SPECTRA

***S**ophisticated*
***P**lant*
***E**valuation*
***C**ode for*
***T**hermal-hydraulic*
***R**esponse*
***A**ssessment*

NRG

***N**uclear*
***R**esearch and Consultancy*
***G**roup*

https://dyfuzjastudio.com/marekstempniewicz/files/SPECTRA-Leaflet.pdf

SPECTRA - Thermal-Hydraulic System Code

System Code

SPECTRA is a thermal-hydraulic system code. SPECTRA is designed for thermal-hydraulic analyses of nuclear power plants. The code main applicability is the area of Light Water Reactors, High Temperature Reactors, Liquid Metal Fast Reactors, Molten Salt Reactors, as well as conventional plants and chemical reactors. The code can be used for analyzing accident scenarios, including loss-of-coolant accidents (LOCAs), operational transients, and other accident scenarios in nuclear power plants. The code structure (Fig. 1) is based on Packages, which contain models on a given topic. The available models include multidimensional two-phase flow, non-equilibrium thermo-dynamics, transient heat conduction in solid structures, general heat and mass transfer package, with natural and forced convection, condensation, boiling. Detailed description of the code is available [Spectra, 2021].

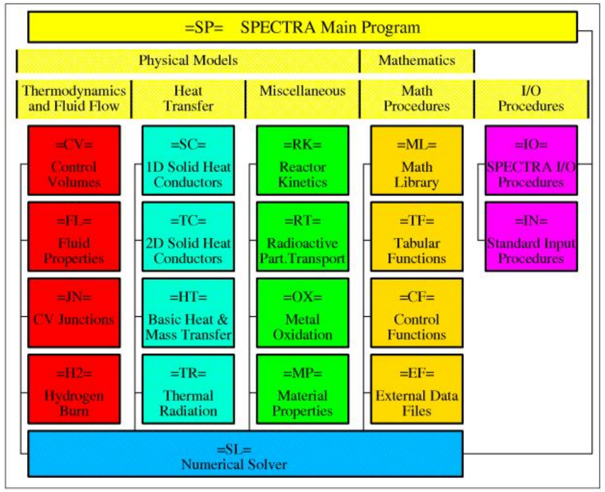

Fig. 1

Fluid Properties

- ***Built-in fluids***
  Built-in fluid property tables consist of the properties of water, steam, and several non-condensable gases, all of them treated as real gases. The steam properties were obtained using the NRC/NBC steam tables program, covering the range from 270 K to 3070 K, and from virtually 0.0 Pa to 2.1×10^7 Pa.
- ***User-defined gases***
  Non-condensable gases can be added as user-defined gases, these are treated as perfect gases.
- ***User-defined liquids***
  User-defined liquid properties can be applied as an alternative fluid. This can be used to liquid metal reactors, molten salt reactors, etc.

Basic Heat Transfer

Heat conduction is calculated by the 1-D (Fig. 2) and 2-D Solid Heat Conductor (Fig. 3) Packages, using a general transient heat conduction equation.

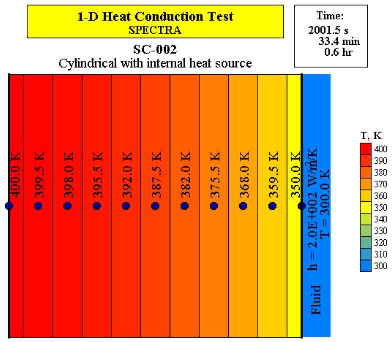

Fig. 2

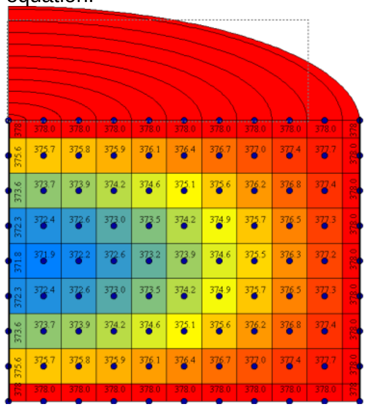

Fig. 3

The heat transfer at the boundaries of the Solid Heat Conductors is obtained from the Heat and Mass Transfer Package, which provides models for natural and forced convection, condensation with and without non-condensable gases, boiling curve (Fig. 4), including nucleate boiling, critical heat flux (CHF), transition boiling, film boiling, heat transfer in two-phase flow, as well as non-equilibrium boiling and condensation ("flashing" and "fogging"). The heat and mass transfer models include correlations valid for wall-fluid, pool-atmosphere, atmosphere-droplets, and pool-bubbles interfaces.

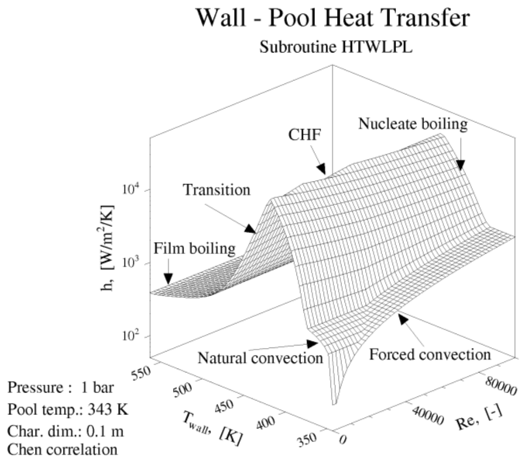

Fig.4

Thermal Radiation

A detailed thermal radiation model is provided, including grey enclosure models, with and without participating gas. The gas radiation model is based on the Hottel gas approach.

Reactor Kinetics

Point kinetics and nodal kinetics models are available, with reactivity feedbacks from control rods, fuel and moderator temperature, void fraction, as well as changes in isotope concentrations. The Isotope Transformation model allows computing core composition changes (due to fuel burn-up, production of poisons, such as Xe-135, fuel reload, etc.), and their effect on reactivity as well as decay heat production. For molten salt reactor applications, the reactor kinetics model has been extended to account for delayed neutron precursor drift, characteristic for molten salt reactors with circulating fuel. The model was validated based on MSRE data (Fig. 5) [Stem et al., 2017a].

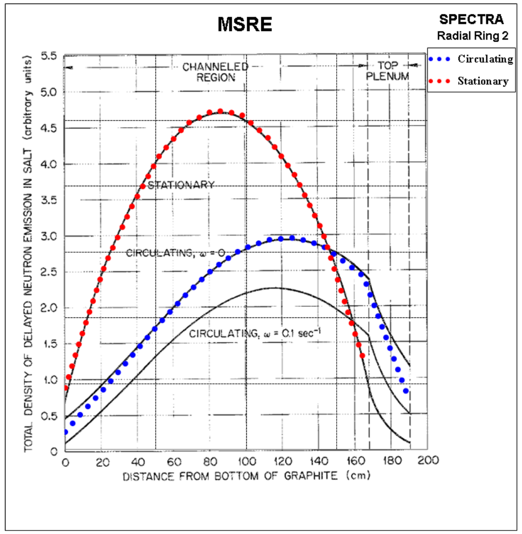

Fig. 5

Radioactive Particle Transport

The Radioactive Particle Transport Package deals with release of fission products, aerosol transport, deposition, and resuspension. Radioactive chains of fission products (Fig. 6) are tracked. The models for the transport and deposition of aerosols include Brownian diffusion, thermophoresis, and gravitational settling. For turbulent flow conditions, turbulent deposition models for the diffusional deposition, turbulent eddy impaction, and inertia impaction regime are included. Moreover, gravitational, Brownian, and turbulent coagulation are modeled.

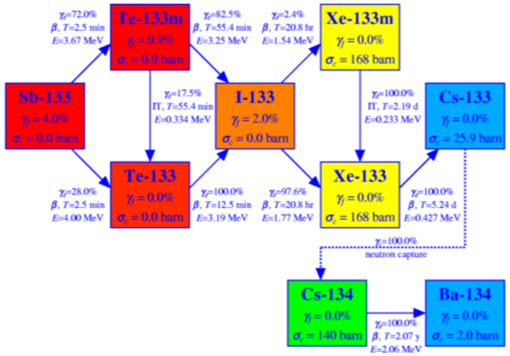

Fig. 6

Two state-of-the-art dynamic resuspension models are available. Alternatively, resuspension can be modeled using a parametric model with user-defined coefficients directly obtained from resuspension experiments. The following fission product models are included:

- FP release models (CORSOR, CORSOR-M, user-defined functions)
- Condensation of FP vapors
- Sorption of FP on surfaces
- Sorption of FP vapors on aerosol particles

Hydrogen Burn

A hydrogen burn model is available, which includes temperature-dependent flammability limits for slow deflagrations, fast turbulent deflagrations and detonations (Fig. 7) as well as ignition criteria.

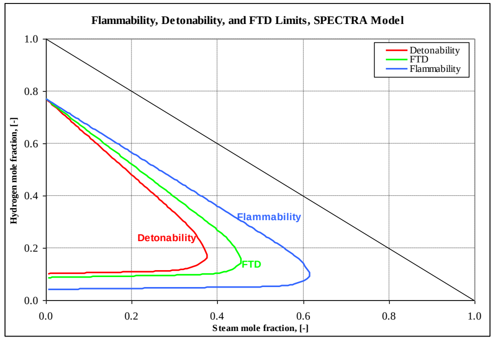

Fig. 7

Oxidation Models

An oxidation model is available, which consists of

- ***Built-in oxidation models***
  - Zr oxidation by steam, (Cathcart or Urbanic-Heidrich)
  - Steel oxidation by steam, (White)
  - Zr oxidation by O_2, (Benjamin et al.)
  - Graphite oxidation by O_2, (Roes)

- ***User-defined oxidation***

  A general oxidation equation with user-defined coefficients is available. Practically any oxidation reaction can modeled. Multiple reactions (e.g. simultaneous oxidation of Zr by steam and air) are possible. Recent additions:
  - reactions occurring during water ingress into HTR were implemented and tested [Stem, 2014].
  - A new correlation for air oxidation (Fig. 8) in spent fuel pools (SFP), including the effect of nitrogen, has been proposed, implemented and tested. The new model results were compared with the results of correlations available in MELCOR and ASTEC [Stem, 2016].

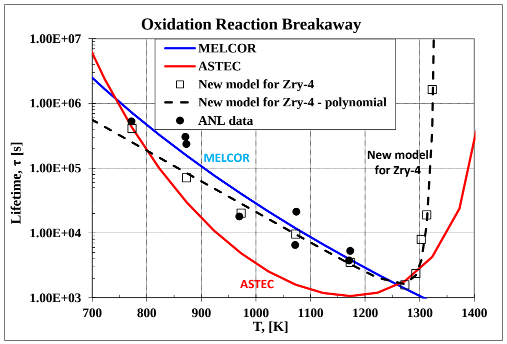

Fig. 8

Validation Based on Experiments - LWR

ORNL THTF Tests

ORNL THTF tests [Mullins et al., 1982] have been selected as one of the Separate Effect Tests (SET) to verify code capability to predict two-phase mixture flow and level swell in a tube bundle. The two-phase mixture level swell is important for a small break loss-of-coolant-accident (SBLOCA) in a PWR. The extent of core uncovery following SBLOCA strongly depends not only on the core liquid inventory but also on the core void fraction distribution. Models were built with SPECTRA and RELAP5 codes and both codes results were compared to measured data (Fig. 9).

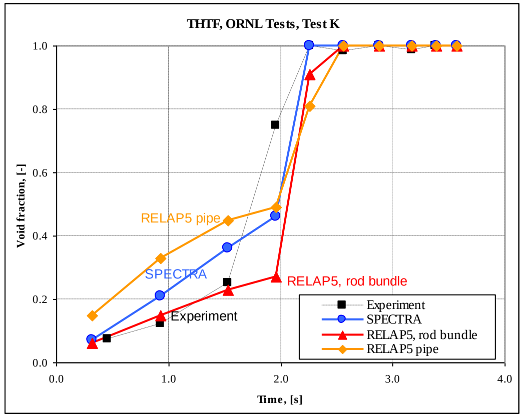

Fig. 9

Boiling in Narrow Channels - Monde Experiments

Code validation for geometries encountered in the research reactors requires data for narrow channels (D_{hyd}~1 mm). Natural convection boiling data obtained by Monde et al. was used to validate odes RELAP5 and SPECTRA for boiling in narrow channels. Models were built with SPECTRA and RELAP5 codes and both codes results were compared to measured data (Fig. 10, 11) [Stem et al., 2016].

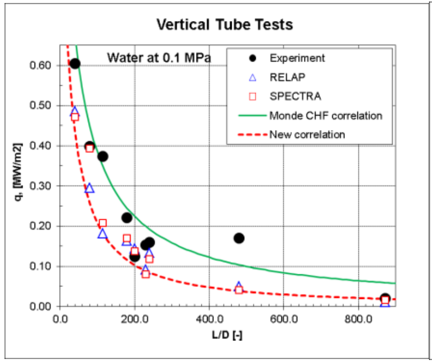

Fig. 10

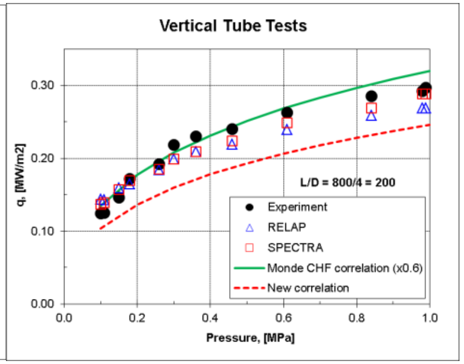

Fig. 11

ISP-35, NUPEC M-7-1 Tests (PWR Containment)

NUPEC hydrogen mixing and distribution test performed in Japan had been selected by CSNI as International Standard Problem No. 35. The purpose of this ISP was to verify the predictive capabilities of computer codes with respect to simulation of light gas (helium) mixing and distribution in a containment. NRG (at that time KEMA Nuclear) participated in ISP-35 with MAAP-4, including blind and open calculations. Later a SPECTRA model was created and open analysis was performed. The conclusions from MAAP4 and SPECTRA calculations:

- Generally good results were obtained with both codes.
- Multidimensional effects of spray were important.
- Modeling of spray in the MAAP / SPECTRA codes was adjusted to mimic the multi-dimensional effects in the open phase. This adjustments improved containment pressure prediction; helium concentrations were practically unaffected.

ISP-42, PANDA Tests (Advanced BWR Containment)

PANDA is a large-scale facility, which has been constructed at the Paul Scherrer Institute (PSI) for the investigation of both overall dynamic response and the key phenomena of passive containment systems during the long-term heat removal phase for Advanced Light Water Reactors (ALWRs). In the PANDA test facility a number of tests were performed for use as the basis of International Standard Problem number 42 (ISP-42). NRG participated in ISP-42 both blind and open phase using SPECTRA. ISP-42 consisted of six “Phases”: A through F (different experiments). The most difficult for simulation with system codes was Phase F, where significant stratification developed in the wetwell volume. Consequently, the containment pressure was clearly overpredicted in the blind phase (Fig. 12). Stratification models applied in the open phase (Fig. 13), allowed to obtain excellent agreement with experiment. The comparison of blind calculations, performed by the organizers, lead them to conclude that in the blind phase “The overall best results were obtained by the lumped parameter code SPECTRA” [Aksan, 2010].

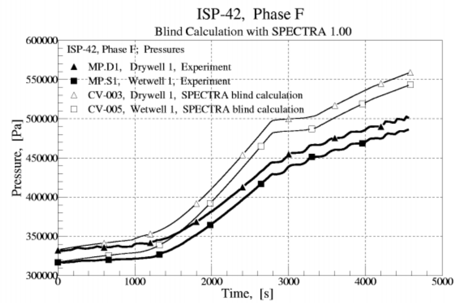

Fig. 12

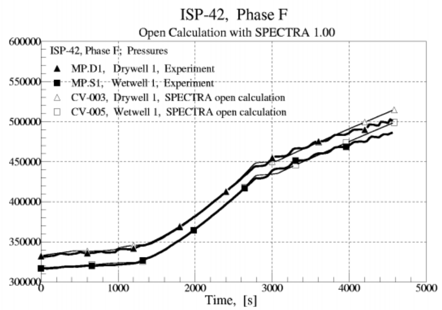

Fig. 13

PANDA PCC Tests (Passive Containment Cooling)

PANDA ("Passive Decay Heat Removal and Depressurization Test Facility") has been constructed at Paul Scherrer Institute (PSI) in Switzerland to study long term performance of the Simplified Boiling Water Reactor (SBWR) passive containment cooling system. The first experiments, conducted at the beginning of 1995, were the so-called S-series tests, performed to investigate the steady state operation of the Passive Containment Cooling (PCC) (Fig. 14) condenser unit at different fractions of non-condensables. The PCC consists of an upper drum, called "steam box", a vertical tube bundle, and a lower drum, called "water box".

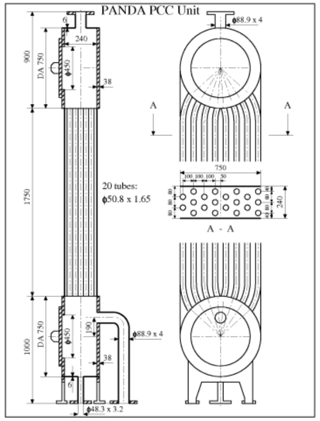

Fig. 14

NRG performed analyses of the S-series tests using four codes: TRACG (GE version), TRAC-BF1 (PennState), MELCOR 1.8.2, SPECTRA.

Comparison of code calculations and measured data showed that all codes under-predicted the PCC efficiency (fraction of steam condensed) [Stem, 2000] (Fig. 15). To explain this fact, it was later postulated that:

- condensate coming from the steam box, forms a stream of liquid in one or two tubes, connected in the lower part of the, steam box, leaving most of the tubes unaffected.
- condensate entering the water box is assumed to fall down in the form of droplets.

With these assumptions, the PCC efficiency increases because of smaller film thicknesses in the tubes and the water box. Results were closer to the experimental data, although small under-prediction still remained [Spectra, 2021].

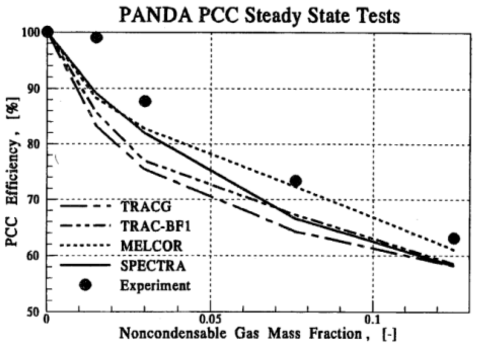

Fig.15

PANDA BC Tests (Building Condenser)

The TEMPEST project was devoted to studying passive decay heat removal systems for advanced BWR: the SWR 1000 reactor designed by Siemens. The passive decay heat removal system for SWR 1000 consists of the Building Condenser (BC), a finned tube heat exchanger placed at the top of the drywell. Experimental investigation of the BC performance has been performed at the PANDA test facility. One of the experiments, BC4 Test, was designed to study the BC performance under severe accident conditions, with hydrogen being generated in the core, and released to the containment (in the experiment helium was used instead of hydrogen). This test was selected for analytical investigation within the TEMPEST project. NRG performed a combined System Thermal-Hydraulic (STH) code / CFD code of the PANDA BC4 test within the TEMPEST project:

- STH code: SPECTRA,
- CFD code: CFX.

Large stratification in drywell-1 (Fig. 16) lead to overestimation of containment pressure in STH. When the stratification data from CFX was included in SPECTRA, containment pressure was very close to the experimentally measured value. The combined SPECTRA/CFX codes showed that stratification was responsible for overestimation of containment pressure obtained by several codes [Wichers, 2003].

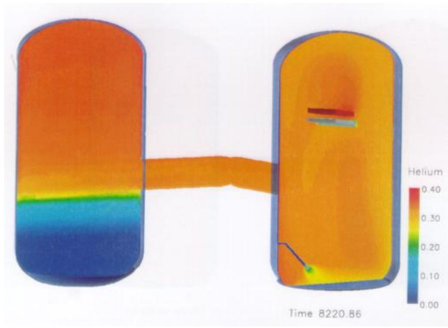

Fig. 16

HYMIT Tests (Hydrogen Burn)

A hydrogen deflagration experiment, performed in the HYMIT experimental facility at Shanghai Jiao Tong University (PR China) between October 17 and October 21, 2016, has been simulated with the lumped parameter system codes MELCOR and SPECTRA. The results were obtained without knowledge of the measured data, so that they are blind predictions. The purpose of the work was to enhance code validation and verification (V&V) of the codes. Some differences between MELCOR and SPECTRA results were observed, which were explained by differences in default burn models in different codes (Fig. 17).

- In MELCOR instantaneous propagation to all CV-s was observed. To prevent this, the value of TFRAC was set to 0.7. Furthermore, the hydrogen limit for ignition without igniters was enlarged from 0.10 to a large value (0.50). This leads to a gradual flame propagation through CV-s and is qualitatively consistent with SPECTRA and also with CFD and ASTEC calculations from [Holler, 2016].
- In SPECTRA flame acceleration to FTD occurs, that finally leads to a detonation. This could be avoided by increasing the constant in the σ-criterion from the conservative value of 3.5 to 4.0. This leads to slow deflagration and is qualitatively consistent with MELCOR and also with CFD and ASTEC calculations from [Holler, 2016].

The results will be further analyzed when the experimental data become available.

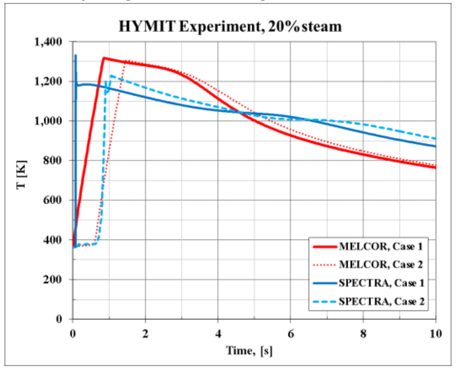

Fig. 17

Validation Based on Experiments - HTR

NACOK (HTR)

Air ingress into to the core after the primary circuit depressurization due to large breaks of the pressure boundary is considered as one of the severe hypothetical accidents for the High Temperature gas-cooled Reactor (HTR). The NACOK (Naturzug im Core mit Korrosion) facility was built at Jülich Research Center in Germany to study the effects of air flow driven by natural convection as well as to investigate the corrosion of graphite.

The NACOK air ingress experiment carried out on October 23, 2008 to simulate the chimney effect, was analyzed at NRG with the SPECTRA code, as well as at INET, Tsinghua University of China with the TINTE and THERMIX/REACT codes [Zheng-Stem, 2012]. The calculated results of air flow rate by natural convection, time-dependent graphite corrosion, and temperature distribution are compared with the NACOK test results. The preliminary code-to-experiment and code-to-code validation successfully proves the codes capability to simulate graphite corrosion (Fig. 18) during air-ingress accident.

Table 4: Average graphite weight loss

	Mass loss (average all blocks)
Experiment	8.4%
SPECTRA, Roes correlation	9.0%
SPECTRA, CEA correlations	9.4%
SPECTRA, NKL correlation	9.7%
TINTE	9.9%
THERMIX/REACT	10.9%

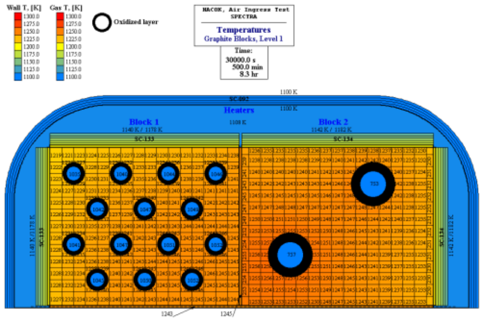

Fig. 18

HTTU

Separate effects tests were conducted to determine the effective thermal conductivity through the pebble bed for temperatures up to 1200°C in the HTTU facility [Rousseau et al., 2014]. The HTTU test section consisted of approximately 25,000 graphite spheres containing no nuclear fuel, with an outer diameter of 60 mm. The spheres were randomly packed within an annular core configuration bounded by inner (electrically heated) and outer (water-cooled) graphite reflectors (Fig. 19).

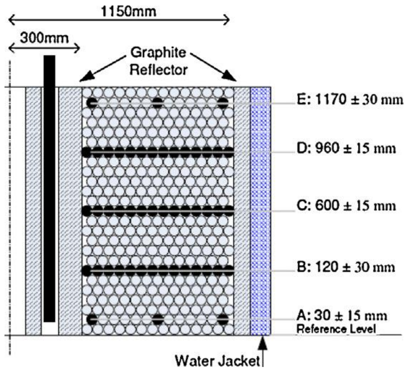

Fig. 19

SPECTRA model of HTTU was prepared. Calculations were performed using two effective conductivity correlations :

- developed based on the HTTU, 82 kW test data
- Zehner-Schlunder and Robold, which were applied in SPECTRA analyses of the HTR-PM reactor [Zheng et al., 2018]

The calculated results were in very good agreement with the measured data (Fig. 20).

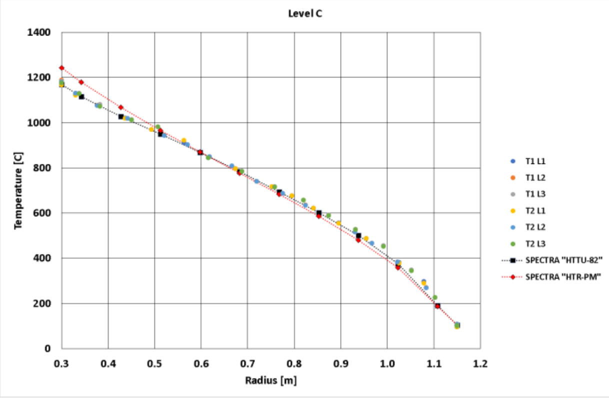

Fig. 20

Resuspension Experiments, Reeks and Hall

Graphite dust that will be generated in an HTR/PBMR during normal reactor operation will be deposited inside the primary system and will become radioactive due to sorption of fission products. A significant amount of radioactive dust may be resuspended and released from the reactor cooling system in case of a depressurization accident. Therefore accurate particle resuspension models are required for HTR/PBMR safety analyses.

A number of well-known resuspension models (Veinstein, Rock'n Roll) are available in SPECTRA. A new resuspension model was proposed by Komen and Stempniewicz [Komen-Stem, 2010]. This model, referred to as NRG4 or the K-S resuspension model, is available in SPECTRA. The resuspension models were investigated using the experimental data of Reeks and Hall experiments (monolayer deposit) [Komen-Stem, 2010].

Resuspension Experiments, STORM, ISP-40

The Committee on the Safety of Nuclear Installations of the OECD/NEA, in its meeting of November 1996, endorsed the adoption of STORM test SR11 as International Standard Problem number 40 (ISP-40). The test took place in April 1997 and included two distinct phases, the first concentrating on aerosol deposition mostly by thermophoresis and eddy impaction and the second on aerosol resuspension under a stepwise increasing gas flow.

NRG performed simulation of the STORM test SR11 using resuspension models in SPECTRA: Veinstein, Rock'n Roll, NRG3, and NRG4 (the K-S model). The resuspension models were compared to the measured data of the STORM experiment (multi-layer deposit) [Komen-Stem, 2010].

Validation Based on Experiments - LMFR

EBR-II (IAEA CRP)

The International Atomic Energy Agency (IAEA) Coordinated Research Project (CRP) “Benchmark Analyses of EBR-II Shutdown Heat Removal Tests” [Briggs et al., 2017] was initiated in 2012 with the objective of improving state-of-the-art SFR codes by extending code validation to include comparisons against whole-plant data recorded during landmark shutdown heat removal tests (SHRT) that were conducted at Argonne’s Experimental Breeder Reactor II (EBR-II) in the 1980’s.

At NRG the multi-scale thermal hydraulic simulation platform, consisting of the system thermal-hydraulic (STH) code SPECTRA and the CFD code ANSYS CFX, was used for transient simulations. Based on comparisons of core inlet/outlet coolant temperatures (Fig. 21), the intermediate heat exchanger (IHX) primary inlet temperatures, IHX secondary outlet temperatures and primary coolant flow rates with the measured data provided by Argonne National Laboratory (ANL), the SPECTRA stand-alone model and the multi-scale thermal hydraulic coupled SPECTRA/CFX model (Fig. 22, 23) proved to be able to provide satisfactory results for this benchmark [Stem, et al., 2017b].

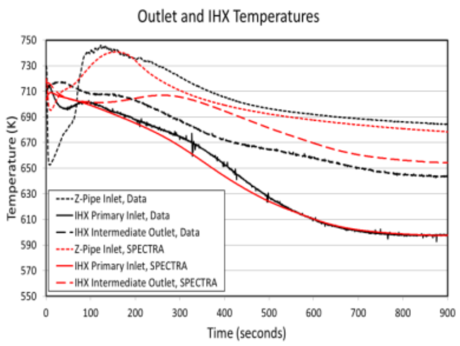

Fig. 21

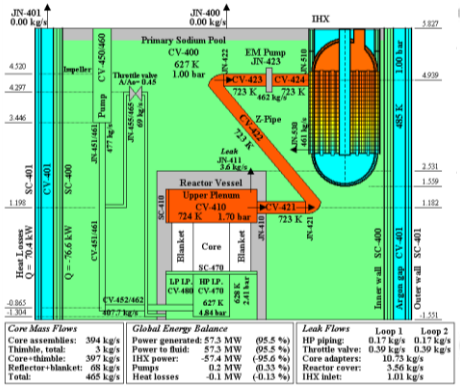

Fig. 22

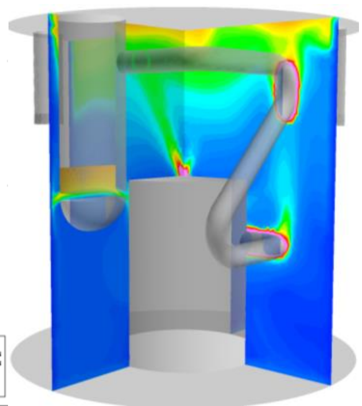

Fig. 23

Validation Based on Experiments - MSR

MSRE

The Molten Salt Reactor Experiment (MSRE) was in nuclear operation at ORNL from June 1, 1965 to December 12, 1969. During that time the reactor generated 13,172 equivalent full-power hours of energy at power levels up to 7.4 MW. Because the fuel is a circulating fluid, the mobility of all the fuel constituents, including the fission products, is an important consideration in the overall performance of molten-salt systems. This mobility is especially important for the noble-gas fission products because they, typically, have very low solubilities in molten salts and because some, notably Xe, are significant neutron absorbers.

The MSRE model for SPECTRA was created using data found in open literature. Steady state calculations were performed at design power of 10 MW (Fig. 24). The steady state model parameters compared to available data showed good agreement [Stem et al., 2017a].

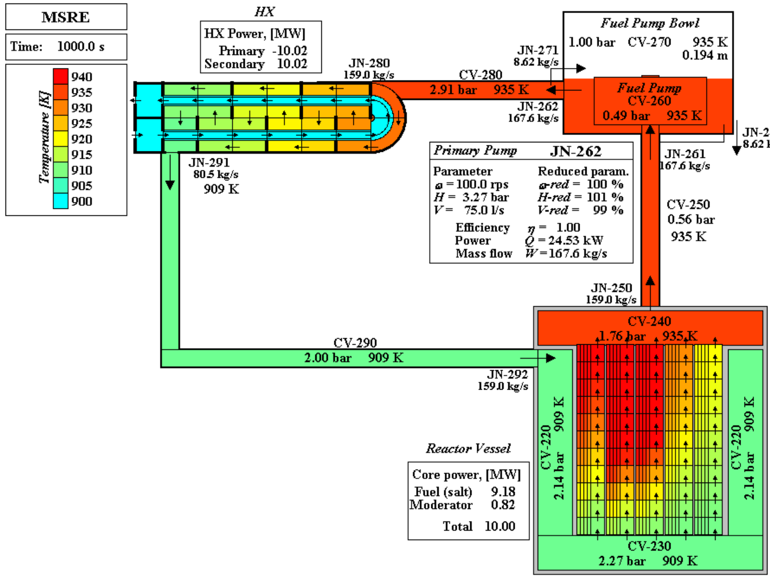

Fig. 24

Fission product behavior was analyzed. As a first step, delayed neutron precursors were modeled because of their importance for reactor kinetics. The calculated DNP behavior showed good agreement with data. As a next step other fission products will be analyzed, including Xenon and noble gases.

International Code-to-Code Benchmarks

GENERIC CONTAINMENT (LWR)

One outcome of the OECD/NEA ISP-47 activity was the recommendation to elaborate a 'Generic Containment' in order to allow comparing and rating the results obtained by different lumped-parameter models on plant scale. Within the European SARNET2 project such a Generic Containment nodalization (Fig 25), was defined based on PWR (1300 MW_e).

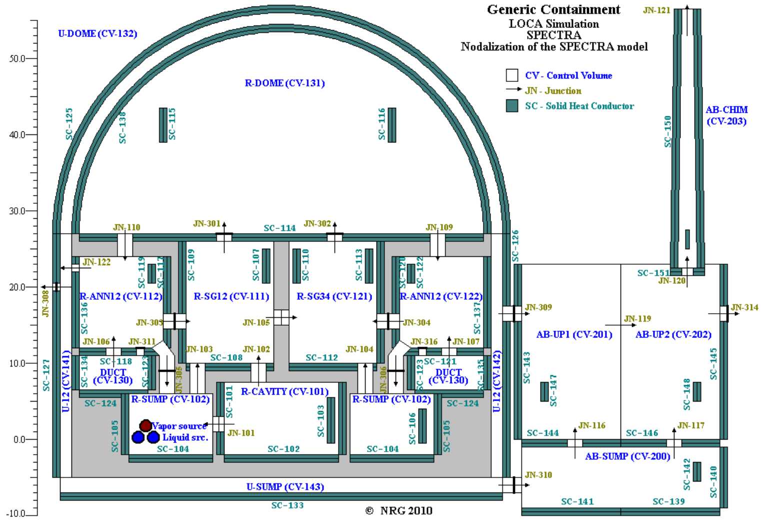

Fig. 25

The methodology applied in order to compare the different code predictions consisted of a series of three benchmark steps with increasing complexity as well as a systematic comparison of characteristic variables and observations:

- Run 0 – initial step
- Run 1 – detailed comparisons
- Run 2 – application to PAR (hydrogen recombiners) modeling

The participant used codes APROS, ASTEC, COCOSYS, CONTAIN, ECART, GOTHIC, MELCOR, SPECTRA. NRG participated in Generic Containment using MELCOR and SPECTRA codes. Both codes provided very similar results in all three steps. A significant user effect was observed, as results obtained with the same code (e.g. MELCOR) by different participants could differ significantly. It was concluded that, even though the problem was well defined, the uncertainty of calculated results due to different modelling approaches and users may be much higher than expected [Kelm et al., 2014].

AIR-SFP (LWR Spent Fuel Pool)

The Fukushima Dai-chi nuclear accident has renewed international interest in the safety of SFPs. In the frame of the SARNET2 FP7 project, several partners performed simulations of accident scenarios in SFP using different Severe Accident (SA) codes (ASTEC, MELCOR, ATHLET-CD, RELAP/SCDAPSIM, ICARE/CATHARE, SPECTRA) [Coindreau et al., 2017]. The studies have raised questions about the reliability of the results obtained since these codes were developed for reactor applications. The code to code comparison of the Air-SFP benchmark project showed not only differences from the different severe accident codes but also user differences by using the same code. NRG participated in AIR-SFP using MELCOR and SPECTRA codes.

ASTRID (LMFR)

In the frame of the ESNII+ FP7 EU Project, participants of the benchmark, using the ASTRID-like core neutronic and thermal-hydraulic specification (including reactivity feedback coefficients), developed the core models with their system codes and 0D neutron kinetics models. Calculations were performed on the most representative design basis accident: the unprotected loss of flow accident (ULOF) up to the initiation of sodium boiling [Bubelis et al. 2017]. Steady-state and dynamic simulation of the ULOF transient was simulated by participants using system codes in combination with neutron point kinetics: TRACE, CATHARE, SIM-SFR, SAS-SFR, ATHLET, SPECTRA, SAS4A. NRG participated with the SPECTRA code [Stem et al., 2018]. The NRG results were roughly in the middle (Fig. 26, 27) of the results of all codes.

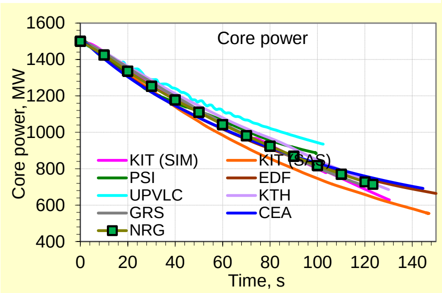

Fig. 26

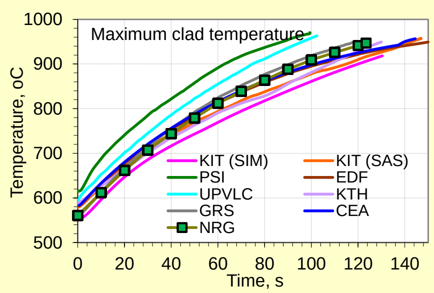

Fig. 27

ESFR (LMFR)

The new reactor concepts proposed in the Generation IV International Forum (GIF) are conceived to improve the use of natural resources, reduce the amount of high-level radioactive waste and excel in their reliability and safe operation. Among these novel designs Sodium Fast Reactors (SFRs) stand out due to their technological feasibility as demonstrated in several countries during the last decades. As part of the contribution of EURATOM to GIF the CP-ESFR is a collaborative project with the objective, among others, to perform extensive analysis on safety issues involving renewed SFR demonstrator designs. The verification of computational tools able to simulate the plant behavior under postulated accidental conditions by code-to-code comparison was identified as a key point to ensure the reactor safety level. In this line, several organizations developed models able to simulate the complex and specific phenomena involving multi-physics studies that this fast reactor technology requires. The participant used codes CATHARE, RELAP5, TRACE, SIM-SFR, SAS-SFR, MAT4-DYN, SPECTRA [Lazaro et al., 2014]. NRG participated in the ESFR benchmark using the SPECTRA code.

LEADER (LMFR)

Lead-cooled European Advanced DEmonstration Reactor (LEADER) is an EU design of liquid lead-cooled fast reactor. Within WP5, several computer codes (SAS-LFR, RELAP, TRACE, CFX, SIMMER, SPECTRA) were applied to evaluate consequences of selected unprotected accident scenarios such as Loss of Flow, Loss of Heat Sink, and reactivity-initiated accidents [Bandini, 2013]. NRG participated with the SPECTRA code (Fig. 28) Eight accident scenarios were analyzed [Stem, 2013]:

- TR-4 - Unprotected Transient Overpower: reactivity insertion
- T-DEC1 - Unprotected Loss of Flow: loss of all primary pumps.
- T-DEC3 - Unprotected Loss of Heat Sink: loss of SCS
- T-DEC4 - loss of off-site power.
- TO-3 - loss of FW pre-heater
- TO-6 - 20% increase of FW flow
- T-DEC6 - SCS failure
- T-DEC5 - Partial blockage of hottest assembly

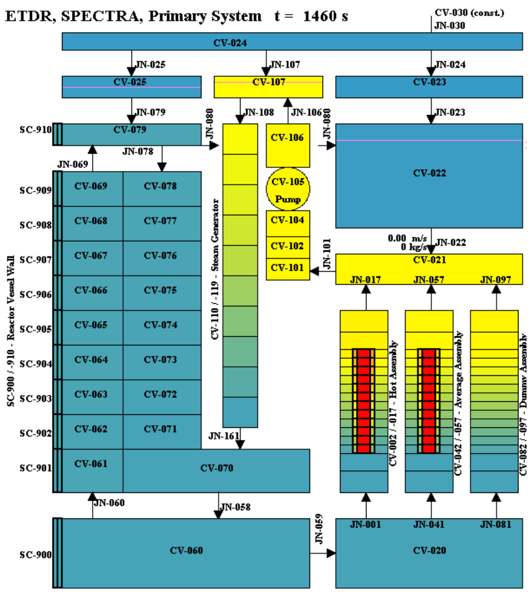

Fig. 28

Dodewaard (BWR Stability)

GKN Dodewaard was a unique reactor, operated in the Netherlands until 1997. This was the only operating BWR with natural circulation. An unstable behavior of the reactor cooling system was observed during one cycle. In the past, NRG performed an analysis of the Dodewaard stability using TRAC-NEM codes. Currently NRG is proposing to perform a code benchmark using the old data. At first this will be an internal benchmark involving computer codes TRACE and SPECTRA. As a next step an international benchmark may be proposed, similar to the Oskarshamn benchmark. The proposal is still to be accepted.

Examples of Past Applications

SWR-1000 (Advanced BWR)

SWR-1000 is a 1000 MW_e advanced Boiling Water Reactor designed by Siemens. Safety analyses of SWR-1000 were performed using the thermal-hydraulic code WAVCO. NRG performed an independent verification of the safety analysis using the SPECTRA code. The work was performed within the TEMPEST project. A SPECTRA model of the SWR-1000 reactor, the containment and safety features (Fig. 29) was prepared and tested. Results of steady state calculations were in agreement with available design data for the nominal operating conditions. Severe accident initiated by a stuck open safety valve, with simultaneous failure to open all valves on the core flooding lines was analyzed. Several runs were performed, investigating the influence of hydrogen stratification on containment pressure, 1-D versus 2-D modelling of reactor vessel.

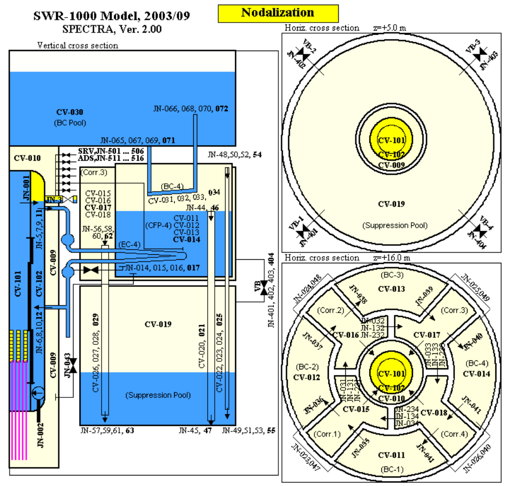

Fig. 29

PBMR (HTR)

PBMR is a South African design helium-cooled, direct cycle (Brayton) High Temperature Reactor. The design-support calculations were performed by PBMR Pty Ltd. using FLOWNEX code. PBMR contracted NRG for independent assessment of the thermal hydraulic calculations performed by FLOWNEX. SPECTRA has been selected as the primary tool for the verification and validation of the PBMR thermal hydraulic analyses. Calculations were performed for the PBMR design versions 5.02, 7.04 (Fig. 30), S201. Good agreement with FLOWNEX results were obtained in most cases. Differences were investigated in detail.

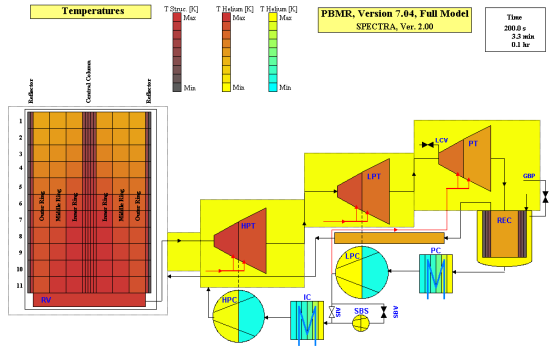

Fig. 30

Design-support calculations of the PBMR Reactor Cavity Cooling System (RCCS) (Fig. 31) were performed by PBMR Pty Ltd. using the RELAP5 code. PBMR contracted NRG for independent assessment of the thermal hydraulic calculations performed by RELAP5. SPECTRA has been selected as the primary tool for the verification and validation of the RCCS thermal hydraulic analyses, including normal operation and selected accident scenarios. Good agreement with FLOWNEX results were obtained.

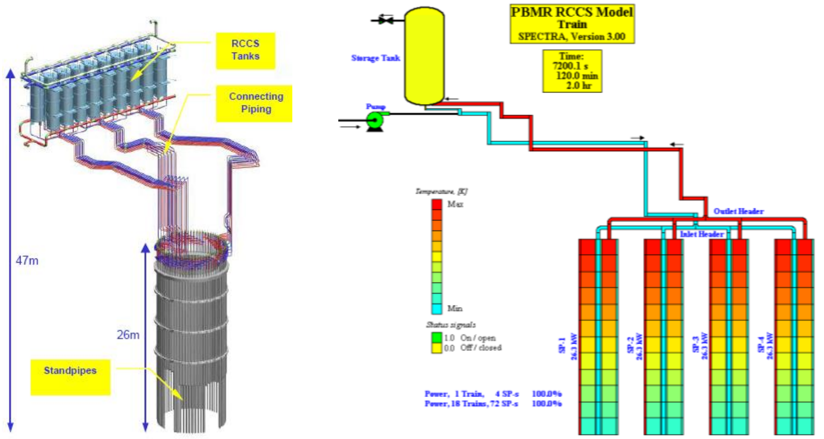

Fig. 31

PBMR Pty Ltd contracted NRG to perform dust analyses, including long-term (plant life time) deposition of dust. Calculations were performed for the PBMR main components: reactor, turbine [Stem-Wessels, 2014], recuperator plates (Fig. 32), coolers, as well as several sub-systems: Core Conditioning System (CCS), Core Barrel Conditioning System (CBCS), Fuel Handling System (FHS).

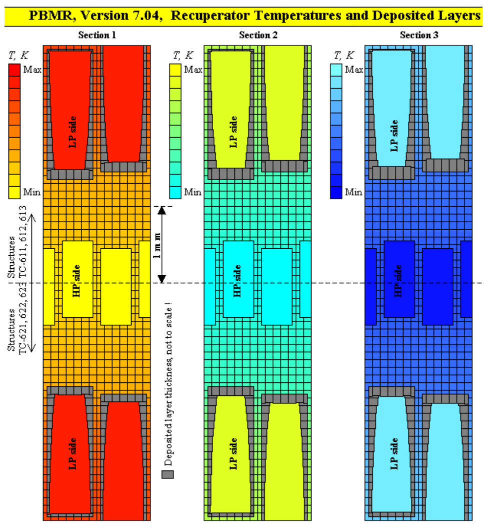

Fig. 32

HTR-PM

INET has a major program on High Temperature Gas Reactors. It has been operating the HTR-10 test reactor successfully for several years now and has finished construction of the HTR-PM demonstration reactor. In addition, large research projects are underway on several HTR subjects, amongst others fuel, thermal-hydraulics and graphite dust. INET contracted NRG to build SPECTRA model (Fig. 33) and perform analysis of dust and fission product behavior during postulated accident scenarios. The contract deliverables consisted of six parts:

- Model description and steady state results
- Analysis of PLOFC, DLOFC
- Dust deposition during plant life-time
- Dust behavior during operational transients
- Dust behavior during accidents
- Fission product behavior during accidents

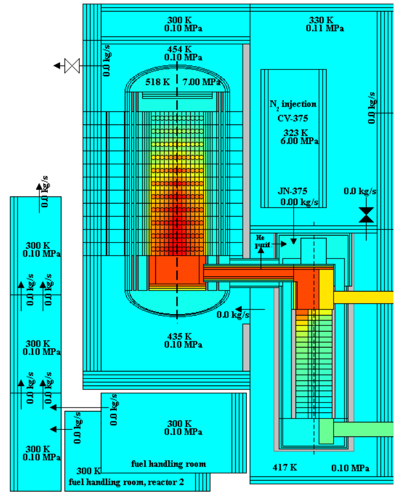

Fig. 33

Chemical Reactors (Shell)

Shell was designing chemical plants involving a number of large multi-tubular reactors in which the so-called Heavy Paraffin Synthesis (HPS) is performed. HPS was an up-scaled version of an earlier design, called PEARL. The cooling system of these reactors involves natural circulation boing water. Shell contracted NRG to perform an independent verification of the design for both PEARL and HPS reactors at various operating conditions (load levels). NRG performed the analyses using two different codes: RELAP5 and SPECTRA (Fig. 34). Very similar results were obtained with both codes.

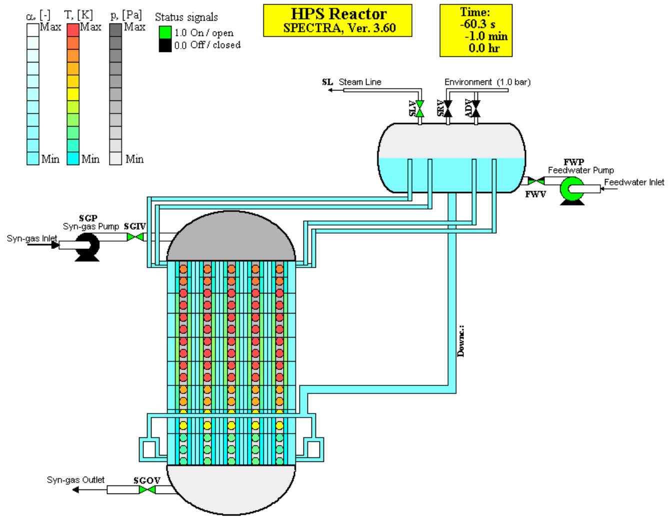

Fig. 34

Interactive Simulators

BWR

A simple BWR interactive simulator (Fig. 35) was created at NRG as a part of a training course for new members of the thermal-hydraulic analysis team, users of computer codes RELAP, MELCOR, SPECTRA.

- Plant: a hypothetical BWR-type reactor.
- Scenario: break at an unknown location of the primary system + simultaneous loss of grid power for one hour.
- Objectives:
  - Prevent any core damage by keeping the core covered using available pumps and emergency (battery) power.
  - Prevent containment venting and failure by using spray if necessary.

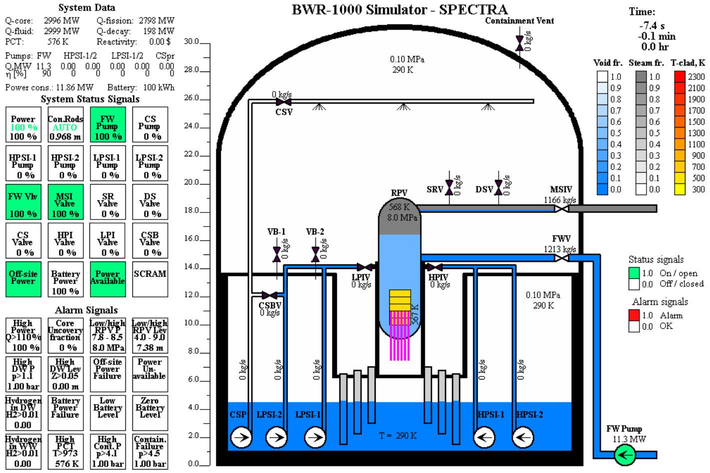

Fig. 35

PWR

PWR interactive simulator (Fig. 36, 37) was created for a course of reactor physics students "Cursus Kerntechniek".

- Plant: typical PWR, 3000 MW$_{th}$, 2 cooling loops.
- Scenario: primary system leakage in an unknown location + loss of grid power + loss of diesel generators
- Objectives:
  - Prevent release of activity to atmosphere.
  - Guarantee core cooling for as long as possible.

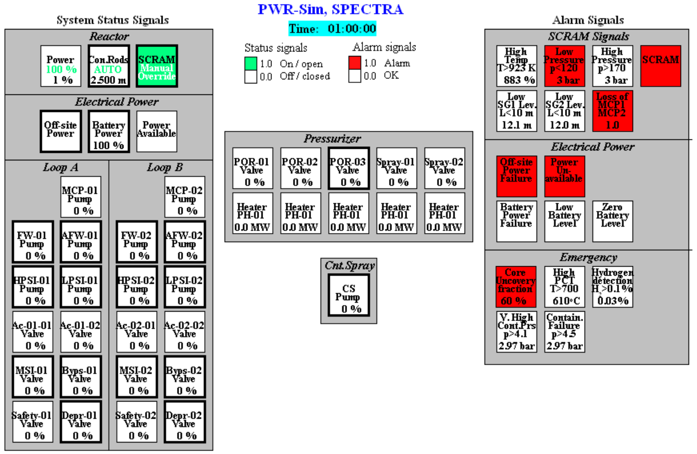

Fig. 36

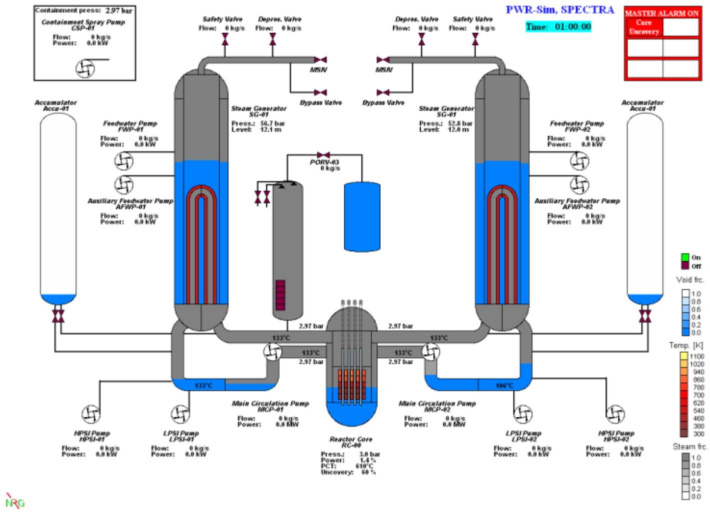

Fig. 37

LFR

An interactive simulator of LFR (Fig. 38, 39) was created for training of HFR operators.

- Plant: Low Flux Reactor operated in Petten (Netherlands) in the period 1960 - 2010.
- Model: full replica of the LFR control room and all its functionalities.
- Objectives: simulation of training exercises performed in the past on the LFR, e.g.:
  - Approach to criticality
  - Reactivity excess measurement
  - Estimation of control rod reactivity worth, etc.

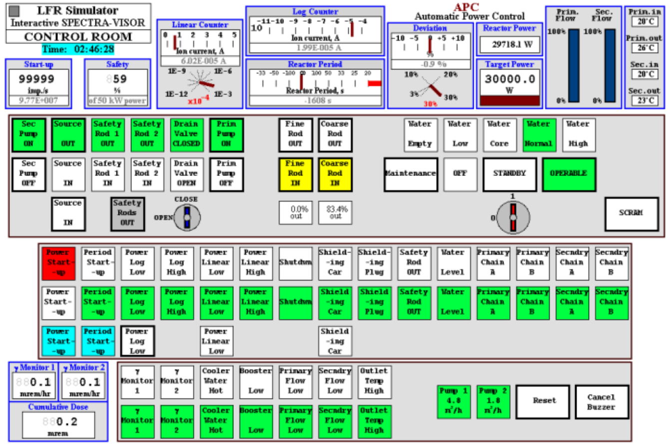

Fig. 38

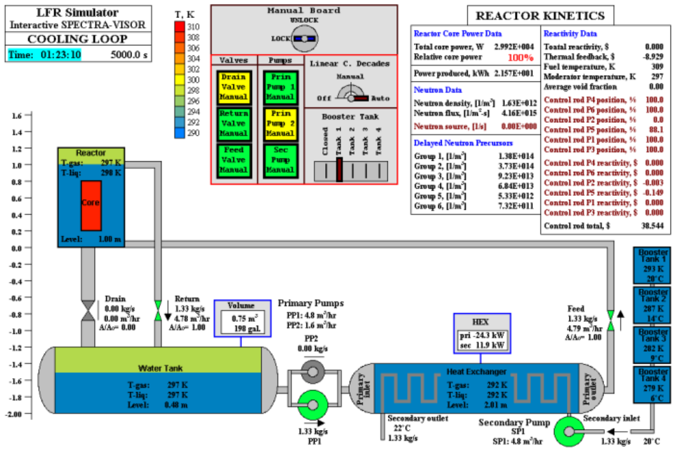

Fig. 39

Applicability

- **Light Water Reactors** (PWR, BWR). Past applications:
  - Accident analyses of General Electric SBWR design
  - Accident analyses for the Siemens design SWR-1000
- **High Temperature Reactors** (HTR, PBMR). Past applications:
  - South African PBMR design support analyses, dust transport analyses, RCCS analyses.
  - Chinese HTR-PM reactor accident (LOCA, LOFC) analyses, dust and fission product transport and release analyses.
- **Liquid Metal Fast Reactors** (LMFR). Past applications:
  - EBR-II shutdown heat removal tests (IAEA CRP).
  - Benchmarks on the new sodium-cooled reactor designs within ELSY, ESFR, ASTRID projects
  - Benchmarks on the new lead-cooled reactor within LEADER project
- **Molten Salt Reactors** (MSR). Past applications:
  - MSRE analyses of design operating conditions, fission product behavior, including delayed neutron precursor behavior.
  - Design-support analyses of the experimental molten salt loop, designed at NRG.
- **Conventional Power Plants, Chemical Reactors** (CPP, CR). Past applications:
  - Design-support analyses of the SHELL design chemical reactors with natural convection cooling system.

Documentation

- **General Overview Documents,** related to individual applicability areas, LWR, HTR, LMFR, MSR, etc. (Fig. 40)
- **SPECTRA Code Manuals**, containing the most detailed information on the modelling, the input requirements, and the V&V status.
  - *Volume 1: Model Description*
  - *Volume 2: User's Guide*
  - *Volume 3: Verification and Validation*
  - *Volume 4: Subroutine Description*
- **Reports and publications** from analyses performed with the code.

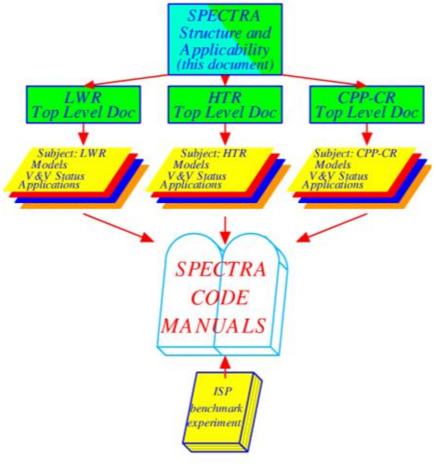

Fig. 40

References

[Aksan, 2010]
N. Aksan, “Overview of the PANDA Test Facility and ISP-42 PANDA Tests Data Base”, IAEA Training Course on Natural Circulation in Water-Cooled Nuclear Power plants, Trieste, Italy, 17- 21 of May, 2010.
[Bandini, 2013]
G. Bandini, "LEADER Project - Task 5.5 Analysis of representative DEC events of the ETDR", DEL022: Work Schedule, LEADER 5th WP5 Meeting, Petten, February 2013, (https://www.slideserve.com/wilma-mathis/leader-project-task-5-5)
[Briggs, et al., 2017]
Briggs, et al., “Benchmark Analyses of EBR-II Shutdown Heat Removal Tests”, IAEA-TECDOC-1819; (ISBN:978-92-0-105517-0); Vienna 2017.
[Bubelis et al., 2017]
Bubelis et al., “System codes benchmarking on a low sodium void effect SFR heterogeneous core under ULOF conditions”, Nuclear Engineering and Design 320 (2017) 325–345.
[Coindreau et al., 2017]
Coindreau et al., “Severe accident code to code comparison for two accident scenarios in a spent fuel pool”, The 8th European Review Meeting on Severe Accident Research – ERMSAR 2017, Warsaw, Poland, 16-18 May 2017.
[Holler et al, 2016]
T. Holler, E.M.J. Komen, I. Kljenak, “Pre-Test Simulations of Hydrogen Deflagration Experiment in HYMIT Experimental Facility”, IJS Report, IJS-DP-12192, Revision 0, October 2016
[Kelm et al., 2014]
S. Kelm, et al., “Generic Containment: Detailed comparison of containment simulations performed on plant scale”, Annals of Nuclear Energy, Volume 74, December 2014, Pages 165-172
[Komen-Stem, 2010]
M.M. Stempniewicz, E.M.J. Komen, “Comparison of Several Resuspension Models Against Measured Data”, Nuclear Engineering and Design, 240 (2010) 1657–1670.
[Lazaro et al., 2014]
A. Lazaro et al., “Code assessment and modelling for Design Basis Accident Analysis of the European Sodium Fast Reactor Design Part I: System description, modelling and benchmarking”, Nuclear Engineering and Design, Volume 266, January 2014, Pages 1-16.
[Mullins, et al., 1982]
C.B. Mullins, et al., “ORNL Rod Bundle Heat Transfer Test Data”, UREG/CR-2525, Vol. 3, ORNL/NUREG/TM-407/V3, 1982.
[Rousseau et al., 2014]
P.G. Rousseau, C.G. du Toit, W. van Antwerpen, H.J. van Antwerpen, “Separate effects tests to determine the effective thermal conductivity in the PBMR HTTU test facility”, Nuclear Engineering and Design 271 (2014) 444–458.
[Spectra, 2021]
M.M. Stempniewicz, “SPECTRA Computer Code Manuals, Volume 1 – Program Description, Volume 2 – User’s Guide, Volume 3 – Verification and Validation, Volume 4 - Code Structure, Development, Hardware and Software Requirements”,
https://dyfuzjastudio.com/marekstempniewicz/files/Spectra-Vol1.pdf
https://dyfuzjastudio.com/marekstempniewicz/files/Spectra-Vol2.pdf
https://dyfuzjastudio.com/marekstempniewicz/files/Spectra-Vol3.pdf
https://dyfuzjastudio.com/marekstempniewicz/files/Spectra-Vol4.pdf
[Stem, 2000]
M.M. Stempniewicz, “Analysis of PANDA Passive Containment Cooling Steady-State Tests with the Spectra Code”. Nuclear Technology, Vol. 131, No. 1, pp. 82-101, July 2000.
[Stem, 2013]
M.M. Stempniewicz, “LEADER, Task 5.5 ETDR Transient Analyses with the SPECTRA Code”, LEADER Project meeting, JRC, Petten, February 26, 2013
[Stem, 2014]
M.M. Stempniewicz, “Correlation for Steam-Graphite Reaction”, Nuclear Engineering and Design Vol. 280, pp. 285-293, December 2014.
[Stem, 2016]
M.M. Stempniewicz, “Air oxidation of Zircaloy, Part 2: New model for Zry-4 oxidation”, Nuclear Engineering and Design 301 (2016) 412–422.
[Stem et al., 2016]
M.M. Stempniewicz, M.L.F. Slootman, H.T. Wiersema, “Validation of system codes RELAP5 and SPECTRA for natural convection boiling in narrow channels”, Nuclear Engineering and Design 307 (2016) 130–143.
[Stem et al., 2017a]
M.M. Stempniewicz, E.A.R. de Geus, B. Mikuž, F. Roelofs, “A New Model to Simulate Molten Fuel Systems with the SPECTRA Code”, NURETH-17 (paper 21065), Xi’an, China, 3-8 September 2017.
[Stem et al., 2017b]
M.M. Stempniewicz, P.A. Breijder, H.J. Doolaard, F. Roelofs, “Multi-Scale Thermal Hydraulic Analysis of the EBR-II Loss of Flow Tests SHRT-17 And SHRT-45R”, NURETH-17 (paper 21062), Xi’an, China, 3-8 September 2017.
[Stem-Wessels, 2014]
M.M. Stempniewicz, D. Wessels, “Analysis of Fission Products in PBMR Turbine”, Proceedings of the HTR 2014 Weihai, China, October 27-31, 2014, Paper HTR2014-61288.
[Stem et al., 2018]
M.M. Stempniewicz, E.A.R. de Geus, F. Roelofs, "NRG analysis of SFR heterogeneous core under ULOF conditions", Nuclear Engineering and Design 339 (2018) 65–74.
[Wichers, 2003]
V. Wichers, “TEMPEST, Testing and Enhanced Modeling of Passive Evolutionary Systems Technology for Containment Cooling”, TEMPEST presentation, FISA-2003, Luxembourg, 10-13 November 2003.
[Zheng-Stem, 2012]
Zheng Y., Stempniewicz, M.M., “Investigation of NACOK Air Ingress Experiment Using Different System Analysis Codes”, Nuclear Engineering and Design, 251 (2012) 423– 432.
[Zheng et al. 2018]
Zheng Yanhua, Marek M. Stempniewicz, Chen Zhipeng, Shi Lei, “Study on the DLOFC and PLOFC accidents of the 200 MWe pebble-bed modular high temperature gas-cooled reactor with TINTE and SPECTRA codes”, Annals of Nuclear Energy 120 (2018) 763–777.