ISSN PRINT 2319 1775 Online 2320 7876
Research Paper , Journal Volume 10, Iss 08, 2021

A Review of Analytical Approaches to Sodium Aerosols in Fast Reactors

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1 INTRODUCTION

The concept of An Analytical Approaches On Sodium Aerosols For The Fast Reactor. Fast reactors were one of the first types of reactors built. New SFR designs have been created for the Generation IV Nuclear Energy Systems project. The goals of these new designs are to explore non-traditional applications of nuclear energy while developing new reactor designs that meet the demand for clean and reliable power generation and also focus on enhanced safety and the reduction of cost and proliferation risks. Many of these new reactor concepts involve reactors that use recycled fuels or metallic fuels, like the SFR [1].

Currently there is not a licensing process for SFRs because there is not a regulatory experience base comparable to the extensive safety reviews that have been performed for Light Water Reactors (LWR). Nor are there safety analysis tools availablewith the pedigree required to support the development of a safety case. The purpose of the research for this thesis is to make realistic assessments of the offsite consequences of characteristic severe accident scenarios for SFRs. Because there is no integrated code package for the analysis of SFR scenarios analogous to the MELCOR package used for the analysis of severe accidents in LWRs [2], it was necessary to patch together results from separate computer codes and to write a code to specifically treat radionuclide release and transport in the reactor coolant system.

2 BACKGROUND

2.1 Research Objectives

This work is part of school of Applied Science, Sanskriti University, Mathura research title An Analytical Approaches On Sodium Aerosols For The Fast Reactor. It is a collaboration of efforts from the Department of Physics, M.V. College, Buxar, VKSU Ara State University of Bihar. Intended to address the challenges of developing future SFR system with a level of economic competitiveness. SFR systems with a level of economic competitiveness. Specifically, this research focuses on the development of methods for the minimization of power generation costs for a Generation IV (Gen IV) sodium-cooled fast reactor (SFR) within constraints of acceptable safety and proliferation resistance. The goals of the Gen IV Nuclear Energy Systems include exploring non-traditional applications of nuclear energy in new reactor designs that will meet the demand for clean and reliable power generation. Gen IV reactors will have to be economically competitive. They will also have to be at least as safe as the Gen III+ plants currently undergoing design certification and provide an acceptable level of proliferation resistance. Many of these new reactor concepts that are being developed and researched involve reactors that use recycled fuels or metal fuels, like the SFR [1]

2.2 Reference SFR Design

With public and political resistance to a national radioactive waste repository in the United States, consideration has been given in recent years to develop reactors that can convert long-lived radioisotopes found in spent nuclear fuel to short-lived ones.

SFRs could play a large part in the burning of actinides found in spent light water reactor (LWR) fuels. By burning a substantial portion of the spent fuel not only would the amount of waste needing to be stored in a repository be drastically reduced, but a considerable amount of energy would be produced as well [3].

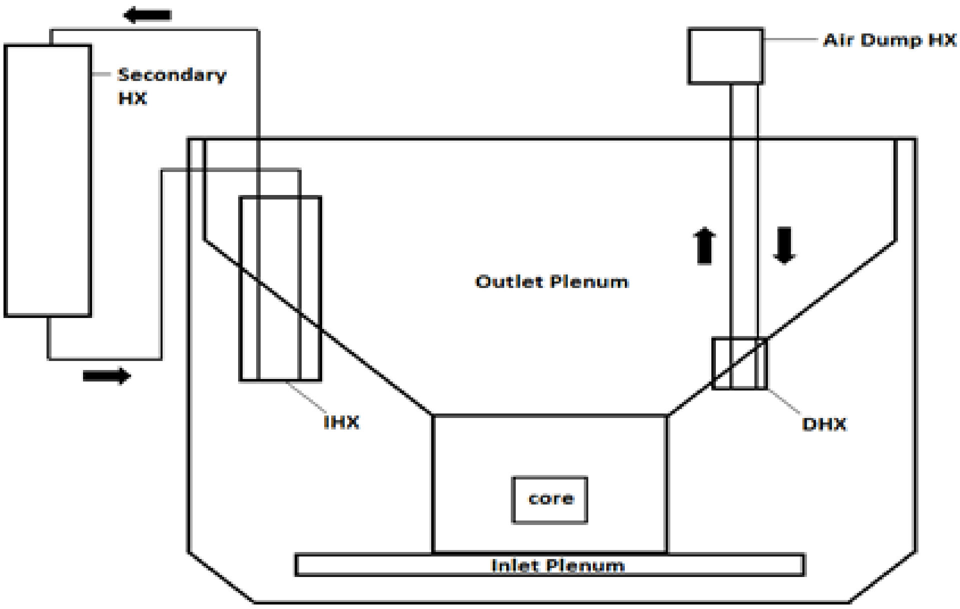

Figure 1: ABR-1000 Reactor Design [5]

SFR Severe Accident Scenarios

The most important aspect of nuclear reactor design to consider is the overall safety of the facility. Despite all of the precautionary design features put in place in a nuclear reactor, accidents can still occur. In comparison to LWRs, SFRs have some inherent safety advantages. The principal advantage is the low pressure of the primary system. The likelihood of a loss of coolant accident and core uncovery in a pool-type SFR is extremely small. SFRs also do not have to contend with depressurization loads on the containment or the potential for pipe whip. Transient tests in the EBR-II reactor also demonstrated the ability of inherent shutdown mechanisms to enable a metal-fueled, pool-type SFR to withstand a broad range of "unprotected" transients [5][1]. Sodium is an excellent coolant. However, one of the safety disadvantages of SFRs is that sodium reacts exothermally with both water and air. The other safety issue is that SFRs tend to have a positive void coefficient. Because compaction of SFR fuel also increases reactivity, the potential exists for an

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energetic event in which local fuel compaction and voiding of sodium propagate across the core. Within the context of the broader NERI project, the objective of the work described in this thesis was to estimate the magnitude of offsite consequences for a variety of characteristic severe accident scenarios. The emphasis in this thesis is on containment processes associated with loads that could threaten containment integrity and radionuclide transport and deposition processes that Unprotected transients are those in which the reactor protection system fails. affect the offsite consequences. The MELCOR computer code is the principal computercode used for these analyses.

The primary system boundary is defined by the reactor vessel and a deck structure above the pool. The deck structure provides shielding for people working in the containment building. Rotatable plugs penetrate the deck structure to enable refueling activities. Polymer seals and the boundaries of the plugs prevent leakage from the primary system. The reactor containment structure serves as the final barrier to the release of radioactive material. Since the offsite dose will be negligible if the primary system remains intact during a severe accident, the research performed for this thesis only considered scenarios where the primary system was failed [7].

Other advantages of the SFR design include the ability to use metallic fuels as well as oxide fuels. The metallic fuel, made up of U-15Pu-10Zr, has high thermal conductivity which, in combination with the good heat transfer characteristics of sodium, results in low operating temperature of the fuel. The metallic fuel has a lower melting point than that of LWR fuels leading to an inherently smaller release of radionuclides during melting. It also has a reduced positive Doppler reactivity feedback response during cool down transients [2]. The large mass of sodium in a pool-type SFR provides substantial thermal inertia in loss of heat removal accidents. It also results in the scrubbing of radioactive material released from the fuel in a severe accident.

Accidents that could occur in an SFR and result in an offsite dose consequence would have to experience multiple failures. In LWRs much of the accident concerns stem from a loss of coolant accident resulting in fuel melt. In general, if there is a fuel melt scenario in a LWR the release of radionuclides occurs in four phases involving gap release, the dominating in-vessel release, release from core concrete attack, and a delayed release from residual fuel and fission products deposited inside the reactor pressure vessel [5]. Although there is some activation of water as it flows through the core of an LWR, the half-life of the activation product is short. In an SFR the sodium coolant is activated as it passes through the core. The extent of activation is high enough that in an accident involving the release of radionuclides from containment the contribution to dose from the primary system sodium must also be taken into account.

Although the gap release tends to be minor in comparison to the subsequent in- vessel release for LWRs, the gap release can dominate consequences in an SFR accident. During operation a large fraction (up to 75%) of the noble gases and volatile radionuclides can be released from the fuel. The noble gases will migrate to the gas plenum. Cesium is likely to be captured in the sodium bond of the pin or to be deposited on surfaces as well as to be airborne in the gas plenum at low concentration. If there is a fuel pin failure, the radionuclide gases form gas bubbles

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and are swept out of the pin.

Bond sodium and molten fuel will also be swept into the channel. The fuel is likely to be carried out of the channel and freeze onto structures. Since the temperature of the molten metallic-SFR fuel is much lower than the LWR fuel and the SFR fuel is molten for much less time, lower releases and offsite consequences are expected in SFRs than in LWRs.

Two general primary system failure modes were investigated for this research: a limited failure scenario where a major seal failure occurs in the deck structure separating the sodium pool and the containment and a gross failure where the sodium pool is open to the containment, with no deck structure. For a gross primary system failure there would be convective heat transfer from the sodium pool surface to the atmosphere of the containment with a large rate of sodium vaporization of 17 g/s. For the limited seal failure mode it was assumed that there would be a two-way flow of air flowing down one part of the seal from the containment and a mixture of vapors from the cover gas region above the pool flowing up another part of the seal back into the containment [7]. This flow would be driven by natural circulation at a rate of 1.3 g/s and would result in contamination of the containment atmosphere with radionuclides from the cover gas of the sodium pool. Both of these scenarios were also modeled for situations in which the containment was both intact and failed. An intact containment structure was assumed to have a modeled leakage path area of 3.86E-5 m^2 corresponding to 1 volume percent leak per day at one atmosphere overpressure. The failed containment structure was modeled with a 1 m^2 hole to the environment

3 CONCLUSIONS AND FUTURE WORK

The first activity performed as part of this Master's Thesis effort was to reformulate the RCS computer code in order to decrease the running time by limiting the number of calculations performed using the smaller time step. This thesis is mostly focused on containment processes associated with the transport, deposition and release ofradionuclides to the environment using the MELCOR computer code. This research was performed to help reach the goals of developing a risk-informed approach to the design optimization and licensing of a sodium-cooled fast reactor. This was completed by analyzing various accident scenarios and their subsequent offsite dose consequences.

The methodology used to analyze these accidents includes a variety of computer codes. The ORIGEN2 code was used to determine the initial amount of radionuclides in the SFR at the time the accident is initiated. The SAS4A computer code was used to examine the transient behavior of accident scenarios and to determine the conditions in the fuel and the time of severe fuel damage. The RCS code was developed at OSU to analyze radionuclide releases from the fuel and observe how they are transported within the primary system. MELCOR was then used to investigate the containment radionuclide transport and deposition, and environmental releases. Offsite dose consequences are calculated using a spreadsheet with an algorithm based on the Regulatory Guide 1.145 approach to determining a 95th percentile meteorology. The WinMACCS code was also used to determine the probability of early fatality consequences within one mile from the site boundary and latent cancer fatalities within ten miles.

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