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EVALUATION OF MATERIAL AGING EFFECTS ON EXTENDED LIFE OPERATION OF EBR-II*

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EVALUATION OF MATERIAL AGING EFFECTS ON

EXTENDED-LIFE OPERATION OF EBR-II*

ABSTRACT

Reactor material aging mechanisms in the EBR-II liquid metal reactor are identified and their effects on extended-life operation are evaluated.

1 INTRODUCTION

The Experimental Breeder Reactor No. II (EBR-II) is a sodium-cooled, metal-fueled, pool-type fast reactor that generates 62.5 MWt and 20 MWe power. It began power operation in 1964. Recently, EBR-II has undertaken the Integral Fast Reactor (IFR) prototype demonstration program, and other programs that require reliable operation to a lifetime of forty years. Since the originally planned life of EBR-II was only five to ten years, the current plant-life goal of forty years is significant. A careful evaluation of key material aging and degradation issues, as well as the features that enhance extended-life potential for the EBR-II design has been performed to establish the feasibility of extended-life operation of EBR-II.

Key aging issues that have been evaluated are material degradation of critical reactor components due to the effects of long-term exposure to the high-energy neutron environment of the fast reactor and the component effects of direct long-term exposure to high-temperature sodium. Of particular interest are neutron embrittlement and swelling of the reactor grid-plenum assembly and swelling of the graphite canisters that provide neutron shielding around the reactor vessel.

2 PLANT DESCRIPTION

For increased understanding of the reactor material aging effects discussed herein, a description of the EBR-II plant configuration and normal operational parameters is in order. EBR-II is a pool-type sodium-cooled reactor. The primary cooling system components and the reactor vessel are submerged in the primary tank containing 340 cubic meters (86,000 gallons) of sodium maintained at 371°C (700°F). The primary sodium never leaves the primary tank. Primary pumps, submerged in the primary sodium, pump the sodium into the reactor vessel, through the core, and then through the reactor outlet piping to the Intermediate Heat Exchanger (IHX), all of which are located within the primary tank (Fig. 1). The secondary cooling system transports the heat from the IHX to the steam generators, where superheated steam is generated and used to drive a turbine generator to provide electrical power to the site and the local power grid. The secondary coolant is also sodium, but it does not come into contact with the primary sodium and therefore remains uncontaminated.

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3 REACTOR MATERIAL AGING MECHANISMS

Although there are many material aging mechanisms at work throughout any type of nuclear power plant, the aging mechanisms affecting critical nonreplaceable reactor and primary system components are of most concern when considering extended-life operation of the facility. For EBR-II, the nonreplaceable components, of most interest from a material aging perspective, are the grid-plenum assembly, the reactor vessel neutron shield, and the primary tank.

The aging mechanisms of concern for the grid-plenum assembly (Type 304SS) are embrittlement and void swelling due to fast neutron irradiation. For the neutron shield (graphite blocks sealed in Type 304SS canisters) the concern is swelling of the graphite caused by irradiation or interaction with sodium; and for the primary tank, thermal embrittlement of the tank material (Type 304SS) and the Type 308SS filler material used to weld the 304SS plate sections together to form the tank. Another potential aging mechanism that affects all these components is long-term exposure to high temperature sodium. These aging mechanisms have been studied and evaluated since the early days of operation. In the 1960s, several archive samples of various reactor materials were assembled into surveillance subassemblies (called SURVs) and inserted into the reactor and the in-tank subassembly storage basket. These samples, having undergone long-term irradiation and exposure to sodium, have been removed and evaluated and provide most of the data used to evaluate the aging effects for the critical reactor components.

3.1 Irradiation embrittlement

Irradiation-induced embrittlement is a common concern for all types of nuclear reactors. The concern for EBR-II is not the reactor vessel, as in most other reactors (the EBR-II reactor vessel is not the primary coolant boundary), but the grid-plenum assembly which supports the reactor core and provides the inlet plena and flow distribution for the primary sodium coolant entering the reactor core. The grid-plenum assembly is subjected to relatively high loads and high neutron flux. Embrittlement concepts to be evaluated are the effects of neutron flux and spectrum and the possible saturation of embrittlement at 371°C sodium inlet temperature.

Some of the SURV samples are grid-plenum assembly archive material and have been irradiated to neutron fluences well beyond what the grid plenum assembly will accumulate by the end of a forty-year life. The SURV-8 samples which were tensile-tested after irradiation exhibited a minimum residual ductility of 17% (elongation) (Longua and Laug 1983) with an accumulated fluence equivalent to that at the grid-plenum assembly after more than 45 years. A reasonable minimum allowable ductility is 5-10% based on Fast Flux Test Facility and Clinch River Breeder Reactor guidelines.

Irradiation embrittlement of the reactor vessel and the primary tank is of little concern because these components are well shielded from the reactor core and have fast neutron exposures that are orders of magnitude less than the grid-plenum assembly.

3.2 Void swelling

The other aging mechanism of concern for the grid plenum assembly is void swelling caused by neutron irradiation. Void swelling is a concern for the grid-plenum assembly because of the two parallel horizontal grid plates that

are part of the structure. These plates have in-line holes through them for subassembly lower pole piece positioning. The upper plate is expected to swell more than the lower plate because of the higher neutron flux at the upper plate which shields the lower plate. The differential swelling between the upper and lower plates could eventually cause misalignment of the in-line holes resulting in binding of the subassembly lower pole piece during removal or insertion into the grid (Hofman 1975). In anticipation of this, subassembly lower pole piece diameters have been reduced to provide additional clearance through the grid plates. Additionally, periodic grid hole gauging is performed to check for decreased clearance caused by misalignment (King 1982); no evidence of grid-plate misalignment has been detected to date. If further reduction in pole piece diameter is required, grid-gauging information should provide adequate lead time in order to make the necessary modifications. Monitoring of fuel handling operations to detect an increase in required subassembly insertion/removal forces is also an important part of this surveillance.

3.3 Graphite swelling

Irradiation-induced graphite swelling has been investigated to determine possible adverse effects in the neutron shield surrounding the reactor vessel and in the reactor vessel cover. The graphite swelling concern is that in certain areas there is a possibility of the swelling causing distortion of the reactor vessel, which may result in fuel handling difficulties in the outer rows of the reactor. There is also potential for distortion of the reactor vessel cover if significant graphite swelling occurred. This may cause difficulties in proper seating of the cover on the vessel or cause binding of a control rod drive shaft.

Samples from the SURV subassemblies have indicated that the graphite was in a densification phase at the fluence level at the time of the evaluation. Further evaluation of the effects of neutron irradiation on the graphite have indicated that at the fluence level of the graphite at forty years, the graphite in the canisters will have started swelling again, but the volume change will be small. The canisters themselves also are sized to permit some swelling of the graphite inside. It appears from this study that irradiation-induced swelling of the graphite should not be a problem for a forty-year life.

However, two graphite sample canisters in the SURVs have ruptured and significant graphite swelling (~50%) due to interaction with the sodium was observed when these were removed from the reactor (Longua, et al., 1984). In-depth examination and study have shown that the graphite swelling occurred as a result of direct contact with the primary sodium. In other words, the seal-welded canister developed a leak, which permitted sodium to enter the canister and cause the reaction and swelling which in turn split open the canister. Further evaluation of the cause of the initial leak indicates that the canister welds had failed; possibly due to internal pressure buildup inside the canister due to helium generation in the borated graphite. Although the design of the sample canisters is not as substantial as the in-service canisters, a graphite-can rupture in the radial shield or in the reactor vessel cover is a possibility and the consequences identified above have to be considered. Even with the possibility of graphite-canister rupture, it is very unlikely that reactor operation would be affected because of the canister-to-canister stacking clearances built into both the radial shield and

the axial shield in the cover, and the fact that few of the canisters are positioned such that rupture would cause operational difficulties. Ways to detect graphite-canister rupture and resulting effects continue to be investigated.

3.4 Thermal embrittlement

Another embrittling phenomenon that may become important to a forty-year lifetime is thermal embrittlement of Type 304 stainless steel, and particularly of welds mating Type 304SS to Type 304SS using Type 308SS as a filler material. The primary tank is constructed of Type 304SS plate sections welded together using Type 308SS welding electrodes.

The embrittlement phenomenon is characterized by degradation of the free ferrite in cast 304SS or in the 304/308SS welds after long periods of time due to phase transformations and formation of precipitates. EBR-II did not use a pure cast form of 304SS (CF-8 or CF-8A) in the construction of the primary tank, so this should not be a problem. In addition, the potential embrittlement problem was anticipated by the designers of the reactor and weld criteria were specified to limit the ferrite content in both weld rods used and in the welds themselves.

Weld samples have had long-term exposure to high temperature sodium in the SURV subassemblies and initial evaluation of the samples indicates that the weld area is not significantly embrittled. Further evaluation and exposure is planned.

4 OTHER AGING MECHANISMS

Other aging mechanisms in EBR-II have also been studied in-depth. Of interest here is fatigue due to thermal cycling of major plant components and the potential for corrosion or erosion of steam generator components. These are discussed briefly in this section.

4.1 Thermal fatigue

Evaluation of thermal stresses and the allowable number and severity of thermal cycles for the major plant components has shown that the most severe operational transient is a reactor scram. The component that is most affected by the induced thermal stress in the case of a scram is the IHX. Also, as part of an evaluation of the plant to determine the impact of a series of operational transient tests, a wide range of more severe test transients were evaluated, as well as their effects on the major components (Chang and Lee 1982). Overall, the most severe of the test transients was a ramp-up in reactor power level to 123% of normal full power followed by a reactor scram. Again, the IHX was the most severely affected.

As a conservative evaluation of the residual fatigue life of the IHX at forty years, an assumption was made that <u>ALL</u> at-power scrams over the life of the plant had and will have equivalent fatigue damage as the transient test. With this assumption, the number of cycles at forty years will be 60% or less of the allowable lifetime cycles for the IHX based on ASME code criteria.

4.2 Erosion and corrosion

The effects of erosion and corrosion in sodium systems have been evaluated since the early days of operation. The most significant study was based on the removal and destructive examination of a superheater after 17 years of service. The unit was found to be in excellent condition. The sodium side exhibited no observable effects of corrosion or erosion, and the steam side showed the normal expected effects of exposure to saturated and superheated steam (Penney et al., 1982; Buschman et al., 1981). Evidence of heat transfer degradation which prompted the removal of the superheater was determined to be caused by relaxation of the prestress in the mechanically-bonded duplex tubes. The second superheater has metallurgically bonded tubes, as does the unit installed to replace the original superheater. No degradation has been observed with these units.

Corrosion and erosion due to sodium has been investigated in other systems and components with similar results. As long as the sodium can be maintained at very low oxygen concentrations, interaction of the sodium with reactor material is negligible.

5 CONCLUSION

Although several potential aging mechanisms are at work in the EBR-II facility, it appears that none of these should prevent EBR-II from achieving a forty-year operational lifetime. Considering that there were no specific design-life criteria required at the time of planning, design, and construction of EBR-II and that it was intended to serve a mission requiring only a five to ten year lifetime, it appears that a very long operational life could be achieved by this type of reactor if long life is specified as a design goal at the beginning of design. The operational performance of EBR-II to date and the evaluation of key aging mechanisms appear to support this conclusion.

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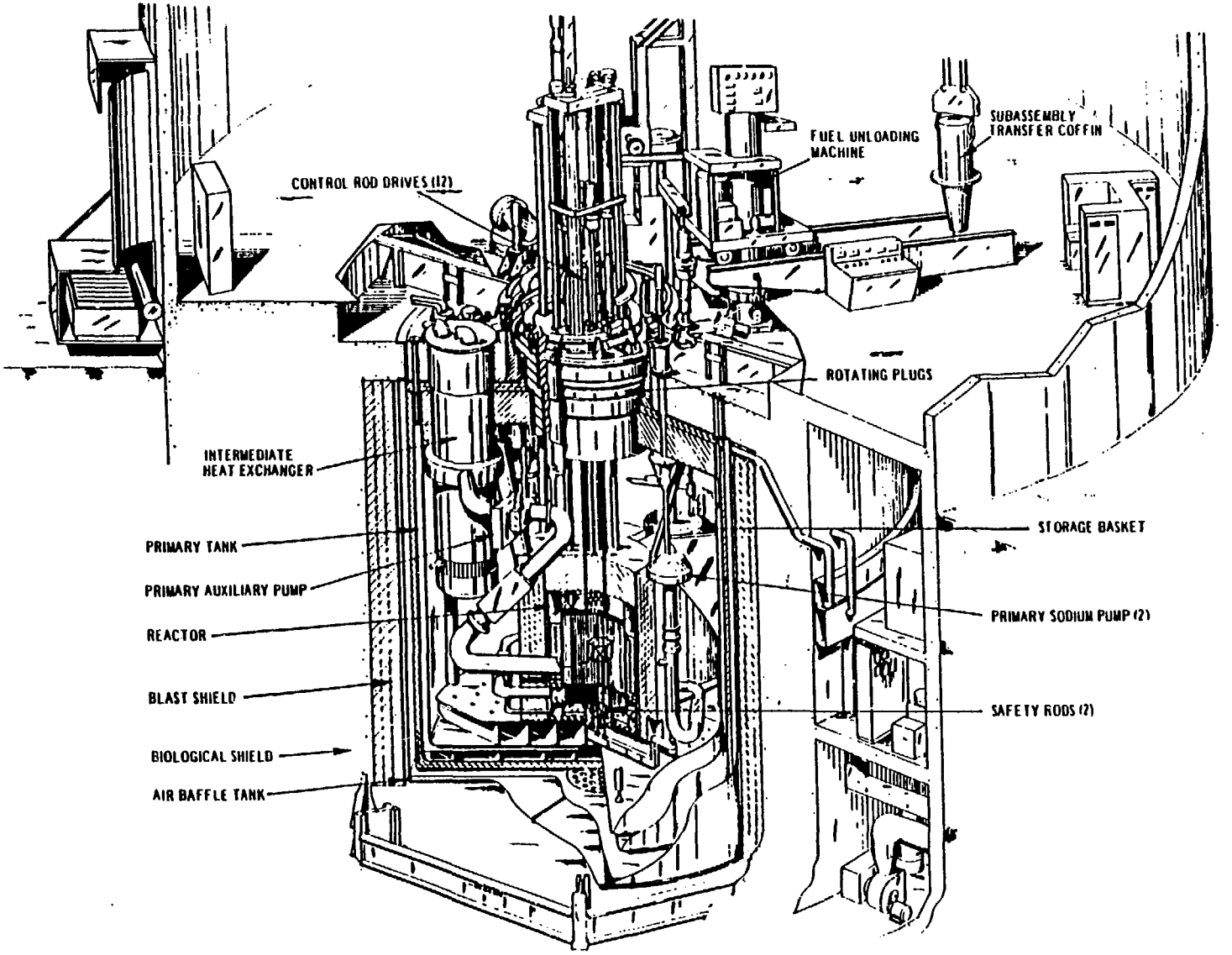
CONTROL ROD DRIVES (12)
FUEL UNLOADING MACHINE
SUBASSEMBLY TRANSFER COFFIN
ROTATING PLUGS
INTERMEDIATE HEAT EXCHANGER
PRIMARY TANK
PRIMARY AUXILIARY PUMP
REACTOR
BLAST SHIELD
BIOLOGICAL SHIELD
AIR BAFFLE TANK
STORAGE BASKET
PRIMARY SODIUM PUMP (2)
SAFETY RODS (2)

Figure 1