Development of Safety, Irradiation, and Reliability Databases based on Past U.S. SFR Testing and Operational Experiences

T. Sofu

Argonne National Laboratory (ANL), Argonne, Illinois, U.S.A.

E-mail contact of main author: tsofu@anl.gov

Abstract. Inherent and passive safety are key aspects of Generation-IV SFRs to achieve licensing assurance and reduce plant costs, but they require demonstration and validation of key features. To address this need, U.S. DOE-NE's Advanced Reactor Technologies (ART) program has been supporting development of EBR-II, FFTF, and TREAT safety testing databases, metal fuel irradiation database, SFR component reliability database, EBR-II physics analysis database and IFR materials information system. These activities complement broader U.S. knowledge preservation and management efforts to facilitate the science-based R&D goals by providing data needed for validation of the state-of-the-art codes and advanced methods [such as those pursued under ART fast reactor methods and the Nuclear Energy Advanced Modeling & Simulation (NEAMS) programs] for design and analysis of advanced fast reactors. This paper summarizes the progress made to date in achieving these goals.

Key Words: SFR, safety testing, database development.

1. Introduction

In 2012, a series of "SFR Safety and Licensing Gap Analyses" were conducted in the U.S. through participation of 42 experts from the national laboratories, academia, industry, and international R&D organizations.[1] The effort aimed at understanding current DOE capabilities in five topical areas to support deployment of SFR concepts in the U.S. and identifying gaps that could hinder future licensing efforts:

- Accident Initiators and Sequences,
- Sodium Technology Phenomena,
- Fuels and Materials,
- Source Term Characterization,
- Computer Codes and Models.

After completion of individual reports for these topical areas, a smaller group of experts from SNL, ANL, INL, ORNL and BNL developed a comprehensive SFR safety and licensing research plan to prioritize gap closures and provide insight into potential future research programs with two major recommendations [2]:

- Coordinated knowledge management and preservation effort,
- Improvements to U.S. fast reactor design and safety analysis codes.

During the last four years, the focus of the DOE-NE's Advanced Reactor Technologies (ART) Fast Reactor Methods and Safety R&D program shifted toward implementation of these recommendations. Consistently, one of the objectives and scope of multi-laboratory efforts under the current ART Fast Reactor Methods and Safety R&D Program is preserving and managing data, knowledge, and experience related to past U.S. fast reactor operations and safety tests.

Achieved through collaborations mainly among of ANL, PNNL, and SNL, this effort focuses on archiving available safety testing data for validation of design and safety analysis codes and new models/methods for normal operations (including anticipated operational occurrences) and postulated accidents. It aims at providing focus, direction and validation basis for modelling and simulation programs pursued under DOE-NE's ART Fast Reactor Methods and Nuclear Energy Advanced Modelling and Simulation (NEAMS) programs.[3] It also leverages the data from this effort to support international collaborations conducted under various bilateral and international collaboration frameworks such as the Generation IV International Forum and IAEA coordinated research projects.[4]

2. Overview of U.S. Fast Reactor Database Development

To address the need for past U.S. fast reactor knowledge preservation, highlighted as a key priority in previous gap analyses[1,2], the ART Fast Reactor Methods and Safety R&D program supports development of EBR-II, FFTF, and TREAT test databases, metal fuel irradiation testing database, EBR-II physics analysis database, IFR material information system, and SFR component reliability database.

Most databases are established on a common software platform based on relational database management system (usually the open source MySQL RDMS or Microsoft SQL Server) with custom-designed (using Perl) web interface for controlled access to the Apache HTTP server on linux machines.

2.1. EBR-II Test Database

Experimental Breeder Reactor-II (EBR-II) is a 62.5 MW-thermal metallic fueled SFR with a "loop-in-a-pool" type design. The original mission of EBR-II was to demonstrate a closed fuel cycle with on-site reprocessing and testing fuels and materials for future, larger SFRs. It achieved first criticality in 1965 and operated as the Integral Fast Reactor prototype for 30 years. Its mission in later years shifted toward safety testing to demonstrate its inherent and passive safety characteristics. This experimental program also included the landmark inherent safety demonstration test when the reactor's supply of electricity was intentionally turned off, causing the coolant pumps to stop while the EBR-II reactor running at full power and the emergency shutdown systems disabled.[4]

EBR-II Test Database has been developed as an online searchable archive of measured data collected between 1984 and 1987 during this comprehensive test program.[5] It encompasses archival of data from 100+ tests and calibration runs that have considerable influence on inherent and passive safety of today's advanced fast reactor concepts. The spectrum of tests includes loss-of-flow with scram to natural circulation, scram with delayed loss-of-flow to natural circulation, reactivity feedback characterization, loss-of flow without scram, loss-of-heat-sink without scram, dynamic frequency response, and steam-drum pressure reduction assessments.

The tests are organized based on the five testing windows during which they were performed, each with a unique core composition and operating conditions:

- June 1984: These early experiments were all designed to test the passive shutdown heat removal capability of the reactor. The window includes a total of 26 such tests.

- May 1985: The experiments conducted during this window included 11 shutdown heat removal tests (some of which were the repeat of the tests conducted in June 1984 testing window), 11 balance of plant tests, and three rod drop tests.
- February 1986: The experiments conducted during this window included 17 shutdown heat removal tests (some of which were the repeat of the tests conducted in earlier and this testing windows).
- March/April 1986: The experiments conducted during this window included two shutdown heat removal tests including the IFR passive safety demonstrations, three balance of plants tests (one of which is repeated), and two rod drop tests.
- November 1987: A final sequence of plant inherent control tests was performed to demonstrate “load-following” features of EBR-II.

Depending on their nature, the tests were also categorized into several groups. Each test file includes typically 300-400 channels of measured data for the tests conducted during the first four test windows, and almost 900 channels of measured data for the tests conducted during the last test window. The instruments recorded by the data acquisition system are arranged in 60 relevant categories.

2.2. FFTF Passive Safety Testing Database

A similar effort has been recently concluded at PNNL to archive the data from several tests conducted at the Fast Flux Test Facility (FFTF) aimed to improve the understanding of the passive safety characteristics of the reactor.[6]

FFTF was a mixed-oxide-fueled SFR operating at 400 MW-thermal for the U.S. Department of Energy by the Westinghouse Hanford Company. The construction of the FFTF was completed in 1978, and the first reaction took place in 1980. From April 1982 to April 1992 it operated as a national research facility to test various aspects of commercial fast reactor design and operation as well as testing advanced nuclear fuels, materials, components. During its ten year of lifetime, its mission also eventually shifted toward safety testing to demonstrate its passive safety characteristics.

In late 1980's, a series of passive safety tests were conducted at FFTF to demonstrate the safety margins and obtain data to validate improved computational capabilities. Since FFTF design includes unique passive reactivity reduction devices called Gas-Expansion Modules (GEM) for loss-of-flow conditions, and a novel core restraint system to control its radial expansion during transient at elevated temperatures, these safety testing are uniquely important.

Of particular interest was a series of loss-of-flow without scram tests from power levels up to 50% to demonstrate the effectiveness of a reactor self-shutdown device called the Gas Expansion Module (GEM). The most recent PNNL team effort also involved expansion of the database to include FFTF start-up tests to measure various reactivity feedback coefficients and thermal-hydraulic response of the reactor to anticipated operational occurrences.

2.3. TREAT Test Database

TREAT Test Database was created as part of a knowledge management and preservation effort to advance the DOE’s R&D mission for design and analysis of fast reactors and integrated safety systems. It is created to store archived information from numerous reactor transient tests conducted from 1959 to 1994 at the Transient Reactor Test (TREAT) Facility at Argonne-West (now Idaho National Laboratory's Materials and Fuels Complex).[7]

The development of the database began with the assumption that roughly 150 tests had been performed in TREAT. However, in the years since its inception, exhaustive searching through document collections at ANL, the DOE Office of Scientific and Technical Information (OSTI), in open literature, and in reference lists of experiment documents has revealed that ~884 experiments were performed in TREAT. At this time, it is believed that the list of experiments performed at TREAT is probably complete and that the vast majority of reports and open literature publications essential for describing the experiments performed in TREAT have been included or identified for inclusion in the database.

The TREAT experiments were performed to support the development of fuel for numerous types of nuclear reactors and to investigate the response of various fuels forms to off-normal transients. The response of test fuels to the transients varied broadly: some fuels exhibited only minor damage, and others, complete meltdown. The TREAT investigations included a range of fuels, cooling environments, accident sequences, and diagnostic methods. Their results supported reactor fuel development and qualification efforts as well as computational model development. The transient tests also demonstrated various material interaction phenomena, their range of relevance, and the fuel limitations attributable to them.

The scope of TREAT experiments evolved in parallel with the developing needs of the U.S. reactor development program through investigations of key phenomena related to severe-accident energetics, demonstration of integral interactions among multiple phenomena related to accident progression, transient-performance limitations of fuels, etc. Early experiments addressed fuel-coolant interactions for many different fuel types for light-water-cooled reactors (LWRs) and sodium-cooled fast reactors (SFRs). Later, the experiment programs shifted to predominantly address oxide fuels for the liquid-metal-cooled fast breeder reactor (LMFBR) development, both for severe accident evaluations and for fuel qualification for use in demonstration reactors. Toward the end of the TREAT experiment program, the tests focused on both LWR radiological source term investigations and SFR metallic fuel over-power transient behavior.

2.4. Fast Reactor Metallic Fuels Irradiation Database

This database aims at preserving the information from metal-alloy fuels irradiation tests conducted at EBR-II reactor during the U.S. Advanced Liquid Metal Reactor (ALMR) and Integral Fast Reactor (IFR) programs to support fuels qualification as part of a future fast reactor license application as well as development and validation of fuel performance codes.[8] The selected experiments cover a wide range of fuel performance information, including prototypic fuel behavior and failure mode during operational modes, fabrication parameters, lead tests, high temperature swelling behavior, and fuel-cladding mechanical interaction. These steady-state irradiation experiments provide important fuel performance data in support of demonstrating the viability of metallic U-Zr alloy fuel and qualifying the fuel for use in fast reactors. The data is also used in calibration and validation of fuel performance codes, LIFE-METAL in particular.

The information maintained in the database include post irradiation examination information and documentations related to specific experiments with cross references to detailed pin-by-pin fabrication data and EBR-II operational history discussed in Section 2.5 below. The database main page identifies the list of irradiation tests currently available with each test identifying the cycles under which irradiation took place, core load configuration and position of the test assembly in each cycle, and the list of all the pins in that experiment. The data for individual pins include the location of the pin in the test subassembly (the initial position and the position for each pin reconstitution) with links to neutronics (such as the linear power, total, and fast neutron fluence as functions of pin length), thermal, and isotopic data (U-235 atom density as a function of pin length).

In addition to archival of data from the most essential fuels irradiation tests of interest, efforts also expand into the large set of documents related to the fuel post irradiation examinations program at ANL's Alpha-Gamma Hot Cell Facility (AGHCF). This effort involves digitizing and reconstructing the hard copy data such as cladding strain data (profilometry graphs) and isotopic gamma scans generated during the experiments using an automated pattern-recognition program.

The quality assurance plan following the ASME NQA-1 standards, and guidelines for evaluation of historical data from different sources of metallic fuel irradiation tests are also being developed as a parallel activity. In 2017, an extension of this activity will be initiated at PNNL to cover the metal fuel irradiation experiments conducted in FFTF with longer fuel pins.

2.5. EBR-II Physics Analysis Database and IFR Materials Information System

As a complementary activity to the Fast Reactor Metallic Fuels Irradiation Database described in Section 2.4, the EBR-II Physics Analysis Database (PADB) and IFR Materials Information System (IMIS) are two parallel efforts that focus on collection and organization of detailed pin by pin data fabrication data, core loading configurations and operational history, core physics and temperature calculations, and other miscellaneous information regarding the EBR-II operations. EBR-II operated from 1964 through 1994. There were 170 distinct runs, plus sub-run segments within an overall run, in EBR-II's irradiation history. Around EBR-II Run 130 (in Fall of 1984), zirconium-alloyed Integral Fast Reactor (IFR) fuel was introduced into the reactor.

The PADB covers all available information required to model the EBR-II irradiation histories, such as fabrication data for all of the Zr-alloy core assemblies, core maps starting from Run 130A (when the U-10Zr metal alloy fuel was first introduced), blanket maps starting from Run 1, operational history since Run 1, and any remaining data such as hardware and blanket assembly descriptions. Included in the PADB are element and assembly (collections of elements) ISOZ files. These ISOZ files are generated using the ORIGEN isotopic irradiation and depletion code, plus element fluxes and cooling times between runs, to generate end-of-life structural, actinide and fission product isotopics and burnups. The axially-dependent masses and burnups in the ISOZ files are given by quadratic fits; equations coefficients are given over specific axial ranges in the EBR-II models.

Developed during 1990's as part of DOE's Integral Fast Reactor (IFR) program, IMIS database has also been recovered and placed under a similar relational database management system configuration with an online interface for access. For all zirconium-fueled runs of the EBR-II reactor, PADB and IMIS contain some overlapping data, such as the fuel assembly and element information (name and jacket number), grid position, fuel and structural compositions, estimated atomic burnups and neutron fluences. An effort has recently been devoted to assessment and comparison of PADB and IMIS to address discrepancies in overlapping information among them and achieve a potential consolidation.

2.6. SFR Component Reliability Database

The purpose of this effort is to establish a database for SFR structures systems and components (SSCs) failure rates. During the EBR-II and FFTF era, the U.S. compiled an SFR-specific PRA information into the Centralized Reliability Database Organization (CREDO) database, maintained at ORNL.[8] Although this effort ended in 1994 following termination of the IFR program, information contained within CREDO was shared with Japan (PNC, now JAEA) in the late 1980's and early 1990's.

In 2013, an effort began at SNL and ANL to reclaim the reliability data using historical documents such as the original reliability data submittal forms used by CREDO developers. These forms specified the engineering, operations, and event data that would be submitted to CREDO by member facilities based on operating and event data from reactor run logs and event reports. Each event is associated with a specific reactor, a corrective action, and a specific sub-system of that reactor. Actions include repair, replacement, addition to, or removal from the system, and the operational data are separated into run-based and daily history entries.

Because reactors differed in the frequency of operational reporting, the run history table is flexible and allows a start and end date to be set for each entry. This permits data to be recorded whether it was provided for each week, quarter, or time-variable reactor run. A system hierarchy for each reactor has been assembled from documents and is used to sort the sub-systems affected by each event. In addition to EBR-II and FFTF operational data, the reliability information regarding the Sodium Reliability Experiment (SRE), FERMI-I, HNPF, and SEFOR reactors has also been discovered and undergoing a review in terms of its applicability to the SFR component reliability database.

In 2016, original CREDO data was received back from JAEA and is currently undergoing a review. It includes 1306 event records (i.e., what, when, and why did something happened and what was done about it), 408 facility operating records (i.e., describing a given facilities operating state as function of time), and 8102 engineering data records (i.e. component descriptions). Since a portion of the original CREDO database has been obtained, the current efforts are underway all data into a single database referred to as NaSCoRD, short for "Sodium System Component Reliability Database".[9] Currently, tools are being developed to resolve likely overlaps for the event and operational data from two sources. It includes development of a scheme to match events based on their timing and affected systems.

Acknowledgments

Argonne National Laboratory's work was supported by the U.S. Department of Energy, Office of Nuclear Energy, Office of Nuclear Reactor Technologies under contract DE-AC02-06CH11357.

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