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System Variables Design of Safety Analysis for Fast Reactors

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Abstract

This research aims to examine the risk in the technology design of fast breeder reactors while the development depends on safety considerations. The project explored the variables, which could affect positively the expected average fuel burn-up, breeding ratio, and decay heat removal. That is accomplished using features such as guard vessels and elevated pipe routing to prevent the cracked state of both core components and fuel cladding interface conditions. So, the cracked region of fuel was detected by thermal-hydraulic analysis. We used ZrFeCr alloys to estimating of the rise in fuel cladding and coolant that can be incorporated in the design ZrFeCr alloys to uniform corrosion in temperature and 10.3 Mpa pressure. Fast creep of the reactor vessel during the coolant heat-up transient is another issue to be considered corrosion resistance of structural material can be achieved by controlling oxygen content in steel alloy. In this trend, S4337 S5140 steels are wide and can be used in future fossil power plants because of their excellent high-temperature strength.

Keywords

Reactor, Safety, Crack, Temperature, Safety Management System

1. Introduction

Nuclear power plants must always ensure the highest level of safety that can reasonably be achieved to protect workers at these plants, the public, and the environment from any harmful effects of ionizing radiation or other hazards present at the installation. This statement is valid for all existing nuclear power plants and serves as a guide for developing Generation IV nuclear reactors [1] [2].

There is a large gap between the high-level safety fundamentals and the detailed codes and standards, which can be summarized in Figure 1.

Because risk cannot be eliminated, it must ideally be optimized. This means a cost/benefit analysis, although such analysis often looks only at the risk of technology, without factoring in benefits. At a risk optimum, the additional resources used to provide additional risk reduction would come at a disproportionate cost, and any resources removed from risk reduction would cause a disproportionate increase in risk. This situation is shown in Figure 2. This optimum is never achieved in practice. This is partly because the risk from nuclear power is perceived to be greater than that of other technologies, even if the numerical risk is the same, due to social factors such as unfamiliarity with the technology and its association with atomic [3] [4].

One of the most important safety considerations is relative to the motion of the reactor core and control rods within the containment affected by the site-specific ground motion to the seismic response. This approach may present the opportunity for refinement of the reactor design by revealing components that are unnecessary or possibly overdesigned components imposed according to current deterministic licensing requirements.

The objective is not only to design a sodium-cooled fast reactor (Figure 3), but also to develop a systematic process to be used in the design, the challenge of this project is obtaining reliable cost estimates in sufficient detail to test to methodology. To avoid absolutes, this project will focus on comparative assessments of design and cost estimates completed in the past. The advanced metal reactor is well-documented since it has detailed costs at the component level and a probabilistic risk assessment [5].

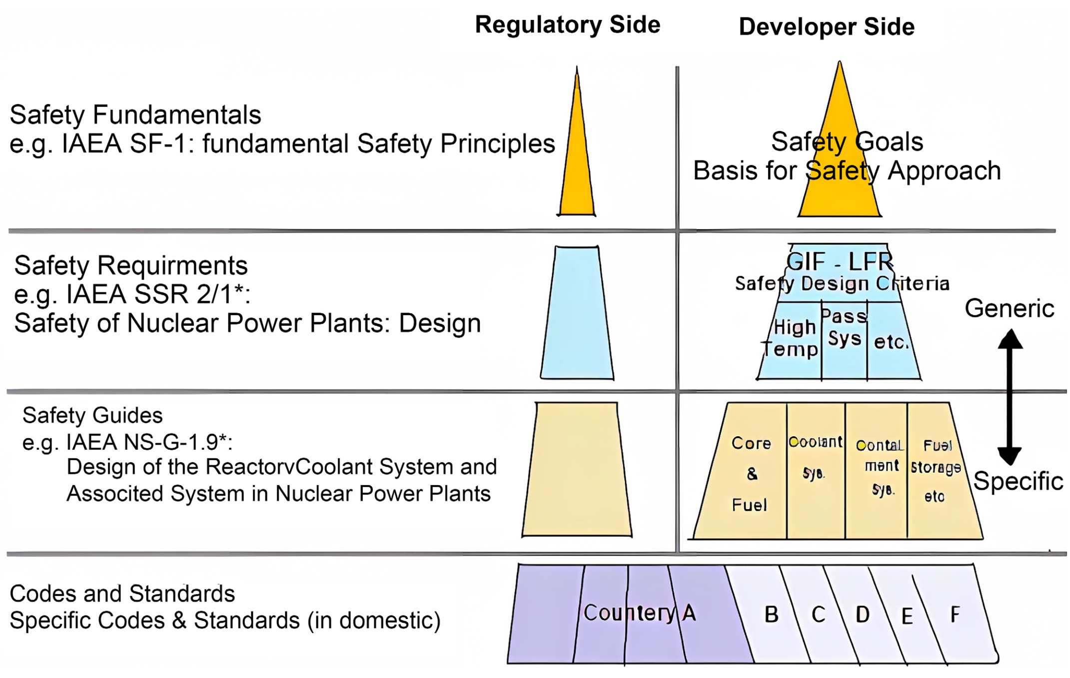

Figure 1. Hierarchy of safety standards.

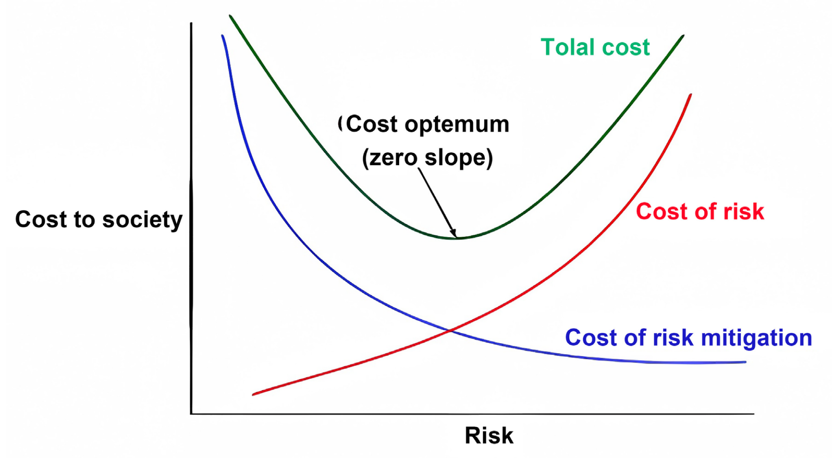

Figure 2. Risk optimization.

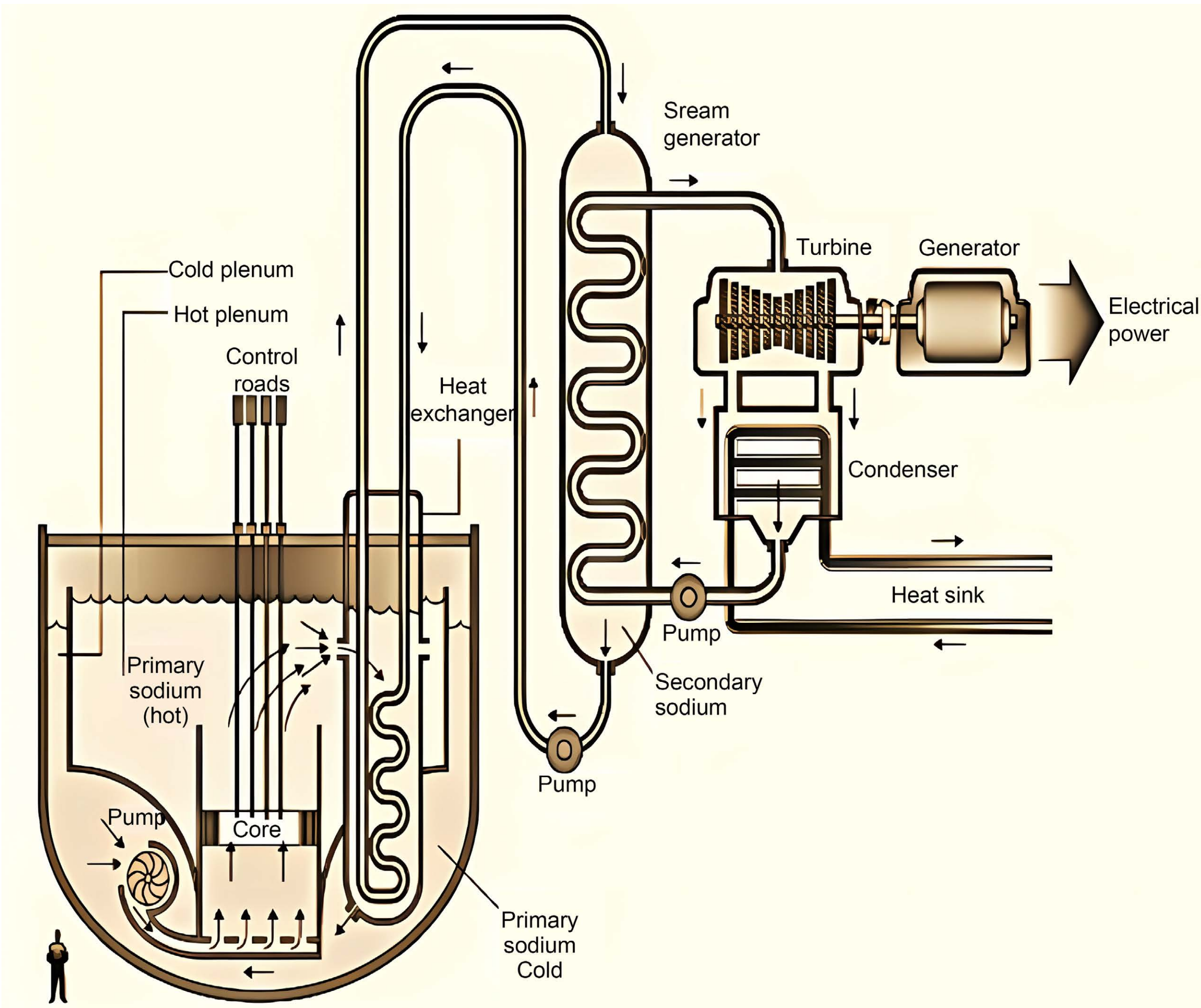

Figure 3. Pool-type Sodium-cooled Fast Reactor (SFR).

The pressure control system refers to that of a BWR the same as in the two-pass cooling SCW fast reactor as shown in Equations (1)-(2) where $V(t)$ is the opening of the turbine control valve with the control's parameter of K [6]:

$$V_r(t) = 1 - \frac{P_{set} - P(t)}{K} \tag{1}$$

$$V(t) = V_r(t) + 2\frac{\mathrm{d}V_r}{\mathrm{d}t} - 5\frac{\mathrm{d}V}{\mathrm{d}t} \tag{2}$$

The early outlet temperature control system was designed by considering only one signal of steam temperature deviation, as the feedback is shown in Equation (3) which u is the control signal, and KP is the control parameter [7].

$$\frac{\mathrm{d}u}{\mathrm{d}t} = K_P \frac{T - T_{set}}{T_{set}} \tag{3}$$

The goal is to show meaningful fuel reductions in cost and safety which will be compared to a thermal reactor once a reasonable model has been developed areas for reduction in cost with safety aspects of the design will be determined and specific items can be addressed expected that the methodology could be used in the design of any future reactor since the methodology will be generic. Sustainability: Concerning sustainability, the main concern was the management of the environment through clean air restrictions, waste management restrictions, and conservation of resources.

Fast reactors use Lead-Bismuth Eutectic (LBE) and lead as coolants and possess a very high level of inherent self-protection and passive safety against severe accidents. That type of reactor can be simultaneously more safely and more cheaply. As all other coolants, LBE and Lead Coolant (LC) possess particular virtues and shortcomings. The conclusion is made about the promising usage of FRs with these coolants in future NP after the experience in operating the prototypes of such reactors has been obtained [8].

There a study has shown a relationship between safety management system implementation and safety performance, where the analyses indicated a significant difference between compliance with the safety management system and safety performance. Therefore, it is concluded that a safety management system is critical to improving safety performance, and the safety management system is. There is a need for ongoing commitment and determination from all parties concerned for the improved safety performance to be sustained [9].

1.1. System Thermal-Fluids Modeling

SAM is being developed as a system-level modeling and simulation tool with higher fidelity yet computationally efficient. As a new code development, the initial effort has been focused on the modeling and simulation capabilities of the heat transfer and single-phase fluid dynamics responses in reactor systems. The transient simulation capabilities of typical reactor accidents have been demonstrated in the transient simulations of various advanced reactor types and validated against the EBR-II, FFTF, and MSRE (**Figure 4**), and many integral effects test results. The key features include [10] [11] [12]:

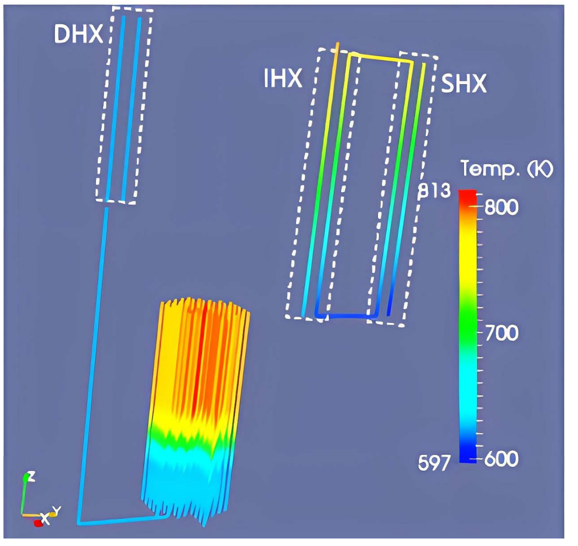

Figure 4. Temperature distributions in the simulation of Advanced Burner Test Reactor (ABTR).

- Robust and high-order FEM model of single-phase fluid flow and heat transfer;
- Component-based system modeling;
- Flexible coupling between fluid and solid components enables a wide range of engineering applications;
- Enhanced built-in closure models and flexible modeling of fluid properties, friction, and convective heat transfer.

1.2. Reduced-Order Multi-Dimensional Flow Model

A computationally efficient multi-dimensional flow model is under development for thermal mixing and stratification phenomena in large enclosures for safety analysis. An advanced and efficient thermal mixing and stratification modeling capability embedded in a system analysis code is desirable to improve reactor safety analyses' accuracy and reduce modeling uncertainties [13] [14] [15] (**Figure 5**).

1.3. Flexible Core Modeling

A pseudo-3-D full-core conjugate heat transfer modeling capability has been developed in SAM for efficient and accurate temperature predictions of SFR structures. A multi-channel rod bundle model is developed to account for the temperature differences between the center region and the edge region of the coolant channel in a fuel assembly [16] [17] [18] [19]. The hexagon lattice core can be modeled with automatically-generated 1-D parallel channels representing the subassembly flow, and 2-D duct walls and inter-assembly gaps (**Figure 6**).

1.4. Multi-Physics Multi-Scale Integration

Flexible coupling interfaces have been developed to allow for convenient integration with other advanced or conventional simulation tools for multi-scale and

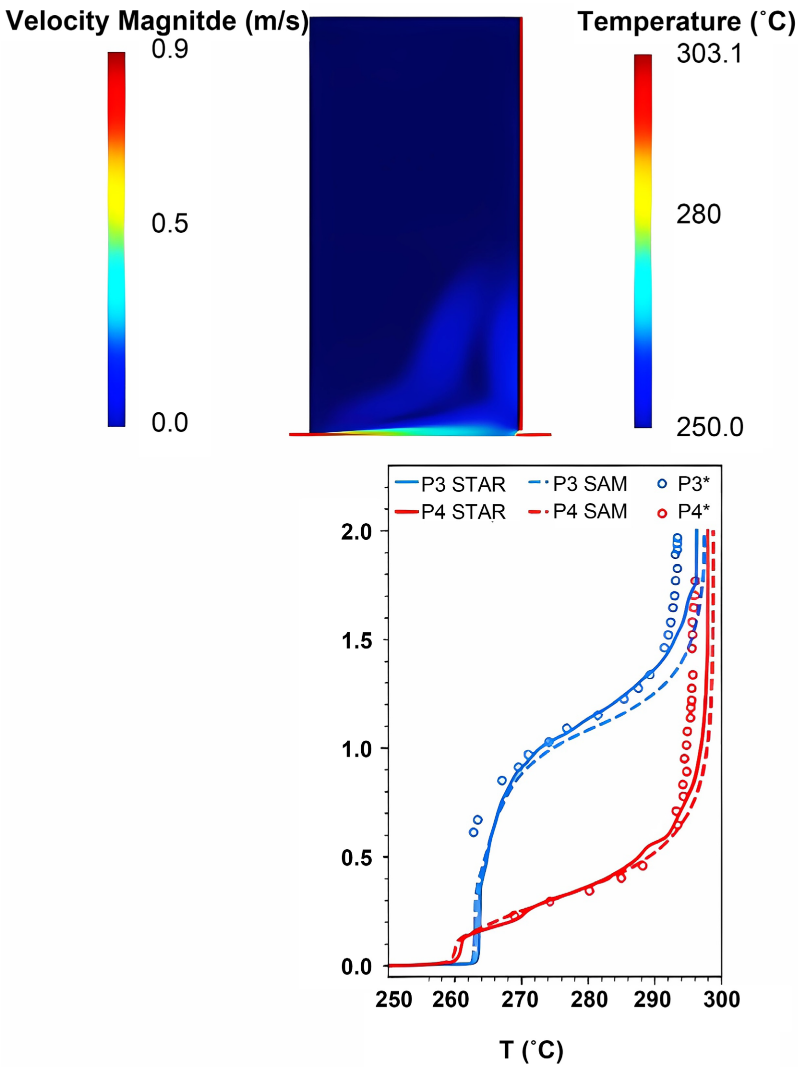

Figure 5. Simulation results of flow fields and comparisons with experiments for test cases in the SUPERCAVNA facility.

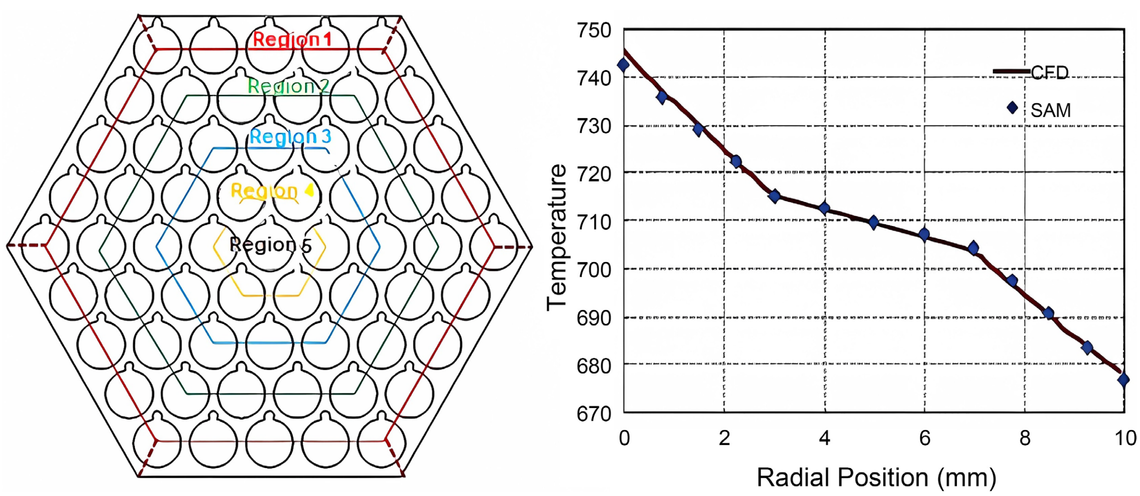

Figure 6. Comparison of average radial wall temperature distributions between SAM and CFD in a 7-assembly demonstration problem.

multi-physics modeling capabilities. SAM simulation results of a heat-pipe-cooled microreactor are shown in Figure 7. This effort utilized several MOOSE-based submodules under NRC's Comprehensive Reactor Analysis Bundle (CRAB), including SAM, MAMMOTH/Rattlesnake, and MOOSE's Tensor Mechanics module (Figure 7).

2. Method

S4337 S5140 steels are also regarded as promising structural materials for high-temperature nuclear power plants such as liquid metal-cooled fast and high-temperature gas-cooled reactors. Their crack growth property is often required to evaluate the structural integrity under the presence of detected or postulated flaws. In addition to fatigue crack growth property, crack growth property under creep

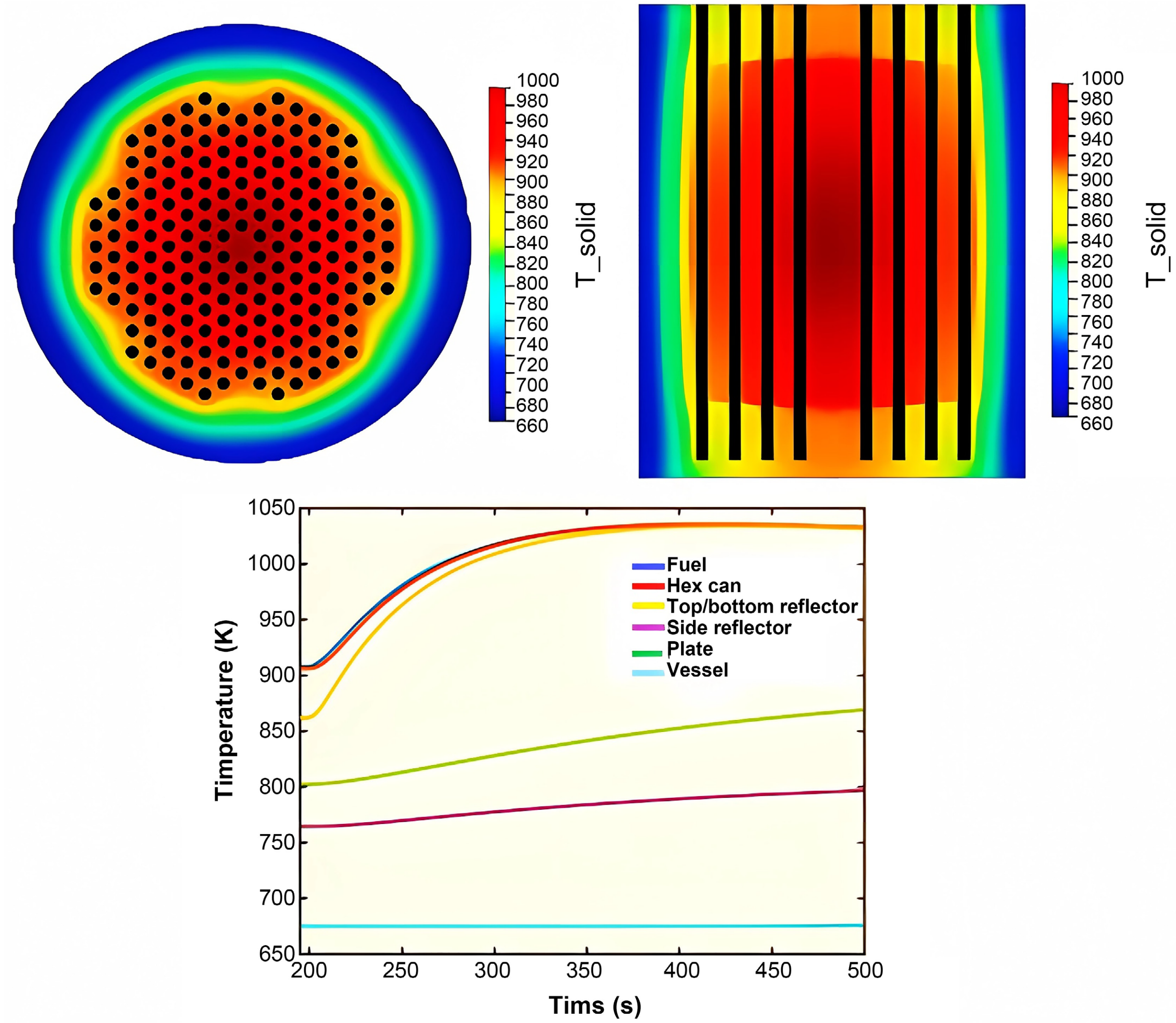

Figure 7. Multi-physics simulation of an unprotected loss of heat sink event of a heat-pipe-cooled mmicroreactor horizontal cut view of temperature profile (top left); vertical cut view (top right); the transient average solid temperature of different blocks (bottom).

and creep-fatigue loading is important because of the high operating temperatures of these plants. As a study in the working group organized in Yarmouk Industrial Complex, experimental and analytical studies have been conducted on the high-temperature crack growth property of representative high-quality steels. Temperature- and specimen-dependency of crack growth properties in terms of J-integral type fracture mechanics parameters were examined in detail. Standard equations were derived to estimate the crack growth rate as a function of these parameters. The effect of constraint on creep crack growth behavior was found to be quite large in designing the primary system against seismic loading.

3. Experimental Work

The experimental measurements were carried out in the Yarmouk industrial complex. An analytic study was carried out to determine the applicability of thermal stress fragmentation of the UO_2-Na fuel-coolant interaction. The primary emphasis was put on the fracture mechanics approach to assess whether or not the solidifying UO_2 would fracture under thermally-induced stresses. It was found that the stress levels were sufficient to generate KI values (inhibitor constant) substantially over the UO_2 fracture toughness. Thus, instantaneous propagation of inherent flaws anticipated for tensile tests, fatigue tests, and creep tests was conducted for smoothed specimens to grasp deformation behavior under various conditions. The higher tensile lists have been used to improve fabrication processes, control bearing quality, and determine steel mechanical properties. One of these tests is to defining of aging effects which leads with result that relaxation of pressure equal to 50% of the initial values at the luminal operational temperatures. Modification has been made on a system test to allow the tension to affect the bearing, and that was accomplished when the loads were in vertical mode. The result of the tests on the modified bearings showed a reduction of vertical effects of shape factor, cross-sectional area, and total thickness when they are at minimum values. The results have confirmed the validity of the simplicity of the formulas' stiffness, and it proved that the adequacy of the single-dowel at attachment system minimized the deformations of the bearing end plates.

Two types of ferritic steel were developed: high-strength ferritic/martensitic steel, considered a good material suitable for tube wrapping, and strengthened oxide ferritic steel tubing technology for such cladding has progressed by the working process. Structural tests are to improve strength the adequacy at elevated temperatures the design rules and verify advanced nonlinear structural analysis methods. The discontinuity structural model to investigate the crack initiation and its behavior prediction was made, and their evaluation and buckling lest for cylindrical shells subjected to shear loads are being conducted. Deterministic and probabilistic fracture mechanics approaches are developed for the integrity assessment of the flawed or cracked structure.

Crack propagation tests of flawed structural elements like pipe, plates, and elbows subjected to mechanical loadings have been performed to validate the system's applicability on creep and non-creep regions. The crack propagation tests of a cylinder with circum referential and surface flaws were at the air cooling thermal transient. Another test for fast reactor equivalent alloys has been conducted to clear the relationship between creep rupture strength and metallurgical variables such as chemical composition grain size. Long-term tests were being run concerning creep fatigue and the effect of thermal aging on the embrittlement of base material and welds. Long-term tests were carried out to investigate the influence of cast-to-cast variations on the creep rupture response of type 316 LN to its equivalents S4337S5140.

The data extrapolation for end of the life condition indicates a minor effect on tensile properties, but a decrease in the toughness concerning irradiation remains greater at irradiation doses up to 5 dpa. High-temperature irradiation is an interim reduction factor for ccreep fatigueat around 550 c, but this value still should be measured. So, the objective is to evaluate the effects of irradiation and composition at high-temperature, and cast-to-cast variation on long-term properties of types S4337 and S5140 steel and weld metals. Concerning toughness, a comparison of different results showed that the fracture toughness properties of S4337 and S5140 alloys are superior to those of 316 L(N) steel, the weld metals results indicate that the toughness properties of the above alloys welds lay close to the lower bands of materials test are also being in progress to investigate the fatigue and creep-fatigue response of type S4337 welded with. For filler metal, furthermore, creep-fatigue tests including a hold period at the maximum strain of the cycle have recently commenced fatigue test has been completed at 650 c and further tests are underway between 650 and 750 c. The preliminary comparison of creep-fatigue data at 650 c indicated the need to make tests using long hold times from 1000 h to 1500 h at low strain ranges to produce creep damage levels more relative to the steam generator operating condition creep-crack growth.

4. Results and Discussions

Chemical composition for the alloy S4337 can be used in future Fast Breeder Rector (FBR) component DIN OXN3FA average of measurements/concentrations: C, Si, Mn, Cr, Mo, Ni, Al, Co, Cu, Nb, Ti, V, W, Pb, and Fe is 0.34, 0.36, 0.34, 1.40, 0.42, 3.18, 0.033, 0.044, 0.25, 0.0004, 0.002, 0.11, 0.007, 0.002, and 93.5, respectively.

Chemical composition for the alloy S5140 also can be used in future Fast Breeder Rector (FBR) components: 0.014, 0.009, 0.002, 0.025, 0.0009, 0.010, 0.003, 0.009, 0.002, and 97.7 is: C, Si, Mn, Cr, Mo, Ni, Al, Co, Cu, Nb, Ti, V, W, Pb, and Fe, respectively.

5. Conclusions

The result of the tests on the modified bearings showed a reduction of vertical effects of shape factor, cross-sectional area, and total thickness when they are at minimum values. The results have confirmed the validity of the simplicity of the formulas' stiffness, and it proved that the adequacy of the single-dowel at the attachment system minimizes the deformations of the bearing end plates.

ZrFeCr alloy is a promising component for use in high-temperature nuclear energy structural materials such as liquid metal-cooled fast reactors and gas-cooled high-temperature reactors.

In future studies, the chemical composition of alloy 40X can be used.

Conflicts of Interest

The authors declare no conflicts of interest regarding the publication of this paper.

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