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APPENDIX 14A - COMPUTER CODES USED IN CHAPTER 14

This appendix contains brief descriptions of the computer codes used within the framework of the accident analysis of PCC-events (Chapter 14), RRC-A sequences (Sub-chapter 16.1) and overpressure protection analyses (section 1 of Sub-chapter 3.4).

The codes occur in the following sections of this appendix:

- Section 1 : S-RELAP5
- Section 2 : COCO
- Section 3 : NLOOP
- Section 4 : PANBOX/COBRA
- Section 5 : ORIGEN-S
- Section 6 : PRODOS-B
- Section 7 : ACARE
- Section 8 : CATHARE
- Section 9 : THEMIS
- Section 10 : SMART
- Section 11 : FLICA III-F
- Section 12 : ALICE 2
- Section 13 : CONPATE 4
- Section 14 : MANTA
- Section 15 : MANTA/SMART/FLICA
- Section 16 : COMBAT

Appendix 14A – Tables 1 to 3 show the computer codes used for each event:

- a single slash means no computer code used,
- a slash between code names means coupled computer codes (e.g. MANTA/SMART/FLICA),
- a comma indicates the use of non-coupled computer codes (e.g. CATHARE, CONPATE),

- "App. 14B" means that code was used in a transient calculation performed in the BDR-99 for EPR at 4900 MWth, with no re-calculation performed in the PCSR for EPR at 4500 MWth. This calculation is presented in Appendix 14B.

APPENDIX 14A – TABLE 1

Computer codes used in Chapter 14
Safety Analyses (PCC)

Section(1)	Event	Computer Codes
14.3.1	Feedwater malfunction causing a reduction in feedwater temp.	/
14.3.2	Feedwater malfunction causing an increase in feedwater flow	/
14.3.3	Excessive increase in secondary steam flow	THEMIS (App. 14B), SMART
14.3.4	Turbine trip	/
14.3.5	Loss of Condenser Vacuum	NLOOP (App. 14B)
14.3.6	Short term loss of offsite power (≤ 2 hours)	PANBOX/COBRA, NLOOP (App. 14B)
14.3.7	Loss of normal feedwater flow (loss of all the ARE [MFWS] pumps and of the start-up and shutdown pumps)	/
14.3.8	Partial loss of core coolant flow (Loss of one RCP [RCS] pump)	PANBOX/COBRA, NLOOP (App. 14B)
14.3.9	Uncontrolled RCCA bank withdrawal at power	THEMIS, FLICA (App. 14B)
14.3.10	Uncontrolled RCCA bank withdrawal from hot zero power conditions	SMART, FLICA
14.3.11	RCCA misalignment up to rod drop without control system action	SMART, FLICA
14.3.12	Start up of an inactive reactor coolant pump at an incorrect temperature	/
14.3.13	RCV [CVCS] malfunction resulting in a decrease in boron concentration in the reactor coolant	SMART
14.3.14	RCV [CVCS] malfunction causing increase or decrease of reactor coolant inventory	/
14.3.15	Primary side pressure transients (spurious operation of pressuriser spray/heaters)	/
14.3.16	Uncontrolled RCP [RCS] level drop (states C,D)	/
14.3.17	Loss of one cooling train of RIS/RRA [SIS/RHRS] in residual heat removal mode	/
14.3.18	Loss of one train of the fuel pool cooling system (PTR [FCPS]) or of a supporting system (state A)	/
14.3.19	Spurious reactor trip (state A)	/
14.4.1	Small steam or feedwater system pipe failure (DN < 50) including break of connecting lines (< DN50) to SG	/

Section[(1)]	Event	Computer Codes
14.4.2	Long term loss of offsite power (> 2 hours)	PANBOX/COBRA, NLOOP (App. 14B)
14.4.3	Inadvertent opening of a pressuriser safety valve	/
14.4.4	Inadvertent opening of an SG relief train or of a safety valve (state A)	/
14.4.5	Small Break LOCA (≤ DN50), including a break occurring on the RBS [EBS] injection line (states A and B)	CATHARE, CATHARE/CONPATE (App. 14B)
14.4.6	Steam Generator Tube Rupture (1 tube)	CATHARE, S-RELAP5 (App. 14B)
14.4.7	Inadvertent Closure of one/all main steam isolation valves	THEMIS, FLICA (App. 14B)
14.4.8	Inadvertent loading of a fuel assembly in improper position	/
14.4.9	Forced Decrease of Reactor Coolant Flow (4 pumps)	SMART, FLICA, THEMIS (App. 14B)
14.4.10	Leak in the gaseous or liquid waste processing system	/
14.4.11	Loss of primary coolant outside the containment	/
14.4.12	Uncontrolled RCCA bank withdrawal (states B, C and D)	SMART, FLICA
14.4.13	Uncontrolled single control rod withdrawal	SMART, FLICA
14.4.14	Long term loss of offsite power (> 2 hours), fuel pool cooling aspect (state A)	/
14.4.15	Loss of one train of the fuel pool cooling system (PTR [FCPS]) or of a supporting system (State F)	/
14.4.16	Isolable pipe failure on a system connected to the fuel pool (states A to F)	/
14.5.1	Long term loss of offsite power in state C (> 2 hours)	/
14.5.2	Main steam line break	MANTA/SMART/FLICA THEMIS, PANBOX/COBRA (App.14B)
14.5.3	Feedwater line break	CATHARE CATHARE (App.14B)
14.5.4	Inadvertent opening of an SG relief train or safety valve (state B)	/
14.5.5	Spectrum of RCCA ejection accidents	SMART, FLICA, COMBAT
14.5.6	Intermediate and Large Break LOCA (up to the surge line break, states A and B)	CATHARE CATHARE/CONPATE (App. 14B)
14.5.7	Small break LOCA (< DN50) including a break in the RBS [EBS] injection line (states C and D)	CATHARE
14.5.8	Reactor Coolant Pump seizure (locked rotor)	PANBOX/COBRA (App. 14 B)

Section[1]	Event	Computer Codes
14.5.9	Reactor Coolant Pump shaft break	PANBOX/COBRA (App. 14B)
14.5.10	Steam Generator Tube Rupture (2 tubes in 1 SG)	CATHARE
14.5.11	Fuel handling accident	/
14.5.12	Boron Dilution due to a non-isolable rupture of a heat exchanger tube	SMART
14.5.13	Rupture of systems containing radioactivity in the Nuclear Auxiliary Building	/
14.5.14	Isolable safety injection system break (≤ DN 250), in residual heat removal mode (states C, D)	CATHARE
14.5.15	Non-isolable small break (≤ DN 50) or isolable safety injection system break (≤ DN 250) in residual heat removal mode - fuel pool drainage aspect (State E)	/
14.6	Radiological Consequences	ORIGEN-S, ACARE, ALICE 2, PRODOS-B

Note (1): Section 14.3.1 refers to section 1 of Sub-chapter 14.3, etc
Section 14.6 refers to Sub-chapter 14.6

APPENDIX 14A – TABLE 2

Computer Codes Used in Sub-chapter 16.1
RRC-A Studies

Section(1)	Event	Computer Codes
16.1.3.1	ATWS by rods failure	MANTA/SMART/FLICA
16.1.3.2	ATWS by RPR [PS] failure	MANTA/SMART/FLICA
16.1.3.3	Station blackout (at power)	CATHARE
16.1.3.4	Total loss of feedwater (at power)	CATHARE
16.1.3.5	TLOCC inducing a break on RCP [RCS] pumps seals (at power)	/
16.1.3.6	LOCA (break size up to 20 cm^2) with loss of partial cooldown signal (at power)	CATHARE
16.1.3.7	LOCA (break size up to 20 cm^2) without MHSI (at power)	CATHARE
16.1.3.8	LOCA (break size up to 20 cm^2) without LHSI (at power)	S-RELAP5/COCO (App. 14B)
16.1.3.9	Uncontrolled level drop without SI-signal from PS (in shutdown state)	/
16.1.3.10	Non-RCV [CVCS] homogeneous dilution with failure of isolation by operator (in hot shutdown state)	SMART
16.1.3.11	TLOCC (in shutdown state)	/
16.1.3.12	Total loss of the cooling chain or the ultimate heat sink (state A to C), for 100h	/
16.1.3.13	Loss of the two main trains of the fuel pool cooling system during shutdown for refuelling (state F)/Station blackout	/

Note (1): Section 16.1.3.1 refers to section 3.1 of Sub-chapter 16.1, etc

APPENDIX 14A – TABLE 3

Computer Codes Used in section 1.5 of Sub-chapter 3.4
Overpressure Protection Analysis

Section	Event	Computer Codes
3.4.1.5.1	Overpressure protection at power	MANTA, MANTA/SMART/FLICA
3.4.1.5.2	Overpressure protection at cold shutdown	MANTA

Note (1): Section 3.4.1.5.1 refers to section 1.5.1 of Sub-chapter 3.4, etc

1. S-RELAP5

S-RELAP5 [Ref-1] is a PWR system transient analysis code that can be used for simulation of a wide variety of PWR system transients of interest in LWR safety. The primary system, secondary system, system controls, and core neutronics can be simulated. The code models have been designed to permit simulation of postulated accidents ranging from LB(LOCA) to non-LOCA transients. Transient conditions can be modelled up to the start of metal-water reaction (beginning of fuel damage).

The code was developed by Siemens Power Corporation (SPC) in Richland, USA, to perform realistic analysis of LOCA for PWRs (See Appendix 14A – Figure 1). It is a RELAP5-based thermal-hydraulic system code, which incorporates features of RELAP5/MOD2 [Ref-1] and RELAP5/MOD3 [Ref-1], as well as SPC and KWU improvements [Ref-2]. In general, the improvements and modifications included are those required to provide congruency with the unmodified literature correlations and those required to obtain adequate simulation of key LB(LOCA) experiments. The code structure for S-RELAP5 was modified to be essentially the same as that for RELAP5/MOD3, with the same code portability features. The coding for reactor kinetics, control system and trip systems was replaced by those of RELAP5/MOD3.

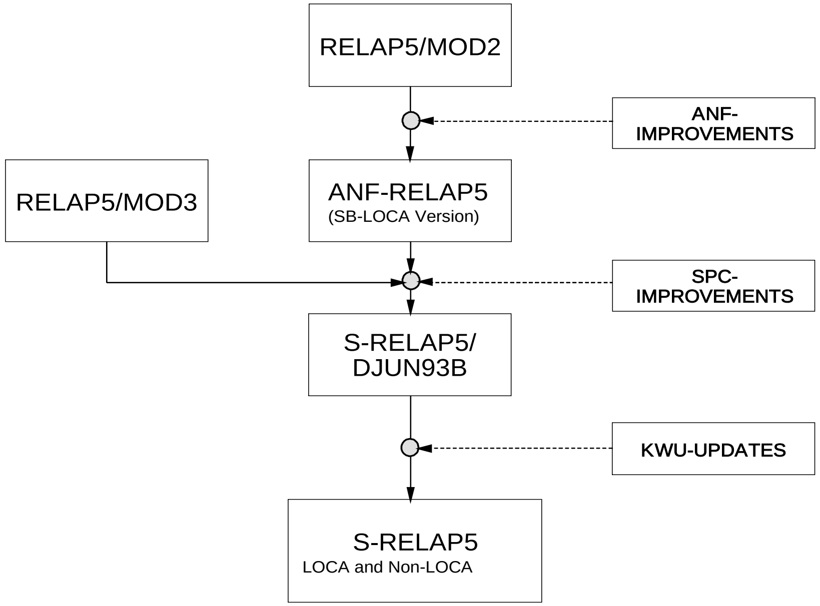
S-RELAP5 is used in Nuclear Regulatory Research in the resolution of current safety issues, in the evaluation of plant operator guidelines, and as a tool for auditing safety analyses submitted by licensees. In addition, the code is widely used by the nuclear industry worldwide for design and safety analyses.

The most important features of S-RELAP5 are:

- History.
- Model description and numerical scheme.
- Major modifications/improvements compared to RELAP5/MOD2.
- Nodalisation of EPR {CCI Removed} b
- Coupling of S-RELAP5 with I&C routines as part of the NLOOP code (see section 14.3 within this appendix as well as Appendix 14A – Figure 3).
- Coupling of S-RELAP5 with COCO code for calculation of pressure and temperature transients in the containment atmosphere and IRWST (see section 14.2 within this appendix as well as Appendix 14A – Figure 4).
- Verification and qualification of S-RELAP5 (see Appendix 14A – Table 4).

APPENDIX 14A – FIGURE 1 [REF-1]

History of S-RELAP5



1.1. S-RELAP5: MODEL DESCRIPTION AND NUMERICAL SCHEME

S-RELAP5 is a PWR system transient analysis code used for simulation of a wide variety of PWR system transients of interest in LWR safety. The primary system, secondary system, system controls, and core neutronics can be simulated. The code models have been designed to enable simulation of postulated accidents ranging from LB(LOCA) to non-LOCA transients. Transient conditions can be modelled up to the start of metal-water reaction (beginning of fuel damage).

1.2. HYDRODYNAMICS

The S-RELAP5 hydrodynamic model is a two-dimensional, transient, two-fluid model for flow of a two-phase steam-water mixture that can contain a non-condensable component in the steam phase and/or a non-condensable component in the liquid phase.

The S-RELAP5 hydrodynamic model contains several options for invoking simpler hydrodynamic models. These include homogeneous flow, thermal equilibrium, and frictionless flow models. These options can be used independently or in combination. The homogeneous and equilibrium models were included primarily to be able to compare code results with calculations from the older homogeneous equilibrium model based codes.

The two-fluid equations that are used as the basis for the S-RELAP5 consist of two-phase continuity equations, two-phase momentum equations, and two-phase energy equations. The equations are recorded in differential stream-tube form with time and one space dimension as independent variables and in terms of time and volume average dependent variables. Events that depend upon transverse gradients, such as friction and heat transfer, are formulated in terms of the bulk potentials using empirical transfer coefficient formulations.

The two-fluid model has seven dependent so-called primary variables (eight if a non-condensable component is present). These are pressure, internal energies of steam and liquid, void fraction, boron density, steam velocity and liquid velocity. The non-condensable quality, defined as the ratio of the non-condensable gas mass to the total gaseous phase mass, is the eighth variable. The eight so-called secondary dependent variables (determined on pressure and internal energies) used in the equations are phase densities, steam generation rate per unit volume, inter-phase heat transfer rates per unit volume, phase temperatures, and saturation temperature.

The difference equations are based on the concept of a control volume (or mesh cell) in which mass and energy are conserved by equating accumulation to rate of influx through the cell boundaries. This model results in the definition of mass and energy volume average properties and requires knowledge of velocities at the volume boundaries. The velocities at boundaries are conveniently defined through use of momentum control volumes (cells) centred on the mass and energy cell boundaries. This approach results in a numerical scheme having a staggered spatial mesh. The scalar properties (pressure, energies, and void fraction) of the flow are defined at cell centres, and vector quantities (velocities) are defined on the cell boundaries. The term, cell, means an increment in the spatial variable, x, corresponding to the mass and energy control volume.

The system model is solved numerically using a semi-implicit finite difference technique. The user can select an option for solving the system model using a nearly-implicit finite difference technique, which allows violation of the material Courant limit. This option is suitable for steady state calculations and for slowly-varying, quasi-steady transient calculations.

The semi-implicit numerical solution scheme uses a direct sparse matrix solution technique for time step advancement. The method has a material Courant time step stability limit. However, this limit is implemented in such a way that single node Courant violations are permitted without adverse stability effects. Thus, single small nodes embedded in a series of larger nodes will not adversely affect the time step and computing cost. The nearly-implicit numerical solution scheme also uses a direct sparse matrix solution technique for time step advancement. This scheme has a grid time that is 25% to 60% greater than the semi-implicit scheme but allows violation of the material Courant limit for all nodes.

1.3. MODEL COMPONENTS

The code includes models for defining flow regimes. These include flow regime related models for inter-phase drag, wall friction, heat transfer, inter-phase heat and mass transfer and re-flood heat transfer. The models include flow regime effects for which simplified mapping techniques have been developed to control the use of component correlations. Three flow regime maps are utilised - vertical and horizontal maps for flow in pipes, and a high mixing map for flow in pumps.

1.4. HEAT TRANSFER

A boiling curve is used in S-RELAP5 to govern the selection of heat transfer correlations. In particular, the heat transfer regimes modelled are classified as pre-critical heat flux (CHF), CHF and post-CHF regimes. Condensation heat transfer is modelled and the effects of non-condensable gases are included.

The pre-CHF regime consists of models for single-phase liquid convection, sub-cooled nucleate boiling and saturated nucleate boiling. The post-CHF regime consists of models for transition film boiling, film boiling and single-phase steam convection. The CHF is calculated with a separate correlation. Heat structures provided in the code permit calculation of the heat transferred across solid boundaries of hydrodynamic volumes. Modelling capabilities of heat structures are general and include fuel pins or plates with nuclear or electrical heating, heat transfer across steam generator tubes, and heat transfer from pipe and vessel walls. Heat structures are assumed to be represented by one-dimensional heat conduction in rectangular, cylindrical, or spherical geometry.

Surface multipliers are used to convert the unit surface of the one-dimensional calculation to the actual surface of the heat structure. Temperature dependent thermal conductivities and volumetric heat capacities are provided in tabular or functional form either from built in or user supplied data. Finite differences are used to advance the heat condition solutions.

1.5. SPECIAL FEATURES

S-RELAP5 has a number of special features that have proven to be very useful in thermal-hydraulic analysis of PWRs. A description of those features and their applications would be too lengthy to be presented in this report. The most important special features of S-RELAP5 are listed below:

- Abrupt area change for single-phase and two-phase flows.
- Accumulator component, including specific hydrodynamic and heat transfer models.
- Centrifugal pump performance model with two-phase degradation effects.

- Choked flow, including special treatments for:
  - Horizontal stratified choked flow.
  - Non-homogeneous, equilibrium two-phase flow.
  - Sub-cooled choking.
- Control system.
- Cross-flow junction.
- Decay heat, including actinides contribution.
- Fine mesh-renodalisation scheme for heat conduction.
- Jet mixer for single-phase and two-phase flows.
- Liquid entrainment in horizontal stratification.
- Motor valve model.
- Reactor kinetics (point model) with reactivity feedback from thermal-hydraulic variables.
- Relief valve model.
- Servo valve model.
- Steam separator.
- Steady state initialisation capability.
- Trip system.
- Turbine component model.
- Vertical stratification.

1.6. INPUT DESCRIPTION

S-RELAP5 has a very general input description capability that allows modelling of any thermal-hydraulic facility. The system can be as simple as a pipe, a single volume of fluid, or as complicated as a multi-loop PWR with many external connections. The input deck for each problem is organised in a number of card blocks. Each block has a unique numbering sequence that is used for automatic sorting by the computer, regardless of the position of each card in the input deck. This is a user facility that makes it easy to change the inputs for different runs. The re-stating and renodalisation capabilities of S-RELAP5 provide additional user convenience for modifications in the system modelling at different stages of the transient.

1.7. MAJOR MODIFICATIONS AND IMPROVEMENTS IN S-RELAP5

The following list summarises the major modifications and improvements in S-RELAP5:

1) Multi-Dimensional Capability

Full 2-D treatment was added to the hydrodynamic field equations. The 2-D capability can accommodate the Cartesian and the cylindrical coordinate systems and can be applied anywhere in the reactor system. Some improvements were also made to the RELAP5/MOD2 cross flow modelling. If necessary, 3-D calculations can be approximated by 2-D plus one direction of cross flow.

2) Energy Equations

The energy equations of RELAP5/MOD2 and MOD3 have a strong tendency to produce energy error when a sizeable pressure gradient exists between two adjacent cells (or control volumes). This deficiency is a direct consequence of neglecting some energy terms which are difficult to be handled numerically. Therefore, the energy equations were modified to conserve the energies transported into and out of a cell (control volume). Omission of some energy terms is still needed to make numerical computation feasible. For LOCA calculations, there are no significant differences in the key parameters (such as clad surface temperature, mass flow rate through a break, void fraction, etc.) between the two formulations of the energy equations. For analyses involving a containment volume, the new approach is more appropriate.

3) Numerical Solution of Hydrodynamic Field Equations

The reduction of the hydrodynamic finite-difference equations to a pressure equation is obtained analytically by algebraic manipulations in S-RELAP5, but is obtained numerically by using a Gaussian elimination system solver in RELAP5/MOD2 and MOD3.

4) State of Steam/Non-condensable Mixture

Calculation of state parameters for the steam/non-condensable mixture at very low steam quality (i.e. the ratio of steam mass to total gas phase mass) was modified to allow the presence of a pure non-condensable gas below the ice point (0°C). The ideal gas approximation is used for both steam and non-condensable gas at very low steam quality. This modification is required to correctly simulate the accumulator depressurisation and to prevent code failures during the period of accumulator RIS [SIS] water injection.

5) Hydrodynamic Component Models

Significant modifications and enhancements were made to the RELAP5/MOD2 inter-phase friction and inter-phase mass transfer models. The component models are flow regime dependent and constructed from the correlations for the basic flow patterns. Some flow regime transition criteria of RELAP5/MOD2 were modified to make them consistent with published data. When possible and applicable, literature correlations are used as published. A component formulation for a particular flow regime may be composed of two different correlations. Transition flow regimes were introduced for smoothing the component models. Partition functions for combining different correlations and for transitions between two flow regimes were developed based on physical reasoning and code-data comparisons. Most of the existing RELAP5/MOD2 partition functions were not modified or only slightly modified. The vertical stratification model was further improved. The RELAP5/MOD2 approximation to the Colebrook equation of wall friction factor is known to be inaccurate and was, therefore, replaced by an accurate explicit approximate formulation of Jain [Ref-1].

6) Heat Transfer Model

The use of a different set of heat transfer correlations for the re-flood model in RELAP5/MOD2 was removed. Some minor modifications were made to the selection logic for heat transfer models (or regimes), single phase liquid natural convection and condensation heat transfer. The Lahey correlations for steam generation in the sub-cooled nucleate boiling region were used [Ref-1]. No changes were made to the RELAP5/MOD2 Critical Heat Flux correlations.

7) Choked Flow

The state calculation at the choked plane was modified by using an iterative scheme to determine the state rather than using the previous time step information. Some minor modifications were also made to the under-relaxation scheme to smooth the transition between sub-cooled single-phase critical flow and two-phase critical flow.

8) Counter current Flow Limitation (CCFL)

A Bankoff type CCFL correlation was implemented in S-RELAP5, which can be reduced to either a Wallis type or a Kutateladze type CCFL correlation [Ref-1]. RELAP5/MOD3 also uses the Bankoff correlation.

9) Component Models

The EPRI pump performance degradation data [Ref-1] was included in the S-RELAP5 pump model. The calculation of pump head in the fluid field equations was modified to be more implicit. The accumulator model was removed because of its well-known problems. With S-RELAP5 the accumulator has to be modelled as a pipe with nitrogen or air as a non-condensable gas.

10) Fuel Model

The plastic strain and metal-water reaction models from RELAP5/MOD3 were included in S-RELAP5 with minor modifications.

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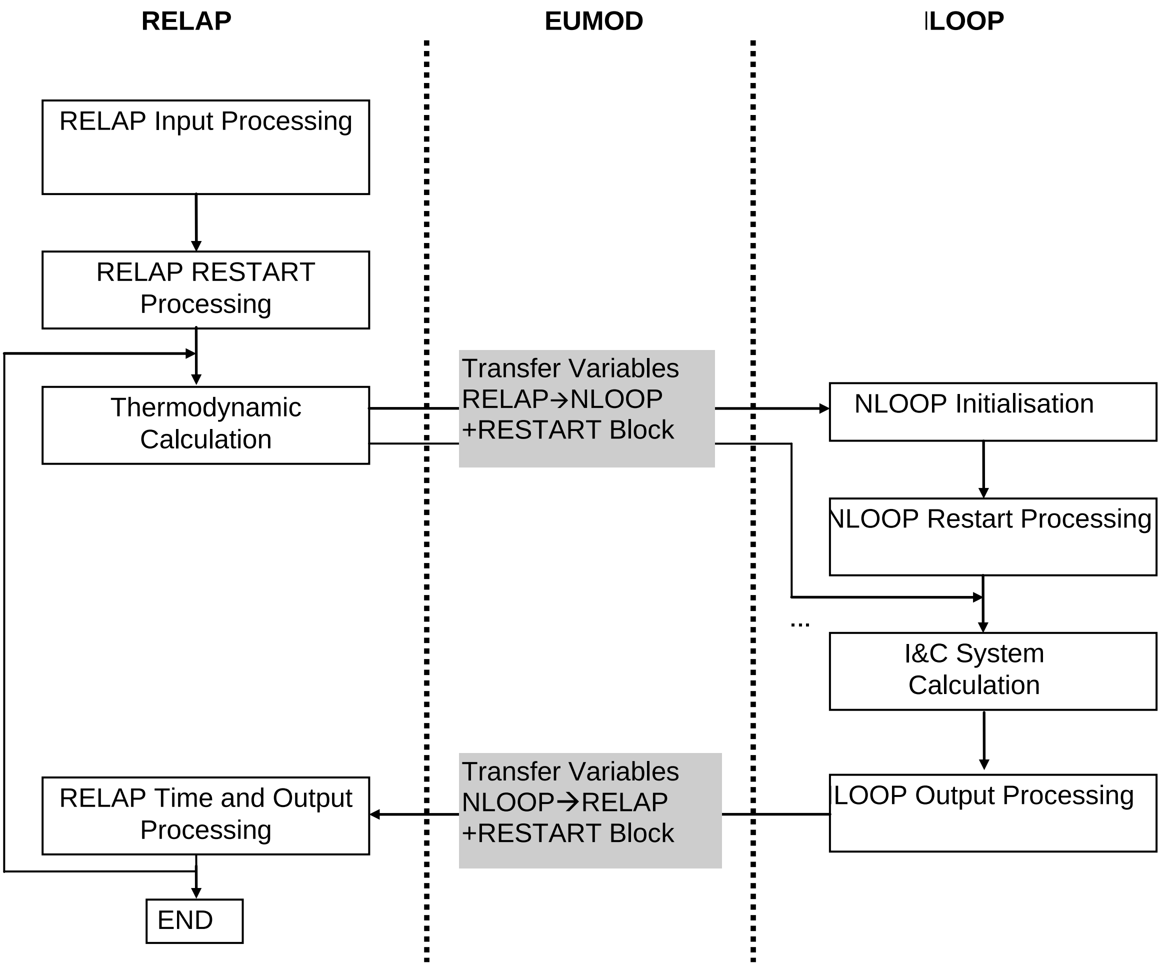
APPENDIX 14A – FIGURE 3

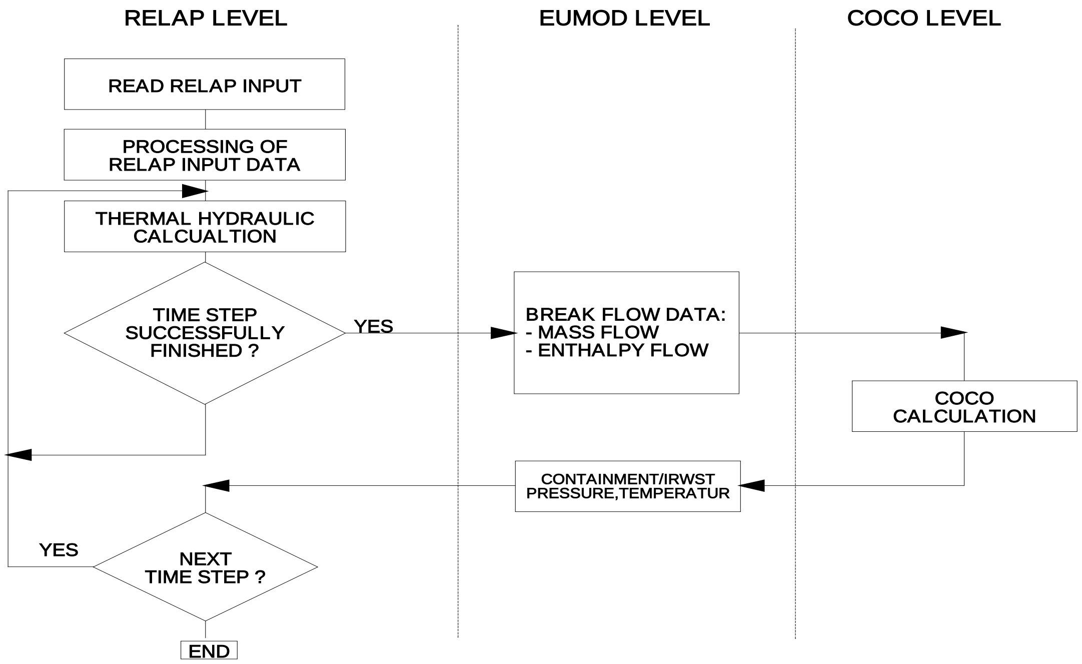
Variables transfer



Transfer Variables RELAP→ NLOOP	All possible RELAP output, i.e. pressures, temperatures, flow rates, valve positions, reactivity
Transfer Variables NLOOP→ RELAP	Valve positions → i.e. MSSV, VDA [MSRT], VIV [MSIV], Pressuriser safety and spray valves Flow rates or fluid velocities→ i.e. RCV [CVCS], MHSI, LHSI, RRA [RHRS], HLBS, FW Heat structure table → i.e. Pressuriser Heaters Reactivity Table → i.e. RCCA control Volume conditions (TDV) → i.e. RCV [CVCS], MHSI, LHSI, RRA [RHRS]

APPENDIX 14A – FIGURE 4

Description of analysis methodologies used by SIE
Data Transfer between S-RELAP5 and COCO (coupled calculation)



APPENDIX 14A – TABLE 4 [REF-1] [REF-2]

S-RELAP5 Code Verification and Qualification

	INTEGRAL EFFECT TEST FACILITY							SEPARATE EFFECT TEST FACILITY									
TEST FACILITY / PHENOMENON	LOFT L2-5 L2-6	CCTF RUN 54	SCTF S35H1	PKL II TEST 3.2	PKL III B3.2	BETHSY #6.2	BETHSY #9.1	FLECHT SEASET 31504	FLECHT SEASET 33056	ORNL THTF 3.09.10	CE/EPRI PUMP TESTS	GE LEVEL SWELL 1004-3	MARVIKEN TESTS 22, 24	BENNETT TESTS 5358, 5379	UPTF TEST 6, 7	UPTF TEST 11	UPTF TRAM A5
BREAK FLOW CRITICAL FLOW	●	●										●	●			LB-LOCA	
DNB, POST-CHF, REFLOOD HEAT	●	●	●	●				●	●	●				●			
REWET, REFLOOD, QUENCHING	●	●	●	●				●									
FLOW REVERSAL, STAGNATION IN CORE	●	●		●													
1/2-PHASE CONVECTION, CCF, CRITICAL FLOW	●	●									●				●	●	
VOID GENERATION, CONDENSATION	●	●	●	●									●		●		
VOID LIQUID LEVEL	●		●	●				●		●		●			●		
STORED ENERGY, GAP CONDUCTANCE (FUEL ROD)	●																
BREAK FLOW	●					●	●						●			SB LOCA	
NATURAL CIRCULATION	●				●												
LOOP SEAL CLEARANCE					●	●											●
ECC FLOW															●		●
SG HEAT TRANSFER	●				●	●		●									
PHASE SEPARATION						●							● ●				●

2. COCO

The computer code COCO [Ref-1] was developed to calculate the temperature and pressure transient of a PWR dry containment under LOCA conditions. The code is used for containment design (peak pressure) as well as for calculation of containment backpressure under LOCA conditions. For the calculation of containment pressure the program COCO was replaced by CONPATE4 code.

COCO thermal-hydraulic calculation is based on a lumped parameter model with balance equations for mass, volume and energy. The containment is divided into two subsystems: the atmosphere inside the containment building and the containment sump. Thermal non-equilibrium between the subsystems as well as mass and heat exchange are modelled. Heat can also be exchanged with adjacent walls or containment internals.

For both sub-systems a balance of incoming and outgoing mass and energy fluxes is performed; the variation over time of containment pressure and temperatures, in the sump water and in the various walls and internals (e.g. steel shell, concrete walls, secondary shield, etc.) is then calculated.

The containment sump is modelled as a pool which collects water lost from the primary system, water produced by condensation processes and RIS [SIS] water injected by safety systems. During the residual heat removal (RHR) phase, sump water is drawn off by RIS/RRA [SIS/RHRS] pumps and is injected via the residual heat exchanger either into the primary system or through the break back into the sump. Steam from the containment atmosphere may condense into the sump. Conversely, the sump water may boil and release steam into the containment atmosphere.

The atmosphere inside the containment building contains a mixture of steam, air and hydrogen with finely distributed water droplets. Air supply and exhaust are accounted for. The cold structures of the building act as heat sinks and condensation surfaces. Heat can also be removed from the atmosphere by air coolers. The concrete walls and steel internals within the containment building retain heat removed from the containment atmosphere.

To simplify the model, the following assumptions are made: Containment atmosphere and sump region are treated as only one node each. Supplied air and steam are immediately mixed with the entire containment atmosphere. Heat transfer to the walls is greatly influenced by the flow velocity of the fluid. Since velocity distribution is not calculated, heat transfer coefficients (HTCs) should be determined with an appropriate correlation. The correlation of Tagami and Uchida [Ref-1] is used in COCO, but HTCs can also be input in the form of a table, as a function of time or temperature.

The outflow through the break is divided according to its energy, into water falling into the sump and steam mixing with the containment atmosphere. The distribution is based on the assumption that water and steam are saturated at the overall containment pressure and at the partial pressure of the water steam in the containment atmosphere, respectively.

Additional to the outflow rate through the break, other heat inputs to the containment may be included, allocated separately to the sump and to the containment atmosphere. This option is useful if the release of heat by hot parts of the system is to be simulated. Another example of heat-up of the containment atmosphere is through the combustion of hydrogen, which may be released in the reactor pressure vessel by an exothermic Zr-H_2O reaction. The release of hydrogen or another non-condensable gas into the containment atmosphere can be input as a function of time. The gas is taken into account in the mass and energy balances. The combustion of hydrogen is not simulated.

The COCO model also includes a sump cooler system (e.g. LHSI/RHR cooler) and a spray system for cooling the containment atmosphere.

3. NLOOP

NLOOP [Ref-1] is a computer program developed by SIE/KWU for the simulation of the overall plant behaviour in the design, licensing and operational survey and safety analysis of a PWR system. It simulates the plant response to a wide range of non-LOCA events including special events such as steam generator tube rupture and transients without scram (ATWS). Multiple asymmetric transients including flow reversal are considered as up to four loops can be simulated.

The code contains models for major thermal hydraulic systems of the primary and secondary sides (RCP [RCS], VVP [MSSS], ARE [MFWS]), for safety and auxiliary systems, and for all essential control and protection systems

The fluid in the primary coolant system is treated as a homogeneous flow. Non-equilibrium conditions with respect to temperature are allowed in the pressuriser, the steam generators, and the reactor pressure vessel head only. Consequently the applicability of NLOOP is limited to scenarios with a very low void fraction in the RCP [RCS] i.e. typical LOCAs cannot be analysed by NLOOP.

Simple adaptation to other plants is obtained by replacing input data and/or modules.

On the following pages the essential features of the code concerning the scope of simulation, nodalisation are summarised (see Appendix 14A – Table 5 and Appendix 14A – Figures 6 and 7).

The status of validation/verification is also addressed (see Appendix 14A – Table 6).

3.1. MODEL DESCRIPTION

3.1.1. Reactor Coolant System (RCP [RCS])

3.1.1.1. Coolant Circuit

The primary coolant system is simulated by a zone model. The division into zones is specified by input parameters {CCI Removed}[b].

The number of plant loops (up to four) and the calculation loops (up to four) are specified in the input data.

Mass and energy balances are applied to each zone. Each zone of the coolant circuit is treated as homogeneous at constant volume. To avoid small integration time steps, the momentum equation is used integrally on each coolant loop.

The mixing of the coolant flow inside the reactor pressure vessel is implemented in the lower and upper plenum by mixing coefficient matrix (input).

Coolant flow patterns without and with reverse flow in one or more loops are differentiated.

3.1.1.2. Reactor Pressure Vessel Head

The reactor pressure vessel and the pressuriser are connected via the hot leg and the surge line.

The RPV head is normally filled with sub-cooled water. The homogeneous state is separated into steam and water volumes, if steam occurs, e.g. by lowering of coolant pressure. Steam and water phase do not have to be in thermal dynamic equilibrium. The simulation model of the RPV head is similar to that of the pressuriser.

3.1.1.3. Pressuriser

The pressuriser is simulated by a variable steam and water volume not necessarily in thermal dynamic equilibrium. The state of the thermal dynamic system is defined by the variables pressure, specific enthalpy and mass, and thus three equations are required to describe the time response of the system. The energy, volume and mass balances are applied to calculate the state of steam and water mass.

The status of pressuriser "full" (two-phase conditions) as well as "empty" (steam only) is also considered.

Mass flow rate through the surge line is determined by the mass balance over the entire coolant circuit.

The pressuriser spray valves, heaters and safety valves are taken into account.

3.1.1.4. Pressuriser Relief Tank

The model of the relief tank assumes the mixture of water, steam and nitrogen is in thermal dynamic equilibrium.

The mass and internal energy is increased by the mass flow rates of the pressuriser relief or safety valves to the relief tank. An iterative method is used to solve the equations for internal energy and pressure.

3.1.1.5. Reactor Coolant Pumps

The pump pressure head of operating pumps, allowing for the pressure loss of non-operating pump(s) with forward and backward flow, is calculated depending on the pump speed and on the pump flow rate (similarity laws).

The pump characteristic (pump head versus volume flow rate) is an input parameter.

The transient of the pump speed is given by input functions (e.g. coast down behaviour of the pump, load rejection to house load).

3.1.1.6. Reactor Core Model

3.1.1.6.1. Fuel Rod/Coolant Channel

The fuel rods of each core section are presented by an average fuel rod/coolant channel.

The axial power distribution of the fuel rod is selected by input parameters. The fuel rod is partitioned in n axial zones in the input data. Each fuel rod zone consists of two radial sections with equal volumes and the cladding of one radial section. The fraction f in the input data is the fission and decay power released in the fuel and the fraction (1-f) in the moderator.

The heat transfer coefficient between cladding surface and coolant is determined by the Dittus-Boelter correlation [Ref-1] which considers the Reynolds (e.g. coast down of reactor coolant pump(s)) and Prandtl numbers. The specific heat capacities of uranium and cladding are input data parameters.

3.1.1.6.2. Fission power

The calculation of the fission power is normally based on the point kinetics model with six groups of delayed neutrons.

However, for cases where both the overall plant behaviour and a detailed core simulation are required, the point kinetics model is replaced by an iterative coupling of NLOOP with the 3D neutronic/thermal-hydraulic core code PANBOX, see section 4 of this appendix.

3.1.1.6.3. Decay power

The decay power is simulated by either eight groups, whereby the coefficients and the time constants are determined e.g. with reference to DIN25463, [Ref-1] or via direct input function.

3.1.1.6.4. Reactivity balance

The reactivity balance is determined from coolant temperature/density effects, Doppler effects, boron concentration inside the core and rod worth contributions.

3.1.1.7. Steam Generator Model (Secondary Side)

The system boundaries of the steam generator model are the inner area of the SG tubes, the feedwater inlet and the main steam outlet. In the standard SG model, the secondary volume consists of a steam and water volume, i.e. a two-point model, which considers thermal dynamic imbalance. In this way the steam generator secondary side model and the pressuriser model are quite similar.

On the tube side, the tube bundle is divided into n zones in the input data. The heat transfer coefficient between reactor coolant and tube surface is determined by the Dittus-Boelter correlation and hence change of coolant flow rate is taken into account by the Reynolds number. The specific heat capacity of the SG tubes is an input parameter.

On the shell side the heat transfer coefficient between the tube surface and secondary coolant is determined by a nucleate boiling correlation. This heat transfer coefficient is dependent on the heat flux density considered during transient conditions.

Steam generator tube rupture can be assumed at any location of the tube bundle. The break size is specified in the input data. The tube leakage to the secondary side is determined by the IHE (Isentropic Homogeneous Equilibrium) model [Ref-1] and is corrected by a function taking into account friction in the tubes.

Apart from this standard SG model, other SG models simulating the circulation in the SG with downcomer and riser etc can be explicitly modelled.

3.1.1.8. Main Steam Supply System (VVP [MSSS])

A nodal model with compressible flow simulates the main steam line system. The nodes refer to the locations of MS relief/safety valves/isolation valves, turbine and MS header.

Isolation valves are simulated by variable pressure loss coefficients, control valves (main steam relief, turbine, bypass station) are simulated by their characteristics (capacity versus valve position).

The IHE model corrected for friction calculates leaks or breaks in the main steam line system.

3.1.1.9. Main Feedwater System

The main feedwater line system is a nodal network like the main steam line system. The geodetic conditions and pressure losses from the feedwater tank up to steam generators are taken into account. This model is valid for the main feedwater pump system as well as the start-up/shutdown pump system.

High and low load feedwater valves are considered by variable resistances depending on the valve positions.

APPENDIX 14A – TABLE 5

Range of NLOOP Simulation [Ref-1]

Thermal-Hydraulic Components		Electronic Systems incl. Measurement Devices and Actuators			
Primary	**Secondary**	**Control**	**Limitation**	**Reactor Protection**	**Secondary Protection**
☑ Reactor, Core ☑ Coolant Lines ☑ Pressuriser ☑ Relief Tank ☑ Pressuriser Safety Valves	☑ Turbine ☑ Generator ☐ Condenser+Pumps ☑ MS-Lines ☑ FW-Lines	☑ RC Pressure ☑ RC Temperature ☑ Pressuriser Level ☐ RCV [CVCS] Tank Level ☐ RCV [CVCS] Boric Acid/ Demin. Water	REACTOR POWER ☑ PUMA ☑ FEED ☑ Rod Drop ☑ LOOP ☐ PEEK RELEB	☑ Reactor Scram ☑ Turbine Trip ☑ Extra Borating ☑ HP Injection ☑ LP Injection ☑ EFW Injection	☑ Turbine ☑ Condenser ☑ Steam Generator
☑ Pressuriser Relief Valves ☐ Accumulator ☐ Bor. Water Storage Tanks ☐ Borating Tanks ☑ Safety Injection Pumps	☑ FW-Tank ☑ FW-Valves ☐ Preheaters ☐ Demineralised Water Pool ☐ Demineralised Water Storage Tank	☐ Neutron Flux ☑ Turbine ☑ SG Level ☑ MS Pressure ☑ FW Tank Level	RC Pressure/ Inventory/ Temp.Gradient ☑ KMD ☐ KMT-Gr. ☑ KÜMM ☑ DEL MADTEB	☑ Cooldown ☑ Cut-off RCP[RCS] pump ☑ Cut-off FWP ☑ Shut-off FW Valves ☑ Shut-off VIV [MSIV] ☑ Shut-off MSSV ☑ Shut-off VDA [MSRT] ☐ Containment isol. ☑ Emergency Diesels	☐ FW Tank
☑ Borating Pumps ☑RCV [CVCS] Pumps ☑ HP-Reduction Station ☑ RRA [RHRS]	☑ Bypass Station ☑ VIV [MSIV] ☑ MSSV ☑ VDA [MSRT] ☑ MFWP ☑ AAD [SSS] Pumps ☑ EFWP ☑ Passive Secondary RHR System		Rod Dropping ☑ RELEB ☑ LAW ☑ PUMA STEW ☐ Power Distribution	☑ Emergency FW ☑ Diesels	
☑ Steam Generator ☐ Containment					

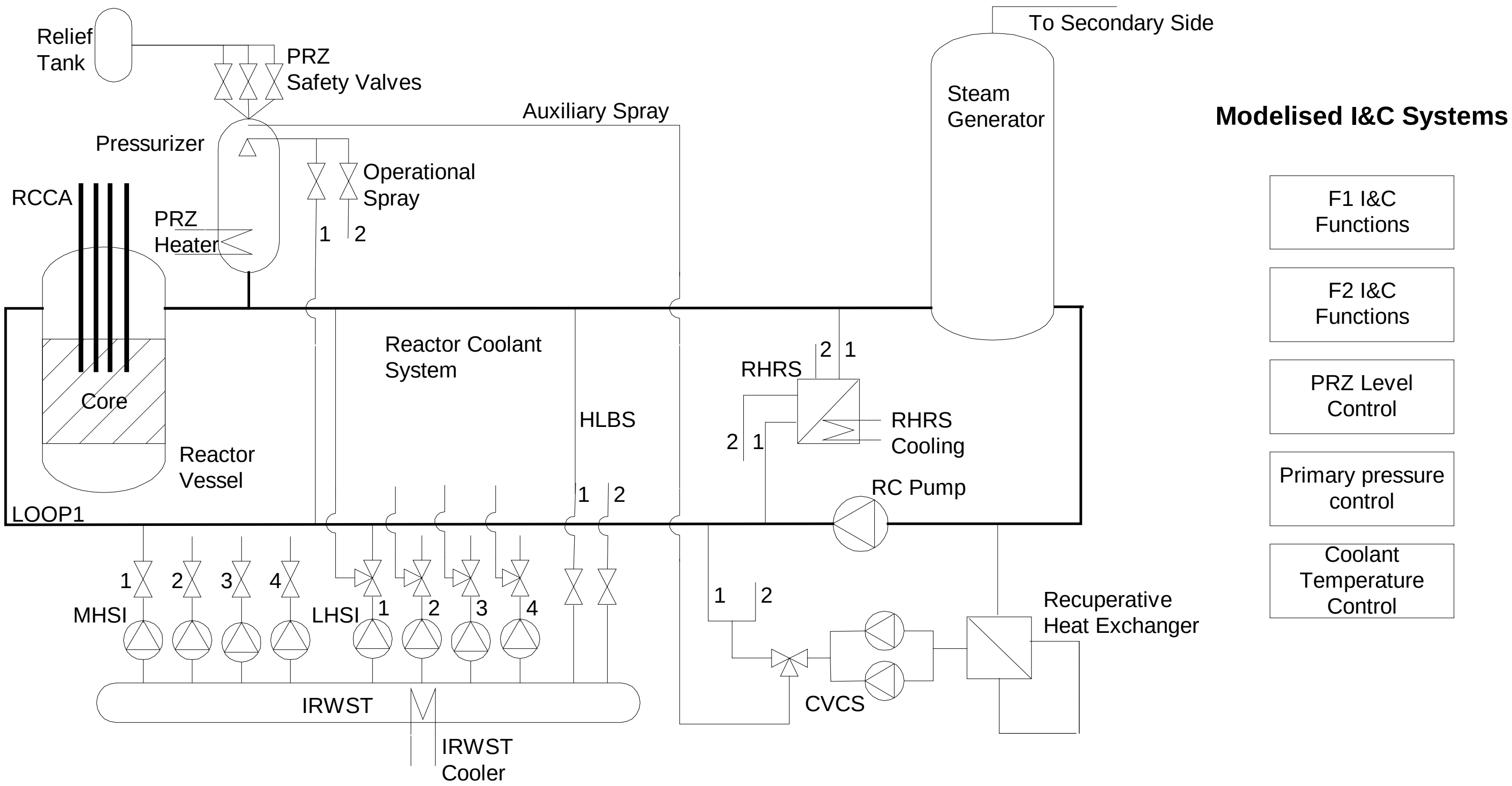
☑ Simulated in NLOOP
☐ Not simulated in NLOOP

{CCI Removed}

b

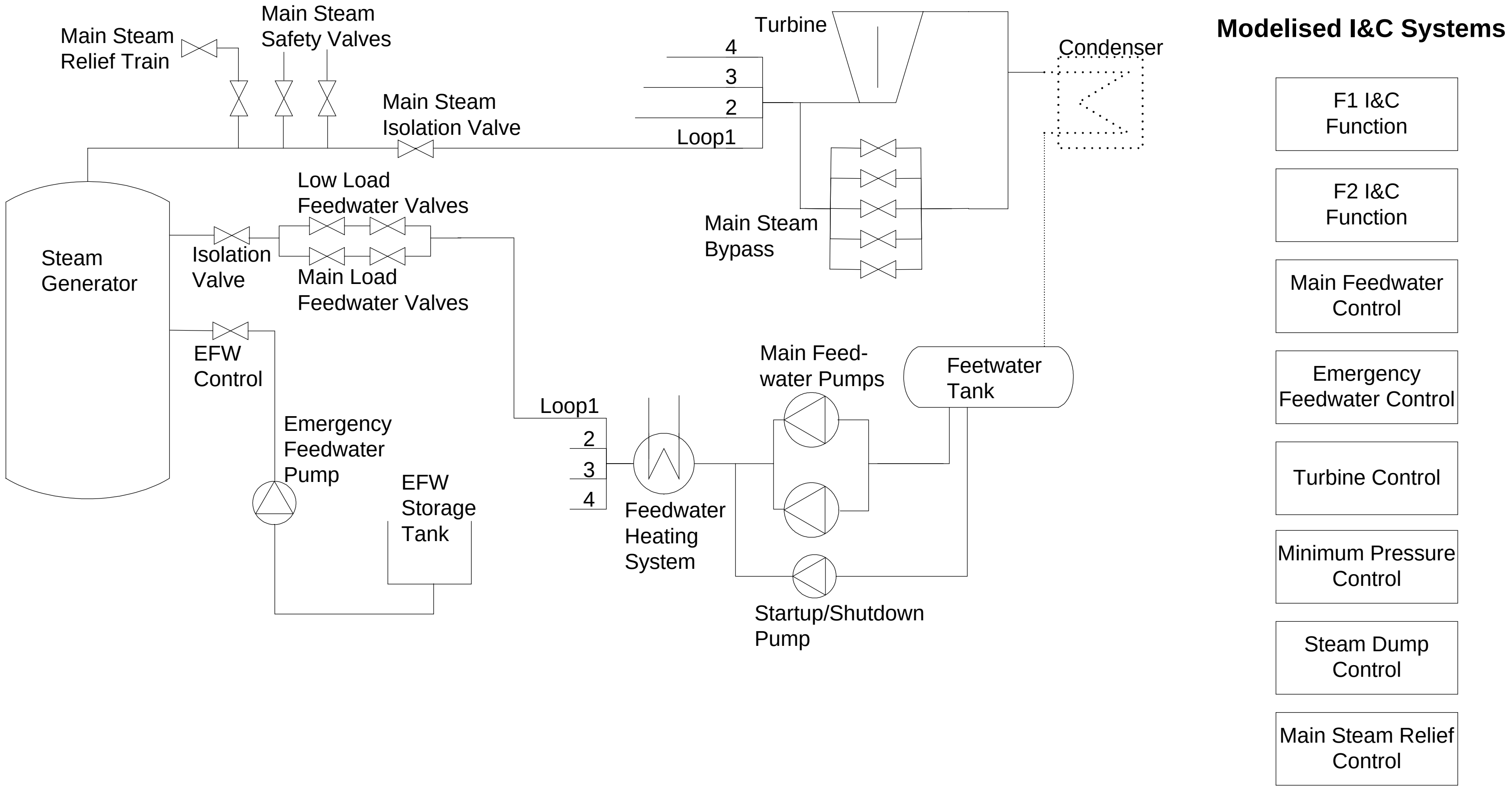
APPENDIX 14A – FIGURE 6

Systems Simulated within EPR NLOOP Package (Primary Side) [Ref-1]



APPENDIX 14A – FIGURE 7

Systems Simulated within EPR NLOOP Package (Secondary Side) [Ref-1]



APPENDIX 14A – TABLE 6

Scope of NLOOP Validation [Ref-1] to [Ref-23]

Recalculation of commissioning tests and unplanned plant events

Incident	Plant	Suppl.	Remark
Reactor trip	KKI2	KWU	Commiss.test
Turbine trip without GCT [MSB]	Biblis B	KWU	Event
Load rejection	KWO	KWU	Commiss.test
Load rejection	GKN1	KWU	Commiss.test
Load rejection	KCB	KWU	Commiss.test
Break of reactor coolant pump shaft	KKG/BAG	KWU	Event
Failure of one reactor coolant pump	KKG/BAG	KWU	Event
Fail closing of one MS isolation valve	GKN1	KWU	Event
Fail closing of one MS isolation valve	CNT1	KWU	Commiss.test
Failure of two feedwater pumps	KKE	KWU	Commiss.test
Sec. Pressure drop transient	KWO	KWU	Event
Load rejection	KWG	KWU	Commiss.test
Fail opening of GCT [MSB]	KKP2	KWU	Event
Reactor trip and turbine trip	KKP2	KWU	Commiss.test Comparison to RELAP
Break of reactor coolant pump shaft	KKG	KWU	Event
Pressuriser relief valve tests	GKN1	KWU	Test
Loss of external load	Ringhals 2	WESTING.	Event
Loss of external load	Doel3	WESTING.	Commiss.test Comparison to RELAP
Trip of one turbine	Ringhals 2	WESTING.	Event
Load ramp 100-50-100%	Doel3	WESTING.	Commiss.test Comparison to RELAP
Failure of one feedwater pump	BEZNAU2	WESTING.	Event
Emergency power mode	KNU1	WESTING.	Event

Recalculation of tests at PKL test facility

Incident	Test Facility	Remark
Total loss of feedwater supply and isolation of 3 of 4 SGs	PKL3	Test
SGTR (one tube) and isolation of affected SG	PKL3	Test

4. PANBOX/COBRA

4.1. OVERVIEW

PANBOX [Ref-1] is designed to calculate the steady-state and transient reactor core conditions (short term and xenon transients) in three-dimensional geometry. Neutronic and thermal-hydraulic modules are applied separately or coupled, evaluating the respective feedback mechanisms.

The history of PANBOX and COBRA development:

History of PANBOX (Codes IQSBOX, BOXER) 1975 - 1989

- PANBOX2 1989 - 1998
- PANBOX2.0 May 27 1992
- PANBOX2.1 Dec.04 1992
- PANBOX2.2 Sept.30 1993
- PANBOX2.3 Oct.07 1994
- PANBOX2.4 Jan.31 1996
- PANBOX2.5 Oct.23 1997

History of COBRA [Ref-2]

- COBRA (-I) 1967
- COBRA II 1970
- COBRA III 1971
- COBRA IIIC 1973
- COBRA IIIC/MIT(-1) 1980
- COBRA IIIC/MIT -2 1981
- COBRA 3-CP today

The following section gives an overview of the actual PANBOX 2 system. It consists of the data base generator PANDAT which couples PANBOX 2 to the core design systems SAV90/SAV95, and of the main program PANCOS which performs the steady-state and transient analysis of a reactor core.

PANCOS consists of the following individual modules:

- FLUXS steady-state neutronics
- FLUXT transient neutronics
- FLUXA adjoint neutronics
- COBRA3-CP thermal-hydraulics
- PFF power form factor calculation
- XENDYN xenon dynamics
- EVAL evaluation of preceding PANCOS runs

Neutronics data, geometry data and necessary mappings for PANBOX 2 are available on special tapes of the PANBOX 1 system.

4.1.1. Nodalisation

Core: in axial direction 28 nodes

in radial direction:

- o 1 node per fuel assembly (FA)
- o for Hot Channel (HC) analysis the FA's within the HC are subdivided into 3 channels

Fuel rod: in axial direction 28 nodes

in radial direction:

- o 5 nodes in the pellet
- o 1 node in the gap
- o 2 nodes in the cladding

4.1.2. Neutronics

The neutronic modules FLUXS/FLUXT solve the steady-state and time-dependent neutron diffusion equations for an arbitrary number of neutron energy groups. Polynomial and (semi-) analytical nodal expansion methods (NEM) can be applied, coupled with an efficient time integration procedure. Multi-level coarse-mesh rebalancing and asymptotic extrapolation are used to accelerate convergence of the iterative solution procedure. Cartesian coordinates with variable mesh sizes in all directions can be treated for full, half, quarter and octant core geometries. Application of mirror and rotational symmetries is possible.

The adjoint FLUXA module solves the space-dependent adjoint neutron diffusion equations for an arbitrary number of neutron energy groups. It serves for determining neutron importance and effective delayed neutron fractions. As in FLUXS/FLUXT, multi-level coarse-mesh rebalancing and asymptotic extrapolation are used to accelerate the convergence of the adjoint solution process.

4.1.3. Thermal-hydraulics

The thermal-hydraulic code COBRA3-CP is coupled to the neutronics modules FLUXS/ FLUXT in PANBOX 2. It solves the time-dependent conservation equations for mass, momentum and energy for the mixture quantities. Separated slip flow is assumed in each sub-channel, and the void fraction distribution is evaluated as a function of enthalpy, flow rate, heat flux and pressure. The conservation equations allow for lateral mixing between adjacent channels by considering both diversion cross-flow and turbulent cross-flow effects. Boundary conditions for system pressure, mass flow and temperature (enthalpy) of coolant at channel inlet can be specified as functions of time. Fuel pin temperatures are calculated from the radial heat conduction equation and from them, fuel pellet enthalpies are derived. The coupling to the coolant determining heat transfer dynamics is realised by appropriate models. Various correlations for calculation of safety-related parameters (e.g. critical heat fluxes) are available.

4.1.4. Neutronics/Thermal hydraulics Coupling

In the coupled system, the steady-state solution is found by repeating the calculation sequence between neutronics and thermal-hydraulics (with the associated updating of powers and cross sections) until both solutions in two successive iterations meet a specified convergence criterion.

Any transient calculation will start from an established converged steady-state reactor core solution. Time-dependent changes can be specified as neutronic disturbances (control rod movement, variation in boron concentration) and/or as variations in thermal-hydraulic boundary conditions (inlet mass flow, inlet temperature, exit pressure). At each time step, the sequence of neutronics calculation followed by thermal-hydraulics calculation, with the respective updating processes, is applied once. The transient time is then moved forward. Time step widths are automatically controlled by checking the behaviour of relative changes of the neutronic and thermal-hydraulic solutions during the time steps.

4.1.5. Power Form Factors

The highly efficient flux reconstruction method MSS-AS based on analytical solution of the diffusion equation is used to determine local flux and power values inside the fuel assemblies. Heterogeneous power form functions can be modulated to account for the heterogeneous structure within a fuel assembly.

4.1.6. Xenon Dynamics

Transients induced by load follow operations are calculated by the XENDYN module. In such cases, the iodine/xenon and/or the promethium/samarium differential equations are solved iteratively with the steady-state flux solution process which itself consists of a sequence of coupled FLUXS/COBRA3-CP iterations. The flux shape is assumed to be linear within the considered time interval.

The iteration process is performed until the flux shapes and the concentrations of xenon and/or samarium are converged to a prescribed criterion.

4.1.7. Evaluation of Calculations

The EVAL module is intended to evaluate preceding PANCOS calculations. Statistics for 2-D pin power form factors can be performed by evaluating the PIN2D_TAPE. Transient ex-core detector signals stored on EXCORE_TAPE can be evaluated following a reactor trip.

4.1.8. Integrated Safety Analysis

The continuing progress in computer technology, characterised by the ever-increasing calculation speed of various computer architectures, enables the direct coupling of code systems used separately in the past. Consequently different areas of analysis such as reactor physics, core thermal-hydraulics and plant dynamics can be integrated to increase the accuracy of simulation above that obtained from imposing conservative boundary conditions at the interfaces. The coupling of thermal-hydraulic sub-channel analysis with nodal space-time kinetics calculations (HOSCAM, hot sub-channel analysis model) is an important step towards an even more extensive integration of complex code systems.

In standard nodal kinetics applications, reactor core subdivision is based on the actual fuel assembly arrangement. The coupled neutronics/thermal-hydraulics system is treated in a comparatively coarse geometry, and the corresponding node-averaged values of power and coolant conditions are the primary results of global reactor calculations. In safety calculations, however, local values of these quantities are needed. To this purpose, as already mentioned above, the automatic procedure HOSCAM renodalises the core wide channel geometry of the PANBOX 2 thermal-hydraulics module COBRA 3-CP. The renodalisation, based on the results of the pin power reconstruction module of PANBOX 2, refines the channels to include hot sub-channels and appropriate sub-channel windows surrounding them.

4.2. VALIDATION AND BENCHMARKING OF THE PANBOX/COBRA CODE

The PANBOX code is qualified against real events (without COBRA coupling) and by benchmark calculations with the coupled PANBOX/COBRA codes.

The following list summarises the validation efforts for the PANBOX code [Ref-1]:

- Eccentric single rod drop, PWR plant KKU.
- Reactor trip, PWR plant KKU.
- RCP [RCS] pump shaft break, PWR plant KKG
- Xe-Transient after shut -down, PWR KWB.
- International main steam line break (MSLB) benchmark.
- NEACRP for rod ejection transient (benchmark).
- NEA-SNC for uncontrolled rods withdrawal at zero power transient (benchmark).

The validation for the COBRA code is performed similarly as for the PANBOX code.

The qualification of the COBRA code coupled with the PANBOX code is demonstrated by the above listed benchmark calculations. The validation for COBRA as a stand alone code was mainly done on the test facility of "Battele Pacific Northwest Laboratory" and some benchmark calculations. The following list summarises the validation efforts for the COBRA code [Ref-2]:

Validation against test facility "Battele Pacific Northwest Laboratory" with different channel geometry:

- Recalculation of flow fields, cross-flow by variation of steam quality.
- Mass flux and enthalpy distribution at single and two-phase flow.
- Void drift, slip flow and wall viscosity effects.
- Blockage of sub-channel:
    - Recalculation of transients of test facility OECD Halden Project.
    - Benchmark tests against VIPRE, THINC and CARO.

5. ORIGEN-S

ORIGEN-S [Ref-1] is a versatile point-depletion and radioactive-decay computer code used to simulate nuclear fuel cycles and calculate the nuclide compositions and characteristics of materials contained therein.

It represents a revision and update of the original ORIGEN computer code, which was developed at the Oak Ridge National Laboratory (ORNL) and distributed world-wide beginning in the early 1970s. Included in ORIGEN-S are provisions to incorporate data generated by more sophisticated reactor physics codes, a free-format input, and a highly flexible and controllable output. With these features, ORIGEN-S has the capability to simulate a wide variety of fuel cycles.

ORIGEN-S uses a matrix exponential method to solve a large system of coupled, linear, first-order ordinary differential equations with constant coefficients.

6. PRODOS-B

Program PRODOS-B [Ref-1] (probabilistic-dose) is developed for the calculation of doses on the basis of observed atmospheric conditions and to determine their occurrences.

Assessments of the doses are programmed following German regulations for accident-related releases. In this version of the PRODOS code the plume rise due to thermal buoyancy is considered. The reflection of the plume at mixing height is also considered.

The program is able to carry out the calculations for sites where 4-dimensional meteorological hourly observations are available (e.g. wind speed, wind direction, atmospheric stability classes, and precipitation).

The program has the ability to extend the duration of the emission episode from one hour to hundreds of hours. Beginning and end of the time period of interest is input (e.g. one year, several years, or summer periods).

The radiation exposures in the neighbouring sectors are calculated in addition to the main sector transport direction.

The program distinguishes four nuclide types (noble gases, aerosols, elemental and organic iodine) and calculates the radiological exposure via four pathways - gamma radiation from the cloud, inhalation, ingestion and ground radiation.

7. ACARE

The program ACARE (Activity in interrelated Compartments And Release to the Environment) calculates the activity flow of up to 99 nuclides in six coupled compartments. For the calculation of radiation doses the program chain ACARE-PRODOS was replaced by MACCS [Ref-1].

Radioactive decay and production of daughter nuclides are taken into account. Deposition of aerosols and iodine in the single compartments can be considered in four different ways. The last compartment represents the atmosphere.

The results are presented, in a time dependent form for:

- the activity inventory in every compartment,
- the release rates to the atmosphere,
- the total release,

and, summed over all nuclides, the nuclide specific release multiplied with dose factors, for four different exposure pathways.

8. CATHARE

CATHARE [Ref-1] is an advanced, two-fluid, thermal-hydraulic code designed for use in realistic studies of accident thermal-hydraulics in pressurised water reactor (PWRs). The transients of interest are those in which core degradation is limited to fuel cladding deformation and bursting. While this excludes the severe accident domain, it does cover all loss-of-coolant (LOCA) accidents, all degraded operating conditions in steam generators (SG) secondary systems, following ruptures or system malfunctions, and insofar as all PWR systems can be simulated, all of the incident or accident transients in which they are involved as initiators or participants.

All fluid systems are represented by a set of CATHARE modules: axial 1-D module for pipes, tubes or channels where velocity has a preferential direction, and volume modules for other zones (vessel plenums, SG channel heads, etc.) and tee junction modules for connections, accumulators, valves, etc. A topological description of each system is also supplied, together with indication of thermal coupling between the various systems (primary/secondary).

Since most PWR system components are pipes, tubes or channels, the basic CATHARE module consists of a 1-D module with six conservation equations, and uses a numerical implicit scheme. These equations represent conservation of mass, energy and momentum, for separate processing of liquid and steam. This system of equations is closed by a complete set of momentum, mass and energy transfer laws for exchange at liquid/steam interfaces or at walls. CATHARE includes a thermo-mechanical model for fuel rods, a point kinetics model for core neutronics and a model for power generation by fission product decay, and a special two-dimensional fuel cladding heat conduction model for core re-flooding, applied near the quench front. Heat sink, mass, energy and momentum terms can be added for simulation of injection, breaks and pumps (0-D pump model).

In addition, CATHARE features:

- a volume module designed to describe fluid behaviour in large volume regions where it is not channelled in a preferential direction of flow (for example, vessel and SG plenums),
- a tee junction module enabling representation of the joining or dividing of currents at connections between two pipes (with phase separation correlations according to direction of diverted flow),
- a 0-D pump model using homologous curves of head and torque,
- a 1-D pump module describing the evolution of the flow lines along the different parts of the pump with use of the basic 6-equations model,
- a 2-D reactor vessel down comer module,
- specific models for accumulator, valve, boron or radioactive product transfer calculations and
- boundary conditions for transient management.

CATHARE can also account for two non-condensable gases.

CATHARE validation is based on widespread qualification by separate effect or component testing and verification on a large matrix of integral effect tests, equivalent in scope to ICAP or OECD matrixes [Ref-1].

More than 300 separate-effect or component tests performed on nearly thirty different test facilities are analysed to validate the equation closure laws and perform the code qualification addressing the following phenomena:

- critical flows (flashing, interface friction),
- phase separation in volumes or tees, type of flow and heat exchange with walls, core thermal-hydraulics (friction and transfer in depressurisation and re-flooding),
- rod thermo-mechanical behaviour,
- pump behaviour,
- steam/water interaction,

- condensation in tubes or by direct contact with cold water stream, bubble rise and swell level.

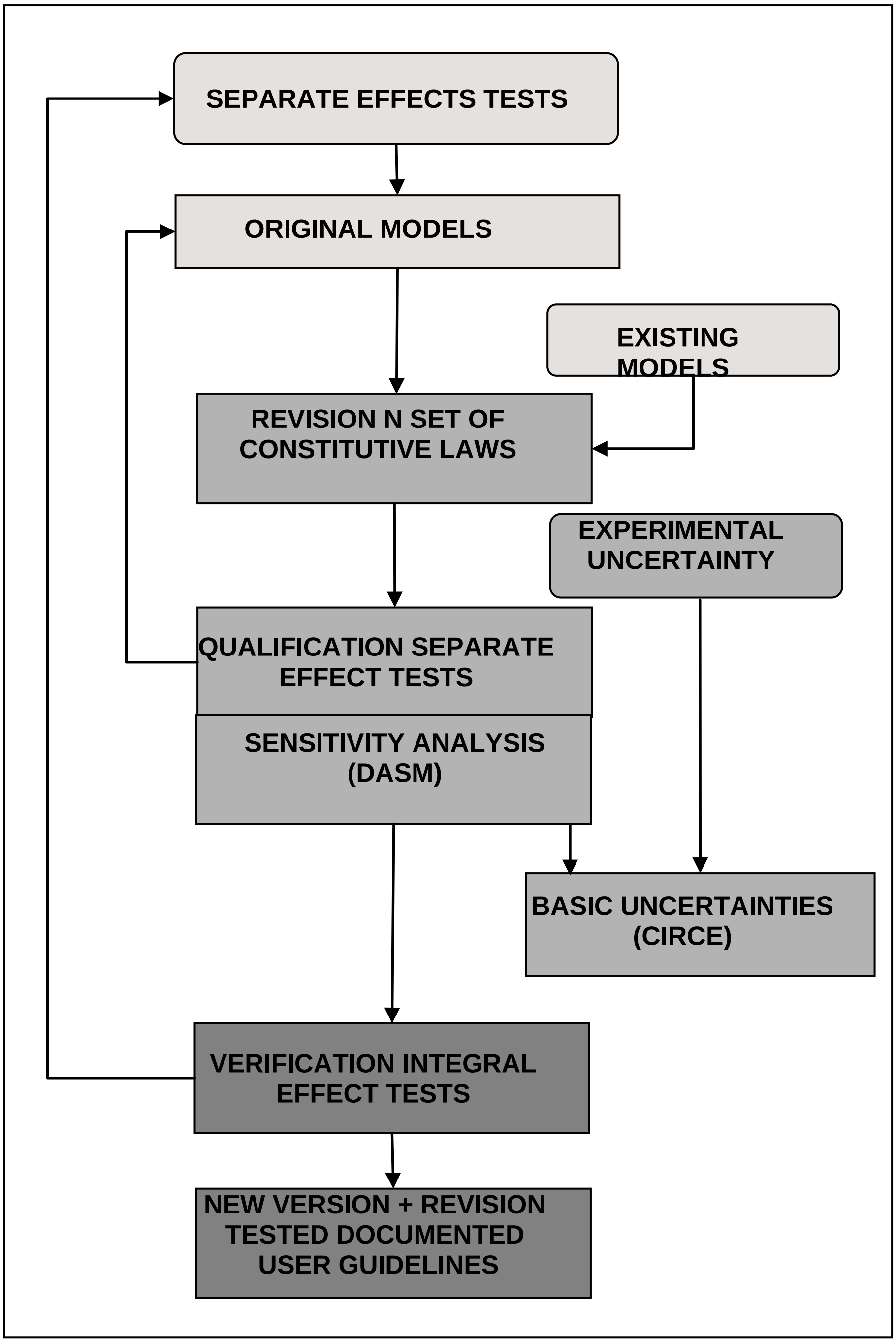
The reconstitution of integral effect tests (BETHSY, LOFT, LSTF, LOBI, PKL, SPES, PACTEL, PMK) [Ref-1] provides experience with large system simulation (schematic layout, meshing, boundary conditions, operator action), and permits verification of code capability for combining phenomena and coupling components to represent a whole reactor system.

The following pages list the most important features and characteristics of CATHARE:

- CATHARE strategy (see Appendix 14A – Figure 8).
- Equation of the 1D module (see Appendix 14A – Table 7).
- CATHARE qualification test matrix (see Appendix 14A – Table 8).
- BETHSY analytical tests and experiments relative to VVERs (see Appendix 14A – Table 9 and 10).
- Assessment and verification on integral tests (see Appendix 14A – Table 11).
- Validation of CATHARE on integral test facilities (see Appendix 14A – Table 12).
- CATHARE Primary side nodalisation of EPR {CCI Removed} [b].
- CATHARE Secondary side nodalisation of EPR {CCI Removed} [b].

APPENDIX 14A – FIGURE 8

CATHARE Strategy



APPENDIX 14A – TABLE 7

Equations of the 1D Module [Ref-1]

MASS BALANCE EQUATION OF PHASE K

$$A\frac{\partial}{\partial t}\alpha_K\rho_K + \frac{\partial}{\partial z}A\alpha_K\rho_K V_K = \Gamma_{iK}$$

TRANSPORT EQUATION FOR NON CONDENSABLE GAS

$$A\frac{\partial\alpha\rho_G X_i}{\partial t} + \frac{\partial A\alpha\rho_G X_i V_G}{\partial z} = S_i$$

MOMENTUM BALANCE EQUATION OF PHASE K

$$A\frac{\partial\alpha_K\rho_K V_K}{\partial t} + \frac{\partial A\alpha_K\rho_K V_K^2}{\partial z} + A\alpha_K\frac{\partial P}{\partial z} = AI_{iK} + \chi_F\tau_{WK} + A\alpha_K\rho_K g_z$$

ENERGY BALANCE EQUATION OF PHASE K

$$A\frac{\partial\alpha_K\rho_K(H_K + \frac{V_K^2}{2})}{\partial t} - A\alpha_K\frac{\partial P}{\partial t} + \frac{\partial A\alpha_K\rho_K(H_K + \frac{V_K^2}{2})}{\partial z} = AQ_{iK} + \chi_F Q_{WK} + A\alpha_K\rho_K V_K g_z$$

INTERFACE RELATIONSHIP

$$\sum_K \tau_{iK} = 0 \quad \sum_K I_{iK} = 0 \quad \sum_K Q_{iK} = 0$$

INTERFACE ENERGY TRANSFERS

$$Q_{ik} = q_{iK} + \Gamma_{iK}(H_K + \frac{V_i^2}{2})$$

q_{iK} is the interface to phase K heat flux

$\Gamma_{iK}(H_K + \frac{V_i^2}{2})$ is the energy transfer due to mass transfer

INTERFACE MOMENTUM TRANSFERS

$$I_{iK} = \tau_{iK} - p_i \frac{\partial \alpha_K}{\partial z} + \varepsilon_K A_m - \Gamma_{iK} V_i$$

τ_{iK} : interface friction term = stationary part of interface forces

$A_m = \beta\alpha(1-\alpha)\rho_m \left[\frac{\partial V_G}{\partial t} - \frac{\partial V_L}{\partial t} + V_G \frac{\partial V_G}{\partial z} - V_L \frac{\partial V_L}{\partial z} \right]$ is the added mass term which controls the speed of sound of the model $(\varepsilon_L = +1, \varepsilon_L = -1)$

$p_i \frac{\partial \alpha_K}{\partial z}$ is due to the non-homogeneous transverse pressure field. It controls the propagation of void fraction waves.

The P_i expression can be analytically derived in case of horizontal stratified flows (assuming a hydrostatic transverse pressure gradient).

For non stratified flows, the P_i term does not play an important role and the expression of P_i is chosen to provide the hyperbolic functionality of the system.

$\Gamma_{iK} V_i$ is the momentum transfer due to mass transfer.

APPENDIX 14A – TABLE 8 [REF-1]

CATHARE Qualification Test Matrix

Experiment	Principal phenomenon	Mech. Trans.	Interf. Heat Flux	Wall Heat Flux	Component
Moby Dick		•	•		Break
SMDLT	Critical	•	•		Break
SMDTC		•	•		Break
SMDT Bet	Flow rate	•	•		Break
Markiven		•	•		Break
Rebeca		•	•		Break
Dadine		•	•	•	
SMD vert		•			
SMD Horiz	Flow	•			
CANON V Tu		•			
TAPIOCA	Regime	•			
CANON V rod		•			Core
ECTHOR IL		•			Int. Leg
ECTHOR HL	Interface	•			Hot leg
G2		•		•	Core
PERICLES Bo	Friction	•		•	Core
PATRICIA SG1		•	•	•	SG Prim
PATRICIA SG2		•		•	SG Sec
ERSEC Tub		•	•	•	
ERSEC Rod	Reflooding	•	•	•	Core
ERSEC Osc		•	•	•	Core
PERICLES Rn		•	•	•	Core
ROSCO		•	•	•	Core
OMEGA Tub		•	•	•	
OMEGA Rod	Wall	•	•	•	Core
TPTF	Flux	•	•	•	Core
FLECHT GV		•	•	•	SG
EPIS	Condensation		•		IS
COSI	At RIS [SIS]				IS & Accu
COSI Inc	Incond		•		IS
CREARE	CCFL	•	•		Downcomer
UPTF Do	Downco	•	•		Downcomer
SEROPS	Déentr	•			Upper Plenum
PIERO	Voiding	•			Lower Plenum
MHYRESA	CCFL	•			Hot leg
SMD Tee	Phase	•			Break
INEL	Separation	•			Break
EDGAR	Fuel			•	Fuel
REBEKA	Model			•	Fuel
EVA	Pump	•			Pump
Pericles 2D	Multi-D	•		•	Core
UPTF	Effets	•			Upper Plenum

APPENDIX 14A – TABLE 9 [REF-1]

BETHSY Analytical Tests

Experiment	Principal phenomenon	Mech. Trans.	Interf. Heat Flux	Wall Heat Flux	Component
Pressu			•	•	Pressu
ti Core		•			Core
8.1 a	Pump	•			Pump

APPENDIX 14A – TABLE 10 [REF-1]

Experiments Relative to VVER's

Experiment	Principal phenomenon	Mech. Trans.	Interf. Heat Flux	Wall Heat Flux	Component
IVO Loop seal	Voiding	•			Int. Leg & Hot leg
IVO CCFL	CCFL	•			Core
REWET II	Reflooding	•	•	•	Core
Guidopress	Horiz. SG	•		•	SG

APPENDIX 14A – TABLE 11 [REF-1]

Assessment - Verification on Integral Tests

Several integral tests are calculated for:

- validating the general coherence of the models,
- testing the code capability to represent system effects,
- verifying the description of scaling effects,
- highlighting points needing further physical investigation.

SELECTED TEST FOR THE ASSESSMENT OF CATHARE 2 Version 1.3 Revision 5

LB(LOCA)'s		
	LOFT L2-5	LB(LOCA)
	LOFT LP-LB 01	LB(LOCA)
	BETHSY 6.7 a4	Reflooding
	BETHSY 6.7 c	Reflooding
	+ UPTF	Refill

CALCULATION FROM BETHSY ANALYSIS GROUP		
	BETHSY 6.2 TC	6" CL break
	BETHSY 4.1 a TC	Natural circulation
	BETHSY 5.2 c	Loss of feedwater
	BETHSY 4.3 b	6 tubes SGTR
	BETHSY 8.1 a	2-phase Forced Circulation
	LSTF SBCL 09	10% CL break
	LSTF SBCL 15	0,5% CL break

SB(LOCA)'s & OTHER TRANSIENTS		
	BETHSY 9.1 b	2" CL break
	BETHSY 6.1 a	6" CL break
	BETHSY 6.1 b	3" CL break
	BETHSY 6.3	10" CL break
	BETHSY 6.5	Break at Pressuriser
	BETHSY 3.4 b	1 tube SGTR
	LOFT L9-3	Anticipated Transient Without Scram
	BETHSY 4.1 b	Natural circulation

INDEPENDENT ASSESSMENT (JAERI, JRC, ISPRA, PISA University, IPSN)		
	LSTF SBCL 5	5% CL break
	LSTF SBCL 21	6" CL break
	LSTF TH-RH 02	Loss of RIS/RRA [SIS/RHRS]
	LSTF TR-LF 03	Loss of electrical power
	LOBI A2 81	1% CL break
	LOBI BL 12	1% CL break
	LOBI BL 34	6% CL break
	LOBI BT 12	100% Steam Line Break
	LOBI BT 01	10% Steam Line Break
	SPES isp 22	Loss of Feedwater
	PACTEL isp 33	Natural circulation
	PMK spe 1	7.4% break
	PMK spe 2	7.4% break
	PACTEL	LOFW

APPENDIX 14A – TABLE 12 [REF-1]

Validation of CATHARE on Integral Test Facilities

System test facilities for CATHARE verification

LOOP	VERT. SCALE	VOLUME SCALE	POWER	PRESSURE MPa	LOOP NB	CORE
LOFT	1/2	1/48	100%	16	2	Nucl
LSTF	1/1	1/48	14%	16	2	Elect
BETHSY	1/1	1/100	10%	16	3	Elect
PKL	1/1	1/134	5%	4	3	Elect
LOBI	1/1	1/700	100%	16	3	Elect
SPES	1/1	1/427	100%	16	3	Elect
PACTEL	1/1	1/305		8	3	Elect
PMK	1/1	1/2070	100%	16	1	Elect

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9. THEMIS

THEMIS [Ref-1] is a computer code for simulating the transient behaviour of a multi-loop pressurised water reactor (PWR). Its field of application extends to transients that involve:

- reactivity changes,
- the heat removal capacity of the steam generator (SG) secondary sides,
- the reactor coolant flow,
- the reactor coolant system pressure,
- water inventory.

In general THEMIS is applicable to all of the transients in which the reactor coolant system remains intact (thus not including LOCAs, but SG tube breaks can be modelled with THEMIS).

THEMIS simulates the PWR reactor coolant system under sub-cooled (single-phase) or saturated (two-phase) conditions, with a low void fraction, under the hypothesis of homogeneous flow with thermal equilibrium between the water and the steam. This restriction does not apply to the pressuriser or to the volume under the reactor vessel closure head, for which single-phase and two-phase conditions can be simulated with phase separation and thermal equilibrium.

THEMIS models the reactor vessel and the reactor core, the hot and cold legs of the reactor coolant loops, the SGs (primary and secondary sides), the reactor coolant system pumps, and the pressuriser. The number of reactor coolant system loops can vary from one to four. A point neutron kinetics model is used for the core. The SG secondary side is simulated by a saturated homogeneous model (point model) or by a multi-variable volume model (axial model) for most thermal transients. The exception is the primary low-temperature overpressure transient, for which axial modelling with thermal imbalance between nodes is used for the SG secondary side. The control systems (RCCAs, pressuriser level and pressure, turbine bypass), protection systems (various reactor trips), and engineered safeguard systems (safety injection and accumulators) are also modelled.

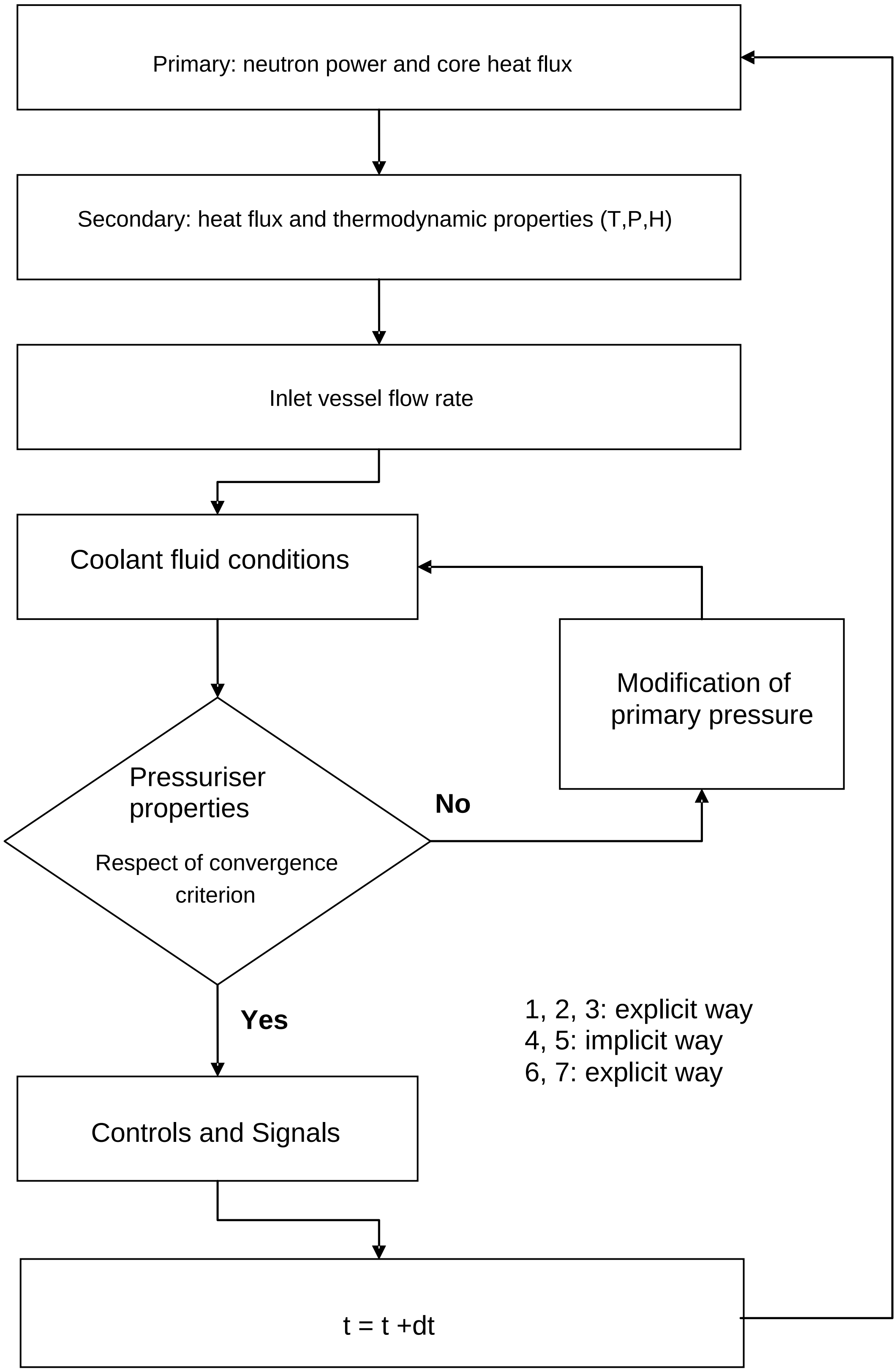
Revision 4 of the THEMIS code has been used for the BDR accident analyses, Appendix 14B.

The major features and characteristics of the THEMIS code are summarised below:

- Numerical scheme (see Appendix 14A – Figure 11).
- Physical law (see Appendix 14A – Table 13).
- Primary nodalisation {CCI Removed} [b].
- Secondary axial steam generator nodalisation {CCI Removed} [b].
- Qualification (see Appendix 14A – Table 14).

APPENDIX 14 – FIGURE 11

THEMIS - Numerical scheme



APPENDIX 14A – TABLE 13

THEMIS – Physical laws

Primary side:

- basic balance equations (motion, mass and energy)
- homogeneous fluid without non-condensable gas
- two separate phases in the pressuriser and in the upper plenum
- one pressure for the thermal-hydraulic properties

Secondary side:

- basic balance equations (motion, mass and energy)
- two phases with drift flux and a grid of heat transfer correlations
- one pressure for the thermal-hydraulic properties

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APPENDIX 14A – TABLE 14 [REF-1]

THEMIS - Qualification

Validation of code models	Phenomenon/or general aspect	Plant/Specific experimental loop
Reactor vessel upper head	Cooling and depressurisation with natural circulation (including coastdown of RCP [RCS] pumps)	Gravelines 1 Neyrpic mock-up
Vessel	Vessel mixing flow	Blayais 1, Cruas 4, Paluel 3 Lacydon (EDF/CEA mock-up)
Pressuriser	Measurement of pressuriser heat loses Adjustment of the continuous spray Start-up of the normal/auxiliary spray Start-up of heaters Insurge transient (piston effect) Water-solid pressuriser	Cruas 2 , Bugey 4, Tricastin
Axial steam generator	Steady state calculations Transient calculation: reactor trip	Bugey 4, Paluel Paluel 1
	Axial pre-heater: - steady state (30% to 100%NP) - transient: reactor trip & alternate steam flow steps	Megeve mock-up
	Velocity between liquid and steam phases	Patricia mock-up
Overall verification	Atmospheric steam dump	Paluel 3
	ATWS	Loft mock-up
	House load	Paluel
	Comparison with the LOFTRAN code (approved by NRC): • certain number of physical phenomena are represented in the same way, • core model & point steam generator are identical, • general hydraulic model identical: control volume method, homogenous flow with thermal equilibrium between phases, slug flow.	

10. SMART

The core calculations are performed with the SMART code [Ref-1]. SMART is the 3D code of the Science channel.

SMART is a two-energy-group, 3-D nodal diffusion code, incorporating the most recent nodal technologies.

SMART uses the nodal expansion method characterised by the use of nodal coupling equations with discontinuity factors. SMART solves the coupled nodal balance and leakage equations using three different levels of iteration: inner iterations, fission source iterations, and nodal coupling coefficient updates. The nodal coupling coefficients are updated by solving the two-node interface problems with a quadratic transverse leakage approximation.

The SMART feedback model, which includes a closed channel thermal-hydraulic module, is based on a multi-parameter tabulation for cross-sections. Fuel depletion is modelled using microscopic depletion.

Local reconstruction of the flux, power, burn-up, and reaction rates is based on a combination of homogeneous intra-nodal fluxes computed at each step and tabulated power form factors. The homogeneous intra-nodal flux is reconstructed using surface currents, surface fluxes, corner point fluxes, and nodal average flux. Power form factors are from the lattice calculation in APOLLO 2 [Ref-1].

11. FLICA III-F

The FLICA III-F [Ref-1] [Ref-2] computer program determines in a very general way the steady state and transient flows of a fluid flowing in separate or connected channels. It is thus a suitable tool for the thermal-hydraulic analysis of reactor cores or experimental loops with heating rod bundles.

Using a system of coordinates composed of an axis parallel with the axis of the channel and axes perpendicular to the interfaces between sub-channels, it is possible to avoid the imposition of a rectangular geometry.

The channels are essentially described by cross-sectional area, hydraulic diameter, and heating perimeter. The connections between sub-channels are mainly defined by the width of the gap, the hydraulic diameter representing the resistance to cross-flow and by a representative length to calculate the derivatives of the physical values at the interfaces.

A given heat flux can be imposed to any channel, with either the same or varied axial distributions.

The boundary conditions are prescribed:

- inlet or outlet pressure,
- flow rate inlet distributions or pressure drop between the inlet and outlet,
- inlet enthalpy distribution or fluid temperature.

The equations are solved with the finite difference method by an iterative scheme at each level.

The two-phase fluid model is homogeneous with slip. The code also includes an equation enabling the calculation of the real steam quality up to the dry steam state.

The axial representation of the assembly consists of 31 mesh elements. The 26 mesh elements dealing with the active fuel length are nearly all of the same height.

In most cases, the calculations are performed with a design radial power distribution shape. 68 channels are considered for one eighth of the core. In the hottest assembly, each physical channel is described by a numerical channel, whereas for the rest of the core, a numerical channel describes a whole assembly.

12. ALICE 2

The computer code ALICE 2 determines the activity released into the compartments of a nuclear installation: cells, rooms, filters, etc. The activity release into the environment is also determined.

ALICE 2 calculates the radionuclide migration in gaseous phase from a source at a time dependent rate and the release into the environment.

The isotopes are released in the gaseous and in the steam-gas mixing flow.

The time dependent thermal-hydraulic conditions in the whole circuit are assumed known.

The release rate to the environment is limited by the following phenomena:

- Particle deposition.
- Volatile condensation and coagulation.
- Radioactive decay.

13. CONPATE 4

The CONPATE 4 computer code determines the change of pressure and temperature inside the containment of a nuclear reactor building following the release of large quantities of water and steam into the containment, for example in the event of accidents involving loss of primary or secondary coolant.

The model simulates a single-volume containment with two-phase representation as follows:

- a gas phase, consisting of a mixture of air and steam,
- a liquid phase, consisting of the water in the containment sumps.

The break flow released into the containment is separated into a fraction of steam and a fraction of liquid, which are added to the containment gas and liquid phases, respectively.

The condensation of steam and the absorption of heat by the containment structures are calculated, together with the distribution of temperature in these structures.

The code enables modelling of the different types of possible safeguard systems inside the containment:

- a sump cooler system (e.g. LHSI in residual heat removal mode (LHSI/RHR) cooler),
- a containment spray system,
- a containment fan cooler system.

and modelling the switch to sump water recirculation configuration. The code can also model other non-condensable gas such as hydrogen for which combustion can be simulated.

The containment is divided into two subsystems:

- The gaseous region is composed of an ideal mixture of air and steam. It is defined by its temperature, its volume, its internal energy, the masses and partial pressures of the constituent air and steam.
- The liquid region exchanges mass and energy with the gaseous region; it is defined by its mass, temperature and pressure.

In each phase (liquid or gaseous), the pressure and temperature are homogeneous. The two phases are in mechanical equilibrium but in thermal non-equilibrium.

The heat conducting systems are the cold materials and structures present inside the containment. They are capable of exchanging heat with the gaseous and liquid regions and other heat sources. They are defined by temperature as a function of depth, and by their thermal properties (i.e. thermal conductivity, density and specific heat). The heat conduction is assumed one dimensional; several layers, each with specific nodalisation, might be modelled within each structure.

Mass and energy transfers between the systems are calculated using equations which allow for the temperature (and pressure) of steam, air, liquid water and heat conducting systems. This produces a set of differential equations, which includes the following:

- state equations for water, steam and air,
- mass balance equations (in the gaseous region and the liquid region),
- energy balance equations (in the gaseous region, in the liquid region including the heat exchangers after switch over to recirculation mode and in heat conducting systems),
- conservation equations for the total containment free volume.

This set of differential equations is transformed using the finite difference method into a system of n equations with n unknowns, which is resolved by matrix methods.

The fluid released through the break has a quality which depends on the break location and on the time after the break occurrence. Different ways of sharing this fluid between the gaseous and the liquid regions exist:

- water below saturation directly joins the liquid region,
- superheated steam directly joins the gaseous region,
- emulsion is divided into steam and liquid water by boiling at the break location. Steam which is saturated at the total pressure of the containment joins the gaseous region. Liquid water which is saturated at the total pressure of the containment joins the liquid region. This type of separation at the break is called the "Pressure Flash Method", it provides pessimistic values for pressure and temperatures in accordance with international practice [Ref-1].

The thermal power exchanged by convection and condensation between the containment structures and the fluid (air-steam mixture or sump water) is governed by TAGAMI-UCHIDA correlation as described in the NUREG 0588.

13.1. QUALIFICATION OF THE CONPATE 4 CODE

The qualification of the CONPATE4 code is based on the following principles:

- the use of general physics laws,
- the use of recognised and proven correlations,
- the global comparison of the calculation results with those obtained using other qualified codes.

13.1.1. General laws of physics

The main physics laws used by the code are the conservation of mass and energy, the state equation for ideal gas and steam-water characteristics, the physical properties of materials and the laws of conduction and heat transfer.

13.1.2. Recognised and proven correlations

The correlations used for heat transfer with internal structures are mainly the Tagami and Uchida correlations which have been established on the basis of experimental test and which are widely used and recognised.

13.1.3. Comparison with other codes

The global qualification of CONPATE 4 is based in particular on its comparison with the code PAREO 8 [Ref-1] developed and qualified by EDF/SEPTEN, and devoted to the same kind of analyses with the same code modelling principles.

14. MANTA

MANTA [Ref-1] is a computer code for simulation of all non LOCA PWR plant transients, including variations of core reactivity, SG heat removal capability, primary flow, primary pressure, and primary mass inventory. Its objectives address:

- the simulation of any complex fluid system, such as the primary and secondary circuits of the PWR, but also any other fluid system such as the Reactor Heat Removal System (RIS/RRA [SIS/RHRS]) or Chemical and Volumetric Control System (RCV [CVCS]) if needed,
- flexible coupling of thermal-hydraulics with different neutronics modelling (0D, 1D, 3D): the code properly describes the neutronic core behaviour and its feedback on thermal-hydraulic behaviour, according to the analysed transient and dominant phenomena,
- user friendly detailed modelling of Instrumentation and Control (I&C) systems, in order to save time and effort for description of these systems.

MANTA addresses design and optimisation of I&C systems, equipment design, systems performance verification, incidents and accidents analyses for plant safety evaluation.

MANTA is applicable for the following transients:

- Steam Generators (SG) feedwater flow abnormal conditions: Feedwater malfunction, Feedwater Line Breaks (FWLB), loss of normal feedwater flow, total loss of feedwater flow (normal and emergency), coincidence with reactor trip failure (Anticipated Transients Without Scram or ATWS).
- SG steam flow abnormal conditions: steam line breaks (SLB), spurious opening of steam relief device, excessive load increase, total loss of load.
- primary flow rate abnormal conditions: loss of main reactor coolant pumps, frequency forced reduction, locked rotor, reactor coolant pumps start-up (in particular during solid state RCP [RCS] condition), natural circulation.
- miscellaneous incident or accident events: spurious Safety Injection System (RIS [SIS]) actuation, spurious pressuriser safety relief valve opening (PSRV), SG tube rupture (SGTR), SLB in coincidence with SGTR, total loss of electric supply, etc.

MANTA applicability range excludes those transients exhibiting a large draining of the RCP [RCS] (LOCA events, feed and bleed transients), possibly resulting in core uncovering.

14.1. PHYSICAL MODELS

MANTA thermal-hydraulics feature a five equation model, for thermodynamic non equilibrium between liquid and steam phases, within the basic 1D element of the code:

- two mass conservation equations (one for total mass, one for steam phase),
- two energy conservation equations (one for each liquid and steam phase),

- one momentum equation for average mixture flow, supplemented by a drift flux model providing differential velocities for both phases, in co-current and counter-current situations, with respect to counter-current flow limit.

Reactivity calculation accounts for moderator density, boron concentration, fuel effective temperature (Doppler Effect), and control rod positions. The reactivity is input to a point kinetics model included in MANTA. This basic core model may be switched off if needed and replaced by coupling with a 1D or 3D neutronic core model.

14.2. MODEL DESCRIPTION

The meshed pipe is the basic element of the 1D code modelling of fluid systems.

Junctions allow pipe connections; each meshed pipe receives one upstream and one downstream junction; each junction can be connected to any number of pipes. Junctions are point fluid capacities, with thermal and hydraulic inertias; energy balances are treated in full non equilibrium, specified by the user.

Thermal walls allow heat exchange between two objects (pipe, junction, thermal boundary condition, 0D thermal mass). Thermal walls are characterised by their geometry (flat or cylindrical), area and thickness, thermal conductivity and heat capacity of different layers; each layer can be divided into thermal meshes allowing for numerical solution of the heat conduction equation.

System objects are pumps, flow restrictors, relief or safety valves, or breaks connected to junctions as hydraulic boundary conditions and driven by input data or external computer code coupled with MANTA or modelled I&C.

Thermal boundary conditions, imposed on thermal walls, and hydraulic boundary conditions, imposed on junctions at pipe ends, can be user defined or from an external coupled code.

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14.3. REACTOR VESSEL FLOW MIXING

Some transients, such as the SLB accident, develop temperature differences between the faulted and intact primary loops. MANTA propagates this asymmetric behaviour in the reactor vessel by user modelling: the vessel down comer and lower plenum are split into N regions (N = number of primary coolant loops), while the core is divided into K parallel pipes. MANTA models the flow distribution between the N modelled lower plenum regions into the K modelled core channels by the following relationship for total mass balance:

$$w_k = \sum_{i=1}^{N} p_k . f_k^i . W^i$$

w_k = mass flow-rate at core channel k inlet
W^i = mass flow-rate from primary coolant loop i
p_k = ratio of mass flow-rate at core channel k inlet, to average core inlet flow-rate
f_k^i = distribution factor of flow-rate from primary loop i into core channel k ($\sum_{k=1}^{K} f_k^i = 1$)

Similar relationship applies for total energy balance, using the enthalpy flow-rates.

This flow distribution model relies heavily upon the pk and fki coefficients: they can be obtained from representative test results of flow mixing inside the reactor vessel and flow distribution at core inlet.

These tests can also be used to derive similar coefficients for characterisation of the distribution of core channels outlet flow-rate.

14.4. MAIN REACTOR COOLANT PUMP

Unless externally imposed, MRCP rotation speed is deduced from the momentum equation of the MRCP shaft, accounting for applied electrical, hydraulic, and friction torques. Hydraulic torque and pump head are derived from homologous curves, using the rotation speed and the volumetric flow rate.

14.5. I&C SYSTEMS

The definition of the automatic control & protection systems is modular. It can be adapted to the plant configuration (number of primary loops, structure of the systems, etc), the selected modelling and the requirements of the studies.

Any existing technology (analogue or digital) can be represented. This is taken into account especially in the management of the time step for the I&C systems. The occurrence of events such as "manual actions" (for example operator action for stopping a pump) or the switchover of an I&C threshold are detected. The time step automatically adjusts to represent the time coupling between I&C systems and physical objects.

A library of elementary functions has been developed (filters, controllers, etc). This library could be extended to represent the functionalities of new equipment.

A graphic interface is used to develop the models of I&C systems. Therefore, the control systems can be described in as much detail as necessary (e.g. links between the different control systems, interlocks, alarms, etc).

Control and protection systems actuate automatic actions of reactor systems such as control rods, valves, and pumps. The kinetics of these systems (open/close, start/stop, etc) are modelled by MANTA.

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14.6. MANTA ASSESSMENT

With respect to PWR primary circuit, all transients are either heat-up or cool-down events, which result in the pressuriser filling or draining. Therefore, primary pressure transients are strongly linked with pressuriser behaviour, possibly also with reactor vessel upper head behaviour in the case of void formation occurring in this area. Asymmetric transients, due to the initiating event occurring in the secondary side of one SG, are handled by the mixing matrix in the reactor vessel, leading to the core inlet temperature distribution. Basic phenomena are:

- pressuriser filling (piston effect) or draining,
- pressuriser swell level increase (opening of relief or safety valve),
- steam ingress into the RCP [RCS] in the case of pressuriser draining and void formation in the RCP [RCS],
- pressuriser spray and heaters effects,
- water plugs propagation in the RCP [RCS] (cold, hot, borated water),
- primary temperature evolution and feedback on core reactivity,
- primary loops flows imperfect mixing in the reactor vessel for asymmetric transients.

With respect to secondary side, important phenomena are related to SG pressurisation or blow-down, cool-down or heat-up, draining or filling, and to primary–secondary heat transfers.

MANTA assessment tests have been selected to address these phenomena.

14.6.1. Qualification

MANTA assessment relies upon the use of qualified correlations in their applicable range for flow models, wall heat exchanges and inter-phase thermal transfers, and comparison with test facilities or actual plant transients. The selection of qualification tests (separate effects and component tests) and of verification tests (reactor transients) are intended to address important physical phenomena appearing during those transients of the simulation area of the code.

Pressuriser tests performed on the Cruas 2 French 3-loop plant provide a means to assess the MANTA pressuriser response during heaters operation, spray operation, and level increase due to changing system operation [Ref-1].

Actual reactor trips were also used to assess the MANTA predicted response, in terms of primary to secondary heat transfer, primary temperatures, SG measured level and pressure changes (a Bugey 4 trip at 50% nominal power [Ref-1], and a Paluel 1 trip at 100% nominal power [Ref-1]).

An overpressure transient performed on the Bugey 4 plant, consisted in the start-up of one reactor coolant pump with a heterogeneous RCP [RCS] temperature distribution due to the cooling by the RIS/RRA [SIS/RHRS] during the preceding phase. For this test, MANTA predicted this pressuriser level increase within 1% of the measured value [Ref-1].

Some tests performed on the MB2 test facility (Mendler, 1986) were also used to assess the behaviour of the SG secondary side:

- one loss of normal feedwater flow rate test, for which the most significant parameter is the SG dryout instant, depending on the initial mass inventory and on the primary to secondary heat transfer during degradation of that mass inventory. MANTA predicted the SG dryout at 110 seconds, very close to the experimental value of 112 seconds,

- one SLB test during hot shutdown conditions (2% nominal power). MANTA predicted the important parameters to within 15% of the measurements, during the first minute of the transient, up to peak extracted power.

Another SLB type test was performed by MANTA. A five minute opening of one atmospheric steam dump valve at Paluel 3 during hot shutdown conditions, while the three other SG were kept isolated. This event produced SG depressurisation and level decrease, some primary cool down, and some pressure and level decrease in the non affected SG, due to reverse heat exchange with the cooled primary system. MANTA under predicted the faulted SG pressure within 10% of the measurement, with other parameters showing consistent impact. In addition, this asymmetric transient verified the adequacy of the reactor vessel mixing matrix from LACYDON hydraulic tests [Ref-2].

Finally, a natural circulation transient at the Gravelines plant (French 3-loop plant) was calculated. This test was of particular interest because of void formation within the reactor vessel upper head during the cool down and depressurisation operations. The measurements were taken until the key event occurred at 200 minutes, i.e. reaching saturation temperature in the reactor vessel upper head.

15. MANTA/SMART/FLICA

15.1. MANTA/SMART COUPLING

MANTA simulation area is in thermal-hydraulics. In addition, MANTA is supplied with an Instrumentation and Control Systems model, with 0D (point) neutronic kinetic model, and with fuel models (0D and 1D radial).

The SMART code has been coupled with the MANTA code. SMART neutronic model provides the 3D nuclear power distribution. The nuclear power is split into a part f deposited in the fuel pins and a part (1-f) directly deposited in the water. SMART fuel pin model provides the Doppler temperature (Tceff) to the neutron model and the heat flux across the clad (Ptherm) to MANTA thermal-hydraulic model. MANTA provides in turn water specific volume (vs) and boron concentration (Cb) to the neutron model, and the wall temperature at clad internal surface (Tpig) to the SMART fuel thermal model.

This current limitation ("closed" channels) in the MANTA calculation of the global thermal-hydraulic response of the primary circuit, is one incentive for a further coupling plan, which will add the core thermal-hydraulics 3D code FLICA to the already coupled MANTA and SMART codes.

15.2. MANTA/SMART/FLICA III-F COUPLING

The thermal-hydraulic and neutronic codes are coupled together in this analysis:

- for each time step, the thermal-hydraulic conditions of the RCP [RCS] and SG are calculated by the MANTA code,
- the thermal-hydraulic conditions at the inlet of the core (temperature map, flow, pressure, boron concentration) are transferred to the FLICA code, to calculate initial core thermal-hydraulics,
    - from the core thermal-hydraulic conditions, SMART calculates the neutronic parameters and transfers the heat flux across the clad and a part of the nuclear power directly deposited in the water to FLICA, until convergence,
    - from the neutronic parameters, FLICA calculates core thermal-hydraulics, and transfers them back to SMART.

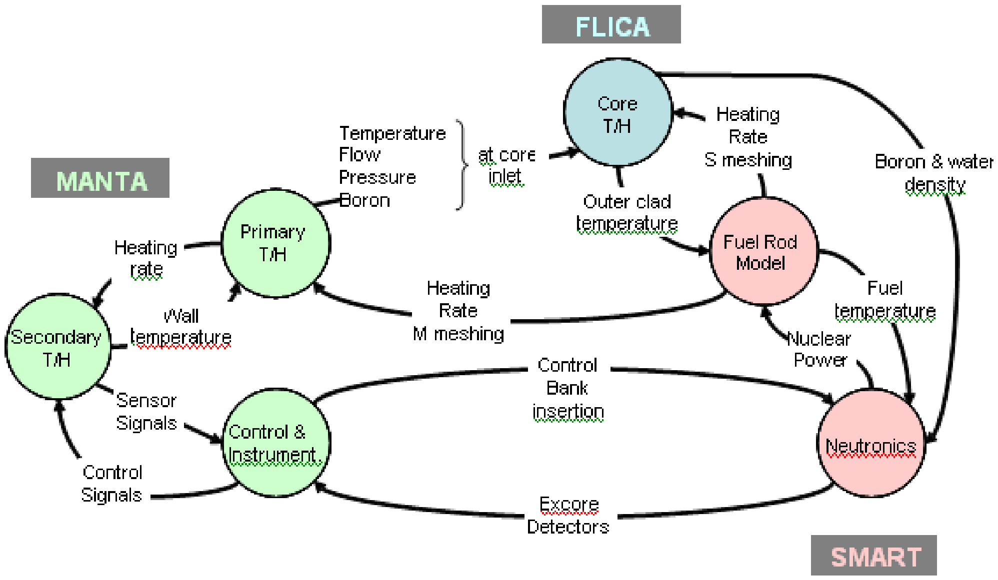
SMART returns to MANTA the power generated in each of the 241 core assemblies for MANTA to redistribute in the 4 MANTA vessels modelled (one corresponding to each primary loop).

See Appendix 14A – Figure 16 for a summarised description of MANTA/SMART/FLICA coupling.

Qualification of MANTA-SMART-FLICA relies on qualification of each of its elements.

APPENDIX 14A – FIGURE 16 [REF-1]

MANTA/SMART/FLICA Coupling principles



16. COMBAT

The COMBAT computer program [Ref-1] calculates the transient temperature distribution in a cross-section of a fuel rod (cladding, pellet-cladding gap, UO_2 pellet) and the transient heat flux at the surface of the cladding, using as input the nuclear flux, the fuel neutronic and mechanical characteristics with or without burnable poisons in the core, and the time dependant coolant parameters.

COMBAT uses a fuel model having the following characteristics:

- a sufficiently large number of radial regions to handle fast transients such as RCCA ejection accident,
- calculation of temperature as a function of the physical properties of the fuel and evaluation of heat transfer in the gap between fuel and cladding,
- transient analysis after boiling crisis: heat transfer correlations for film boiling, water-zircaloy reaction and partial fuel melting.

APPENDIX 14A – REFERENCES

External references are identified within this appendix by the text [Ref-1], [Ref-2], etc at the appropriate point within the appendix. These references are listed here under the heading of the section or sub-section in which they are quoted.

1. S-RELAP5

[Ref-1] S-RELAP5 Models and correlation. Code Manual EMF-2100 Revision 6. (E)

[Ref-2] Validation of the S-RELAP5 program. KWU NDS1/97/1002. (E)

APPENDIX 14A – FIGURE 1

[Ref-1] S-RELAP5 Models and correlation. Code Manual EMF-2100 Revision 6. (E)

1.7. MAJOR MODIFICATIONS AND IMPROVEMENTS IN S-RELAP5

[Ref-1] S-RELAP5 Models and correlation. Code Manual EMF-2100 Revision 6. (E)

APPENDIX 14A – FIGURE 2

[Ref-1] S-RELAP5 Models and correlation. Code Manual EMF-2100 Revision 6. (E)

APPENDIX 14A – TABLE 4

[Ref-1] S-RELAP5 Models and correlation. Code Manual EMF-2100 Revision 6. (E)

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[Ref-1] NLOOP – A multiple Loop Code to Determine PWR Plant Transients – KWU Technical Report R15/85/e 1008. August 1985. (E)

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3.1.1.6. Reactor Core Model

3.1.1.6.1. Fuel Rod/Coolant Channel

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3.1.1.6.3. Decay power

[Ref-1] NLOOP – A multiple Loop Code to Determine PWR Plant Transients – KWU Technical Report R15/85/e 1008. August 1985. (E)

3.1.1.7. Steam Generator Model (Secondary Side)

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APPENDIX 14A – TABLE 5

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APPENDIX 14A – FIGURES 5 TO 7

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APPENDIX 14A – TABLE 6

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[Ref-1] A Barbier. CATHARE2 – Code Synthetic qualification assessment. EPD DC 490 Revision C. AREVA. September 2007. (E)

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