Development of Integrated Analytical Tools for Level-2 PSA of LMFBR

December 8, 2009

K. Haga, H. Endo, T. Nakajima, T. Ishizu
Nuclear Energy System Safety Division
Japan Nuclear Energy Safety Organization (JNES)

H. Iizuka
MHI Nuclear Engineering Company
Contents

1. Background and Objectives
2. Computer codes
   2.1 For the plant response phase
      NALAP-II
      ACTOR
   2.2 For the core disruptive phase
      ARCADIaN-FBR
      APK
   2.2 For the Containment vessel
      AZORES
3. Phenomenological Relation Diagram : PRD
4. Summary
1. Background and Objectives

- JNES is engaged in the technical support for the regulatory activity for Monju, the Nuclear and Industrial Safety Agency (NISA).
- The major areas of the supportive work are Inspection and Safety analysis.
- To the safety analysis, JNES has been developing the own tools for several years.
- Our reset big activity is the effectiveness evaluation of the proposed accident management (AM) of Monju by using PSA.
- The developed tools for the PSA are not only computer codes but also the decision method for the balancing points of event trees.
- This presentation presents the outline of each tool.
Process of severe accidents of LMFBR and JNES’s Analysis tools

Plant response phase
- Sodium temp. & press. increase
- Boundary failure

Core disruption phase
- Sodium boiling
- Core disruption
- Re-criticality
- FP release
- Keff and reactivity coeffs. of degraded core
- FP evap. from fuel
- FP transport in Na, from Na to cover gas
- Cover gas heating by FP within cover gas space
- FP deposition on piping wall

CV response phase
- Na-concrete reaction
- Debris-concrete reaction
- Na spray burning
- H2 burning
- Aerosol release
- FP release
- Power and reactivity transient of Super prompt criticality

Containment Vessel responses & FP transport
- Hydrogen behavior due to Na-concrete and debris reactions
- Hydrogen burning
- Aerosol and FP transports

Analysis tools
- NALAP-II
  Plant kinetics
  - Coolant temp.
  - System pressure
- ARCADIAN-FBR
  Neutronics
  - Keff and reactivity coeffs. of degraded core
- APK
  Super-prompt criticality
- ACTOR
  FP behavior in PHTS
  - FP evap. from fuel
  - FP transport in Na, from Na to cover gas
  - Cover gas heating by FP within cover gas space
  - FP deposition on piping wall
- AZORES
  Containment Vessel responses & FP transport
- ABAQUS
  Structure Analysis
  - Large deformation and inelastic deformation analysis
- EFD
  Phenomenological ET expansion
- PRD
  Quantification pf phenomenological ET
- CFF
  Containment failure Frequency
2. Computer codes
2.1 For the plant response phase (1)

NALAP-II code

Functions: Plant kinetics of the primary and secondary loops
Sales point: A function to evaluate the degree of high-temperature creep of the location by SCDF (Structural cumulative damage factor, $D_c$).

\[ D_c = \sum \left( \frac{\Delta t_i}{t_r} \right), \quad (1) \]

where,

$\Delta t_i$: computation time step (s),
$t_r$: creep failure time (s).

In eq. (1), $t_r$ depend on the material, temperature and pressure. It is calculated by using the Larson-Miller Parameter.

It was considered that the location will fail when $D_c = 0.2 \sim 1.0$. 
• The RV outlet temperature exceeded 800°C at 51hr.
• PHTS pressure began to increase at 22 hr due to the sodium volume expansion.
• The SCDF is evaporator cylindrical wall made of 2.25Cr-1Mo steel firstly exceeded the line of 0.2.
• Another locations exceeded the line of SCDF=0.2 after about 10 hr.
2.1 For the plant response phase (2)

[ACTOR code]

Functions: FP behavior from fuel to cover gas or flowing sodium

Sales point: Analysis of FP gas bubble behavior with other FP nuclides
Calculations by ASFOR

Amount of iodine transferred per bubble volume: Experiment & Calculation

Distribution of FP nuclides in reactor 100s after cladding failure in PLOHS


2.2 For the core disruption phase (1)

ARCADIAN-FBR code

Functions: Neutronics for degraded core

Sales point: The combination of

(1) a deterministic calculation part

to provide the various reactivity coefficients
and those spatial distributions

and

(2) a Monte Carlo calculation part

to provide reference results against the
deterministic calculations and to evaluate
approximation effects employed in the
deterministic calculations
Constitution of ARCADIAN-FBR

Deterministic Calculation Part

JFS-3-J3.3
-> Cell Code SLAROM
-> 70-group Effective Cross Sections $\sigma_{\text{eff}}$
- Diffusion Code DIF3D
- Transport Code VARIANT
- $k_{\text{eff}}, \phi, \phi^*$
- Burnup Cal. Code REBUS
- Perturbation Code PERKY
- $\rho, \ell, \beta$

Monte Carlo Calculation Part

JENDL-3.3
-> Monte Carlo Code GMVP
- $k_{\text{eff}}, \phi, \phi^*$
- Kinetics Parameter Calculation Module
- $\ell, \beta$
- Burnup Cal. Code MVP-BURN
- $N$
2.2 For the core disruption phase (2)

APK code

Functions: Reactivity evaluation by solving 6-group point reactor kinetics equations to debris beds of some temperature

Sales point: Simple using one point temperature

Validation: Comparison with SAS4A

(a) Power history  (b) Net reactivity  (c) Average fuel temp.  (d) Input energy
2.3 For the CV response phase

AZORES code

Functions: To analyze various reactions caused by the sodium and core materials leaked through a failed coolant boundary.

Sales point: Radioactive materials behaviours to the environment are also solved with the thermo-chemical reactions.
Example of CV response calculated by AZORES

Fig. 7-22(1) Core Temperature (Case 15: PLOHS, Non-CVBp, PL-NSL-H-γ)

Fig. 7-22(2) Pressure (Case 15: PLOHS, Non-CVBp, PL-NRL-H-γ)

Fig. 7-22(3) Hydrogen Mol Fraction (Case 15: PLOHS, Non-CVBp, PL-NRL-H-γ)

Fig. 7-23(1) Xe Gr Initial Inventory Ratio (Case 15: PLOHS, Non-CVBp, PL-NRL-H-γ)

Fig. 7-23(2) I Gr Initial Inventory Ratio (Case 15: PLOHS, Non-CVBp, PL-NRL-H-γ)

Fig. 7-23(3) Ce Gr Initial Inventory Ratio (Case 15: PLOHS, Non-CVBp, PL-NRL-H-γ)
3. Phenomenological Relation Diagram : PRD
Probability decision method for the blanching points in event trees (ET)

- **Objectives of PRD:** To eliminate the subjective decision for the probability of blanching points in ETs.

- **Procedure:**
  1. Constructs an event tree from the top event (an event considered at a branching point of an phenomenological event tree) to lower events.
  2. Function gate is settled to calculate by a certain function with lower events.
  3. To lower events the probabilities are to be give by any method as much as possible. The probability distributions are transferred to the upper events.
**Japan Nuclear Energy Safety Organization**

**-PRD Approach for the top event of acceleration -**

**Function gate**

\[ \text{Motion equation: } F = ma \]

**Events** (outputs from the function gate)

**Top event**

[Acceleration: \( a \)]

**Conditional branch**

- Yes
- No

**Sub-PRD**

[Mass: \( m \)]

**Events**

- Force: \( F \)

**Probability distribution function**

- Events are composed of several events. This case represents that events are composed of elementary events\( \text{① to ③} \).

**Incidence probability distribution**

To be calculated by a certain function with lower events as inputs. The details of calculation are indicated in the adjoining blue frames.

**Probability distributions**

- Probability distributions are given to events that are considered to have a margin of uncertainties. Distributions to be given are defined by “the probability distribution function.”

**To be expanded in detail in a lower PRD**

**Bottom event**

**Incidence probability distribution**

- The probability distribution function to be given to a certain variable: Example
  - \( U[1,10] \): A uniform distribution between 1 and 10
  - \( N[10,5] \): A normal distribution with the average value of 10 and the standard deviation of 5
  - \( L,U=[0,100] \): Lower limit – 0; Upper limit - 100

**Elementary event ①**

- [F1]

**Elementary event ②**

- [F2]

**Elementary event ③**

- [F3]
Evaluation procedures of ROAAM used LWR and PRD

Both methods seem basically identical.
Summary

● With the developed integrated analytical tools, one-through evaluation for severe accidents including FP release ratio became possible to LMFBR. The tools were used for the effective evaluation of AM for Monju.

● Now, further efforts are being made to make analyses more realistic for the Monju with an advance core and the Japanese demonstration LMFBR.

● These improvements are very helpful in constructing data-bases of the Emergency Response Support System (ERSS) for Monju by conducting more reliable analyses to the conceivable scenarios after initiating events.
Many thanks for your audience.
Characteristics of PLOHS

- After the reactor shutdown, the plant temperature increases slowly but monotonously both in the primary cooling system (PHTS) and in the secondary cooling system (SHTS).
- Many plant locations will fail due to the high-temperature creep.
- Among several conceivable accident scenarios, the containment vessel by-bass (CVBP) would be most important event due to the high probability and the high potential of large scale FP release to the environment.
- We evaluated the frequency of PLOHS/CVBP in this study.
FP leak passes in PLOHS/CVBP

In the case of PLOHS/CVBP, some early failure in SHTS is postulated. When some interface between PHTS and SHTS failed, FP from the molten fuel will run through PHTS and SHTS piping.
Boundaries of IHX that have a high potential of first failure in PHTS

- **Cover gas bellows**
  - Diameter: 700mm
  - Thickness: 1.5, 2.0 mm
  - Material: SUS3316

- **IHX bellows**
- **Bottom head** (interface between PHTS/SHTS)
  - Diameter: 2190mm
  - Thickness: 25mm
  - Material: SUS304

- **Heat transfer tubes**
An vent tree of PLOHS

<table>
<thead>
<tr>
<th>Power supplies</th>
<th>Leakage from the shield plug of RV</th>
<th>Earlier secondary system failure</th>
<th>Failure in some areas of primary system</th>
</tr>
</thead>
<tbody>
<tr>
<td>yes ↑</td>
<td></td>
<td>(a)</td>
<td></td>
</tr>
<tr>
<td>no ↓</td>
<td></td>
<td>(b)</td>
<td></td>
</tr>
<tr>
<td></td>
<td>0.25</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>PLOHS</td>
<td>0.96</td>
<td>(a)</td>
<td>CVBP</td>
</tr>
<tr>
<td></td>
<td>0.75</td>
<td></td>
<td>(d)</td>
</tr>
<tr>
<td></td>
<td></td>
<td>(b)</td>
<td></td>
</tr>
<tr>
<td></td>
<td>0.044</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

The probabilities of branching point (a), (b), (c), (d) are obtained through present analyses.
2. Analyses to obtain the probability of the branching points

<table>
<thead>
<tr>
<th>Failure mechanism</th>
<th>Objects for analysis</th>
<th>Analysis code</th>
</tr>
</thead>
<tbody>
<tr>
<td>High-temperature creep</td>
<td>Vessel, Piping</td>
<td>NALAP-II</td>
</tr>
<tr>
<td>Buckling</td>
<td>Bellows and bottom head of IHX</td>
<td>ABAQUS</td>
</tr>
<tr>
<td>Tensile (or breaking) stress</td>
<td></td>
<td>FINAS</td>
</tr>
</tbody>
</table>

*Thermal-hydraulic Analysis*

*Structural Analysis*
2.1 Thermal-hydraulic Analysis

FBR plant thermal-hydraulic analysis code NALAP-II

- NALAP-II has been developed in JNES.
- For the present study a function was added to evaluate the degree of high-temperature creep of the location: SCDF (Structural cumulative damage factor, $D_c$).

- $D_c = \sum (\Delta t_i/t_r)$, \hspace{1cm} (1)

  where,

  $\Delta t_i$: time step (s),
  
  $t_r$: creep failure time (s).

In eq. (1), $t_r$ depend on the material and it is calculated by using the Larson-Miller Parameter.

It was considered that the location will fail when $D_c=0.2\sim1.0$. 
Specifications of proto-type LMFBR plant and locations where high-temperature creep was evaluated

(a) Spec. of LMFBR

<table>
<thead>
<tr>
<th>Spec. of LMFBR</th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>Thermal power</td>
<td>714 MW</td>
</tr>
<tr>
<td>Number of loops</td>
<td>3</td>
</tr>
<tr>
<td>Primary heat transfer system (PHTS)</td>
<td></td>
</tr>
<tr>
<td>Temperature at reactor inlet/outlet</td>
<td>397/529 °C</td>
</tr>
<tr>
<td>Flow rate</td>
<td>5100 t/h</td>
</tr>
<tr>
<td>Pressure at reactor inlet/outlet</td>
<td>0.8/0.1 MPa</td>
</tr>
<tr>
<td>Main circulation pump</td>
<td>One unit per loop</td>
</tr>
<tr>
<td>IHX</td>
<td></td>
</tr>
<tr>
<td>Number of unit</td>
<td>3</td>
</tr>
<tr>
<td>Type</td>
<td>Vertical parallel flow with no sodium surface</td>
</tr>
<tr>
<td>Secondary heat transfer system (SHTS)</td>
<td></td>
</tr>
<tr>
<td>Temperature at IHX inlet/outlet</td>
<td>3700 °C</td>
</tr>
<tr>
<td>Flow rate</td>
<td>0.8/0.1 MPa</td>
</tr>
<tr>
<td>Main circulation pump</td>
<td>One unit per loop</td>
</tr>
<tr>
<td>Auxiliary cooling system</td>
<td>By-pass of the secondary cooling system with one air cooler per loop</td>
</tr>
</tbody>
</table>

(b) Locations of plant evaluated high-temperature creep

<table>
<thead>
<tr>
<th>Location</th>
<th>D (inner dia.) (mm)</th>
<th>t (thickness) (mm)</th>
<th>D/t</th>
<th>Material</th>
</tr>
</thead>
<tbody>
<tr>
<td>RV cylindrical wall</td>
<td>7100</td>
<td>50</td>
<td>142</td>
<td>SUS304 stainless steel</td>
</tr>
<tr>
<td>Reactor outlet piping</td>
<td>788</td>
<td>11</td>
<td>72</td>
<td>SUS304 stainless steel</td>
</tr>
<tr>
<td>Reactor inlet piping</td>
<td>591</td>
<td>9.5</td>
<td>62</td>
<td>SUS304 stainless steel</td>
</tr>
<tr>
<td>IHX vessel cylindrical wall</td>
<td>2940</td>
<td>30</td>
<td>98</td>
<td>SUS304 stainless steel</td>
</tr>
<tr>
<td>IHX heat transfer tube</td>
<td>19.3</td>
<td>1.2</td>
<td>16</td>
<td>SUS304 stainless steel</td>
</tr>
<tr>
<td>IHX SHTS-side outlet piping</td>
<td>540</td>
<td>9.5</td>
<td>56</td>
<td>SUS304 stainless steel</td>
</tr>
<tr>
<td>EV cylindrical wall</td>
<td>2900</td>
<td>50</td>
<td>58</td>
<td>2.25Cr-1Mo steel</td>
</tr>
<tr>
<td>Air-cooler heat transfer tube</td>
<td>45</td>
<td>2.9</td>
<td>16</td>
<td>SUS304 stainless steel</td>
</tr>
<tr>
<td>Fuel pin cladding</td>
<td>5.56</td>
<td>0.47</td>
<td>11.8</td>
<td>20% cold-worked modified 316 stainless steel</td>
</tr>
</tbody>
</table>

The locations with large value of D(diameter)/t(thickness) were chosen for the present calculation of SCDF.
2.2 Structural analysis ABAQUS 3-d analysis to IHX bottom head
- Buckling critical stress and fracture critical stress -

The buckling of IHX bottom head became marked at the end of calculation: Temperature: 838.7°C; Pressure: 0.89MPa

Stress-pressure relationship equation in IHX bottom head
\[ \sigma (\text{MPa}) = \theta \times a \times (P - 0.1) \]
a = 89.89, P (MPa): Primary system pressure
\( \theta \): A margin of uncertainties in an approximate expression
Probability variable (0.95~1.05)

The obtained stress-pressure relation curve encountered with the buckling critical stress line and the fractional critical stress line at one point, respectively. The IHX bottom head is expected to fail at the condition between the two points.
The results of FINAS 2-d analysis to IHX bellows
- Yield stress and tensile stress -

Deformed IHX bellows (Temperature: 750°C; Pressure: 0.388 MPa) (The results of the FINAS 3-d calculation)

IHX bellows (SUS316) buckling critical stress and fracture critical stress

\[ \sigma (\text{MPa}) = \phi \times a \times (P-0.1) \]

- \( a = 136.25 \)
- \( P(\text{MPa}): \) Primary system pressure
- \( \phi: \) A margin of uncertainties in an approximate expression (0.95~1.05)

The obtained stress-pressure relation curve encountered with the yield stress line and the tensile stress line at one point, respectively. The IHX bellows is expected to fail at the condition between the two points.
3. Discussion

Pressure range for failure of PHTS components

Five locations (RV cylindrical wall, RV outlet piping, IHX cylindrical wall, IHX bellows and IHX bottom head) fail at the pressure between 0.77 to 2.3 MPa.
The five locations fail at a narrow time period (between 52hr to 59hr).
Thus, it will be hard to say the failure order in these locations.
In PSA, it will be reasonable to assign the equal probability of first failure to these locations, that is 0.2.
4. Conclusion

(i) A model to analyze the high temperature creep progression by introducing the calculation function of SCDF was added to the LMFBR plant thermal-hydraulics code, NALAP-II.

(ii) The model was applied to the PLOHS event of the typical proto-type LMFBR. The results indicated that the evaporator made of 2.1/4Cr-1Mo steel in SHTS firstly failed when the system temperature exceeded 800 °C. The main plant components and piping made of SUS 304 stainless steel failed when the system temperature exceeded 870 °C.

(iii) In addition, detailed structural analyses were performed by using ABAQUS and FINAS codes and the temperature and pressure histories to the locations where the buckling and the tensile stresses are the causes of failure.

(iv) Comparing the failure time information of each location, it was concluded that the probability of CVBP was 0.2 to the plant.
Present trial evaluation shows that the frequency of CVBP is one order smaller than that of retained RV integrity scenario. However, the frequency is more than one order larger than that of other scenarios.
Flow network of the NALAP-II code for RV and PHTS

Fig.5 Flow network of the NALAP-II code for RV and PHTS
IHX inlet and outlet temperatures, and PHTS flow rate
Temperature and the Mises stress behavior of IHX bellows
(Stress with the maximal element of equivalent creep strain: the center of plate pressure)
(The results of the FINAS-based two-dimensional elasto-plastic large-scale deformation and creep analysis)
Temperature and the Mises stress behavior of IHX bellows
(Stress with the maximal element of equivalent creep strain:
the center of plate pressure)
(The results of the FINAS-based two-dimensional elasto-plastic
large-scale deformation and creep analysis)
Comparison of initiating events frequency between JAEA and JNES

Japan Nuclear Energy Safety Organization
4. Results of PSA

4.1 Effects of AM measures to keep the RV coolant level

- Primary main coolant system circulation pump trip failure
- Isolation of argon supply line to RV
- Keeping sodium temperature high
- Siphon break of maintenance cooling system
- Loss of guard vessel integrity
- Double leak in area below SsL following leakage from the primary coolant system
- All loss of reactor level (LORL) event

**Graph:**
- Before AM Measures
- After AM Measures

<table>
<thead>
<tr>
<th>Event Description</th>
<th>Core Damage Frequency (Reactor Year)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Primary main coolant system circulation</td>
<td>$10^{-12}$</td>
</tr>
<tr>
<td>Isolation of argon supply line to RV</td>
<td>$10^{-11}$</td>
</tr>
<tr>
<td>Keeping sodium temperature high</td>
<td>$10^{-10}$</td>
</tr>
<tr>
<td>Siphon break of maintenance cooling system</td>
<td>$10^{-9}$</td>
</tr>
<tr>
<td>Loss of guard vessel integrity</td>
<td>$10^{-8}$</td>
</tr>
<tr>
<td>Double leak in area below SsL</td>
<td>$10^{-7}$</td>
</tr>
<tr>
<td>All loss of reactor level (LORL) event</td>
<td>$10^{-6}$</td>
</tr>
</tbody>
</table>
4.2 Effects of initiating events frequency

- Loss of reactor shutdown functions
- Loss of functions to maintain a proper reactor coolant level
- Loss of decay heat removal functions
- All relevant events

Frequency (/ry)

- JNES before AM
- JAES before AM
- JNES after AM
- JAEA after AM

Frequency (1.0E-09 to 1.0E-06)
4.3 Effects of choosing common cause failure

(In the PSA of JAEA, common cause failure is considered between the main reactor shutdown system and the backup reactor shutdown system)

No AM Measures for ATWS

<table>
<thead>
<tr>
<th>Event Type</th>
<th>JNES Before AM</th>
<th>JNES After AM</th>
<th>JAEA Before AM</th>
<th>JAEA After AM</th>
</tr>
</thead>
<tbody>
<tr>
<td>LORL</td>
<td>1.0E-09</td>
<td>1.0E-09</td>
<td>1.0E-09</td>
<td>1.0E-09</td>
</tr>
<tr>
<td>PLOHS</td>
<td>1.0E-09</td>
<td>1.0E-09</td>
<td>1.0E-09</td>
<td>1.0E-09</td>
</tr>
<tr>
<td>All</td>
<td>1.0E-09</td>
<td>1.0E-09</td>
<td>1.0E-09</td>
<td>1.0E-09</td>
</tr>
</tbody>
</table>

Frequency (/ry)

- ATWS increases
- 2.7 times
- 3.8 times

ATWS increases with common cause failure consideration.
4.4 Contribution of initiating events to CDF (b) after AM  (by JNES)
4.5 Comparison of estimated CDF between JNES and JAEA