Annals of Nuclear Energy 64 (2014) 220–229

Contents lists available at ScienceDirect

Annals of Nuclear Energy

journal homepage: www.elsevier.com/locate/anucene

Severe accident source terms for a sodium-cooled fast reactor

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ARTICLE INFO

Article history:
Received 10 April 2013
Received in revised form 4 October 2013
Accepted 9 October 2013
Available online 5 November 2013

Keywords:
Source terms
SFR
Metallic fuel

ABSTRACT

In order to support the demonstration of a risk-informed approach to the design optimization of a sodium-cooled fast reactor (SFR), it was necessary to make realistic estimates of the consequences of severe accident scenarios. This paper describes the database, models, and assumptions used to estimate the offsite consequences of characteristic severe accident scenarios. As required for comparison with the NRC's technology neutral framework limit curve, the offsite dose at one mile from the plant boundary is calculated using conservative meteorology.

The reference plant design is a 1000 MWt pool-type design with metallic fuel. Because an integrated analysis tool comparable to MELCOR does not exist for SFR accident scenario analysis, it was necessary to write a computer code that would assess release of radionuclides from the fuel and transport within the reactor primary system and to link those analyses with results from existing computer codes that assess the dynamic response of the reactor, containment thermal–hydraulics, and radionuclide transport processes within the containment. The analyses indicate that the offsite source terms for SFR severe accident scenarios tend to be small because of the low melting temperature of the fuel, likelihood of significant retention of fission products within the sodium pool, augmentation of containment deposition processes by interaction with sodium oxide aerosols, and small driving force for release from the containment to the environment. A number of major sources of modeling uncertainty are identified as requiring further development effort. An integrated modeling capability, similar to the MELCOR code, is required to direct further research and ultimately to support the licensing of an SFR.

1. Introduction

The objective of this work is to identify and assess the processes that affect the magnitude of offsite consequences for severe accident scenarios in metal-fueled, pool-type Sodium-Cooled Fast Reactors (SFRs). The effort was performed within a Nuclear Energy Research Initiative (NERI) grant, funded by the US Department of Energy (DOE), to develop methods for a risk-informed optimization of fast reactor designs. The major participants in the NERI study were Massachusetts Institute of Technology, The Ohio State University, and Idaho State University. The overall project is described in the paper, "Investigation of Risk-Informed Innovations to Improve Sodium-Cooled Fast Reactor Economics, Safety, and Non-Proliferation Resistance" (Apostolakis et al., 2011). This paper describes the methods developed and results obtained for estimating the release of radionuclides from fuel, transport within the primary system, release to containment, containment deposition, release to the environment and assessment of offsite consequences.

The state of understanding of light water reactor source term phenomena is much more advanced than that for SFRs. Following the Three Mile Island Unit 2 accident the US Nuclear Regulatory Commission (NRC) undertook a major research initiative to develop the capability to estimate the release and transport of radionuclides in severe accidents (Silberberg et al., 1986) and the NRC adopted an improved source term for regulatory purposes (Soffer et al., 1995). In recent years, through cooperative international research programs, these methods have been further improved and validated. The MELCOR/MACCS (Gauntt et al., 2005 and Chanin and Young, 1998) code combination enables a consistent, integrated analysis of severe accident scenarios for Light Water Reactors (LWRs) from the initiating transient to the calculation of offsite consequences. In 2010, the Department of Energy performed a state of the art assessment of SFR accident analysis capability. The results of the panel addressing source term behavior (Powers et al., 2010) identify a number of important phenomena for which there is a high need for additional experimental research:

- High temperature release of radionuclides from fuel during an energetic event.
- Energetic interactions between molten fuel and sodium coolant and associated transfer of radionuclides from the fuel to coolant.

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http://dx.doi.org/10.1016/j.anucene.2013.10.007

- Entrainment of fuel and sodium bond material during the depressurization of a fuel rod with breached cladding.
- Rates of radionuclide leaching from fuel by liquid sodium.
- Surface enrichment of sodium pools by dissolved and suspended radionuclides.
- Thermal decomposition of sodium iodide in the containment atmosphere.
- Reactions of iodine species in the containment to form volatile organic iodides.

Other high importance phenomena identified as having a medium priority for research include: bubble transport and radionuclide scrubbing, chemical forms in the fuel and fuel-cladding gap, chemical activities of radionuclides in the fuel and in sodium, and rates of fuel dissolution in sodium. Our efforts to quantify radionuclide release and transport in characteristic SFR severe accident scenarios provide some additional insights regarding the relative importance of these modeling deficiencies.

2. Reference design features

The reference plant for this study is a 1000 MWt, 380 MWe pool type reactor with metal fuel. It is very similar to the Advanced Burner Reactor (ABR) design developed by Argonne National Laboratory (ANL) (Yang et al., 2007) and in many respects to the PRISM reactor design (General Electric, 1986). As illustrated in Fig. 1a barrier (redan) separates an upper pool of sodium from a lower pool. Four intermediate heat exchangers transport heat to intermediate sodium loops. There are also two independent passive decay heat removal systems, a Direct Reactor Auxiliary Cooling System (DRACS), which removes heat from the pool and a passive secondary loop heat removal system. The DRACS has both an active and a passive cooling mode, capable of rejecting decay heat. A guard vessel surrounds the primary vessel. The fuel is a metallic alloy of zirconium, uranium and plutonium. Each fuel pin is sodium bonded with a large plenum above the fuel to accommodate the release of gases from the fuel during normal operation.

At the top of the pool is a cover gas (argon) region that is separated from the containment floor by a reactor vessel top enclosure that forms the top head of the reactor vessel, supports the various equipment that penetrates the deck, and provides some shielding to protect personnel within the containment. Two rotatable plugs penetrate the top enclosure to enable the refueling machine to access locations within the core. Elastomeric seals provide a leak-tight barrier. The containment is a conventional design with a steel-lined, reinforced concrete structure. It is approximately 50 m in height and has a free volume of 2.86×10^4 m^3. The design pressure is approximately one atmosphere. The design leak rate at one atmosphere is one volume percent per day. Although this is somewhat larger than the design leak rates for Generation II commercial reactor designs, the actual leakage expected in an accident tends to be small because of the small driving force for release in characteristic SFR scenarios.

3. Key events affecting source term

Although severe accident behavior in LWR accidents can differ substantially among scenarios, many scenarios follow a standard pattern similar to the behavior of the TMI-2 accident in which water inventory decreases within the core region, fuel fails, a pool of molten fuel develops, the molten pool falls into the lower head, the head fails, and the molten core falls into the reactor cavity and attacks concrete. The release of radionuclides from fuel is typically divided into phases involving a gap release, which is typically small, an in-vessel release that is dominant, release during core concrete attack, and a long term release from residual fuel in the vessel and the re-evolution of radionuclides deposited within the vessel.

SFR scenarios differ substantially. During the period of reactor operation extensive relocation of noble gases and volatile radionuclides occurs from the fuel. These radionuclides are either captured in the sodium bond, airborne in the pin gas plenum or deposited on a surface internal to the pin. When a pin fails, the noble gases and airborne radionuclides are swept from the pin and form a gas bubble in the flow channels within an assembly. Sodium bond and molten fuel are also swept from the pin. Whereas, in the typical LWR scenario the time frame of radionuclide release from over-heated fuel ranges from minutes to hours, in an SFR accident fuel is typically only molten within the fuel pin for seconds and after release from the pin is rapidly quenched. The temperature

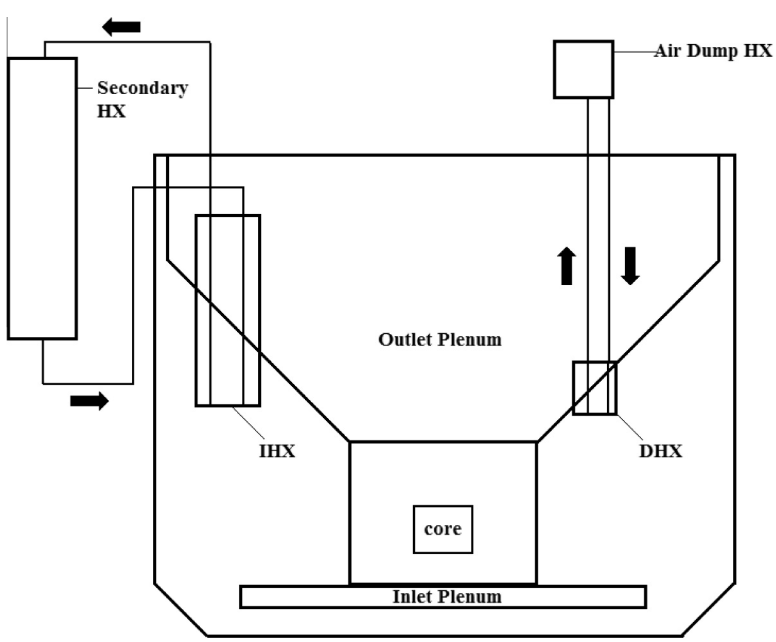

Fig. 1. Reference design concept.