A Risk-Informed Approach in the Design of a Molten Salt Reactor

Diego Mandelli, Peilai Zhang, Tunc Aldemir, Richard Denning,

The Ohio State University, Nuclear Engineering Program, 201 W. 19th Ave, 43210 Columbus (OH)
Email: mandelli.1@osu.edu

1. INTRODUCTION

A Molten Salt Reactor (*MSR*s) concept is one of the options selected for evaluation in the international Generation IV nuclear reactors [1] reactor development program. These systems are particularly attractive because they make available the world's extensive thorium resources as a source of fission energy. The MSR concepts utilize fuel in liquid form instead of the solid fuel pellets [3]. A breeding ratio greater than one can be achieved in a seed-blanket configuration with ^{233}U as the fissile material and ^{232}Th as the fertile material. On-line reprocessing is an essential aspect of the design.

A preliminary design study has been undertaken at the Ohio State University for a two fluid *MSR* [5]. This paper demonstrates how the safety and risk analyses were carried out as an integrated element of the design process.

The reactor vessel (see Fig.1) contains both the blanket circuit (a mixture of ^{232}Th and FLiBe) and the fuel (mixture of ^{233}U and FLiBe) circuit. The reactor core consists of a large number of small tubes containing fuel salt surrounded by the blanket salt.

The heat is removed from both the fuel and the blanket circuits through the fuel and the blanket HXs (heat exchangers) respectively on an intermediate loop (see Fig.1). Furthermore, this heat is discharged to a Brayton cycle through an intermediate circuit which contains FLiBe salt (see Fig.1).

The Reprocessing system is responsible to control the chemical composition of the fuel and blanket salt and to refuel the fuel circuit with the ^{233}U generated from the neutronic absorption of ^{232}Th nuclei in the blanket circuit (see Fig.1).

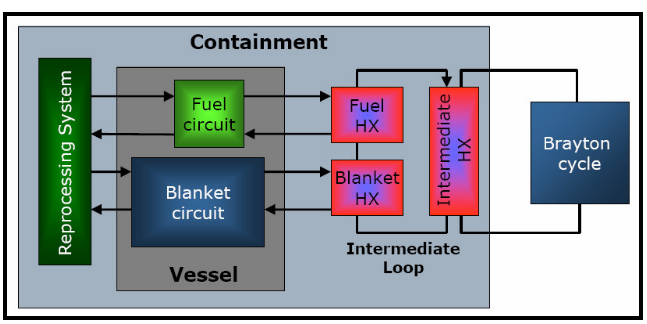

Fig.1. MSR Reactor System

2. PROPOSED METHODOLOGY

The concept for the design process involves a series of systematic procedures which are able to follow changes in the design phase of the project without involving radical changes in the safety analysis itself while maintaining high level risk objectives. Figure 2 shows how these procedures (represented in terms of blocks) are connected.

The methodology has four steps (see Fig.2):

1. Definition of the accident scenarios given the layout of the system.
2. Construction or update of the *FMEA* (Failure Modes and Effects Analysis) chart based on the current design concept.
3. Design (or update of the design) of safety systems.
4. Event Tree (ET)/Fault Tree (FT) modeling and the semi-quantitative assessment of critical scenarios.

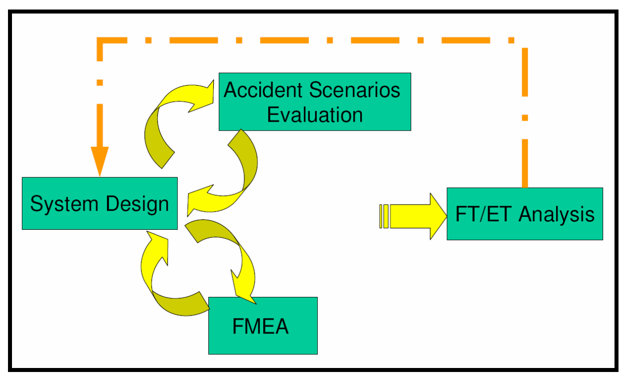

Fig.2. Interaction between Safety Analysis and System Design

The driving idea is develop the capability for risk informed design when components are added to or deleted from during the design process without involving major changes in the system safety analysis. After the identification of the initiating events of interest for accident scenario analysis (Section 2.1), a FMEA for each component (e.g. a valve or also an entire new sub system) is performed (Section 2.2) to identify its impacts on the system behavior. The safety system is updated (Section 2.3) if the introduction of the new component adversely affects the system behavior. Finally, an ET/FT analysis is performed [7] (Section 2.4) to quantify the impact. The process is repeated until the ET/FT analysis yields satisfactory results.

2.1 Accident Scenario Analysis

The starting point for the identification of initiating events is based on Regulatory Guide 1.70 [2]. (Standard Format and Content Analysis Reports for Nuclear Power Plants, *LWR* edition) which lists the initiating events that have to be analyzed.

Since RG 1.70 refers to light water reactors (*LWRs)*, some modifications have been made based on the differing features of the *MSR*s. The initiating events under consideration in this study are the following [6]:

1. Decrease in intermediate circuit temperature;
2. Decrease in fuel flow rate;
3. Decrease in blanket flow rate;
4. Decrease in intermediate circuit flow rate;
5. Reactivity and power distribution anomalies;
6. Decrease in fuel salt inventory;
7. Decrease in blanket salt inventory;
8. Decrease in intermediate loop inventory;
9. Radioactive release from a subsystem or component.

For each of these initiating events, and given the design of the system, a list of possible causes and consequences is developed in addition to the FMEA (Section 2.2).

2.2 Failure Modes and Effect Analysis

The purpose of the *FMEA* chart is to classify and analyze the possible failure modes of the components of the system and determine the effects of these failures on the overall system. This is an activity that is often used by PRA analysts as a basis for event tree development. For each component, the following are considered:

- A list of all the possible failures modes (e.g. crack or breaking of fuel tube inside the vessel).
- An analysis of the possible detection methods (e.g. Thorium detector installed in fuel circuit and ^{233}U detector installed in the blanket loop).
- An analysis of the impact of this failure (e.g. mixing of fuel and blanket inside the vessel).

2.3 Safety System Design

The accident scenario (see Section 2.1) and the *FMEA* analysis (see Section 2.2) presented earlier constitute the starting point to identify and to design safety systems. These systems have to be able to drive the primary circuit (i.e. the blanket and the fuel salt) into a safe and stable state without radioactive releases outside the containment.

Essentially, the functions and the number of these systems have been such that in each accident scenario (see Section 2.1) they:

- prevent formation of a critical fuel configuration, and,
- provide heat removal from the fuel and blanket salts,
- provide heat removal to the ultimate heat sink, and,
- contain radionuclide releases.

Figure 3 shows the proposed safety systems. In order to provide redundancy and diversity, two reactor protection systems (RPS) have been developed. The first line RPS (i.e., Main *SCRAM* System) consists of three systems (see Fig.3):

- Fuel Drain System,
- Blanket Drain System,
- Intermediate Salt Drain System.

Each of these systems is responsible to drain the fuel, blanket and intermediate salts into tanks using both passive systems such as freeze valves (i.e. plugs which melt when the temperature of the salt reach the melting point of the plug) and active components (i.e. motor driven pump). The backup RPS (i.e., Auxiliary *SCRAM* System) injects neutron poisons in the fuel salt circuit.

A dual containment system concept has been developed, which includes an external shield wall to protect against external threats. In fact, the containment system consists of an Inner and an Outer Containment as well as systems which provide cooling to the SCRAM systems and to containment itself (i.e. the Containment Cooling System and the Concrete Cooling System respectively, shown in Fig.3). Passively safe features are employed in the design to the extent practical [4].

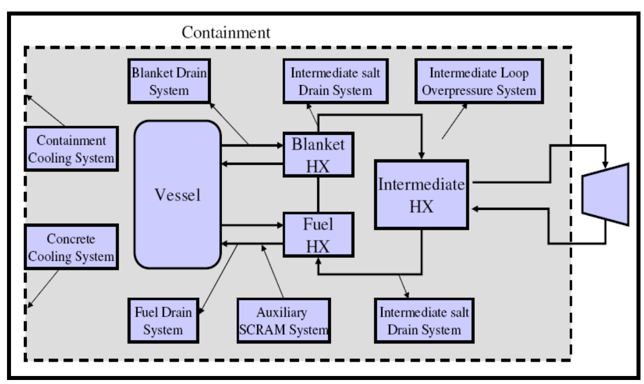

Fig.3. Proposed Safety Systems

2.4 Risk Assessment

The last step of the methodology is the evaluation of the risk during the accident scenarios given the overall system (see Fig.1) and the safety systems (see Section 2.3 and Fig.3).

As shown in Fig.4 for the Fuel Drain System, for each safety systems presented in Section 2.3, the corresponding fault tree is developed [7].

For a typical *LWR* the Top Event is core damage. Since the system under consideration differs from a *LWR,* the Top Event has to be changed. The fuel is in liquid form and melting of the fuel is not of concern in this system. From the initiating events listed in Section 2.1, two scenarios are identified that can lead to radioactive releases outside the Inner Containment:

- reactivity excursion in the fuel salt,
- the safety system does not provide sufficient cooling of the Inner Containment.

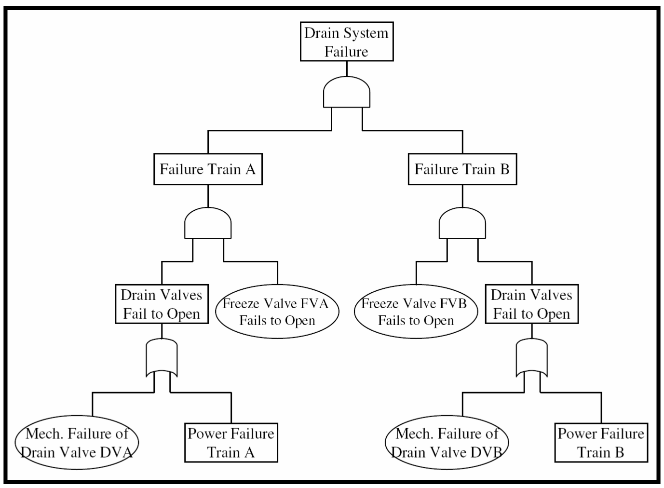

Fig.4. Fault Tree for the Fuel Drain System

For each accident scenario (see Section 2.1), the corresponding ET [7] has been developed, as shown in Fig.5 for the Decrease of fuel inventory.

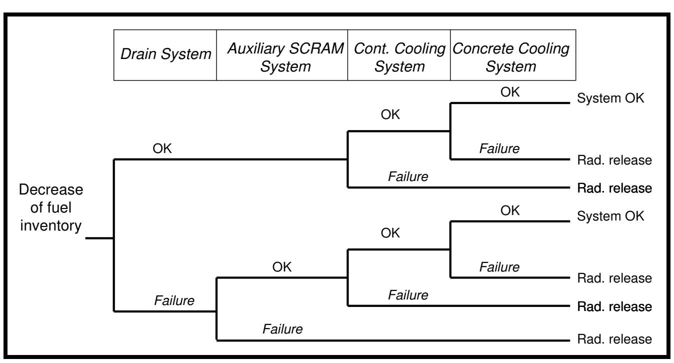

Fig.5. Event Tree for the Decrease in Fuel Inventory Scenario

3. CONCLUSION

This paper proposes a procedure for the risk informed design of Generation IV reactors. The objective of the design process presented is to assure the maintenance of high level risk goals as the design is optimized to attain operational performance goals.

Although no consequence frequency quantification was performed, the implementation of the procedure on the *MSR* concept in Fig.1 led to improved SCRAM and containment designs through the identification of large loss of fuel and blanket scenarios that would have been difficult to identify without the proposed procedure.

REFERENCES

[1] *The U.S. Generation IV Implementation Strategy*, U.S. Department of Energy, Office of Nuclear Energy, Science and Technology, September 2003.

[2] *Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, LWR Edition*, Regulatory guide 1.70 (Revision 3), Office of Standards Developments U. S. Nuclear Regulatory Commission, November 1978.

[3] *Two fluid molten salt breeder reactor design study*, R.C. Robertson, O.L. Smith, R.B. Briggs, E.S. Bettis, ORNL 4528, Oak Ridge National Laboratories, 1970.

[4] *Physical properties of molten salt reactor fuel, coolant and flush salts*, S. Cantor, J.W. Cooke, A.S. Dworkin, G.D. Robbins, R.E. Thoma, G.M. Watson, ORNL-TM-2316, Oak Ridge National Laboratories.

[5] *Summary of the objectives, the design and a program of development of molten salt breeders reactors*, R.B. Briggs, ORNL-TM-1851, Oak Ridge National Laboratories, 1967.

[6] *Analysis of filling accidents in MSRE,* J.R. Engel, P.N. Haubenreich, S.J. Ball, ORNL-TM-497, Oak Ridge National Laboratories, 1966.

[7] *Reliability and Risk Analysis*, N.J. McKormick, Academic Press, 1981.