Research and Development on Simulator of Fast Reactor in China

Prof. ZHANG ZHIJIAN

Presented by Prof. ZHANG ZHIGANG

College of Nuclear Science and Technology
Harbin Engineering University (HEU) of China
Contents

1. Background for CEFR Simulation
2. Technical Solutions for CEFR Simulator
3. Modeling and Simulation
4. Comparison and Effect
5. Proposal
1. Background

China began to develop the sodium-cooled fast reactor since March, 1992, through international cooperation, China Experimental Fast Reactor (CEFR) is constructed since May, 2000 and reaches the first critical condition in July, 2010.
1. Background

CAEA invited HEU to develop the full scope real-time simulator of CEFR in collaboration with China Institute of Atomic Energy (CIAE) in 2009. The aim is:

- to research operation characteristics
- to validate system design
- to research emergency operating procedure
- to train operator
1. Background

Related research introduction

<table>
<thead>
<tr>
<th>Designer</th>
<th>Code</th>
<th>Simulation Technology and Range</th>
</tr>
</thead>
<tbody>
<tr>
<td>Russia</td>
<td>GRIF</td>
<td>A large-scale SFR transient thermodynamic code; Based on BN-350, BN-600; To calculate three-dimensional single-phase flow in sodium pool; Be able to represent complex geometry and structure; Covering neutron kinetics, reactivity feedback, PDHR.</td>
</tr>
<tr>
<td>Russia</td>
<td>GRIF-SM</td>
<td>Transient analysis of SFR under severe accident condition; Considering sodium boiling, fuel melting, reactor dynamics, reactivity feedback. Temperature distribution of coolant, fuel rod, core, and primary loop is available. Be tested by ULOF and LOFA of BN-800.</td>
</tr>
<tr>
<td>France</td>
<td>CATHARE</td>
<td>Be applied to SFR with implementation of sodium physical properties, adapted pressure drop and heat transfer correlations, fast reactor point kinetic model for neutronics; Be tested by modelling of the Phenix primary circuit.</td>
</tr>
</tbody>
</table>
1. Background

<table>
<thead>
<tr>
<th>Designer</th>
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<th>Simulation Technology and Range</th>
</tr>
</thead>
<tbody>
<tr>
<td>America (WH)</td>
<td>IANUS</td>
<td>Analysis on dynamic response of loop type LMFBR under accident conditions, such as pipe rupture, pump failure.</td>
</tr>
<tr>
<td>America (WH)</td>
<td>DEMO</td>
<td>Based on IANUS, the natural circuit calculation is added.</td>
</tr>
<tr>
<td>America (ANL)</td>
<td>NATDEMO</td>
<td>Base on system equilibrium model in DEMO-IV; Suitable for both loop and pool type fast reactor. Tested by EBR-II</td>
</tr>
<tr>
<td>America (ANL)</td>
<td>DSNP-ND</td>
<td>Based on the models of the primary system components of NATDEMO; Simulating EBR-II and compared with NATDEMO.</td>
</tr>
<tr>
<td>America (ANL)</td>
<td>SASSYS-1</td>
<td>A reactor core thermal hydraulic analysis code; To calculate decay heat removal character, transient and accident conditions.</td>
</tr>
<tr>
<td>America (BNL)</td>
<td>SSC-L</td>
<td>Thermo hydraulic simulation code for Transients in LMFBRs; Dynamic analysis on heat transport system of loop-type.</td>
</tr>
<tr>
<td>Korea</td>
<td>SSC-K</td>
<td>Based on SSC-L; Analysis of the KALMR; Representing core with multiple coolant channels; A point kinetics model for reactivity feedback; Considering sodium boiling in core and PDHR system.</td>
</tr>
</tbody>
</table>
1. Background

CEFR adopted a pool-type FR technology with three-loops, which has 216 subsystems.
1. Background

The full scope real-time simulator was finished in 2013, which has been validated by CEFR.
2. Technical Solutions

According to the principle, characteristics of system, structure and operation of CEFR, the following research has been accomplished:

I. To determine the scope and degree of simulation

II. To establish the simulation models with a modular method

III. To design CEFR simulation system consisting of 13 systems

IV. 71 subsystems were simulated which are closely related with operating

V. To setup 86 initiating events, including leaking of pipes and equipment, loss of heat sink, etc., and 1,043 common failure points in 11 categories
2. Technical Solutions

CEFR Simulation System

Training Room
- Large Scale Teaching Screen
- Teaching Terminal

Main Control Room
- Integrated Display Console of Main Control Room
- I/O System
- Control Workstation

Instructor Room
- Printer
- Instructor Station

Reactor Physics Simulation Code
- Primary Coolant System Simulation Code
- Secondary Coolant System Simulation Code
- Auxiliary System Simulation Code
- Steam-Power Conversion System Simulation Code
- Control and Logic System Simulation Code
- Safety and Protection System Simulation Code
- Radiation Monitoring System Simulation Code

Shared Memory
- Network Hub System

Database
- Monitor System Simulation Computer
- Local Operating Workstation
- System Control Workstation
- Operation Analysis Workstation

Main Computer
- Developing Engineer Computer
- Developing Engineer Computer
- Developing Engineer Computer

Developing and Maintenance Station
rsity
3. Modeling and Simulation

The model and software have been developed for 71 CEFR subsystems. The main systems are listed as below:

- Reactor Physics
- Primary Coolant System
- Intermediate Loop System
- Third Coolant System
- Auxiliary System
- Passive Decay Heat Removal System
- Reactor Control and Protect System
- Electrical System
- Reactor Radiation Monitoring System
3. Modeling and Simulation

(1) Thermal-hydraulic Model of Primary Coolant System (PCS) - Structure

- Sodium pump (2)
- Shield layer
- Core structure
- Core
- Rod driven system (8)
- Individual heat exchanger (2)
- Intermediate heat exchanger (4)
- Core vessel

Inner structure and composing for PCS of CEFR
Thermal-hydraulic processes for sodium pool

Complex inner structure, include conduction, convection, natural and forced circulation, three-dimensional, dynamic and physical process.
3. Modeling and Simulation

High precision thermal-hydraulic modeling and real-time simulation code of sodium pool

Solutions

- Use fine control volume
- Divide the three-dimensional sodium pool to multiple one-dimensional flow in different directions
- Establish liquid sodium flow heat transfer equation and neutron physical coupled model
- Convergence and calculation stability are realized
The flow distribution of core are decided by the pressure of some key locations

\[
\frac{L_1}{A_1} \frac{dW_1}{dL} = P_{pool} - P_{11} - \frac{fW_1^2}{l_1 D_e \rho A_1^2} dz - \frac{\xi W_1^2}{2 \rho A_1^2}
\]

\[
\frac{L_2}{A_2} \frac{dW_2}{dL} = P_{pool} - P_{12} - \frac{fW_2^2}{l_2 D_e \rho A_2^2} dz - \frac{\xi W_2^2}{2 \rho A_2^2}
\]

\[
\frac{L_3}{A_3} \frac{dW_3}{dL} = P_{0} - P_{11} - \frac{fW_3^2}{l_3 D_e \rho A_3^2} dz - \frac{\xi W_3^2}{2 \rho A_3^2}
\]

\[
\frac{L_4}{A_4} \frac{dW_4}{dL} = P_{0} - P_{12} - \frac{fW_4^2}{l_4 D_e \rho A_4^2} dz - \frac{\xi W_4^2}{2 \rho A_4^2}
\]

\[
\frac{L_5}{A_5} \frac{dW_5}{dL} = P_{21} - P_{11} - \frac{fW_5^2}{l_5 D_e \rho A_5^2} dz - \frac{\xi W_5^2}{2 \rho A_5^2}
\]

\[
\vdots
\]

\[
\frac{L_{11}}{A_{11}} \frac{dW_{11}}{dL} = P_{0} - P_{pool} - \frac{fW_{11}^2}{l_{11} D_e \rho A_{11}^2} dz - \frac{\xi W_{11}^2}{2 \rho A_{11}^2}
\]

\[
V_{P11\_in} = V_{P11\_out}
\]

\[
V_{P21\_in} = V_{P21\_out}
\]

\[
V_{P0\_in} = V_{P0\_out}
\]

\[
V_{P12\_in} = V_{P12\_out}
\]

\[
V_{P22\_in} = V_{P22\_out}
\]

\[
V_{Pc1\_in} = V_{Pc1\_out}
\]

\[
V_{Pc1\_in} = V_{Pc1\_out}
\]
Setup of control volume for flow and heat transfer of sodium pool

- **Wall cooling for hot pool**
- **Wall cooling for cold pool**
- **Natural circulation**
- **Forced circulation**
- **Thermal process of argon**
- **Heat convection**
- **Wall conduction**

Develop the high-fidelity thermal-hydraulic analysis code including 513 control volumes for primary coolant system of pool type.
Setup of control volume for flow and heat transfer of sodium pool

<table>
<thead>
<tr>
<th>ID</th>
<th>Control volumes</th>
<th>ID</th>
<th>Control volumes</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>Argon region: 1 volume (001)</td>
<td>8</td>
<td>2# Primary pump: 27 volumes (075<del>077, 408</del>431)</td>
</tr>
<tr>
<td>2</td>
<td>Hot sodium pool: 8 volumes (002<del>005, 099</del>102)</td>
<td>9</td>
<td>Pressure chamber: 3 volumes (078~080)</td>
</tr>
<tr>
<td>3</td>
<td>1# IHX: 120 volumes (006<del>035, 144</del>233)</td>
<td>10</td>
<td>Reactor core: 108 volumes (081<del>098, 104</del>143, 451~500)</td>
</tr>
<tr>
<td>4</td>
<td>1# Cold sodium pool: 3 volumes (036~038)</td>
<td>11</td>
<td>Public region of reactor vessel cooling system: 1 volume (103)</td>
</tr>
<tr>
<td>5</td>
<td>1# Primary pump: 27 volumes (039<del>041, 384</del>407)</td>
<td>12</td>
<td>1# primary vessel cooling regions: 36 volumes (324<del>353, 432</del>436, 443)</td>
</tr>
<tr>
<td>6</td>
<td>2# IHX: 120 volumes (042<del>071, 234</del>323)</td>
<td>13</td>
<td>2# primary vessel cooling regions: 36 volumes (354<del>383, 437</del>442)</td>
</tr>
<tr>
<td>7</td>
<td>2# Cold sodium pool: 3 volumes (072~074)</td>
<td>14</td>
<td>Independent heat transfer: 20 volumes (444<del>450, 501</del>513)</td>
</tr>
</tbody>
</table>

**TABLE 3  DETAILS OF THE CONTROL VOLUMES**
3. Modeling and Simulation

(2) Neutron Physics Model

Technical challenges

- Breeding zone and fuel zone are consisted by multiple fuel assemblies.
- Fast neutrons have a wide spectrum, some are used for sustaining fission, some are used for breeding.
- The neutron dynamic process is quite complex.
3. Modeling and Simulation

(2) Neutron Physics Model

Solutions

- 3D, four groups neutron diffusion equations are to describe the neutron behavior.
- The calculation zone is divided into 732 nodes, including breeding zone and fuel zone.
- Reactivity coupling equations have been modeled.
- Hexagon fuel assemblies neutron program has been developed.
3. Modeling and Simulation

(2) Neutron Physics Model

\[
\frac{1}{\nu_g} \frac{\partial \phi_g(r,t)}{\partial t} = \nabla D_g(r,t) \nabla \phi_g(r,t) - \Sigma_{ag}(r,t)\phi_g(r,t) - \Sigma_{sg}(r,t)\phi_g(r,t)
\]

\[
+ \sum_{g'=1}^{G} \Sigma_{g'g}(r,t)\phi_{g'}(r,t) + (1 - \beta)\chi_{pg} \sum_{g'=1}^{G} \nu_{g'} \Sigma_{fg'}(r,t)\phi_{g'}(r,t)
\]

\[
+ \chi_{dg} \sum_{d=1}^{D} \lambda_d C_d(r,t) + S_g(r,t)
\]

\[
\frac{\partial C_d(r,t)}{\partial t} = -\lambda_d C_d(r,t) + \beta_d \sum_{g=1}^{G} \nu_g \Sigma_{fg}(r,t)\phi_g(r,t)
\]
3. Modeling and Simulation

(3) Precise Modelling and Real-time Simulation of OTSG and Other Devices

Technical challenges

- Complex phase change in OTSG water side.
- Rapid volume change caused by water-steam transition.
- Dynamic change of steam-water interface with load.
- Sodium-water reaction accident from sodium leakage.
3. Modeling and Simulation

(3) Precise Modelling and Real-time Simulation of OTSG and Other Devices

Solutions

The two-phase drift flux model is used to simulate phase change.

A non-equilibrium heat transfer model including single-phase water, steam and two-phase boiling is established.

By adjusting control volumes to solve convergence and stability.
4. Comparison and Effect

- Simulator has been applied to instruct the debugging and experimental operation of CEFR and improve control methods.
4. Comparison and Effect

### TABLE 4

<table>
<thead>
<tr>
<th>Simulation accuracy</th>
<th>CEFR simulation system</th>
<th>A-PWR simulation system</th>
</tr>
</thead>
<tbody>
<tr>
<td>Key parameter error</td>
<td>Key parameter error less than 1%</td>
<td></td>
</tr>
<tr>
<td>Reactor power</td>
<td>Reactor power</td>
<td></td>
</tr>
<tr>
<td>Coolant temp. at the outlet of core</td>
<td>Coolant temp. at the outlet of core</td>
<td></td>
</tr>
<tr>
<td>Coolant temp. at the inlet of core</td>
<td>Coolant temp. at the inlet of core</td>
<td></td>
</tr>
<tr>
<td>Generator power</td>
<td>Generator power</td>
<td></td>
</tr>
<tr>
<td>Primary system pressure</td>
<td>Primary system pressure</td>
<td></td>
</tr>
<tr>
<td>SG pressure</td>
<td>SG pressure</td>
<td></td>
</tr>
<tr>
<td>Other parameter error</td>
<td>Other parameter error less than 10%</td>
<td></td>
</tr>
<tr>
<td>Continue operation 60min, key</td>
<td>Continue operation 60min, key</td>
<td></td>
</tr>
<tr>
<td>parameter Max. change less</td>
<td>parameter change less than 2%</td>
<td></td>
</tr>
<tr>
<td>than 1.8%</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

**Reference standard:**
ANS/ANSI 3.5-2009 《Nuclear Power Plant Simulators for Use in Operator Training and Examination》
4. Comparison and Effect

Reactors scram analysis by simulator

- Control rod drop, reactor scram
- Secondary loop sodium pump speed down, flux down
- Sodium pump speed down, flow rate down
- Feed water pump speed keeps

Cold pool
Hot pool

- Feed water into the superheater
- Sodium-water reaction occur
- Large thermal stress caused on heat transfer pipe
- Pipes may crack, the sodium-water reaction may occur

Application example: Improve the control methods when reactor scram
4. Comparison and Effect

Reactor scram analysis by simulator

Two-phase state in Superheater outlet

Before improvement

The superheat degree of steam is zero

Saturated Temperature
$T_{out}$ of Evaporator
$T_{out}$ of Superheater
4. Comparison and Effect

Improvement to avoid sodium-water reaction

Flow rate of feed water should be regulated as the total heat transfer of OTSG.
4. Comparison and Effect

Efficiency Analysis

To test the real-time ability of this developed code, three simulations for accident conditions have been analyzed.

<table>
<thead>
<tr>
<th>Accident case</th>
<th>CPU time /s</th>
<th>Simulation time /s</th>
<th>Ratio (CPU time/simulation time)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Reactivity Insertion Accident</td>
<td>100</td>
<td>400</td>
<td>0.25</td>
</tr>
<tr>
<td>Loss of Coolant Accident</td>
<td>185</td>
<td>600</td>
<td>0.31</td>
</tr>
<tr>
<td>Loss of Heat Sink Accident</td>
<td>175</td>
<td>600</td>
<td>0.29</td>
</tr>
</tbody>
</table>
5. Proposal

China is going to develop a fast reactor power plant 600MWe, So we will go on undertaking the related simulation projects such as

• simulation validation platform for digital control system (DCS)
• full scope real-time simulator of 600MWe for training
• principle simulator of 600MWe for education
5. Proposal-principle simulator to IAEA

Main interface
Thank you for your Attention!
5. Comparison and Effect

<table>
<thead>
<tr>
<th>Variables</th>
<th>Relative Error based on the values of design conditions / %</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>100%</td>
</tr>
<tr>
<td>Core coolant flow rate (kg/s)</td>
<td>0</td>
</tr>
<tr>
<td>Coolant temp. at the inlet of core (°C)</td>
<td>1.269</td>
</tr>
<tr>
<td>Coolant temp. at the outlet of core (°C)</td>
<td>0.100</td>
</tr>
<tr>
<td>Flow rate at the primary side of IHX (kg/s)</td>
<td>0.006</td>
</tr>
<tr>
<td>Flow rate at the secondary side of IHX (kg/s)</td>
<td>0</td>
</tr>
<tr>
<td>Coolant temp. at the inlet of the primary side of IHX (°C)</td>
<td>0.120</td>
</tr>
<tr>
<td>Coolant temp. at the outlet of the primary side of IHX (°C)</td>
<td>0.603</td>
</tr>
<tr>
<td>Coolant temp. at the inlet of the secondary side of IHX (°C)</td>
<td>0</td>
</tr>
<tr>
<td>Coolant temp. at the outlet of the secondary side of IHX (°C)</td>
<td>0.325</td>
</tr>
</tbody>
</table>

**TABLE II: VALIDATION AND ANALYSIS FOR STEADY-STATE CONDITIONS**
4. Modeling and Simulation

(3) System Model for Secondary Loop Coolant System
4. Modeling and Simulation

(4) System Model for Main Steam System
4. Modeling and Simulation

(5) System Model for Deoxidizing/Purification System