

SAFETY CRITERIA AND GUIDELINES FOR MSR ACCIDENT ANALYSIS

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ABSTRACT

Accident analysis for Molten Salt Reactor (MSR) has been investigated at ORNL for MSRE in 1960s. Since then, safety criteria or guidelines have not been defined for MSR accident analysis. Regarding the safety criteria, the authors showed one proposal in this paper. In order to establish guidelines for MSR accident analysis, we have to investigate all possible accidents. In this paper, the authors describe the philosophy for accident analysis, and show 40 possible accidents. They are at first classified as external cause accidents and internal cause accidents. Since the former ones are generic accidents, we investigate only the latter ones, and categorize them to 4 types, such as power excursion accident, flow decrease accident, fuel-salt leak accident, and other accidents mostly specific to MSR. Each accident is described briefly, with some numerical results by the authors.

Key Words: **molten salt reactor, safety criteria, safety guideline, accident analysis**

1. INTRODUCTION

Accident analysis for Molten Salt Reactor (MSR) has been investigated and several calculations were made for the experimental reactor: MSRE [1] [2]. However, it was 50 years ago, and it may not be applicable from the standpoint of recent licensing approach for Light Water Reactor (LWR). Since then, safety criteria or guidelines have not been defined for MSR accident analysis.

Regarding the safety criteria, the authors show one proposal in Section-3, based on temperature limitation of the component material: Hastelloy-N. In order to establish guidelines for MSR accident analysis, we have to investigate all possible accidents. This study is also required to prepare a preliminary safety analysis report (PSAR). The authors describe the philosophy for accident analysis, and show 40 possible accidents based on the philosophy with some numerical results in Section-4 and 5.

2. DEFINITION OF “ACCIDENT”

In the current LWR licensing, there are two major guidelines. One is used as a guide for safety design of the reactor at conceptual stage [3], which is called “General Design Criteria”. And another is used for accident analysis [4], which defines the events to be studied in licensing and provides the criteria of the analyzed results. In this paper, the authors describe the latter issue.

In the current LWR licensing, all abnormal events are categorized to three types, which are abnormal operating transients (AOT), design basis accidents (DBA), and severe accidents (SA) which are beyond DBA, as is shown in Figure-1 and Table-1. In this paper, the authors include all three types as “accident”, because their definitions for LWR may not be applicable to MSR, and also may be changed depending on different safety designs. Therefore, each accident shown in Section-5 must be categorized to AOT or DBA or SA at later stage, depending on its severity, probability and safety design.

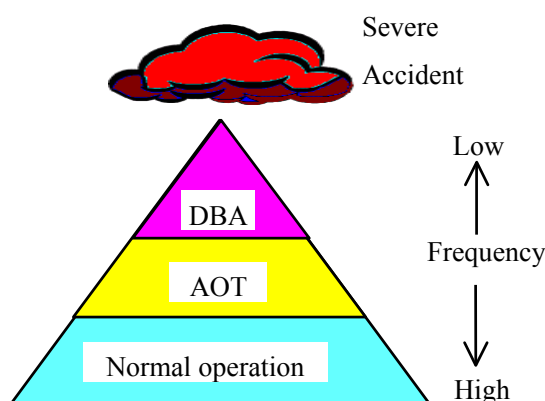


Figure 1. . Classification of reactor status

Table 1. Classification of reactor status and definition

| | |
|------------------------------------|--|
| Severe accident (SA) | Events beyond DBAs, which will cause core melt-down for LWR and/or large release of radioactivity. |
| Design basis accident (DBA) | Events beyond AOTs, which will cause LWR fuel failure. Initiated by 2 equipment malfunctions or 2 operator errors, or combination of each. |
| Abnormal operating transient (AOT) | Anticipated events to occur once or more during a plant service lifetime. Initiated by a single equipment malfunction or a single operator error |
| Normal operation | Reactor shutdown or startup, besides operation at power |

3. SAFETY CRITERIA FOR MSR

Since there are no safety criteria for MSR, the authors started from safety criteria for LWR, and propose the possible criteria for abnormal operating transients (AOT). Safety criteria for design basis accident (DBA), severe accidents (SA) and normal operation are future discussions.

3.1. Safety Criteria for Abnormal Operating Transients (AOT)

As for LWR, the objective of safety criteria for abnormal operating transients (AOT) is to keep re-startup capability of the reactor after AOT occurs. Therefore, in order to keep fuel integrity, heat flux index such as DNBR or MCPR shall not violate the limit, fuel cladding tube shall not mechanically fail, and fuel enthalpy shall be less than the limit. Also, in order to keep primary system integrity, reactor system pressure shall be less than the limit.

Since the fuel of MSR is molten salt, mechanical fuel failure is impossible. Then, the above criteria for fuel integrity cannot be applied directly. Also, since molten salt has high boiling temperature and low vapor pressure, boiling accident is incredible. Therefore, pressure criteria for primary system integrity is not appropriate to apply.

Therefore, appropriate criteria for primary loop boundary should be considered. Of course, reactor vessel of MSR must satisfy the provision of the ASME code (ASME Boiler and Pressure Vessel Code) such as ductility, brittleness, fatigue failure, or stress corrosion cracking, and so on, but these are mostly for steady state and for long-term design conditions. .

There are two candidates of safety criteria as follows. The first one is maximum allowable stress. In the MSBR design report, they defined maximum allowable stress as 250 Kg/cm² or at 704 deg-C for Hastelloy-N [5]. The maximum allowable stress of Hastelloy-N is shown in Figure-2, but there are no experimental data beyond 704 deg-C [6]. If this curve is extrapolated beyond that point, limiting temperature may be somewhere between 900 and 1,000 deg-C. However, these limiting temperatures are applicable for long-term operation. Therefore, these criteria are conservative for a short time event such as AOT.

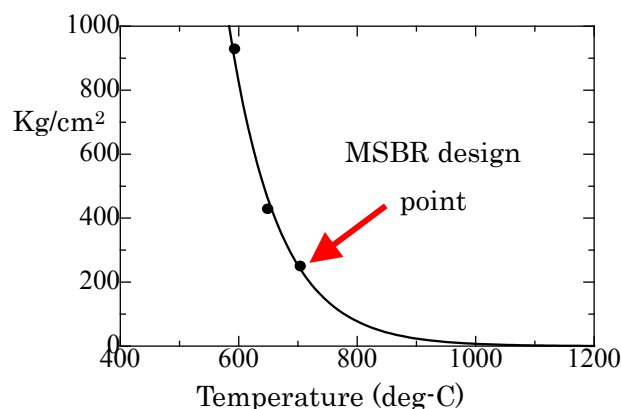


Figure 2. Maximum allowable stress of Hastelloy-N

One more possible candidate is the tensile strength of Hastelloy-N. Since a reactor vessel for MSR is not a “pressure vessel” and pressurization transient is incredible, loss of strength of the material could be limiting. In general, tensile strength is used as strength of metals. Based on the existing data for tensile strength as is shown in Figure-3, if it is extrapolated, the maximum allowable temperature could be somewhere between 900 and 1,000 deg-C [6]. For other Hastelloys such as Hastelloy-B / Hastelloy-C / Hastelloy-X, the maximum allowable temperature could be similar [7].

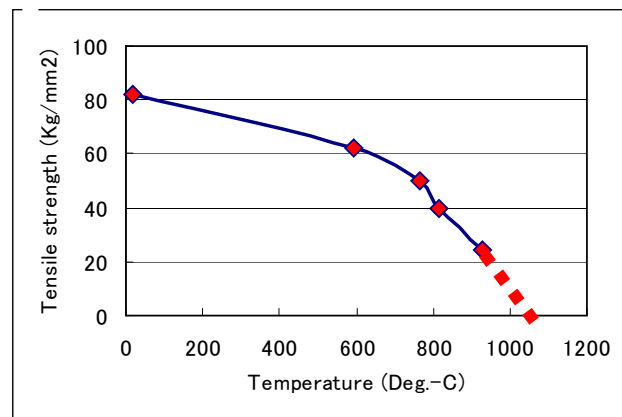


Figure 3. Tensile strength of Hastelloy-N

Based on the above consideration, limiting temperature may be somewhere between 900 and 1,000 deg-C. In order to decide the limiting point, the authors propose to apply the same concept of safety criteria for LWR fuel cladding tube to reactor vessel and piping of MSR. That is, current LWR licensing requires that the plastic strain of fuel cladding tube shall be less than 1% to prevent mechanical failure of fuel cladding tube. Assuming MSR's vessel temperature identical with fuel-salt, the allowable fuel-salt temperature should be less than the temperature, which causes plastic strain of 1% during 1 hour for primary loop boundary under the design load. Here, the authors conservatively chose the maximum duration to be 1 hour, which is long enough to shut down the reactor and take corrective actions. This is similar to the fuel rod cladding tube criteria of LWRs, although time endurance is not defined in LWR.

Using Larson-Miller plot of Hastelloy-N [1,6], the limiting temperatures can be estimated. For example, 930 deg-C for outlet and 790 deg-C for inlet fuel-salt in MSBR reactor vessel [8]. Violating this fuel temperature criteria, fuel-salt must be drained to the drain-tank shortly. The drain-tank of MSR can confine radioactivity, thus no radiation release will occur.

Of course, LWR fuels are discharged after 4-5 years even if AOT occurs, and they do not experience second AOT. On the other hand, reactor vessel and piping of MSR are used for several decades. This issue must be discussed in the future.

Another possible candidate may be oxidation criteria. The ASME pressure vessel standard requires the allowable oxidation, if it occurs. The report of Hastelloy-N manufacturer says that about 1,000 deg-C may be the limit, as is shown in Figure-4 [6]. But, since the primary loop of MSR is enclosed with inert gas such as nitrogen gas within the containment, oxidation will not occur. Therefore, oxidation criteria is not applicable.

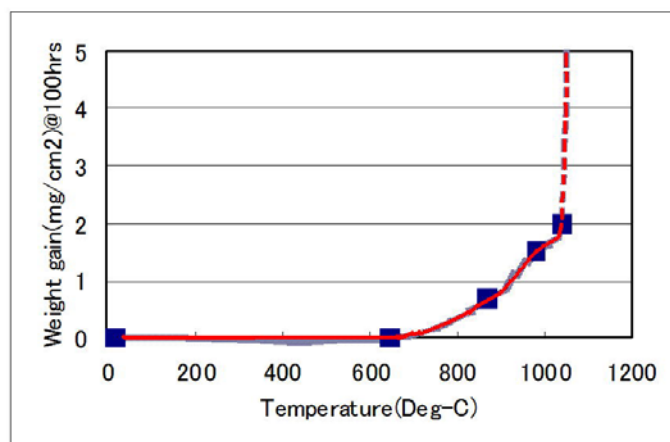


Figure 4. Oxidation resistance (weight gains at 1000hrs) for Hastelloy-N

3.2. Safety Criteria for Design Basis Accidents (DBAs) and Other Events

As for LWR, the objective of safety criteria for design basis accidents (DBA) is to prevent the occurrence of large release of radioactivity from the reactor, even if fuel failure occurs and radioactivity is released outside the fuel cladding tube. In order to prevent this situation, it is required that reactor is not damaged severely, reactor maintains coolable geometry, fuel enthalpy is lower than the limit, pressure of primary loop boundary is lower than the limit, pressure of containment boundary is lower than the limit, and radiation risk to the public is not severe. Based on the possible scenarios for MSR accidents, it is incredible that any DBAs will cause large radioactivity release. Therefore, safety criteria for DBAs should be investigated in the future.

Regarding the safety criteria for severe accidents (SA), which are beyond DBA, there is no global standard at this time. Also, SA is more incredible for MSR than DBA. Therefore, this issue should be also discussed in the future.

As for the limit for normal operation, criteria is defined for LWR, in order to prevent the violation of AOT criteria even if AOT occurs. MSR may also need similar criteria, for example criteria for maximum outlet temperature of fuel-salt at normal operation in order to satisfy the AOT criteria even if AOT occurs. This issue should be also discussed in the future.

4. CLASSIFICATION OF ACCIDENTS

The objective of accident analysis is to prevent or mitigate radioactivity release in case of accidents, which may affect human health. In MSR, this is caused by rupture or break of primary loop boundary, which is composed of a reactor vessel, pipes, pumps, heat exchangers and so on. The following two scenarios, as is described in 4.1 and 4.2, may cause rupture or break of primary loop boundary.

4.1. External Cause Accidents

External cause accidents are initiated by external events, such as earthquake, tsunami, flood, wind, fire, turbine missiles, terrorism and so on. In addition to these accidents, there are other standard accidents such as turbine trip accident, generator trip accident (load rejection accident), loss of external electric source accident, which are caused by initiating events outside the reactor

building but affect the reactor. These ten accidents are generic ones, and not discussed here.

4.2. Internal Cause Accidents

Internal cause accidents are caused by internal events, such as over-pressure, over-heat and other mechanical failures of the primary loop boundary. These are discussed in Section-5.

Of course, even if the above accidents occur, it does not mean direct release of radioactivity to outside of the reactor. Based on the defense-in-depth principle, there are containment and the reactor building, which prevent or mitigate radioactivity release, and therefore, these systems are sometimes called “mitigation system (MS)” in current LWR licensing. MS includes containment, reactor building, control-rod scram system, ECCS (Emergency Core Cooling System) and so on. Meanwhile, primary loop boundary such as a reactor vessel is called “prevention system (PS)”, whose failure initiates the accident.

5. ACCIDENTS TO BE CONSIDERED

The above internal cause accidents are categorized to the following four types (Table-2), which must be evaluated or analyzed in the framework of safety guidelines.

Regarding accidental causes such as over-pressure or over-heat, they are caused by power increase accident (5-1) or flow decrease accident (5-2), because temperature rise or enthalpy rise is proportional to power and inversely proportional to flow. In MSR, vapor pressure is very low, and over-pressure accident is incredible with one exception, which is described in Section 5.4.1. Besides these two accidents, fuel-salt leak accident (5-3) may be caused by other mechanical failures of the primary loop boundary. Therefore, the above three types of accidents must be considered. Of course, some of the second and the third accidents may cause reactivity increase as a result. Meanwhile, the first accident is usually called reactivity initiated accident (RIA), because it is initiated by the insertion of positive reactivity at first.

Besides these three categories, the fourth category, as named as “other accidents” (5-4), is considered. These accidents are mostly specific to MSR. In some cases, this category is just a cause of the above three accidents, and then these accidents may be re-categorized in the future.

Table 2. Accident category

| | | | |
|-------------------------------|-----|---|----------|
| 4-1) External Cause Accidents | | Accidents by external causes, such as earthquake, tsunami, etc.. | 10 cases |
| 4-2) Internal Cause Accidents | 5-1 | Power increase accident or RIA (Reactivity Initiated Accident) For example, control rod ejection accident. | 11 |
| | 5-2 | Flow decrease accident For example, fuel-salt pump stopping accident | 6 |
| | 5-3 | Fuel-salt leak accident For example, rupture of primary loop pipes | 1 |
| | 5-4 | Other accidents Mostly MSR specific. For example, molten salt freeze accident | 12 |

In this paper, the following typical MSR concept is assumed as is shown in Figure-5, which was originally proposed by ORNL for MSBR design [5], applying graphite moderator, single-fluid design, with gaseous fission product removal system. One big improvement from the original design by the authors is removal of online reprocessing equipment, in order to simplify the reactor [9,10].

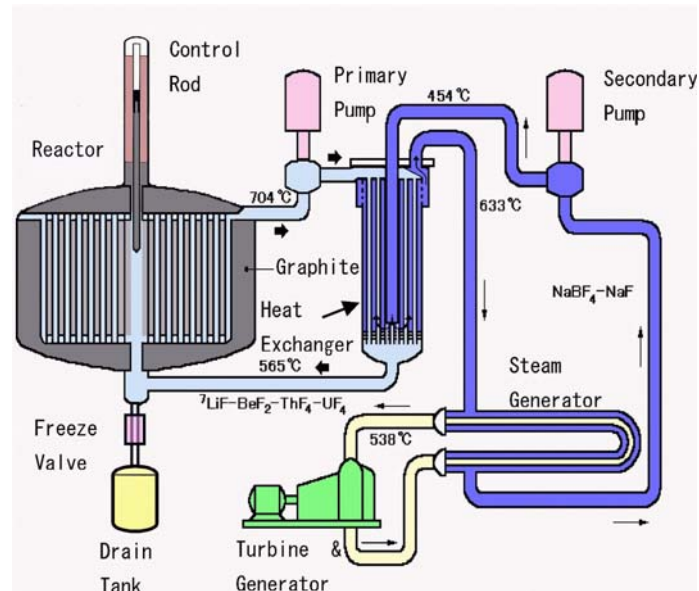


Figure 5. General concept of Molten Salt Reactor (Loops are redundant.)

5.1 Power Increase Accident or RIA (Reactivity Initiated Accident)

5.1.1. Control rod withdrawal/ejection accident

This is a most typical reactivity initiated accident (RIA). If we adopt a control rod made of neutron absorbing material, and when this control rod is inserted in operation and is withdrawn or ejected by some equipment failure or operator error, then RIA occurs. If we adopt a control rod made of graphite, which was proposed for MSBR [5], insertion of graphite control rod increases more neutron moderation and it may cause RIA. Owing to large negative reactivity coefficient of fuel-salt temperature, power excursion terminates, even if control rod scram function fails. Meanwhile, reactivity coefficient of graphite temperature is slightly positive, but this does not cause any problem, because heat transfer to graphite is slow. RIA was analyzed for MSBR as is shown in Figure-6 [11], and for small sized MSR: FUJI case [8].

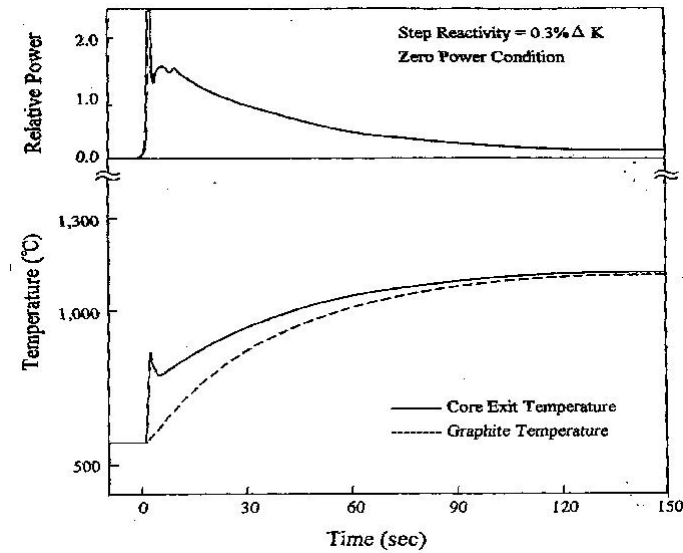


Figure 6. Typical RIA (Reactivity Initiated Accident)

Although delayed neutron fraction (β) of ^{233}U is smaller than ^{235}U , this does not affect dynamic behavior of MSR. In order to confirm this, the authors show one-point kinetic equations using adiabatic model without core cooling [12].

$$\frac{dn}{dt} = \frac{\rho - \beta}{\ell} n + \lambda \cdot C$$

$$\frac{dc}{dt} = \frac{\beta}{\ell} n - \lambda \cdot C$$

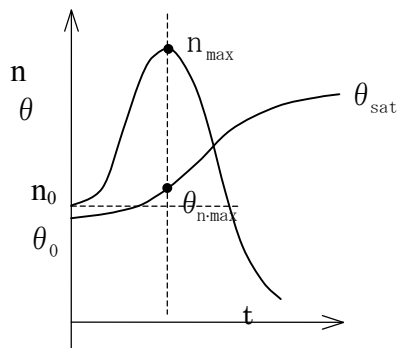
$$\frac{d\theta}{dt} = \frac{1}{C_p} n$$

C_p : Heat capacity

$$\rho = \rho_0 - |\alpha| \cdot (\theta - \theta_0) \quad \alpha : \text{Temperature reactivity coefficient}$$

ρ_0 : Inserted reactivity

For prompt critical situation in a short time, concentration of delayed neutron precursor “C” does not change and can be assumed as constant. Then, the above equation can be solved as follows.



$$n_{\max} = n_0 + \frac{C_p \cdot \rho_0^2}{2|\alpha| \cdot \ell}$$

$$\theta_{n-\max} = \theta_0 + \frac{2 \rho_0}{|\alpha|}$$

$$\theta_{\text{sat}} = \theta_0 + \frac{\rho_0 + \sqrt{\rho_0^2 + 2\ell \cdot |\alpha| \cdot n_0 / C_p}}{|\alpha|}$$

Figure 7. Number of neutrons and temperature after step reactivity insertion

It is clear that delayed neutron fraction does not affect the maximum neutron flux nor maximum fuel-salt temperature. On the other hand, MSR has a longer prompt neutron lifetime than LWR, and this fact mitigates the maximum neutron flux.

5.1.2. Cold-loop startup accident

Since reactivity coefficient of fuel-salt temperature is negative, if fuel-salt pump is inadvertently restarted from stand-by condition, and cold fuel-salt is injected into the core, then positive reactivity is inserted. However, lowest possible temperature of fuel-salt is the melting temperature, and the consequence is limited.

Besides cold-loop startup accident, if fuel-salt is more cooled than current status, similar consequence will occur as follows, although 5.1.3 and 5.1.4 are trivial.

5.1.3. Secondary-salt flow increase accident

5.1.4. Secondary-salt temperature decrease accident

5.1.5 Fuel-salt flow increase accident

If fuel-salt flow is increased, core temperature is decreased, and then reactor power is increased due to negative temperature coefficient. However, the reactor power is stabilized to a certain value, which is called “consistent power”, as is shown in Figure-8 [13].

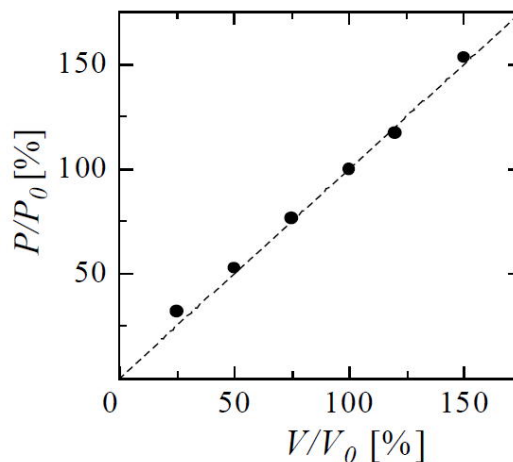


Figure 8. Reactor power versus fuel-salt flow

5.1.6. Fuel-salt filling accident

When fuel-salt is filled from a drain-tank to the core without any safety protection systems, criticality accident may occur. Since accident consequence depends on safety design, its design must be described.

5.1.7. Excessive fuel addition accident

MSR is equipped with a system that can adjust fuel-salt composition. If excessive fissile material is injected into fuel-salt, positive reactivity is inserted. Removal of thorium salt, which is neutron absorber, causes the same result. Therefore, safety design must be described.

5.1.8. De-pressurization accident

MSR has a slightly positive void reactivity coefficient of fuel-salt, and small amount of He-bubbles are circulating within the primary loops in order to remove gaseous fission products (Xe/Kr/T) from fuel-salt. Therefore, if de-pressurization occurs, bubbles become larger and positive reactivity is inserted, as is shown in Figure-9 [14]. De-pressurization may be caused by break of primary loop boundary, stopping of pumps and so on, because in normal operation, fuel-salt pumps give about 0.5 MPa (Mega Pascal) pressure to the primary loop. Malfunction in a He-bubble injection system may cause similar result.

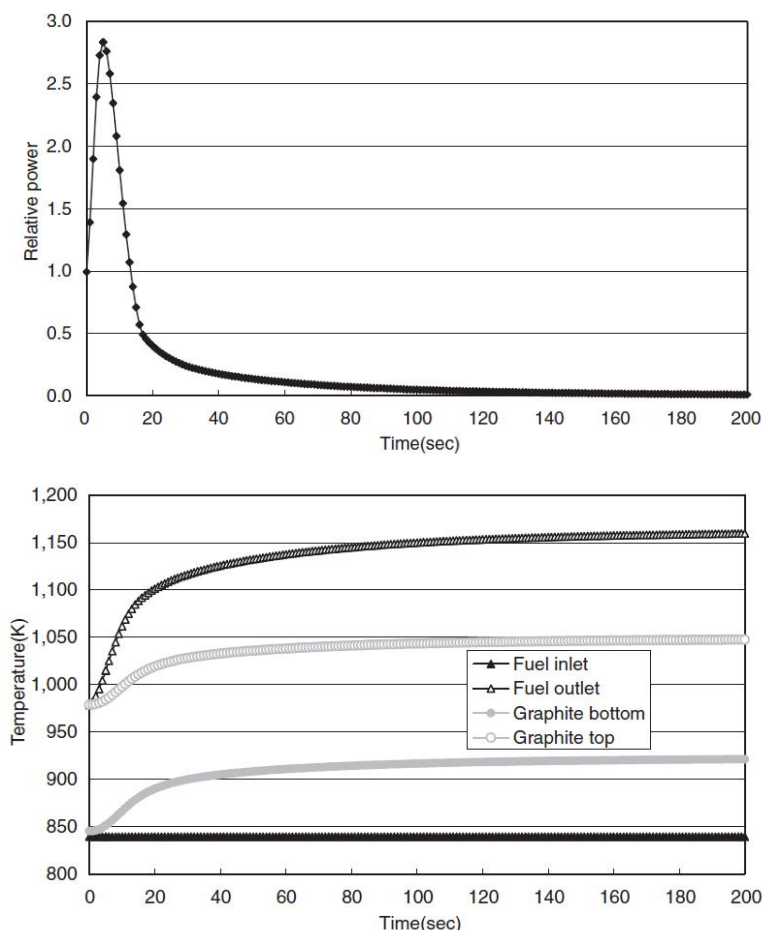


Figure 9. De-pressurization accident

5.1.9. Fissile precipitation accident

Fissile material such as uranium (U) has a high solubility in fuel-salt. However, if U precipitates and this U deposit is suddenly injected into the core, then positive reactivity is inserted. The cause of this accident may be invasion of moisture or oxygen in fuel-salt.

5.1.10. Graphite loss accident

If graphite breaks and flows out from the core, increased volume of fuel-salt may causes positive reactivity insertion. However, as is explained in 5.1.1, graphite control rod insertion causes positive reactivity by increasing neutron moderation. Therefore, loss of graphite will insert negative reactivity.

5.1.11. Fissile penetration to graphite accident

If uranium (U) penetrates to the surface of graphite, and if this U is suddenly released into fuel-salt, then positive reactivity may be inserted.

5.2. Flow Decrease Accident

5.2.1. Pump trip accident

If all fuel-salt pumps trip (stop), heat removal function is lost. (In LWR licensing, one pump trip is categorized to AOT, and all pumps trip is categorized to DBA.) Then, fuel-salt temperature increases. Also, delayed neutrons increase in the core when salt circulation stops, and it causes the same effect as positive reactivity insertion. This is because normally some of delayed neutrons are lost out of the core; for example, about 10% for FUJI-U3 case, and about 40% for MSBR design case. However, owing to the negative reactivity coefficient, its consequence is not as severe as pump seizure accident as is described in 5.2.2. In this pump trip accident case, control rods are inserted and nuclear fissions stop. In this situation, since pump has a free rotation, natural circulation is expected, as is shown in Figure-10 [15]. Installing small motor (pony motor driven by battery, if possible) to the fuel-salt pump is effective, as is demonstrated in the Japanese FBR “Monju”.

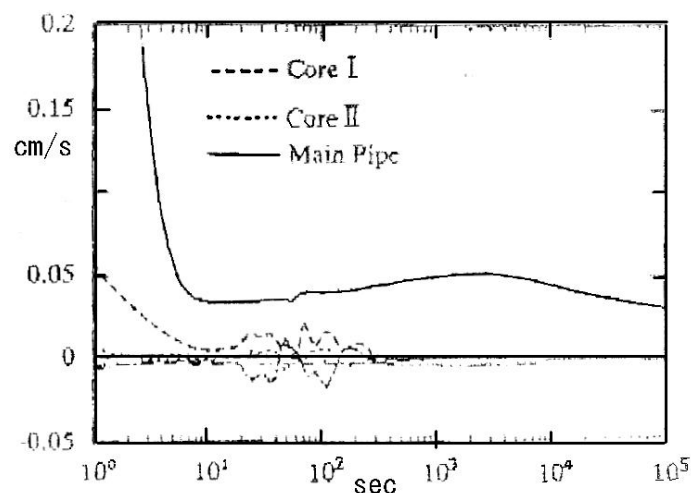


Figure 10. Fuel-salt flow under natural circulation in MSR

5.2.2. Pump seizure accident

If pump shaft is stuck, more severe situation than the above pump-trip accident occurs. Because fuel-salt flow becomes almost zero in a short time, and core-cooling function is lost, and then fuel-salt temperature increases. Therefore, numerical evaluation is required. Based on the author’s evaluation that if control rods are not scrammed, maximum fuel-salt temperature reaches about 900 deg-C after 300 seconds for all fuel-salt pumps seizure accident, as is shown in Figure-11 [16]. In the actual case that control rods are scrammed, consequence will be milder than this result. If this happens in all fuel-salt pumps, fuel-salt must be drained to a drain-tank.

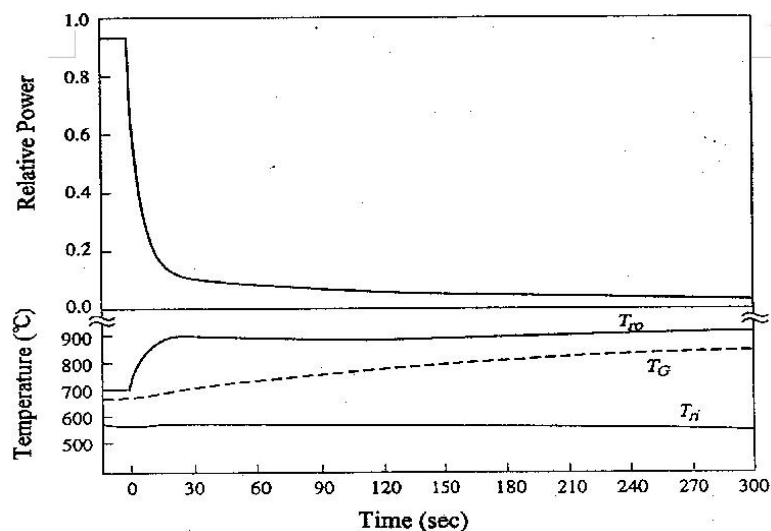


Figure 11. Pump seizure accident in MSBR

5.2.3. Flow blockage accident

If blockage occurs at anywhere in the primary loops, fuel-salt flow stops. The consequence is similar to pump trip/seizure accident. Total mechanical blockage is incredible, and in case 20 flow channels are blocked among totally 100 flow channels, the result is shown in Figure-12 [17]. Another possible cause may be fuel-salt freeze accident, which is discussed later (see 5.4.3).

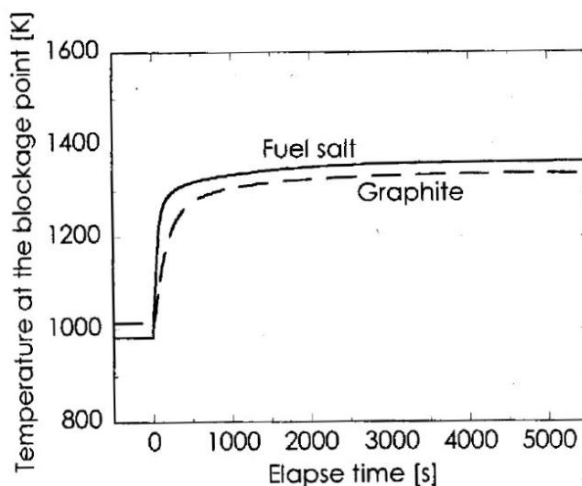


Figure 12. Flow blockage accident in MSR.

5.2.4. Loss of secondary-salt cooling accident

If secondary-salt pump trips or pump seizure occurs, fuel-salt is not cooled, and then fuel-salt temperature increases. In some severe cases, fuel-salt must be drained to a drain-tank. Loss of secondary-salt cooling will occur in other scenarios, such as rupture or break of secondary-salt loop, or steam-generator loop failure, and then same situation will occur.

5.2.5. Loss of decay-heat cooling accident (in core)

5.2.6. Loss of decay-heat cooling accident (in drain-tank)

After the reactor is shut down, fuel-salt must be cooled, because it has decay heat. This function is required in any cases, as far as fuel-salt stays in the core. Therefore, appropriate decay-heat cooling system must be equipped, unless fuel-salt is not drained. As is shown in Figure-10, natural circulation may be applicable for this purpose. If fuel-salt is drained to a drain-tank, cooling function must be provided. MSBR proposed a passive cooling system for the drain-tank [5].

5.3 Fuel-salt Leak Accident

5.3.1. Primary loop break accident

If rupture or break of primary loop boundary such as vessel, pipes, pumps, heat exchangers occurs by some reasons, then the integrity of primary loop boundary is lost, and fuel-salt will leak out. Of course, leaked salt is caught by a catch-pan, and collected in a drain-tank or an emergency drain-tank without passing freeze valve, as is shown in Figure-13 [5]. Regarding the rupture of heat exchanger, mixing of fuel-salt and secondary-salt must be evaluated. The causes of these accidents may be manufacturing flaw, excessive increase of boundary temperature, pressure increase by sudden boiling, corrosion, thermal stress cycling and so on.

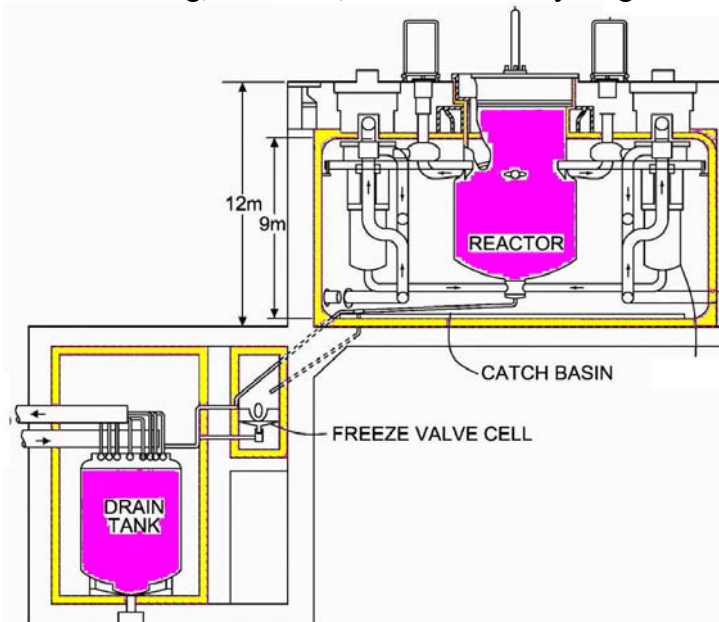


Figure 13. Freeze valve and drain-tank in MSBR

5.4 Other Accidents

5.4.1. Steam-generator break accident

As is described at first, vapor pressure of molten salt is very low, and over-pressure accident is incredible. The only one exception is steam-generator (SG) break accident. MSR applies super-critical SG, and its pressure is about 25 MPa, and higher than LWR case of about 7 MPa. If SG breaks, this high-pressured steam is injected into the secondary loop. In order to avoid the propagation to the primary loop, appropriate protection systems must be equipped, such as isolation valve or rupture disc.

5.4.2. Re-criticality accident in drain-tank

Since there is no graphite as neutron moderator in a drain-tank, re-criticality accident is incredible to occur. Possible cause may be isolation or precipitation of fissile material.

5.4.3. Fuel-salt freeze accident

Volume of frozen fuel-salt (at 20 deg-C) is about 15% smaller than that of molten fuel-salt (at 600 deg-C), mostly due to thermal expansion property of fuel-salt. For example, the formula for density of typical fuel salt is $3.628\text{g/cc} - 0.00066 \times \text{Temp}(\text{deg-C})$ [18]. Therefore, even if freezing occurs, it does not cause pipe break. Also, since primary loop is within the high temperature containment (see Figure-13 and 15), which is heated higher than freezing temperature of fuel-salt, fuel-salt freezing is incredible to occur. Of course, malfunction of the heating system at high temperature containment is to be considered, although this event will be gradual.

The most probable scenario is fuel-salt freezing at heat exchanger, because the inlet temperature of secondary-salt (ca. 450 deg-C) is lower than the freezing temperature of fuel-salt (ca. 500 deg-C). If one of fuel-salt pumps stops but secondary-salt pump does not stop, then fuel-salt is over-cooled, and starts freezing at heat exchanger (a kind of “over-cool” accident). This situation is similar to the above flow blockage accident.

In some cases, fuel-salt must be drained to a drain-tank. However, opening the freeze valve may take 5-10 minutes, if there is no heater to hasten melting of fuel-salt at the freeze valve. Therefore, faster mechanical valve or rupture disc may be required. Salt freezing may occur at the drain-tank, as is shown in Figure-14 [19].

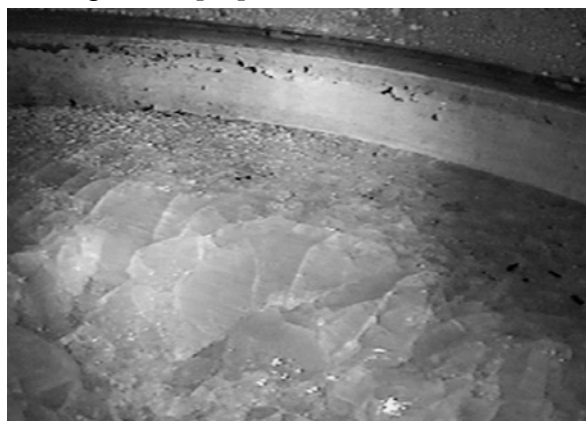


Figure 14. Frozen salt (FLiBe in MSRE tank)

5.4.4. Secondary-salt freeze accident

Similar accident as fuel-salt freeze accident (5.4.3) may occur at secondary loop. If the reactor is shut down and its power decreases, but if the steam-generator does not stop, then secondary-salt is over-cooled and will freeze (a kind of “over-cool” accident). The actual consequence must be evaluated, and in some severe cases, fuel-salt must be drained to a drain-tank.

5.4.5. Re-melt accident

Volume of molten fuel-salt (at 600 deg-C) is about 15% larger than that of solid one (at 20 deg-C), mostly due to thermal expansion of fuel-salt [18]. Therefore, careful re-melting is required to avoid unexpected pressure/stress increase in pipes or in tanks. This may happen on

secondary-salt loops too. Therefore, safety design must be described in the safety report for licensing.

5.4.6. Freeze-valve failure accident

Opening freeze valve is just to switch off a valve cooling system. Therefore, failure probability of freeze valve is very low, but it is not zero. Since freeze valve is the last countermeasure to keep the integrity of primary loop boundary, some verification is required, and/or faster mechanical valve or rupture disc may be required.

5.4.7. Graphite fire accident

It is incredible that graphite fire occurs in MSR. Because reactor-grade graphite has a high density and it does not burn without continuous external heat. In MSR, there is no heat source near the graphite, after fuel-salt is transferred to a drain/emergency tank. Also, there is no oxygen within the reactor vessel and containment, because inert gas is enclosed.

5.4.8 Wigner effect accident

Another cause of graphite fire is known as Wigner effect, which was discovered by Eugene P. Wigner (the first research director of ORNL), and actually Windscale fire accident occurred in 1957 [20]. However, Wigner effect occurs only at low temperature operation (< 200 deg-C), and MSR operates at higher temperature (> 500 deg-C) than this effect. Therefore, there is no possibility of Windscale type fire.

5.4.9. Off-gas system failure accident

Since gaseous Fission Products (FP) such as xenon/krypton/tritium, as shown in Table-3 [5], are removed from fuel-salt and collected in normal operation of MSR. Therefore, it must be confirmed which FPs will be collected by the FP gas removal system, and these FPs must be confined in secure tanks for enough decay periods.

Table 3. Gaseous FPs to be considered

| Nuclides | Half life of decay |
|-------------------|--------------------|
| ^3H | 12 years |
| ^{133}Xe | 5.3 days |
| ^{135}Xe | 9.1 hours |
| ^{137}Xe | 4.2 min. |
| ^{138}Xe | 17 min. |
| ^{85}Kr | 11 years |
| ^{87}Kr | 1.3 hours |
| ^{88}Kr | 2.8 hours |
| ^{89}Kr | 3.2 min. |

5.4.10. Reactor oscillation accident

Safety guidelines require that the reactor can be operated with enough stability, because reactor oscillation may occur even if reactivity coefficient is negative. One example is xenon oscillation in PWR, and another is flow instability in BWR.

As for xenon issue, gaseous FPs such as Xe/Kr are always removed from the core in MSR, and xenon oscillation does not occur.

Preliminary study on MSR controllability suggests that core can be stabilized within several minutes, when reactor power is changed by core flow from 100% to 50/75/125/150% [21]. But, wider analysis must be performed based on reactor control design.

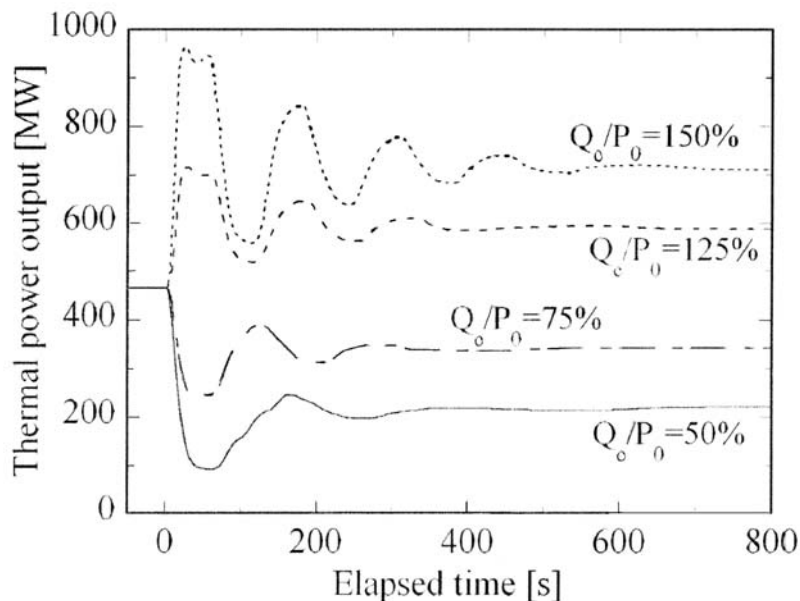


Figure 15. Reactor power behavior for core flow change

5.4.11. Fuel-salt or beryllium release accident

Any radioactive isotopes in fuel-salt are toxic to human health, and also Beryllium (Be) in fuel-salt has chemical toxicity. As is explained in 5.3.1, if there is a leak of fuel-salt, it is collected to a drain-tank, and finally it freezes (solidifies) to glassy material below its melting temperature. In any cases, fuel-salt will be confined within the containment. Therefore, release of radioactive elements of fuel-salt or Be to outside of the reactor building is incredible, but it must be confirmed.

5.4.12. Rupture of containment accident

MSR has three level containment safety, as is proposed in MSBR design and shown in Figure-15 [9,10]. First level is a reactor vessel and pipes made of Hastelloy-N. Second level is a high-temperature containment (HTC) composed of three layers, which contains a reactor vessel, pipes, and heat exchangers. Third level is a reactor building (RB) composed of two layers. In any of the above accidents, HTC and RB prevent or mitigate radioactivity release. Based on the above all scenarios, loss of HTC/RB integrity is incredible, but possible damages on HTC/RB must be evaluated in each accident analysis. As is explained in Section-4, HTC or RB is identified as part of the mitigation system, whose failure does not initiate accident, but its mitigation capability must be evaluated.

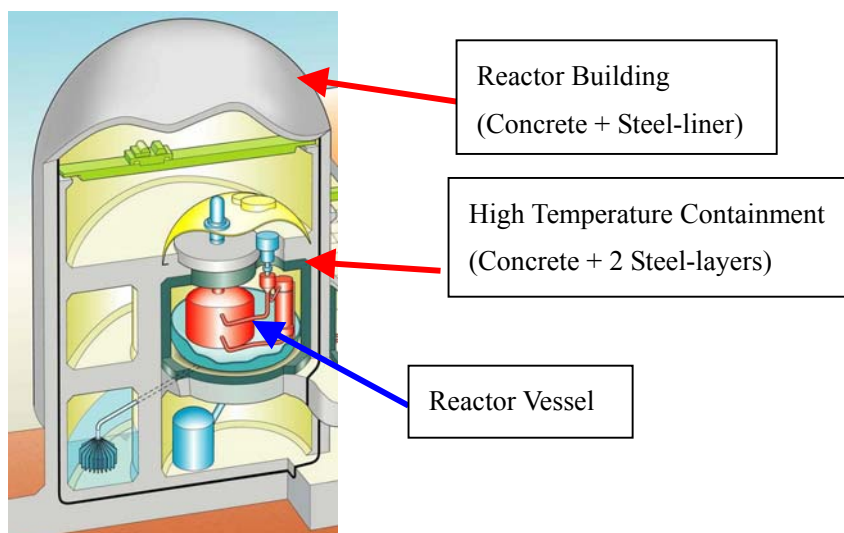


Figure 15. Containment function of MSR

5.4.13 Source Term Issue

Source term issue is considered not to be required in MSR at this moment because of the following reason.

Based on the NRC definition, “source term” is “Types and amounts of radioactive or hazardous material released to the environment following an accident”. In order to evaluate the source term, it is necessary to assume the accident scenario how the radioactive materials in nuclear fuel are discharged to outside of the reactor. For example, types of radioactive materials if they are gas or solid or liquid, initiating events of accidents, effectiveness of defense barriers, mitigation systems and so on, must be assumed. In LWR, some amount of radioactive materials is discharged to the environment in DBAs, and large amount of radioactive materials are discharged in severe accidents [22].

In MSR, there will be no release of radioactive materials to the environment in DBAs, except the failure accident of off-gas system. Actually, based on the above preliminary evaluations for internal cause accidents such as pipe break accident, rupture of containment is incredible to occur, and hence there will be no release of radioactive material to the environment in DBAs.

As for the “Station Black Out” (loss of all AC electric power), passive cooling system, which does not need electricity, is proposed for drain tank cooling [5][23].

As a result, simultaneous loss of cooling function and containment function will not occur except the external cause accident by terrorism or missiles. Since this paper describes only AOT and DBA and does not refer severe accident, if source term issue is considered same as LWR, it is not required to include source term issue in MSR.

Although source term issue is not required to be included at this moment, but for future study, preliminary prospect on simultaneous loss of cooling function and containment function is discussed in this section as follows, because unfortunately this happened at Fukushima accident in 2011.

In MSR, since gaseous FPs such as Xe and Kr are always removed from the fuel salt, instantaneous release of radioactive materials will not occur even in simultaneous failure of reactor vessels/pipes and containment. In this scenario, fuel salt is liquid state, and most of radioactive materials will stay within fuel salt as an ion form, but their behavior must be confirmed. Some of them such as Iodine and Cesium, which were dominant source term at

Fukushima accident, are described in the ORNL report [24]. After decay heat decreases, fuel salt solidifies and all radioactive materials will be confined.

In this paper, online reprocessing is not adopted, but if it is included in the design, the above discussion must include its safety, as is discussed in the reference [25].

6. CONCLUSION

Regarding the safety criteria for MSR accident analysis, based on the allowable stress and tensile strength of Hastelloy-N, temperature between 900 and 1,000 deg-C will be applicable for long-time limitation of the fuel salt. But, for short-time events such as abnormal operating transients, the authors propose to limit the fuel-salt temperature based on the plastic strain of 1% during 1 hour for primary loop boundary under design load.

Safety criteria against design basis accident (DBA), severe accident beyond DBA, and normal operation should be discussed in the future.

As for the safety guidelines for MSR accident analysis, the authors showed 40 possible accidents for MSR. They were classified to external cause accidents and internal cause accidents. We investigated the latter ones, and categorized them to four types, such as power excursion accident, flow decrease accident, fuel-salt leak accident, and other accidents mostly specific to MSR. Each accident was described shortly with some numerical results by the authors. In several accidental scenarios, fuel-salt must be transferred to a drain-tank, and this system assures high safety of MSR. However, its consequence depends on freeze valve function, because its operation is slow, and this means that some verification is required. Also, some other accidents need quantitative evaluation. As for the source term issue, it is not required to be included at this moment, but preliminary prospect for future study was discussed. These guidelines described here may be varied for different designs, and should be improved depending on the progress of design. Also, PRA (Probabilistic Risk Assessment) is required at detail design stage [26].

As a summary of this paper, it can be concluded that MSR has superior safety, and it may be concluded that MSR has an intrinsic safety after completion of these evaluations.

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