passive safety features are especially important when active systems such as emergency shutdown systems for reactor shutdown are not functioning properly. Passive shutdown systems (PSSs) can operate either continuously or function as a backup actuation method for the conventional reactor scram system. The innovative fast reactor concepts are presumed to include substantial passive safety features. In nearly all the fast reactor programmes, specific systems to improve reactor safety performance during accidental transients have been developed, and a large number of proposed systems have reached various stages of maturity. In order to promote the exchange and collection of information on projects and programmes related to PSSs for fast reactors at the national and international level, as well as to present and review the advanced engineered safety shutdown system concepts, a Technical Meeting on Passive Shutdown Systems for Liquid Metal Cooled Fast Reactors was held in Vienna during October 2015.

The meeting observed that designers of the PSSs still have to face several challenges (e.g. speed, reliability and predictability of actuation during accident scenarios, testability during operation, lifetime and performance degradation issues, impact on core operation and neutron economy, costs). Therefore, the fast reactor community called for a comprehensive publication based on a combined study, to provide detailed information on the basic design principles for PSSs and the related operational experience gathered so far, and also to provide a review of the innovative concepts under development along with their R&D needs and qualification tests. This IAEA Nuclear Energy Series publication is a response to that call.

1.2. OBJECTIVE

The overall objective of this publication is to assist existing and potential interested stakeholders by presenting a consistent state of the art review of fast reactor PSSs. The specific objective of this publication is to present both the historic and current developments in PSS design, evaluation and modelling in the various development programmes across IAEA Member States.

1.3. SCOPE

This publication provides a comprehensive set of existing options for those interested in applying PSSs to fast reactors. This publication presents the views of contributors regarding the requirements for the functioning of such systems, and is expected to interest organizations involved in both the development and operation of fast reactors. Guidance provided here, describing good practices, represents expert opinion but does not constitute recommendations made on the basis of a consensus of Member States.

1.4. STRUCTURE

Sections 1 and 2 of this publication present a historical perspective, definitions and functional requirements of PSSs in general. Section 3 briefly presents the design and function of a wide array of PSSs, both historical systems not currently in development as well as new system designs. Section 4 presents an overview of the operational experience with fast reactor shutdown systems to date. In Section 5, the report presents in more detail the PSSs that are under active development (in 2017/2018). Section 6 presents results from transient analysis of fast reactors equipped with PSSs, and where available, comparisons of analysis with experimental results. In Section 7, the general needs for research, development and qualification of novel systems is discussed.

2. FUNCTIONAL REQUIREMENTS

2.1. OVERVIEW

Unprotected transients, also called anticipated transients without scram (ATWSs) are a group of design extension conditions (DECs) that can significantly challenge sodium cooled fast reactor (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 low probability events combined with a reactor protection system (RPS) failure (no reactor scram) would lead to coolant boiling and a core melting scenario. These events include: unprotected transient overpower (UTOP), unprotected loss of flow (ULOF) in the primary coolant circuit, unprotected loss of heat sink (ULOHS), unprotected loss of flow and heat sink, and unprotected station blackout event.

In general, passive devices may, in combination with other measures, contribute to reducing 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.

Achieving safety in design extension events, in case of unavailability of the active safety systems, implies that the reactor can be shut down and the decay heat can be removed from the reactor automatically by means of natural processes (gravity, the coolant flow, the thermal principles, etc.). In this way, the reactor is brought into a safe state and temperatures are kept well below the boiling point of the coolant. The focus of this report is on the systems acting to passively shut the reactor down; the report does not consider the systems for decay heat removal. The main design principle of a PSS is: under a DEC, even if the RPS fails, the reactor can be shut down only by either inherent passive reactivity feedback mechanisms or an engineered, independent PSS. Therefore, it is a functional requirement for the engineered system to be able to handle all unprotected transient events, as listed above.

Based on reactor R&D activities within the United States of America (USA), general functional requirements for the development of SFR PSS designs were specified as follows:

(a) The system should have sufficient worth, and should be sufficiently fast acting such that an appropriate amount of negative reactivity is introduced in time to prevent damage to the core in postulated faulted conditions.
(b) The system should be fault tolerant to deformations caused by DEC energetic events.
(c) The system should be able to function under the loads imposed by design basis seismic conditions including a margin.
(d) The system's location relative to the core should be possible to identify based on reactor instrumentation.
(e) The system should be built of materials that can withstand fast neutron flux and the coolant temperature environment for its design lifetime.

Some other characteristics are desirable but not mandatory:

(a) The system should be testable in situ.
(b) The system should be resettable in situ.
(c) The system's influence on any mode of normal reactor operation (i.e. startup, shutdown, full power and partial power with partial flow) should be as low as reasonably achievable.

2.2. GENERIC REQUIREMENTS

The principal objective is to provide an independent shutdown system which will efficiently operate passively/inherently to improve the plant's response to the transient. Thus, major functional requirements of the system are the following:

— Requires no operation of any intelligent sensing instrumentation or the control system;
— Fails in the fail-safe condition.

These systems are required to perform their safety function for internal and external hazards (DECs), such as a severe earthquake. Incorporation of such a shutdown system will substantially reduce the probability of failure to shut down in a timely manner.

2.3. SAFETY REQUIREMENTS

Specific safety requirements include the following:

(a) Limit failure modes that introduce positive reactivity to acceptable levels, limiting any potential power increase;
(b) Avoid common mode failure as much as reasonably achievable;
(c) Assure design can be licensed within current licensing frameworks.

The reliability of the whole shutdown function should be improved by the introduction of these engineered, passive systems by providing for a redundant, independent shutdown system which avoids any common mode failures of the design basis RPS.

In addition, the self-actuated shutdown system (SASS) must meet high availability and reliability requisites: the Generation IV goal for scram failure probability is set to be less than 10^{-8} [1]. Since the value of the RPS failure probability is about 10^{-6} [2], a value of at least 10^{-2} should be allocated for the third shutdown system failure probability.

2.4. FUNCTIONAL REQUIREMENTS FOR PASSIVE SHUTDOWN SYSTEMS FOR BN-TYPE REACTORS

The main requirements set out in the preliminary safety analysis report for BN-800 reactors were formulated at the Institute of Physics and Power Engineering in the Russian Federation:

(a) According to the purpose, the efficiency and response of a PSS should be sufficient to warn of sodium boiling and heavy damage in the reactor core and also for the damping of reactor power while maintaining mean sodium temperature in the reactor core at a secure level.
(b) The threshold of a PSS should be selected so that it does not operate for transient events and design accidents.
(c) The devices of PSSs and their modes of operation should eliminate the capability of them introducing positive reactivity at any event connected with a change of sodium flow rate or sodium temperature in the reactor.
(d) The PSS is best initiated by a rise in coolant temperature, as this is the most universal sign of an accident connected with a loss of coolant flow rate and with increase of reactor power.
(e) The efficiency of each PSS should guarantee the insertion of sufficient negative reactivity to maintain the reactor in a subcritical state.
(f) The design of a PSS should not change a nominal system of refuelling.
(g) The PSS design and the physical principles of its activation and performance should, whenever possible, replace the nominal system of accident protection in case of its general failure.
(h) The functionality of a PSS can be tested by modelling accident processes in the reactor. When it is not possible to securely model accident processes directly in the reactor, the PSS can be tested through indirect signs or in a special test section.

PSSs should function in BN reactors under difficult conditions of exploitation (e.g. temperature ~600°C, fluence of fast neutrons with $E > 0.1$ MeV $\sim 2 \times 10^{21}$ (neutron/cm^2)). In other words, PSSs should not be sensitive enough to cause shutdown under normal operational factors.

The following types of PSS worked:

— Hydraulically suspended rods;
— Magnetic PSS, activated on the basis of Curie point;

— PSS activated on the basis of a hyperthermal shape memory effect;
— PSS activated on the basis of a series of non-traditional physical effects including lyophobic capillary porous systems.

Based on physical effects in a temperature sensitive element where the PSS is activated by an excess of coolant temperature, it is possible to divide the functioning of working elements into the following groups:

— Linear temperature expansion of a rigid body, including bimetals;
— Volumetric expansion of a liquid;
— Magnetic properties, including Curie point effect;
— Shape memory effect;
— Melting of the working body in a lyophobic capillary porous system.

2.5. SPECIAL CONSIDERATION FOR OTHER SYSTEMS

2.5.1. Heavy metal fast reactors

Heavy liquid metal cooled fast reactors (HLMFRs) are cooled by lead or lead–bismuth eutectic (LBE). The response of HLMFR designs to DECs is somewhat different from that of SFRs.

HLMFRs have some intrinsic advantages such as inherent safety characteristic regarding CDA in the case of the unprotected events listed at the start of Section 2.1. This is due to the neutronic properties of lead and the larger amount of coolant in the core, which gives them a negative coolant temperature reactivity coefficient, thereby avoiding core damage in most unprotected cases. Furthermore, the temperature limit is no longer the coolant boiling temperature but, in the short term, the cladding thermal resistance, and in the longer term, the chemical resistance of the cladding to LBE attack. The aim of an engineered shutdown system is therefore to bring the reactor to a cold shutdown state where corrective action can be taken, such as defueling.

HLMFRs have their own issues. With lead and LBE chemistry, a coolant fault may drive an in-core local flow blockage or overall flow blockage, such as with the accident on the K27 submarine (also known as Project 645). During an undetected (and thus unprotected) event, control[1] may still be exercised, even if it is manual. If the formation of a blockage is slow, the lack of a sudden change can hinder detection until after the blockage becomes problematic.

These specific characteristics lead to different requirements for the engineered shutdown systems. The emphasis on ‘passive’ is less important; nevertheless, the system has to be passive in execution, but could be active in initiation, with more emphasis on system monitoring. The minimum expectation for ‘monitoring’ is knowledge of the system status: withdrawn or inserted. When initiation is active, robustness is provided by two independent and redundant systems, triggered by different sets of sensors measuring different physical parameters. This is feasible for system-wide events but is difficult for local events, such as local flow blockages, as a large number of sensors are already required owing to the necessary distribution of sensors over the core.

As HLMFRs already have negative natural feedback (an inherent safety characteristic), designing engineers tend to prefer systems that toggle between fully withdrawn and fully inserted (bringing the system to a cold shutdown) in the case of incident or accident rather than a having a progressive reversible system (which only reaches ‘hotter than normal operation’ shutdown).

2.5.2. Accelerator driven systems

In accelerator driven systems (ADSs), a subcritical ($k_{\mathrm{eff}} \approx 0.95$) core is fed by a neutron source. This source[2] is driven by an accelerator, and neutron output is proportional to the beam current; thus, reactor power is driven

[1] All the more so for an engineered passive regulation system.
[2] Spallation in high power ADSs or fusion in low power ADSs.

by the beam current. A scram is performed by shutting down the accelerator. The scenario wherein the accelerator is unable to be tripped[3] can be excluded. Unprotected events could only reasonably occur through non-detection.

The inherent behaviour of an ADS is completely different from that of a critical reactor. In an ADS, reactivity perturbations only weakly affect the overall power. Even reactivity changes that would rapidly shut down a critical machine would only decrease the overall power of an ADS by a few per cent, and would not noticeably decrease the power in the target area. The major consequence is that in-core actions are inefficient (reactivity feedback is useless), and only actions taken on the accelerator or beam are effective.

Reactivity effects are concealed by current variation[4], making indirect detection impossible. System passivity can only be reached by 'non-intelligent' sensors. These sensors have direct effect on the signal to the accelerator (e.g. a thermal switch). Note that reactivity measurement through beam interruption and current to power ratio via a computerized system are under development.

During an unprotected event, power is basically unchanged, with the exception of events involving beam current. Note that severe target disruptions generally result in an overall power reduction, due to target motion away from its optimum position.

2.5.3. Gas cooled fast reactors

In recent years there has been an increased interest in gas cooled fast reactor (GFR) systems. The helium gas cooled fast reactor is one of six advanced reactor concepts selected by the Generation IV International Forum and offers a number of advantages over liquid metal cooled core designs. There are significant safety, economic and technical benefits to be gained from using an inert readily available gaseous coolant which is compatible with both air and water, compared to sodium, which reacts rapidly with water and requires special treatments. The low neutron absorption cross-section of a helium nucleus results in one of the hardest neutron spectra among fast reactors, making it ideal for minor actinide transmutation. The lower coolant void coefficient in helium cooled cores, compared to sodium cooled systems, allows for the loading of a far greater quantity of degraded plutonium and minor actinide fuels. The core outlet temperature, not limited by the coolant characteristics, leads to attractiveness for hydrogen production and other high-potential heat applications.

Therefore, GFRs offer considerable flexibility in core design. The safety challenges associated with current GFR Generation IV designs are mainly related to high power density, low heat transfer coefficient of the materials used in the core, low coolant thermal inertia and the low natural convection mass flow rates available for decay heat removal. These basic facts lead to the possibility of rapid temperature excursions during an ATWS event. The results show that current fuel cycle economic factors for a uranium startup core would set a minimum power density range on the order of 70–100 W/cc. Regardless of fuel form (plate or pin type), with the limited conductivity of core fuel and structural materials, just a small amount of decay heat (approximately 2%) would lead to core meltdown.

For the Generation IV GFR it would be appropriate to ensure that no core meltdown would occur on account of the failure of reactivity scram systems, failure of active and passive decay heat removal systems, total loss of electric power and depressurization events. Taking into account GFR specifics, as described in this section, passive safety mechanisms would need to be utilized. According to calculations, the actuation time of passive systems is in some cases relatively short. In case of ULOF, the rapid increase of core temperature leads to melting of the fuel pin cladding within 100 s.

The safety and control rods in available GFR designs are currently based on a common design concept using hexagonal rods controlled by rod followers passing through the lower part of the core. Therefore, the inherent/passive system should address common mode failures that prevent rod insertion, such as the following:

— Gross core distortion (beyond basic seismic events) or presence of debris from failed primary circuit components preventing motion of rods within the core;
— Trapped absorber elements above the core (high temperature environment);
— Distortion of the lower part of the reactor where rod mechanisms are placed.

[3] Present day accelerators have rather poor reliability, but the problem is keeping them running, not stopping them.
[4] About ±2% with present linear accelerator technologies.

The new passive system needs to provide a method for reactivity control without jeopardizing the safety and function of the reactor core. A variety of commonly used methods of reactivity control can be used in GFR design, including:

— Adding absorber material into the core;
— Removing fuel or reshaping the core;
— Increasing neutron leakage from the core, causing a decrease of core reactivity.

The previously mentioned approach has been considered within a range of reactor designs, including molten salt and fast reactors for outer space applications. In the GFR, owing to the smaller coolant transport cross-section, the migration area of neutrons is exceptionally large, and the possibility of controlling reactivity by the movable reflector appears to be a promising additional safety solution. To increase the effectiveness of this system in emergencies, the neutron leakage can be supported by the additional absorber material in the movable reflector region. Owing to the fast neutron spectra, a combination of materials with moderator and absorption properties can increase the absorption rate and possibly decrease the overall cost of additional reactivity control systems.

The additional movable reflector system may be actuated by using Curie point latches or by a system which enhances reflector rod drive line thermal elongation. Both these systems are reliable, robust and reusable. A movable reflector would have only a minor impact on neutron economy during normal operation but may suffer from actuation, time lag or a potential inadvertent operation, unless optimally calibrated. These and more questions need further investigation.

2.6. HISTORICAL PERSPECTIVE OF PASSIVE SHUTDOWN SYSTEMS

In the 1970s, a review of the available commercial light water reactor (LWR) experience displayed that the failure of a typical RPS occurs with a probability of 10^{-5} per demand [3], leading to an expected frequency of ATWS of 2×10^{-4} per reactor year [4]. Generally, this is considered an unacceptably high probability considering the possible consequences. In some countries, additional regulations have been adopted, such as requiring an additional independent reactor trip system, to reduce the frequency of an ATWS. As with LWRs, designers made an attempt to improve the scram function reliability for prototypes or commercial size SFRs designed in the late 1970s, like BN-600, Monju or Superphénix. These improvements were achieved through enhancements of redundancy and diversity of the shutdown systems.

However, given the state of the art of safety features in LMFR, ATWS sequences remain the major contributor to the risk of core meltdown or CDA. Indeed, it is to be noted that, unlike in the pressurized water reactor concept, the shutdown function of existing LMFRs relies only on insertion of absorber rods in the core (no bore injection system). Hence, an approach to achieve passive shutdown has been to utilize fuel or existing core structural materials to add negative reactivity (Experimental Breeder Reactor II is an example). In case of ATWS, such an approach may avoid the potential reactivity overshoot caused by sodium boiling and then reduces the risk of CDA. Nevertheless, the inherent feedback effects that would tend to stabilize the core power in case of ATWS are of a thermal nature and are difficult to increase. In addition, these thermal reactivity effects are effective as long as temperatures remain significantly high in the primary circuit [5]. For most of the LMFR designs, these temperatures would lead to rapid damage of the primary sodium confinement. As a consequence, even if inherent core characteristics might be used to stabilize the core power in case of ATWS, LMFRs have to be shut down. Thus, it has been considered that the main way to prevent the risk of core meltdown in LMFRs was to enhance the reliability of the engineered shutdown function as far as reasonably possible, for all potential situations.

In this context, the enhancement of shutdown system reliability brought the shutdown function of SFR concepts developed in the 1990s (like the Power Reactor Innovative Small Module (PRISM), Japan Sodium Cooled Fast Reactor (JSFR) or Prototype Fast Breeder Reactor (PFBR)) to very low failure probability [1, 2, 6] (around 10^{-7} to 10^{-6}). Nevertheless, owing to the lack of sufficient statistical basis, important uncertainties impaired the credibility of the failure probability calculation for the insertion of the control rods. In particular, a common cause failure that could lead to a ‘rod stuck’ event is important in the reliability evaluation as compared with the failure probability of the command control electronics [1].

The above concerns prompted designers to envisage a ‘third level’ of reactor shutdown based on passive features to achieve an efficient diversification of the shutdown function. Such devices have been studied for all LMFR projects developed since the 1990s, like PRISM (1990 design evolution), the European fast reactor (EFR) or, more recently, BN-800 [7].

Finally, following the accident at the Fukushima Daiichi nuclear power plant, the fast reactor design community emphasized the need for a PSS for fast reactor designs to prevent a DEC (ATWS) leading to a core meltdown accident [8].

2.7. DEFINITION OF PASSIVE SHUTDOWN SYSTEMS

Each type of PSS is associated with advantages and disadvantages. It is difficult to determine unambiguously which of them will be most suitable in the specific conditions of nuclear power plants. Owing to this fact, a large number of research activities and investigations are under way for different kinds of PSS.

The analysis of scientific, technical and patent sources has shown that more than 200 different devices that can act as PSSs for fast reactors have been already proposed.

According to the feedback, the basic PSS devices can be divided on the following basis:

— Operating at maximum allowed fuel temperature excess;
— Operating at low coolant flow rate;
— Operating at high coolant temperature.

The operation of the first type of device is based on a phase transition, for example, melting [9] and sublimation, and nuclear fuel moving. These types of device will provide the most effective safety functions, but at present they are developed poorly from the technological viewpoint, and it is therefore necessary to develop one very carefully before any implementation. That will require time and money.

There has been significant progress in development of devices operating at low coolant flow rate. They can be further classified based on the following:

— Increased neutron leakage (Japan, gas expansion module (GEM)) [9];
— Principle of hydrodynamic containment of the absorber (Russian Federation) [8].

The disadvantage of these types of device is the low sensitivity to changes of the reactor temperature (or power).

PSSs that operate based on an increase in the coolant temperature are the most widespread now. Such PSSs are placed, as a rule, at the core outlet and their temperature sensitive element is near the coolant. When coolant temperature reaches the set point, the element activates, and a neutron absorber is set free and falls under its own weight into the reactor core, changing it into the subcritical state. Such devices are sensitive to increase in temperature at the core outlet for all accidents owing to a misbalance of the ratio between power and flow rate. These devices are being developed intensively.

In some devices of similar type, the absorber is not injected by gravity but by an active driving force. In this case, the working elements producing the active force actuate as a result of temperature expansion of a solid or liquid medium. However, according to the estimates, known devices [6, 8] are massive and have considerable inertia. This does not provide an opportunity to quickly shut down the reactor, in case of an event beyond the design basis accident. A detailed classification has been done in IAEA-TECDOC-626 [9] and could be used for reference. Some useful definitions from that publication are provided below:

— Inherent safety characteristic: Safety achieved by the elimination of a specified hazard by means of the choice of material and design concept.
— Passive component: A component which does not need any external input to operate.
— Passive system: Either a system which is composed entirely of passive components and structures or a system which uses active components in a very limited way to initiate subsequent passive operation.

3. BASIC DESIGN PRINCIPLES OF PASSIVE SHUTDOWN SYSTEMS

3.1. LITHIUM EXPANSION MODULES

The concept of the lithium expansion module (LEM) for inherent reactivity feedback is illustrated in Fig. 1 [10]. LEMs consist of one or more large reservoirs of ^{6}Li located above the core, with closed ended tubes leading down through the active core region. Lithium-6 placed in the positive void region of the core would result in negative reactivity insertion because neutron absorption by ^{6}Li dominates over scattering.

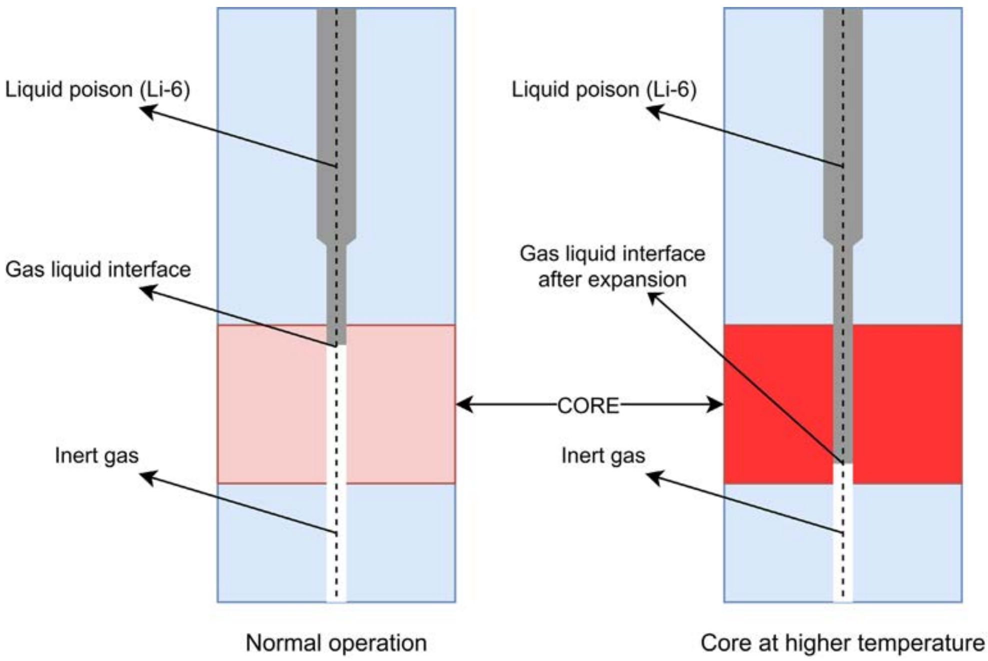

FIG. 1. Lithium expansion module concept.

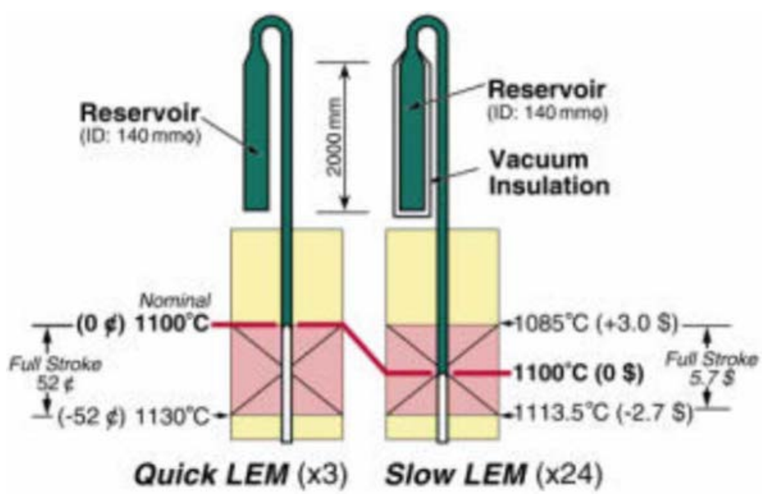

FIG. 2. Elevation of the lithium expansion module gas–liquid interface.

In the upper part of the envelope, ^{6}Li is suspended by surface tension exerted on the gas–liquid interface. The volume expansion of ^{6}Li itself actuates the LEM. During standard operation, the ^{6}Li in the tubes is suspended above the active core by argon gas. As the temperature rises, the ^{6}Li inside the reservoir expands, which causes ^{6}Li to be pushed down into the tube and into the core region, compressing the argon gas.

During the nominal operation, the gas–liquid interface is placed at the active core top. In case of decrease of the core outlet temperature, the gas–liquid interface increases without any positive reactivity insertion as expected.

The LEM can also provide positive reactivity insertion (slow LEM). In fact, the gas–liquid interface in the nominal operation is placed in the active core region as shown in Fig. 2. If the core outlet temperature decreases, the gas–liquid interface increases and positive reactivity is added, and vice versa. Slow LEMs are affected only by moderate thermal transients resulting from burnup reactivity swing and primary flow rate control, and thus have the role of automated burnup compensation and partial load operation in accordance with the primary flow rate.

3.2. LITHIUM INJECTION MODULES

In recent years, a lithium injection module (LIM) innovative system has been proposed for the RAPID fast reactor concept in Japan [11]. The concept of the LIM for inherent ultimate shutdown is illustrated in Fig. 3. The LIM also comprises an envelope in which 95% 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).

Reactivity insertion of the LIM occurs in 0.24 s, which is shorter than the as much as 2 s required for free drop of conventional scram rods.

Similar 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.

The difference between LEMs and LIMs is that the former can achieve both negative and positive reactivity feedback reversibly, and the latter can only achieve negative feedback permanently. The injection temperature, which depends on the requirements of the core design, can be selected by choosing among several candidate materials for the freeze seal. Thus, freeze seal design is the key issue to ensuring accurate injection temperature

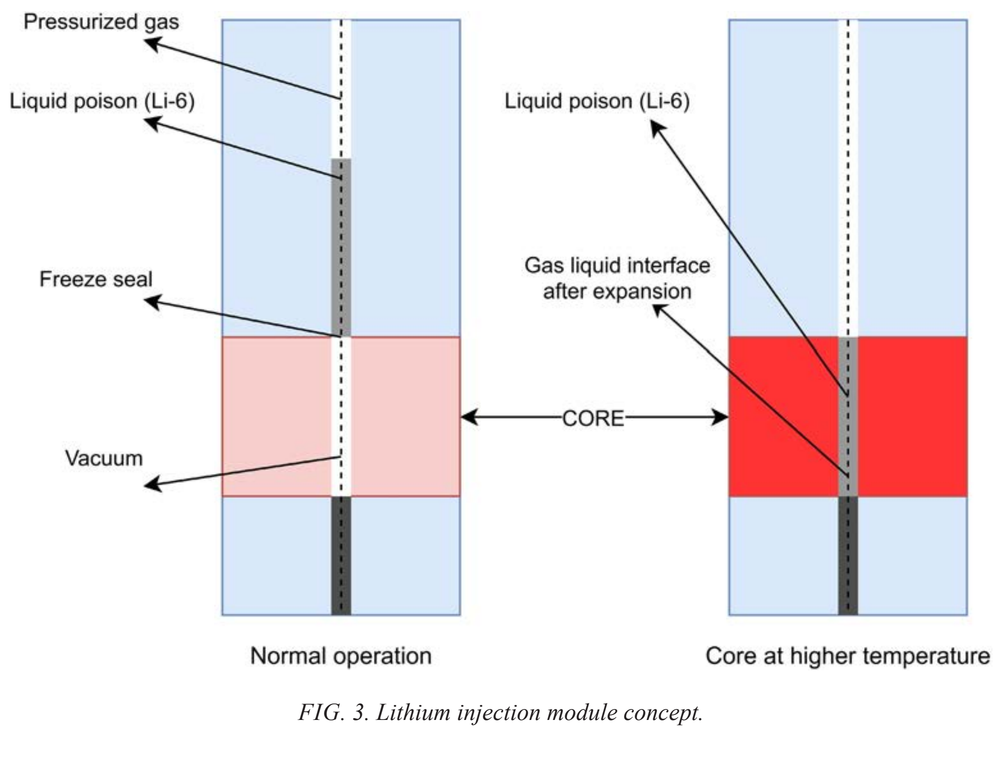

FIG. 3. Lithium injection module concept.

over the design lifetime. A freeze seal segment consisting of copper–nickel alloy (trade name L-30) assures an injection temperature of 1240°C. When adopting aluminium for the freeze seal, LIM injection would be performed at 660°C. This innovative concept has undergone some experimental verifications of its performance, such 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 in Ref. [11].

3.3. 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 Fig. 4. 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 delatches at the detach surface and drops together with the control rod into the reactor core. The Curie point SASS is a simple structure and has flexibility in its detaching position.

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

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

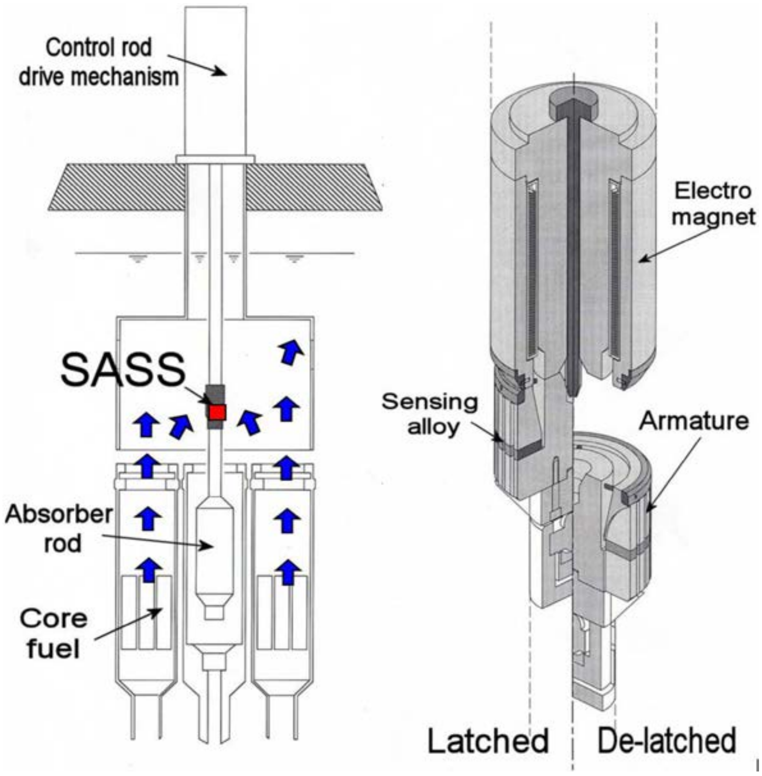

FIG. 4. Self-actuated shutdown systems concept (reproduced from Ref. [12]).

3.4. THERMOSTATIC SWITCHES

Thermostatic switches are classified as non-intelligent sensors. Here, non-intelligent means that they have direct[5] action on the circuitry of the safe shutdown system. This safe shutdown system is engineered to fail into a

[5] 'Direct' here means via reasonably few simple components in an obvious manner.

fully inserted state upon this action. Thermostatic devices are sensitive to temperature rises, a symptom of a large number of accidental events.

3.4.1. Advantage

An advantage of sensing switches is that they may be distributed all over the core, at different positions, to improve sensitivity to local effects, while also providing redundancy for overall effects. Switches can be placed close to fuel assembly outlets, where temperature variation is an important and fast indicator in the majority of events. Variation of temperature is less sharp in some passive sensor positions in the hot plenum or in the rod channel. The switches' sensing probes have low thermal inertia, giving a fast response time.

The antireactivity worth is that of a full shutdown system. It is a good way to give 'passivity in triggering' to classic systems, which are already passive in action.

Of course, extra wiring is required, but this wiring is external to the vessel, is largely unaffected by internal events and could easily be hardened and monitored for external events. It could also be made fail-safe for events such as cables becoming cut. Connections will be keyed to avoid incorrect mating. External wiring also gives access to the circuitry, enabling online monitoring of its state, something that is impossible or difficult for 'self-contained' devices. These robust switches can trigger 'out of core' safety systems, such as the accelerator tripping system in the case of an ADS.

3.4.2. Future development

The philosophy of having robust switches may be extended to earthquake sensors such as the well known tilt switch used in pinball machines. Note that more advanced[6] switches are under study, but are not considered here.

3.4.3. Design consideration

The system could be designed using reversible switches but with irreversible triggering mechanisms. For example, releasing an electromagnet that is unable to reconnect, even if the current is switched back on.

Here we are considering simple mechanical switches, as they are mature and robust, and they can be radiation hardened and resistant to high temperatures. They can be placed reasonably close to the core and are easily integrated through classic wiring.

Designers should take into account potential arcing due to transient[7] effects and show preference to systems that toggle from open to closed with no stable intermediate positions, in a similar manner to a micro switch[8]. Even if a switch can tolerate arcing for a limited time and for a limited number of actuations, it is better to keep occurrences of the phenomena reasonably low, even if an electric arc is unavoidable with classic mechanical switches.

3.5. LYOPHOBIC CAPILLARY POROUS SYSTEMS

A lyophobic capillary porous system (LCPS) consists of a capillary porous matrix and lyophobic liquid (i.e. non-wetting of matrix). Compensation (stabilization) of pressure is the result of a change in the LCPS's volume at reversible filling–emptying of the porous matrix by the liquid.

Processes in fusible accident protection devices using the lyophobic effect in comparison with the process in traditional fusible elements have a number of characteristic features connected to their nature [13]: stored energy, generation of force and hyperdilatation at melting.

In most cases, volume increases when a substance melts (usually by some percentage points). Experimental data received by Lipchin [14] testify that the value of linear shrinkage at crystallization of aluminium is ~1.8% (under pressure P = 0.1 MPa) and ≈ 0.17% (at P = 100 MPa). The shrinkage porosity of aluminium which was crystallized under pressure P = 0.1 MPa is ≈ 0.3%, and at P = 100 MPa it is ≈ 0.05% [14].

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[6] Thermionic switches, for example, seem promising.

[7] Most safety system are inductive in nature. Their overvoltage may be reduced by use of snubber.

[8] The micro switch was invented and trademarked in 1932; it is 'micro' in the sense of microelectromechanic, not in the present sense of microelectronic.

The volumetric shrinkage for crystallization of aluminium at atmospheric pressure is about ≈ 5.5%. In order to decrease the shrinkage, it is possible to perform the crystallization under high pressure (~100 MPa), but this is technically difficult.

Thus, there is a need for complex systems that have a high and controllable value of dilatation at melting, for example, LCPS.

Calculation results of dilatation at working medium melting in LCPS are shown in Fig. 5.

From Fig. 5 it is visible that the process of getting the fused working medium out of the pores at appreciable (>5%) open porosity is prevailing. At high porosity, the dilatation effect in LCPS is tens of per cent; it is higher by far than the dilatation of the working medium.

This explains the interest in using the effect of the hyperdilatation of LCPS in a number of technical devices, first of all in protective ones.

In patent No. 2138086 filed in the Russian Federation [16], a thermal sensitive starting device is proposed. The device contains an elastic container filled by the temperature sensitive substance with both fixed and mobile ends. The mobile end is connected to the trigger mechanism. The capillary porous material, which is not moistened by the temperature sensitive substance it is filled with, is introduced into the elastic container. The melting temperature of the substance corresponds to the temperature of the device in operation. The radius of the pores in the material satisfies the condition [17]:

$$r < \frac{2\sigma|\cos\theta|}{P} \tag{1}$$

where r is the radius of pores; σ is surface tension of the temperature sensitive substances in the liquid state; θ is the contact angle ($\theta > 90°$); and P is the elastic container (sylphon) pressure from external factors.

In the device, the melting of the temperature sensitive substances releases the stored energy determined by:

$$E = \int_{V2}^{V1} P_{\mathrm{L}} \cdot dV = \int_{\Omega 1}^{\Omega 2} \sigma \cdot \cos\theta \cdot d\Omega \tag{2}$$

where $P_{\mathrm{L}} = \frac{2\sigma|\cos\theta|}{P}$ is the capillary pressure of Laplace and Ω is an interphase surface of contact between the liquid temperature sensitive substance and capillary porous material.

The device consists of a trigger mechanism with claws holding a neutron absorber (Fig. 6). The PSS operates (releases the absorber) at the given rise of the sodium temperature beyond nominal. Operation of the PSS prevents the process of sodium boiling in the reactor under the worst beyond design basis accident conditions of the BN-600 (BN-800) reactor, with shutdown of primary coolant circuit pumps, failure of active protective systems and removal of the rods with an efficiency of 0.24% $\Delta k/k$. The availability of three working elements allows the use of substances with different melting temperatures and raises operation reliability.

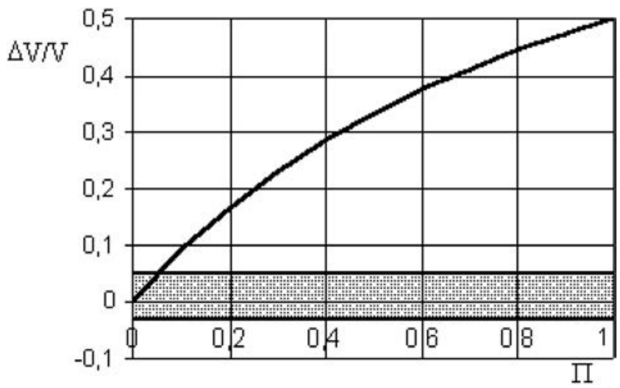

FIG. 5. The relation between ΔV/V in a lyophobic system and the matrix porosity (Π) (reproduced from Ref. [15]).

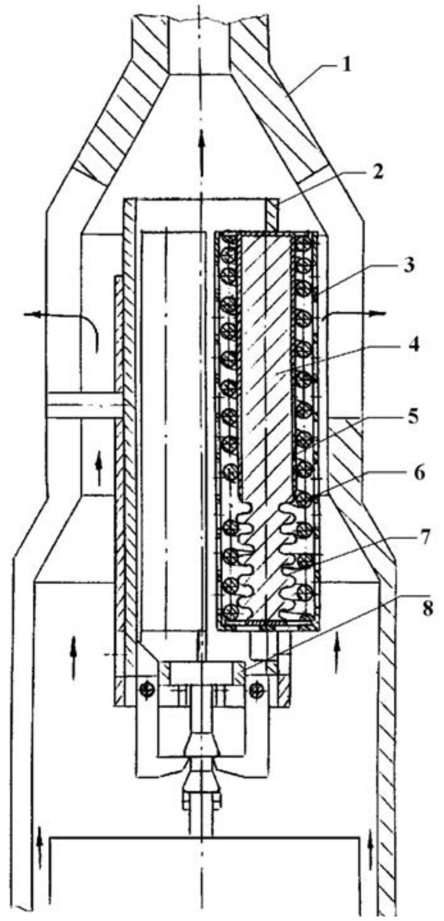

FIG. 6. The passive shutdown system with three sylphon working elements (reproduced from Ref. [15]): (1) head of a core assembly of passive type; (2) movable bush; (3) movable cylinder; (4) working medium; (5) container; (6) high temperature spring; (7) sylphon; (8) stop ring.

If the ambient temperature is more than the critical value (600–650ºC), the claw is pushed out from the vessel and releases an absorber of neutrons. The device operates based on several independent force factors:

— Melting of a temperature sensitive substance (for example, aluminium or magnesium) first of all on the sylphon contour (its corrugations are released and under the influence of a high temperature spring are extended up to the necessary size for releasing the claws and dumping an absorber into the core).
— Hyperdilatation at melting of the LCPS.
— Temperature sensitive substance increasing in volume owing to phase transition, temperature and volumetric expansion. It also provides sylphon lengthening and causes dumping of the neutron absorber. At the same time, significant force is generated.

Thus, the device works to dump an absorber by itself and it provides high reliability of operation. For operation of the device, full melting through the section of temperature sensitive substances is not necessary since first of all there is a melting of temperature sensitive substances along the elastic container contour. Therefore, the capillary porous material can be placed close to the walls of the container.

Hence, using a small elastic container for a capillary porous matrix filled by non-wetting temperature sensitive substances, it is possible to cause significant lengthening of the container. Using non-wetting capillary porous matrices with various sized pores and capillaries allows for adjustment of the force.

Experimental study of characteristics of the sylphon working elements was carried out for a special liquid metal installation. The maximal lengthening of thesylphon was achieved in 5 s, while the velocity of lengthening was approximately constant. After the tests, the sylphon was visually examined. There were no leakages of aluminium or traces of corrosion. Residual axial deformation of the sylphon was noted.

The design of the 'sylphon container' system has much higher axial stability. In the proposed design, the problems related to placing a high temperature spring are solved. The scheme of the PSS with three sylphon container working elements is demonstrated in Fig. 6. The experimental justification of the lengthening size is carried out for models of the sylphon container working elements (d_s = 16 mm, l_s = 44 mm, d_c = 28 mm, l_c = 50 mm). Comparison of the experiments and calculation results is shown in Fig. 7.

There is good accordance between the calculations and measurements shown in Fig. 7. This provides proof of applicability of the techniques developed for calculations of characteristics of working elements of the PSS.

Figure 8 demonstrates the estimated calculations of sodium temperature dependence on time at the outlet of a reactor core and of the subassembly with maximal power under beyond design basis accident conditions in the BN-800 reactor [15].

When modelling conditions of a subassembly with maximal power, the operation of the PSS ranges from 9 s up to 12 s (on various effects) from the beginning of a beyond design basis accident. The coolant temperature

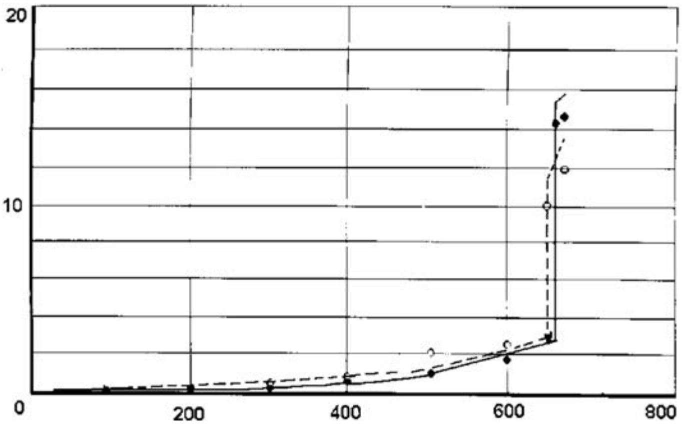

FIG. 7. Lengthening of passive shutdown system aluminium and magnesium as temperature rises: O — aluminium (experiment, calculation); ● — magnesium (experiment, calculation) (reproduced from Ref. [15]).

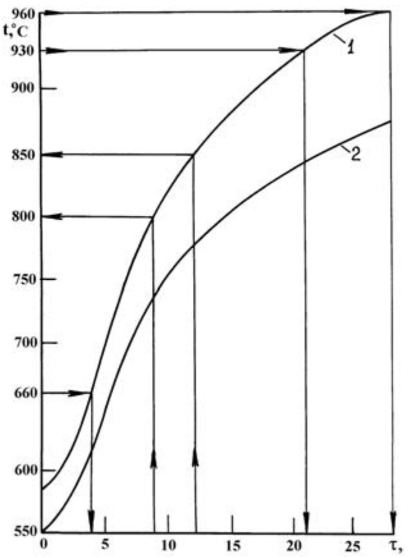

FIG. 8. Change of coolant temperature under beyond design basis accident conditions in the BN-800 reactor: (1) a subassembly with maximal power; (2) the reactor.

at the outlet of the subassembly with maximal power will be ~800–850°C (i.e. there is an extreme underheating (more than 100°C) up to maximum permissible temperature (930°C) and up to the boiling temperature of sodium (960°C), and there is almost double the time reserve to reach those temperatures).

Reliability of the devices (according to patent No. 2138086 filed in the Russian Federation) is experimentally proved at temperatures of operation equalling 650°C (magnesium) and 660°C (aluminium).

3.6. FLOW LEVITATED ABSORBERS

During the reactor's normal operation, the absorber rods are hydraulically suspended in the coolant flow, with their absorbing part above the core. When the coolant flow rate through the core decreases (e.g. in ULOF accidents), the rods are released into the core under their own weight.

A lot of work — theoretical and experimental — has been performed in order to assess SASS feasibility (e.g. the Russian Federation developed and tested different versions of flow levitated absorbers (mainly based on boron carbide) for BN-600 and BN-800 reactors).

As presented in Ref. [18], the flow levitated rods are placed in a standard shutdown subassembly in the core which is actuated by a decrease of the coolant flow in the reactor core (Fig. 9). Their flow rate actuation threshold has been set at 0.6 of the nominal value, with the reactor being able to run safely on two out of three working loops [13, 14]. The actuation of the rods has been tested in the BR-10 reactor, including during operation.

During the design stage of the device, provisions were considered in order to avoid the accidental rise of the rods during reactor shutdown [18].

Moreover, a full scale simulation of this type of PSS for the BN-600 reactor has been tested in a hydraulic (water) mock-up with the main goal being the assessment of the main characteristics of the device, including the insertion time of the rods [9, 18].

As revealed through experiments, some of the main advantages of the PSS based on hydraulically suspended rods are the following: a simple actuation principle, a time response able to ensure a consistent temperature margin to sodium boiling, a minimum inspection requirement, and an operational configuration that can be restored through a simple procedure [18].

Also, some drawbacks have been identified: the same principle of insertion as the standard shutdown systems is in use, a limited range of the coolant flow rates can be managed and some operational parameters can be affected by changes in some hydraulic features (i.e. deposition of impurities and oxides).

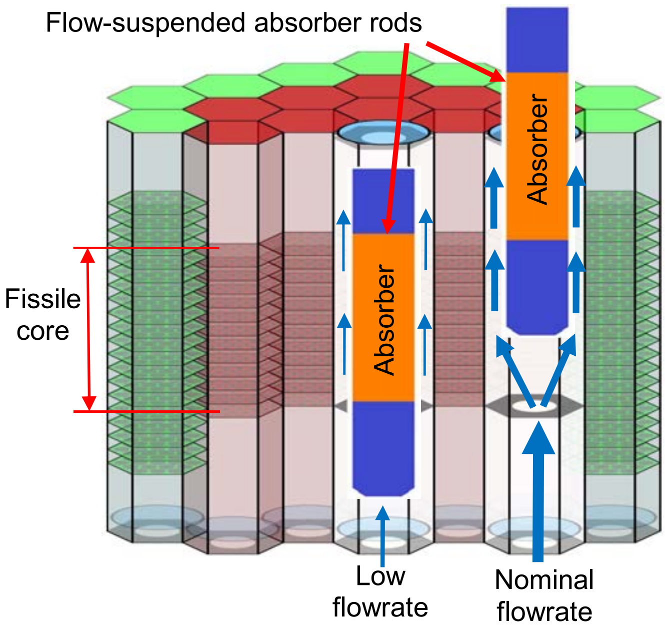

FIG. 9. Passive shutdown system with flow levitated rods for BN-800 (adapted from Ref. [18]).

3.7. CARTESIAN DIVERS

In general, a Cartesian diver is a device which contains a liquid in which a neutrally buoyant object (the diver) is placed. Pressure is communicated to the inside of the device and the gas in the floating object is compressed, causing the object to sink. In a reactivity feedback device, the diver portion would be an absorbing material incorporated into a neutral-buoyancy float [19, 20]. This is shown schematically in Fig. 10.

The liquid here is molten lead and the device would have communication with the coolant pressure, such as bellows, shown here at the top of the device. A loss of pressure would allow the gas containing float to expand, increasing its buoyancy. The diver would then float up into the active core region and help to shut the reactor down.

Another variation on this concept is the 'Galilean thermometer' in which a rise in liquid temperature causes the float to expand, increasing its buoyancy, and again rise into the active core region. A device combining both features can be envisioned which would protect against both loss of coolant accident (LOCA) and UTOP events.

Another scheme for implementation of the Cartesian diver concept is shown in Fig. 11. In this device, a molten Pb–Bi filled canister is placed in the reflector region and vented to the coolant. Inside the molten Pb–Bi, a poison float is placed consisting of an absorber and a gas core with bellows.

Upon either loss of pressure or excessive rise in temperature, the diver becomes more buoyant and floats into a midcore position, causing the core to shut down. When in the bottom (or withdrawn) position, the molten Pb–Bi provides excellent reflection of neutrons during normal operation. The device could be made quite large without displacing core fuel. The radial reflectors could consist of several of these devices with solid reflectors taking up the rest of the core periphery.

3.8. LEVITATED ABSORBER PARTICLES

This SASS is similar in its principle to the flow levitated rods. The difference is that it employs flow levitated absorber balls instead of rods. In addition, in order to widen the range of accidents by which the device is actuated,

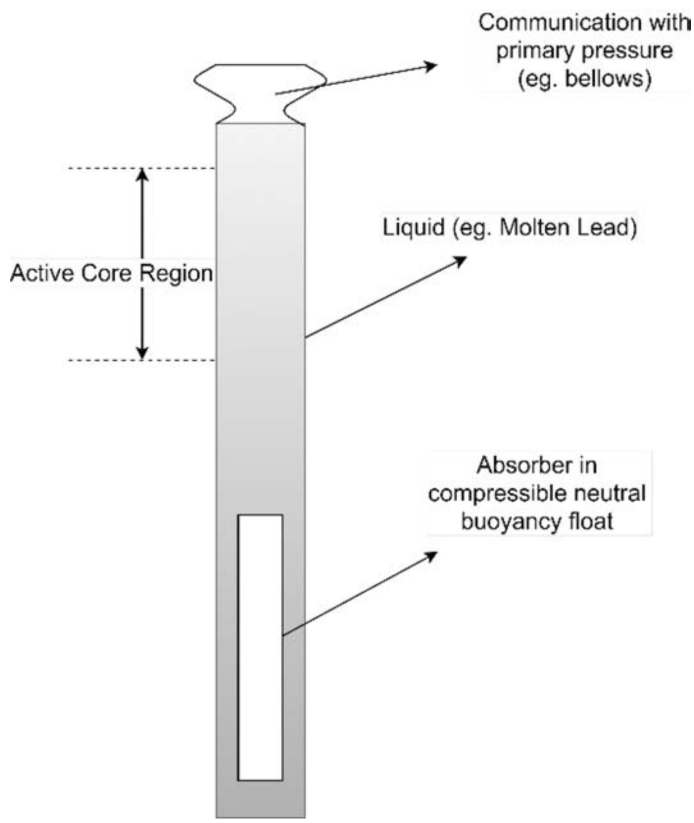

FIG. 10. Cartesian diver reactivity feedback device (adapted from Ref. [20]).

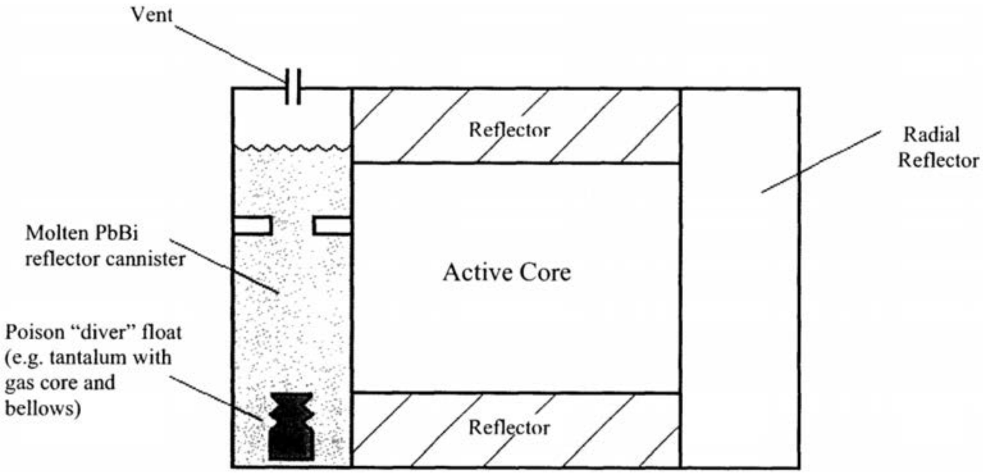

FIG. 11. Alternate Cartesian diver design for use in reflector (reproduced from Ref. [20]).

it contains a shutoff valve actuated by a thermal device based on a Curie point magnet (CPM). The CPM is actuated by the increase of the coolant temperature and blocks the passage of coolant through the shutdown device, which also results in the insertion of absorber balls into the core.

Figure 12 shows a schematic diagram of the levitated absorber system.

This SASS has some advantages over the flow levitated rods (e.g. it is more versatile and ensures safety not only in ULOF, but also in UTOP and ULOHS accidents [18], and a significant deformation of the core cannot prevent the insertion of the absorber).

There are also some drawbacks identified in Ref. [18]: the leak of the absorber balls from the device if it loses integrity, the possibility of the absorber balls jamming in the upper or lower position and the possibility of the absorber balls self-welding.

A more recent study found in Ref. [21] investigated a new concept of SASS that is a combination of some of the strong points of the currently existing concepts, including the hydraulically suspended absorber balls.

Figure 13 presents the new concept, which consists of an empty tube as wide as a fuel pin in the centre of every fuel assembly. The tube is divided into two sections by means of an aluminium seal similar to the melting seal in the LIM concept. In the upper region, above the active core, spherical neutron absorbing boron carbide particles are placed. In case of overpower and loss of coolant transients, the seal will melt. The absorber balls are then no longer supported and fall into the active core region, inserting a large amount of negative reactivity.

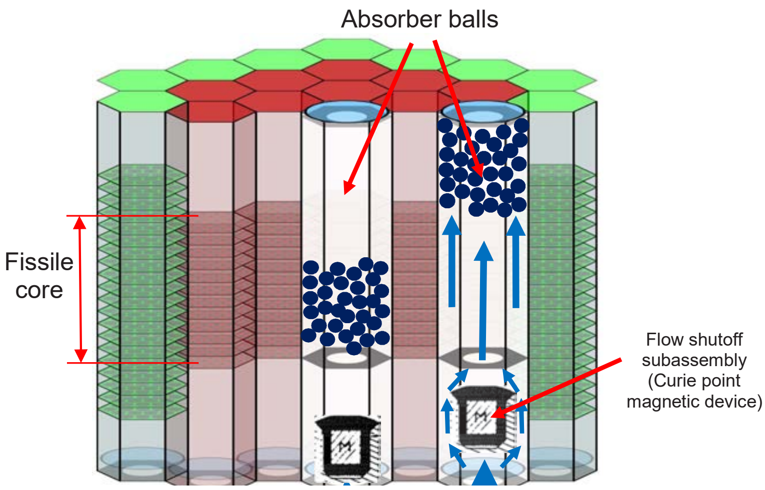

FIG. 12. A levitated absorber system for self-actuated shutdown system (adapted from Ref. [18]).

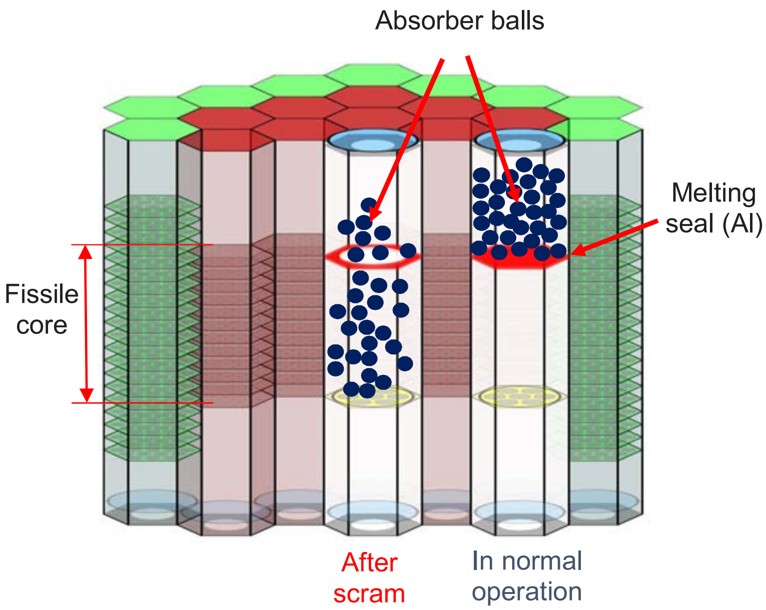

FIG. 13. Concept for a secondary scram system using absorber balls and an aluminium melt seal.

The concept is also self-actuating, just like the hydraulically suspended absorber balls and the liquid insertion with melting seal. It is, however, mechanically simpler than the hydraulically suspended balls, which might contribute to the overall reliability. Owing to the flow-like behaviour of the absorber balls, it is less prone to failure due to channel deformation/blocking than the absorber rods are, even though this issue has to be investigated.

Given the novelty of the concept, the study investigates issues such as neutronic effectiveness (based on a variant of the Multi-purpose Hybrid Research Reactor for High-tech Applications (MYRRHA) core), flow dynamics of spherical absorber particles by simulations and experiments, ideal particle size and blocking probability.

3.9. ENHANCED THERMAL ELONGATION OF CONTROL ROD DRIVELINES

A control rod enhanced expansion device (CREED) is the second component of the ‘third shutdown level’ for the EFR [18]. It is a SASS that inserts an absorber rod into the core because of an increase in the core outlet temperature. The purpose of the device is to prevent sodium from boiling in the core.

The EFR project studied three variants of the CREED concept. One is based on thermal expansion of a fixed mass of liquid sodium (Fig. 14), the second depends on the elongation of a stack of bimetallic washers (Fig. 15) and the third relies on the thermal differential expansion between the rod shaft and a jacket (Fig. 16). The first two variants have a common principle for absorber rod delatching: a temperature rise results in displacement of one element of the socket in a ball and socket joint, releasing the balls and allowing the absorber rod to drop into the core. For the third system, the differential displacement of the two concentric parts results in a force being applied on the control rod head which separates it from the electromagnet, even if it is not demagnetized. The temperature threshold for delatching was estimated at about 500°C to 600°C and the time delay of the enhanced expansion estimated at about 14 s at nominal flow.

CREED is an effective remedy for ULOF and ULOHS accidents. It ensures that the maximum permissible sodium temperature, 930°C (boiling temperature in core), is not exceeded.

Among the advantages of CREED are its complete independence from the main reactor shutdown systems (RSSs) and its compact form.

Application of the principle of solid absorber rod insertion, which is employed in the standard shutdown systems, constitutes the main drawback of CREED. In addition, the latch is not immune to failure (e.g. as a result of coolant impurity deposition on it, or when two elements in contact with each other get stuck together). The actuation temperature threshold for these devices may vary widely. The first two devices cannot be reset, but the third one can.

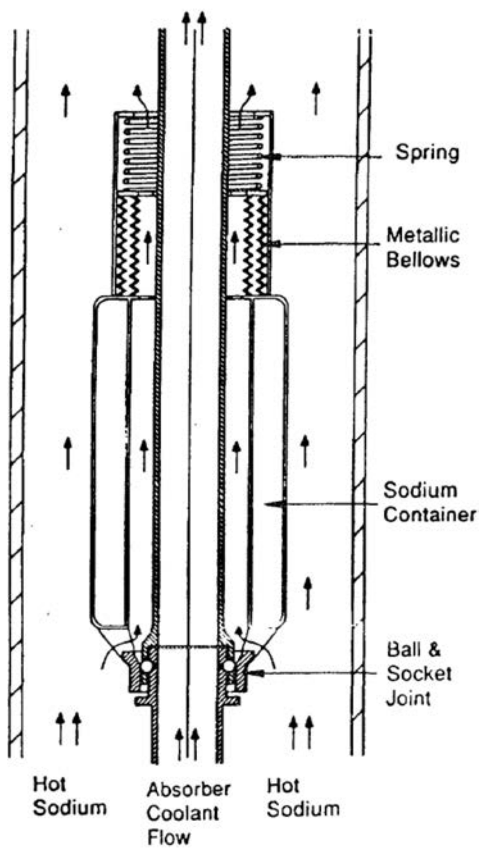

FIG. 14. Enhanced thermal expansion hydraulic concept (reproduced from Ref. [18]).

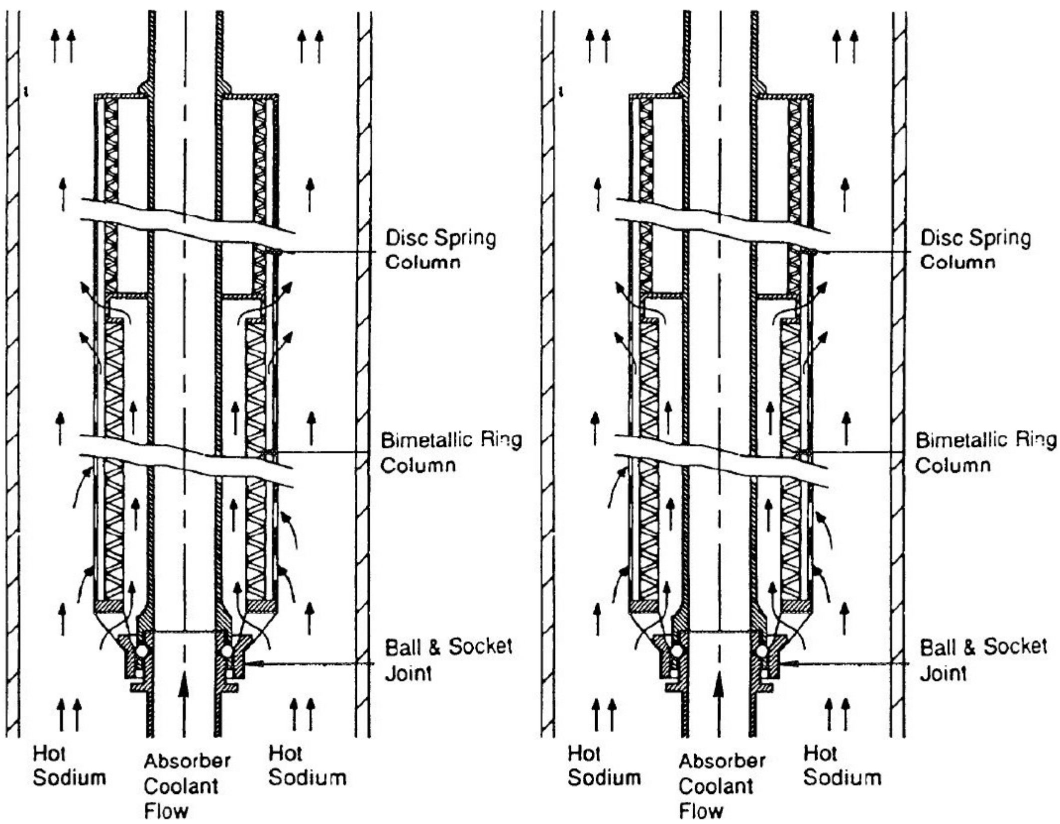

FIG. 15. Enhanced thermal expansion bimetallic ring concept (reproduced from Ref. [18]).

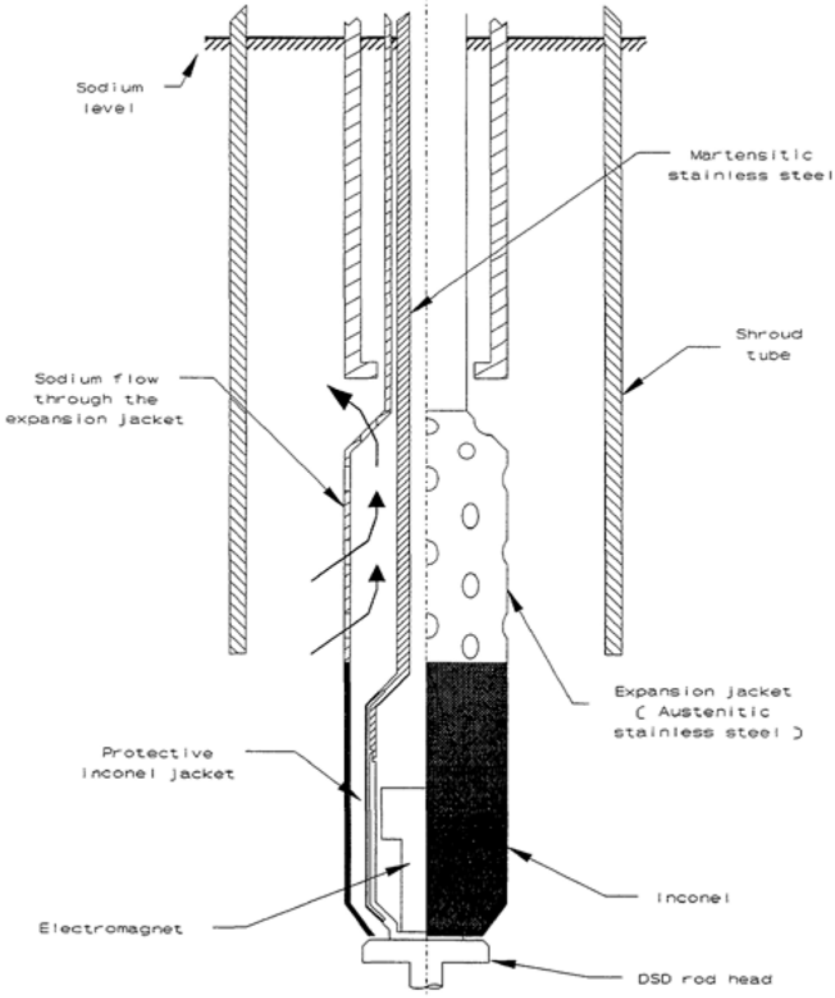

FIG. 16. Thermal expansion device for passive insertion of backup rods in European fast reactor (reproduced from Ref. [18]).

3.10. GAS EXPANSION MODULES

A GEM is essentially a passive device for inherent shutdown which inserts negative reactivity during an unprotected loss of coolant flow in a primary system. The device is basically a hollow removable subassembly sealed at the top and open at the bottom. The gas trapped inside the subassembly expands when core inlet pressure decreases owing to flow reduction, which expels sodium from the subassembly. Neutron leakage increases and negative reactivity is inserted, as shown in Fig. 17. By using a 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, a GEM is not sufficient in large cores and produces negative reactivity only when hydraulic pressure is lost.

The GEM is conceived for advanced liquid metal demonstration reactors in the USA (ALMR) and in the Republic of Korea (KALIMER-150).

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

3.11. AUTONOMOUS REACTIVITY CONTROLS

The autonomous reactivity control (ARC) system in its standard configuration is installed as a modification to a conventional nuclear fuel assembly design. The system consists of two reservoirs, located at the top and bottom

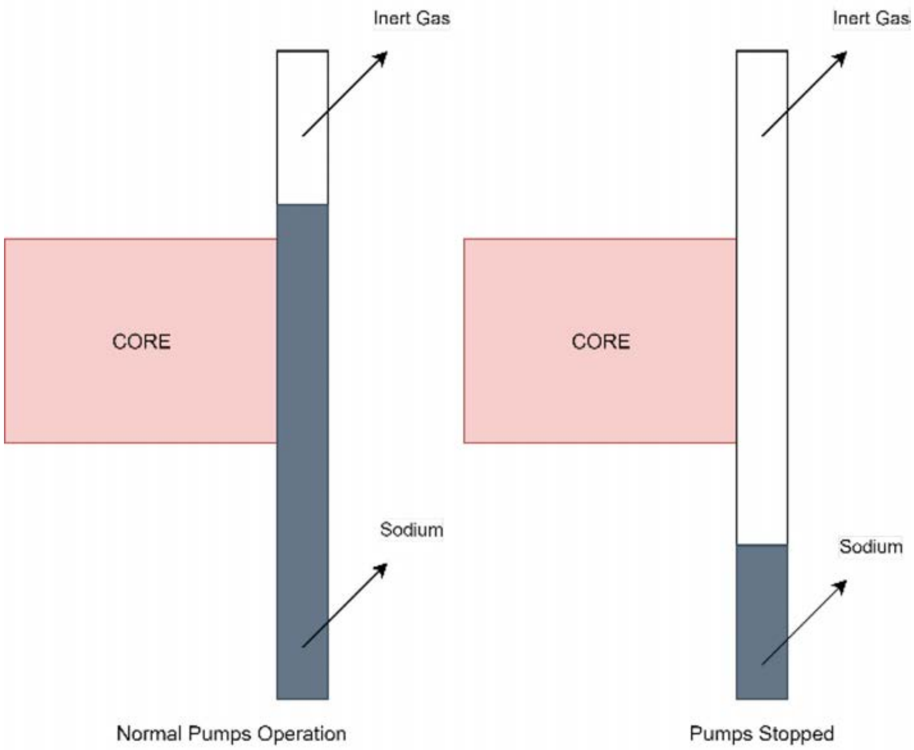

FIG. 17. Gas expansion module design concept.

of the assembly, and two concentric tubes that link the reservoirs. The inner tube is open at both ends and connects the insides of both reservoirs, while the outer tube is open only at the bottom (connected to the lower reservoir) and its top connects to a closed, gas filled reservoir. During operation, the upper reservoir is completely filled with a liquid (henceforth referred to as the expansion liquid), while the lower reservoir contains the same expansion liquid and, floating on top of it, a separate immiscible liquid (henceforth referred to as the absorber liquid). The remaining free volume between the two concentric tubes in the closed system is filled with an inert gas. The outer ARC tube has the same outer dimension as the fuel rods. Installing an ARC tube therefore requires replacing one of the fuel rods in the assembly. A schematic view of the operation of an ARC installation is shown in Fig. 18, and a full detail computer-aided design model of a fast reactor fuel assembly equipped with an ARC system is shown in Fig. 19. During an accident/transient scenario in the reactor, the ARC system responds in the following way, starting from a standard operating condition:

(a) Some event raises the temperature in the core, which heats up the coolant.
(b) The heated coolant flows to the top of the assembly and transfers heat to the expansion liquid inside the upper reservoir.
(c) The expansion liquid in the upper reservoir thermally expands. Since the reservoir is completely filled and sealed at the top, this expansion is directed down the inner ARC tube that connects the two reservoirs.
(d) As expansion liquid enters the lower reservoir from the upper reservoir (through the inner ARC tube), the level of absorber liquid rises toward (and finally into) the active core, while compressing the inert gas above.
(e) The absorber liquid, which has a high neutron capture cross-section, introduces negative reactivity by absorbing neutrons in the core, which in turn causes a reduction in power and temperature.
(f) As the core cools down, the temperature of the expansion liquid starts to fall. Thermal contraction combined with the pressure of the inert gas again lowers the axial level of the absorber liquid until the system reaches a stable critical configuration.

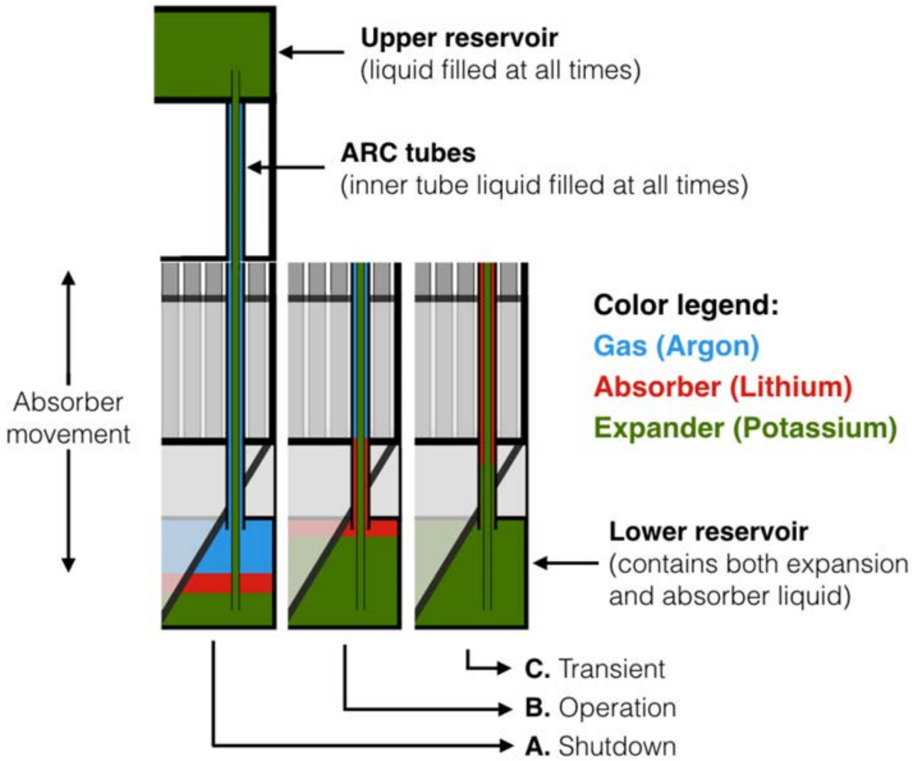

FIG. 18. Schematic view of the autonomous reactivity control system at different states/temperature (courtesy of S. Qvist, University of California, Berkeley).

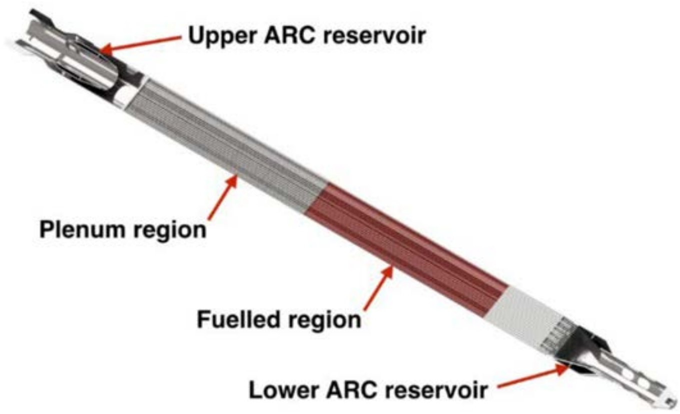

FIG. 19. Fast reactor fuel assembly with autonomous reactivity control system (courtesy of S. Qvist, University of California, Berkeley).

3.12. TRAVELLING WAVE REACTOR THERMOSTATS

The use of ^{6}Li for reactivity control was introduced along with the original travelling wave reactor (TWR) design by Teller et al. [22]). The system devised by Teller et al. [22] for the TWR consists of two connected metallic compartments, one filled with ^{6}Li and the other with ^{7}Li, fed by capillary tubes. The ^{7}Li, which is permanently located within a compartment in the fuel region, expands upon a temperature increase, which in turn actuates a piston that injects ^{6}Li into a separate compartment located inside a coolant channel. When temperatures decrease, the ^{6}Li retracts down a tube and leaves the in-core compartment. In this way, a passive thermostating reactivity control system with a small impact on core neutronics during standard operation was devised. The system is shown in Fig. 20.

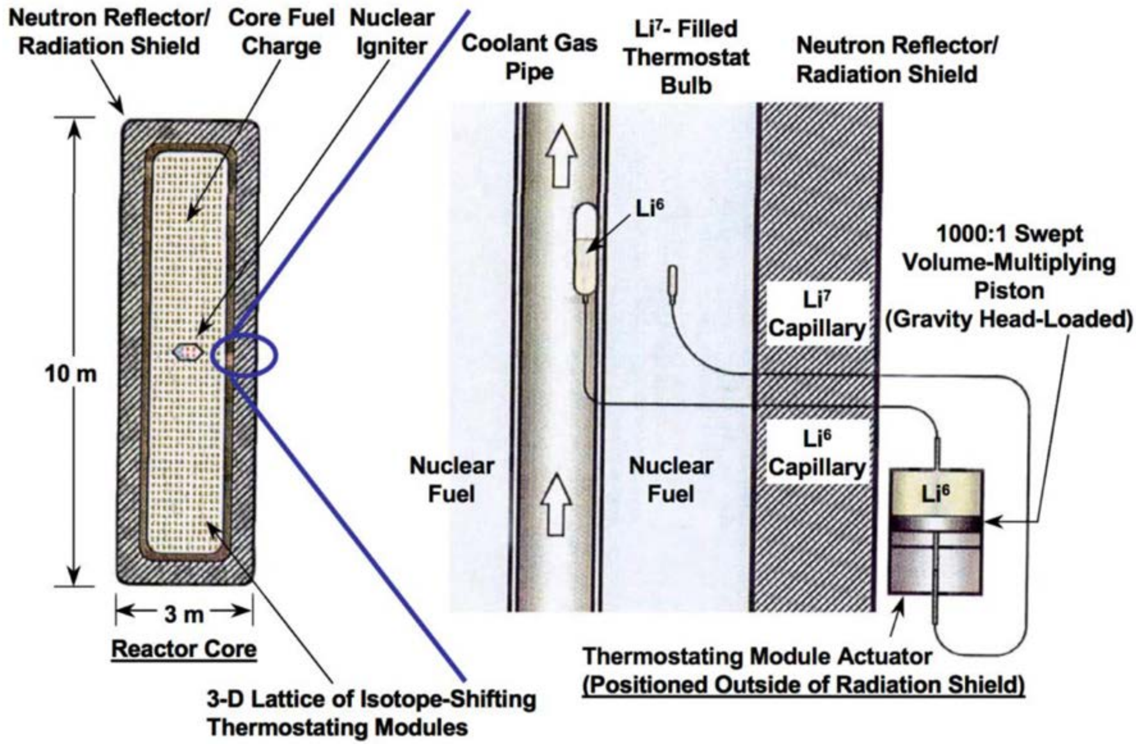

FIG. 20. The travelling wave reactor ^{6}Li thermostating control system.

3.13. THERMO SIPHON BASED PASSIVE SHUTDOWN SYSTEMS

The basic concept behind the proposed system is that one equilibrium configuration is transferred to another equilibrium configuration irreversibly when there is a severe thermal imbalance in the reactor.

There are three immiscible liquids with different densities and they are initially configured as shown in configuration A in Fig. 21. The density of liquid A is about eight times greater than that of liquid B, and ~16 times greater than that of liquid C. Column h_1 of liquid A and h_2 of liquid C are balanced by column h_3 of liquid B and column h_4 of liquid A. When column h_4 of liquid A is heated, it expands to $h_4 + \Delta h$. As long as the level in column h_4 remains below point 1, the left and right limbs are balanced. Once the free level of liquid in column h_4 reaches the top (point 1) owing to thermal expansion (configuration B), it starts falling into volume V_1 and equilibrium gets disturbed. Then the interface at level 2 starts moving down to maintain the equilibrium shown in at the bottom of Fig. 21 (transition from A to B). The area ratio of the tube containing liquid column h_3 to the tube containing column h_4 is such that it further amplifies the imbalance triggered by thermal expansion and causes greater imbalance in the circuit as liquid in the left limb is no longer balanced by liquid column h_4. This cascading effect along with further heating of the liquid in column h_4 sustains the siphoning of liquid from column h_4 into volume V_1. Once the siphon starts, the liquid in column h_4 is totally siphoned to volume V_1. This continues until the equilibrium configuration II is achieved, as shown in configuration B. In this configuration, liquid C in column h_2, which is a neutron absorber, is displaced to the active core zone. Thus shutdown of the reactor is achieved passively by the siphon which is triggered by a thermal imbalance in the reactor.

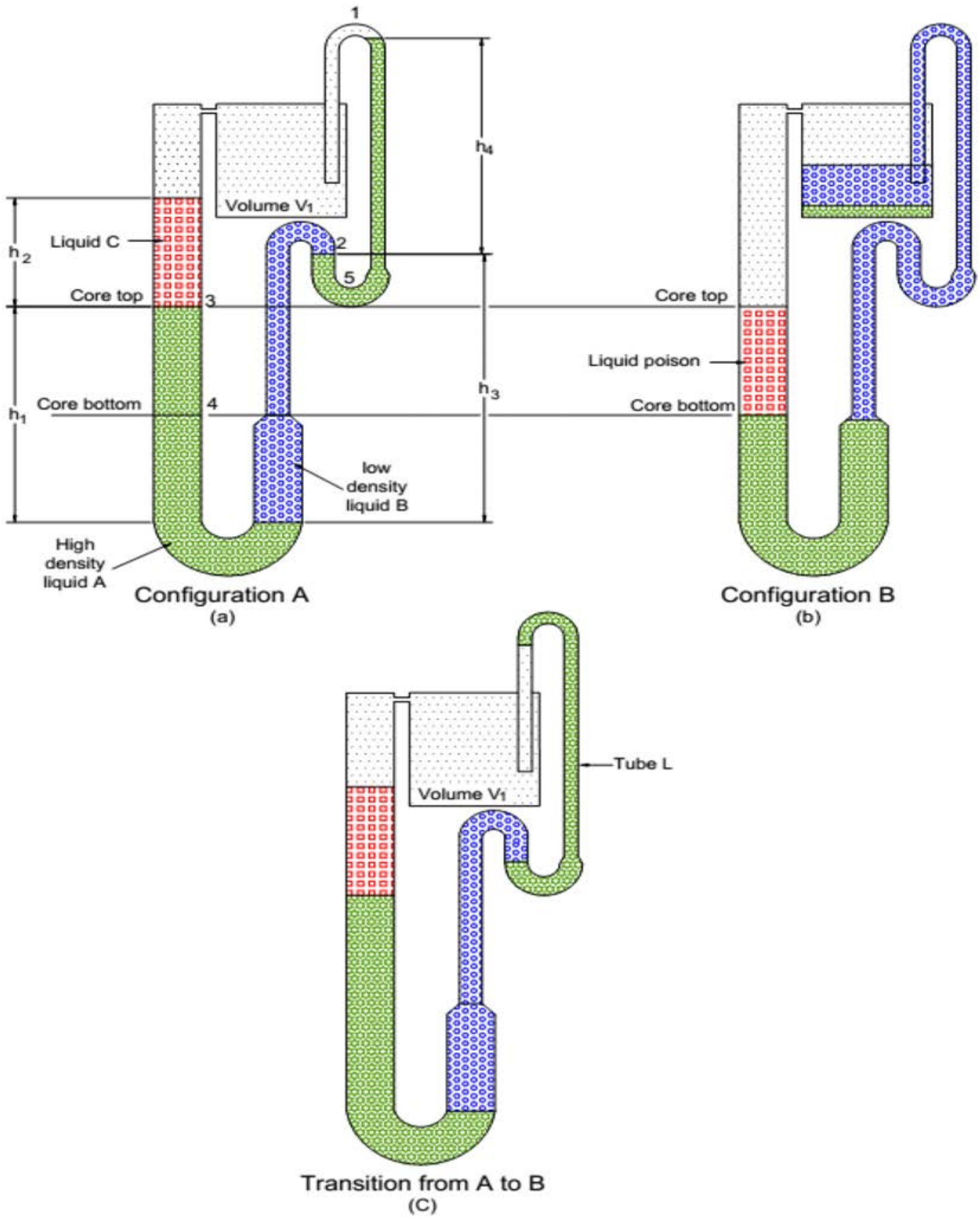

FIG. 21. Stages of the thermo siphon concept: (a) initial configuration, (b) final configuration, and (c) transition from initial to final configuration (courtesy of V. Raju, Indira Gandhi Centre for Atomic Research).

3.14. STATIC ABSORBER FEEDBACK EQUIPMENT

The objective of the static absorber feedback equipment (SAFE) [23] is to deal with the positive coolant temperature coefficient of the SFR; thus, it should be able to provide negative reactivity as soon as the coolant temperature increases for any reason. The SAFE concept is inspired by the control rod driveline (CRDL) expansion feedback in the SFR. It is well known that in the SFR core, the CRDL provides strong negative feedback since it expands with a coolant temperature increase or shrinks for a coolant temperature decrease [24]. Furthermore, the CRDL feedback effect is quick because any change in the coolant temperature in the core region can be quickly transferred to the CRDL owing to the good heat transfer coefficient of the sodium coolant and high thermal conductivity of the CRDL material (usually stainless steel). However, the negative feedback of the CRDL is only available when the control elements are inserted into the core region. Therefore, the CRDL feedback will be very small near the end of life of an SFR core or when the excess reactivity is very small and control elements are largely withdrawn during operation. The principle of the SAFE device is basically the same as in the CRDL feedback. Meanwhile, the SAFE device is designed to provide negative feedback whenever needed. Figure 22 shows the schematic concept of the SAFE device.

Unlike the conventional control assembly, in the SAFE concept depicted in Fig. 22, a fixed absorber is always slightly inserted into the core and it is held by a long steel holding line similar in length to the control element driveline (about 9 m). The long holding line is attached to the upper core structure. It should be noted that the SAFE module is placed in the central region of the control assembly. A SAFE device is installed in a single control assembly. Just like the CRDL, most of the holding line of the SAFE device is immersed in the hot coolant pool. Therefore, the coolant temperature change can be easily conveyed to the SAFE holding line made of stainless steel, and the long holding line can either expand or shrink to provide negative reactivity feedback to the core.

The absorber in the SAFE device is B_4C contained in stainless steel. The size and the enrichment of the absorber can be adjusted depending on the requested negative reactivity feedback. Insertion depth of the SAFE absorber can also be optimized such that the negative reactivity loss is not too high and the resulting feedback coefficient is big enough when required. It should also be mentioned that the burnout of the boron absorber in the SAFE device can be significant because the absorber is always located near the active core. However, since the SAFE device is attached to the reactor cover modules, it can be replaced, if necessary.

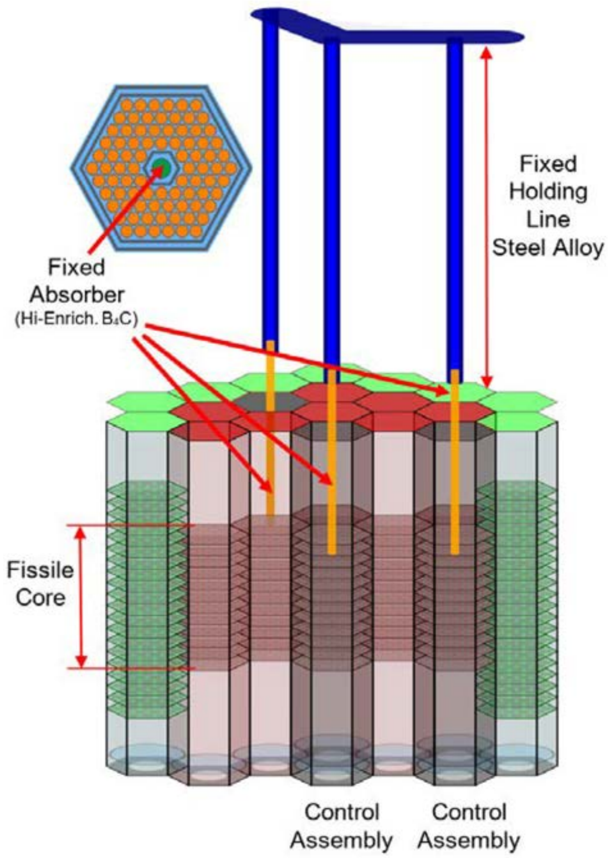

FIG. 22. Static absorber feedback equipment (adapted from Ref. [23]).

4. OPERATIONAL EXPERIENCE WITH APPLIED PASSIVE SYSTEMS

4.1. FAST FLUX TEST FACILITY

The fast flux test facility (FFTF) was operated at the US Department of Energy's Hanford site as a reactor with a fast neutron spectrum [25]. The FFTF was a 400 MW(th) loop type reactor with oxide fuel in two enrichment zones. In the late 1980s, a series of passive safety tests were conducted to achieve the following two objectives:

— Demonstrate safety margins for SFR designs;
— Provide data for validation of computational models.

In the area of applied passive systems, of particular interest was a series of ULOF tests (without scram) that were initiated at power levels ranging up to 50% of the core thermal power rating. Owing to large Doppler feedback and stored heat, oxide fuelled SFRs considered in the USA have smaller margins to coolant boiling and fuel failures after ULOF in comparison to metal fuelled SFRs. To overcome this deficiency, a reactor GEM self-shutdown device was introduced into the core design of the FFTF to mitigate potential consequences of a ULOF event.

The GEM is essentially an empty assembly, sealed at the top but open at the bottom, fitted with FFTF core compatible hardware at both ends to permit insertion into the inner row of the reactor radial reflector. The FFTF GEM assembly design is shown in Fig. 23, while the locations of the nine GEMs installed in the first row of the FFTF reflector region are shown in Fig. 24.

During normal operation, the sodium level in each device rises until the core inlet pressure equals the compressed argon gas pressure, about 30 to 40 cm above the active core height. This device thus provides a mechanism for automatic removal of reactivity if primary flow is lost. In particular, it is a passive feature that protects against a reduction in inlet plenum pressure caused by a loss of primary flow. The loss of pressure causes the trapped argon gas to expand, forcing the sodium in the internal volume back down below the core level. The displacement of sodium increases the neutron leakage from the core, introducing 1.50$ reactivity worth.

The outlet temperature from most core assemblies was monitored via thermocouples located in the above-core instrument trees. These instruments were inside gas filled wells and thus had an inherently slow response time (i.e. the time constant was ~4 min). There were eight core positions into which 'open test assemblies' could be inserted in the FFTF core. These positions had their own instrument leads to support in-core instrumentation. During GEM testing, two core positions were provided with special thermocouples that had faster response times than the thermocouples in the instrument trees. In addition, these fast response thermocouples were located closer to the top of the fuel pins, reducing the delay in their response due to coolant transport time.

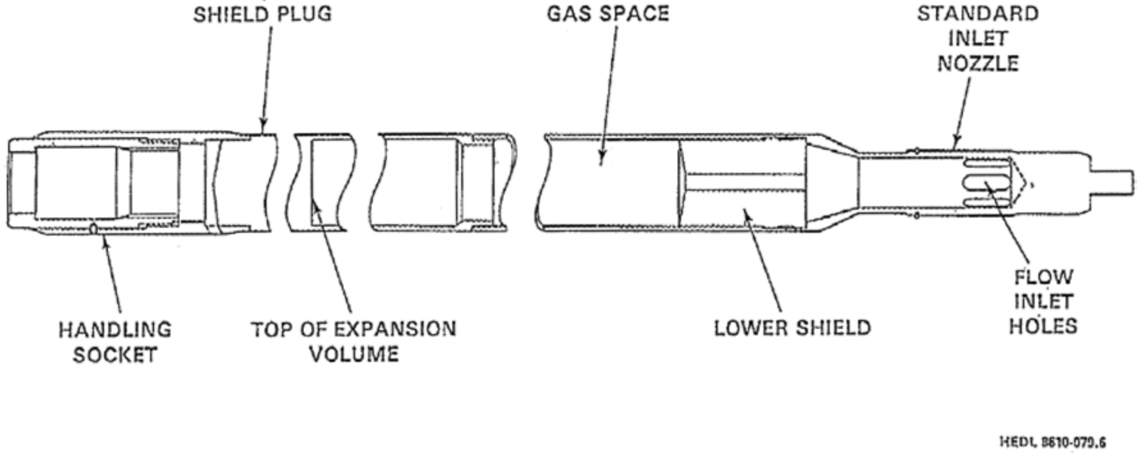

FIG. 23. Fast flux test facility gas expansion module assembly (courtesy of D. Wootan, Pacific Northwest National Laboratory).

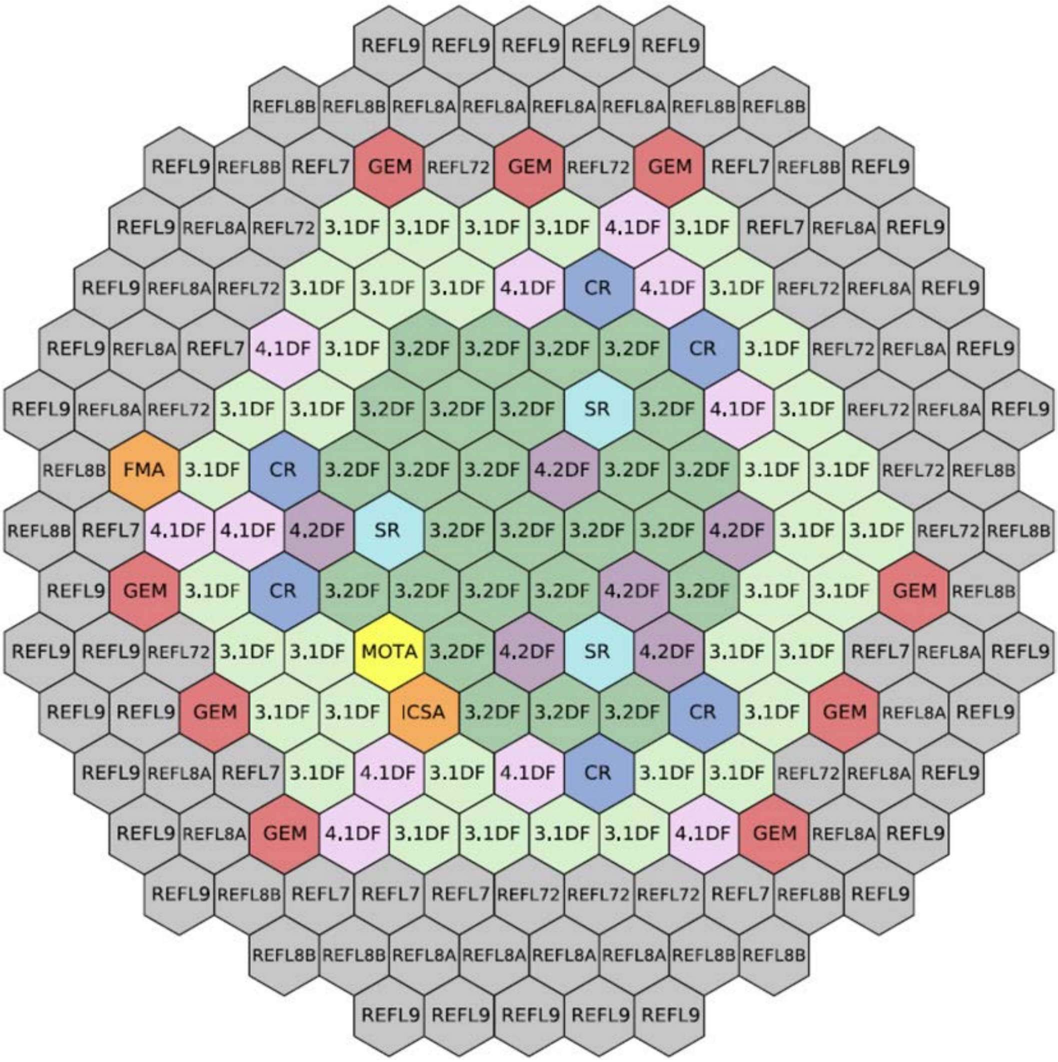

FIG. 24. Locations of gas expansion modules in the core of a fast flux test facility (courtesy of D. Wootan, Pacific Northwest National Laboratory).

The first series of ULOF tests were conducted with the primary pump pony motors kept on throughout the transient so that the minimum flow reached in each test was 9%. Peak coolant temperature for this test series was ~493°C. With knowledge gained from these initial tests, ULOF tests were then repeated with the same initial conditions, except the primary pony motors were left off so that a direct transition to natural circulation flow in the primary system would occur. The tests were repeated at 10, 20, 30, 40, 45 and finally 50% of full power. The core temperatures measured with the fast response thermocouples are shown in Fig. 25.

The peak core temperature for this test series was 509°C and was measured for the highest power level test (50%). The temperatures show double peak behaviour. The first, sharp peak after trip is associated with rapid initial flow coast down before the GEM sodium levels fell sufficiently to start inserting negative reactivity. Once the GEMs start inserting negative reactivity, the power dropped faster than the flow, which caused the core temperature to drop. As the GEM sodium level approached the bottom of the core and reactivity insertion slowed, core temperatures began to increase. The second, broad peak is associated with flow reaching a steady value while power continued to fall slowly.

These in-reactor tests successfully demonstrated the effectiveness of GEMs in mitigating the consequences of a ULOF event for an oxide fuelled fast reactor core. Passive shutdown was achieved without automatic scram or operator intervention with power levels ranging up to 50% of full power. When assessing the effectiveness of this system for other oxide core designs, one needs to keep in mind the overall core size, as neutron leakage effects at the core radial boundary (which GEMs promote under loss of flow conditions) decrease as core size increases.

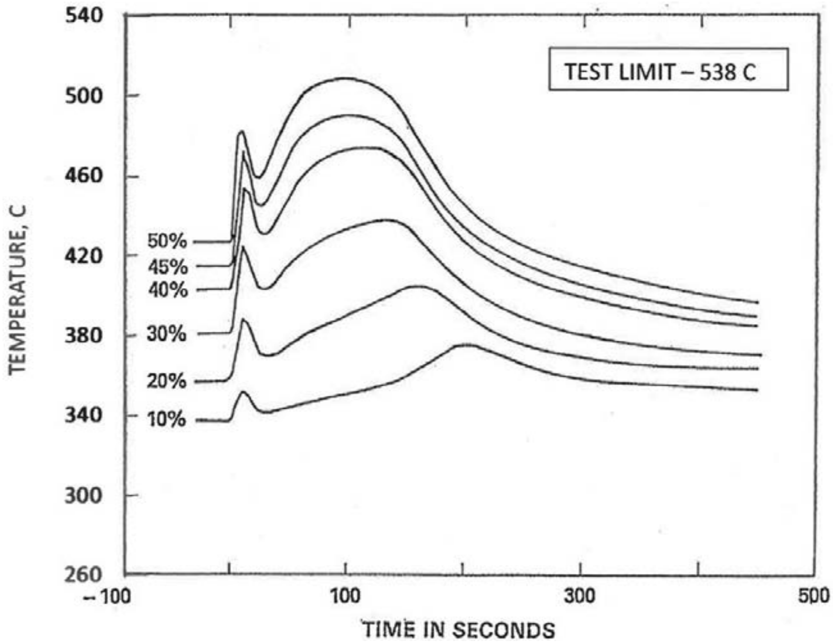

FIG. 25. Temperatures measured at the outlet of assembly row 2 during an unprotected loss of flow sequence initiated at different initial core power levels with nine gas expansion modules located at the core periphery of the fast flux test facility (courtesy of D. Wootan, Pacific Northwest National Laboratory).

4.2. BN-600 (BR-10)

Today, the only SFRs under commercial operation are the BN-600 and BN-800 at the Beloyarsk nuclear power plant in the Russian Federation. BN-600 has been under continuous operation since 1980 and has been instrumental in development of the closed fuel cycle in the Russian Federation.

4.2.1. Main features

Experience acquired during the initial period of BN-350 operation was used to make changes in the BN-600 design.

The BN-600 is a three loop design with reactor and primary pumps submerged in a large pool of liquid sodium (Fig. 26).

The secondary circuits comprise three loops each with a steam generator and a secondary sodium pump. The steam supplies three 200 MW turbines. Each steam generator consists of eight sections comprising an evaporator, superheater and reheater which are connected by a manifold and can be isolated on both the sodium and the steam sides. A detailed description of BN-600 components and their operation is presented in Ref. [27].

Over its operating life, the BN-600 reactor has been upgraded and the lifetime of its key components extended (including the steam generators, sodium pumps and intermediate heat exchangers (IHXs)). As a consequence, the Russian nuclear regulator, Rostekhnadzor, issued in April 2010 a licence for the operation of Beloyarsk 3 until 31 March 2020. During the licensing process, studies were performed to validate the lifetime of the reactor components for 45 years of operation, which means that another five year licence extension might be possible in the future.

4.2.2. Passive shutdown system with hydraulically suspended absorber rod

One of the most important components of BN-600 is the reactor control and protection system, which contains emergency shutdown elements, automatic control elements, temperature and power effect shim elements, and burnup elements [25].

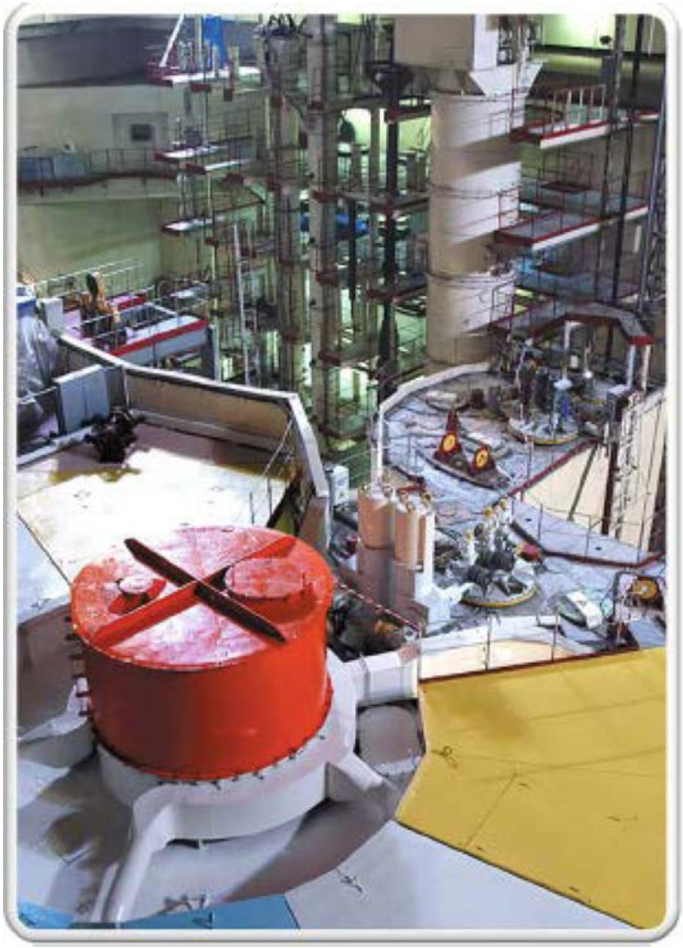

FIG. 26. BN-600 power unit at the Beloyarsk nuclear power plant in the Russian Federation (reproduced from Ref. [26]).

Reactor safety is to be ensured in the case of any accident, even under conditions of total active system failure (i.e. by means of inherent safety features and specific self-actuated systems and elements). Reactor power can be lowered to the safety level under the conditions of active RPS failure via effective negative reactivity feedback, self-actuated shutdown systems or simultaneous application of both [18].

The general requirements of PSSs in SFRs are determined taking into account several factors to ensure the termination of the accident and transition of the reactor to a safe condition.

The channel of the RPS that monitors coolant flow is the most important channel [28]. In the core, the rates at which the temperature rises during the most dangerous failures of sodium circulation through the reactor have reached 100°C/s. Moreover, operating experience indicates that accidents involving various coolant circulation failures in the loops are the most common type of accident. A drop in the rate of coolant flow through the reactor is the most dangerous when all the primary pumps shut down at the same time. A shutdown such as this may be due to a power supply failure from the pump motors. It is for just such a contingency that the protection system channels, which function during a loss of coolant flow through the reactor, have been designed.

A wide range of experiments to prove the feasibility of SASSs have been conducted in the Russian Federation in parallel with theoretical studies.

4.2.2.1. 1988–1989 — development of various passive shutdown system versions

Several design versions of the subassembly were tested. Numerous in-core and simulation experiments were performed in order to confirm the correct operation of these flow levitated rods [15]. Their flow rate actuation threshold was set at 0.6 of the nominal value. Thus, reactors can be also run on two out of three working loops. The actuation mode of the rods has been tested in the BR-10 reactor, including during operation. In case of main circulating pump stop and malfunction of the accident protection system, a rod falls into the core under its own weight at a flow rate of sodium below 0.6 Q_{nom}. To predict accident development, it is also necessary to know the insertion time of the rod into core. Results concerning the characteristics of such devices (including the rod's insertion time) when they are used in power reactors are shown in Table 1 for the various models of PSS investigated.

The rod fall time measurements in the mock-up were conducted on a full scale model of a passive accident protection assembly through simulation/imitation of the flow rate change over time. Experimental results were compared with the calculated values.

The experimental value of the insertion time in the case of the assembly with a guide pipe of 78 mm was ~6.2 s. An increased experimental value of 7.2 s was obtained when a guide pipe of 76 mm was tested. A graph of the agreement among the experimental results and the computed values is presented in Fig. 27.

TABLE 1. RESULTS OBTAINED FOR VARIOUS MODELS OF PASSIVE SHUTDOWN SYSTEM [15]

No. of breadboard versions (rods)	1 (initial)	2 (prototype)	3 (modify)	4 (prototype)	5 (final)	6 (final)
Diameter of a guide pipe (mm)	78	78	78	78	78	78
Wire wrap on an operating link	—	—	—	—	—	—
No. of hinges	3	3	3	2	2	2
Length of a rod (m)	2.1	2.1	2.1	2.06–2.1	2.08	2.08
Weight of a rod (kg)	17.75	18.25	18.25	17.8	17.8	18.0
No. of rows and flow area of perforation (m^2)	—	Six $f = 1.88 \times 10^{-3}$	Six $f = 1.88 \times 10^{-3}$	Six $f = 1.88 \times 10^{-3}$	Six $f = 1.88 \times 10^{-3}$	Six $f = 1.88 \times 10^{-3}$

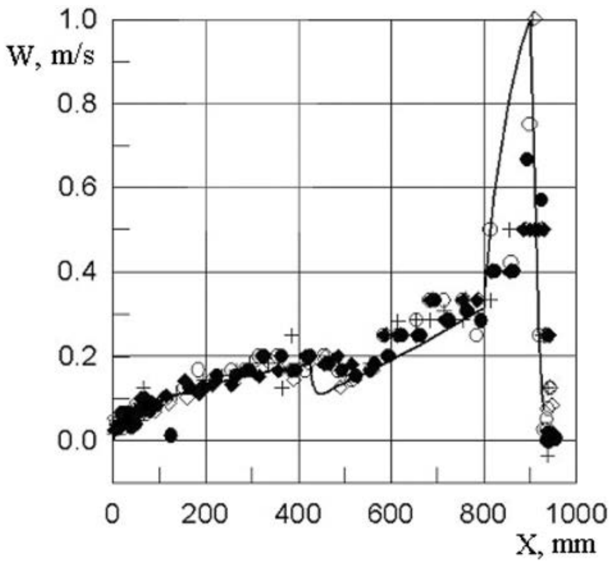

FIG. 27. Fall velocity variation of a rod in passive accident protection assembly as a function of distance (from its actuation point) in a regime cloning a rundown of main circulation pump 1 of the BN-600 reactor (reproduced from Ref. [15]).

The recalculated values of some hydraulic parameters of the passive accident protection assemblies for the transition of water to sodium are listed in Table 2.

Among them, two PSSs (hereafter referred to as PSS 1 and PSS 2) based on the BR-10 standard subassemblies were developed, fabricated and tested in the BR-10 reactor (Table 3 and Fig. 28) [29].

In 1994, the programme of lifetime testing of the two subassemblies in the reactor, including on power actuation, was completed. The tests confirmed the design parameters of the subassemblies, which are recommended

TABLE 2. VALUES OF MAIN PARAMETERS OF EXPERIMENTAL PASSIVE ACCIDENT PROTECTION ASSEMBLY FOR BN-600 OBTAINED FROM EXPERIMENTS AND THE RESULTS OF THEIR RECALCULATION FOR SODIUM [15]

Model version	2	3	4	5	6	AP
Water						
Q_{nom} (1/s)	5.45	4.08	5.33	3.7	5.33	
Q_{fr} (1/s)	7.22	7.22	8.62	10.5	10.5	
τ (s)	6.5	5.0		7.0	5.8	
Sodium						
Q_{nom} (1/s)	6.1	4.57	5.97	4.14	5.97	2.4
Q_{fr} (1/s)	8.09	8.09	9.65	11.8	11.8	
Q_{rod} (1/s)	1.04	1.1	1.01	1.16	1.01	1.03
τ (s)	6.1	4.7		6.6	5.5	<8.0
Q^{l}_{rod} (1/s)	0.14	0.10	0.12	0.08	0.12	>0.053

Note: Q_{nom} — nominal flow rate through assembly; Q_{fr} — flow rate through assembly at emersion of rod from lower to upper position; Q_{rod} — flow rate through a rod at nominal flow rate through the assembly; Q^{l}_{rod} — flow rate through a rod in the lower position at $q = 0.25\ Q_{nom}$.

TABLE 3. TECHNICAL PARAMETERS OF PASSIVE SHUTDOWN SYSTEMS

No.	Parameter	Units	PSS 1	PSS 2
1	Absorber rod cladding diameter (thickness)	mm	22.5 (0.3)	21.5 (0.3)
2	Absorber rod weight	g	~242.0	~225.0
3	Absorber rod efficiency	$\%\Delta k/k$	0.146	0.22
4	Flow rate through the reactor at a rod rise	m^3/h ($\%Q_N$)	96 (48)	81 (40.5)
5	Flow rate through the reactor at a rod drop	m^3/h ($\%Q_N$)	70 (35)	63 (31.5)
6	Rated flow rate through the PSS at a raised rod	m^3/h	0.93	0.97
7	Time of rod drop into the core	s	1.14	0.67

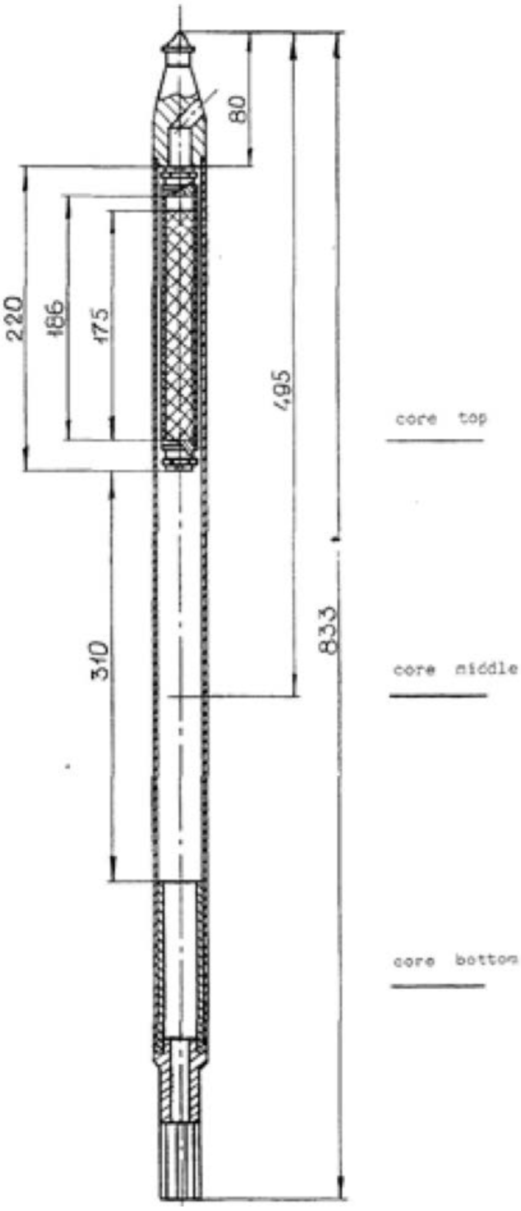

FIG. 28. Passive shutdown system subassembly with hydraulically suspended rod for the BR-10 reactor (reproduced from Ref. [29]).

for use in the BR-10 reactor as standard assemblies. The results of the lifetime testing of the two subassemblies are presented in Table 4.

The total time of operation in the reactor was 218 days for PSS 1 and 1020 days for PSS 2. The number of rod drops during the tests was 38 for PSS 1 and 116 for PSS 2. It should be noted that no rod jamming in the subassemblies occurred and the flow rate values through the reactor at rod rise and drop during the tests did not change.

In 1988–1989, a test absorber subassembly for the BN-600 reactor was designed on the basis of its standard shutdown absorber subassembly, and its full scale mock-up was manufactured for testing in the hydraulic (water) rig (Fig. 29).

The logic of its operation is described in detail in Ref. [29] and refers to three situations: when the reactor is in shutdown, when a signal for shutdown is received or when refuelling is happening.

At a signal for reactor shutdown, the rod is automatically put into the lower working position by the drive stem at the open gripper. In this case the absorber remains in the gripper, as during the time of its movement (~1 s) from the upper working position to the lower one, the primary coolant flow rate does not change. With the reduction of flow rate when changing over the primary pumps to reduced rotations (0.25 Q_{nom}), a decrease of

TABLE 4. RESULTS OF THE LIFETIME TESTING OF THE TWO SUBASSEMBLIES

No. of passive shutdown system	Core cell	Duration (dates)	On power operation, eff. (days)	Accumulated fluence (E > 0, MeV) n/cm^2	No. of actuations (including those on power)
	110	1989-01-03–1989-01-05	0		
PSS 1	110	1989-03-29–1989-08-01	39.55	2.6×10^{21}	38 (10)
	95	1992-09-14–1992-11-17	21.96		
	95	1991-05-15–1992-08-11	151.07		
PSS 2	95	1993-11-23–1994-11-25	127.42	1.7×10^{22}	125 (10)
	95	1994-11-25–1995-06-15	55.5		

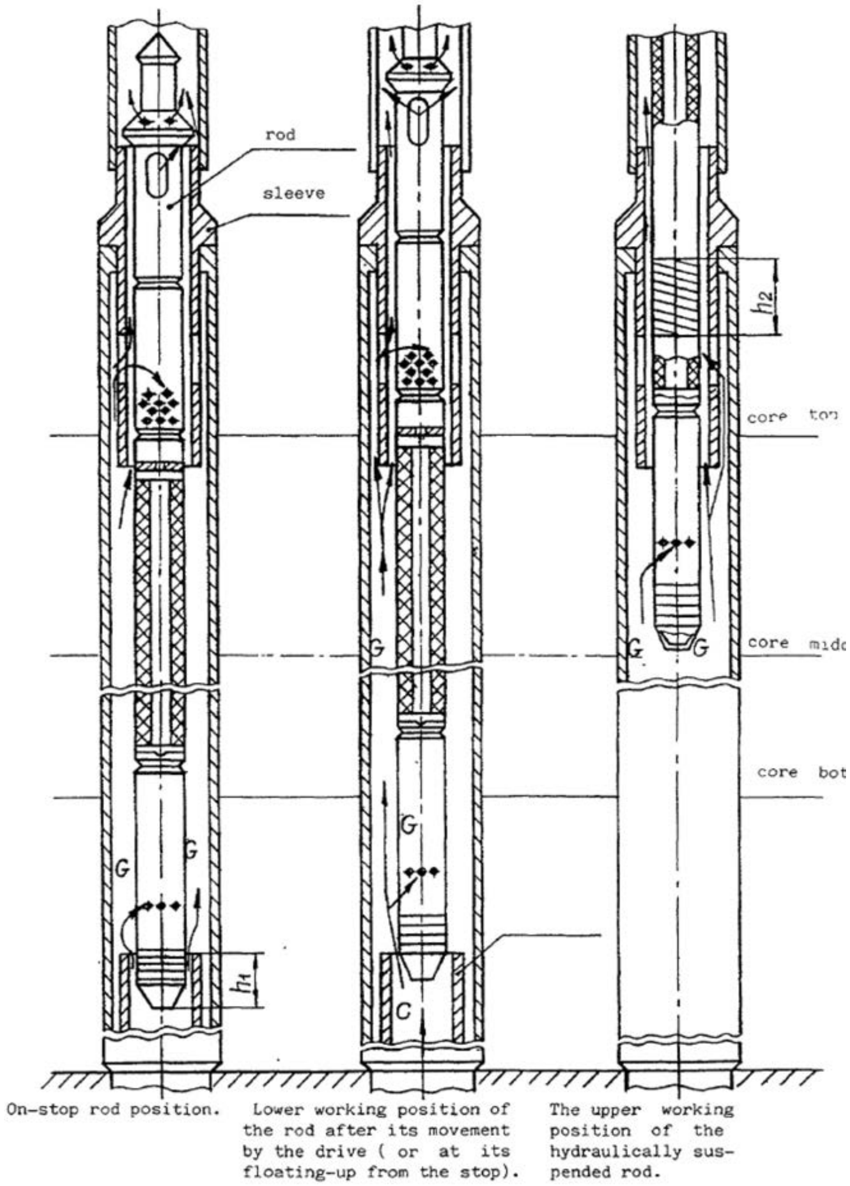

FIG. 29. Passive shutdown system subassembly with hydraulically suspended rod for the BN-600 reactor (reproduced from Ref. [29]).

the hydrodynamic force takes place; at its decrease below the rod weight, the rod drops from the lower working position (80 mm above the stop) into the brake and is held in it. At further flow rate reduction, the rod softly descends from the brake to the stop.

In the case of a drive failure, the rod starts falling into the core by gravity at a flow rate decrease down to <0.6 Q_{nom}. As it falls, it stops at first in the brake (40 mm above the stop) and then, at further flow rate reduction, it softly comes to rest against the stop.

Table 5 shows the main hydrodynamic characteristics of the BN-600 standard safety system subassembly and of the two PSS subassemblies tested in the water rig; the coolant flow rates (L/s) are given scaled to sodium at an operating temperature.

Figures 30 and 31 show the effect of PSS actuation with various rod worths (0.6 and 1.2% $\Delta k/k$) in a beyond design basis accident in the BN-600 reactor with total loss of electric power supply and a failure of all active reactivity control systems for various times of the beginning of PSS actuation (τ_1 = 4 and 14 s) from the beginning of an accident and from absorber insertion into the core (τ_2 = 2, 4 and 7 s). The figures show that for the rod worth values under consideration, τ_1 and τ_2, the core outlet sodium temperature level at an early stage of accident development is mainly determined by the beginning of absorber insertion (Fig. 30), and at a later stage by the value of the reactivity worth inserted (Fig. 31).

TABLE 5. HYDRAULIC CHARACTERISTICS OF THE BN-600 STANDARD AND EXPERIMENTAL PSS SUBASSEMBLIES [29]

S/A type	τ_t (s)	τ_2 (s)	Q_{susp}	Q_{nom}	Q_{float}	Q_b	Q^{nom}_{rod}	Q^{min}_{rod}	η
Safety S/A				2.0			1.0	0.25	2.2
PSS 1	10.1	6.1	3.6	6.0	11.5	2.1	1.0	0.36	2.5
PSS 2	8.7	4.7	2.7	4.5	11.5	2.1	1.0	0.25	3.6

Note: τ_t — total time of rod insertion into the core after the beginning of an accident; τ_2 — time of rod insertion into the core from the beginning of its movement (primary flow rate reduction to 0.6 Q_{nom}); Q_{susp} — coolant flow rate through the subassembly by which rod suspension in the upper working position by coolant flow takes place; Q_{nom} — rated coolant flow rate through the subassembly; Q_{float} — coolant flow rate through the subassembly where the rod floats up from the lower position into the upper one; Q_b — coolant flow rate through the subassembly where the rod floats up into the brake; Q^{nom}_{rod} — coolant flow rate through the rod in the upper working position at a rated flow rate through the subassembly; Q^{min}_{rod} — coolant flow rate through the rod in the lower position at a flow rate through the subassembly of 0.25 Q_{nom}; η — margin for the rod to not float up.

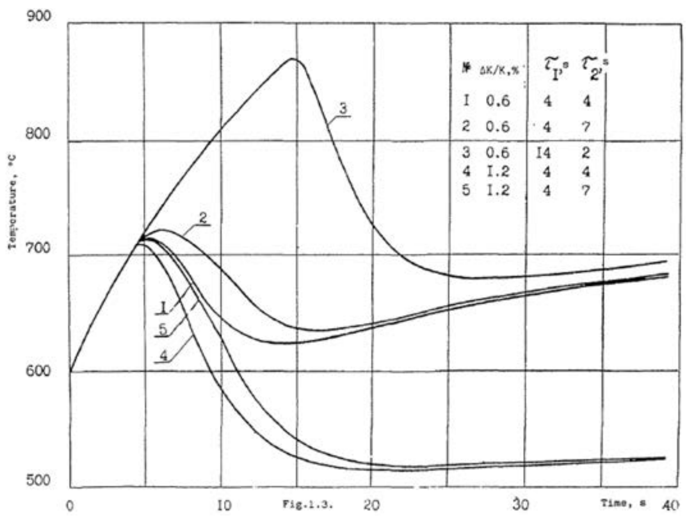

FIG. 30. Sodium temperature at the core outlet as a function of time since the onset of an accident (reproduced from Ref. [29]).

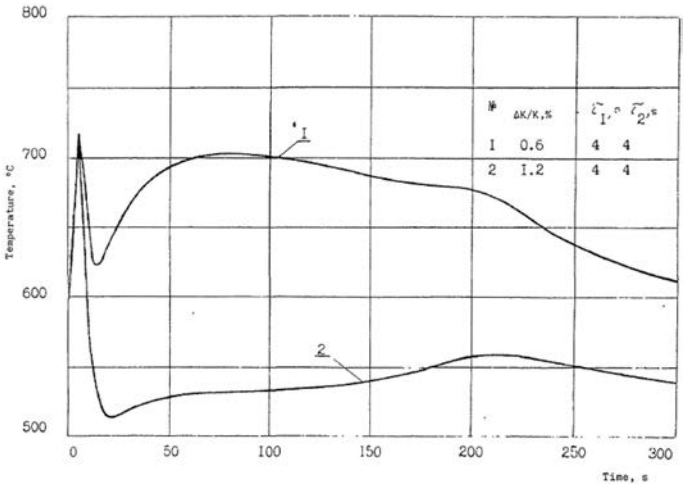

FIG. 31. Sodium temperature at the core outlet as a function of time since the onset of an accident (reproduced from Ref. [29]).

From Fig. 30 one could conclude that for a PSS efficiency value of ~6% $\Delta k/k$ (efficiency of one standard safety rod), when actuation occurs at τ_1 = 4 s (during this time from the beginning of the accident the coolant flow rate decreases to a critical value of 0.6 Q_{nom}) and when absorber insertion into the core happens at τ_2 = 6 s (see Table 5), the core outlet sodium temperature will not exceed 720°C (i.e. considerable margin (200°C) is ensured to the sodium boiling point).

4.2.3. BN-600 operating experience history

A synthesis of the main BN-600 developments and expertise gained is presented in Ref. [26]. It is generally agreed that the selected design options were proved to be correct through BN-600 operation. By the main performance indicators achieved, the BN-600 nuclear power plant is reported by the World Association of Nuclear Operators to be ahead of 50% of nuclear power plants worldwide. However, some components had to be modified in order to improve the operational safety, reliability and efficiency.

The first, most important step was taken between 1986 and 1987 when the BN-600 reactor core was changed (to the second generation core). The linear heat ratings were reduced owing to different fuel enrichments (17, 21 and 26%) and improved radiation resistant structural materials.

During 1991 to 1993 the core was modified once again and a higher burnup was achieved. Moreover, the best radiation resistant and commercialized structural materials were used. This third generation core was qualified as the standard one and improvements are envisaged for the next, fourth generation core.

Much experience was accumulated during operation with regard to minimizing the risk of sodium leaks and sodium–water reactions, and ensuring the efficiency of the sodium fire extinguishing systems if a leak did occur. Thus, the operating experience proved the selected steam generator modular concept to be correct, since 13 sodium–water reactions resulted in loss of only 3% of electrical generation. The evaporator was replaced only once, during the period from 1991 to 1997, instead of three planned replacements during the entire lifetime of the nuclear power plant. The longer lifetime was justified by the results of an extensive evaporator condition examination programme and supported by more stringent water chemistry, fewer transients and emergency situations against the rated value, and periodic reagent cleaning.

The primary sodium pumps are characterized by successful operation. Over the initial period, the shaft couplings experienced damage leading to unplanned loop disconnections. The damage was caused by the shaft resonance frequencies coinciding with the torsional vibration frequencies. After cause remediation, the main results of the work on the pump reliability improvement are reflected in the extension of the lifetime of the main pump components, including the extension of the impeller lifetime up to 50 000 hours.

During the operation at the BN-600 nuclear power plant, a number of important actions were performed in order to improve the equipment reliability and the facility safety, including an extensive R&D programme:

(a) Introduced the sector-type failed fuel detection system;
(b) Mastered the evaporator stage reagent cleaning following the standard procedure involving the use of the feedwater pumps;
(c) Mastered the startup procedure without auxiliary boilers;
(d) Backfitted the steam generator and water/steam drain pipelines;
(e) Upgraded the extinguishing system.

4.2.4. Conclusions

The operating experience from the BN-600 reactor power unit over more than 35 years is positive in terms of demonstrating the feasibility of using an SFR for commercial electricity generation.

The BN-600 reactor is a key link ensuring the continuity of fast reactor development in the Russian Federation; the operational experience confirms the good perspective of this direction of the nuclear power industry [26].

During BN-600 power unit operation, valuable operating experience with the individual systems and components was accumulated and it should be preserved and used in the development of advanced SFR designs.

4.3. BN-800

Construction of the BN-800 and plans for a larger BN-1200 began in 1984, with startup of the BN-800 then planned for 1992. For various reasons, the project was delayed and construction began in 2006 (Beloyarsk-4). The plant reached its full power production in August 2016.

Over the years, the project was improved based on the accumulated expertise on BN-600 operation. The new unit includes a large number of design and technological improvements.

4.3.1. Main features

The BN-800 (Fig. 32) has been designed not only for power generation but also for the demonstration of enhanced safety features which will be used in future SFR projects.

It is envisaged to be refuelled 20 times during the 40 years of operation, with a 730 day fuel cycle length [7]. The reactor core has been designed to support 100% mixed oxide (MOX) fuel, to burn weapons grade plutonium and to produce isotopes.

The most important design changes and enhanced capabilities with respect to BN-600 are the following [7]:

- An additional PSS using flow levitated absorber rods which will drop into the core under their own weight when coolant flow drops under 50% of its nominal value;
- Passive decay heat removal based on air heat exchange connected to the secondary sodium loop;
- A core catcher to avoid interaction between the melted core and the core vessel;
- A sodium plenum instead of the core upper axial blanket to increase the neutron leakage and to compensate for the sodium positive void reactivity;
- Enhanced systems and devices able to prevent sodium leaks and fires (e.g. modular steam generators, leak detection system);
- Increased thermal power (2100 MW(th)) with respect to BN-600 (1470 MW(th)).

4.3.2. Development of passive shutdown systems for BN-800

Over the years, the Russian Federation performed a large number of studies (theoretical and experimental) dedicated to various concepts of PSSs, including: flow levitated absorber rods, magnetic materials (Curie point PSS) or shape memory effect [15].

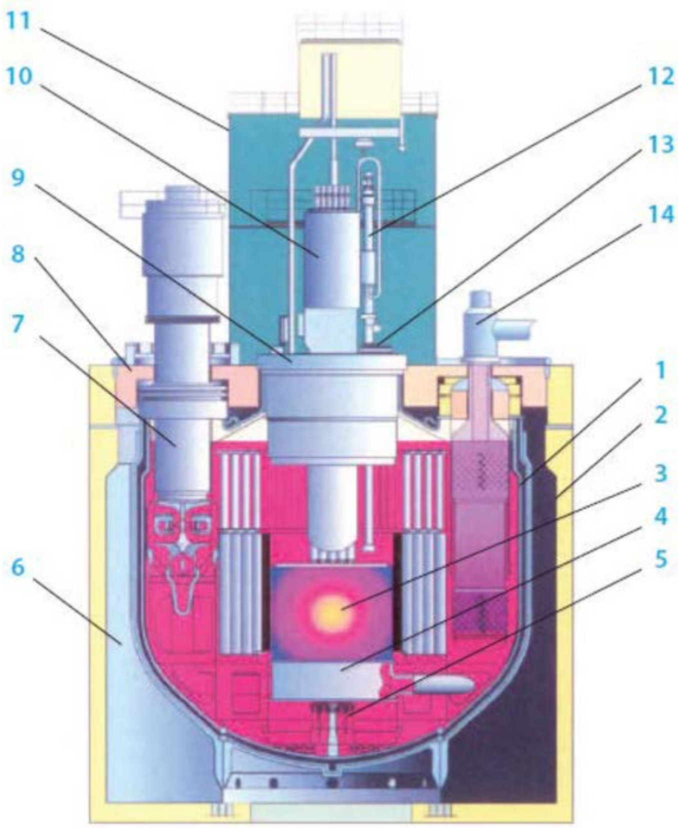

FIG. 32. BN-800 reactor components (courtesy of ROSATOM): (1) reactor pressure vessel; (2) guard tank; (3) reactor core; (4) pressure chamber; (5) corium catcher; (6) reactor cavity; (7) reactor coolant pump; (8) fixed upper shield; (9) large rotating cover; (10) central rotating cover; (11) protected hood; (12) reloading device; (13) small rotating cover; (14) intermediate heat exchanger.

4.3.2.1. Passive shutdown systems based on flow levitated absorber rods

Special attention has been paid to hydraulically suspended absorber rods [15], which are suspended in the coolant flow during the normal operation of the reactor. The rods (Fig. 33) are released into the core under their own weight when the coolant flow rate through the core decreases below a certain value (e.g. 0.6 of the flow nominal value).

Extensive work has been performed in order to assess the characteristics and performance of hydraulically suspended absorber rods; the experiments developed in the water test rig as well as some of the results obtained are presented here in Section 4.2.

4.3.2.2. Passive shutdown systems based on magnetic materials

The PSSs based on CPM have been intensively investigated with regard to their eventual implementation on BN-800.

The concept is based on magnetic materials that lose their magnetic properties at a defined temperature. In the PSS design, these magnetic materials are used in the latch that holds the absorber rods above the core. During accidents induced by a temperature increase, the absorber rod is released and dropped into the core (Fig. 34).

The most important step in the development of a Curie point type PSS was to find appropriate materials which significantly change their magnetic properties in a desired temperature range. Moreover, an important feature of these materials is the weight they could hold.

To ensure the desired rated load, the configuration presented in Fig. 34 has been used [18]; it consists of a permanent magnet made of magnetic alloy with axial magnetization, a surrounding screen made of ferrous–nickel alloy with a Curie point of 620°C, and the armature made of Armco iron which is connected to the absorber rod.

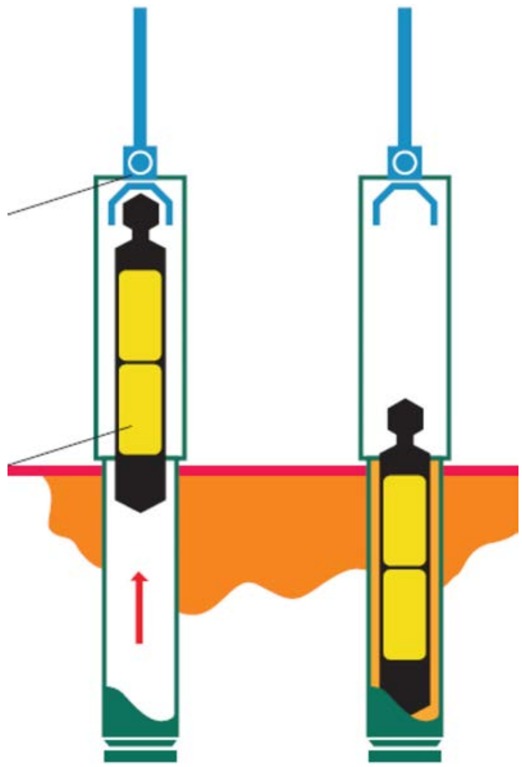

FIG. 33. Passive shutdown rods: pumps in operation (left); pumps out of operation (right) (reproduced from Ref. [15]).

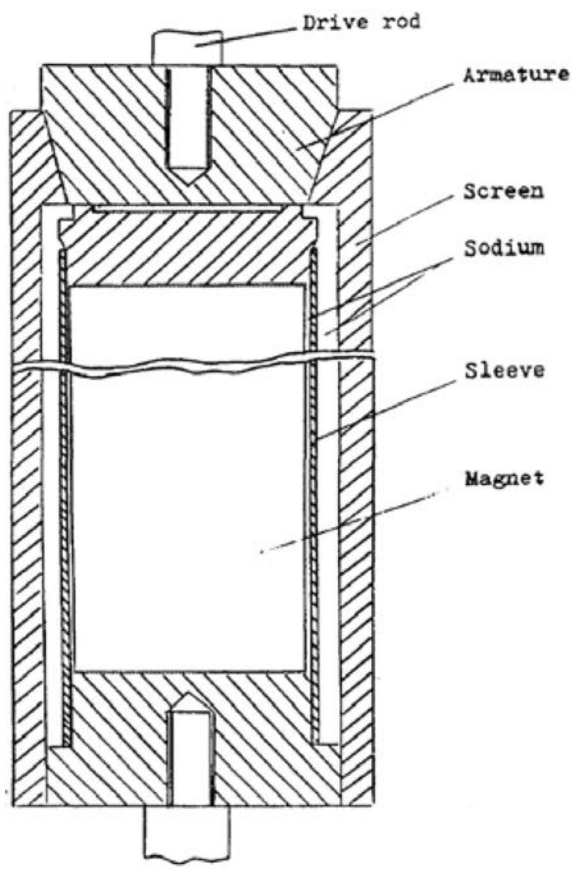

FIG. 34. Mock-up of magnetic actuating device (reproduced from Ref. [18]).

Various experiments [15] have shown that during the beginning of the temperature increase (from 20°C up to 300–400°C), a small decrease of its carrying capacity (~10%) occurs owing to the changes of the magnetic induction as a function of temperature. A further rise in temperature leads to a fast decrease in the material's holding capacity because of deterioration in the material's ferromagnetic properties. Near the Curie point, holding capacity is lost and the rod drops into the core under its own weight.

A magnetic, hard alloy with a Curie point equal to ~850°C has been considered [15] the most appropriate material for the permanent magnet. The investigations performed showed that permanent magnets made of this alloy are efficient during 10 000 hours at 550°C and during 50 hours at 650°C without deterioration of their magnetic properties. As for the magnetic wire, two options have been selected: the soft, magnetic iron–nickel–cobalt alloy Fe–67Ni–Co with a Curie point equal to ~635–640°C and the iron–nickel alloy Fe–65Ni with a Curie point of ~620°C. The analysis of the carrying (holding) capacity of various alloys as a function of temperature is presented in Fig. 35.

The main conclusion drawn [15, 18] referring to the Curie point based PSS is that these devices have the advantage of versatility and can be used to prevent any type of accident.

Some of their drawbacks have been identified in Ref. [18]: the use of the same principle as the standard shutdown rods with regard to their insertion into the core, the possibility that the device may fail to demagnetize owing to either insufficient temperature increase in the temperature sensitive material or increase in its actuation

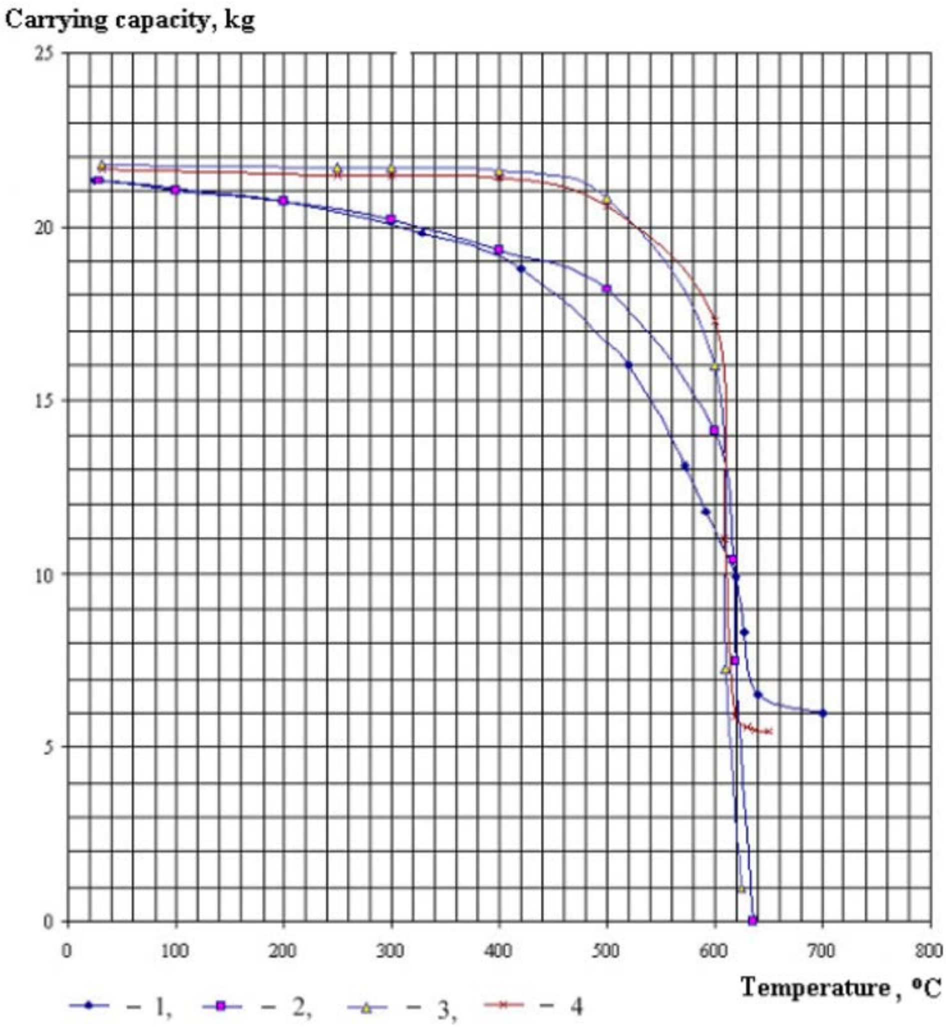

FIG. 35. Carrying capacity as a function of temperature for various samples (reproduced from Ref. [15]): (1) screen material — Fe–65Ni, armature material — St.3; (2) screen material — Fe–65Ni, armature material — Fe–65Ni; (3) screen material — Fe–67Ni–Co, armature material — Fe–67Ni–Co; (4) screen material — Fe–67Ni–Co, armature material — St.3.

temperature or the possibility that the latch may fail to release despite demagnetization of the material because of various issues, including the adhesion of the magnet and the rod.

4.3.2.3. Passive shutdown systems based on shape memory alloys

The functioning principle of this type of PSS is that the absorber rod is held above the core during the normal operation of the reactor with the help of a device based on shape memory alloys, and when the temperature of the latch reaches a certain value the latch changes form and releases the absorber rod into the core. As mentioned in Ref. [18], it has been developed as a method for shaping the material for various rod designs.

Hundreds of alloys with shape memory have been fabricated and studied all over the world. The large diversity of the investigated alloys offered the opportunity to make the appropriate selection with regard to the most suitable one to be used in BN type reactors. Table 6 presents some of the main alloys with shape memory at high temperatures [15].

In the perspective of implementing this type of PSS in the BN type reactor design and development, alloys based on titanium (Ti–Ta 30% and Ti–Ni–Ni–Rh) having a hyperthermal shape memory have been produced and tested.

An extensive computational and experimental investigation was performed in support of these PSSs [15] envisaging the following aspects: development of computational techniques able to evaluate the thermomechanical characteristics of the working elements of different geometries (laminated, cylindrical springs, Belville's springs), investigation of the corrosion stability of alloys (e.g. titanium in liquid metals) or the effect of neutron irradiation on these alloys.

Figure 36 presents one of the most promising memory shape PSSs for BN type reactors based on Belville's spring.

TABLE 6. THE MAIN ALLOYS HAVING SHAPE MEMORY AT HIGH TEMPERATURES

No.	Composition	Temperature (°C)
1	Ti–Ni	<130
2	Cu–Al–Mn	150–600
3	Cu–Al–Mg	200–300
4	Co–Al–Ge	360–400
5	Co–Al–Si	525
6	Fe–Ni	525
7	Ni–Mn–Ti	−200–700
8	Ti–Nb	260
9	Ti–Nb–V	50–500
10	Ti–Au	290–630
11	Ti–Ta	230–670
12	Ti–Ni–Ti–Me, Where Me: Pd, Pt, Rh	100–1000

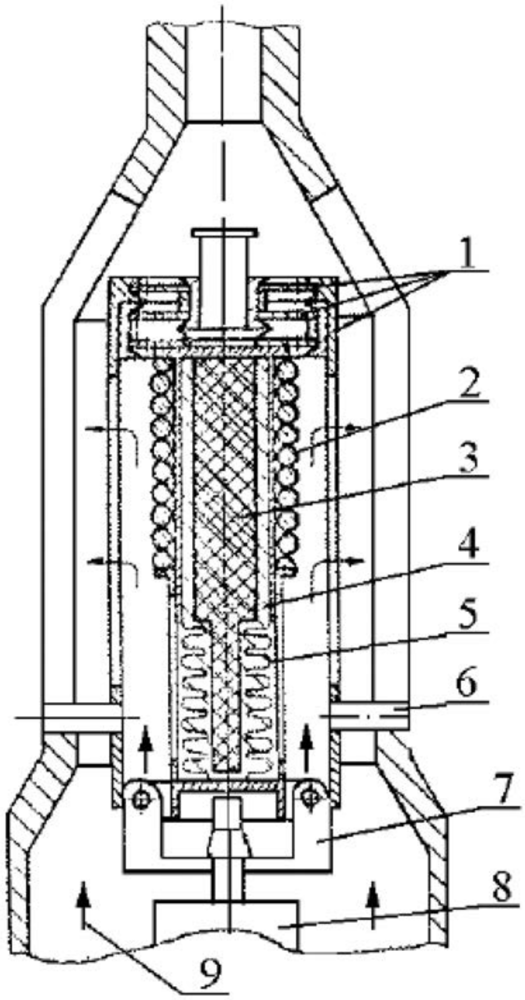

FIG. 36. Passive shutdown system operating on the basis of magnetic memory shape and on other physical effects (reproduced from Ref. [15]): (1) springs with magnetic memory shape; (2) high temperature spring; (3) capillary porous material; (4) temperature sensitive substance; (5) sylphon; (6) support; (7) trigger mechanism; (8) absorber of neutrons; (9) coolant.

Also, Ti–Ta and Ti–Ta–Hf alloys with a shape recovery temperature of 650°C (of interest in BN type reactors) [18] have been selected and studied (including corrosion and mechanical tests).

The investigations lead to the conclusion [15, 18] that the PSSs with magnetic memory shape can be used in different types of accidents, and their main drawbacks are the principle of rod insertion which is the same as for the standard shutdown system and the possible changes in the material properties due to irradiation.

4.3.3. Main tasks to be solved by construction and operation

Following are the main tasks that need to be solved by construction and operation [12]:

— Further improvement of the safety aspects.
— Demonstration of the technology’s competitiveness.
— Optimization of the closed fuel cycle and non-proliferation strategies.
— Development of innovative technologies for future liquid metal fast breeder reactors:
  - Advanced fuel and structural materials testing and qualification;
  - Minor actinide burnout technology demonstration;
  - Testing of novel technical solutions;
  - Sustaining competency in the liquid metal fast breeder reactor technology.
— Ensuring the local demand for energy.

4.4. INHERENTLY EFFECTIVE SHUTDOWN SYSTEM WITH CURIE POINT CONTROLLED SENSOR/SWITCH UNIT

By definition [30], an inherently effective shutdown system ensures reactor shutdown also in cases when all reactor protective systems fail and severe mechanical damage has occurred in the reactor system.

Moreover, an inherently safe shutdown system meets a number of criteria, the most significant of which are the following:

— The system operates in a fail-safe mode.
— The system is diverse with respect to the other shutdown systems.
— The response and shutdown intervals are sufficiently short to be able to manage specified incidents.
— All functions of the system are amenable to testing at any moment.
— The components are suited for long service lives (at least equal to the lifetimes of the fuel elements).
— They are protected against human error.

Within the framework of activities devoted to the safety of fast breeder reactors, a concept for an inherently effective shutdown system was worked out at the Karlsruhe Nuclear Research Center [31]. The key element of this system is a Curie point controlled sensor/switch unit. In case of an inadmissible rise in the sodium temperature at the fuel element outlet, the unit will automatically unlatch the mechanism holding the control rods in position, dropping the absorber rods into the reactor. It is possible to arrange the corresponding switch units directly above the heads of selected fuel elements, thus achieving a very short response time.

4.4.1. Function and layout of the overall system

The proposed system consists of two parts, the absorber rod unit and a Curie point controlled sensor/switch unit (Fig. 37). The absorber rod unit can consist of several rod bundles interconnected through joints. It is so

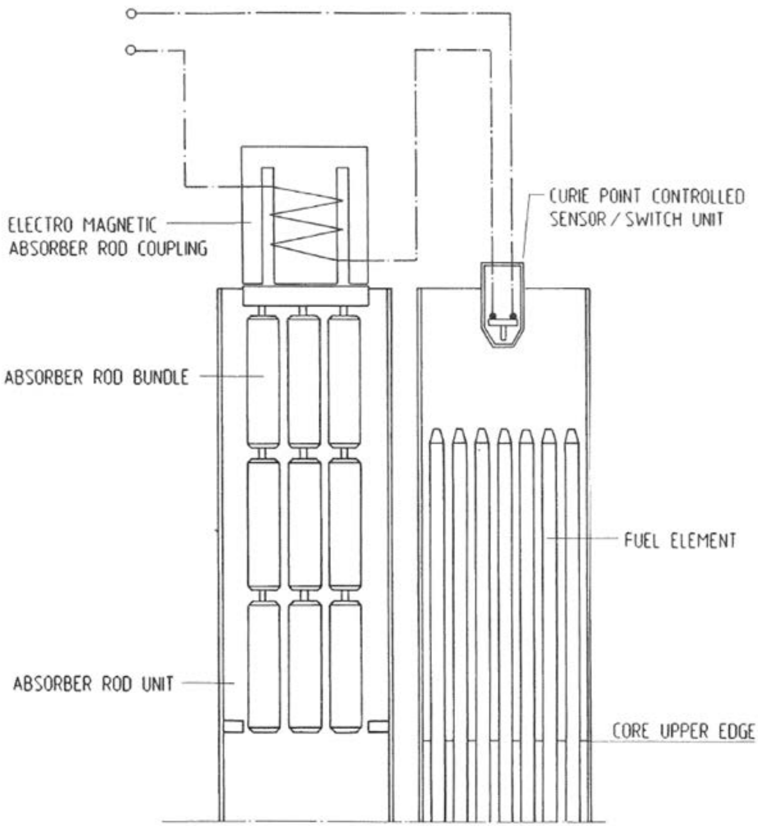

FIG. 37. Temperature sensitive inherent reactor shutdown system [30].

flexible that even in case of maximum deformation of the duct, it glides into the core zone. The absorber unit is connected with the suspension system by means of an electromagnetic coupling. The joint face of the coupling in the withdrawn position lies at the upper edge level of the absorber element duct in the primary sodium. Thus, the function of the coupling is not even affected when the absorber rod suspension system gets damaged.

The sensor/switch unit is to be placed above the core in such a manner that it is directly exposed to outflowing sodium from a selected fuel element. Subsequent to an increase in the sodium temperature at the core outlet, reactor shutdown is initiated after interruption of the current supply to the electromagnetic absorber rod coupling by means of the self-acting sensor/switch unit. Unlike concepts where the temperature sensitive element is a component of the coupling, this concept offers the advantage that a much smaller mass of material has to be heated for the switching operation to be activated. This ensures the fastest possible response. Moreover, the electromagnetic coupling can be made of a material with a higher Curie point. Thus, it constitutes — independent of the prevailing operating condition — an almost constant holding force.

4.4.2. Curie point controlled sensor/switch unit

Figure 38 shows the Curie point controlled sensor/switch unit. Permanent magnets together with the magnetically soft iron yoke, the sensor element and switching weight make up a closed magnetic circuit. The switching weight is electrically insulated from the housing; in the extracted condition, it bridges the electric contacts through which the holding current of the absorber rod coupling flows. Being part of the switch housing, the sensor element is directly surrounded by the flow of reactor sodium and is provided with external fins to improve heat transfer. The material used for the sensor element is a binary nickel alloy whose Curie point is ~600°C. When the

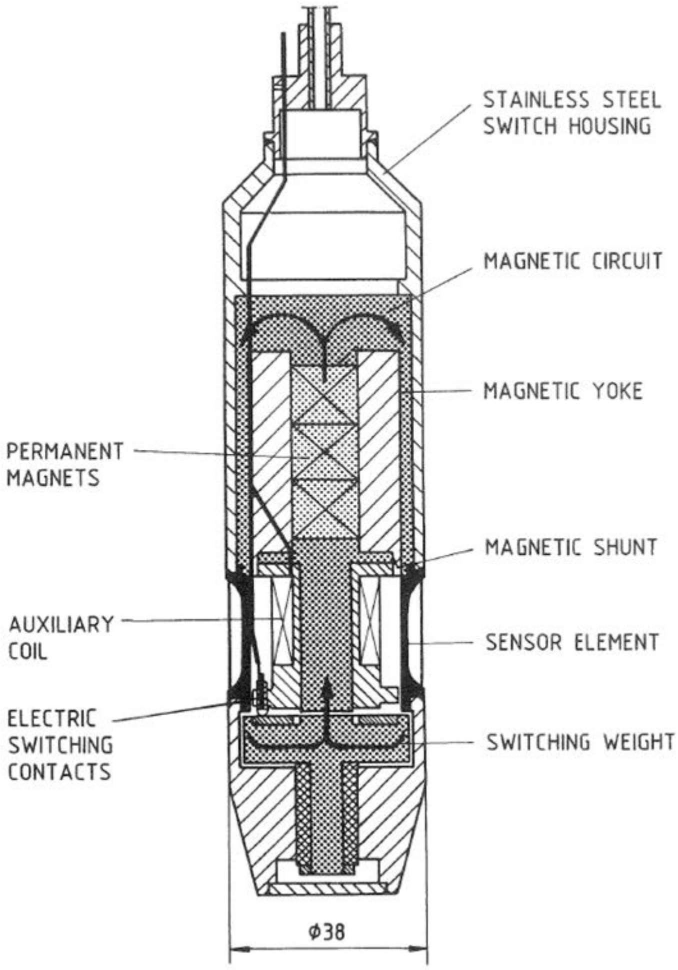

FIG. 38. Curie point controlled sensor/switch unit [30].

sodium reaches this temperature limit, the magnetic flux passing through the switching weight is interrupted. Most of the magnetic flux then flows through an appropriately sized shunt. Consequently, the switching weight drops and interrupts the current supply of the magnetic coupling which causes the absorber bundle to drop into the core with very little time delay.

An auxiliary, additional coil causes a secondary magnetic circuit to become established, if necessary. Its direction can be set either equidirectional with or reverse to the permanent magnetic flux by reversal of the coil polarity. In case it is equidirectional with the permanent magnetic flux, this flux is intensified to the extent that the dropped switching weight is lifted again. On the other hand, by reversing the coil polarity, the switching weight can be dropped and hence the function of the shutdown device gets tested.

The issue of coil malfunction during steady state operation can be precluded by a simple safety circuit (Fig. 39). The load current of the coil is supplied from a capacitor preceded by a resistor. Therefore, the current in the coil passes in both directions by short current impulses only or, in case of malfunction, by a negligible low steady state current.

The switch unit is 38 mm in outside diameter, 208 mm in total length and filled with helium to ensure thermal conductivity. The electric lines connecting it with the switching contacts and the auxiliary coil are routed so as to leave the switch on top.

In designing the magnetic circuit of the Curie point controlled sensor/switch [32] for use at elevated temperatures and in the radiation field of the reactor, the choice of the magnetic circuit materials was of particular importance. These materials must have stable structure, keep their dimensional stability under irradiation and, as to the sensor, be chemically inert with respect to sodium. Characteristic magnetic properties of the magnetic circuit components are the current flux density and the permeability. The Curie point is of particular interest.

Preparatory work for the experiments started early and served as a basis and for the validation of the magnetic circuit design of the Curie point controlled sensor/switch unit. A test bench was built in which a first test switch was operated under a cover gas at temperatures ranging from 550 to 600°C. The switch was passed by a coil current of a simplified absorber rod coupling.

The laboratory experiments served to optimize the magnetic circuit gaps in terms of the impact of forces on the switching weight in steady state operation and at the moment of shutdown. This was accompanied by improvements in the design of the mobile switch components and switch contacts. Also, various material combinations suited for bridging the absorber rod holding current were tested. Molybdenum was found to be the most suited contact material.

Collected operating experience during these experiments finally led to a fully functional switch concept. On that basis a prototype switch was operated successfully for 440 days under cover gas at ~560°C and heated at

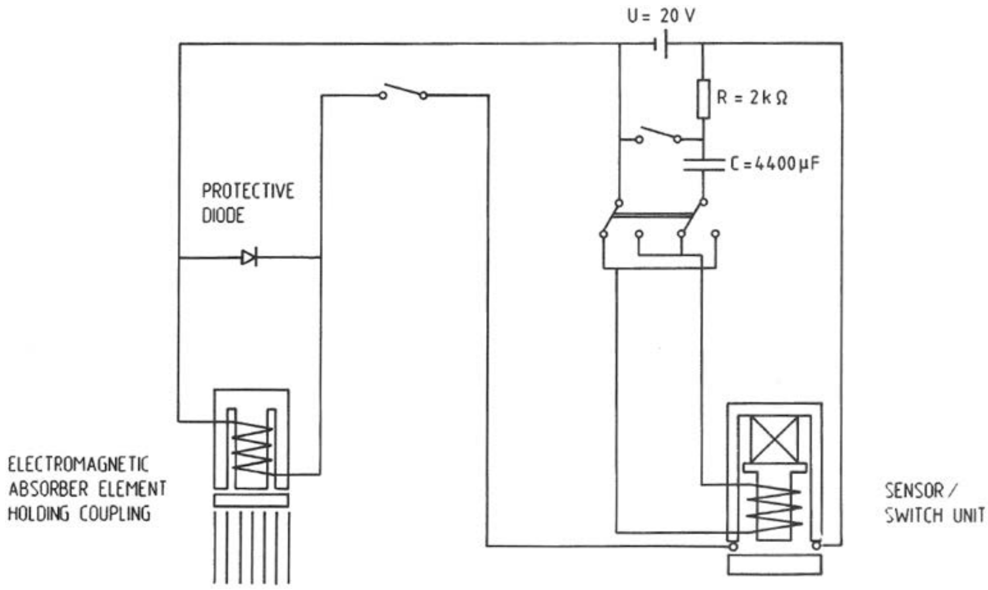

FIG. 39. Electric circuit of the shutdown system [30].

irregular intervals until the switching point was activated. Experience accumulated during this long term experiment resulted in the detailed design of a reference concept for the switch unit whose main data are shown in Table 7. This concept served as a basis for the irradiation of several switches in an in-pile experiment at the KNK-II reactor in Karlsruhe.

TABLE 7. PRINCIPAL DATA OF THE CURIE POINT CONTROLLED SENSOR/SWITCH UNIT [30]

Parameter	Value
Dimensions	
Length	208 mm
Diameter	38 mm
Materials	
Permanent magnet	Al–Ni–Co 450 Al 7 wt% Cu 4 wt% Ni 15 wt% Ti 5 wt% Co 32 wt% Fe 37 wt%
Yoke components	Armco iron
Switching contacts	Mo
Magnet support and bobbins	1.4541
Sensor	Ni–Fe 6535
External parts	1.4571
Auxiliary coil and leads to the switching contacts	
Jacket	1.4306
Wire	Cu/Zr stainless steel coated
Insulation	Mg–O
Cover gas in the switch	He
Forces	
Weight force of the switching weight	0.7 N
Magnetic force acting on the switching weight	
Normal operation	1.4 N
Moment of switching	0.3 N
Temperatures	
Operating temperature	560°C
Shutdown temperature (Curie temperature of the sensor material)	600°C

The response time of the Curie point controlled sensor/switch unit is of particular importance. The response time is the interval between reaching of the inadmissible sodium temperature limit at the core outlet and the time when the absorber rod bundle actually drops.

This time interval is determined in two steps. First, the delay in switching upon interruption of the holding current of the coupling was determined in an experiment for a potential absorber rod unit. Second, the transient temperature development for the switch unit was calculated for a defined accident scenario.

The result of computation is visible in Fig. 40. The upper curve represents the coolant temperature at the fuel element outlet and the lower curve the temperature development at the inner wall of the sensor. Accordingly, the temperature at the sensor wall attains the value of the Curie point after 5.2 s. The difference with respect to the coolant temperature rise yields a 1.5 s delay in switching. The 0.45 s switching delay of the absorber rod holding the coupling has to be added to this value. Thus, the investigations have shown that response times of less than 2 s can be achieved. This is a very favourable value for an inherent shutdown system.

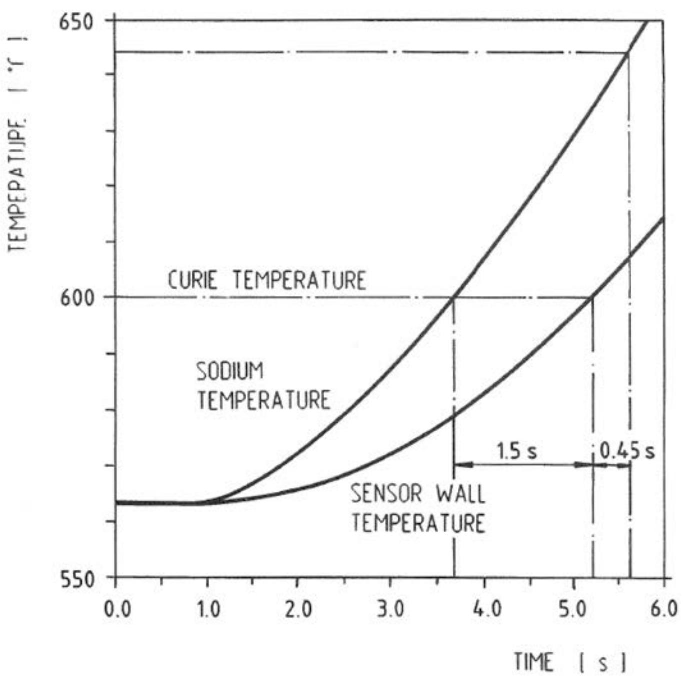

FIG. 40. Sensor wall temperature in the SNR-2 reactor during a loss of flow accident [30].

4.4.3. In-pile testing in KNK-II

As an investigation of individual material properties in the hot cells would have been too expensive, and moreover not sufficiently effective, an in-pile experiment at KNK-II was designed and performed as an integral function test. It was devised to demonstrate that the additional influence exerted by the radiation field has no adverse effect on the operational behaviour of the switch unit.

When specifying the irradiation conditions, it was assumed that the switch units in the SNR-2 reactor are placed near the upper edges of the fuel elements, directly above the heads of selected fuel elements so that interference could be recorded as quickly as possible. In that zone, the fast neutron flux ($E > 0.1$ MeV) is 3×10^{13} n (cm^2·s). The desired service life was 1200 full power days.

Eight test switches based on the reference concept were manufactured at the Karlsruhe Nuclear Research Center (KfK) central workshop in the context of the in-pile experiment. The performance of each switch unit was examined during assembly of the units. All switch units responded perfectly in their switching operation. The bandwidth of the cut-off temperature of each switch was <1 K.

The in-pile experiment was performed during the period of extended KNK-II/2 operation. The switches remained fully operative until the end of testing. During the time of irradiation, their switching temperature remained almost constant. The scatter band of the switching points ranged from 2.5 K for the central switch to 4 K for the upper switch.

4.4.4. Conclusions

Development work concerned the Curie point controlled sensor/switch unit. To be able to evaluate the sensor/switch unit and the proposed system as a whole, the following criteria must be fulfilled:

— Provisions for fail-safe behaviour;
— Protections against errors in manipulation;
— Possibilities for in-plant testing;
— Specifications for long term operation.

The materials used should allow the switch unit to reach a service life of several years. These requirements have been successfully tested and confirmed on laboratory scale tests performed at elevated temperatures and in a long term irradiation, in-pile experiment under actual reactor operating conditions. In these integral tests, the switches remained fully operational until the end of the testing. Their switching temperatures remained nearly constant during the entire irradiation period.

5. INNOVATIVE CONCEPTS

Depending upon the functional requirements and details of the reactor design, different methods for passively inserting negative reactivity under off-normal conditions have been considered or are being developed. The two principal parameters that change in the core during a transient are temperature and flow rate. Thus, most passive methods for inserting negative reactivity are triggered by changes in one of these two variables. Concepts that rely on changes in flow are described first, while those triggered by changes in temperature are described later in the section.

5.1. FLOW ACTUATED PASSIVE SHUTDOWN SYSTEMS

This section describes PSSs that are triggered by changes in coolant flow rate. There are two primary concepts: those based on coolant levitation of absorber material, and those based on increasing neutron leakage at the core boundary when coolant pressure decreases (proportional to flow). Although concepts like the GEM that increase neutron leakage with reductions in coolant pressure have been shown to be effective in terminating ULOF transients for oxide cores, this concept is limited to smaller core sizes as these devices act to increase leakage at the core boundary. On this basis, several different concepts were developed in the 1970s as part of the US fast reactor programme for application to large SFR cores. These efforts were carried out predominately by industry: levitated absorber balls (Atomics International [33]) and levitated absorber assemblies (Combustion Engineering [34]). Although the designs for these concepts were refined to various degrees, they were never tested in-core due to termination of the US fast reactor programme. The summary here focuses on levitated absorber assembly design concepts as those were the most thoroughly developed concepts of the two.

5.1.1. Flow levitated absorbers

The basic design requirements of flow levitated absorbers established for this work [34] were that the hardware must fit into a secondary control assembly, and that insertion could be initiated independently by the plant protection system or by self-actuation. In this manner, the core design is not penalized by requiring additional non-fuelled assemblies.

The first approach [34] is based on an orifice concept for levitating absorber material. The poison assembly is held out of the core during normal operation by the pressure drop across a support seal at its upper end. The concept is illustrated in Fig. 41. Upon loss of flow, when the hydraulic pressure becomes insufficient to hold up the poison assembly, the assembly drops by gravity into the core, thereby achieving reactor shutdown.

At plant startup, the assembly is removed from the core by the control rod drive mechanism. As coolant flow increases during startup, the pressure drop across the orifice eventually exceeds a design value that is sufficient to hold up the weight of the assembly and hydraulically lock it into position, suspended above the core. The control rod drive is then delatched from the assembly so that it is free to fall under gravity if the flow rate falls below the minimum required to hold up the assembly, thereby inserting negative reactivity. Note that the hydraulic lock can also be broken by insertion of the control rod drive so that normal plant shutdown function is maintained with this design.

Depending upon design specifics, this basic orifice concept can have an inherent coast down delay before actuation that may not be acceptable in some plant designs. In recognition of this possible limitation, a variable orifice concept was also developed [34] to maintain a constant pressure drop across the support seal at any operating flow rate, so that the coast down delay can be avoided. The pressure difference across the seal is controlled by axial movement of the control rod drive. This concept is illustrated in Fig. 42. As shown, the control rod drive guide tube has several holes on the exterior surface near the gripper that can be aligned with corresponding holes through the duct wall, thereby providing orifice steps that can be adjusted based on the position of the control rod drive. The minimum number of orifice steps is determined by the need to maintain both an acceptable release time and support pressure margin at any flow rate within the operating range. Although this approach is attractive, it would require in-core calibration to determine hydraulic hold up points as a function of control rod drive position.

A second design variant was also developed to address concerns regarding the coast down delay inherent in the basic orifice design concept. This system operates on the same basic principle as an aviation rate of climb indicator by responding to local changes in pressure. The concept is shown in Fig. 43. During normal operation, the bellows position and the residual fluid volume in the cylinder gradually adjust by flow through the orifice. This acts to equilibrate the accumulator bellows pressure with that of the coolant inside and outside the cylinder. However, during the rapid pressure change that is indicative of pump trip, the stored energy in the sodium bellows

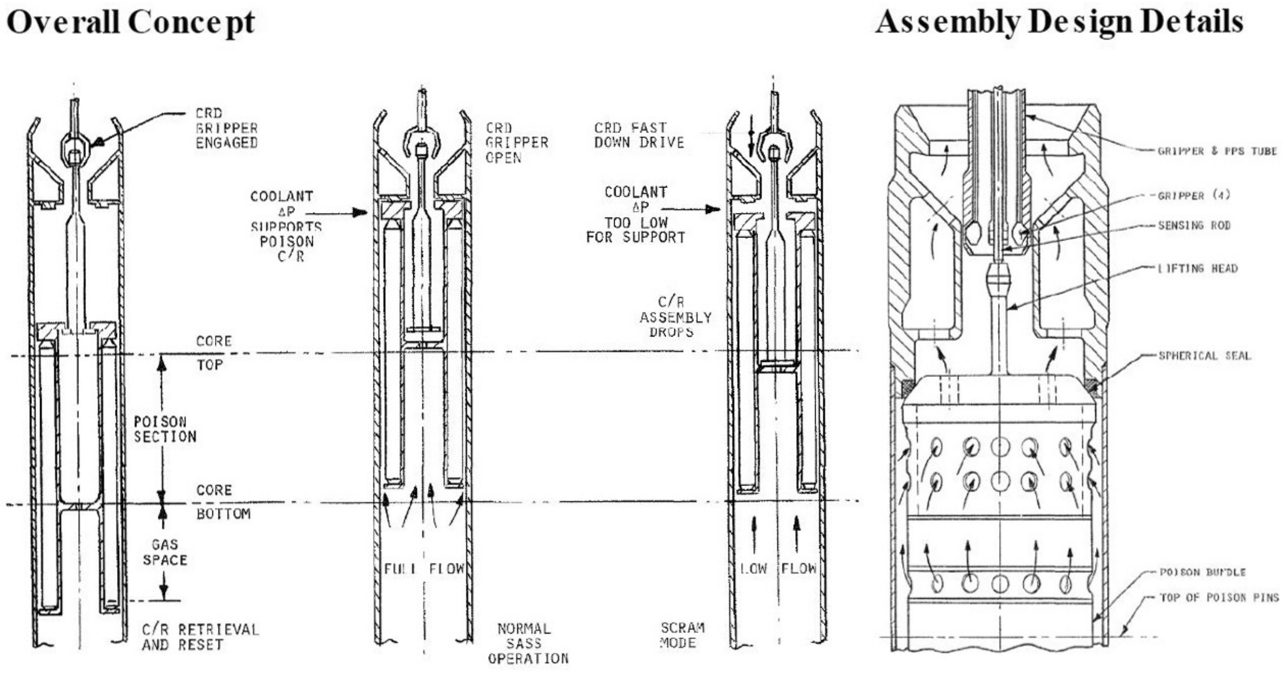

FIG. 41. Basic orifice concept for a levitated absorber assembly design [34].

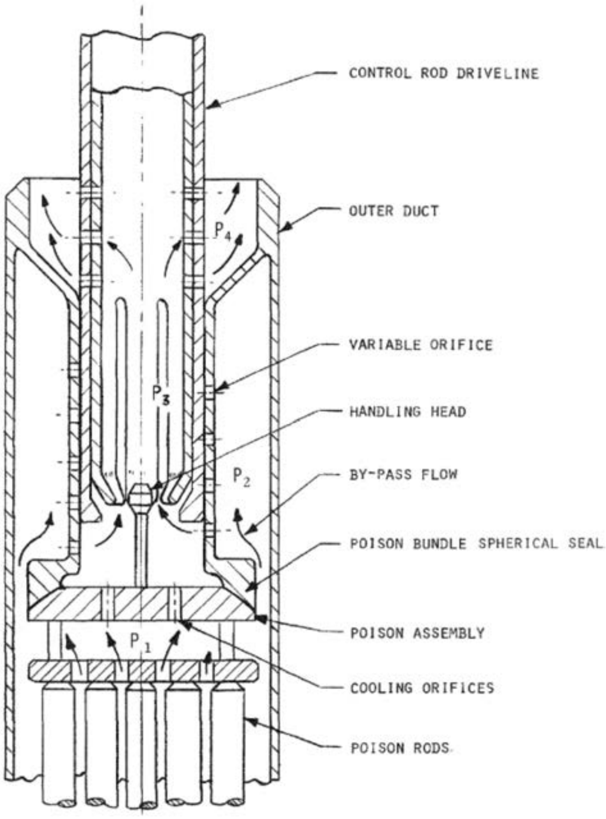

FIG. 42. Variable orifice concept allowing operation at various power levels [34].

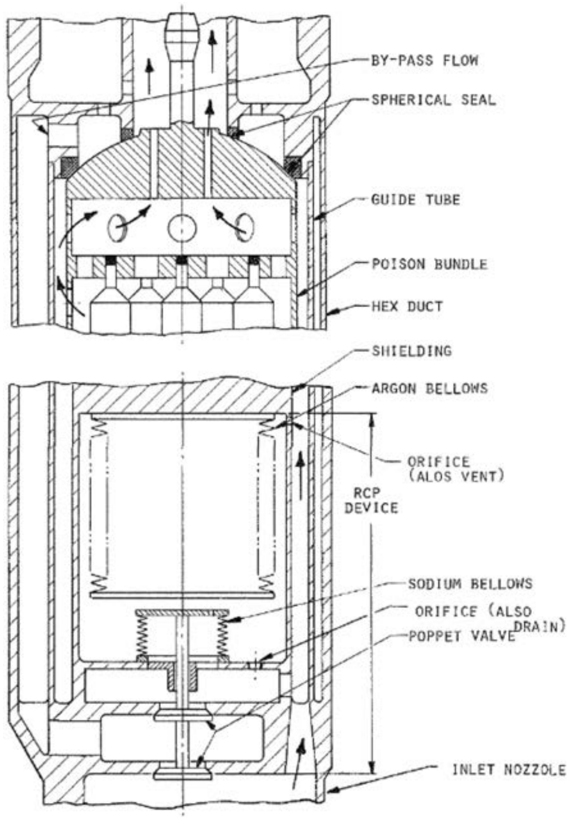

FIG. 43. Rate of change pressure switch trip actuator concept [34].

acts to open a valve, breaking the seal and allowing the assembly to drop. Since this rate of change pressure device is insensitive to normal (slow) variations of coolant flow, pressure and temperature, spurious scrams are avoided.

As noted earlier, although designs for these levitated absorber concepts were refined to various degrees, they never underwent in-core testing owing to termination of the US fast reactor programme.

5.1.2. Gas expansion modules

The main focus of reactivity feedback improvement connected to the coolant loss effect has been centred on leakage based methods. The loss of coolant is an important safety problem for LMFRs, especially those with sodium as a coolant. In GFRs this effect is not so significant; this is one of the major advantages of the gas cooling. The loss of coolant reactivity effect is extremely space dependent. Loss of coolant from the centre of the core yields a highly positive reactivity effect, but coolant loss from near the boundary area of the fuel gives a negative effect. This behaviour is driven by two main phenomena which contribute to the overall coolant void effect: spectral hardening and increased leakage. Both effects are large and of opposite sign. The spectral hardening effect is positive. The loss of coolant from the core results in decreased moderation of the neutrons, so the average neutron energy increases. This produces a positive reactivity effect because of the increase in neutron importance with increasing energy. At the same time, the loss of coolant results in an increased mean free path for the neutrons, which in turn yields a negative contribution to coolant loss reactivity. The strong spatial dependence of the coolant loss effect is due primarily to the different spatial behaviour of the leakage component relative to that of the spectral hardening component. The effect of sodium loss near the centre of the core is highly positive, since the large spectral hardening effect is not counterbalanced by a large leakage contribution. The same loss of coolant near the edge of the core increases the leakage significantly and adds much to the negative leakage contribution to the loss of coolant effect. For sodium based systems, the coolant loss effect is slightly affected by the elimination of sodium capture and the change in self-shielding [24].

One of the first systems specifically designed to reduce reactivity through leakage in accident scenarios is the GEM system developed at the FFTF in the 1980s. A GEM is essentially a passive device for inherent shutdown to insert negative reactivity during a primary system ULOF. The device is basically a hollow removable subassembly sealed at the top and open at the bottom. The gas trapped inside the subassembly expands when core inlet pressure decreases due to flow reduction and expels liquid coolant from the subassembly. Neutron leakage increases and negative reactivity is inserted, as shown in Fig. 44. The integrity of the envelope has to be assured in order to avoid ingress of the gas into the core and the consequent positive reactivity insertion [36].

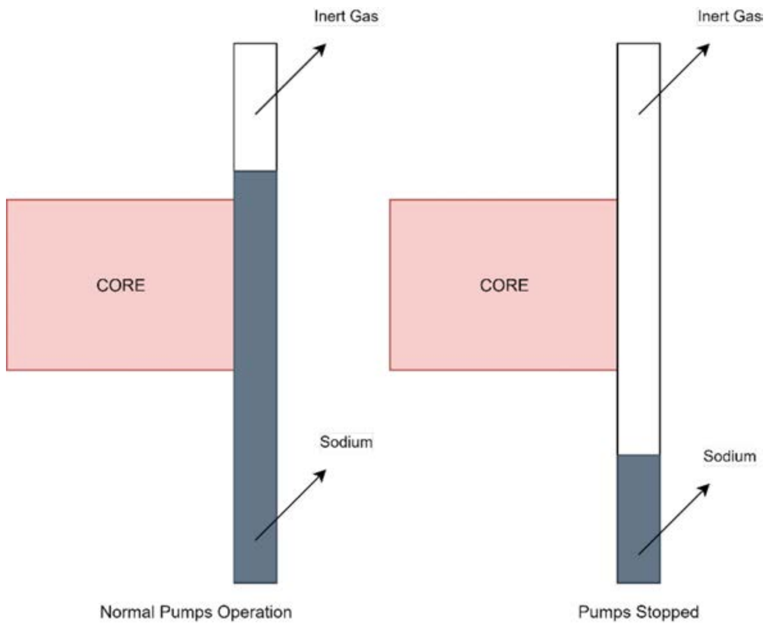

FIG. 44. Design concept of a gas expansion module [35].

As an alternative to a GEM system compensating only ULOF consequences, the GEM actuated by a temperature increase, with modifications usable also for GFR, can be utilized. In this system, a boron carbide rod is submerged into a liquid metal. The level of the liquid metal, and thus the boron carbide rod, is controlled by the liquid metal vapour pressure in the system. In ATWS, temperatures will increase, resulting in increased vapour pressure, decreasing the level of the liquid metal and thus inserting the neutron absorbing boron carbide rod into or near the active core region. Although this system is self-actuating and even self-resetting, it has an operational similarity to the control rod based systems [35].

In GFR, the voiding effect of non-fuelled zones surrounding the active core has little impact on the overall coolant loss effect. Reflection of the neutrons is provided mainly by the solid structural materials in reflector assemblies. For these kinds of system, a solution can be considered in which parts of the reflector are moved out or replaced by an absorber. Two alternative solutions were recently proposed for GFRs: a flow levitated absorber [37] combined with a high void content radial reflector assembly, and a movable reflector assembly. During normal operation, the movable reflector assemblies are suspended in their upper positions and the reflector material is in the core. During the emergency situation, the assemblies are dropped whole, right to the lower edge of the fuel part. With the reflector in its original position, empty space exists where an absorber or special neutron trap can be placed to increase leakage efficiency. The principle of the neutron trap is that it slows down the neutrons in the inner cavity and then traps them with higher efficiency in the outer region of the assembly. A schematic of the construction of the neutron trap is shown in Fig. 45. The movable reflector assembly's advantage is in its functionality independent of the actuation system. If the system is initiated by high temperature or loss of power, all motion of the assembly is actuated by gravity alone.

However, GEM or other systems positioned radially around the core are not sufficient in large cores. A common approach to improve positive coolant loss reactivity in fast reactors has been to design cores to have a large axial neutron leakage probability by restricting the height of the active core region to the vicinity of 100 cm. By sufficiently reducing the core height, it is even possible to achieve lower void worth. Most of the modern, large, fast reactor designs feature a large plenum of coolant above the active core region. Coolant voiding is likely to initiate in the upper part of the fuelled region where the coolant temperature is highest. This boiling will quickly

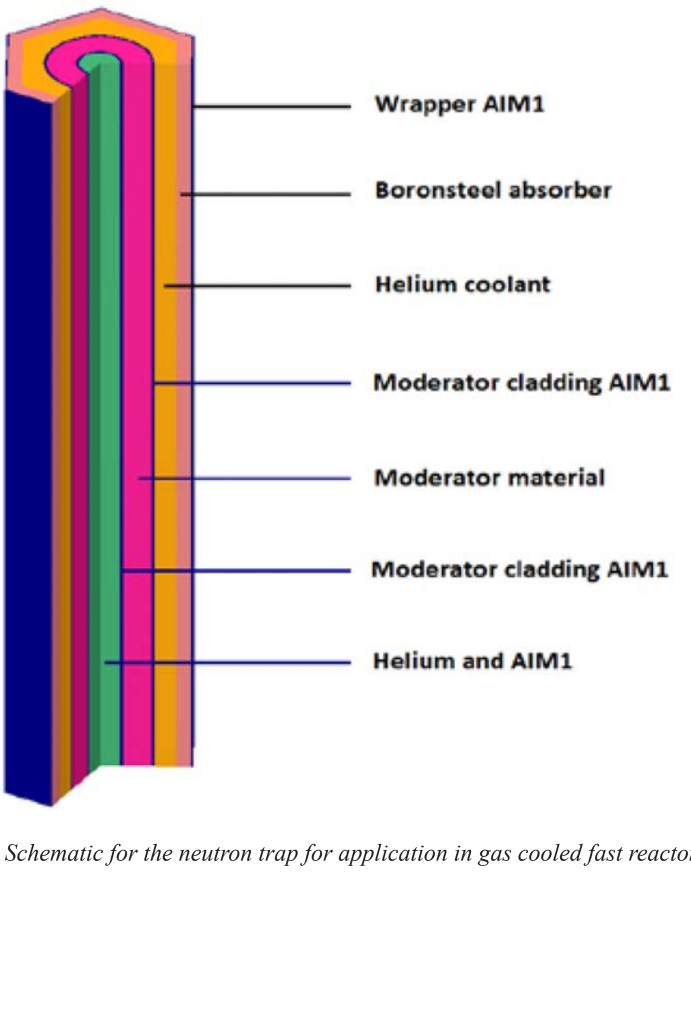

FIG. 45. Schematic for the neutron trap for application in gas cooled fast reactors [38].

spread to the above-core plenum region. Voiding of coolant outside of the active core region reduces neutron reflection back into the core and provides a way of reducing void worth with low impact to the neutron economy of the core in standard operation. The addition of an absorption layer over the above-core coolant plenum further increases the void worth reduction efficiency due to a decrease in back scatter from upper reflectors in voided conditions [39].

5.2. TEMPERATURE ACTUATED PASSIVE SHUTDOWN SYSTEMS

5.2.1. Lithium expansion modules

The LEM system is advantageous since it does not have moving mechanical parts and relies solely on physical phenomena (thermal expansion). The LEM does, however, rely on the stability of a gas–liquid interface, where the heavier liquid is suspended above the gas. Calculations, balancing the buoyancy force with surface tension, show that a stable interface can be upheld within a tube with a diameter such as the inner cladding diameter of a standard fuel pin. These calculations have also been validated experimentally [40].

The principle of LEMs has been verified by neutron radiography conducted at JRR-3M reactor at the Japan Atomic Energy Research Institute. The gas–liquid interface goes up and down in accordance with temperature as shown in Fig. 46.

An innovative LEM system has been envisioned for the RAPID fast reactor concept [10] in Japan. RAPID-L is equipped with four quick LEMs and 24 slow LEMs. A quick LEM is characterized by a quick response. It can provide only a negative reactivity insertion. Three (out of four) quick LEMs ensure a reactivity of −50¢. Accordingly, it is effective to mitigate the anticipated transient without scram.

Conversely, slow LEMs can provide both negative and positive reactivity insertion with moderate thermal response [11]. Reactivity varies between 22.7 and 13.0$ with 24 slow LEMs. Slow LEMs have the role of automated burnup compensation. In addition, slow LEMs also realize partial load operation in accordance with the primary flow rate. The gas–liquid interface in nominal operation is placed in the active core region as shown in Fig. 2. In case the core outlet temperature decreases, the gas–liquid interface rises, and positive reactivity is added, and vice versa. To avoid quick positive reactivity addition, slow LEMs have a reservoir of double envelopes for vacuum insulation. Therefore, slow LEMs are affected only by moderate thermal transients resulting from burnup reactivity swing and primary flow rate control.The design parameters of LEMs are described in Table 8.

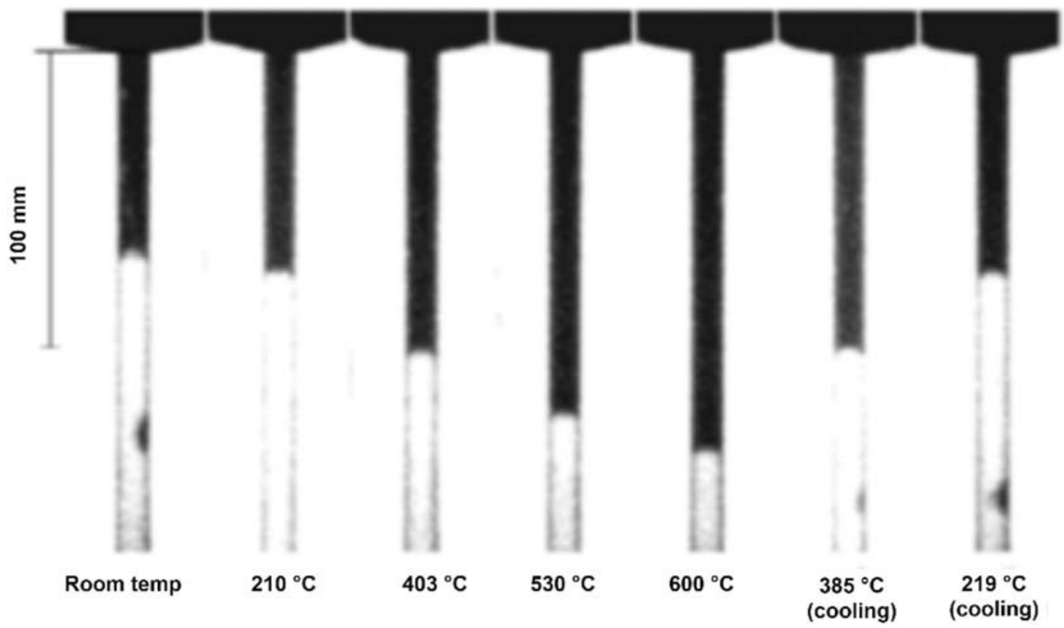

FIG. 46. Neutron radiograph of lithium expansion module [10].

TABLE 8. DESIGN PARAMETERS OF QUICK AND SLOW LITHIUM EXPANSION MODULES [11]

	Quick LEM	Slow LEM
Envelope		
Inner diameter (mm)	20	20
Full stroke (mm)	640	640
Material	Mo–Re	Mo–Re
Reservoir		
Inner diameter (mm)	140	140
Length (mm)	2000	2000
Total LEM sensitivity (¢/K)	2.3	16.7
Single LEM sensitivity (¢/K)	0.77	0.77

5.2.1.1. Experimental verification of the concept

The purpose of the experiment was to confirm the lithium holding ability of the LEM envelope. Test specimens of various envelope inner diameters and materials (6, 8, 10 and 12 mm diameters made of type 316L stainless steel, and 8 mm diameter made of Mo–41Re alloy) without a reservoir were manufactured by IKE Stuttgart. The inner surface roughness of the 316L and Mo–41Re envelope is <0.7 and 0.1 μm, respectively. Natural lithium of special quality supplied by Chemetall GmbH was enclosed in the upper part of the envelope. An inert gas mixture of 90% argon and 10% helium (0.1 MPa at room temperature) was enclosed in the lower part. These specimens were vertically installed in a vacuum furnace with trace heaters attached and heated up to 900°C with a gradient of 10°C/min. After holding for one hour, the specimen was cooled.

A neutron radiograph of the specimens after cooling to room temperature showed that lithium remained in the upper part of the envelope in all the specimens.

The stroke of the gas–liquid interface in this experiment was only a few millimetres because no reservoir was equipped. This test revealed that an envelope with an inside diameter of 10 mm is feasible with a sufficient margin.

5.2.1.2. Installation of the lithium expansion module in the reactor

Figure 47 illustrates how the reactor can be controlled by a LEM. An elevation of the gas–liquid interface is indicated with a quasi-steady-state equilibrium temperature. At the nominal core outlet temperature, the gas–liquid interface is set at the top of the active core region in order to avoid positive reactivity addition with decreasing core outlet temperature.

A LEM can be placed in any type of fast reactor. For the 60 MW(e), metal fuelled RAPID fast breeder reactor, LEMs are expected to be placed in the upper plenum of the integrated fuel assembly as shown in Fig. 48. The upper plenum is isolated from the hot plenum by the fuel cartridge; therefore, primary sodium passed through the fuel pin bundle flows around the reservoir without any mixing with the hot plenum sodium. A LEM can also accommodate conventional fast reactors and achieves the same safety objectives. In this case, a LEM can be placed in guide tubes among fuel subassemblies as shown in Fig. 49. Each guide tube comprises three LEMs. The role of the guide tube is to hold the three LEMs and to provide a better thermohydraulic situation for the LEM reservoir. The primary sodium from the fuel subassemblies enters into the guide tube and comes up without any mixing with the hot plenum sodium.

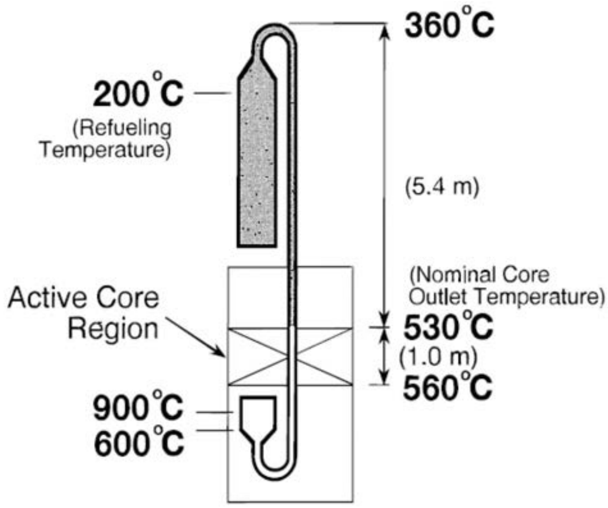

FIG. 47. Elevation of the gas–liquid interface in the liquid expansion module envelope.

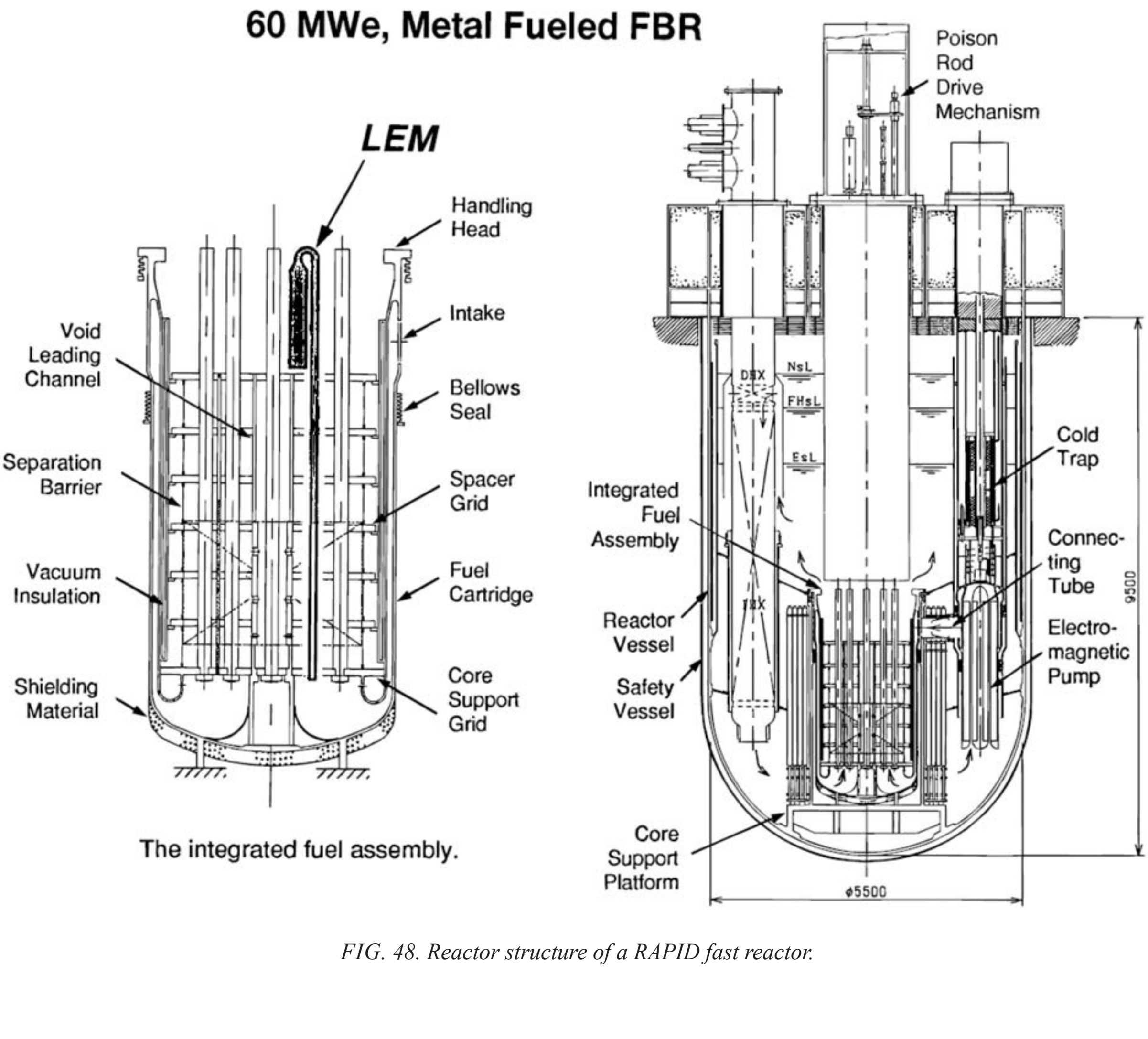

FIG. 48. Reactor structure of a RAPID fast reactor.

Just prior to the refuelling operation, the LEM guide tube should be hung up in the upper internal structure. In the case of a 1000 MW(e), MOX fuelled reactor, 12 guide tubes (36 LEMs) can provide 50¢ of reactivity worth. In any case, a LEM can be installed in the reactor with hardly any influence on the core performance or reactor design.

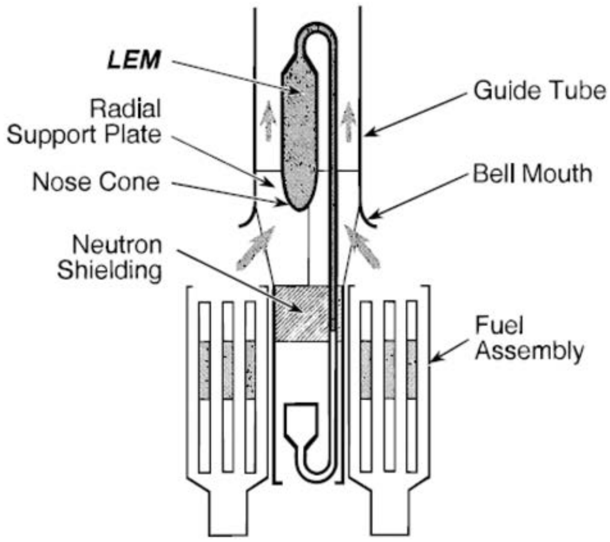

FIG. 49. Lithium expansion module installed in a guide tube to accommodate a conventional core.

5.2.2. Enhanced control rod driveline expansion

Two types of device for enhancing driveline thermal expansion are represented in Table 9. One is a prototype of a hydraulic device called ATHENa (an abbreviation of the German words for 'shutdown by thermal expansion of Na') manufactured and tested for implementation in the EFR. For ATHENa, a sodium filled container was chosen as a hydraulic expansion module, as shown in Fig. 50. It provides for both rod drop and forced insertion of jammed control rod assemblies (CRAs) [5, 41].

TABLE 9. DESIGN FEATURES OF ENHANCED THERMAL EXPANSION DEVICE

Enhanced thermal expansion type		Gripper type	Driveline expansion stroke (mm)	Function
Hydraulic expansion module with liquid sodium filling (ATHENa, CSD of EFR)		Mechanical and electromagnetic	~200	Unlatching CRA gripper Forced insertion of CRA jamming
Bimetallic device	Long thermal expansion device length	Mechanical and electromagnetic	~10	Unlatching CRA mechanical and electromagnetic gripper
	Short thermal expansion length	Electromagnetic	~3	Unlatching CRA electromagnetic gripper only

Note: CRA — control rod assembly; CSD — control and shutdown rods.

Several bimetallic devices were suggested in several SFR designs as represented in Fig. 51. This device serves in a relatively small thermal expansion distance compared to the hydraulic device because the distance is generated by the thermal expansion difference of the two structural metals as the coolant temperature rises. So it is used only to directly unlatch the CRA gripper holding it. In early PRISM design, an enhanced thermal expansion device with multilayer bimetallic cylinders was used to unlatch the CRA hold by a finger type gripper. The later EFR has also adopted a bimetallic device to unlatch the CRA's hold by an electromagnetic force as shown in Fig. 51 [42]. This type has been adopted in the Prototype Generation IV Sodium Cooled Fast Reactor (PGSFR) design concept developed in the Republic of Korea. Design data for enhanced thermal expansion devise for SFR reactors are presented in Table 10.

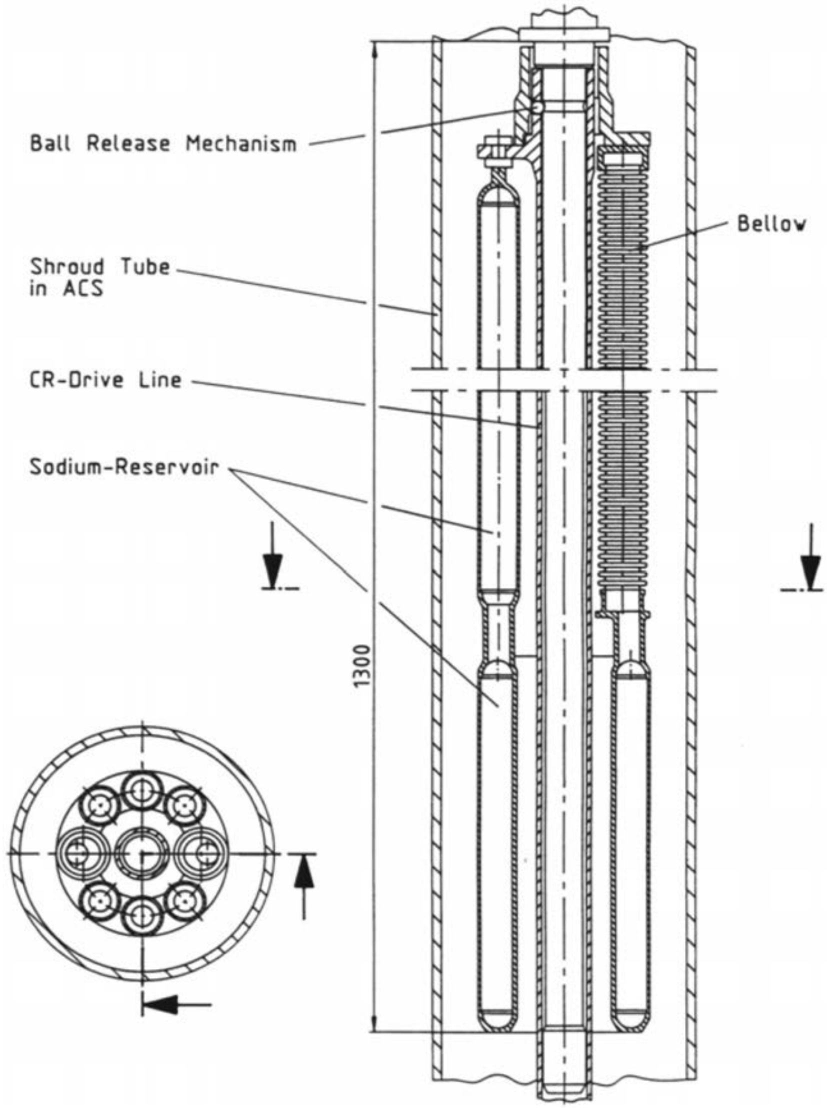

FIG. 50. The ATHENa technical design (reproduced from Ref. [41]).

5.2.2.1. Hydraulic thermal expansion device

A hydraulic expansion concept was tested in ATHENa [5, 41]. The hydraulic expansion device consists of two coaxially arranged drivelines. The primary part corresponds to the conventional CRDL. The secondary one is an enhanced thermal expansion part, which has an axially expandable container filled with sodium. The two parts are linked together by a special release mechanism. The ball release mechanism needs less than 5 mm expansion distance to separate the two parts. The sodium has a fairly large thermal expansion coefficient, which is more than four times as large as the value of the stainless steel container.

Consequently, with increasing temperature, the sodium blows up the expandable bellows part of the container. The expanded length of the bellows moves up the movable end of it. The elongation of the container is dependent on the ratio of its volume and the cross-section of its expandable part. Reducing the bellows cross-section decreases the force by which the absorber can be pushed into the core. The technical design of the ATHENa device shown in Fig. 50 was done in such a way. Only about 1.3 m at the lower end of the normal CRDL of the EFR CRDLs can be used for ATHENa. The diameter is limited to 190 mm.

The sodium container is composed of an annular cylinder and a tube bundle made of austenitic steel. The total volume is 14.6 L. Two metal bellows with a total cross-section of about 35 cm^2 provide for the axial expansion by the expanded sodium volume. The resulting thermal expansion coefficient is about 1 mm/K (i.e. ten times the value of the conventional CRDL). The system can withstand an internal pressure of 2.7 MPa at a maximum

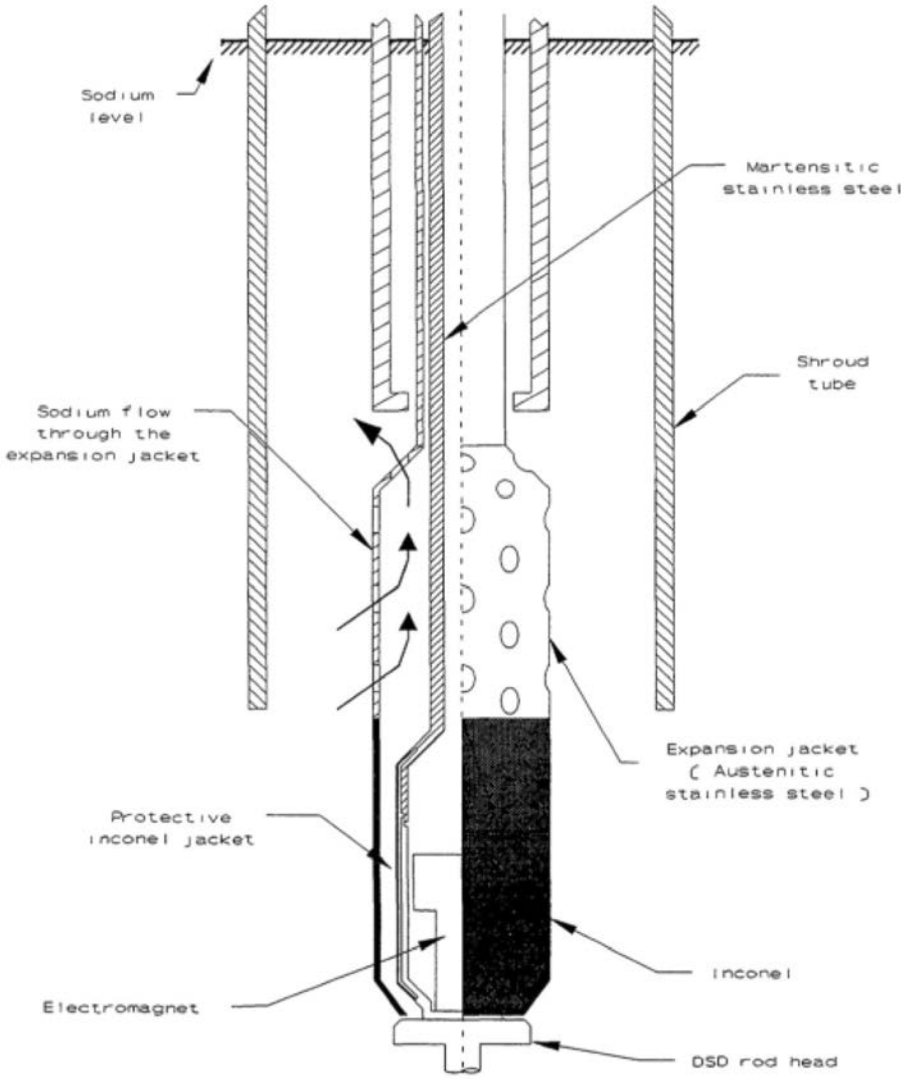

FIG. 51. The European fast reactor third shutdown concept (reproduced from Ref. [42]).

TABLE 10. DESIGN DATA FOR ENHANCED THERMAL EXPANSION DEVICE IN SFR REACTORS

Design data		PRISM (one of kind, USA)	EFR (France, DSD)	EFR (France, CSD)	PGSFR (Republic of Korea)
Operation temperature (°C)		538 (1000°F)	600	600	650
Thermal expansion device	Length (m)	6	6~7	1.3	~3.0
	Materials	316SS/Mo–Re alloy	316SS/martensitic steel	316SS/sodium	316SS/Inconel 718
Gripper opening actuation length (mm)		12.7 (collet type)	7.5 (electromagnet)	5 (bayonet)	1~2 (electromagnet)
Forced insertion distance of jammed CRA (mm)				200	

Note: CRA — control rod assembly; CSD — control and shutdown rods; DSD — diverse shutdown rods.

temperature of 800°C. The resulting force for pushing the control rod into the core is roughly 10 kN. It decreases with increasing expansion (20 N/mm) due to the spring effect of the bellows.

Maximum bellows expansion is 200 mm. This is sufficient to bring the reactor into a safe subcritical state. The ball release mechanism releases the secondary part when temperatures increase by 5 K. At nominal coolant temperature, the container is not completely filled with sodium and the internal pressure is equal to the sodium vapour pressure. Because of the higher external pressure (0.16 MPa), the metal bellows are 28 mm precompressed. This keeps the ball release mechanism closed and makes ATHENa fail-safe. Any leak in the sodium reservoir or metal bellows would increase the internal pressure, causing an extension of the bellows. The resulting expansion would open the ball release mechanism and drop the corresponding control rod.

The reengagement of the ball release mechanism is rather easily accomplished by simply pushing together the two parts of the primary driveline. When the release mechanism is open, the primary driveline can expand like a telescope: the upper shaft slides within the lower one, keeping the balls in the open position. In the prototype, this stroke is limited to 350 mm: 150 mm for the upper end of the expansion module plus 200 mm maximum bellow expansion. After returning to predelatching temperature and/or pressure, the metal bellows always try to contract and thus close the release mechanism again. This happens automatically when the telescopic primary driveline is shortened to its original length using the control rod drive mechanism.

The contracting force of the bellows resulting from the inside–outside pressure difference is 560 N, which is not enough to lift the 800 N absorber weight. The absorber weight is the maximum load on the six balls in the release mechanism. Thus, each has to withstand a force on the order of 150 N, which poses no problem for the materials. For the experimental investigations, the ATHENa prototype was equipped with sensors that measure the sodium level and temperature profiles.

Even through the implementation in the EFR design was not adopted, this device could expect to prevent hypothetical core disruptive accidents during unprotected transients in fast reactors.

5.2.2.2. Bimetallic thermal expansion device

As an example of a passive shutdown device using bimetallic thermal expansion, the thermal expansion device of the PGSFR shown in Fig. 52 is described here.

The PSS is implemented in the secondary control rod drive mechanism of the PGSFR. The system consists of a thermal expansion device, an electromagnet to hold and trigger the CRA, and a flow guide structure. The coil structure encloses the electromagnet coil and stator, and protects the components from the core exit sodium. The flow guide directs the core exit sodium to the thermal expansion device to make sufficient contact with the sodium. An electromagnet armature is attached to the CRA extension rod top. The extension rod plays the role of the armature of the electromagnet system.

The device length is determined to be about 2 to 3 m, which will be updated based on the coolant thermal transient analysis and the reactor operation basis. The device size is limited by a control assembly duct with an inner space 100 mm in diameter. The outer diameter of the thermal expansion device is determined to be 90 mm, taking into account a clearance of 5 mm. The inner diameter at the lower part is 86 mm. The length at the lower part is about 1.18 m. The inner and outer diameters of its upper region are 64 mm and 70 mm, respectively. The region extends up to the lower end of the bushing, and the whole length of the thermal expansion device is 2.86 m.

The thermal expansion differential coefficient of the thermal expansion device is represented in Table 11. The total expansion difference between SS316 and Inconel 718 is expected to be between 1.7 and 2.6 mm when the environment fluid temperature rises to between 100°C and 150°C more than the normal operation temperature. The calculation results are as follows:

$$1.72\ \text{mm} \approx (6.0 \times 10^{-6}/°\text{C}) \times (100°\text{C}) \times (2.86\ \text{m})$$
$$2.57\ \text{mm} \approx (6.0 \times 10^{-6}/°\text{C}) \times (150°\text{C}) \times (2.86\ \text{m})$$

The size of the electromagnet is to be 80 mm in outer diameter, and about 300 mm in length, as shown in Fig. 52. The electromagnet outer core encloses the coil and protects it from sodium ingress. The inside diameter of the outer core is 66 mm. The outer diameter of the inner electromagnet core is 49 mm.

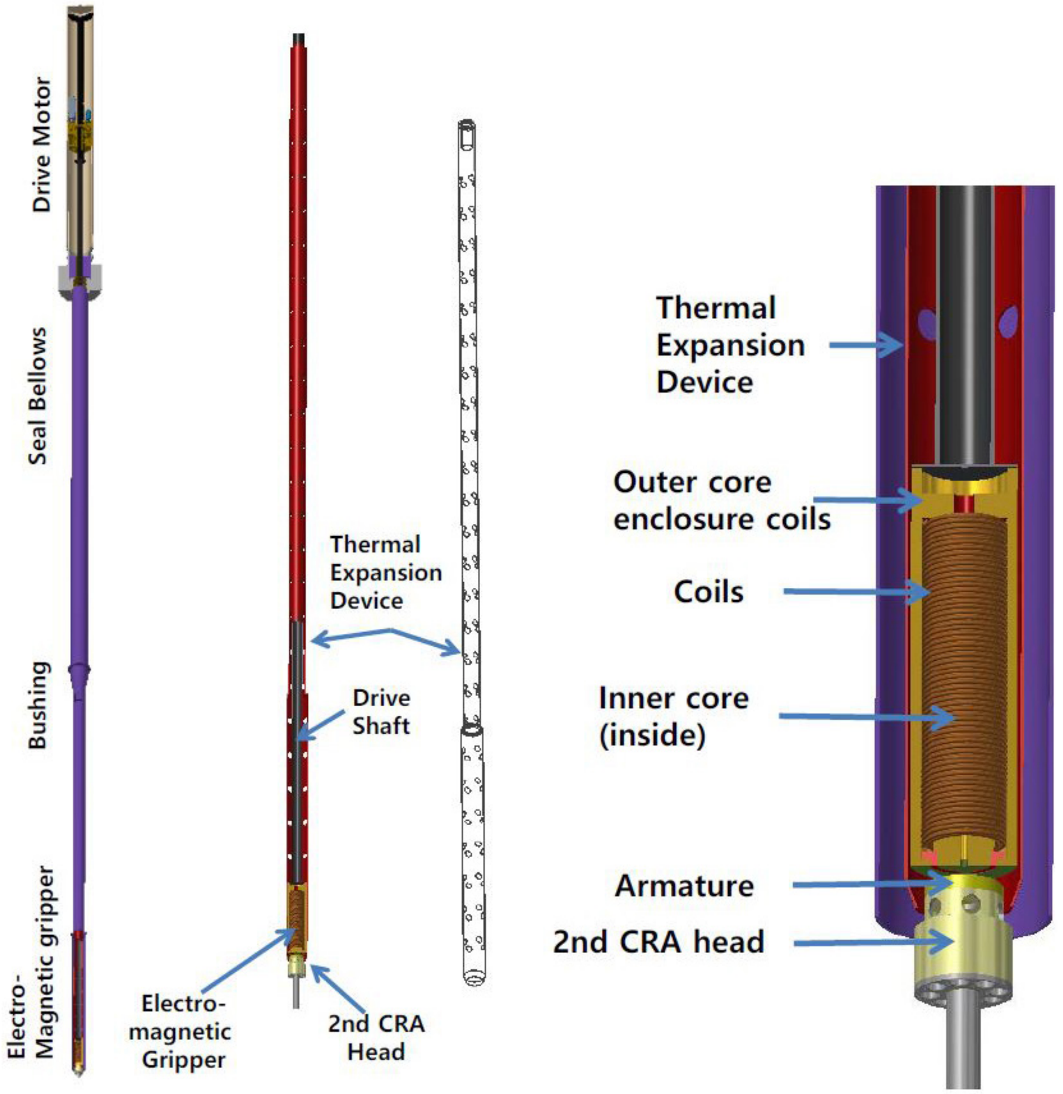

FIG. 52. Passive shutdown concept using a bimetallic thermal expansion device with an electromagnet (courtesy of J. Lee, KAERI).

TABLE 11. DESIGN FEATURES OF ENHANCED THERMAL EXPANSION DEVICE OF THE PGSFR

Parameters	Requirements
Installation space	<100 mm in diameter
Gripper off mechanism	Bimetal thermal expansion difference: 2–3 mm (operating temperature: 600–650°C)
Secondary CRA mass	~50 kg
Gripper type	Electromagnet with solenoid coil
Drive shaft material corresponding to the thermal expansion region	9Cr-1Mo-V (or Inconel 718)
Thermal expansion device material	SS316
Average linear thermal expansion difference coefficient	~6.0×10^{-6} mm/(mm × °C)

Note: CRA — control rod assembly.

A DC power of ~15 V and 7.5 A is selected to be supplied on the coil. A total of 264 turn coils are allowed inside the electromagnet core, and the cross-section of each coil is 1.54 mm^2. The ampere turns are calculated to be 1980 AT.

The electromagnetic forces on the CRA within a 1 mm gap are in the range of 300 to 2100 N. Thus, the thermal expansion difference of the thermal expansion device to trigger the CRA should be controlled within 1 mm at the set temperature. Design feasibility tests using mockups of the thermal expansion device as a passive concept of the PGSFR are being performed.

5.2.3. Autonomous reactivity control

Properly designed, the ARC system can act as a thermostat in the core, autonomously controlling temperature without the need for any operator action, electrical systems or moving mechanical parts. This actuation responds to temperature and relies solely on the laws of physics, and is therefore an inherent feedback mechanism. By having a separate liquid (or gas) push the neutron absorbing liquid into the core in the event of a deviation from nominal operating conditions, rather than having the absorber liquid expand itself into the core, there is no dependence on a gas–liquid interface, the absorber inventory is radically reduced and the actuation speed of the system increases. The recommended material selections for the ARC system are potassium as the expansion liquid (with potential alternatives of indium, caesium and rubidium) and lithium as the absorber liquid. The components and assembly steps for an ARC system installation in a typical fast reactor core have been worked out in full detail in Ref. [43].

In the reference design, the lower reservoir is created by extending the coolant inlet tube part of the nose piece 10–15 cm into the assembly. The volume created by this minor design alteration is sufficient to house the liquids for the lower reservoir in a typical fast reactor. A view of the foot (nose piece) of a fast reactor fuel assembly with this type of design is shown in Fig. 53. The upper ARC reservoir, which provides the main driving force for system actuation, is located above the rod bundle and is shaped as an annulus to maximize the rate of heat transfer and thereby minimize the time delay (Fig. 54).

At full flow conditions in a typical SFR, the volume averaged temperature of the liquid in the upper ARC reservoir will lag the coolant outlet temperature by slightly more than one second. This means the system actuates fast enough to counteract the effects of any transient that raises the coolant temperature considerably. The complete installation of an ARC system in a fast reactor core increases the axial length of the assembly by ~30 cm, and raises the pressure drop by ~0.5%.

The steps for assembling an ARC assembly of this type are:

(1) Mounting rail pins are inserted in to their slots in the nose piece and stabilization rods are inserted in their slots through both the pin mounting rails and the nose piece.
(2) The open gas reservoir is welded onto the outer ARC tube.
(3) The lower duct section is welded onto the nose piece.

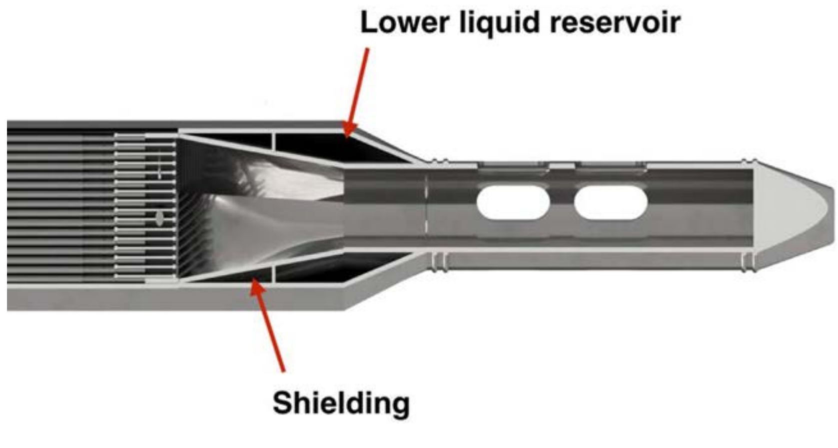

FIG. 53. Foot of a fast reactor fuel assembly with a lower autonomous reactivity control reservoir (courtesy of S. Qvist, University of California, Berkeley).

(4) The liquid reservoir is positioned on top of the gas reservoir and slid into its positioning slots in the duct. The inner ARC tube, which is prewelded to the liquid reservoir, is inserted into the outer ARC tube through the gas reservoir. The liquid reservoir piece is then welded (outside and inside) onto the gas reservoir.
(5) The uppermost duct and the handling head are welded to the lower duct.

The first four steps and the corresponding components are shown in Fig. 55.

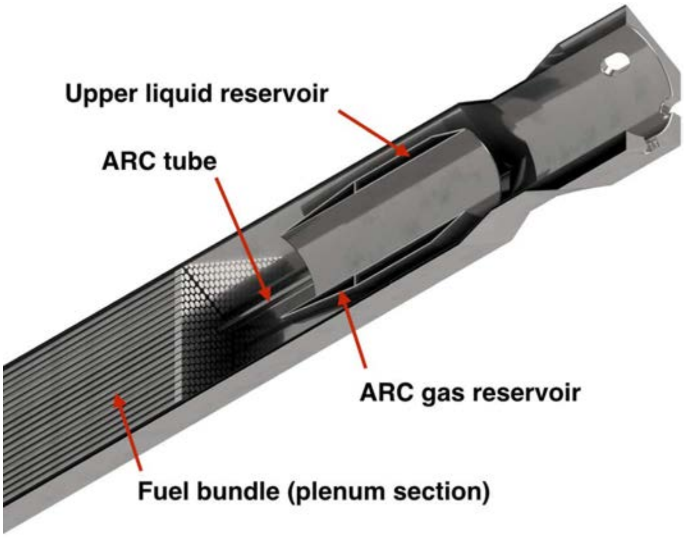

FIG. 54. Top of fast reactor fuel assembly with upper autonomous reactivity control reservoir (courtesy of S. Qvist, University of California, Berkeley).

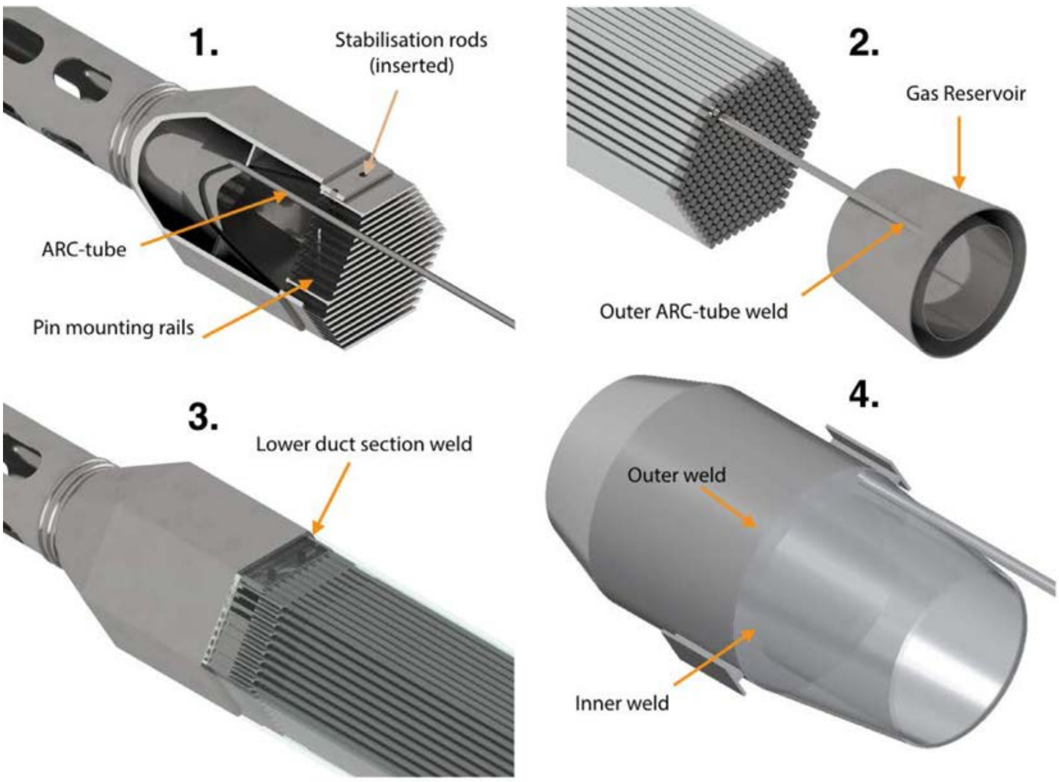

FIG. 55. Main assembling steps of a fuel assembly equipped with autonomous reactivity control (courtesy of S. Qvist, University of California, Berkeley).

5.2.4. Thermo siphon

5.2.4.1. Description of a thermo siphon

The basic concept behind the working of the proposed system is that one equilibrium configuration is transferred to another equilibrium configuration irreversibly when there is a severe thermal imbalance in the reactor. The basic principles are given in Section 3.13.

5.2.4.2. Description of a thermo siphon system

Figure 56 shows the passive shutdown assembly whose outer profile and handling features are the same as that of core assemblies. It contains a poison module at the centre and two rows of fuel pins surrounding it. Such assemblies are proposed at suitable locations inside the core. The choice of location, number and enrichment of the poison is such that in the case of an unprotected transient event, poison injection gets triggered in this module and brings the reactor to a cold shutdown state. The nominal outlet temperature of sodium exiting from this assembly is 550°C during normal operating conditions. The system is designed to trigger poison injection when the sodium temperature goes about 100°C above normal operating condition. In a power reactor for a typical ULOF event initiated by loss of off-site power, bulk boiling of sodium in the hottest subassembly top starts at ~25 s and fuel melting starts at ~75 s. In order to ensure that no bulk boiling begins even in the event of ULOF, the system is designed such that about 80% of the poison gets injected within 25 s. The margin of 100°C above nominal is considered adequate to prevent spurious triggering with expected temperature increase during protected transients.

Liquid lithium is the selected poison material and it is kept above the active core by high density liquid A (cadmium) in the left limb. Potassium is the low density liquid B in the right limb. A layer of potassium is kept in between the cadmium and lithium to avoid mixing [44]. In the proposed concept, thermal expansion of the cadmium triggers the siphoning of the potassium in the right limb. In order to enhance the expansion of the triggering liquid, it is kept inside a spiral tube about 4.5 m long (see Figs 56 and 57). The triggering liquid is exposed to liquid sodium exiting out of the assembly after removing the heat from the fuel pins inside the same assembly. Dimensions of the components were worked out such that they would meet the functional requirement and they can be fitted inside a typical hexcan.

(a) Estimated typical time required for poison injection

The time required for the sodium temperature to rise to the triggering temperature depends upon the transient event, but is about 5 s for a ULOF initiated by an off-site power failure. Once the temperature of the sodium increases, it takes about 2 s for the temperature of the liquid inside the spiral tube to reach the same temperature as the sodium. Once the temperature is more than 65°C above nominal, expanding liquid starts entering the siphon tube, and at 100°C above nominal, the liquid column in the siphon tube provides enough head for quick siphoning.

The flow of liquid is initially highly transient, and it can be treated as a rigid column flow. For rigid column flow with a constant friction factor and constant head, the discharge behaviour is governed by the following equation [45], which gives the time to reach a steady state velocity:

$$t = \frac{L}{v_0 C} \ln\left[\frac{v_0 + v}{v_0 - v}\right] \tag{3}$$

where v_0 is the steady state velocity and v is the transient velocity.

By rearranging the equation, we can calculate the transient velocity (v_{tr}) at any instant:

$$t = v_{tr} = \frac{\left(v_0 \times e^{\frac{tv_0 C}{L}} - v_0\right)}{1 + e^{\frac{tv_0 C}{L}}} \tag{4}$$

The steady state velocity (v_0) can be calculated based on Bernoulli's equation:

$$v_0 = \sqrt{\frac{2 \times \Delta h \times g}{(1+c)}} \tag{5}$$

where c is the loss coefficient and Δh is the available head.

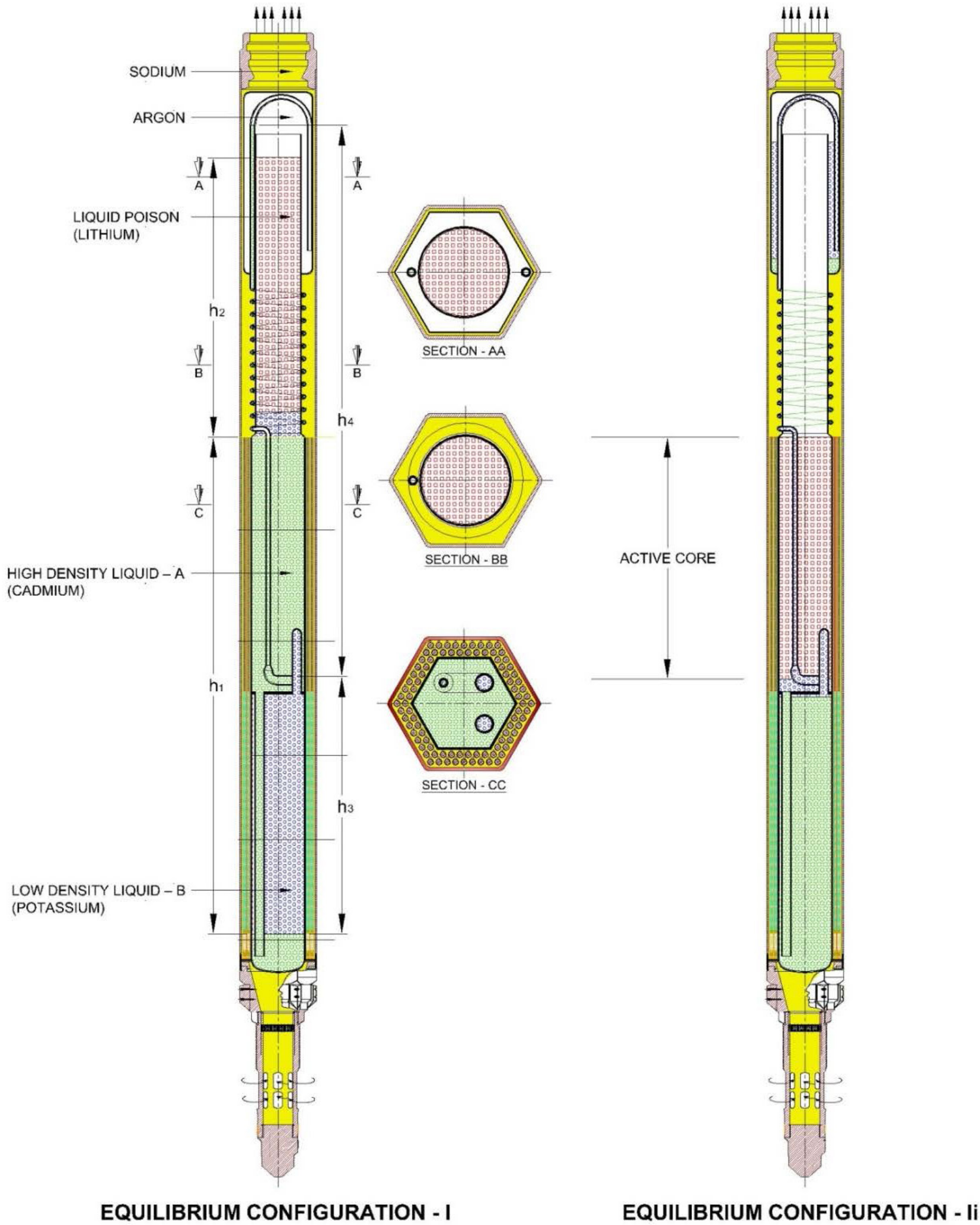

FIG. 56. A thermo siphon passive shutdown subassembly.

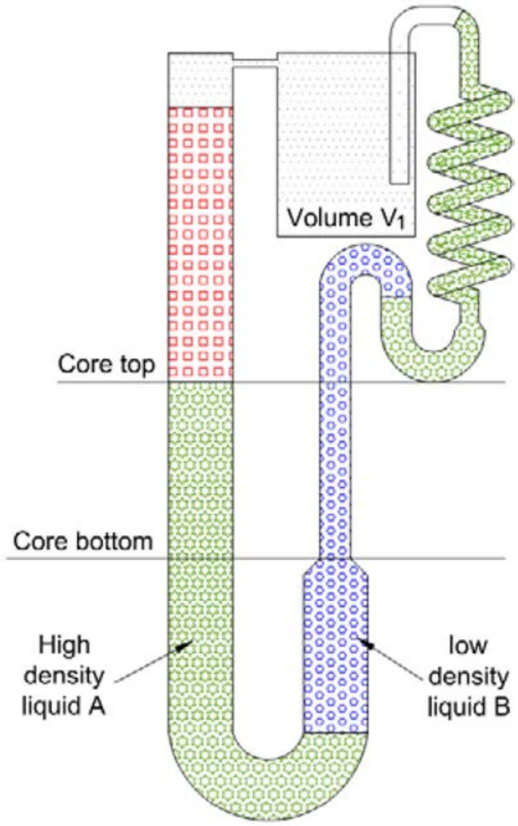

FIG. 57. Thermo siphon (courtesy of V. Raju, Indira Gandhi Centre for Atomic Research).

At the start of siphoning, the head is provided by the thermal expansion. The head increases as the flow covers the siphon pipe, after which the head remains constant. Hence the flow in the siphon was divided into the following three processes and the time was estimated using the above governing equations.

— Transient flow of the liquid until it covers the pipe length (h_c = 0.8 m) is about 4.6 s. In this condition, head also varies as flow progresses. See Fig. 58.
— Flow of the liquid until it reaches a steady state velocity with a constant head (travel distance 2.4 m) is about 1.5 s. See Fig. 59.
— Flow of the liquid at steady state velocity until the tube (spiral tube is about 4.5 m) gets drained is about 1.6 s.

By this time almost 23% of the core gets covered by poison. After siphoning, liquid B in column h_3 of Fig. 57 is pushed into volume V_1 owing to pressure imbalance. The driving potential keeps changing during flow. At the beginning, the driving potential is at its maximum, and it falls to zero after reaching the equilibrium shown in configuration II of Fig. 56. It is treated as a quasi-steady process and the time required to reach configuration II is estimated.

In total, it takes about 17 s to insert the poison up to 80% of the core height and a further 7 s for complete insertion.

The safety studies performed with such an insertion rate show that a reactor can be safely shut down, as shown in Fig. 60. Hence, based on the preliminary estimates, it is concluded that it is feasible to address unprotected transients with such a system and further R&D is needed to realize such a system for future reactors. The following challenges and R&D requirements are foreseen:

— Immiscibility and stability of liquid metals in the system;
— Suitable container material for the liquid metals in all operating conditions;
— Suitable liquid metals for all reactor conditions (i.e. maintaining liquid phase during shutdown condition and DECs);

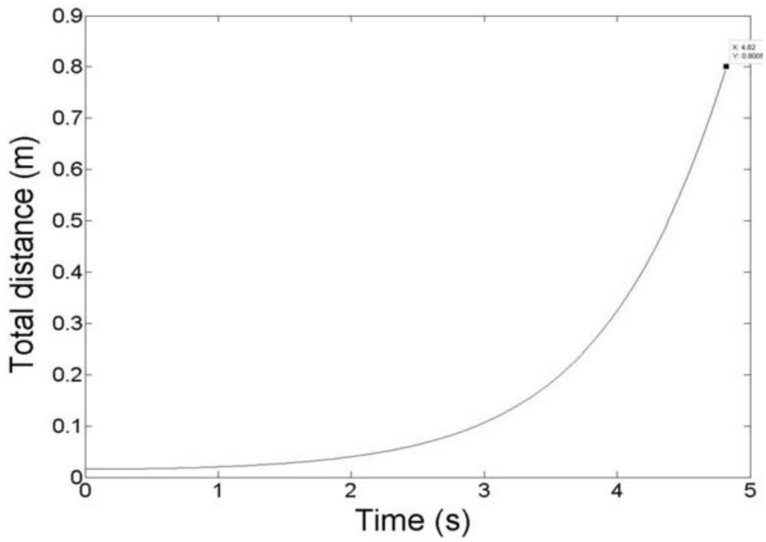

FIG. 58. Transient flow of liquid until it covers the siphon pipe length.

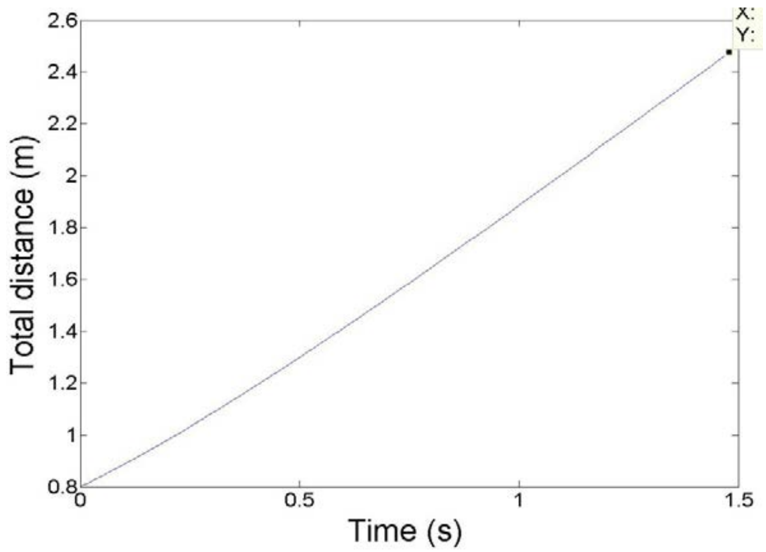

FIG. 59. Flow of liquid until it reaches steady state velocity.

— Production and containment of tritium during operation;
— Effect on permanent components in case of leak.

5.2.5. Floating absorber for safety at transient

The floating absorber for safety at transient (FAST) is a passive safety device designed to counteract an unacceptable increase of the coolant temperature and a loss of coolant from the core region. In other words, the FAST device provides a practical solution to the positive coolant void reactivity of fast reactors. As shown in Fig. 61, a single FAST device is simply a cladding-like guide thimble containing an absorber rod and a buoyancy canister. Basically, the guide thimble for a FAST device is identical to the cladding tube, except it has small holes at the top and bottom so that the coolant flows through with a negligibly low speed during normal operation.

The working principle of the FAST device is extremely simple and fully passive. During normal operation, the guide thimble is filled with coolant, and the absorber rod floats owing to the buoyancy of the absorber part and empty void canister below. Once the coolant temperature increases to a certain level, the buoyancy force should be

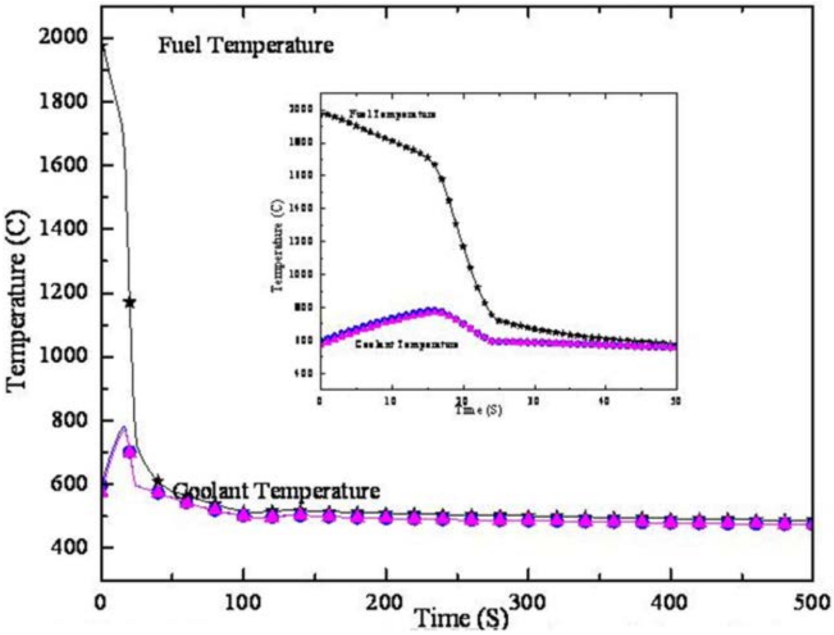

FIG. 60. Evolution of temperature with poison insertion during an unprotected loss of flow event (courtesy of V. Raju Indira Gandhi Centre for Atomic Research).

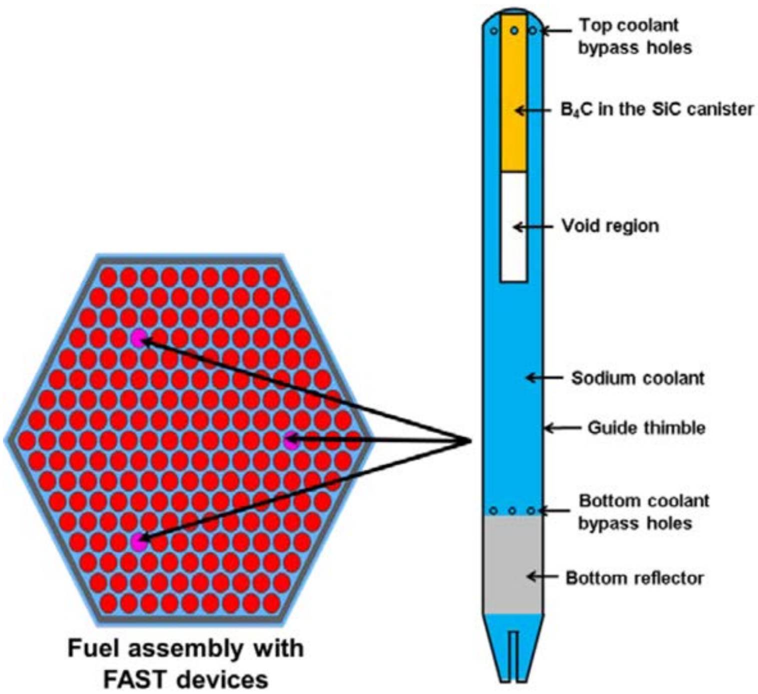

FIG. 61. Floating absorber for safety at transient passive safety device.

weaker owing to the reduction of coolant density. When the effective density of the absorber and empty canister region is higher than the coolant density, the absorber section starts to sink to the core region, thereby inserting negative reactivity in a passive way. Therefore, the coolant can be passively protected from overheating by the FAST device. The FAST module can work even when a local coolant blockage or coolant loss takes place for some reason. In the case of coolant loss, the absorber section should be inserted to the core region by gravity.

In the FAST device, a strong neutron absorber is loaded into a thin tube. The absorber needs to be light. Additionally, the FAST device needs to have a long lifetime to be used in a long life SFR core. Taking into account the design goals of the FAST device, materials and dimensions of the FAST components were determined. The FAST module uses an enriched B_4C neutron absorber enclosed in a SiC/SiC composite [46] canister. For reliable operation of the FAST device, the absorber rod and buoyancy canister should be much smaller than the inner radius of the guide thimble. It should be noted that the absorber section is not joined with the lower buoyancy canister. This allows flexible movement in the narrow guide thimble. A SiC/SiC composite material was chosen for the thin canister material since it is immune to neutron irradiation and is compatible with the sodium coolant and the B_4C absorber.

To ensure that the B_4C absorber section floats during normal operation, the effective density of the B_4C region should be much lower than the theoretical one. The low density of the B_4C also ensures that the absorber will not swell due to the helium gas produced by the ^{10}B. A large amount of ^{10}B can be depleted in a FAST device during the whole irradiation period. It is assumed that the resulting helium gas can be vented through microholes made in the SiC/SiC composite canister and the helium pressure will remain low within the SiC/SiC composite canister. The SiC/SiC composite can be helium permeable depending on the fabrication process [47]. Thus, the helium gas can be vented if a helium permeable and low density SiC/SiC composite is used for the absorber section. Reference [47] also mentions that thermal expansion of the SiC based FAST components is much smaller than that of the sodium coolant and the buoyancy force for the FAST components should decrease with a coolant temperature increase. The linear thermal expansion coefficient of SiC is known to be about $4 \times 10^{-6} \cdot K^{-1}$ for 500 to 800°C [48], while it is about 10^{-4} L/K for the sodium coolant [49].

A longer buoyancy canister may be needed for a longer and heavier absorber section. The allowable length of the absorber rod and void canister of the FAST device will depend on the extent of the fuel pin bowing. It is expected that fuel pin bowing will be relatively smaller than that of the duct. However, if a shorter length is necessary, the absorber section can be divided into two parts. A reflector or shield is loaded into the bottom of the FAST device to support the absorber when it sinks. The thickness of the bottom shield depends on the length of the FAST device. The FAST device is designed so that the absorber section is fully out of the core during the normal operation and the top of the absorber rod contacts the upper cover of the thimble, as shown in Fig. 62. In the case of a full insertion of the FAST device, the absorber section of the FAST device should be in the middle of the

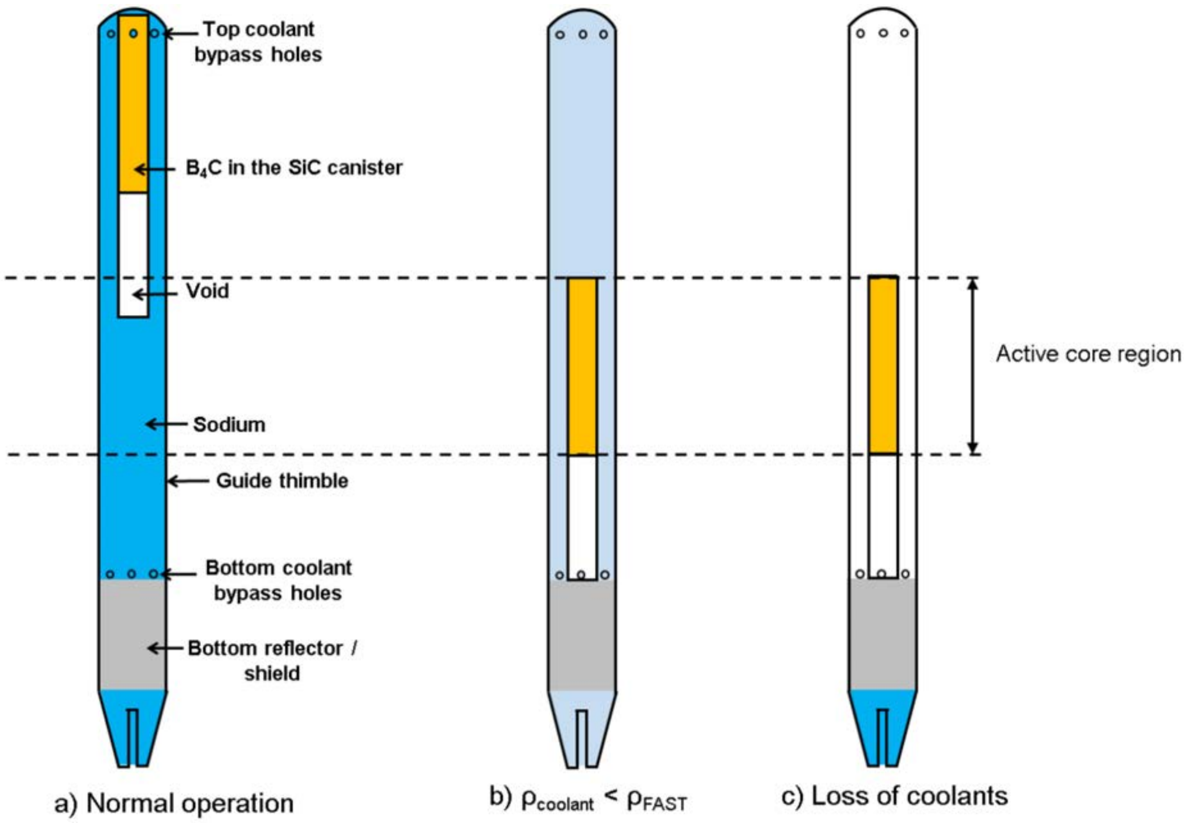

FIG. 62. Positions of a floating absorber for safety at transient device in several reactor conditions.

active core. If it is necessary to improve the sinking speed and depth, an alternative design of a FAST device has an absorber rod below the buoyancy canister (reversed design from the current one). It should be noted that in the alternative design, the absorber rod and the buoyancy canister need to be flexibly jointed.

It should be mentioned that the FAST device can also effectively counteract a local coolant flow blockage, which results in a local coolant temperature rise. In the case of a local flow blockage, only the FAST device installed in the specific fuel assembly will respond and prevent a local power increase and propagation of the detrimental impacts of the local flow blockage. Meanwhile, in the case of LOCA, the absorber will also passively drop into the core region due to gravity. In other words, the FAST device can provide strong negative reactivity in the case of coolant void as well. Figure 62 shows the position of the FAST device during the three scenarios.

It is clear that the FAST device can be easily installed into the fuel assembly by replacing a fuel pin. The number of FAST devices per fuel assembly can be determined based on the negative reactivity to be provided by the FAST.

5.2.6. Curie point temperature actuated absorbers

Development of SASSs based on magnetic materials and Curie point is under way in the Russian Federation [50, 51] and in Japan [52].

Magnetic materials are used in the latch that holds the absorber rods above the core. The materials lose their magnetism at a definite temperature. Thus, in accidents involving temperature increase, the absorber rod is released and drops into the core.

In Japan, research on such devices is based on electromagnets, while in the Russian Federation, devices of this kind are being developed to be fully autonomous [18].

An important task is to find materials that significantly change their magnetic properties within the desired temperature range. For the device under development in the Russian Federation, the actuation temperature (i.e. the temperature of the coolant) is taken to be 650–670°C with an actuation time under 5 s. A very important feature of the magnetic material for the autonomous variant is the weight it can hold.

In order to ensure the desired rated load, the following configuration of the CPM is used in the autonomous variant: a permanent magnet made of magnetic alloy with axial magnetization is surrounded by a screen made of ferrous–nickel alloy that has a Curie point of 620°C, inside an armature made of Armco iron which is connected to the absorber rod.

Figure 63 shows a mock-up of the CPM device for an experiment in a sodium rig. The experiment tested its rated load in a flow of sodium in the temperature range of 300–680°C, with a rate of temperature increase in the device of about 12°C per second. The rated load of the device in a gaseous atmosphere is about 8.2 kg at room temperature and 2.8 kg at 680°C. To study the effects of irradiation, magnetic material specimens have been placed in the BR-10 reactor.

These SASSs have the advantage of versatility and can be used to prevent any type of accident.

Their drawbacks are as follows:

- They employ the same principle of solid absorber insertion into the core as the standard shutdown system.
- They may fail to demagnetize owing to either insufficient temperature increase in the temperature sensitive material or an increase in its actuation temperature. The former may result from changes in thermohydraulic features of the device and the basic, thermophysical properties of the thermosensitive material. The latter may be due to changes in temperature or magnetic properties as a result of gap reduction (due to accumulation of some magnetic material from the coolant, swelling of the material, etc.), or a rise in Curie point and changes in magnetic permeability.
- The latch may fail to release despite demagnetization of the material (e.g. due to adhesion of the magnet to the rod).

5.2.7. Fusible melt system for sodium cooled fast reactors

In order to accommodate unprotected transients, as well as other unintended abnormal operational conditions, AREVA proposed the implementation of a thermally activated dedicated absorber device [53] which is fully diverse compared to the normal RSSs.

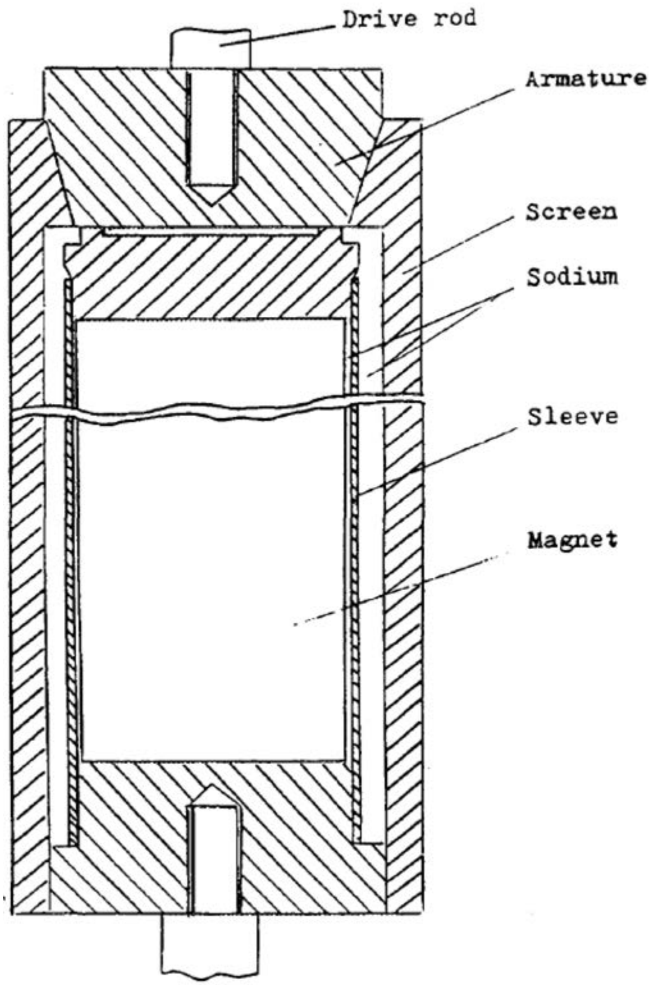

FIG. 63. Mock-up of magnetic actuating device [18].

One of the possible design options is the insertion of absorber B_4C pebbles inside the central fuel pin of the subassemblies, either in a limited number of subassemblies, or in dedicated subassemblies. Under nominal operational conditions, the B_4C pebbles are located above the fissile fuel region. They are released into the fuel core centre when a local coolant temperature, just above the fissile fuel region, exceeds preset limits. This solution is referred to as a 'self-protected' subassemblies concept, and uses a fusible melt system for SFRs.

In this particular case, all fissile subassemblies are assumed to have their central pin replaced with a pin of identical dimensions containing only gas and no fissile fuel. B_4C pebbles are located above the fissile zone supported by a fusible aluminium layer linked to the cladding just above the fissile zone outlet. In case of a significant core coolant outlet temperature increase, the fusible support layer melts, leading to the absorber B_4C pebbles dropping into the fissile core region, and subsequently to reactor shutdown. The self-protected additional shutdown device (ASD) schematics are presented in Fig. 64.

The melting temperature of the fusible devices should be high enough to prevent unintended delatching of the B_4C pebbles during normal reactor operations (as well as during category two and three protected transients).

Therefore, assuming relatively homogeneous temperature within the subassemblies at the fissile zone outlet (to be further assessed and confirmed), the fusible melting point should be above 640°C.

Technical characteristics of the proposed ASD/fusible melt system are:

— Melting temperature of the fusible device:
  - Reference value: 660°C;
  - Parametric cases: 640–715°C.

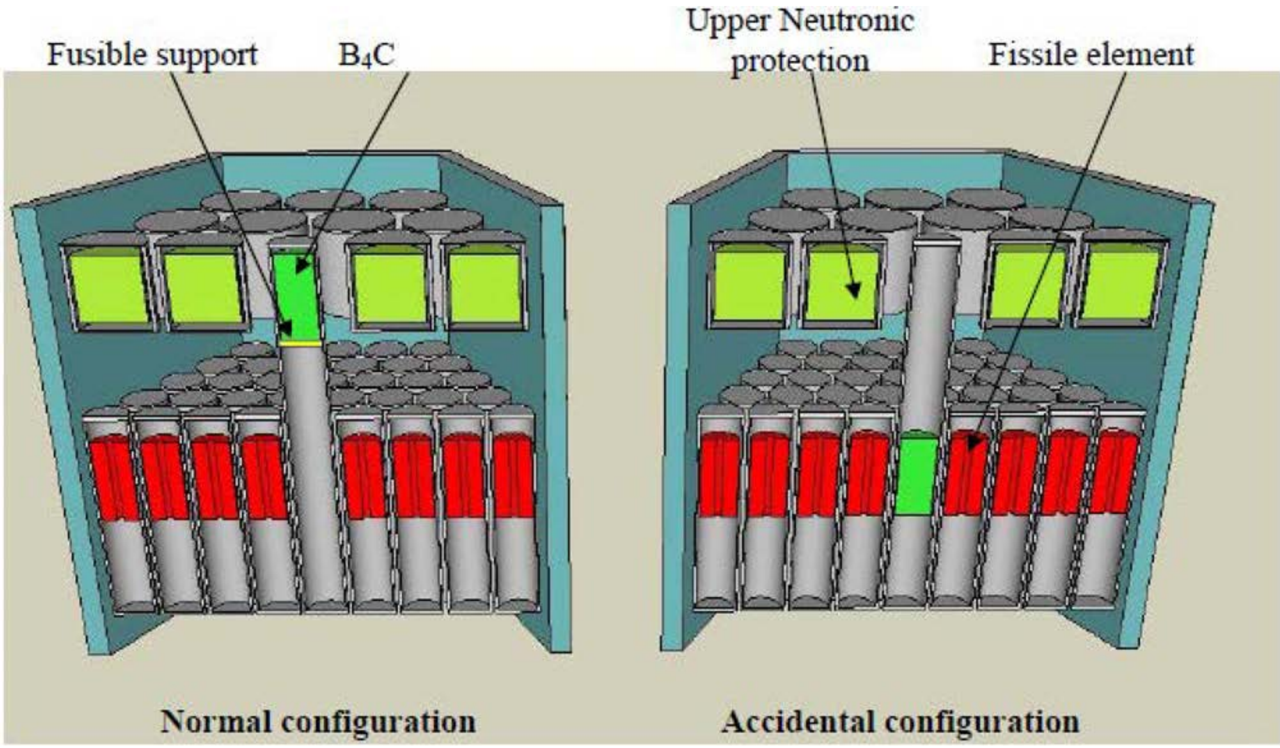

FIG. 64. Self-protected additional shutdown device schematics [53].

— Response time[9] of the fusible devices:
  - Reference value: 5 s;
  - Parametric cases: 2–8 s.

— Reactivity worth: −2.5 pcm per ‘self-protected’ subassembly.

The evaluation of severe transient behaviour in SFRs on the basis of the analysed ULOF transient and the impact of newly conceived mitigation measures were the main tasks of the study [53]. The reactor design used for the analysis was the SFR(v2b-ST) [54], and the system code used was the SIM-SFR code [55].

The main results of the parametric ULOF study for SFR(v2b-ST) with an ASD in operation can be summarized as follows:

— In most of the analysed cases, an ASD is capable of safely shutting down the reactor in a timely manner during a ULOF transient, thus avoiding sodium boiling and a related possible power excursion.

— The only requirement for the ASD implementation is that not less than 50% of the total number of the fuel assemblies in the reactor core should be equipped with the ASD. Equipping a lower number of fuel assemblies with the implemented ASD does not provide sufficient reactivity potential for reactor shutdown in case of a ULOF transient, in particular under bona fide end of equilibrium cycle (EOEC) core conditions (control rods essentially totally withdrawn, yielding minor rod extension reactivity feedback potential).

In general, the study clearly demonstrated that the implementation of an ASD driven by fusible devices as proposed by AREVA is a very effective system — with inherent safety feature characteristics — in mitigating unprotected transients (i.e. ULOF, etc.).

[9] This is the length of time between the sodium surrounding the ‘fusible pins’ reaching the melting temperature of the fusible devices and the B_4C pebbles dropping into the core. Therefore, the response time accounts for the fusible heating up and melting, as well as for the B_4C pebbles dropping into the core.

6. PLANT TRANSIENT ANALYSIS

6.1. TARGET PLANT RESPONSE: EXPERIMENTAL BREEDER REACTOR II

One approach for achieving SFR passive shutdown capability is to develop engineered systems that can insert negative reactivity under off-normal conditions and that are incorporated into the design of the core and/or surrounding structures. A second approach is to capitalize on features of the reactor design itself that are able to passively shut down the reactor based on inherent reactivity feedback. This latter approach was examined in the shutdown heat removal tests (SHRTs) performed at the Experimental Breeder Reactor II [56], which was a 60 MW(th) metal fuelled test reactor operated by Argonne National Laboratory in the USA. These tests were conducted with the specific goal of demonstrating the potential for an SFR to survive severe accident initiators with no core damage.

During the most ambitious of these tests, referred to as SHRT 45, the primary and intermediate pumps were tripped while the core was operating at full power, and the control rods were not inserted. The sodium core outlet temperature measured during this ULOF sequence is shown in Fig. 65. The coolant outlet temperature is noted to rise rapidly (~200°C in 30 s). However, the thermal expansion of core components enhances neutron leakage and, owing to the net negative reactivity feedback, the reactor power is shut down autonomously and the coolant temperature rise is terminated. Past this point, residual decay heat is removed by natural convection through the core.

The second notable experiment in this series was a loss of heat sink without scram initiated from 100% power. The core inlet and exit temperatures measured during this ULOHS test are shown in Fig. 66. The reactor inlet temperature rise from the sudden loss of heat rejection introduces negative reactivity from expansion of the grid support, lower reflector and other core materials. By the time the inlet temperature has risen about 40°C, the reactor power is shut down. It is noteworthy that the core outlet does not overshoot, but decreases from the value at full power. This type of behaviour (i.e. the low, long term reactor outlet temperature and the lack of transient temperature overshoot) is typical of metal fuelled SFRs owing to the high thermal conductivity of the fuel which results in low Doppler reactivity feedback at power [56].

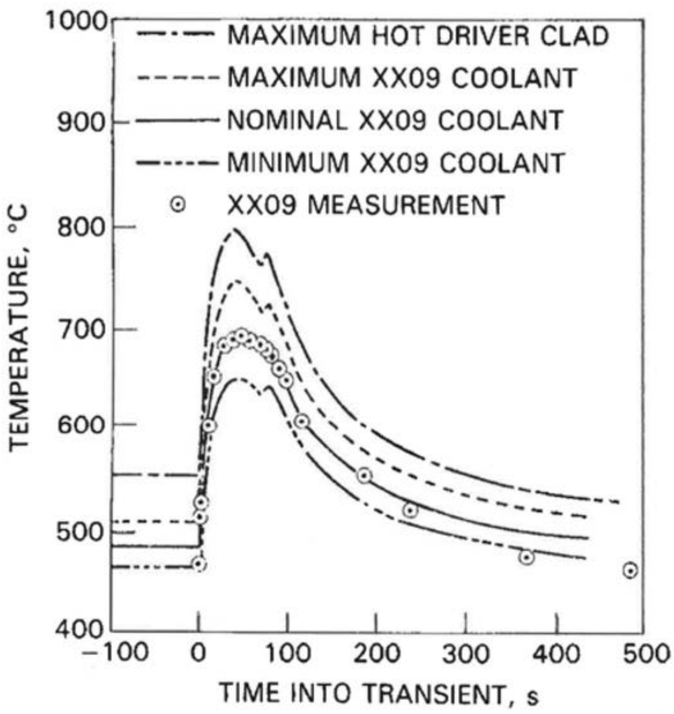

FIG. 65. Unprotected loss of flow from 100% power with 95 second pump coastdown time — predictions and measurements of in-core temperatures (courtesy of Argonne National Laboratory).

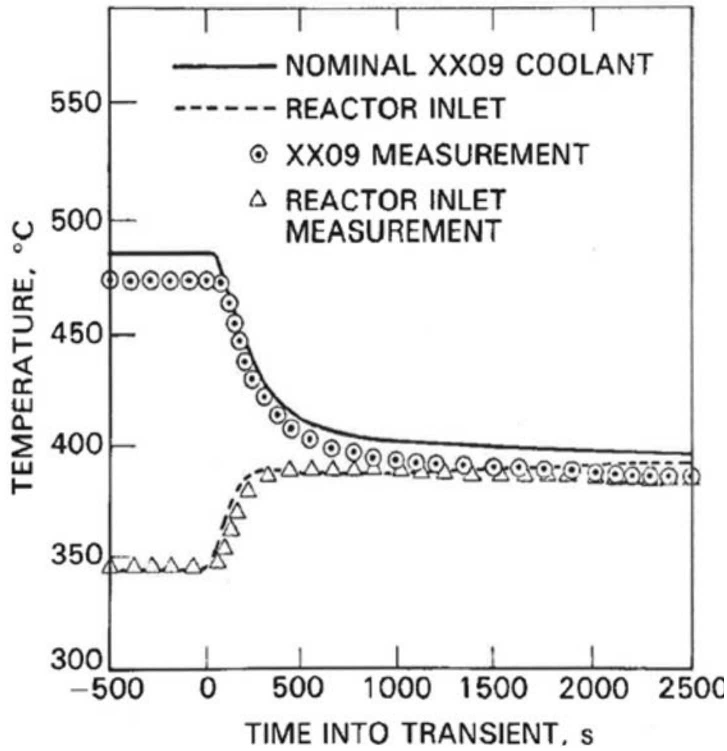

FIG. 66. Unprotected loss of heat sink from 100% power — pretest predictions and measurements of reactor temperatures (courtesy of Argonne National Laboratory).

The implications of these tests are that an SFR can be designed to be inherently safe with respect to severe undercooling accidents. In particular, the technical feasibility of passive shutdown and subsequent heat removal by natural circulation for ULOF and ULOHS sequences was demonstrated for a metal fuelled pool type reactor design [56]. Passive shutdown was achieved without automatic scram, operator intervention or special in-core devices. The most important features to achieve this type of response are the ones that provide reactivity feedback. These types of integral feedback coefficient are typical of metal fuelled reactors of all sizes, principally because of the high thermal conductivity that is characteristic of metal fuel. This can result in low temperature fuel and a reactor with relatively small Doppler feedback. In contrast, for oxide fuel, the low thermal conductivity can lead to higher fuel temperatures and significantly higher Doppler feedback, particularly in larger reactors that have a softer neutron spectrum and consequently a large Doppler coefficient. As a result, the power reactivity decrement is large in these reactors and is dominated by terms proportional to power or fuel temperature. The temperature response to a ULOF or ULOHS in a uranium oxide core therefore tends to be much less favourable than the response in a metal fuelled core.

6.2. FAST BREEDER TEST REACTOR EXPERIENCE

6.2.1. Introduction

The Fast Breeder Test Reactor (FBTR) is a 40 MW(th) loop type sodium cooled reactor currently operational in Kalpakkam, India. The currently running campaign of the FBTR has a core configuration comprising 37 Mark-I, 7 Mark-II and 8 MOX subassemblies. The thermal power of the plant with this core configuration is 26.1 MW(th). The process parameters envisaged for the operating campaign are given in Table 12. In order to demonstrate the safety of the plant, analyses of various design basis events (DBEs) were carried out using the plant dynamics code DYNAM.

TABLE 12. PLANT PROCESS PARAMETERS FOR 26.1 MW(th) OPERATION

No.	Parameter	Value
1	Reactor power	26.1 MW(th)
2	Primary sodium flow	203 kg/s
3	Secondary sodium loop flow	48 kg/s
4	Feedwater flow	10.54 kg/s
5	Reactor inlet temperature	399°C
6	Reactor outlet temperature	498°C
7	Steam generator sodium inlet temperature	497°C
8	Steam generator sodium outlet temperature	290°C
9	Feedwater temperature	190°C
10	Steam temperature	460°C
11	Steam pressure	12.5 MPa

Temperature and power coefficients of reactivity are −4.28 pcm/°C and −11.12 pcm/MW(th), respectively, for this campaign.

6.2.2. Results of event analysis without safety actions

Various enveloping DBEs whose consequences are likely to challenge core safety were analysed without safety actions. Analyses without safety actions confirmed that the response of the plant is benign. The following paragraphs describe the behaviour of the plant under five DBEs without safety actions.

6.2.2.1. One primary sodium pump trip

This event is characterized by the speed coast down of one primary sodium pump (PSP) under its own drive inertia. The evolution of primary sodium flow through the two pumps supplying flow to the core is shown in Fig. 67. Because two PSPs are operated in parallel, the flow supplied by the other pump increases. The flow supplied by the tripped pump decreases and even starts bypassing the flow supplied by the other pump. The non-return valve on the suction side of the pump prevents excessive reverse flow through loop 1. Net core flow decreases to 77% in about 25 s. The reduction in core flow causes an increase in sodium and structural temperatures. Negative feedback reactivity effects, due to increased temperature, reduce reactor power as shown in Fig. 68. Reactor power decreases to a minimum of 46% in about 250 s and then finally stabilizes close to 55%. The cladding hotspot in the central subassembly increases to a maximum of 728°C at 23 s and then starts falling owing to power reduction. The maximum temperature of sodium reached at the exit of the central subassembly is 556°C. The junction of two primary pipes at the reactor inlet (culotte) sees a hot shock of 1°C/s after a 30 s delay.

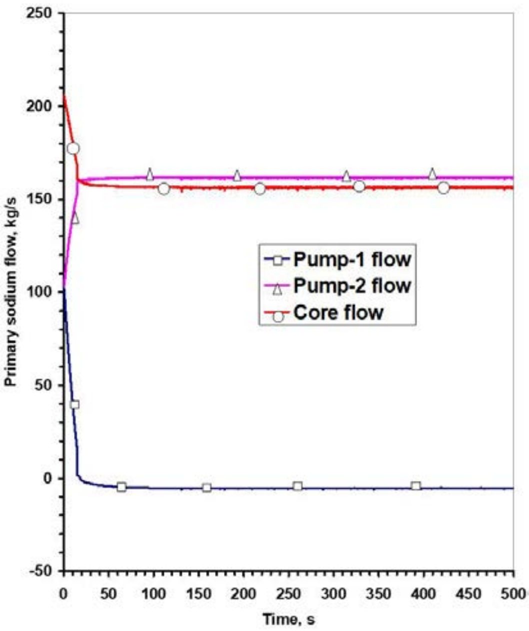

FIG. 67. Primary sodium flows during one primary sodium pump trip event without safety action.

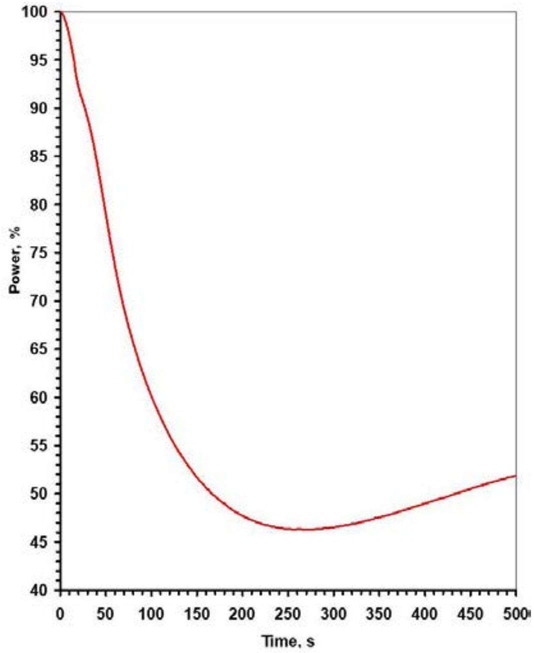

FIG. 68. Reactor power during one primary sodium pump trip event without safety action.

6.2.2.2. One primary sodium pump seizure

The primary sodium pump failure is simulated by coasting down one PSP in 1 s. The pump 1 flow decreases and the pump 2 flow increases very sharply as shown in Fig. 69. A non-return valve prevents reverse flow thorough pump 1. Core flow decreases rapidly to 77%. The reduction in core flow causes the core temperatures to increase. The resulting negative reactivity feedback effects reduce the reactor power as shown in Fig. 70. The reactor power stabilizes at about 55%. Cladding hotspots reach a maximum temperature of 745°C in 1 s.

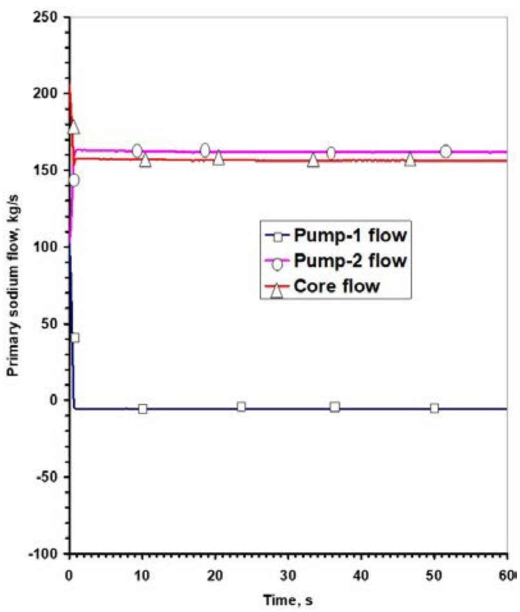

FIG. 69. Primary sodium flows during one primary sodium pump seizure event without safety action.

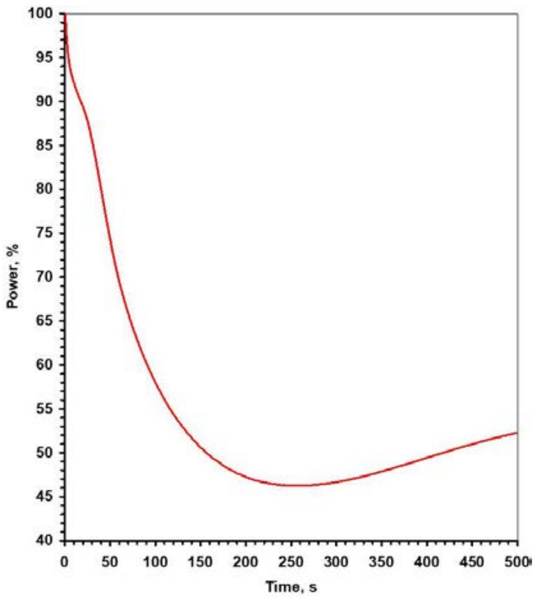

FIG. 70. Reactor power during one primary sodium pump seizure event without safety action.

6.2.2.3. One secondary sodium pump trip

This event is characterized by the speed coast down of one secondary sodium pump (SSP) governed by its own inertia. The evolution of secondary sodium flows in the two loops is shown in Fig. 71. Sodium flow in loop 1 decreases to 50% in 25 s and subsequently decreases to natural convection flow conditions (25%). Reduction of the secondary sodium flow through the IHX causes a reduction in heat transfer from the primary to the secondary in that loop. This increases the IHX primary outlet temperature of loop 1 and hence increases the reactor inlet temperature. The reactor inlet temperature increase introduces the negative reactivity feedback effect owing to a grid plate expansion, and the reactor power decreases. The power finally stabilizes at about 68%. Thus, the cladding temperature will only decrease during this event, as evident in Fig. 72.

6.2.2.4. One secondary sodium pump seizure

An SSP failure is simulated by coasting down one SSP in 1 s. The pump 1 flow reduces to natural convection flow conditions very sharply. The qualitative behaviour of the plant is similar to that following a one SSP trip event with a marginal difference: because of the faster reduction of sodium flow, the thermal transients are a little faster. Because of the negative reactivity feedback effects reducing the reactor power, the core temperatures will only decrease from their full power values. The hot shock seen by the primary cold end of the pump 1 IHX is 5°C/s for about 20 s.

6.2.2.5. Total loss of feedwater flow

A total loss of feedwater flow event is simulated by the reduction of water flow in the steam generators to 0.1% in 0.3 s. Because of the loss of heat sink, the reactor inlet temperature increases after a delay which in turn causes reactor power to decrease continuously owing to negative reactivity feedback effects. The evolution of reactor power and primary circuit temperatures is shown in Figs 73 and 74, respectively. The reactor power decreases to 9% in 500 s. The cladding hotspot temperature also decreases continuously. Hence, there is no concern for core safety during this event.

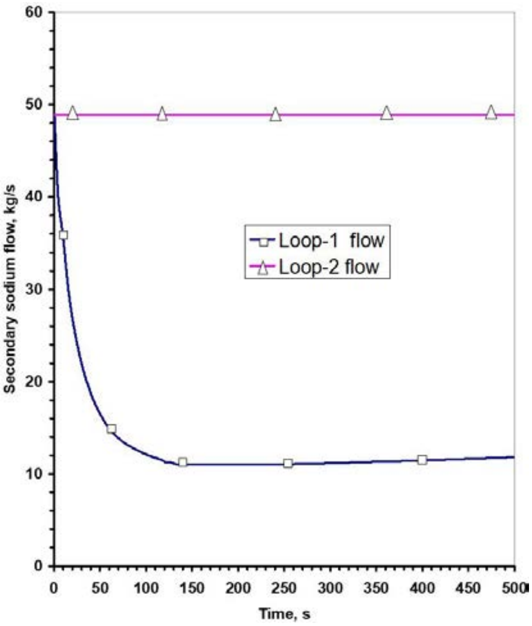

FIG. 71. Secondary sodium flows during one secondary sodium pump trip event without safety action.

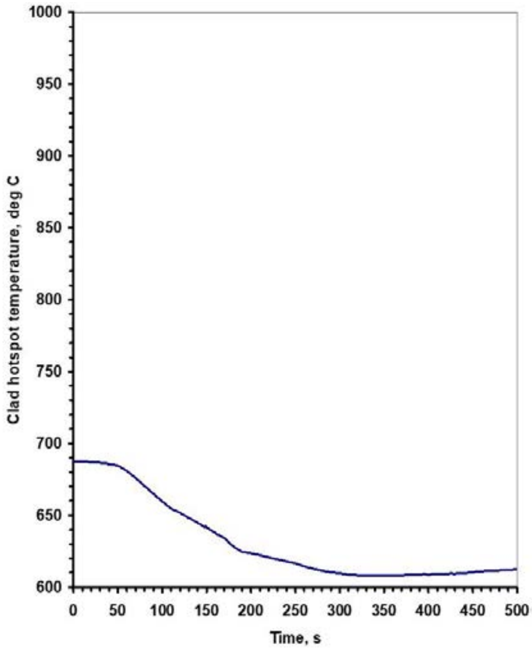

FIG. 72. Cladding hotspot temperature during one secondary sodium pump trip event without safety action.

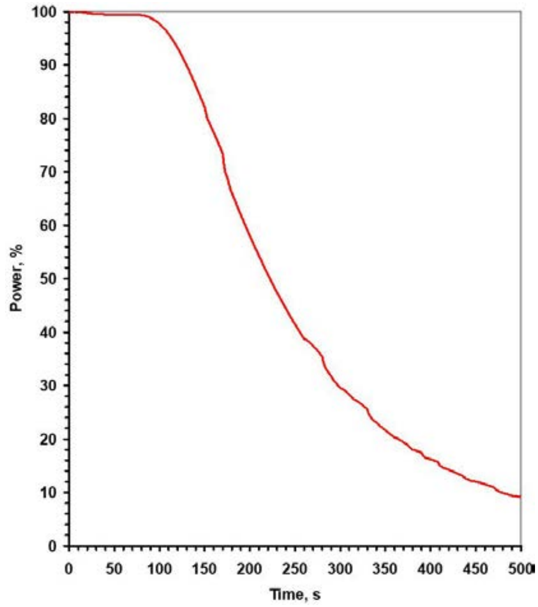

FIG. 73. Reactor power during loss of feedwater without safety action.

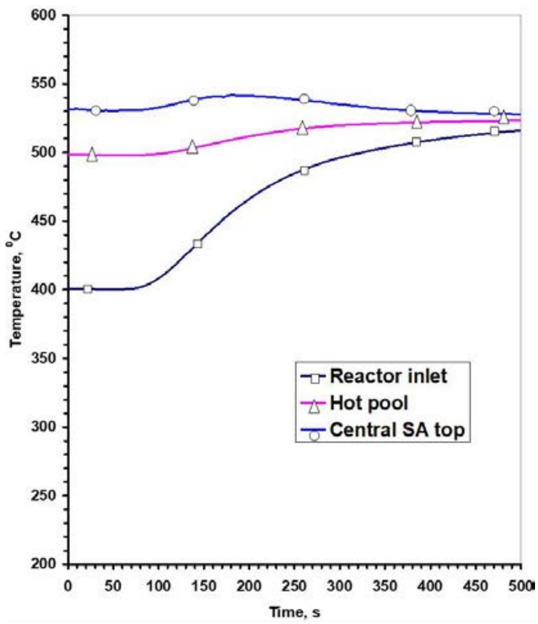

FIG. 74. Primary circuit temperatures during loss of feedwater flow event without safety action.

6.2.2.6. Conclusions

Detailed event analyses of ULOF events at the FBTR were carried out using the plant dynamics code DYNAM. Consequences of the following DBEs were analysed:

— One PSP trip;
— One PSP seizure;
— One SSP trip;
— One SSP seizure;
— Total loss of feedwater flow.

The results indicate that even without safety actions, the reactor stabilizes at a lower power level and important safety parameters like the cladding hotspot temperature and the fuel centre line temperature are within the acceptable limits. Hence, it can be concluded that the FBTR core is inherently safe for the above mentioned ULOF events.

6.3. RESULTS OF TRANSIENT ANALYSIS WITH SOME INNOVATIVE CONCEPTS

6.3.1. Efficiency of a passive safety system as an additional shutdown device for the prevention of severe accidents in sodium cooled fast reactors

The evaluation of severe transient behaviour in SFRs and the impact of newly conceived mitigation measures were the main tasks for this study. The SFR used for the analysis was the SFR(v2b-ST) reactor design [54], and the system code used was SIM-SFR [55].

The transient initiators chosen for testing the mitigation measures for SFR(v2b-ST) reactor design were ULOF and ULOHS transients. More information on these safety studies can be found in Ref. [53].

6.3.1.1. Unprotected loss of flow transient analysis without an additional shutdown device

ULOF is a transient event characterized by a combination of several failures occurring simultaneously, namely an unintentional trip of the primary pumps followed by a failure to trip the reactor by inserting the control and/or shutdown rods. The ULOF initiator event is judged to be highly improbable, with an estimated frequency of occurrence of less than 10^{-8} per year.

The ULOF event is essentially characterized by a mismatch between the production of power and the removal of heat by forced or natural convection from the reactor core. Tripping the primary pumps will lead to a continuous decrease of the mass flow rate through the reactor core during the pumps' coast down phase, until the natural convection process takes over as governed by buoyancy and the frictional forces of the flow paths, largely determined by the chosen design and layout characteristics of the primary system.

During the flow coast down process, the power level should ideally decrease in proportion to the decrease in the mass flow rate. However, the mass flow rate during ULOF usually decreases faster than the power level, leading to an overheating of the core materials. The parameter that is most impacted by the ULOF is the core coolant outlet temperature (T_{outlet}), as the core coolant inlet temperature (T_{inlet}) remains largely unchanged during the initial phase of the pump coast down process. The sensitive parameter of primary concern during a ULOF process is thus the T_{outlet}.

In large, sodium cooled, oxide fuelled fast reactors of the 1500–3000 MW(th) class of conventional core design, extensive boiling of sodium (T boil ~880°C at 0.1 MPa, and 937°C at 0.157 MPa) in the central core region will most likely insert significant positive reactivity in excess of 1$, leading potentially to a power excursion. Conditions that could lead to sodium boiling in the central core region are thus to be avoided under all circumstances in large SFRs. On the other hand, limited boiling of sodium in the upper core regions, or the core outlet, can be tolerated to some extent, as in this case, negative reactivity is inserted into the core. Local boiling of sodium in the upper core regions can thus be tolerated as long as it can be assured that the boiling front does not propagate downward towards the central region of the core, where reactivities then can become large and positive. However, limiting boiling of sodium to just the upper core outlet regions has been shown experimentally to be a very delicate task, as sodium boiling can be a highly unstable process [57]. Any minor imbalance between power production and heat removal (flow rate) could trigger a very rapid progression of the boiling front downward towards the core centre. In addition, sodium boiling is known to drastically decrease the mass flow rate within the affected flow channel (subassembly) owing to the increased two phase flow frictional head loss in the dried out portion of the flow channel, causing the boiling front to naturally propagate towards the core centre.

It is of interest to see how the design of the various core and primary system components can be influenced or preselected to optimize the SFR(v2b-ST) plant response during a ULOF, meaning essentially that it avoids initiating local sodium boiling. Seventeen different ULOF parameter cases were analysed using the SIM-SFR code, assuming that no so called ASD is implemented. Adopting a conservative approach, the cases consider reactivity feedback coefficients according to EOEC core conditions, and control rods inserted to ~25% of their full insertion length (corresponding to core conditions at about the middle of the equilibrium cycle), thus providing average control rod extension reactivity feedback effects under ULOF conditions. The purpose of this parameter study was to isolate a viable combination of primary system design modifications that would allow accommodation of a ULOF transient without running into a power excursion.

The main results of this parametric ULOF study for SFR(v2b-ST) without an ASD can be summarized as follows:

— No single design modification alone will 'fix' the problem (i.e. prevent a ULOF power excursion). At least two or three design modifications in conjunction are needed to have a design that can accommodate a ULOF in SFR(v2b-ST) without entering into unstable boiling, leading to a power excursion.
— To have any chance at all of accommodating a ULOF, a different primary pump with a longer rundown half-life (similar to the SPX1) is needed foremost to provide more grace time for ex-core driven thermal inertia reactivity feedback effects, especially CRDL thermal expansion, to become effective.
— Without a different primary pump, no alternative design option — or combination thereof — will fundamentally change the outcome of a ULOF transient in SFR(v2b-ST), which is that the plant will eventually enter into a power excursion: not changing the height differential between core and main heat

exchanger (MHX) midplanes, not decreasing the primary system pressure drop, and not a combination of both.

— To sustain a ULOF in SFR(v2-ST), it appears that a minimum height differential of about 4.6 m is needed between core and MHX midplanes, in conjunction with three additional design modifications, such as:
  - SPX1-like primary pump;
  - Low nominal primary system pressure drop of about 0.215MPa;
  - 50 cm rod insertion depth.

From the above, the necessity of implementing a third, independent ASD, such as the one proposed by AREVA[10], could be foreseen.

6.3.1.2. Unprotected loss of flow transient analysis with an additional shutdown device

In order to accommodate unprotected transients, as well as other unintended abnormal operating conditions, AREVA proposed the implementation of a thermally activated dedicated absorber device as an ASD, fully diverse compared to the normal RSSs.

Table 13 provides a summarized overview of the ULOF transient sensitivity to the variation of different ASD design parameters. From this table, one can determine the influence of certain parameters of the ASD system on the response to ULOF in SFR(v2b-ST).

TABLE 13. UNPROTECTED LOSS OF FLOW ANALYSIS RESULTS FOR SFR(v2b-ST) UNDER END OF EQUILIBRIUM CYCLE CONDITIONS USING AN ADDITIONAL SHUTDOWN DEVICE (ASD)
(including the results from the sensitivity analysis of different ASD design parameters)

Case	Prim. pump half-life (s)	Subassemblies with ASD (%)	Trigger temperature (°C)	Delay time (s)	ASD dropping (%)	Result
Impact on	Flow	Power	Power	Power	Power	Power excursion at transient time (s)
R0	10	0	660	5	100	37
R1 (Ref. case)	10	100	660	5	100	n.a.
R2	10	100	640	5	100	n.a.
R3	10	100	715	5	100	n.a.
R4	10	100	660	2	100	n.a.
R5	10	100	660	8	100	n.a.
R6	10	100	715	8	100	n.a.
R7	10	75	715	8	100	n.a.
R8	10	50	715	8	100	n.a.
R9	10	25	715	8	100	158 (local boiling starts at 95 s)
R10	10	25	715	8	50	158 (local boiling starts at 105 s)

Note: n.a. — no power excursion.

[10] In the EFR project, an enhanced thermal expansion of the control rods was investigated [58].

During the ULOF transient in SFR(v2b-ST) with an ASD, the core outlet temperature increases quite rapidly, reaching about 660°C between 5 and 6 s into the transient. If the melting temperature of the fusible device is set at 660°C, a clock counter is activated and, depending on the chosen delay time (i.e. 5 s (input parameter)), the highest temperature subassemblies will release their B_4C pebbles from above the active core zone to enter the central part of the core region between 10 and 11 s into the ULOF transient. The percentage of subassemblies releasing their B_4C pebbles into the core region is determined based on outlet temperature differences between the peak and the average subassembly and the outlet temperature gradient of the average subassembly. This SIM-SFR internally calculated rate (ASD subassemblies/sec) can then be multiplied by a factor (rate_SA_fail, nominal value = 1.0) to be supplied as an input parameter in order to investigate the importance of the B_4C pebbles' drop rate on the course of the transient under consideration.

The main results of this parametric ULOF study for SFR(v2b-ST) with an ASD in operation can be summarized as follows:

— In most of the analysed cases, the ASD is capable of safe shutdown in a timely manner during the ULOF transient, thus avoiding sodium boiling and a resultant power excursion.
— The only requirement for the ASD implementation is that not less than 50% of the fuel assemblies in the reactor core be equipped with the ASD. Installing fewer ASD does not provide sufficient reactivity potential for reactor shutdown in case of a ULOF transient — in particular under bona fide EOEC core conditions (control rods essentially totally withdrawn, yielding a minor rod extension reactivity feedback potential).

6.3.1.3. Unprotected loss of heat sink transient analysis

The ULOHS transient is characterized by a total loss of heat sink — it is assumed that neither the MHX nor the heat exchangers of the decay heat removal system system are available as a heat sink. In addition, the RSS is assumed to have failed. The only potential heat sink available is radiative heat transfer from the vessel wall to the surroundings, assuming the reactor cavity cooling systems are appropriately dimensioned to extract the heat coming from the vessel wall. Despite the high sodium temperature which could be reached, the primary pumps are assumed to continue to operate at nominal flow.

In total, six different ULOHS cases were analysed using the SIM-SFR code. Results of this parametric ULOHS study for SFR(v2b-ST) are as follows:

— During the ULOHS transient, the CRDL expansion reactivity coefficient will lead to a relatively important positive reactivity insertion into the core because the vessel walls heat up significantly. The 'CDRL expansion reactivity feedback effect' is actually composed of two separate, opposing feedback effects: (a) the thermal expansion of the rod drive extensions located in the upper plenum, which causes an insertion of the rod bank into the core during the primary system heat-up phase, and (b) the counteracting effect caused by the thermal expansion of the reactor vessel wall, which leads to an extraction of the entire rod bank out of the core region once the vessel wall heats up. During ULOHS, the vessel wall driven rod extraction process overshadows the rod drive extension insertion process, thereby adding positive reactivity into the core. If no ASD is implemented, SFR(v2b-ST) will enter into a power excursion about 310–370 s into the ULOHS transient as the high reactor power will initiate sodium boiling about 20–30 s prior to initiation of the power excursion.
— If an ASD is available during a ULOHS transient, the ASD's B_4C pebbles are released into the core region about 50 s into the transient, decreasing the reactor power to the decay heat level within 50 s transient time. For SFR(v2b-ST), a total of 300 ASD subassemblies are activated within the first 500 s of the ULOHS, counteracting the positive vessel wall reactivity effect (about +330 pcm for SFR(v2b-ST)) by assuring a net negative reactivity balance. This demonstrates the effectiveness of the ASD in preventing a power excursion during ULOHS for the SFR(v2b-ST) design.

6.3.1.4. Conclusions

This analysis demonstrated that the sodium cooled reactor design SFR(v2b-ST) is a viable core and primary system design under nominal power conditions. Maximum cladding temperatures remain below 600°C and maximum fuel temperatures remain below 2200°C. The nominal pressure drop across the core is calculated to be 0.35 MPa, with an estimated total primary system nominal pressure drop of 0.485 MPa.

Under ULOF conditions and without a third shutdown (ASD based) safety system, a parameter study has shown that at least three design modifications need to be made to the SFR(v2b-ST) system design in conjunction in order to accommodate a ULOF. Foremost, a different primary pump with rundown characteristics similar to those of the SPX1 pump would be required (SPX1 pump half-time of about 50 s), and this would need to be accompanied by an increased height differential between the core and the IHX to at least 4.6 m (up from 2.35 m currently) and a decrease of the drop in primary system nominal pressure to 0.215 MPa (down from 0.485 MPa currently).

Aside from the above indicated design modifications, the only other remaining alternative for an even larger decrease in the risk of a CDA would be the introduction of an additional independent device, as proposed by AREVA, based on fusible materials releasing B_4C absorber pebbles into the fissile core region once certain system temperatures exceed the specified temperature limits. This study has shown the effectiveness of using such an ASD in the central fuel pins in all subassemblies (or a certain fraction not less than 50%) to limit the consequences of ULOF and ULOHS transients. Should the proposed ASD design actually function within the parameter range investigated in this report (fusible melt temperature, B_4C ball drop time delay, fraction of subassemblies in which the ASD system is implemented, etc.), then this system will be effective in shutting down the reactor in time if the two primary shutdown systems fail.

In general, the study has clearly demonstrated that the implementation of an ASD driven by fusible devices as proposed by AREVA is a very effective system — with inherent safety characteristics — in mitigating unprotected transients such as ULOF and ULOHS. However, the inadvertent actuation of some of the ASDs and their potential consequences remains an open issue to be further studied.

6.3.2. Transient analysis of fast reactor cores with autonomous reactivity control systems

Detailed transient analysis has been carried out for both the reference and the ARC equipped versions of many existing fast reactor systems and concepts. Here, we present the transient analysis results for the Argonne National Laboratory Advanced Burner Reactor with oxide fuel core and a conversion ratio of 0.75 [59]. More details regarding this analysis can be found in Ref. [60]. The ARC installation was designed and configured to stabilize the core at an asymptotic mean coolant outlet temperature of 750°C following a ULOF event. The general input data for the core and ARC system operation are summarized in Table 14.

The reactivity feedbacks and corresponding required ARC system reactivity worth are summarized in Table 15. The calculation of the required ARC reactivity worth in Table 15 is based on the limiting (most serious) event, which is the ULOF transient at EOEC.

The transient simulations were made using the code CHD. A detailed description of the code together with comparative benchmarking of other established codes can be found in Ref. [60]. The heat transfer to the ARC reservoirs was precalculated using the COMSOL Multiphysics software package for a wide range of coolant temperatures and different primary coolant flow velocities. The ARC system axial reactivity worth profile in the Advanced Burner Reactor core was calculated using the Serpent neutron transport code [61].

The temperature, power and net reactivity of the core following an ULOF event are shown in Fig. 75. The lithium of the ARC system reaches the bottom of the active core as the temperature of the upper reservoir reaches 560°C. This is the cause of the sharp drop in net reactivity just after 10 s into the transient. After the first few minutes of the transient, the system slowly and monotonously transitions toward its quasi-static critical state with a coolant outlet temperature (as designed) of 750°C.

Figure 76 shows the results of an ULOHS simulation. The reference core (without ARC systems installed) stabilizes at a temperature just 10 K below the atmospheric pressure boiling point of sodium, while the ARC equipped core slowly transitions to a quasi-static critical state with a uniform coolant temperature just above 700°C (with more than 180°C temperature margin to boiling). About two hours into the transient, the core is neutronically shut down, but the decay heat power level is still above the decay heat removal capacity (0.5% of full power) and the temperature of the core continues to increase, bringing it to a subcritical level. After about 13 hours (46 000 s) the decay heat level falls below the capacity of the decay heat removal system and the temperatures start to decrease until the core becomes critical again after about 47 hours (170 000 s). Following a transient transition, the core stabilizes again, in a critical state, at a power level matching the capacity of the decay heat removal system. Since

TABLE 14. GENERAL INPUT DATA FOR AUTONOMOUS REACTIVITY CONTROL EQUIPPED ADVANCED BURNER REACTOR CORE TRANSIENT EVALUATION

Parameter	Value
Average linear power (kW/m)	18.3
Ref. coolant inlet/outlet temperature (°C)	355–510
Effective delayed neutron fraction (β_{eff})	0.0033
Transient overpower initiator (ρ_{ext}, ¢)	28
Coolant temperature rise (ΔT_{co}, °C)	155
Normalized natural circulation flow rate (F_n)	0.03 (3%)
Passive heat removal capacity (P_d)	0.005 (0.5%)
Temperature change to core for the ARC system (°C)	50
Maximum allowable long term temperature (T_{LC})	750
Full actuation ARC temperature rise (ΔT_f, °C)	190

Note: ARC — autonomous reactivity control.

TABLE 15. REACTIVITY FEEDBACK PARAMETERS OF THE ADVANCED BURNER REACTOR CORE

Reactivity feedback coefficient	BOEC[a]	EOEC[b]
Radial expansion coefficient (¢/°C)	−0.35	—[c]
Fuel temperature coefficient[d] (¢/°C)	−0.52	−0.54
Structure density coefficient (¢/°C)	0.08	—[c]
Sodium density coefficient (¢/°C)	0.15	—[c]
Total ARC reactivity worth (¢)	−160.9	

[a] BOEC: beginning of equilibrium cycle.
[b] EOEC: end of equilibrium cycle.
[c] —: no data available.
[d] Sum of fuel axial expansion and Doppler coefficients.

the normalized flow to power ratio reaches 200 ($F = 1$, $P = 0.05$), the coolant temperature rise at this point is less than 1°C.

Finally, Fig. 77 shows a station blackout scenario, which is essentially a combination of the ULOF and ULOHS events. In the reference case, the station blackout scenario leads to coolant boiling at 87 s in the hottest channel. With the ARC system installed, the coolant temperature stays at about an 80°C margin to boiling throughout the transient, and stabilizes at around 720°C. Since the coolant inlet temperature increases in the station blackout event, more negative reactivity is introduced than in the typical ULOF event (in which inlet temperature is

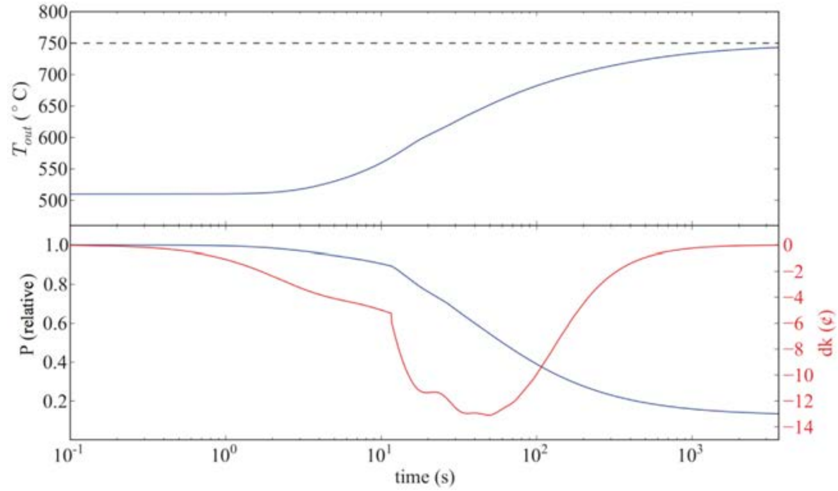

FIG. 75. Advanced Burner Reactor unprotected loss of flow quasi-static temperature prediction and transient simulation results. Upper panel: mean coolant outlet temperature (dotted black line: quasi-static prediction); lower panel: power and net reactivity. Note the discontinuity in the reactivity after about 10^1 s, which is due to the ARC system being engaged.

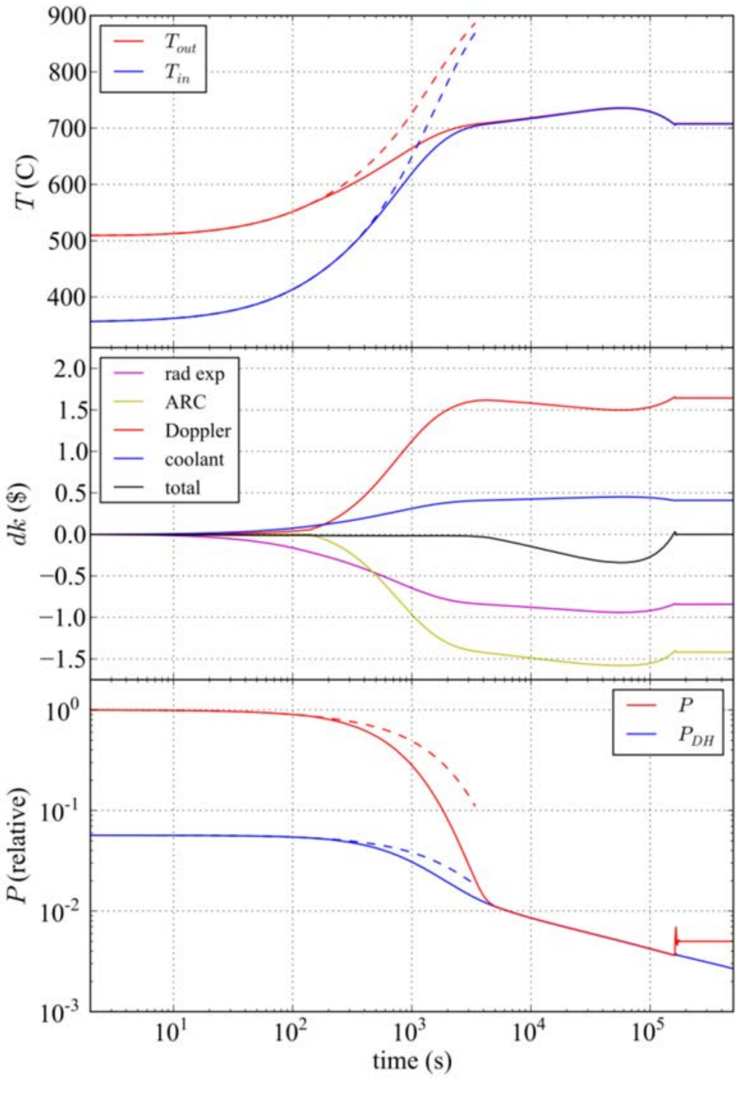

FIG. 76. Unprotected loss of heat sink simulation results. Upper panel: coolant temperatures (dotted lines are without autonomous reactivity control systems installed); middle panel: reactivity components; lower panel: power (total and decay) and flow rate.

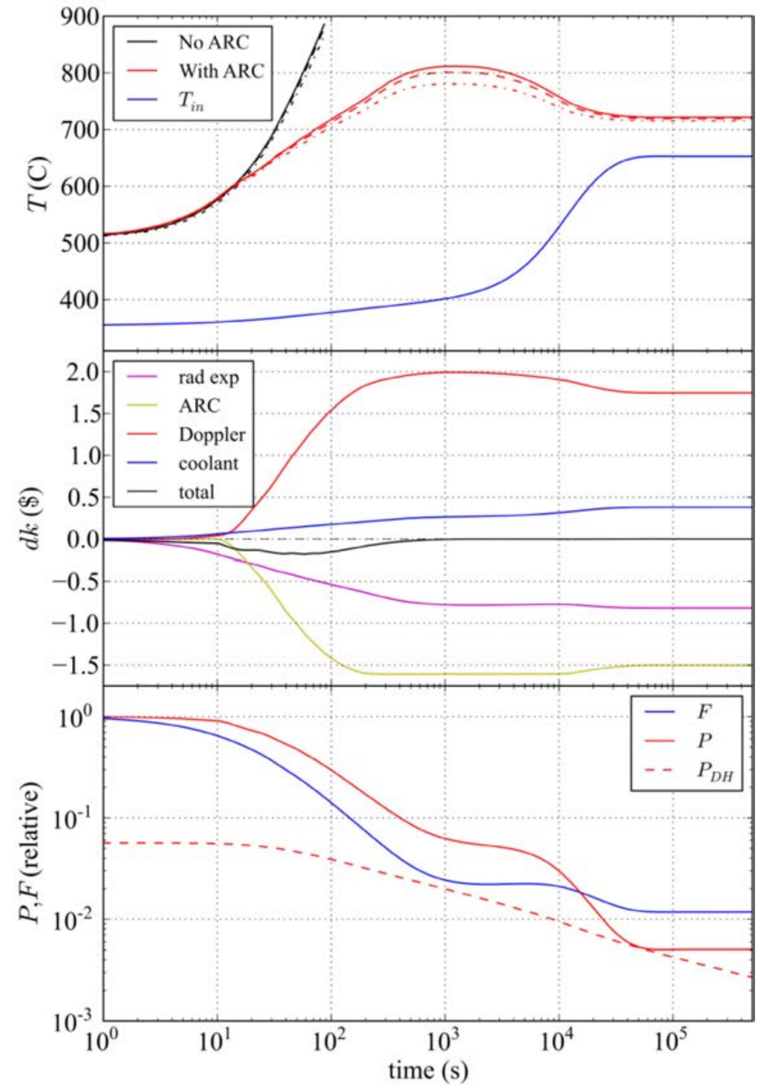

FIG. 77. Station blackout simulation results. Upper panel: coolant temperatures (the three red lines correspond to the three separate channels); middle panel: reactivity components; lower panel: power and flow rate.

assumed to remain constant). Because of this, the peak coolant temperature reached in the station blackout event is about 30°C lower than in the ULOF event.

The UTOP event was simulated but showed no differences between the ARC equipped and reference design cores, since the coolant temperature rise is not high enough for the lithium of the ARC system to reach the active core.

If ARC systems are designed to introduce too much reactivity worth per degree of temperature change, the time delay of actuation, especially in loss of flow events, may cause severe (and unacceptable) oscillations in power and temperature. Such behaviour can be avoided entirely by accepting a higher quasi-static stabilization temperature, thus increasing the operating temperature range of the ARC system and decreasing its total reactivity worth. As can be seen in the transient simulation results presented here, it is possible to design an ARC system to respond to the most severe unprotected transients that are anticipated in a typical fast reactor core while maintaining temperatures at acceptable levels, without introducing any oscillatory behaviour. The economic and neutronics penalty of replacing one fuel rod per assembly (out of 271) and installing the ARC reservoirs (thereby slightly increasing system pressure drop) is likely more than made up by the assured passive safety performance of the ARC equipped core. Perhaps more importantly, incorporating ARC equipped fuel assemblies into the design process of new fast reactor cores may open up a design space that was previously inaccessible due to the requirement that passive safety performance rely solely on existing reactivity feedback components.

6.3.3. Transient analyses for the ALFRED plant

Lead cooled fast reactors (LFRs) have been selected as one of the possible options for Generation IV reactors [62]; along with other fast reactors, they allow the closure of the fuel cycle leading to a reduction of uranium consumption, reduction of the transuranic waste requiring geological disposal and reduction of the fuel waste long term radiotoxicity. The deployment of LFRs in the short term will fulfil the aims of the Sustainable Nuclear Energy Technology Platform to target the goals proposed by the European Union through its Forward-Looking Climate Change Policy and Energy Security Strategy.

With this in view, the European Commission cofunded the European Lead-cooled System (ELSY) project [63] under FP6 (2006–2010), which led to a preliminary design for the ELSY reactor of 1500 MW(th) (renamed European Lead Fast Reactor (ELFR)). Afterwards, the Lead-cooled European Advanced Demonstration Reactor (LEADER) project was funded under FP7 (2010–2013). The work performed has been focused on the revision and further development of ELFR as well as on the preliminary design of its demonstrator reactor (300 MW(th)) named Advanced Lead Fast Reactor European Demonstrator (ALFRED).

The main goal of the ALFRED plant is to play the role of a demonstrator for the European concept of a LFR, by proving the safety and reliability of the simple engineering solutions adopted for all operating conditions, and by reducing the uncertainties in the design, construction and operation to the largest possible extent.

As a first step, based on the outcomes and the accumulated expertise in the previous projects, the main technological constraints, goals and safety performances for the ALFRED core were identified (Table 16).

TABLE 16. ALFRED CORE DESIGN PARAMETERS

Parameter	Value
Thermal power	300 MW
Maximum inner vessel radius	~150 cm
Fuel assembly concept	Closed hexagonal
Fuel type	MOX[a]
Maximum fuel temperature	~2000°C
Maximum Pu content in the fuel	~30%
Peak burnup	100 MW·d/kg
Maximum fission gas plenum pressure	5 MPa
Cladding material	15–15 Ti (density at 20 °C: 7.95 g/cm^3)
Maximum cladding temperature at nominal conditions	550°C
Coolant	Pb
Coolant inlet temperature	400°C
Coolant outlet temperature	480°C
Maximum coolant velocity	2 m/s
Maximum cladding temperature during ULOF	750°C

[a] MOX: mixed oxide.

A comprehensive approach that takes into consideration all the relevant neutronic, thermohydraulics, thermomechanics and safety aspects was adopted in order to fulfil the requirements and to reach the safety performance shown in Table 16. In this approach a key role is played by the choice of the control and safety systems.

In the case of the ALFRED, the design of the control and safety systems has been adapted from the CDT-MYRRHA project [64]. Two independent systems (shown in Fig. 78) have been considered: (a) a control rod system, used for both normal control of the reactor and for scram in case of emergency, and (b) a safety rod system, used only for reactor scram.

The control rods consist of a cylindrical bundle of 19 pins. Their withdrawn position is below the core, and they are actuated by motors during reactor operation. They are also provided with an electromagnetic connection whose release allows a rapid insertion into the core by buoyancy in case of emergency shutdown.

The safety rods consist of a bundle of 12 absorbing pins positioned atop the active zone during normal operation. They are only actuated for scram by unlocking an electromagnet. When the electromagnet is turned off, the resistance to a pneumatic system is simultaneously lost, so that the safety rods are passively but rapidly pushed into the core. Moreover, if the pneumatic system fails, tungsten ballast which is located atop the safety rods has the appropriate weight to counteract the buoyancy and ensure the insertion of the safety rods into the core.

The reference configuration for the ALFRED plant shown in Fig. 79 consists of 171 fuel assemblies (57 inner core fuel assemblies and 114 outer core fuel assemblies fuelled with MOX) and 16 control/safety assemblies. The primary coolant working temperature is in the 400–480°C range. The core power is removed by eight steam generators, installed in an annular zone near the top of the reactor vessel, and connected with eight secondary circuits which are fed with water at 335°C and 18.8 MPa. Superheated steam at 450°C and 18 MPa pressure from the steam generator outlet feeds the turbine to produce a net electrical power of about 125 MW(e). The decay heat removal system (DHR) in the ALFRED is made up of two independent and redundant systems: DHR-1 — four

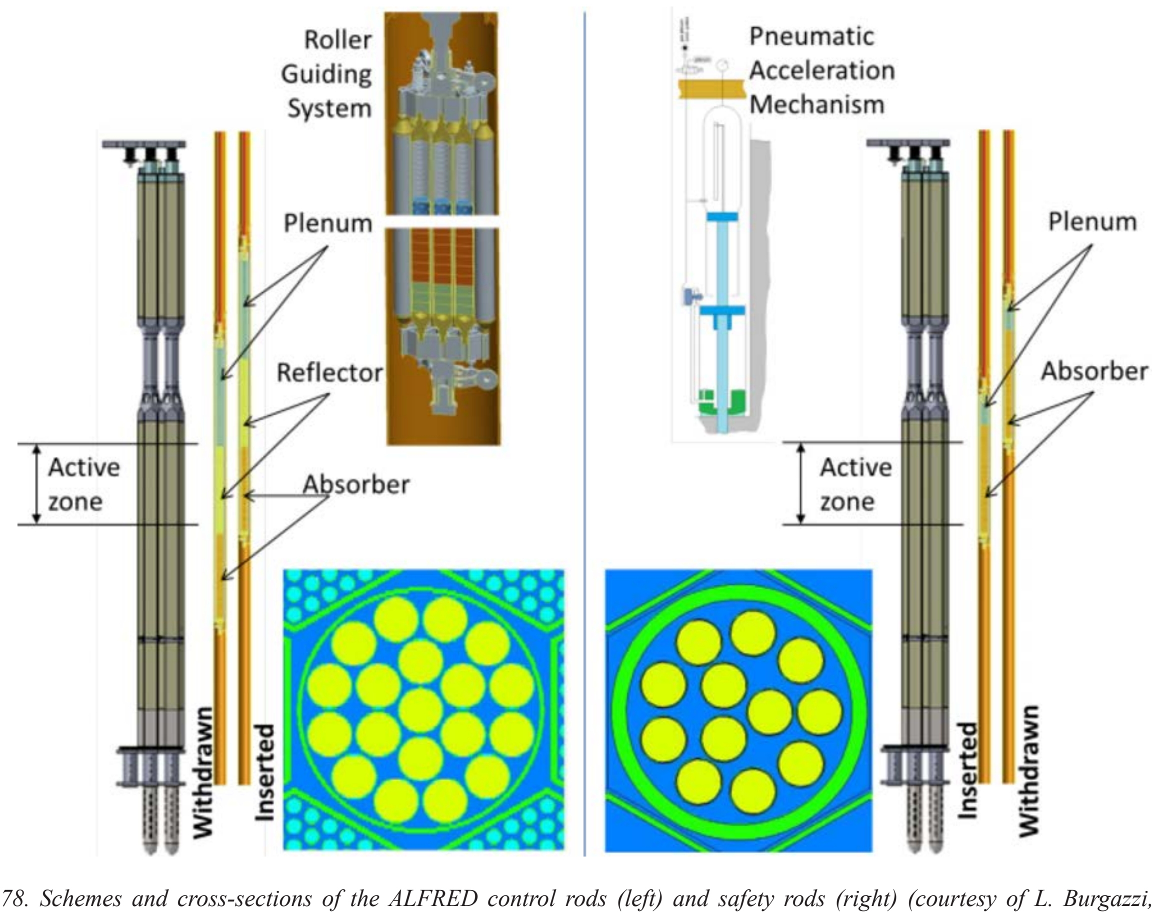

FIG. 78. Schemes and cross-sections of the ALFRED control rods (left) and safety rods (right) (courtesy of L. Burgazzi, Italian National Agency for New Technologies, Energy and Sustainable Economic Development).

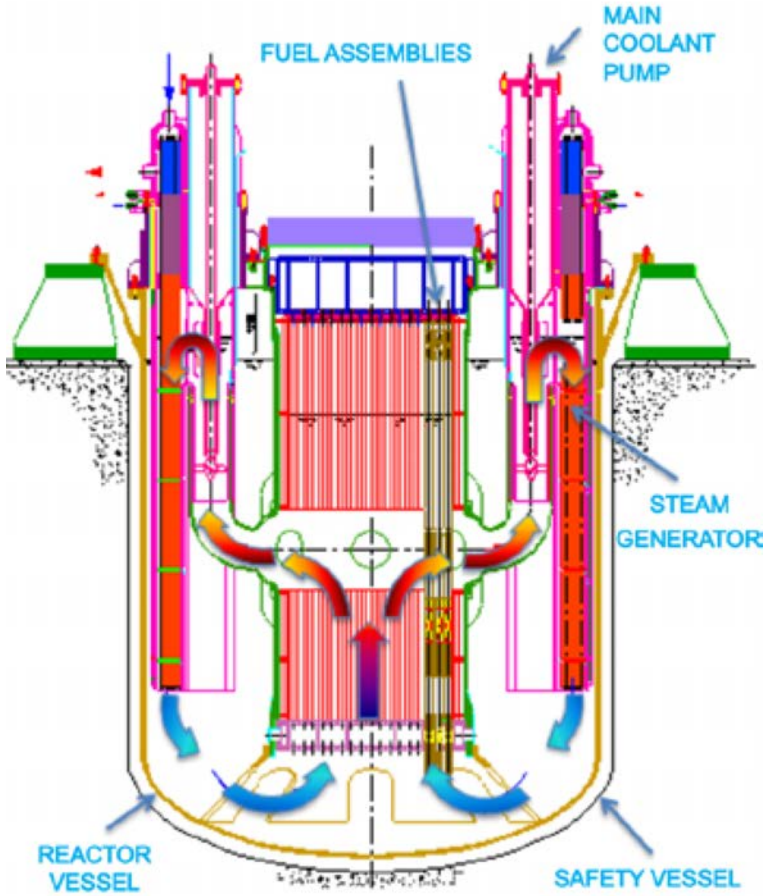

FIG. 79. ALFRED primary systems (courtesy of L. Burgazzi, Italian National Agency for New Technologies, Energy and Sustainable Economic Development).

isolation condenser systems connected to four steam generators, and DHR-2 — four isolation condenser systems connected to the remaining four steam generators.

6.3.3.1. Preliminary safety analysis

The preliminary safety analysis for the ALFRED was performed under the LEADER project based on the deterministic and phenomenological analyses, which represent one of five analytical tools of the integrated safety assessment methodology. The use of deterministic and phenomenological analyses includes the safety analysis for design basis accidents and DECs. Different system codes (RELAP5/MOD3.3 [65], SIM-LFR [55], FAST [66], SIMMER-III [67], CATHARE V2.5_2 [68], SPECTRA [69]) adapted to deal with lead coolant were used for the design basis accident and DEC transient analysis by the partners involved in the LEADER project [70, 71].

6.3.3.2. Preliminary safety analysis — design basis condition events

The postulated protected design basis condition (DBC) events are those in which the reactor is assumed to shut down normally upon demand. The main interest of this study is focused on the determination of the available grace time before critical core temperatures are reached leading to potential cladding failure, and the determination of the available grace time before critical (freezing) temperatures are reached at the outlet of the MHX following the failure/malfunction of the secondary cooling system (usually overcooling event(s)).

For the ALFRED design, a total of 12 transients were identified for detailed DBC (protected) transient analysis covering a wide range of the potential transient initiators that might occur during reactor operation [70]. Some typical DBC transient initiators are loss of flow and loss of heat sink, transient overpower, subassembly blockage, overcooling of the primary side due to malfunction of the secondary side cooling system and heat exchange tube rupture.

The prevention of fuel cladding failure (rupture) during transient conditions was considered one of the primary safety criteria ensuring fuel pin integrity, and thus ensuring the ultimate safety of the plant. Cladding failure can be considered to directly correlate to cladding temperature and internal pin plenum pressure. Thus, during the transients, all the computer codes have monitored either the maximum cladding temperature of the peak pin in the core, or the maximum temperature of the average pin of the hottest subassembly. In the following, the protected loss of off-site power as a significant example of DBCs is presented.

(a) Protected loss of off-site power

This transient is initiated by the total loss of off-site power supply (station blackout). As a consequence, the forced circulation in the primary system is lost (pump coast down) with a simultaneous turbine and feedwater trip on the secondary side. The reactor scram is also a direct consequence of the complete loss of the off-site grid — the power supply to the electromagnet in the locking mechanism of the control rods is lost and they are promptly inserted into the core by buoyancy. The secondary circuits are isolated by the closure of the main steam isolation valve and the decay heat removal by the DHR-1 system (four isolation condenser loops are assumed to be in service) is promptly started through the opening of the triggering valve positioned below the isolation condenser.

The main results of the simulations performed with RELAP-5 are presented in Figs 80–82. Figure 80 shows a small increase of the initial peak cladding temperature (max. T_{clad} = 556°C) owing to a smooth core flow rate reduction in the initial phase of the transient. This is not significant from safety point of view. Moreover, as can be seen in Fig. 81, the core decay power is efficiently removed by the DHR-1 system during the whole transient phase. Before about 1800 s, all system temperatures are below the nominal values and there are no safety concerns (Fig. 82). After, the isolation condenser power overcomes the core power so that the lead temperatures decrease. They eventually approach the freezing temperature in about 2 h. A concern might be raised concerning the lead temperature at the wall of the steam generators, but as it takes about 2 h for the coolant in the steam generator to freeze, it is assumed that the operator has sufficient time to deactivate at least one subsystem of the DHR-1 system in order to prevent this. Coolant freezing in the primary cooling circuit should be avoided also in the long term, by appropriately selecting a core cooling strategy. Starting from the results obtained, one can conclude that in case of a total loss of off-site power supply, the core decay heat can be passively and safely removed indefinitely without the need for active energy sources. Due to the favourable characteristics of lead as a coolant (high thermal inertia, high boiling point) as well as the good natural convection characteristics of the plant design, no relevant, immediate safety issues were identified during the simulation of all DBC transients [70].

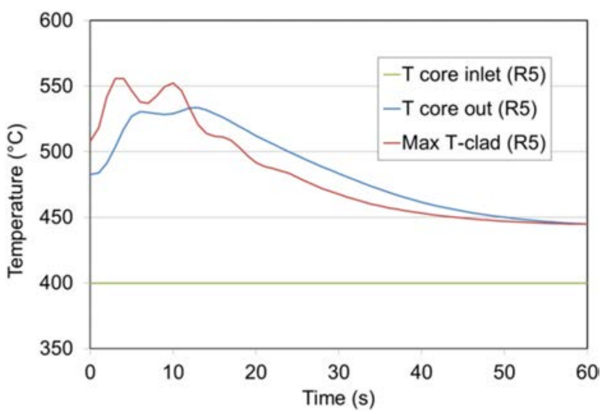

FIG. 80. Core inlet/outlet and maximum cladding temperatures (short term).

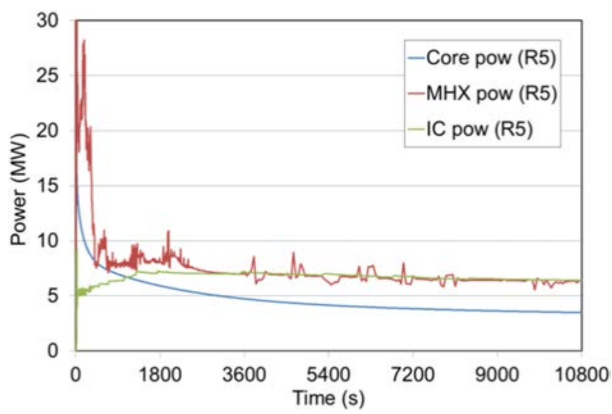

FIG. 81. Core, main heat exchanger (MHX) and isolation condenser (IC) powers (long term).

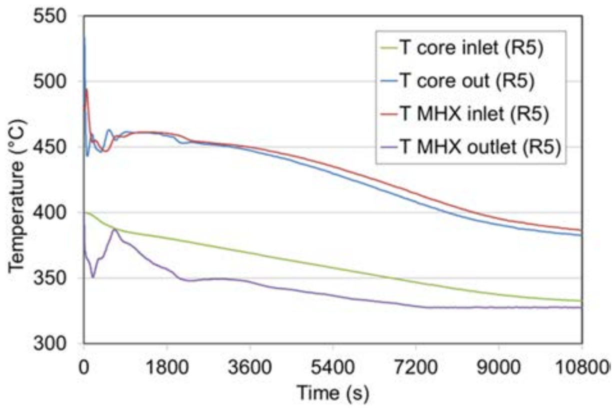

FIG. 82. Core inlet/outlet and main heat exchanger (MHX) inlet/outlet temperatures (long term).

6.3.3.3. Preliminary safety analysis — design extension conditions

For the ALFRED, a total of eight transients were selected as representative of all identified DEC transients reflecting a wide range of potential transient initiators [71]. Two of the most penalizing DEC transients are ULOF, simulated using the SIM-LFR code, and UTOP, analysed with the CATHARE code.

(a) Unprotected loss of flow

A ULOF transient is initiated by the loss of power supply to all primary pumps. The reactor scram is assumed to fail and then the core power is driven by reactivity feedbacks. The secondary heat transport system is assumed to be fully functional (forced convection mode).

Since the ALFRED primary system was specifically designed with the intention of being able to accommodate the ULOF transient without significant pin failures, the main aim is to investigate the cladding and fuel temperature response of the peak pin and the likelihood of peak pin cladding rupture during the flow undershoot. Core temperatures during the ULOF transient are primarily determined by the flow undershoot that must be expected during flow coast down before the asymptotic natural convection flow rate is reached.

In the case of the ALFRED, despite the halving time of <1 s assumed for the pumps' coast down, no relevant flow undershoot is expected in the initial phase of the transient (Fig. 83) and enhanced stable natural convection is established in the primary circuit at about 23% of the nominal value. Reduction of the coolant flow through the core leads to an initial peak in the outlet coolant temperature (around 750°C) at 17.8 s into the transient (Fig. 84). But later, owing to the negative reactivity feedback, the core power is reduced to about two thirds of the nominal value (Fig. 85).

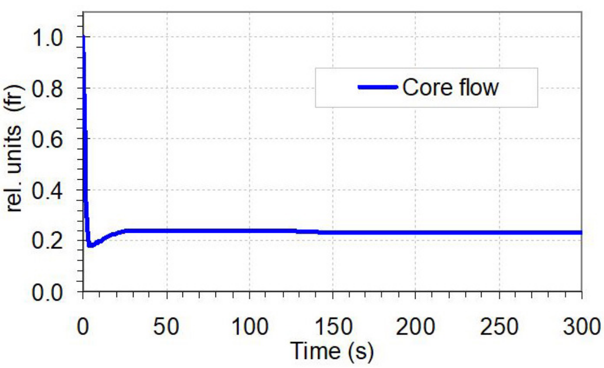

FIG. 83. Core flow rate.

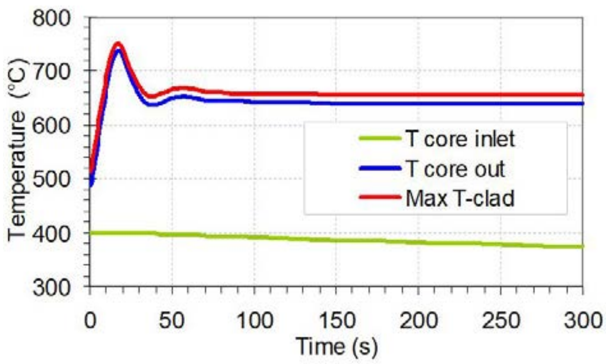

FIG. 84. System temperatures.

As a result, the maximum cladding temperature rather quickly stabilizes at around 650°C, with the cladding failure time for the peak pin stabilizing at about 10^8 s (about 3 years), as calculated by the SIM-LFR code (Fig. 86).

The ULOF transient in the ALFRED reactor can be easily accommodated since the maximum cladding temperature remains well below the creep rupture limit in the long term (maximum cladding temperature stabilizes below 700°C). Moreover, SIM-LFR code simulations show a very long minimum cladding failure time of roughly 160 000 s (about 2 d), under the minimum coolant flow conditions (initial flow undershoot) at both beginning of cycle and end of cycle conditions.

(b) Unprotected transient overpower

A UTOP transient is initiated by an unexpected positive reactivity insertion of 250 pcm in 10 s owing to some possible perturbations such as core compaction, steam generator tube rupture or fuel assembly flow blockage. The reactor scram is assumed to fail, while forced circulation is maintained in the secondary circuits with no control of the feedwater flow rate, which remains constant at its nominal value.

The simulations performed using the CATHARE code show (Fig. 87) that the reactivity insertion is mainly counterbalanced by the Doppler effect, which limits the net core reactivity excursion below 100 pcm and allows the other feedback mechanisms to play their role in reducing the reactivity and hence, the core power. Thus, the core power, which reaches a maximum of 735 MW in 10 s, starts to decrease progressively (Fig. 88), while the power removed by the secondary side increases due to lead temperature increase at the MHX inlet.

The simulations also show that at both beginning of cycle and end of cycle core conditions, the peak fuel pin cladding is not expected to fail since there is a large margin before the creep rupture limit (Fig. 89).

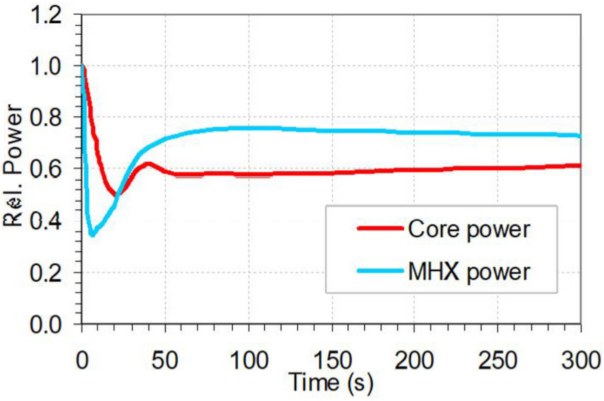

FIG. 85. Core and main heat exchanger (MHX) powers.

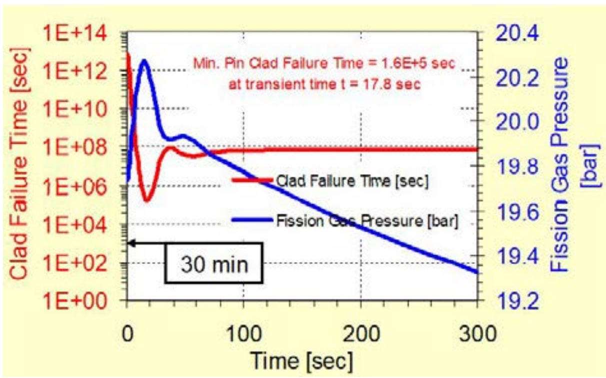

FIG. 86. Cladding failure time and peak pin fission gas pressure.

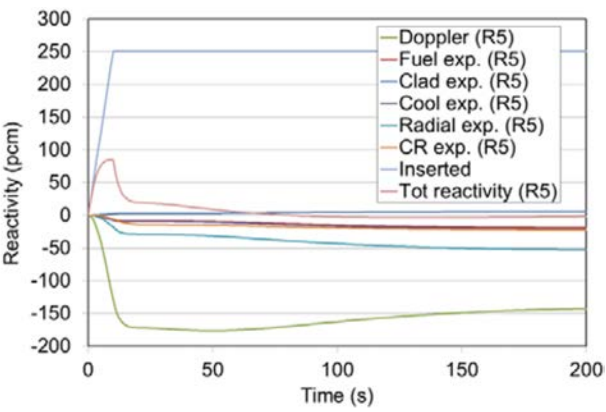

FIG. 87. Total reactivity and feedbacks.

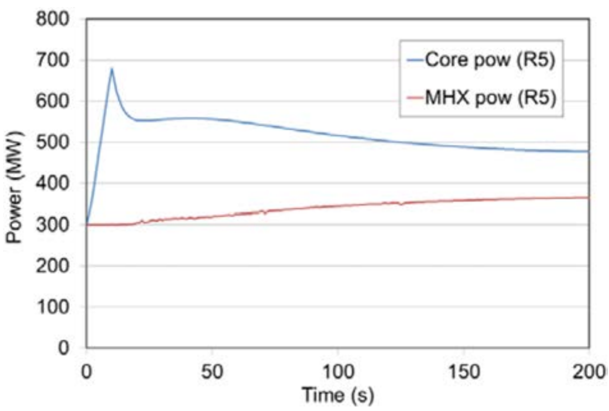

FIG. 88. Core and main heat exchanger (MHX) powers.

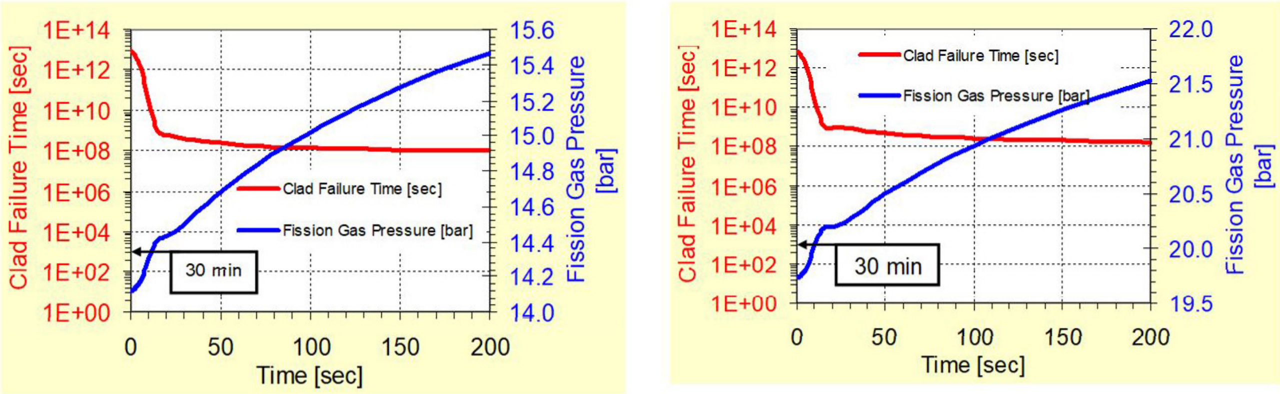

FIG. 89. Cladding failure time and peak pin fission gas pressure at beginning of cycle (left) and end of cycle (right) core conditions.

The more critical issue is the maximum fuel temperature, which is expected to exceed the MOX melting point in the initial phase of the transient because of the initial power excursion. However, all codes agree [71] that fuel melting is observed only locally in the peak fuel pins (for the few pellets around the core midplane in the innermost cylindrical shell of about 1 mm thickness out of 3.5 mm of the solid annulus). Therefore, extended fuel rod damage can be excluded, as well as the likelihood for this transient to evolve towards a severe accident.

6.3.3.4. Conclusions

All selected transients examined proved that the ALFRED plant can accommodate all investigated accidental DBC events [70]. The cladding and fuel temperatures are always well below the safety limits and no significant vessel wall temperature increase is predicted. Moreover, there is an extended time margin (grace time) of several hours for a possible manual operator intervention even under the worst accident conditions (potential of lead freezing in the long term in the case of uncontrolled decay heat removal by the DHR).

The analysis of all the DEC transients performed with the ALFRED [71] has demonstrated the very forgiving nature of this plant design when compared to other similar plant designs owing to an appropriate combination of the inherently large thermal inertia of the lead cooled primary system, the optimization of safety relevant control, safety systems and components, as well as the neutronic core characteristics that assure various reactivity feedback effects able to decrease the reactor power in the case of all DEC transients.

Moreover, the analyses performed show that all the features and the engineering solutions adopted in the design of the ALFRED ensure the safety of the plant in any operating conditions, even in the absence of PSS.

6.3.4. Transient analyses for the Japan Sodium Cooled Fast Reactor

The Japan Atomic Energy Agency has developed a large scale JSFR concept having specific design features and embedding innovative technologies able to simultaneously fulfil the requirements of a next generation plant: safety and reliability, cost competitiveness, sustainability and proliferation resistance.

6.3.4.1. Japan Sodium Cooled Fast Reactor main features

The JSFR will provide an electricity output of 1500 MW(e). Its economic competitiveness is due to an advanced design consisting of a simplified and compact reactor structure that integrates the IHX and the circulation pumps. It is a two loop configuration plant with a reduced piping layout (Fig. 90).

Moreover, efforts have been made in order to meet the safety requirements. These are reflected in the development and enhancement of a PSS and a recriticality-free core concept to protect against ATWS in DECs. A fully passive decay heat removal system with natural circulation is also introduced for DBEs and DECs. These efforts also envisaged the application of various innovative technologies concerning advanced elevated temperature structural design standards, steel with high strength at high temperatures and advanced seismic isolation.

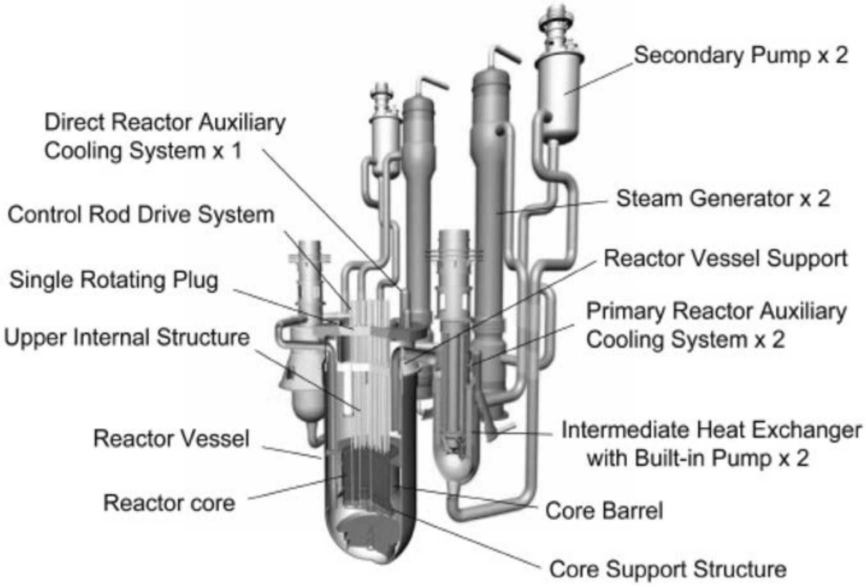

FIG. 90. Japan Sodium Cooled Fast Reactor [12].

6.3.4.2. Safety design

The JSFR safety design relies on the defence in depth principles and takes into consideration all the plant states: normal operation, anticipated operational occurrences, design basis accidents and DECs [72]. Safety functions such as the RSS and the decay heat removal system for core damage prevention are specified through a deterministic approach considering the postulated design basis accidents.

The JSFR design also includes features able to face severe accidents which are considered DECs. Besides defence in depth principles, a risk informed approach has been adopted in the JSFR design [73] for a better understanding of the balance of different levels of defence in depth.

6.3.4.3. Reactor shutdown function

The RSS has two independent subsystems (primary and backup systems) which are activated by the RPS. Each of them consists of control rods and their mechanisms and is designed to allow a rapid shutdown in order to prevent fuel failure during design basis accidents. The primary RSS has mechanical delatch devices with acceleration by gas pressure for insertion of the control rods, while the backup RSS has electromagnets for the delatching devices [73]. The control rods of the backup RSS are inserted by gravity.

In order to prevent core damage, especially during ATWS, a SASS for the backup RSS has been conceived. The main requirements considered in the selection of the most appropriate SASS were: effectiveness against all ATWS (e.g. loss of flow, loss of heat sink, overpower), enough negative reactivity, simple reset after an actuation and testability.

A Curie point electromagnet type SASS is considered the most appropriate for JSFR (Fig. 91) [12, 72]. The Curie point electromagnet SASS consists of an electromagnet and an armature that are parts of its magnetic circuit containing a temperature sensing alloy. The magnetic force is suddenly lost when the alloy is heated up to its Curie point by the hot coolant from the core. Then the armature delatches and drops together with the control rod into the reactor core. The Curie point SASS has a simple structure and its one dimensional movement reduces the uncertainties during ATWS.

Even if the JSFR core restraint structure impedes core deformation — which could be produced by heat, irradiation or an earthquake — additional measures have been considered which consist of a flexible joint structure (see Fig. 92) which is able to avoid the mechanical jamming of a control rod in case of a possible core deformation.

Moreover, taking into consideration that the passive actuation principle does not require the activation of the RPS [73], it has been concluded that it is not necessary to consider a common cause failure between the passive and active shutdown systems.

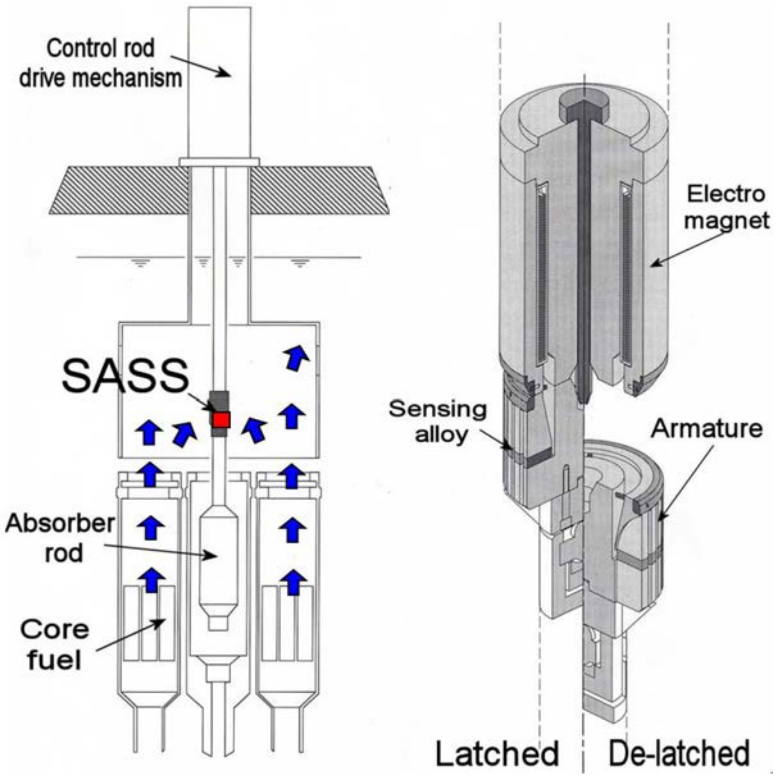

FIG. 91. Curie point concept in self-actuated shutdown systems (reproduced from Ref. [12]).

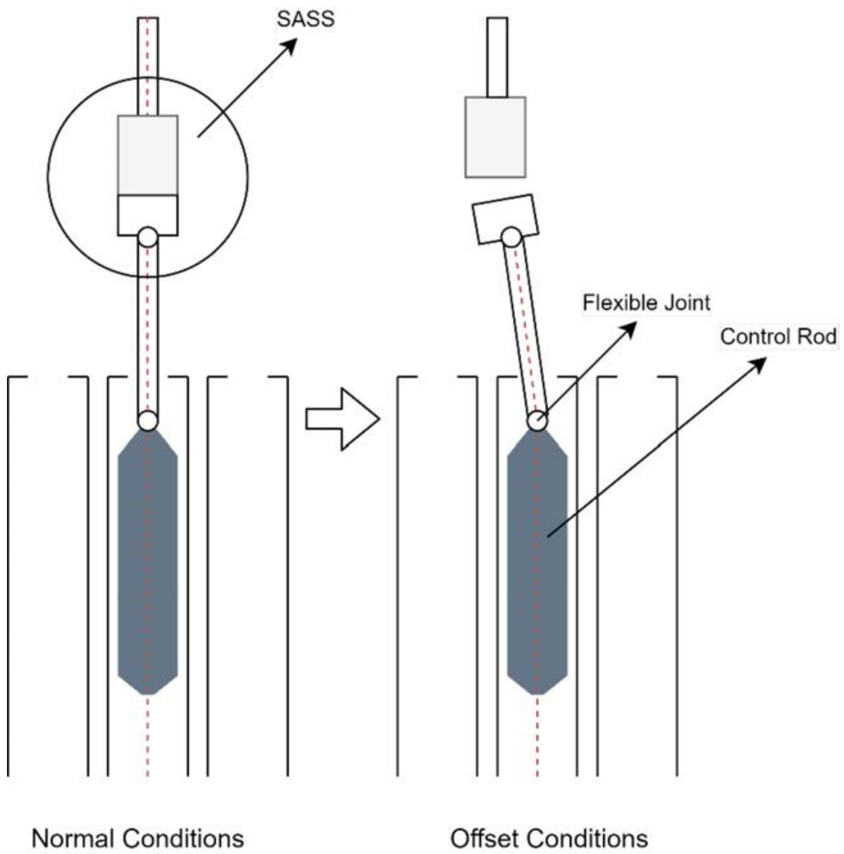

FIG. 92. A self-actuated shutdown system with a flexible joint (adapted from Ref. [73]).

6.3.4.4. Passive shutdown capability evaluation

Some transients have been investigated in order to confirm the operability of the SASS under an ATWS (e.g. ULOF, UTOP, ULOHS [12, 73]). Moreover, in various studies ULOF, UTOP and ULOHS are regarded as important ATWS leading to CDA [74]. However, there are differences among the transient timescales for these ATWSs. Thus, in a ULOF, the flow coasts down with a halving time of several seconds and the core disintegration could occur less than 10–20 s after the accident initiation. In the case of a UTOP, it takes several minutes to reach core disintegration due to the JSFR's control rod drive system which is designed so that reactivity rate increase is less than 0.5¢/s. For ULOHS, the typical time to reach core disintegration is also not less than several minutes. Thus, a ULOF event is considered the most penalizing among the three types of ATWS.

In the ULOF analysis, a flow coastdown of all primary pumps was assumed without a reactor trip from full power operation. The values of the parameters involved in the transients' evaluations have been established based on both the experiment and analysis performed during the Demonstration Fast Breeder Reactor design study in Japan [12].

Thus, the activation temperature of the SASS was set to 680°C in agreement with the conclusion drawn from the experiments performed in support of the selection of materials for the components. The coolant transport duration from the top of the neighbouring fuel assemblies around the backup control rod (BCR) with SASS to the temperature sensing alloy was assumed to be 1.0 s, which becomes longer in a ULOF owing to reduction of the coolant flow rate [12]. Based on further three dimensional computational fluid dynamics calculations [73], the transport time was set to 1.3 s and only five BCR insertions were assumed.

The time constant of the detachment is another parameter that was evaluated using computational fluid dynamics; that is the time difference between when the bulk coolant temperature around the SASS reaches the detachment temperature and when the SASS detaches. Design changes (depending on the individual BCR locations) were performed in order to improve the time constant; thus, their values are 3.4 s for the core centre BCR, and 1.0 s for the remaining four neighbouring rods [73]. With regard to the insertion time of the detached control rod, the necessary time for 85% insertion of a BCR is assumed to be 1.5 s for gravitational insertion, based on the rod insertion test data.

Figure 93 presents the behaviour of fuel, cladding and coolant temperatures during ULOF, where the halving time of the coolant flow rate was set to 6.5 s, leading to the most severe consequence among all the ATWSs.

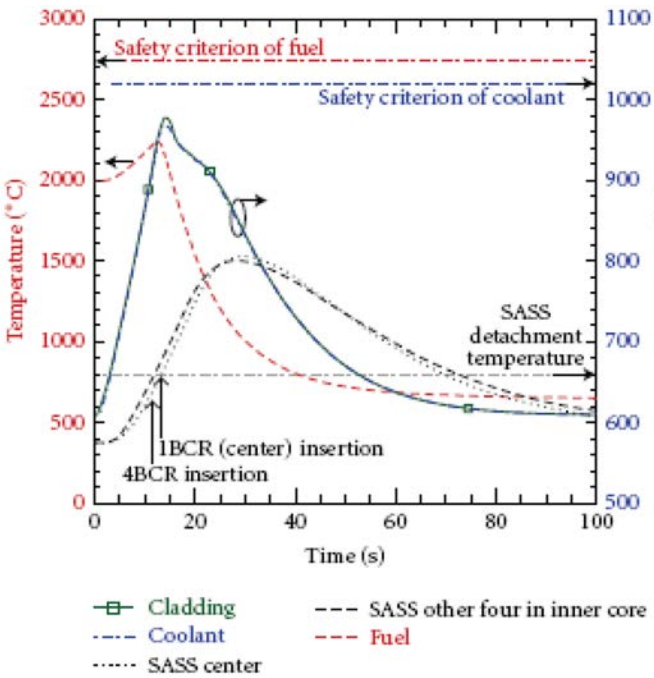

FIG. 93. Loss of flow without scram with passive shutdown [73].

The simulations showed [73] that the calculated coolant temperature around the SASS armature for the four BCRs in the inner peripheral positions reached the 660°C SASS detachment temperature and the delatch occurred 11.9 s after the transient onset. In case of the core centre BCR, the SASS detached the rod at 13.3 s. The computations showed that the maximum temperature reached by the fuel is 2248°C, while the maximum coolant temperature reached is 969°C. Both of those are less than the values imposed by the safety criteria.

In case of a ULOF, the conclusion [73] is that the SASS averted bulk coolant boiling, and core cooling could be maintained. Also, some UTOP and ULOHS calculations [12] showed that the SASS is able to prevent core damage during these ATWSs, which belong to the DEC category.

The JSFR concept design encompasses highly reliable systems as well as specific measures for both accident prevention and mitigation. These include the following [73]:

> "the SASS as the passive feature in addition to the two independent RSSs, the fully natural circulation DHRS [decay heat removal system] that can remove the decay heat without the SHTS [secondary heat transport system], the introduction of double-wall piping against sodium leaks, a core design limiting the sodium void worth, devisal of the special fuel assembly for attaining IVR [in-vessel retention] under ATWS, and the leak-tight containment. … The safety analyses and PSA [probabilistic safety assessment] revealed that the JSFR design fulfilled the safety criteria."

JSFR designers do not exclude further design improvements and intensive R&D efforts, which will be needed for its deployment in the future as a commercial reactor [73].

7. NEEDS FOR R&D AND QUALIFICATION

7.1. GENERAL REQUIREMENTS FOR QUALIFICATION OF SELF-ACTUATED SHUTDOWN SYSTEMS

As defined in Ref. [75], the purpose of the qualification of any safety system is to provide evidence that the equipment fulfils its functional requirements on demand and according to the environmental conditions representative of operation under normal and accident operation. SASS qualification is based on out-of-pile and in-pile testing as well as numerical studies modelling the behaviour in situations that cannot be reproduced in experimental installations. In practice, the qualification programme is carried out between manufacturing and installation of the system on the reactor. Once installed on the reactor, the qualification is completed by performance tests of the SASS.

For advanced reactors, a SASS concept satisfies the need to reinforce core meltdown prevention in DECs. SASSs are foreseen to operate in case of a degraded environment (e.g. loss of power supply or external hazard) that has jeopardized the operation of active protection systems. As a consequence, they are considered a supplementary defence line to prevent core meltdown in case of emergency shutdown malfunction.[11] Specifically, for fast reactors, the design of SASS reactivity worth may take into account potential local degradation of the core, like partial melting of a few subassemblies. This is because the SASS might be activated in case of a local subassembly fault. Such verification of the design can be achieved on the basis of sensitivity computations.

Given these definitions, it is important to outline here several specific SASS features which have an important influence on the qualification programme:

— SASSs are directly triggered by the environmental conditions linked to the physical state of the reactor (no need for an intermediate system to process the information).
— In the common approach used by the designers of advanced reactors, SASSs are required to operate only in DECs (in case of an unprotected transient).
— The system should active within a short delay.

[11] Not applicable for self-controlling cores.

For the SASS, the following functional requirements should be used to define the types of testing and measurement to be performed within the qualification programme [18]:

- The system is triggered directly by the variation of physical parameters associated with core state or with the primary coolant.
- The antireactivity worth of the system, whether it is brought on by insertion of neutron absorbing materials or by increased neutron leakage,[12] is sufficient to shut down the reactor with an adequate safety margin in terms of delay time and temperatures.
- The total response time of the system[13] is compatible with maintaining the fuel, cladding and structural material integrity.
- The system is able to operate under mechanical vibrations and relative displacements between its supporting structures (usually linked to the main vessel closing slab) and the core.

It may be noted that the above items, except the first one, are also relevant for the qualification of the main protection systems (active systems). The list above can be achieved by design requirements which must be taken into account in the qualification:

- The position of the neutron absorbing part is monitored.
- The standard handling machine can unload the system from the core.
- Spurious actuation of the SASS during normal operation is avoided.
- If the SASS is used in a safety demonstration to rule out some of the core meltdown scenarios, its design and qualification provisions are adequately defined to obtain the required reliability.

7.2. GENERAL SCOPE AND CONTENT OF A SELF-ACTUATED SHUTDOWN SYSTEM QUALIFICATION PROGRAMME

SASS qualification relies on a programme established by the designer. The definition and content of this programme should follow the same principles as those applied to the qualification of active protection systems: definition of the qualification tests on the basis of the functional requirements to be fulfilled by the equipment or the system and definition of qualification profiles. The qualification programme covers, in principle, all the DECs in which the SASS needs to operate.

The qualification programme addresses successively: elementary components, the whole system and the system integrated in its environment (taking into account potential interfaces with other systems and functions). It also takes into account inherent physical properties of the SASS[14] such as the material's thermal expansion coefficient, buoyancy, melting point and magnetic permeability.

The tests prescribed by the programme are based on qualification profiles which are determined upon the results of accident studies. The profiles derive partly from the study of the unprotected transients, which provides in particular the evolution of the physical parameter which triggers the SASS. In practice, two families of unprotected transients are considered: loss of core cooling (ULOF, ULOHS, unprotected loss of off-site power) and slow overpower transients. Environmental conditions resulting from combinations of failures that may be considered in DECs and can have an impact on the definition of the qualification profiles have to be considered (combination of loss of flow accidents with a vessel leak is an example). Moreover, it is to be noted that accident studies have to integrate the effects of external hazards (situations initiated by or combined with hazards). As an example, the qualification programme also includes seismic testing.

Finally, uncertainties on functional parameters are of primary importance for passive systems. As a consequence, qualification programmes should be tailored to ensure that the uncertainties considered in the SASS

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[12] For example, when using a GEM.

[13] The total response time of the system is composed of: (a) the time lag between reaching the safety criteria (e.g. maximum cladding temperature) and the triggering of the passive system (absorber rod dropping, plug melting, etc.); (b) the elapsed time between the triggering and the effective insertion of antireactivity (rod falling time, liquid absorber injection time, etc.).

[14] Indeed, one can observe that it is also an important point for active shutdown systems, in particular for the mastering of the thermal expansion of the CRDL during startup phases.

design were correctly evaluated. Indeed, uncertainties have to be set to conservative values during the design phase because of unknown environmental parameters, sensor accuracy and manufacturing tolerances. So, the mechanistic and statistical analyses of the data acquired during the qualification tests give essential feedback to determine the uncertainties more accurately (fall time of absorbers is an example). Two IAEA studies have some related qualification studies [76, 77].

7.2.1. Out-of-pile qualification

The major part of SASS qualification is expected to be done out-of-pile. The main purpose of this step is to validate the overall concept by testing the main components and the complete system in tests rigs. In this frame, the objectives are the following [78]:

(a) Validating component designs;
(b) Verifying compatibility of the SASS materials with the coolant;
(c) Characterizing the physical phenomena on which operation of the SASS is based;
(d) Evaluating the time constant of actuation.

Corrosion tests are necessary to satisfy item (b), and for metallic coolants, the influence of coolant purity (possibility of oxide deposits, etc.) needs to be examined.

It should be emphasized that qualification goes along with the fine characterization of the physical phenomena that trigger the SASS, together with the evaluation of the uncertainties on the piloting physical parameters. As an example, it is not the pump regime — easily correlated with the rotation speed of the pump motor — that actuates the hydraulically suspended rods, but the induced hydraulic forces on the absorber rod subassembly, which are more difficult to measure. Specifically for passive systems that are triggered by the coolant temperature, constant time of actuation is defined as the delay from the time when the sodium reaches the safety threshold up to the full insertion of the absorber.

In some cases, as triggering parameters and actuation thresholds cannot be directly measured during qualification tests, calculations are done to confirm the fulfilment of functional requirements. Measurements are then used to qualify numerical modelling of the SASS in its environment. For systems based on Curie point, tests are recommended to evaluate the uncertainties in the magnetic forces which are induced by potential pollution of the primary coolant (deposits in the magnetic gap between static and mobile parts of the system).

The challenge appears therefore to be in reproducing the physical environment of the system in the test rigs, because the operational functions tested for SASSs are driven by physical phenomena conditioned themselves by the physical environment (e.g. hydraulic forces, fluid temperature). The influence of scale effects are then to be carefully examined, when using a down scaled mock-up of the system (shutdown systems for SFRs are usually large structures with great elongation).

Out-of-pile qualification also includes seismic integrity tests which are not specific to passive systems.

7.2.2. In-pile qualification

Three main objectives are achieved through SASS in-pile testing: the measurement of the reactivity worth, the validation of the triggering threshold and the evaluation of the impact of the irradiative environment. For this purpose, a set of qualification procedures are performed in experimental irradiation devices (zero power mock-up and experimental reactors). It should be noted that the representativeness of such testing may be a real challenge for passive systems because of their great sensitivity to the physical conditions of the reactor.

Hence, a major part of the qualification programme of the SASS is shared with that of active shutdown systems, but it is of particular importance to thoroughly characterize the environmental conditions through the in-pile characterization of the physical parameters of the coolant (pressure, flow, temperature).

Irradiation by neutrons and photons induces the following three phenomena that have to be characterized during in-pile qualification:

— Material damage at the atomic scale (atom displacements);
— Real friction and hydrodynamic forces;

— Heat production in the structures of the device (owing to interaction with photons and neutrons).

Radiation induced ageing also has to be evaluated, taking into account the following:

— The location of the system and the various types of subassemblies that can be loaded in the vicinity (variable shielding effect, etc.);
— The periodicity foreseen for the replacement of the sensitive elements of the system.

Because SASSs are intended to operate in DECs, their qualification may also take into account accident scenarios involving an abnormal irradiative environment. The corresponding qualification profile needs to be characterized based on the results of the associated numerical safety studies.

7.2.2.1. Qualification programme and reliability studies

A viable path towards the implementation of the research efforts is delineated: among the most relevant issues to be addressed in an R&D programme is assessing the performance of a PSS to achieve a preliminary figure of merit.

Evaluation of the degree of inherent safety and reliability of innovative passive systems may be a challenge because of lack of data. Thus, the qualification programme of passive systems has to be tailored to provide the statistical data that are needed to feed the reliability studies.

With regard to the reliability issue, failure probability of these passive features could be preliminarily estimated by considering uncertainty factors of various design parameters such as actuation temperatures and effective reactivity insertions.

Among the various concepts of PSSs which could be implemented on a fast reactor, SASSs obviously benefit from favourable characteristics. One is that a SASS is usually triggered by the overrun of a single threshold related either to primary coolant flow or temperature. And once a SASS has been activated, it does not have to adapt to further evolution of the physical environment, unlike passive cooling systems (for example ‘one shot’ systems). These characteristics tend to simplify their qualification.

Although many elements of the qualification programme are also found in those for active shutdown systems, the main difference lies in the fact that the systems are directly triggered by the environmental conditions, which are sometimes difficult to characterize and reproduce during the qualification tests. In particular, specific environmental conditions have to be considered to be representative of DECs to determine the qualification profiles (partial degradation of barriers, cumulated abnormal events or hazards, etc.). Therefore, careful attention needs to be given to the qualification of the modelling of the reactor behaviour in accident conditions.

Moreover, uncertainties related to physical parameters that trigger the SASS are of particular importance. The qualification programme should allow for the exploration of the sensitivity of the SASS performance regarding these uncertainties, mainly during out-of-pile qualification.

Finally, in-pile qualification should validate the SASS for operation in an irradiative environment, including conditions representative of plant postulated accidents.

It is to be mentioned that SASSs should be liable for periodic in service examination and testing, similar to active shutdown systems. The purpose is to ensure the availability on demand and the constant performance of the system throughout its expected service life (most of the SASSs are replaceable). On that point, several types of SASS, like the neutron injection modules described in Ref. [79], are not reversible systems and may not be periodically tested. For these devices, the qualification programme should be adequate to be able to justify that the devices loaded in the core are identical to those tested outside (as a matter of quality assurance) and that the environmental conditions of the reactor should not modify their performance throughout their service life.

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Annex

DESCRIPTION OF PARTICIPATING ORGANIZATIONS

A–1. BELGIUM

As a research centre dealing with peaceful applications of radioactivity, the Belgian Nuclear Research Centre (SCK•CEN) is an indispensable part of Belgian society. SCK•CEN performs forward looking research and develops sustainable technology for socially valuable purposes. With more than 750 employees, SCK•CEN is one of the largest research centres in Belgium. Throughout all of its work, there are three research topics that receive particular attention:

- Safety of nuclear installations;
- Well thought out management of radioactive waste;
- Human and environmental protection against ionizing radiation.

Thanks to its thorough experience in the field of nuclear science and technology, its innovative research and the availability of large and unique nuclear installations, SCK•CEN is not only a renowned nuclear research institution, but also an important partner for nuclear education and training in Belgium as well as internationally. Within the SCK•CEN Academy, more than 60 years of nuclear expertise and experience gained from its different research projects is collected and transferred.

SCK•CEN, in Mol has been working for several years on the design of a multipurpose irradiation facility in order to replace the ageing BR2 reactor, a multifunctional materials testing reactor in operation since 1962.

Multi-purpose Hybrid Research Reactor for High-tech Applications (MYRRHA), a flexible, fast spectrum research reactor (50–100 MW(th)) is conceived as an accelerator driven system (ADS), able to operate in subcritical and critical modes. It contains a 600 MeV proton accelerator, a spallation target and a multiplying core with mixed oxide fuel, cooled by liquid Pb–Bi.

Following are some MYRRHA applications:

- Sustainable fission energy: Demonstrate the physics and technology of an accelerator driven system for transmuting long lived radioactive waste.
- Sustainable energy: Development of fast spectrum reactor and fusion technology.
- Enabling technologies for a small modular reactor.
- Health care: Production of radioisotopes for nuclear medicine.
- Science: Fundamental research for the generation of new expertise in various fields.

A–2. GERMANY

The Karlsruhe Institute of Technology (KIT) is a higher education and research organization with about 9 500 employees, 25 000 students and a total annual budget of about 860 million euros. It combines the missions of a university of the state of Baden-Wuerttemberg and of a large scale research institution, the Helmholtz Association. Within these missions, KIT operates along the three strategic fields of action of research, teaching and innovation. In establishing innovative research structures, KIT is pursuing joint strategies and visions. KIT is devoted to top research and excellent academic education, as well as to being a prominent location of academic life, lifelong learning, comprehensive advanced training, exchange of know-how and sustainable innovation culture.

KIT's research programme for nuclear waste management, safety and radiation (whose acronym is NUSAFE) has a long tradition, is widely recognized and represents an integral part of national research, addressing new challenges by providing core competences on the highest level of science and technology internationally, regarding nuclear safety and waste management research. The scientific research work is carried out in several academic institutes, whereas the programme management is responsible for the strategy and scientific and administrative coordination work. KIT participated in a number of previous and ongoing European Union projects relating to safety in liquid metal and gas cooled fast reactors.

The NUSAFE programme within the Research Field Energy programme of the Helmholtz Association continues to study scientific and technical aspects of the safety of nuclear waste management and of nuclear reactors. This work constitutes research in the interest of society and, for this reason, must be preserved for a long time to come. Within the NUSAFE programme, scientists from several academic institutes work on four research topics:

— Safety of nuclear waste management;
— Safety of nuclear reactors;
— Radiation protection;
— Decommissioning techniques.

KIT operates several liquid metal experimental facilities. In the Karlsruhe Liquid Metal Laboratory (KALLA), several experimental loops have been established, such as THESYS (Technologies of Heavy Metal Systems), THEADES (Thermal-Hydraulics and ADS Design) and CORRIDA (Corrosion in Dynamic Lead Alloys). The Karlsruhe Sodium Laboratory (KASOLA) facility comprises a large scale facility with flow rates up to 150 m^3/h and temperatures up to 820 K, three small scale facilities for material testing purposes (SOLTEC) up to 1000 K, a small scale DITEFA gallium–indium–tin facility for high precision measurements at room temperature, as well as the ATEFA facility with a ceramic test section to reach a temperature of 1200 K. The facilities are funded by Helmholtz NUSAFE programme, the Helmholtz Energy Materials Characterization Platform (HEMCP) and the Helmholtz Alliance LIMTECH (Liquid Metal Technologies). The MOCKA (German Experimental Facility for Melt Coolability and Concrete Interaction Studies) and LIVE (Large Scale Experiments on In-Vessel Melt Relocation and Retention) test facilities are part of the European infrastructure known as SAFEST.

For fast reactor and ADS safety studies under design basis conditions, the SIM-family codes and the SAS-SFR code are employed. For studies under design extension conditions, the SIMMER family of codes is used.

A–3. INDIA

The Indira Gandhi Centre for Atomic Research (IGCAR) was established in 1971 under the Department of Atomic Energy, Government of India. The centre is engaged in a broad based multidisciplinary programme of scientific research and advanced engineering directed towards the development of fast breeder reactor technology and fuel cycle technologies.

This comprises a fast breeder test reactor based on a unique plutonium–uranium–carbide fuel — the first of its kind in the world — and the KAMINI reactor, the only operating reactor in the world using ^{233}U fuel.

IGCAR has developed the design of the 500 MW(e) Prototype Fast Breeder Reactor (PFBR) based on work in areas such as R&D, engineering development and manufacturing technology development. Performance qualifications of major reactor systems and components have also been undertaken and successfully completed. Commissioning of the PFBR is in progress.

India is planning to construct a series of fast breeder reactors in the future. IGCAR is currently working on the design of future fast breeder reactors that will have enhanced safety and improved economics. IGCAR is also working on metallic fuel development for long term deployment and focuses on allied science and associated technologies.

A–4. ITALY

The Italian National Agency for New Technologies, Energy and Sustainable Economic Development (ENEA) is a public research institution operating in the fields of energy, environment and new technologies to support competitiveness and sustainable development.

The ENEA conducts scientific research and technology development activities that draw on a wide range of expertise, advanced facilities and tools located at its 11 research centres, operating in support of ENEA programmes and the national productive system. The staff — about 2600 persons — is for the most part made up of researchers engaged in R&D activities.

The ENEA is member of the European Sustainable Nuclear Energy Technology Platform. The ENEA also participates in several research projects promoted within the European Atomic Energy Community (Euratom) Framework Programmes, as well as in international committees and working groups of the Nuclear Energy Agency of the Organisation for Economic Co-operation and Development (OECD/NEA) and IAEA in charge of developing strategies and carrying out studies, in particular in the field of innovative fast reactors and advanced nuclear fuel cycles.

The ENEA has been addressing the very important topic of passive systems in the context of the development of advanced reactors (e.g. Gen III+ and Gen IV), with focus mainly on safety and reliability aspects.

In particular, the ENEA aims to further the analysis relative to the implementation of reactor designs with passive systems for safety improvement, including the development of innovative shutdown devices by passive means.

A–5. ROMANIA

The Institute for Nuclear Research (RATEN-ICN) is an R&D centre in Romania established in 1971 to provide scientific and technical support to the national nuclear energy programme. Its activity covers the main technical areas involved in the nuclear field: reactor physics, fuel behaviour, thermohydraulics, accident analysis and nuclear safety assessment, reactor component irradiation tests and examination, plant material investigations, out-of-pile testing, radioactive waste and spent fuel management, radioprotection and environmental impact. The R&D activities are based on a modern and complex infrastructure consisting of two TRIGA reactors (a steady state 14 MW and a pulsed one), postirradiation laboratories, radioactive waste treatment and a conditioning plant, nuclear fuel fabrication and testing, nuclear materials testing, radiochemical laboratories, out-of-pile testing laboratory and other facilities.

RATEN-ICN is member of the European Sustainable Nuclear Energy Technology Platform and the Implementing Geological Disposal of Radioactive Waste Technology Platform, the European Sustainable Nuclear Industrial Initiative, the European Energy Research Alliance and more recently of the OECD/NEA, and it has been involved in many Euratom projects covering a broad range of nuclear energy topics (ESNII+, LEADER, MATTER, SEARCH, STYLE, MATISSE, ARCADIA, CAST, ASAMPSA_E, MARISA, MACSIMA, PLATENSO, EAGLE, NEWLANCER, ELSY, etc.).

Since 2007, RATEN-ICN has been deeply involved in the design and development of lead cooled fast reactor concepts, and it is a founding member of the Fostering ALFRED Construction (FALCON) consortium devoted to implementation of the Advanced Lead Fast Reactor European Demonstrator (ALFRED). According to the FALCON agreement, RATEN-ICN is the reference option for the ALFRED siting.

More recently, in February 2017, by Governmental Decision No. 81, the ALFRED was included in the National Programme for Research, Development and Innovation — PNCDI III as a major European and international project in Romania. This programme supports investment in large infrastructure as well as other types of science based partnerships oriented towards the commercialization of scientific and technological ideas.

Romania confirms its option to host the ALFRED and it continues to contribute to its development under the strategic objective (included in the RATEN Strategy for 2015–2025) devoted to studies and developments in the field of heavy liquid metal technology with a specific focus on its application to lead cooled innovative nuclear fission systems and to ALFRED implementation.

A–6. SLOVAKIA

The Slovak University of Technology in Bratislava (STU) is a modern educational and scientific institution. Since its foundation in 1937, more than 145 000 students have graduated from it. On average, 17 000 students study at the STU every year. The STU is a public university and offers education mainly in technical, technological, technical economic, technical information and technical artistic fields of study using modern methods of education, laboratories and practical training. It is aimed at the areas of study with stable opportunities for students employment.

The Institute of Nuclear and Physical Engineering at STU is responsible for university education in the area of nuclear and physical engineering. Through teaching, scientific and R&D activities, the institute is active in the fields of general physics and mathematical physics, physics of condensed matter and acoustics, nuclear and subnuclear physics, material science, environmental engineering, electrotechnology and materials, nuclear power engineering and technology, and biomedical and physical engineering. The institute's staff consists of 12 professors in physics, nuclear power engineering and material science. There are 50 employees in total (scientific or research workers) and about 20 PhD students.

Priorities in energy R&D in Slovakia follow the objectives of the current national energy policy formulated in 2014. Among others, two priorities are defined: to use nuclear energy as a zero carbon source of electricity and to increase the safety and reliability of nuclear powerplants. One of the most important R&D activities addressing these priorities is the Allegro project — a nuclear energy cooperation project between the Czech Republic, France, Hungary and Slovakia. The Allegro project is a research project focused on R&D for the cooling system for a prototype of a fast breeder reactor cooled with inert helium gas. The project originated as a joint proposal from three Central European organizations (the Czech company ÚJV Řež, the Hungarian Academy of Sciences and the Slovak company VUJE) in negotiations with France's Atomic Energy Commission for the construction of this prototype in either the Czech Republic, Hungary or Slovakia within the the European Strategic Energy Technology Plan (SET-Plan) as an initiative for sustainable nuclear energy. Currently, the preparatory phase of the project is under way, with the Hungarian Academy of Sciences, ÚJV and VUJE having already committed to cooperate on its preparation until a decision is made as to the country in which the prototype will be built. The Slovak University of Technology in Bratislava and Slovak Academy of Sciences are also involved in R&D projects related to core neutronic studies and development of prospective structural materials suitable for GFR technology. Since Slovakia joined the GFR consortium, several bachelor, diploma and PhD theses were defended and number of small projects related to GFR core neutronic studies were supported by the Government's grant institutions. Construction of Allegro is expected to begin in 2020 and operation in 2025. The industrial demonstrator will follow it.

A–7. SWEDEN

Fast reactor related research in Sweden is carried out at three major universities: the Royal Institute of Technology (KTH), Uppsala University and Chalmers University of Technology, as well as by the private reactor developer LeadCold Reactors.

KTH is a technology focused university with more than 12 000 full time students, 2 000 PhD students and approximately 3 700 full time employees. At KTH, fast reactor research is undertaken within the Divisions of Nuclear Power Safety, Nuclear Reactor Technology, Reactor Physics and Technology, and Material Sciences, with a combined nuclear power focused staff of about 40 people. The Division of Nuclear Power Safety aims to promote the safe use of nuclear energy in Sweden and worldwide, with a specific focus on multiphase thermofluid science and computational safety analysis, at the interface of fundamentals and applications. The Division of Reactor Physics and Technology and the Division of Nuclear Reactor Technology conduct research that aims to improve the performance and safety of current and future nuclear power plants through a combination of numerical modelling and experimental investigations. The current research relating to fast reactors focuses on the following areas:

- Development and optimization of advanced Monte Carlo methods;
- Nuclear fuel cycle modelling;
- Design and safety analysis of lead cooled reactors;
- Science of radiation damage in nuclear steels;
- Advanced nuclear fuel development (mixed oxides, nitrides and silicides).

In addition, the Division of Material Sciences has helped to develop novel steels for application in lead cooled fast reactors.

Uppsala University is a comprehensive research intensive university with 42 000 students and 7 000 employees. The Uppsala University Division of Applied Nuclear Physics conducts research in the areas of nuclear reactions (for applications like nuclear energy, cancer therapy or transmutation of nuclear waste), nuclear fuel diagnostics and safeguards (encapsulation of spent nuclear fuel, the future needs of nuclear power plants, nuclear safeguards and non-proliferation issues), neutron diagnostics for fusion energy (studying fusion as a possible future energy source in present day devices and for the ITER international experimental reactor) and interaction of high velocity ions in various materials (with applications ranging from archaeology and medicine to the ageing of materials in nuclear reactors).

Chalmers is a technology focused university with 10 300 full time students and 3 100 employees. At Chalmers, fast reactor related research is mainly carried out in the Divisions of Nuclear Engineering and Nuclear Chemistry. The research at the Division of Nuclear Engineering focuses on development of coupled neutronic–thermohydraulic tools for noise analysis in nuclear reactors, analysis of design basis accidents and severe accidents, and development of efficient schemes to link neutron kinetic codes, system codes and computational fluid dynamics codes. At the Division of Nuclear Chemistry, research is focused on new innovative fuel cycles, such as partitioning and transmutation with subsequent manufacture of new types of nuclear fuels. Other applications related to the nuclear power production are studies of pollution prevention in severe reactor accidents, reactor water chemistry and the chemistry of the disposal of nuclear waste.

LeadCold is a new Swedish fast nuclear reactor company which is developing the Swedish Advanced Lead Reactor (SEALER) line of reactors. SEALER is a lead cooled reactor designed with the smallest possible core that can achieve criticality in a fast spectrum using 19.9% enriched uranium oxide fuel. The rate of electricity production may vary between 3 and 10 MW, leading to a core life between 10 and 30 full power years (at 90% availability). The reactor is designed to maintain a maximum temperature of the lead coolant below 450°C, making corrosion of fuel cladding and structural materials a manageable phenomenon, even over a lifespan of several decades.

A–8. UNITED STATES OF AMERICA

The United States Nuclear Regulatory Commission (NRC) was created as an independent agency by Congress in 1974 to ensure the safe use of radioactive materials for beneficial civilian purposes while protecting people and the environment. The NRC regulates commercial nuclear power plants and other uses of nuclear materials in the United States of America (such as in nuclear medicine) through licensing, inspection and enforcement of its requirements.

With potential near term applicant engagement involving GEN IV reactor technologies, including fast reactor concepts, the NRC participates in various international working groups or projects to remain abreast of current technologies, safety philosophy and licensing considerations in other countries. Lessons learned from other IAEA participants can aid the NRC staff review process and aid in early identification of any potential GEN IV policy or licensing issues, thereby supporting an efficient review process.

ABBREVIATIONS

ADS	accelerator driven system
ALFRED	Advanced Lead Fast Reactor European Demonstrator
ARC	autonomous reactivity control
ASD	additional shutdown device
ATWS	anticipated transient without scram
BCR	backup control rod
CDA	core disruptive accident
CPM	Curie point magnet
CRA	control rod assembly
CRDL	control rod driveline
CREED	control rod enhanced expansion device
DBC	design basis condition
DBE	design basis event
DEC	design extension condition
DHR	decay heat removal system
EFR	European fast reactor
ELFR	European Lead Fast Reactor
ELSY	European Lead-cooled System
ENEA	Italian National Agency for New Technologies, Energy and Sustainable Economic Development
EOEC	end of equilibrium cycle
Euratom	European Atomic Energy Community
FAST	floating absorber for safety at transient
FBTR	Fast Breeder Test Reactor
FFTF	fast flux test facility
GEM	gas expansion module
GFR	gas cooled fast reactor
HLMFR	heavy liquid metal cooled fast reactor
IGCAR	Indira Gandhi Centre for Atomic Research
IHX	intermediate heat exchanger
JSFR	Japan Sodium Cooled Fast Reactor
KIT	Karlsruhe Institute of Technology
LBE	lead–bismuth eutectic
LCPS	lyophobic capillary porous systems
LEM	lithium expansion module
LIM	lithium injection module
LMFR	liquid metal cooled fast reactor
LOCA	loss of coolant accident
LWR	light water reactor
MHX	main heat exchanger

MOX	mixed oxide
MYRRHA	Multi-purpose Hybrid Research Reactor for High-tech Applications
OECD/NEA	Nuclear Energy Agency of the Organisation for Economic Co-operation and Development
PFBR	Prototype Fast Breeder Reactor
PGSFR	Prototype Generation IV Sodium Cooled Fast Reactor
PRISM	Power Reactor Innovative Small Module
PSP	primary sodium pump
PSS	passive shutdown system
RATEN-ICN	Institute for Nuclear Research
RPS	reactor protection system
R&D	research and development
RSS	reactor shutdown system
SAFE	static absorber feedback equipment
SASS	self-actuated shutdown system
SFR	sodium cooled fast reactor
SHRT	shutdown heat removal test
SSP	secondary sodium pump
STU	Slovak University of Technology in Bratislava
TWG-FR	Technical Working Group on Fast Reactors
TWR	travelling wave reactor
ULOF	unprotected loss of flow
ULOHS	unprotected loss of heat sink
UTOP	unprotected transient overpower

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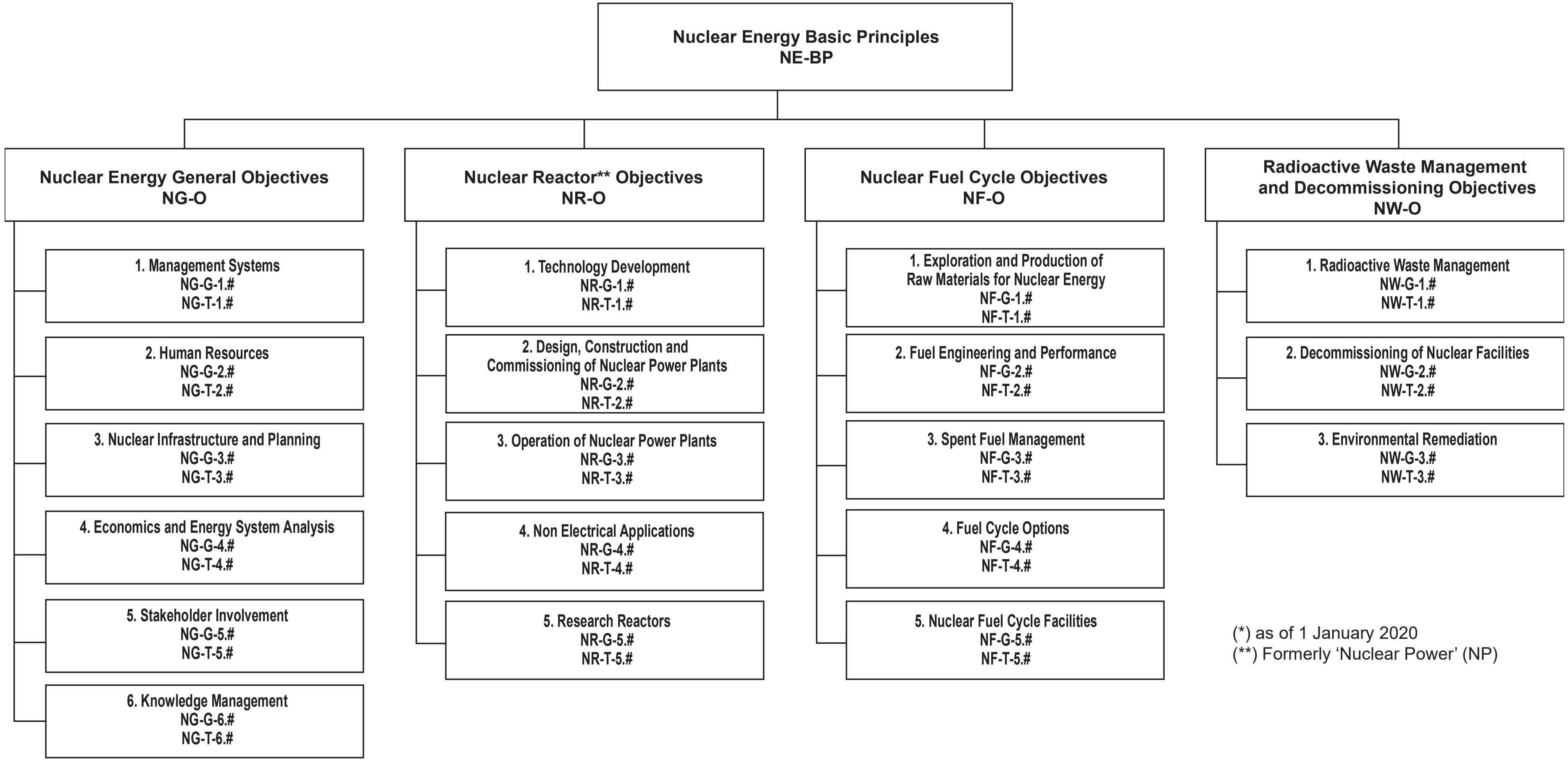

(*) as of 1 January 2020
(**) Formerly 'Nuclear Power' (NP)

Key

BP: Basic Principles
O: Objectives
G: Guides and Methodologies
T: Technical Reports
Nos 1–6: Topic designations
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Examples

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