Analysis Methodologies for Beyond Design Basis Accidents in SFR: Current Status and Future Directions

P. Chellapandii

Indira Gandhi Centre for Atomic Research
Kalpakkam-India

IAEA Workshop on Safety Aspects of Sodium Cooled Fast Reactors
23-25 June, IAEA Headquarters, Vienna
## Typical BDBE Analysed in SFR

### Analysis done to demonstrate that the following events do not occur

- Leakage from both main and safety vessels
- Multiple primary pipe rupture
- Multiple pump seizure
- Multiple primary pipe rupture
- Failure of core support structure
- Ejection of all control rods from the core
- Gas entry into the core

### Analysis done for studying consequences of following Initiating Events

- Total loss of all DHR systems on demand
- Loss of off-site power without SCRAM
- Continuous withdrawal of one CR without SCRAM

### Severe Accident Analysis

- Total and instantaneous blockage of a fuel SA
- Core Disruptive Accident
- Sodium fire on roof slab
Advanced Analyses to Comply Seismic Design Criteria

- After main vessel leak, the net hydrostatic force acting on the main vessel is transmitted to the safety vessel and the force is further amplified significantly during any seismic event due to dynamic pressures generated in the liquid confined in the inter-space between main and safety vessels due to fluid structure interaction effects.

Future direction: Investigation of cliff-edge effects
Analysis of Primary Pipe Rupture Event for SFR

**Important scenarios are:**

- Increase of power to 112 % within 0.6 s
- Increase of clad hot spot temperature to 1267 K
- Increase of coolant hot spot temperature to 1107 K.

- The values are less than the respective category 4 allowable temperature limits
- SCRAM action is initiated by power to flow ratio \((P_0/Q_0)\) signal comes at 0.06 s

**Future Directions:**

- Re-distributed core flow pattern calls for advanced analysis combining multidimensional code for flow analysis and system code for core neutronic and thermal modeling
- Multiple pipe rupture for future reactors, which would have 4 pipes per pump
Advanced Thermal Hydraulics Analyses

**Failure of all DHRs**

- Heat removal by other cooling systems, viz. reactor vault cooling and top shield cooling etc. is insignificant (~ 1 MW)
- Core safety is not challenged since pumps are working.
- Main vessel temperature exceeds the rupture limit after ~ 3 h. Efforts are needed to bring back the DHR systems (either OGDHRS or SGDHRS) before 3 h
- Failure of all DHR along with failure of all pumps is to be studied taking in to account inter wrapper flow effects.

**Failure of all Pumps**

**Approach**

Analysis with multi-dimensional model for pools with inter-wrapper space (StarCD) and 1-D model for equipments and piping (DHDYN).

**Main Conclusion**

Availability any two circuits for 7 h and one circuit subsequently with primary circuit under natural convection is sufficient to limit the temperatures below category 4 limits.
Gas Entrainment in SFR

**Analysis Method:**

- CFD prediction of gas entrainment at its infancy. Codes used: PHOENICS, STAR-CD & OpenFOAM.
- Phenomenon is governed by Froude, Weber & Reynolds (Re) numbers. Respecting Froude and Weber numbers, water model demands a large scale of 5:8.

**Design Solutions:**

- Horizontal baffle in hot pool to minimize free surface velocity to ~ 0.5 m/s. Overflow weir in MV cooling system to avoid flow separation sufficient sodium depth in restitution plenum. Provision of purger SA in grid plate.

**Future Direction:**

- Development of CFD based model to predict physical, chemical and thermal aspects of gas entrainment scenario and coupling it with neutronics for reactivity feedback and power evolution.
TIB of a Single fuel Subassembly

- Blockage adopter holes are provided at top to allow a positive flow in the event of TIB.
- Due to TIB at the top, sodium flow through reduces to 40% of nominal value (Code: STAR-CD).
- This is adequate to prevent sodium boiling.
- Temperature rise recorded by the core monitoring TC leads to reactor SCRAM, for all SA except for the central SA.

Flow distribution in IW space during TIB
Flow rate is determined to be 40% nominal.

Multiple radial entry holes are provided to minimize the probability of TIB at the bottom.

Temperature below Lattice Plate during TIB in peripheral SA. Blockage detection ensured by core monitoring TC.
Analysis of Mechanical Consequences of CDA

- Preliminary Deformations of Components (0 - 50 ms)
- Upward Motion of Sodium Slug (50 – 100 ms)
- Sodium Slug Impact on top shield and development of Transient Forces on Reactor Vault (100-150 ms).
- Sodium Release to RCB during quasi-static state (150 – 900 ms)
- Post Accident Heat Removal Condition depending upon coolability of core bubble (> 900 ms)
- Effects of internals contribute by changing (i) energy release, (ii) transient forces and (iii) duration of major phenomenon, by virtue of their geometrical features, location and inertial characteristics.
International Computer Codes

Lagrangian Codes:
• REXCO-H developed at ANL (FFTF)
• REXCO-HEP and ASTARTE of UKAEA
• ARES of Interatom Germany (SNR-300)
• SIRIUS of CEA France (SPX-1 and SPX-2).

Eulerian Codes:
• ICECO, developed in 1975.
• SEURBNUK-2 by Euratom, Belgonucleaire, and the UK for the analysis CDFR and SNR-300
• PISCES 2 DELK, an Eulerian version of the PISCES code.

Eulerian-Lagrangian Coupled Codes:
• CASSIOPEE (SPX-1)
• EURDYN (SPX-1 and SPX-2)
• ALICE (CRBRP)
• PLEXUS (EFR)
• FUSTIN (PFBR)
Complexities in Numerical Simulations

- Large distortions in fluids
- Large displacements in structures
- Fluid-structure interactions:
  - Structures having part contact with fluid
  - Structures having full contact with fluid
- Intersection of interfaces:
  - Solid-fluid & liquid-gas boundaries
- Sodium slug impact
- Shock wave propagation
- Numerical integration
Features of FUSTIN Code

- Axisymmetric Finite Element Code
- Capability to model fluid, structure, fluid-structure interactions, complicated movements of core bubble, sodium free surface motions, shock wave propagation, large displacement of fluids and structures.
- Arbitrary Eulerian Lagrangean co-ordinate system for fluids. Convected co-ordinate system for structures.
- Special algorithm, viz. ‘multi-phase element’ to compute the pressure in each element which has liquid/liquid or liquid/gas or gas/gas phases.
- Efficient numerical solution techniques (explicit algorithm to calculate initial values and subsequent implicit algorithm to get converged values).
- Efficient automatic rezoning at end of every time step.
- Code has been extensively validated.
MANON Test Programme: CEA/DRNR, Cadarache
Spherical shape, low pressure, low density explosive source. Deformable steel cylinder completely filled with water and closed at top and bottom by a rigid flat roof.

COVA Test Series: UKAEA and Joint Research Centre, Euratom
A series of small scale (the maximum diameter of the vessel is 560 mm and height varied over the range of 700-1120 mm), Well instrumented tests aimed at providing high quality data on stresses, strains and loads when a well characterised energy source is released within a fluid in a containment vessel.

CONT Benchmark Analytical Problem: Sponsored by CEC, Italy.
A full size CDFR (UK) Vessel undergoing a postulated CDA Analysis results using six computer codes: ASTARTE, CASSIOPEE, PISCES–2 DELK, SEURBNUK, SIRIUS and SEURBNUK / EURDYN
Validation of FUSTIN: MANON Test Programme

Vessel mid point displacement

Bubble Pressure: \( P = P_0 \left( \frac{V_0}{V} \right)^{1.535} \) (MPa)

\( P_0 = 288 \text{ MPa} \) \( V_0 = 93 \text{ cm}^3 \)

Evolution of core bubble growth
Validation of FUSTIN: COVA Test Programme

Geometry  stress-strain curve  Thickness variation

0  0.5 ms  1.0 ms  1.5 ms  2.0 ms

Hoop strain  Meridinal strain
Validation of FUSTIN: CONT Benchmark

**Equation of State**

**Core bubble:**

\[ P = P_o \left( \frac{V_o}{V} \right)^{0.75} \text{ (MPa)} \]

\( P_o = 10 \text{ MPa} \quad V_o = 4.2 \text{ m}^3 \)

**Argon:**

\[ P = P_o \left( \frac{V_o}{V} \right)^{1.67} \text{ (MPa)} \]

\( P_o = 0.1 \text{ MPa} \quad V_o = \pi \text{ m}^3 \)

**Sodium:**

\[ P = 4.44 \times 10^9 \mu + 4.328 \times 10^9 \mu \left/ \mu + 1.218 \times E(1+\mu) \right/ \]

\( \rho_o = 832 \text{ kg/m}^3 \quad \text{and} \quad \mu = (\rho/\rho_o - 1) \)

\( E = \text{Energy per unit volume (MJ/m}^3) \)
Pressure Propagation in fluids
Validation of FUSTIN: CONT Benchmark

Vessel strain vessel

Cover gas pressure

Hoop strain

Pulldown load on top shield
Basic Input Data

Initial Core: \( V_0 = 2.5 \text{ m}^3 \) \( P_0 = 4 \text{ MPa} \).

\[
P = P_0 (V_0/V)^{0.72} \text{ MPa}
\]

Sodium: \( P = 4.44 \times 10^9 \mu + 4.328 \times 10^9 \mu \frac{|\mu|}{\rho_0 - 1} + 1.218 \times e(1+\mu) \)

- \( e \) - energy per unit volume (MJ/ m³)
- \( \mu \) - \( (\rho/\rho_0 - 1) \) and \( \rho_0 = 832 \text{ kg/m}^3 \)

Cover gas: \( P = 0.1 [(V_0/V)^{1.67} - 1] \), where \( V_0 = 105 \text{ m}^3 \).
Transient Response under CDA (100 MJ)

Analysis-1: Main vessel without internals

Analysis-2: Main vessel with internals
Important Results

- Structural integrity of MV is ensured with comfortable margin for 100 MJ for CDA in PFBR.

For Future SFRs:
- Development of 3D code to include IHX and DHX in the analysis model
- Development of an integrated code to include reactor physics and sodium release through TS.
Computer Application for Manufacturing & Erection of Reactor Assembly Components

Step-1
- Transportation
- Mounted on inner wall

Step-2
- Placing of pads and construction of upper lateral portion of outer wall

Step-3
- Mounting of grid plate and inner vessel
- Mounting of top shield

Step-4
- Erection of Grid Plate, Inner Vessel & Top shield
- Alignment and fit up of main vessel and top shield shell
- Final assembled view

Erection of Main Vessel along with Core Catcher & Grid Plate
- Transportation
- Mounting on pads
- Transmitting the loads to outer wall through tie rods

Welding of Main Vessel with Top shield

Insert the Bottom Plate Assembly Over Shielding Box Assembly
SIMULATION OF NATURAL CONVECTION IN HOT AND COLD POOLS OF PHENIX REACTOR

Transient flow patterns

Transient Temperature distributions
Benchmark Analyses of Sodium Natural Convection in the Upper Plenum of the MONJU Reactor Vessel
• Excellent thermal hydraulics as well as structural mechanics analysis capability towards SFR safety assessment exists.
• For Structural analyses, CAST3M and PLEXUS codes have been used in India since 1985 (longest experience in the world)
• Coupled codes for comprehensively addressing reactor physics, thermal and mechanical aspects are our next step
• Computer codes for fuel modeling under extreme loading conditions and gas entrainment analysis are being developed
• Many young minds are involved in developing, validating and using the codes
• All these efforts reflect on future SFRs to meet the ultimate objective of achieving enhanced safety and improved economy
• Existing internal collaborations would be strengthened and new collaborations would emerge on the basis of challenges and mutual interest
Thank You