12.4 Computer Codes Used in Safety Analysis

The following computer codes were used to analyze reactor facility responses during AOOs and accidents.

1. BLOOST-J2
2. THYDE-HTGR
3. TAC-NC
4. RATSAM6
5. COMPARE-MOD1
6. GRACE
7. OXIDE-3F
8. FLOWNET/TRUMP

Representative events of AOOs and accidents and corresponding computer codes used to analyze them are shown in Table 12.5.

<table>
<thead>
<tr>
<th>Representative events</th>
<th>CODE NAME</th>
</tr>
</thead>
<tbody>
<tr>
<td>Abnormal CR withdrawal during subcooled condition</td>
<td>O</td>
</tr>
<tr>
<td>Abnormal CR withdrawal during rated power operation</td>
<td>O</td>
</tr>
<tr>
<td>Stop of PGC for IHX</td>
<td>O</td>
</tr>
<tr>
<td>Opening of exhaust valve of primary helium storage and supply system</td>
<td>O</td>
</tr>
<tr>
<td>Increase in revolution of POC for IHX</td>
<td>O</td>
</tr>
<tr>
<td>Increase in revolution of PGC for PPWC</td>
<td>O</td>
</tr>
<tr>
<td>Opening of supply valve of primary helium storage and supply system</td>
<td>O</td>
</tr>
<tr>
<td>Opening of bypass flow control valve of air cooler</td>
<td>O</td>
</tr>
<tr>
<td>Opening of exhaust valve of secondary helium storage and supply system</td>
<td>O</td>
</tr>
<tr>
<td>Increase in heat removal by secondary cooling system</td>
<td>O</td>
</tr>
<tr>
<td>Loss of off-site electrical power</td>
<td>O</td>
</tr>
<tr>
<td>Abnormal reactivity insertion by movement of irradiation specimen</td>
<td>O</td>
</tr>
<tr>
<td>Deterioration of insulation material in irradiation capsule</td>
<td>O</td>
</tr>
<tr>
<td>Transient during safety demonstration test</td>
<td>O</td>
</tr>
<tr>
<td>Channel blockage of standard fuel element</td>
<td>O</td>
</tr>
<tr>
<td>Failure of inner pipe in primary concentric hot gas duct</td>
<td>O</td>
</tr>
<tr>
<td>Failure of inner pipe in secondary concentric hot gas duct</td>
<td>O</td>
</tr>
<tr>
<td>Rupture of secondary concentric hot gas duct</td>
<td>O</td>
</tr>
<tr>
<td>Rupture of pipe in PWCS</td>
<td>O</td>
</tr>
<tr>
<td>Depressurization accident(^1)</td>
<td>O</td>
</tr>
<tr>
<td>Failure of PPWC heat transfer tube</td>
<td>O</td>
</tr>
<tr>
<td>Failure of primary helium purification system(^1)</td>
<td>O</td>
</tr>
<tr>
<td>Failure of gaseous radwaste treatment system(^1)</td>
<td>O</td>
</tr>
<tr>
<td>Failure of iweep gas pipe in irradiation test equipment(^2)</td>
<td>O</td>
</tr>
<tr>
<td>Channel blockage of fuel failure test specimen</td>
<td>O</td>
</tr>
<tr>
<td>Failure of stand pipe(^1)</td>
<td>O</td>
</tr>
</tbody>
</table>

\(^1\): In addition, radiation exposure is estimated

\(^2\): Only radiation exposure is estimated
12.4.1 BLOOST-J2 Code

(1) Outline

The BLOOST-J2 code\textsuperscript{12-21} is used to analyze the effects of reactivity and flow rate change on the reactor power and temperatures of the core.

The BLOOST-J2 code is a modified version of BLOOST-5 code\textsuperscript{12-21} so as to adopt the configuration of the HTTR core.

In the BLOOST-J2 code, the core is simulated by two channels: an average channel and a hot channel. Temperature feedback reactivity is calculated in the former channel and maximum fuel temperature in the latter. The transients of the reactor power is calculated by the use of the point kinetic equation. Temperature transients of fuels, graphite sleeves, coolant and graphite blocks are obtained by solving the equations of the heat transfer and heat conduction by use of a simulated reactor core with the one channel cylindrical model.

An analytical model of the reactor core is shown in Fig.12.1. The reactor response is simulated using an equivalent cylinder representing the core.

![Cross-section of HTTR fuel assembly, Equivalent cylinder, Analytical model for code](image)

**Fig.12.1** Equivalent core channel model for BLOOST-J2 code

(2) Validation of the code

A validation of BLOOST-J2 code was performed by comparing the analytical results with the data of CR withdrawal/insertion experiment with Fort St. Vrain (FSV) at 50% of rated power\textsuperscript{12-21}. Comparisons were made for the following two cases:

1) CR withdrawal with $6.2 \times 10^4 \Delta k/k$ reactivity added ramp-wise for 6 seconds
2) CR insertion with $-4.1 \times 10^4 \Delta k/k$ reactivity added ramp-wise for 21 seconds

A transient of power for the CR withdrawal experiment was shown in Fig.12.2. Analytical results agreed well with the experimental results.
**12.4.2 THYDE-HTGR Code**

(1) Outline

The THYDE-HTGR code\textsuperscript{12-14} is used to analyze the transient thermal-hydraulic characteristics of the PCS, secondary helium cooling system (SHCS), PWCS and ACS during abnormal occurrences. The original code of THYDE-HTGR is the THYDE code\textsuperscript{12-13} which was developed to analyze the thermo-hydraulic transients of Light Water Reactors. The THYDE-HTGR code is a modified version of the THYDE code to treat thermal-hydraulics of the HTGRs. The THYDE code was validated by a comparison with various experiments such as the LOFT experiments\textsuperscript{10-12,13}. 
In the THYDE-HTGR code, the coolant flow paths, including the reactor core, are simulated by a flow network model with nodes of finite volume and junctions which are in contact with the nodes. Thermal-hydraulic transients are calculated by solving conservation equations of mass, momentum and energy in helium gas and water. Temperature distributions in the structure such as fuel rods, moderator graphite, heat transfer tubes in the IHX and pipes are obtained by solving the one-dimensional transient heat conduction equation. An analytical model of the HTTR system is shown in Fig.12.3.

(2) Validation of the code

Since THYDE-HTGR code is a modified version of the validated code of THYDE, some parts of THYDE-HTGR code have already been validated. Then, the validation of the dynamic characteristics in the core and thermal-hydraulics of the heat exchanger was performed.

The validation of core dynamic characteristics was confirmed by comparing the analytical results with the data of CR withdrawal/insertion experiment under the condition of 50% power of the FSV. The analytical results are shown in Fig.12.4. From the analytical results, the nuclear dynamics of THYDE-HTGR code was validated.

To validate thermal-hydraulic characteristics in the heat exchanger obtained by the code, the experimental data obtained in the High Temperature Helium Loop, which was operated by the Engineering Research Association of Nuclear Steelmaking (ERANS), with the helium-helium intermediate heat exchanger (IHX) were used. A comparison between the experiment and the analysis shows good agreement as shown in Fig.12.5.

Fig.12.3 Analytical model of THYDE-HTGR code
Fig. 12.4 Comparison between experimental and analytical results of control rod withdrawal/insertion test for THYDE-HTGR code
12.4.3 TAC-NC Code

(1) Outline

The TAC-NC code\cite{15-16} is used to calculate transient thermal-hydraulic characteristics such as natural circulation caused by the density difference among coolant channels during loss of forced cooling. The TAC-NC code is a modified version of TAC 2D code\cite{15-16} so as to be able to treat natural circulation.

In the TAC-NC code, a temperature change in the structural materials is calculated by solving the two-dimensional transient heat conduction equations. Natural circulation after the accident is calculated from density distribution and pressure drop in the reactor core.

The analytical model consists of a coolant passage model which calculates natural circulation and a heat transfer model which calculates heat conduction in the structural materials, heat transfer...
between the coolant to structural materials and radiation as shown in Fig.12.6.

Fig.12.6 Analytical model of TAC-NC code

(2) Validation of the code

A calculation scheme of heat conduction analysis in TAC-NC code is originated from TAC-2D code. Since TAC-2D code was used for various temperature calculations, heat conduction results obtained by the TAC NC code could have a sufficient reliability. Calculations of the natural circulation flow rate and temperature distributions were validated by a comparison with the air ingress experiment which simulated a rupture of the primary concentric hot gas duct in the HTTR.

A simulated core of experimental apparatus consists of electric pipe heaters placed in a vessel of 3.9m in height. The heater could be heated up to 400°C. Inner regions of the electric pipe heaters are coolant passages which simulate the coolant passages of the HTTR. The experiment was performed by opening a flange at the bottom of the vessel after the heater temperature reached
at 300°C.

A comparison of the results between the experiment and the analysis is shown in Fig.12.7. Calculated temperature agreed very well with the experiment. Physical properties of the structural materials are selected so as to obtain conservative results with respect to the temperature. As shown in the results, the flow rate of natural circulation is larger in the analysis than that in the experiment. It is because the pressure drop in the simulated reactor core was larger than that of the analytical model. In the safety analysis of the HTTR, the pressure drop at the coolant passages was underestimated so as to obtain the large (conservative) flow rate of natural circulation. The effect of the natural circulation on the fuel temperature estimation could be neglected because the flow rate of natural circulation was very small compared with that of forced circulation. On the other hand, a large flow rate was conservative from the viewpoint of oxidation of the graphite structure. Therefore, the results obtained by TAC-NC code could be estimated as being sufficiently conservative.

Fig.12.7 Comparison of analytical with experimental results for TAC-NC code
12.4.4 RATSAM6 Code

(1) Outline

The RATSAM6 code\(^{12-13}\) is used to calculate the amount of mass and energy released from the reactor into the CV with consideration given to heat transfer during the rupture of the primary concentric hot gas duct. This code is used to analyze the release characteristics of the coolant during a depressurization accident of the FSV.

In RATSAM6, the depressurization characteristics of the PCS could be obtained by solving the one-dimensional unsteady mass, momentum and energy equations. An analytical model of the PCS is shown in Fig.12.8.

![Analytical model for RATSAM6 code](image)

Fig.12.8 Analytical model for RATSAM6 code

(2) Validation of the code

Validation of the code was performed by comparing the analytical results with experimental results\(^{12-20}\) which were obtained by using a 1/18-scaled apparatus simulating the PCS of the Colder Hall-type Reactor. Cooling systems of the apparatus consist of a heat exchanger, pipes and orifices which simulate the pressure drops induced by a circulator. Comparisons between the experiment and analysis were made in the case of a pipe rupture. The rupture was simulated as a leak path in the model. Flow resistance of pipes and reactor core was determined\(^{12-20}\) by considering the configuration and pressure drop due to a hydraulic friction loss in the core. A typical time dependent behavior of pressure is shown in Fig.12.9. The analytical results and experimental dates are in good agreement.

![Comparison between experimental and analytical results of differential pressure transient for RATSAM6 code](image)

Fig.12.9 Comparison between experimental and analytical results of differential pressure transient for RATSAM6 code
12.4.5 COMPARE-MOD1 Code

The COMPARE-MOD1 code\textsuperscript{12-20} is used to calculate pressure and temperature behavior in each compartment of the containment vessel (CV) during the depressurization accident. This code was certified by US Nuclear Regulatory Committee as a code for safety analysis to calculate pressure and temperature behaviors in the CV.

In COMPARE-MOD1 code, space of the CV was divided into nodes, then pressure and temperature behaviors in the CV are obtained by solving mass, momentum and energy equations on each node.

An analytical model of the CV is shown in Fig.12.10. All of the compartments in the CV are treated in the model.

![Analytical model for COMPARE-MOD1 code](image)

Fig.12.10 Analytical model for COMPARE-MOD1 code

12.4.6 GRACE Code

(1) Outline

The GRACE code\textsuperscript{12-20} is used to calculate axial and radial distributions of oxidation amount of graphite materials and concentration distribution of oxygen in a mixed gas of air and helium gas by analyzing the oxidation reaction between ingressed air and the graphite structures. Three processes are considered in the analysis, i.e., mass transfer process from ingressed air to surface boundary
layer of graphite, diffusion process in pores from the surface boundary layer of graphite to graphite block and reaction process between oxygen and graphite.

**Figure 12.11** shows a schematic diagram of the ingress path of mixed gas containing air. The mixed gas ingressed from the CV into the reactor pressure vessel (RPV) flows through the reactor core and flows out again to the CV.

![Diagram of reactor core and flow paths](image)

**Fig.12.11** Flow of mixed gas in case of depressurization accident

(2) Validation of the code

In the GRACE code, the mass transfer coefficient was obtained from the heat transfer correlations obtained in the experiment. Correlations of the heat transfer and mass transfer coefficients were compared using graphite oxidation experiment. The mass transfer coefficient obtained from the experiment agreed well with the value derived from the heat transfer correlations\[^{12-21}\]. A graphite oxidation experiment\[^{12-21}\] was also carried out for validation of the GRACE code using IG-110 graphite. In an air-flowing channel with 88mm in inner diameter,
heated specimen of IG-110 graphite with 50mm in outer diameter was installed and the oxidation amount was measured. Experimental conditions were as follows:

- Temperature: 700°C ~ 1000°C
- Air flow rate: 2 ~ 7 l/min
- Concentration of air: 20%

Results are shown in Fig.12.12. An experimental temperature range of 700°C ~ 1000°C was determined by the knowledge that at the temperature below 700°C, oxidation reaction is very slow and above 900°C, oxidation reaction is dominated by mass transfer through the boundary layer of gas flow. As shown in the figure, the analytical and experimental results agreed well at the temperature range of above 800°C. As analytical results below 700°C are considerably higher than the experimental results, the graphite oxidation by the GRACE code is estimated conservatively at this temperature range.

Fig.12.12 Comparison of axial distribution of oxidation depth between experiment and analysis for GRACE code
12.4.7 OXIDE-3F Code

(1) Outline

The OXIDE-3F code is used to analyze the oxidation reaction of graphite structures with steam ingressed in the PCS by rupture of the heat transfer tube of the PWC, considering heat transfer in the core. This code is originated in OXIDE-3 code(12-20) which was developed to analyze the oxidation by steam in the core of the multi-hole-type fuel assembly. The OXIDE-3F code is a modified version of the OXIDE-3 code to treat the configuration of the HTTR reactor core and structures, i.e., hot plenum blocks and support posts. On the basis of estimated amount of ingressed steam, the distribution of the oxidation amount of the graphite structure, transient steam concentration in gas, the partial pressure of steam and total pressure in the PCS are obtained by solving the two-dimensional axi-symmetric transient diffusion equations.

A mixed gas flow during heat transfer tube rupture of the PWC is schematically shown in Fig.12.13. The reactor was modeled by dividing it into several regions. A fuel channel in the core was modeled using a cylindrical channel.

![Diagram of reactor core and components](image)

*Fig.12.13 Flow of mixed gas in case of water-ingress accident*
Analytical models of the PCS and the CV are shown in Fig. 12.14. The mixed gas is released into the CV when the pressure of the gas containing steam and FP's in the PCS exceed the pre-set pressure for the safety valve.

![Diagram](image)

**Fig. 12.14** Analytical model of primary system and containment vessel for OXIDE-3F code

(2) Validation of the code

Corrosion of graphite structures is caused by the reaction between graphite and active gas (H, O, O₂) which diffused into the graphite through the boundary layer. Important factors in estimating the oxidation rate are the mass transfer coefficient and chemical reaction rate.

In the OXIDE-3F code, mass transfer coefficient is derived from the experiments for heat transfer correlations. Correlation between the mass and heat transfer coefficients is obtained by graphite oxidation experiments. The mass transfer coefficient obtained from the experiment is in good agreement with the value derived from the heat transfer correlations. The chemical reaction rate is properly estimated based on the graphite oxidation experiment of steam.

The method to calculate the rate of graphite oxidation in the OXIDE-3F code is basically the same as that of the GRACE code. Therefore, the validity of the OXIDE-3F code was confirmed.

**12.4.8 FLOWNET/TRUMP Code**

(1) Outline

The FLOWNET/TRUMP\textsuperscript{12-24} code is used to calculate the temperature distribution in the fuel
block when a coolant channel was blocked. This code is the combination of the FLOWNET code\textsuperscript{(12-25)} which is a flow distribution analysis code and the TRUMP code\textsuperscript{(15-26)}, which is a generally used three-dimensional heat conduction code. The TRUMP code has been used for heat transfer analysis of the cask of nuclear fuels\textsuperscript{(15-27)}. In the analysis, the nodes are selected in the coolant passages inside or outside the fuel blocks. The nodes are connected by branches which are one-dimensional and have an equivalent crosssection area, length and hydraulic diameter. Coolant pressure and flow rate are calculated by a flow network model consists of the nodes and branches. The temperature in solids and coolant are calculated by dividing the graphite block and fuel into several elements with finite volume.

(2) Validation of the code

A validation analysis of the FLOWNET/TRUMP code was performed by comparing the results of the uniform and non-uniform power distribution tests which were carried out using multichannel test rig of HENDEL. The HENDEL T, simulates 12 fuel cooling channels and is an apparatus to study heat transfer characteristics of the fuel elements in the high pressure and high temperature helium, which is the same condition as the HTTR. A non-uniform power test was performed by varying the power of arbitrary fuel (in practice, No.6 fuel rod, see Fig.12.15). Heat transfer through graphite block is important in case of analysis of the channel blockage accident and the nonuniform power test is effective for the estimation of validity of the code.

![Diagram of fuel channel and graphite block]

**Fig.12.15** Comparison of surface temperature of simulated fuel between analysis and experiment in case of channel blockage for FLOWNET/TRUMP code
As the most severe case, a channel blockage experiment was performed where about 90% of the total crosssection was blocked at the inlet of No.6 channel. The maximum surface temperature of the simulated fuel rod is shown in Fig.12.16. Analytical results show a similar profile as the experimental results and the analysis give conservative results.

![Flow rate and Revolution number graph](image)

**Fig.12.16** Damping characteristics of helium circulator

### 12.5 Major Analytical Conditions

Analytical conditions commonly used and factors to be considered in the safety analyses of abnormal events are described below.