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Fast Reactor Core Design

Module 5 : Safety Assessment

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Lectures on Fast Reactor Core Design

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5.0 INTRODUCTION

Safety assessment is an integral element of nuclear reactor design. Safety is of paramount importance in addition to economic objectives. Though the safety assessment has several common aspects with that of water cooled reactors, fast reactors has many unique features which have to be factored into the safety assessment. The safety principle demands defence in depth approach to be followed in the design. In this section, general safety features of fast reactors, event categorisation, plant protection system are covered. Events connected with subassembly are focussed and an example safety analysis case is presented in detail.

5.1 ISSUES SPECIFIC TO FAST REACTORS

With respect to the general safety principles, defence in depth requirements, safety functions, radiological dose limits to plant personnel and public, site boundary radiological requirements, emergency preparedness etc., the safety requirements are same for both thermal as well as fast reactors. However, the nuclear steam supply systems (NSSS) of fast reactors differ from thermal reactors basically on several counts. To name a few, the major differences are given below.

- Reactor core not in most reactive configuration
- Possibility of either positive or small negative coolant void reactivity of core
- High burnup
- High decay heat content
- Liquid metal sodium coolant, absence of moderator
- Absence of pressurised coolant system and primary boundary
- Components subjected to very high neutron dose with associated large radiation damage
- Large size thin shell structures subjected to high temperature
- Additional failure modes such as those due to high temperature effects like creep & fatigue in additional to the conventional structural failure modes
- Scheme of fuel handling such as offline as against online
- Manual reactor control against automatic control due to large reactivity worth availability
- Sodium fire hazard
- Shutdown systems with limited diverse features
- Large thermal capacity in pool type fast reactors
- Presence of large and massive structures above main vessel
- Different types of initiating events with different unfolding safety scenarios

The above issues have to be specifically addressed in the design and the approach to safety.

5.2 SAFETY APPROACH

5.2.1. Defence in depth principle

Application of the concept of defence in depth throughout design and operation provides a graded protection against a wide variety of transients, anticipated operational occurrences and accidents, including those resulting from equipment failure or human action within the plant, and events that originate outside the plant. Following five levels of defence (inherent features, equipment and procedures) aimed at preventing accidents and ensuring appropriate protection in the event that prevention fails have been listed in IAEA Safety Standard. The safety approach remains based on the prevention and the mitigation of the abnormal situations with the objective being to make acceptable the risks for all the events possible in the installation.

5.2.1.1. The levels of the Defence in Depth

Following the IAEA/INSAG 10, measures relative to defence in depth are generally ranked in five levels of defence. The first four levels are oriented towards the protection of barriers and mitigation of releases; while the last level relates to off-site emergency measures to protect the public in the event of a significant release.

Level 1: Prevention of abnormal operation and failures

Measures at Level 1 include a broad range of conservative provisions in design, from siting through to the end of plant life, aimed at confining radioactive material and minimizing deviations from normal operating conditions (including transient conditions and plant shutdown states). The safety provisions at Level 1 are taken through the choice of site, design, manufacturing, construction, commissioning, operating and maintenance requirements. Furthermore, Level 1 provides the initial basis for protection against external and internal hazards, even though some additional protection may be required at higher levels of defence
.

Level 2: Control of abnormal operation and detection of failures

Level 2 incorporates inherent plant features, such as core stability and thermal inertia, and systems to control abnormal operation (anticipated operational occurrences), accounting for all the phenomena capable of causing further deterioration in the plant status. The systems to mitigate the consequences of such operating occurrences are designed according to specific criteria (such as redundancy, layout and qualification). The objective is to bring the plant back to normal operating conditions as soon as possible.

Level 3: Control of accidents within the design basis

In spite of provisions for prevention, accident conditions may occur. Engineered safety features and protection systems are provided to prevent evolution towards severe plant conditions and also to confine radioactive materials within the containment system. The measures taken at this level are aimed at preventing core damage in particular. To ensure a high reliability of the engineered safety systems, design principles are adhered to such as: redundancy; prevention of common mode failure by physical or spatial separation and structural protection, by diversity or functional redundancy; automation to reduce vulnerability to human failure; testability to provide clear evidence of system availability and performance; qualification of systems, components and structures for specific environmental conditions that may result from an accident or an external hazard.

Level 4: Control of severe conditions including prevention of accident progression and mitigation of the consequences of a severe accident

For the concept of defence in depth as applied to currently operating plants, it is assumed that the measures considered at the first three levels will ensure maintenance of the structural integrity of the core and limit potential radiation hazards for members of the public. Nevertheless, additional efforts are made in order to further reduce the risks. The broad aim of the fourth level of defence is to ensure that the likelihood of an accident entailing severe core damage, and the magnitude of radioactive releases in the unlikely event that a severe plant condition occurs, are both kept as low as reasonably achievable, economic and social factors being taken into account. The most important objective for mitigation of the consequences of an accident in Level 4 is the protection of the confinement It has to be pointed out that accident management may not be used to account for design deficiencies at prior levels.

Level 5: Mitigation of the radiological consequences of significant external releases of radioactive materials

Even if the efforts described in the foregoing are expected to be effective in limiting the consequences of severe accidents, it would be inconsistent with defence in depth to dismiss off-site emergency plans. These plans cover the functions of collecting and assessing information about the levels of exposures expected to occur in such very unlikely conditions, and the short and long term protective actions that constitute intervention. The responsible authorities take the corresponding actions on the advice of the operating organization and the regulatory body.

5.2.2. Categorisation of Events

The safety of the plant is evaluated by analyzing the plant response to various events affecting the plant. They are broadly classified as Design Basis Events (DBE) and Beyond Design Basis Events (BDBE). The categorization into DBE and BDBE depends on the probability of occurrence. There are slight variants to the definition adopted by different countries. Typically, the DBE have the frequency of occurrence more than 10^{-6} per reactor-year (ry). The DBE are further classified into few categories according to the frequency of their occurrence. The basis for the categorization are the frequency of events and their severity. Very unlikely or rare events are classified as BDBE and they typically have the frequency less than or equal to 10^{-6}/ry. Fig. 5.1 shows the frequency of accident Vs. accident severity.

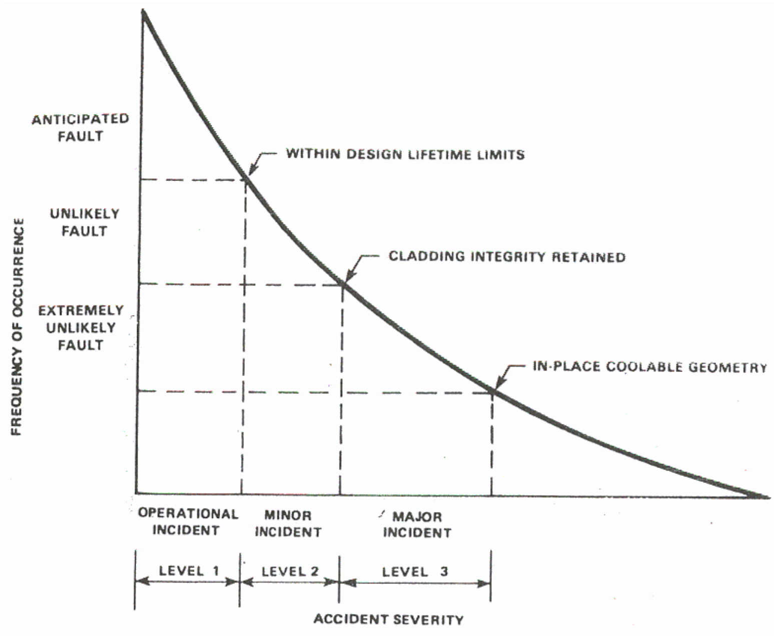

Fig. 5.1 : Accident severity Vs frequency of accident

5.2.2.1. Design Basis Events

All the postulated events will be identified in the design basis established for the plant. The categorization adopted for a typical plant are as follows. The categorization and their implications on the plant are also given in the table.

Category	Definition
1	Normal Operation
2	Upset Condition Abnormal behaviour due to minor faults

	Do not require any major repair or regulatory inspection (> 10^{-2} / reactor year)
3	Emergency Condition Requires mandatory inspection & repair before plant restart (< 10^{-2} / reactor year & > 10^{-4} / reactor year)
4	Faulted Condition Plant Restart not essential & Public health and safety to be ensured (< 10^{-4} / reactor year & > 10^{-6} / reactor year)

The number of such events during the life of the plant will be conservatively estimated. Safety provisions for the events will be provided to satisfy the criteria acceptable and adopted in the design and respecting the radiation protection limits as prescribed in the approved manuals of safety.

5.2.2.2. Beyond Design Basis Events

The events with a probability of occurrence of less than 10^{-6} per reactor year will be typically considered as BDBEs which have to be analyzed to evaluate the consequences of such accidents for consideration of a site emergency plan etc.

5.3 GENERAL SAFETY FEATURES OF FAST REACTORS

General safety features of a fast reactor plant can be broadly classified as inherent safety features and engineered safety features. Inherent safety features are available because of the chosen concept of the reactor system, choice of coolant and core characteristics. Specific inherent features are pool concept, negative reactivity coefficients, efficient sodium coolant and easy natural convection. Some of the important engineered safety features are multiple radial entry sleeves for subassemblies, inertia on primary pump, emergency power supplies for the primary sodium pumps, core monitoring instrumentation, reactor shutdown systems, Safety Grade Decay Heat Removal System, protection against sodium - water reaction and external sodium leaks and the various design provisions to prevent transient over power event and to prevent escape of radioactivity to the environment following the BDBE of core disruptive accident. In the following sections, brief descriptions of these features are given.

5.3.1 Pool Concept

Pool type concept for primary sodium circuit provides large thermal inertia and hence it takes more time to affect the main vessel and core support structure in case of DBE during reactor operation, than in a loop type reactor. The main vessel (MV) has no nozzle penetrations and thus offers high structural reliability. The safety vessel around the MV ensures adequate sodium inventory in the MV for decay heat removal even in case of a leak from the main vessel. Thus, loss of coolant type of accident is not a design basis event in pool type fast reactor.

5.3.2 Negative Reactivity Coefficients

All reactivity coefficients except the sodium expansion effect are negative. Therefore, the reactor operation is stable for any bounded disturbances. The burn-up reactivity changes are small due to breeding of fuel and low absorption cross-section of fission products to fast neutrons. There is no xenon poisoning as in thermal reactors. There is no need for automatic power regulation. This avoids the problems that may arise by failure of the power regulation system.

Inlet temperature and power coefficients of reactivity are negative so that any off normal increase in temperature and/or power leads to reduction in reactivity and the consequent reduction in power. The expansion of coolant and structural steel result in small positive reactivity that is compensated by negative and prompt reactivity effects like Doppler and fuel expansion. There would also be negative reactivity feedback from grid plate, spacer pad and differential control rod expansion that tend to decrease the reactor power for transient under cooling incidents.

5.3.3 Efficient Sodium Coolant

Sodium boils at 1160 K (at atmospheric pressure) while the maximum sodium temperature at the core outlet is typically around 850 K. Thus, a large margin of about 300 K exists between the normal operating sodium temperature at the core outlet and the boiling point of sodium. This can accommodate significant temperature rise in the event of mismatch between heat generation and heat removal, without the system being pressurized. Maximum pressure in the primary sodium circuit during normal operation is less than 1 MPa. All the events associated with the de-pressurization of coolant as in thermal reactors are absent in fast reactors. The primary stresses in vessels and piping remain low leading to low probability of failure.

5.3.4 Easy Natural Convection

High thermal conductivity, low viscosity and large difference between the temperatures of hot sodium at the bulk pool temperature of about 820 K and ambient air at 310 K coupled with significant variation of sodium density with temperature changes facilitate decay heat removal through natural convection mode.

5.3.5 Radial Entry of Coolant into subassemblies

To prevent total flow blockage in fuel and blanket subassemblies, multiple radial openings in the sleeves in grid plate and in the feet of all the core subassemblies are normally adopted in the design as shown in Fig. 5.2. In some designs, total blockage at the outlet of a subassembly is prevented by providing an adapter, which ensures alternate path for flow through side gaps in case of a blockage occurring at the top. A typical design of the adapter assembly is shown in Fig. 5.2.

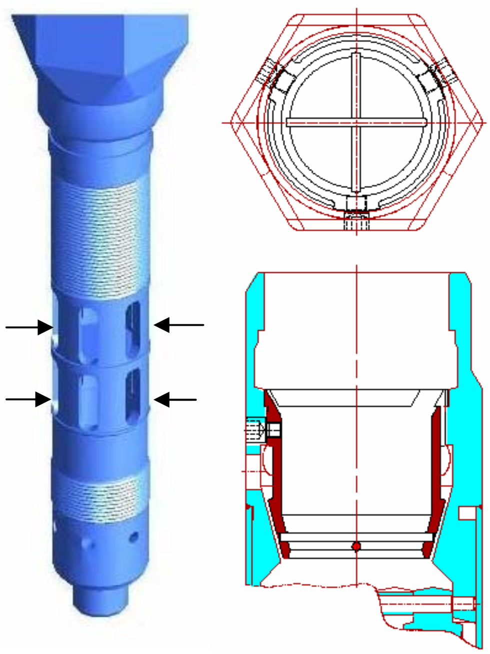

Fig. 5.2 : Mechanisms to avoid total blockages

5.3.6 Inertia of Primary Sodium Pump (PSP)

Flywheel is provided on the PSP shaft to achieve a gradual flow coast down during power failure and PSP trip events. This is necessary to prevent fuel pin clad damage.

5.3.7 Emergency Power Supply for PSP

The PSP will be normally provided with independent Class 3 power supply arrangements to ensure adequate forced convection flow through the reactor for most event situations. The PSP can also be provided with pony motors capable of running the PSP at a little higher speed than the speed envisaged. The pony motors will be provided with Class 3 and Class 2 uninterruptible power supplies (UPS), as a defense in depth. Emergency power from diesel generator will also be provided to the PSP to maintain forced circulation at a given level during loss of class IV power.

5.3.8 Core Monitoring

The objective of core monitoring is to detect any fault and ensure safety of the core. Neutron detectors are provided to monitor the power and provide signals on parameters like power, period and reactivity. Neutron flux monitoring system will be usually triplicated to permit 2/3 voting logic. The other parameters that would normally be monitored are: (i) SA sodium outlet temperature at the outlet of each fuel SA (ii) Reactor inlet temperature monitoring in each PSP suction (iii) PSP flow and derivation of power to flow ratio (iv) Failed fuel detection is done by monitoring the delayed neutrons in the cover gas and primary sodium coolant entering the IHX (v) Delayed neutron detection (DND) by the primary sodium sampling. Safety action on the signals of period, reactivity, power, central SA sodium outlet temperature, deviation of individual SA temperature from an expected value, mean core temperature rise, power to flow ratio and DND causes release of all the absorber rods to fall into the core by gravity (SCRAM) thereby reducing the reactor power. Provisions mentioned above would ensure at least two SCRAM parameters for all the events affecting the core. Of course, for clad failure detection and SA blockage detection, there would be only one parameter.

5.3.9 Shutdown System (SDS)

In the event of any safety parameter crossing its set point, the shutdown systems would be activated which will scram the reactor. Usually, in the shutdown system design, diversity will be provided to the extent possible starting from the sensor and up to the instrumentation logic to avoid common mode failure.

Depending on the number of shutdown systems and their duty, they will have a non-availability of certain value so as to achieve the non availability of the complete shutdown systems being less than 10^{-6} /ry. The two shutdown systems for PFBR is shown in Fig. 5.3

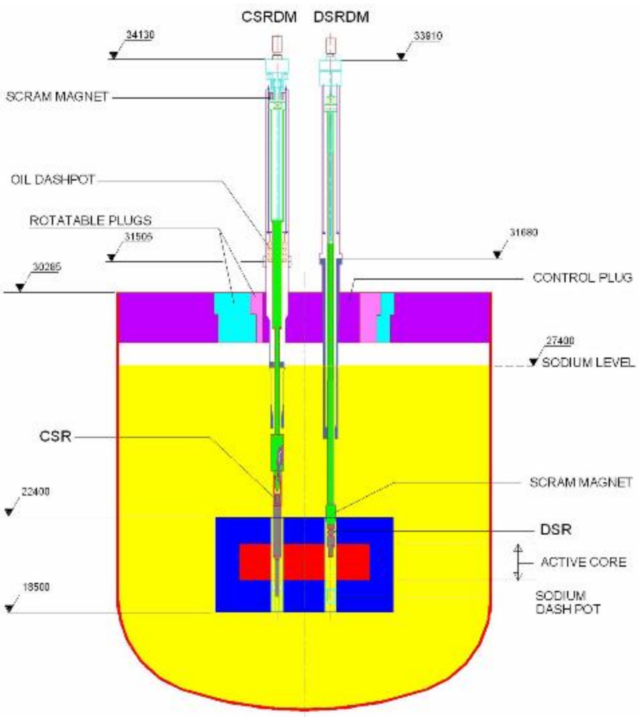

Fig. 5.3 : Shutdown systems in PFBR

5.3.10 Decay Heat Removal

Heat is generated due to residual fission power (which would be significant only for a few minutes after a SCRAM) and decay of fission products even after shutdown of the reactor. It would be about 1-2% of the nominal power after few hours to 1 day after the reactor is shutdown. Therefore, to maintain core integrity after reactor shutdown, the decay heat is to be removed. Thus, Decay Heat Removal (DHR) system is another important safety system that would be designed with a certain failure frequency (typically, less than 10^{-7}/ry). Sufficient redundancy and diversity would also be provided in the system. Decay heat removal in all the normal operating conditions and in some of the upset conditions where the steam - water system is not impaired will be through the normal heat transport system i.e. through the steam generators and steam - water system. During other events, decay heat is removed through dedicated Safety Grade Decay Heat Removal (SGDHR) circuits, which would employ a series of heat exchangers which will reject the heat to the ultimate heat sink. Normally, natural circulation would be resorted to while deciding the SGDHR circuit layout and the design. A typical decay heat removal circuit employed (PFBR) is shown in the Fig. 5.4.

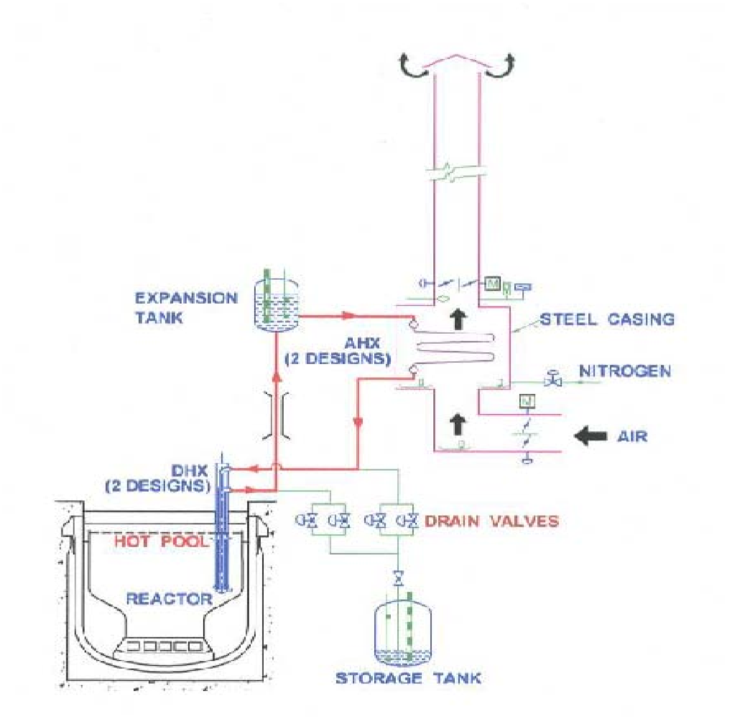

Fig. 5.4 : Safety Grade Decay Heat Removal system of PFBR

5.3.11 Protection Against Sodium – Water Reaction (SWR) in SG

Despite the high quality design and construction of the Steam Generator (SG) modules, steam or water may leak into the sodium and hence SWR is likely to occur. Adequate provisions will be incorporated to detect, terminate and limit damages to secondary sodium circuit components. Micro leaks and small leaks will be detected through continuous on line hydrogen monitors in the sodium streams out of

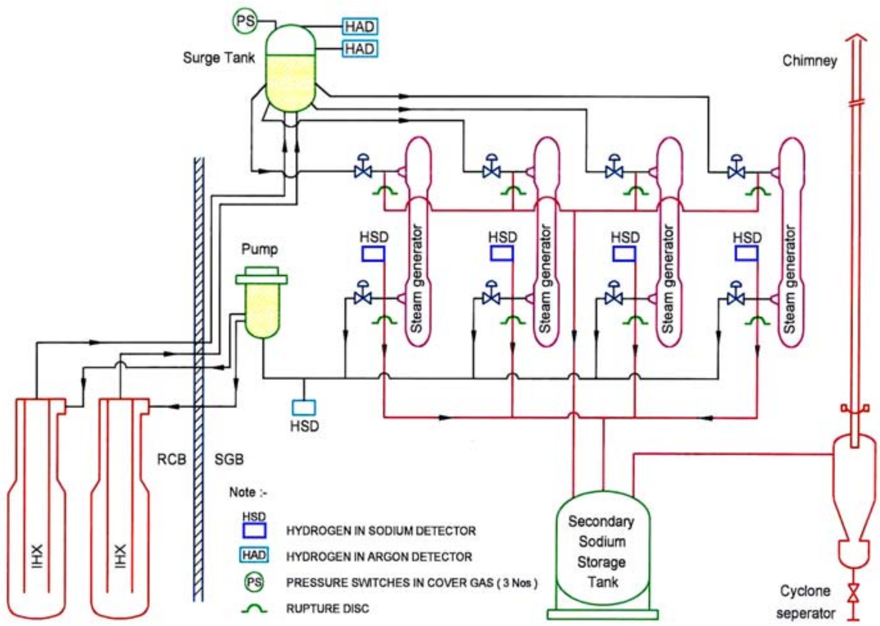

Fig. 5.5 : Secondary Sodium circuit

the SG and in the argon cover gas in the surge tank. When such a leak is confirmed in a SG module, isolation of the module both in the steam – water and sodium sides is carried out, the steam – water side depressurised and nitrogen blanketed automatically. This prevents further leak and tube wastages and hence avoids a large SWR. In case of a large SWR, rupture discs fitted in the inlet and outlet lines of the SG module opens out (by the pressure generated due to the large SWR), direct the SWR reaction products to the sodium storage tank and limits the maximum pressure rise in the circuit. Isolation of the particular SG module is also carried out from sensing the rupture discs opening up. The secondary sodium circuit schematic is shown in Fig. 5.5.

5.3.12 Prevention and Mitigation of External Sodium Leaks

Sodium leaking from pipes or components of the secondary systems reacts with oxygen in air and results in sodium fire. The sodium oxide production results in smoke formation, which affects visibility. Some of the possible design features considered to prevent and detect sodium leaks, and to mitigate the leak effects are given below:

- Quality assurance in design and construction.
- Safety vessel around main vessel with nitrogen in the inter-space and sodium detectors.
- Provision for emergency dumping of secondary sodium.
- All sodium pipes inside the RCB are provided with guard pipe.
- Sodium aerosol, wire type and spark plug type detectors at appropriate places.
- Provision for leak collection trays.
- Use of sodium resistant concrete.
- Providing partition walls to limit the effect of fire on neighbouring equipment.
- Provision of proper vents in different areas to relieve pressure developed due to sodium fire.
- Provision of fire extinguishers at appropriate locations in the plant.

5.3.13 Back-up Control Room

A back-up control room will be normally provided for monitoring and carrying out essential operations to maintain the reactor under safe shutdown conditions in case the main control room is uninhabitable due to fire. Usually, manual SCRAM would be made possible from this room.

5.3.14 Design Measures against Transient Over Power (TOP) Event

TOP arises due to addition of reactivity. This can occur due to any one of the events like ejection of an absorber rod, catastrophic failure of core support, large argon bubble passing through the core, fuel subassembly blockage leading to sodium boiling, core compaction due to fuel melting and slumping, entry of moderating materials into the core and withdrawal of an absorber rod. Hence, suitable and appropriate design measures would be incorporated to preclude or prevent the above.

Ejection of absorber rod can be prevented by providing higher downward force due to its weight than the upward thrust due to core flow under all conditions. Components in the core support path should be designed for the highest design class. Primary system should be designed in such a way that formation of large stagnant volume of bubble that could collect in the pump suction is avoided. Normally, purger subassemblies are employed towards this purpose which would channel the gas bubbles through the outer part of the core where sodium void worth is negative.

Fuel subassembly inlet blockage can be made unlikely by suitable design such as multiple and redundant radial coolant inlet design. Similarly, measures could be thought of to prevent blockage at the top of the SA (similar to the blockage adaptor in the fuel SA of PFBR, which will ensure a by-pass flow path in case of blockage of the outlet). Purity of sodium coolant is maintained to prevent any impurities precipitating inside the core coolant channels and causing internal blockages.

Further, experiments have shown that small internal coolant blockage in rod bundles do not propagate. Failed fuel pins may lead to Fuel Coolant Interaction (FCI) and some internal blockage can take place because of this. However, DND systems provided are capable of detecting a fuel failure. With such events contained within a single fuel SA by the hexcan walls, FCI does not propagate to other subassemblies. Finally, monitoring individual fuel SA sodium outlet temperatures enable early detection of blockage development and reactor shutdown.

By the above measures, fuel melting and potentially high positive reactivity insertions could be excluded as far as possible.

5.3.15 Reactor Containment Building

The potential sources of radioactivity in the plant are the fission products in the fuel, primary sodium, primary argon cover gas and washings from the spent fuel handling and component handling systems. Elaborate design provisions, applying the defence in depth principle would be put in place to avoid exposure of the operating personnel and public at the site boundary in excess of acceptable dose limits. The fuel matrix, fuel pin clad, sodium and main vessel provide multiple barriers. The Reactor Containment Building (RCB) provides the fourth barrier, as an engineered safeguard. Thus there are four barriers before the public can be exposed to the fission products. Even for a core disruptive accident (a BDBE), the RCB would be normally designed such that it can withstand the pressure build-up due to the sodium fire.

5.4 PLANT PROTECTION SYSTEM

5.4.1 Elements of the plant protection system

Despite the assurance offered by careful design, construction and operation, the second and most important level of safety is the plant protection systems (PPS), which can handle a wide range of conceivable abnormal incidents and malfunctions and safety shutdown the reactor. Basically the system consists of :

(i) A system of instrumentation to monitor the plant operating parameters and characteristics.
(ii) Assured shutdown systems triggered by signals from the monitors.
(iii) Decay heat removal systems

Core damage may occur if the shutdown systems or the decay heat removal systems fail on demand. Hence proper operation of the PPS is very important. This is assured by means of redundancy, diversity and fail-safe operation of the components of the PPS. Redundancy refers to the use of two or more similar components in parallel by means of which it is assured that the PPS does not fail when needed, and at the same time, false plant shut-downs are not caused by instrument errors. Diversity in shutdown systems, electric power systems, etc. is provided to prevent common mode failure, i.e. a number of failures resulting from a single cause. By fail-safe operation is meant that failure of a component of the PPS results in shutdown of the plant rather than continued unprotected operation. The PPS often includes a process computer which monitors and processes the information of all signals relevant to the safety of the plant.

5.4.2 Plant monitoring systems

The most important monitoring systems in a fast reactor plant are for neutron, sodium temperatures, coolant flow and clad rupture detection. Other monitoring systems are for sodium levels, sodium leaks, aerosol monitoring and steam generator leak detection.

Neutron monitoring: Neutron detectors and associated electronics are required in order to monitor and control the power level as well as to provide signals for safety action. The neutron monitoring system provides signals for alarms, power reduction or reactor shutdown when the neutron density level is too high or the rate of increase of neutron density is too high. In addition, most fast reactors contain reactivity meters which use measured neutron density as a function of time to derive the reactivity changes, which can also signal for safety action if found excessive. The neutron monitoring system is triplicated and safety action is taken when any two of the three monitors call for it.

On account of high gamma radiation fields, all neutron detectors should discriminate against gammas. The gamma detectors that would be employed are BF_3 counters, boron coated counters, fission counters and boron coated compensated ion chambers. Location of the detectors outside the primary vessel is desirable from considerations of temperature and radiation. Further, for a pool type reactor, adequate neutron levels are not obtained outside the main vessel for shutdown monitoring and use of in-vessel detectors may have to be considered.

Sodium temperature and flow monitoring : Monitoring of sodium temperatures and early detection of sodium boiling is very important for core safety especially with positive coolant voiding coefficients. Most commonly, thermocouples are used to monitor the coolant outlet temperature from each fuel SA. Similarly, the coolant flow from the primary sodium pumps also would be monitored. SCRAM signals are actuated by high sodium temperature, low flows, mismatch of power to flow values, loss of electric power supply, etc. The onset of sodium boiling is also sought to be detected in modern LMFBR's by the methods of neutron noise and acoustic noise detection. Bubbles formed when sodium boils give rise to both reactivity fluctuations (see the discussion in 3.3b) as well as to pressure variations in the metal. These fluctuations can be recorded by neutron monitors or by pressure transducers and the onset of boiling detected.

Clad rupture detection : Detection, localisation and removal of failed fuel with ruptured clad is required to prevent excessive contamination of the primary coolant and to prevent flow blockages that could arise on account of gradual fuel swelling when sodium contact the fuel. The methods of failed fuel detection are by monitoring of the cover gas fission product activity and by monitoring of delayed neutrons in the primary coolant. Excessive activities from these monitors trigger a plant shutdown. The cover gas fission product activity is due to gaseous fission products like Xe and Kr escaping from ruptured fuel pins. These gases require time to reach the cover gas plenum and the response time for clad rupture detection by cover gas monitoring is of the order of minutes. On the other hand, fission products like I^{137} and Br^{87} which are delayed neutron precursors get dissolved in the sodium upon clad rupture and the emitted neutrons are detected in a bypass loop of the primary circuit. The response time of this system is a function of the coolant transport time and is in the range of 20-30 sec. Identification of the failed fuel subassembly is an important issue and one of the methods followed in some reactors is to sample the sodium from each subassembly by selector valves and monitor the delayed neutrons.

5.4.3 Reactor shutdown systems

Besides the function of reactivity or power adjustment, the control rods must be able to shutdown the reactor under any foreseen conditions. It is important to assess the mechanisms of accidental reactivity insertions and ensure that the shutdown system can compensate for these. Such mechanisms include melting of a fuel subassembly, sodium voiding, inadvertent withdrawal of a control rod, introduction of moderator, sudden flow increases, etc. The principle of diversity, independence and redundancy are followed while designing the number and the engineering of shutdown systems.

5.4.4 Decay heat removal

The decay heat is initially about 6 to 7% of the reactor power and gradually falls with time becoming around 1% after an hour. An important part of the PPS is the provision of adequate decay heat cooling under shutdown conditions for as long as necessary (up to a month or more) even under conditions of coolant system failures.

Decay heat removal in fast power reactors is ensured by a diversity of means.

(i) multiplicity of primary heat transport loops ensuring normal decay heat removal by the operation of a single loop alone,
(ii) redundancy and diversity in power supplies to pumps,
(iii) forced circulation in the immediate period following shutdown (when decay heat generation rates are high) by design provisions, such as fly wheels for gradual coast down of pumps,
(iv) continued forced circulation by low speed operation of pumps by means of pony motors backed by alternate power supplies / emergency diesel generator sets / batteries,
(v) provision of independent back-up decay heat removal systems,
(vi) design of normal as well as back-up heat removal systems, such that setting-up of good heat removal by natural convection is facilitated.

In pool type reactors, the large thermal capacity of the sodium bulk acts as a heat sink and allows a long time interval of several hours for alternate measures to be taken before the sodium boils. The possibility of complete loss of sodium from the primary heat removal circuits is also not present in such reactors. Setting-up of natural convection for heat removal would be ensured by suitable elevations of core, IHX and sodium / air heat exchangers. To provide cooling in case of common mode failure resulting in non-availability of all normal loops, an independent back-up decay heat removal system would be provided.

5.5 SCRAM PARAMETERS

The safety action, in the event of scenarios proceeding contrary to the design intent, is accomplished through scram parameters which provide the index of safety on exceeding the design values. The choice of scram parameters and their location and number of signals and the method of deriving the decision form an important element in the design of the plant. At the same time, multiplicity of scram parameters should not lead to frequent and spurious scrams and complexity of the systems. While finer aspects vary between reactors, broadly the principle behind the selection of parameters are same and also the parameters to a large extent. As an example, the scram parameters used in PFBR is briefly explained in the following section.

The Design Safety Limits are given in the Table 5.1 for the different categories of the DBE. The design criteria adopted are as follows:

Table 5.1 Design Safety Limits adopted for PFBR

Parameters	Category of events			
	1	2	3	4
Functional				
Shutdown	Yes	Yes	Yes	Yes
Decay heat removal	Yes	Yes	Yes	Yes
Containment	Yes	Yes	Yes	Yes
Restart	Yes	Yes, after fault clearance	Yes, after inspection and repair if required	Not necessary
Temperatures, K				
Cold pool	670	813	873	913
Hot pool	820	873	898	923
Average SA coolant hotspot (AVSHS)	No Bulk Coolant Boiling No burnout in local hotspots			
Clad hotspot (CHST) – Driver fuel SA	973 and *CDF ≤ 0.25*	974–1023 *for 75 m* and 1023-1073 *for 15 m* for all cate.2 events and *CDF ≤ 0.25*	974-1073 *for 15 m* and 1073–1123 *for 6 m* and 1123–1173 *for 2 m* and *CDF ≤ 0.25*	1473
Clad hotspot (CHST) – Storage SA	823	873	923	1223
	and CDF < 0.25			
Fuel hotspot (Hotspot fuel center line - FHST)	No melting	No melting	No melting	Melting to the extent that there is no clad failure associated with this melting.
Structural				
As per RCC-MR	Level A	Level A	Level C	Level D
Radiation				
Plant Personnel	30 mSv / y / p, 100 mSv / p cumulative for a block of 5 y			250 mSv / p
Public at site boundary	0.1 mSv / y (for PFBR)			100 mSv / event*
Public evacuation	No	No	No	No

- For any DBE in which the consequences on fuel, clad and coolant are below its respective Category DSL, automatic protective action of SCRAM is not called for.
- For all the other DBE, the first SCRAM parameter should limit the consequences within the specified Category DSL of the event and the second SCRAM parameter should limit the consequences within the next higher Category DSL. However for the category 4 DBE, two SCRAM parameters should limit the consequences within the category 4 DSL.

The list of essential SCRAM parameters and their thresholds are obtained by carrying out the transient analyses of all the DBE which challenge the DSL, such that the maximum values of sodium, clad and fuel temperatures reached are limited below the DSL by the action of SCRAM. In addition to this, the thresholds are set close to the normal operating levels after accounting for (i) fluctuations in the signal around their mean value, (ii) operational margins and (iii) errors that could occur in setting the thresholds. The most important plant measurements are neutron flux (Φ), sodium temperatures at the core inlet (θ_{RI}), central SA outlet (θ_{CSAM}), and other individual fuel SA outlets (θ_I), Primary pump speed (N_P), core flows (Q_{PP}) (measured at the PSP discharge) and the delayed neutron flux (DND). Some of the important derived signals from these measurements are reactor power (Lin P and Log P), period (τ_N), reactivity (ρ), power to flow ratio (P/Q), group mean of SA sodium outlet temperature (θ_M), deviation of individual SA sodium outlet temperature from an expected value ($\delta\theta_I$) and the mean core temperature rise ($\Delta\theta_M$). Overpower events can be detected by the power, reactivity or reactor period and temperature rise in the core. Undercooling transient can be detected by pump speed, power to flow ratio, outlet temperature rise in the subassemblies, and reactivity. DND serves for detecting fuel pin clad failures.

Selected SCRAM parameters, their thresholds and the shutdown system to which they are connected are given in Table 5.2 which includes the three SCRAM parameters Log N, Log P and variable linear power (Lin P_V) that caters for specific situations. Log N is the SCRAM parameter originating from the pulse mode signal processor for the case of non-take over of the Campbell mode signal processor with the threshold at 3.1 MW. As and when the Campbell mode takes over successfully (at 250 kW), Log N parameter is automatically inhibited by a signal from the Campbell mode at 500 kW. Similarly Log P is the SCRAM parameter originating from the Campbell mode signal processor for the case of non-take over of the pulse mode linear signal processor with the threshold at 20 % nominal power. As and when the DC mode takes over successfully (at 15 MWt), Log P parameter is automatically inhibited by a signal from the DC mode at 5 % nominal power (62.5 MWt). Analysis has also been carried out to check the adequacy of these parameters' threshold for effectively managing the continuous control rod withdrawal event even while the non take over of the Campbell mode or the DC mode of signal processing. Lin P_V SCRAM parameter has been introduced to prevent the reactor power exceeding 110 % of a target power that is less than the nominal power, during any event or operator error. This parameter is especially useful during initial commissioning phase when the reactor power would be raised to the design power in stages and for prolonged operation of the plant on reduced power for any other reason. Reactivity parameter is inhibited up to 5 % power and period parameter will be active till that time. Period is inhibited beyond 5 % power.

Table 5.2 : List of Reactor SCRAM Parameters in PFBR

Sl. no.	Parameter	Description	Threshold	Shutdown system
1	Log N	Power in pulse mode	3.1 MW	1
2	τ_N	Period	10 s	1
3	Log P	Power in Campell mode	20 % nominal	1
4	τ_P	Reactor period for an ‘e’ fold increase	10 s	1
5	Lin P	Reactor power in linear channel	110 % nominal	1

6	Lin P_V	Variable linear power	110 % target	1
7	ρ	Net reactivity	± 10 pcm	1
8	DND	Neutron flux at delayed neutron detectors	10 cm^2 recoil area	1,2
9	P/Q	Reactor power to flow ratio	1.10	1
10	θ_{CSAM}	Central subassembly sodium outlet temperature	Nominal + 10 K	2
11	$\Delta\theta_M$	Core temperature rise	Nominal + 10 K	2
12	$\Delta\theta_{CSAM}$	Central SA temperature rise	Nominal + 10 K	1
13	$\delta\theta_I$	Deviation from an expected value of a subassembly sodium temperature	10 K	1,2
14	θ_{RI}	Reactor inlet sodium temperature	Nominal + 10 K	1
15	N_P	Primary pump speed	95 % nominal	1

Protection against blockage of SA is achieved though the monitoring of sodium temperature at the outlet of individual fuel SA and comparing it against an expected value. Signals for this parameter are derived through two thermocouples mounted near the exit of each fuel SA. In case of failure of one thermocouple it will be detected and set to SCRAM automatically. From this point, the system will function in one out of one mode. P/Q signal is derived from normalized power P (i.e. P/Po) and normalized flow Q (i.e Q/Qo). The option selected for deriving the P/Q parameter is by considering the total flow from two pumps (i.e. Q=Q1+Q2) and normalized with initial total flow (i.e. Qo=Q1o+Q2o).

θ_{RI} of each pump is considered for SCRAM. There are three thermocouples in each primary sodium pump to monitor the reactor inlet temperature. Each thermocouple signal is compared with set point. If any of the two PSP1-thermocouple reading crosses the set point or any of the two PSP2-thermocouple crosses the set point, then SCRAM signal is generated.

5.6 DBE Analysis – PFBR Case study : One Primary Sodium Pump Trip

As an example case, salient results of the DBE analysis carried out for the event of one primary sodium pump trip in PFBR are given below.

Subsequent to this category 2 event of one primary sodium pump (PSP) trip, the speed of the pump reduces slowly against inertia. This results in the flow supplied by the pump to reduce. Due to the parallel operation of two PSP, the flow supplied by the other pump increases. However, the operating PSP flow increases to a maximum of only 126 % due to NPSH limitation. The tripped PSP flow decreases to 50 % in 2.6 s, to 0 % and reverses in 9.4 s. Subsequent to this, the operating PSP feeds both the core flow and the by-pass flow through the tripped pump. Core flow decreases to 61 % in 10 s with the operating PSP speed maintained constant. The reduction in the flow through core causes the temperatures to increase.. It can be observed that five parameters viz. N_P, P/Q, θ_{CSAM}, $\Delta\theta_{CSAM}$ and $\Delta\theta_M$ are available for protection for this event. Reactor SCRAM action by these parameters will be initiated after the delay time for measurement and electromagnet clutch release. Among the various parameters available the first appearing parameter that independently trigger reactor SCRAM by SDS-1 and SDS-2 are N_P and θ_{CSAM} respectively.

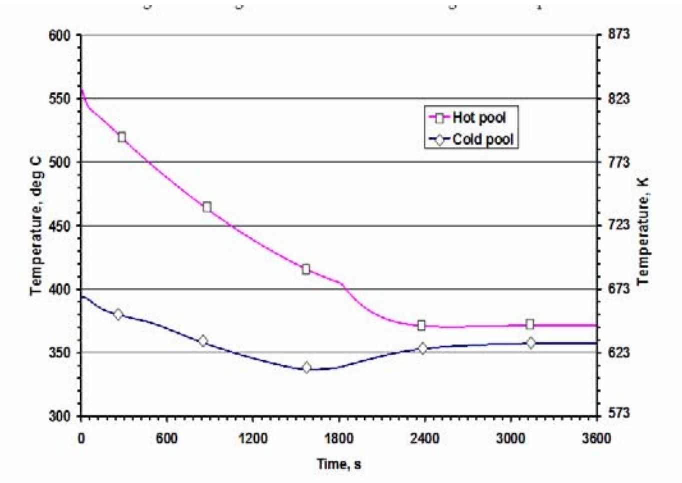

Fig 5.6 : Evolution of pool temperatures following one PSP trip

Maximum values of clad hot spot, average sodium hot spot and fuel hotspot reached were estimated during the event with SCRAM action taken based on various parameters and verified against the respective design safety limit temperatures. The availability of at least two SCRAM parameters for shutdown by SDS-1 and SDS-2 are ensured for this event. The evolution of hot and cold pool temperatures are shown in Fig. 5.6.

5.7 SUBASSEMBLY EVENTS

As mentioned in the beginning, the BDBE are those events that are considered for mitigating consequences of the accidents, identifying offsite emergency plans and are also important for site selection aspects. These events have very low probability of occurrence ($\leq 10^{-6}$ to 10^{-7} /ry) due to various provisions provided in the design in terms of intrinsic safety, sufficient margins in component design and redundancy, diversity and independence in the design of shutdown system and decay heat removal system. Events with probability < 10^{-7}/ry come in the category of residual risk. Local and whole core events that come in the category of BDBE and leading to core meltdown are briefly covered in the following section.

5.7.1 Design Approach

The main emphasis is on the accident prevention rather than accident containment. Hence, it is prudent to optimise the safety provisions with regard to probable events rather being concerned with hypothetical events.

From design point of view, it is necessary to define admissible damage levels in the fuel pin under different categories of event which are given below.

- No fuel pin failure up to design lifetime under category 1 and 2.
- No loss of cladding integrity under category 3 for a single event.
- Clad failure is admissible under category 4 events, however, coolable geometry of the fuel pin should be retained.

While detailed structural analysis is required to check the cladding integrity under various categories of events, the design safety limits would be checked against as discussed earlier. In the design approach, the availability of fault detection system and preventive actions are assumed in limiting the consequences to the desired level. The design strategy towards the local blockage comprises the following.

- Identification of initiating faults, their frequency and all their potential consequences;

- ➢ Specification of Lines of Defence (LOD) to detect, mitigate and safely terminate the event;
- ➢ Definition of Maximum Allowable Defect (MAD) limits taking into account protection systems;
- ➢ Determination of the Maximum Credible Defect (MCD) that may occur;
- ➢ Demonstration of MCD < MAD, taking the LOD's and uncertainties into account.

If this goal (i.e. MCD < MAD) is not achieved, design and/or protection system is to be improved. The lines of defence (LOD) are characterised by their reliability. Basically, three types of LOD's are available as given below.

i. Fault minimisation (prevention)
ii. Benign post-fault behaviour
iii. Active protection systems

Since clad melting may lead to material relocation, the design MAD is chosen to be no clad melting. Wherever small margin exists between boiling and dry-out, this limit means no sodium boiling. This is to maintain coolable geometry.

5.7.2 Local Subassembly Faults

The concern with regard to fuel subassembly flow blockage is melt-down in a SA propagating to whole core and finally resulting in Core Disruptive Accident (CDA). A large-scale melt-down accident took place in Fermi reactor due to inlet port blockage of several SA. However, with the radial entry of the coolant through multiple ports at different orientations, such type of accidents is made improbable in present day fast reactor design. The whole core melt-down accident due to other initiating events have also been made improbable due redundant and diverse methods of monitoring, highly reliable and diversified shutdown systems and highly reliable Safety Grade Decay Heat Removal (SGDHR) system.

However, it is recognised that a local melt-down accident due to any fault at the fuel subassembly level may propagate leading to whole core accident. Hence, understanding of local blockage phenomena is very important for the operation of commercial fast reactors. Because of the presence of large number of pins, the probability of local blockage formation increases.

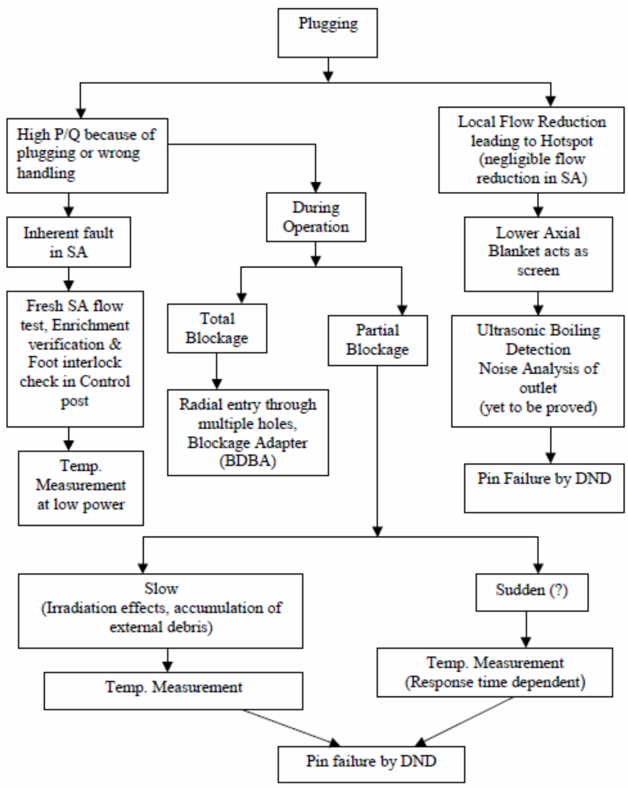

Fig. 5.7 : Protection against plugging

If a local fault reaches a critical size at a given coolant flow rate, it may cause local coolant boiling followed by dry-out resulting in reactivity addition and melting of pin cladding and fuel. The consequences could be either non- energetic dispersion of the fault or accumulation of larger amount of molten material with the possibility of a fast coolant vaporisation, pressure generation and fast damage propagation to the whole fuel SA. The effect of the local faults on the reactor safety depends on several factors: size and thermo-physical properties of blockages, its location in the SA, fuel pin power and coolant velocity in the SA.

Provisions to detect local SA faults

Protection against SA is provided at two levels, in design and safety action. The possible scenario with respect to protection against plugging is illustrated in Fig. 5.7.

The subassembly overheating may be categorised as gross and local. The gross error in a SA lead to the disturbance of the design power-to-flow ratio and may be initiated due to the following reasons:

- Incorrect positioning of a SA,
- Inlet or outlet blockage (gross blockage) of a SA

Loading a higher enrichment fuel SA to a lower enrichment SA position in the core is the most severe case of incorrect positioning of a SA. Prior to loading in the reactor core, the enrichment is checked in the control post. The foot profile is checked against the enrichment and flow zone category. Further, loading errors are detected by different combination of dimensions for coolant entry tube and the bottom discriminators for different SA. If a SA is loaded in an unsafe position by mistake, then full insertion is impossible and due to this design feature and monitoring top level of SA after loading helps in detecting wrong loading. The monitoring of outlet temperatures of the individual SA at low power by thermocouples mounted in the core cover plate acts as a backup to the above position.

Total and large-scale blockage due to the external debris is ruled out by providing radial entry of coolant through multiple holes. The multiple entry points are distributed throughout the circumference thus providing different orientation. Total blockage of fuel and blanket SA at outlet is ruled out by providing an adapter (PFBR design), which provides alternate path for flow through the side gaps.

It generally requires very large blockage to effect substantial flow reduction. As an example, for PFBR, a blockage as high as 83% of the inlet port, results in a flow reduction of 67% whereas for 72% blockage, the flow reduction is 43%. Analysis indicates that a flow reduction of 48% due to gross blockage avoids sodium boiling and hence admissible. For clad temperature limit of 1073 K, the allowable flow reduction is 28%. For a slowly developing blockage, a blockage rate of 4.63 %/s is allowed without exceeding the clad hotspot temperature whereas the allowable value is 9.72 %/s to avoid sodium boiling. Local blockages in a SA may be due to the following:

- Pin deformation,
- Failure of spacers,
- Transport of debris to the pin-bundle from the coolant circuit and getting lodged
- Fragments from failed fuel.

The different mechanisms for local blockage formation lead to different size, shape and physical property of the blockage. Even the geometry of the pin-bundle affects the nature of blockage formation. For wire-wrapped pin- bundle, a blockage spread over a large radial area is unlikely. A failed wire or debris accumulation in a wire wrapped pin bundle would lead to a long thin blockage.

Studies have revealed that the external debris is essentially trapped at the bottom of the bundle. The debris concentration decreases from the bottom of the bundle due to the screening effect due to wire-wrap. In no case, more than one or two adjacent sub-channels are expected to be blocked. Further growth is expected in axial direction. An analysis of the available data reveals that the process of blockage formation of oxides and impurities is very slow compared with the general time scale of failure development. In the case of a cladding fault of one pin, the amount of material available is insufficient for global blockage formation. At the same time, a large blockage in the fast reactor can occur only as a result of fuel release from the pin failure.

5.7.3 TOTAL INSTANTANEOUS BLOCKAGE (TIB)

TIB is defined as an instantaneous blockage at the inlet of fuel SA at normal operating condition. A TIB is considered as the bounding event enveloping all types of blockages that can evolve and has been specified as a BDBE. The safety objective is to demonstrate that TIB will not propagate to cause a whole core accident and coolable geometry of the core is maintained. The SCARABEE (Moxon et al., 1986; Moreau et al., 1991) tests for fresh fuel indicate that

- No significant fuel ejection from the fissile zone and, consequently, automatic end of the accident by fuel dispersion is ruled out. Thus a large power concentration is kept at the initial failure site;
- High heat transfer from the boiling pool;
- Rapid melt-through of the Hexcan and symmetrical propagation;
- No energetic MFCI
- No systematic DND signal from blocked SA.

Because of high heat transfer and no energetic MFCI, the penetration of the melt into neighbouring SA will be partial. Significant sodium flow might be exiting for some time, giving a possibility for thermal detection at the outlet of the the neighbouring SA and slowing down the thermal propagation. In any case, a DND signal can be assumed as soon as there is penetration into the inter-assembly or at latest after propagation into the neighbours. The signal will trigger the scram after a delay time of ~10-30 s. The main question is to know how far the melt propagates in this time and whether the resulting debris can be successfully cooled at decay heat levels. With irradiated fuel, the important phenomena might be different, although it can be expected that the DND signals might be emitted earlier than in the case of fresh fuel. One important conclusion can be drawn from all the tests that the local melting is not likely to lead to any significant core melting.

The question of the detection of the TIB has to be considered critically because TIB is physically impossible. That is to say that even if a total blockage occurs inside the SA, it cannot be instantaneous. An early DND signal is emitted as soon as the first clad rupture occurs. Even in the worst reactor conditions, the residual sodium flow is sufficient to provide significant DND signal from the faulted SA, thereby allowing the early detection of the faulted SA. Still the major task of the designers is to improve the performance and reliability of detection systems.

Two important specifications of the global detection system are the sensitivity and response time of the DND system. The sensitivity determines the detectable size of the failed pin. The response time on the other hand determines the number of SA meltdown following TIB and the core catcher size.

5.8 WHOLE CORE EVENT

A whole core accident may take place if there is a continued mismatch between heat generation and heat removal. This is possible by Transient Over Power (TOP) initiated by reactivity addition like uncontrolled absorber rod withdrawal resulting in heat generation greater than heat removal or by under cooling transients initiated by Loss Of Flow (LOF) or Loss Of Heat Sink (LOHS) resulting in heat removal smaller than heat generation. These accidents may lead to core melting and core disruption under the following two circumstances; (i) Shutdown System (SDS) fails on demand. Since the probability of failure of SDS is less than 10^{-6} / ry, these unprotected accidents come in the Category of BDBE (ii) The SDS works on demand but DHRS fails. This results in LOHS and is termed as protected LOHS accident. In this accident, coolant can boil off and sluggish core disassembly can tale place.

Also, in early days of fast reactor development, structural failures due to extreme external events like earthquake leading to core support failure, and various combinations of TOP/LOF, etc. have been postulated. However, these accidents clearly have sufficiently low probability and need not be considered for evaluation of their consequences. It is found by analysis that UTOP initiated by uncontrolled Control and Safety Rod (CSR) withdrawal in PFBR does not lead to a whole core accident. Both fresh and equilibrium cores were considered at nominal and conservative initial conditions. ULOF event initiated by loss of electrical power supply to the primary pumps resulting in 8 s pump speed halving time. The initial power is taken as the nominal power Only LOFA exemplifies the generic behavior over the whole range of CDA spectrum of circumstances (including pipe rupture, pump seizure, etc.), and hence can be used to adequately characterize the spectra of energetic consequences. For an irradiated core, reduced gap conductance, fission gas accumulated in the pin and irradiation swelling will affect the course of the accident. The effect of these is important when one does fine analysis of the transition phase (defined later) of the accident. In the absence of transition phase analysis, the conservative calculations of ULOF accident provides the envelope of reactivity insertion rate and energy release for fresh as well as irradiated core.

Four characteristics of large fast reactor cores – positive sodium void worth, core geometry not in its most reactive configuration, large margins to coolant boiling, and high fuel reactivity worth against an extensive background of experience in accident analysis, led to choice of ULOFA as the principal scenario for study, internationally. In the safety analysis of both SUPER PHENIX and EFR, only ULOFA was considered as whole core accident for determining energy release. The PFBR case is presented here as an example.

5.8.1 Initiators

5.8.1.1 UTOPA

Uncontrolled reactivity addition leading to UTOPA can occur due to

(a) uncontrolled withdrawal of control rods
(b) passage of large bubble in core
(c) collapse of core support structure leading to core compaction

Uncontrolled withdrawal of control rods is a PIE eventhough control rods are usually moved for a preplanned distance. If the worth of the rod is high enough or more than one rod is involved, it can lead to high reactivity addition and consequent large power increase.

Uncontrolled reactivity addition can also occur due to passage of large gas bubble in the core. Passage of such a large gas bubble in the core is very highly unlikely and the reactivity that can be inserted in present designs (with low coolant void reactivity effect) is low. Small gas bubbles result in very negligible reactivity addition.

UTOPA can perhaps also occur due to sudden structural failure of core support structure which is again considered as residual risk at present. Structural vibrations due to Operational Basis Earthquake (OBE) and Safe Shutdown Earthquake (SSE) can give rise to only reactivity oscillations. The reactivity oscillations from OBE and SSE are small in magnitude (less than 0.5 $, which a peak pulse), and further there is damping due to negative reactivity feedbacks and which can be shown to result in only a small increase in reactor power.

Hence, the uncontrolled withdrawal of control rod is the only initiator considered for UTOPA and possible CDA.

5.8.1.2 ULOFA

Coolant flow starvation can be initiator for ULOFA which are as follows:

a) failure of primary pump
b) seizure of primary pump
c) pipe rupture and
d) total instantaneous blockage in SA

Failure of primary pump due to loss of power supply can be an initiator for CDA. The loss of power supply leads to flow coast down (with flow halving time of about 8s to 10s) and reduced heat removal, leading to mismatch in heat balance. Pump seizure cannot lead to a CDA, if it recovers operation in few tens of seconds and it is a Category 3 DBE. Pipe rupture does not lead to total flow starvation, and is Category 4 DBE. Total Instantaneous blockage is prevented by having multiple holes for coolant entry in the foot of SA, and an adapter at the top of the SA to give an alternate flow path. Hence, pump seizure or stoppage of primary pump due to loss of power supply is the only main initiator for ULOFA and possible CDA.

Different physical phenomena are dominant at various stages of the accident and also the time scales involved are different. Therefore, the accident progression is analysed deterministically in different phases using cause and effect phenomenology and adopting a conservative approach where deterministic analysis is not possible. The phases of accident analysis are discussed below.

(a) Pre-disassembly Phase

In this phase of analysis, the calculations are performed deterministically for core neutronics, reactivity feedbacks, thermal hydraulics, sodium boiling, fuel pin failure, cladding and fuel slumping and their relocation and fuel coolant interaction. This phase lasts till the fuel reaches the boiling point and starts dispersing the core material.

(b) Disassembly Phase

Once the fuel starts dispersing, fuel displacement feedback dominates, time scales are short (of the order of milliseconds) and all other reactivity feedbacks except the Doppler can be ignored. The core loses its integrity. Pre-disassembly phase provides the initial conditions for the disassembly phase. The core neutronics and core hydrodynamics calculations are performed in 2D (r, z) geometry. This phase lasts till the reactor attains subcriticality due to fuel dispersal.

(c) Mechanical Energy Release / System Response Phase

At the end of the disassembly phase, though neutronically the reactor is in shutdown state, core is still expanding and thermal energy release is capable of doing mechanical work on the system. Here, the mechanical work potential (W_{max}) of fuel vapour isentropic expansion up to 0.1 MPa (one atmosphere) pressure or up to the volume equivalent to cover gas volume is calculated. As mentioned above, the results of the disassembly phase and mechanical energy release calculation form the input in evaluating the system (main vessel and top shield) response to the energy release in the CDA.

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