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Eighteen Years of Operation with a Sodium-cooled Reactor: EBR-II

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The Experimental Breeder Reactor II (EBR-II) is a sodium-cooled fast reactor with a design power of 62.5 MWt and net output at full power of 20 MWe. The reactor is operated by Argonne National Laboratory for the United States Department of Energy and is located at the Idaho National Engineering Laboratory in southeast Idaho.

EBR-II (including the adjacent Fuel Cycle Facility) was completed in 1963, after five years of construction, at a cost of $32,500,000. EBR-II is now the longest operating breeder in the world and the only complete (with power plant) breeder operating in the USA.

During its operating history, EBR-II has demonstrated that it is a versatile facility in performing varying programs. The early operation of the reactor successfully demonstrated the feasibility of a sodium-cooled fast breeder reactor operating as an integrated reactor, power plant, and fuel processing facility. In 1967, the role of EBR-II was reoriented from a demonstration plant to an irradiation facility. Since that time, over 10,000 individual experiments have been or are in the process of being irradiated. The experiments consisted of various fuel types (oxides, carbides, nitrides, and metal), cladding and structural materials, and reactor control materials. Peak burnups of 19.0 and 18.5 at.%, respectively, for mixed-oxide and metal fuels have been reached. Beginning in 1977, the reactor was operated for extended periods with failed (breached) fuel elements as part of a program to determine the lifetime and survivability of failed fuel in fast reactors. Data gathered at EBR-II have contributed significantly to the maturing technology of sodium-cooled fast reactors and have been used in the designs of the Phenix and Clinch River reactors.

The reactor has recently completed qualification to perform a series of safety tests, starting in 1983 and extending at least through mid-1986. These tests relate to the off-normal conditions that may be anticipated to occur sometime during the operating life of a commercial reactor. Testing includes mild-overpower transients, duty-cycle transients, and loss-of-forced cooling (shutdown heat removal by natural convection of reactor coolant). It is expected that, as these safety tests proceed, additional programs could be integrated into the program to make use of the unique capabilities of EBR-II.

EBR-II has proven to be a very forgiving reactor and has achieved an excellent operating record. Its safe operation for 18 years can be attributed to the many safety features designed into the primary and secondary systems and certain favorable characteristics of sodium. These features have made EBR-II inherently protected from various component or operator failures that could occur. EBR-II uses the pool-type concept in which the reactor, major primary-system components and piping, and much of the fuel handling equipment are submerged in a large, double-walled tank containing over 341 m^3 of 473°C liquid sodium. This pool concept, with its large thermal capacity and capability for providing natural circulation through the core, ensures continuous cooling for the reactor during a loss of cooling flow incident. During this incident, the natural circulating primary coolant transfers the reactor heat to an intermediate nonradioactive secondary sodium system where it is then transferred to a power plant. In the worst-case situation of no pumping power available and no power plant available for dissipation of the decay heat, natural circulation through the core would still provide sufficient heat removal. This heat would be transferred from the primary coolant to two passive NaK shutdown coolers which exhaust the heat to the atmosphere. Also with the reactor and primary components submerged in liquid sodium, a loss of coolant as a result of a leak in the primary system piping is not a reactor safety problem as the reactor core will always remain submerged in coolant.

The main mechanical components in the primary sodium, secondary sodium, and fuel handling systems have provided excellent performance and demonstrated durability and reliability during 18 years of operation. Of particular interest is the duplex-tube evaporators and superheaters which have never experienced a sodium/water leak. The few problems associated with the reactor and sodium systems have not been of a serious nature and most occurred during the early years and were due to design deficiencies. Successful repairs and modifications were made on the failed components. No plant shutdown from equipment failure has exceeded four months.

During the operating lifetime of EBR-II, a substantial amount of experience and information has been gained in the areas of personnel exposure and radiation/contamination control associated with a sodium-cooled fast reactor.

The major radiation/contamination problems in removing components from the primary tank are contamination from fission products, radiation from ^{24}Na and ^{22}Na isotopes, and if the components are removed from the core region, they are highly radioactive due to neutron activation.

Despite the contamination from ^{24}Na, ^{22}Na, and various fission products (mainly ^{137}Cs), the annual exposure to maintenance personnel has decreased from slightly over 1 rem in 1975 to 145 mR in 1982 and the average exposure to Operations personnel in 1982 was 81 mR. This relatively low average exposure is achieved because: (1) all radiation is confined to the primary tank; (2) the rapid decay (15-h half-life) of ^{24}Na, which is the principal activation product in the primary sodium; and (3) the development and use of a cesium trap to remove the principal fission product ^{137}Cs. The cesium trap is part of the sodium purification system and can be used during reactor operation.

EBR-II is a "near zero" release facility in regard to radioactive gaseous effluents. The total gaseous effluent release in years 1980 through 1982 averaged 153 curies per year. This airborne effluent consists mainly of the noble fission gases, ^{133}Xe and ^{85}Kr. The major reason for the low discharge level even while running fuels beyond cladding breach has been the installation and successful operation of a cover-gas-cleanup system (CGCS) that uses a cryogenic distillation process to remove fission gases from the primary cover gas.

The average annual capacity factor since 1976 has been 72.1%. Although this is not as high as would be desired for commercial application, it is, nevertheless, an excellent record for a research reactor. It is also indicative of the reliability and maintainability that can be achieved in a pool-type sodium-cooled fast reactor.

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