Past Projects & Experience in

Thermal
Hydraulic
Analysis

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

System Thermal-Hydraulic Codes and Severe Accident Codes

The following system thermal-hydraulic (**STH**) and severe accident (**SA**) codes were used:

- **MAAP**. The Modular Accident Analysis Program (MAAP) is a fast-running computer code that simulates the response of light water and heavy water moderated nuclear power plants for both current and Advanced Light Water Reactor (ALWR) designs. It can simulate Loss-Of-Coolant Accident (LOCA) and non-LOCA transients for Probabilistic Risk Analysis (PRA) applications as well as severe accident sequences, including actions taken as part of the Severe Accident Management Guidelines (SAMGs). There are several parallel versions of MAAP4 for BWRs, PWRs, CANDU designs, FUGEN design and the Russian VVER PWR design.

- **MELCOR**. The US NRC severe accident code. It is a fully integrated, engineering-level computer code whose primary purpose is to model the progression of accidents in light water reactor nuclear power plants. One basic model suffices for representing either a boiling water reactor (BWR) or a pressurized water reactor (PWR) core, and a wide range of levels of modeling detail is possible. MELCOR has been successfully used to model East European reactor designs, such VVER and RMBK-reactor classes.

- **RELAP5**. The US NRC thermal-hydraulic code. RELAP5 has been developed for best-estimate transient simulation of light water reactor coolant systems during postulated accidents. The code models the coupled behavior of the reactor coolant system and the core for loss-of-coolant accidents and operational transients such as anticipated transient without scram, loss of offsite power, loss of feedwater, and loss of flow.

- **SPECTRA**. The NRG thermal-hydraulic 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 thermal accident scenarios, involving loss-of-coolant accidents (LOCAs), operational transients, and other accident scenarios in nuclear power plants.

- **TRACE/PARCS**. The US NRC thermal-hydraulic codes. TRACE is a modernized thermal-hydraulics code designed to consolidate and extend the capabilities of NRC's 3 legacy safety codes - TRAC-P, TRAC-B and RELAP. It is able to analyze large/small break LOCAs and system transients in both pressurized- and boiling-water reactors (PWRs and BWRs). This is the NRC's flagship thermal-hydraulics analysis tool. PARCS is a 3-D neutronics code, which may be linked with TRACE.

Areas of Expertise

PWR
PWR-related work include participation in International Standard Problems (ISP), research projects, as well as work under contracts for the Dutch PWR Borssele (KCB) and foreign ATMEA, TRACTEBEL.

BWR
BWR-related work include participation in International Standard Problems, research projects, as well as work under contracts for the Dutch PWR Dodewaard, German advanced BWRs: SWR-1000, KERENA.

HTR
HTR-related work include participation in International Standard Problems, research projects, as well as work under contracts for the South African PBMR and Chinese HTR-PM.

LMFR
LMFR-related work include participation in international code-to-code benchmarks on new designs of liquid metal cooled reactors, performed within several EU projects.

MSR
MSR-related work include code validation based on existing data from the MSRE reactor operated at ORNL, and design-support analyses of a sub-critical molten salt loop in Petten and new MSR concept within Thorizon.

Research Reactors
Work done in the area of research reactors include code validation as well as safety analyses performed under contract for HFR.

Chemical Reactors
Work done in the area of chemical reactors include design-support analyses of cooling systems of several chemical reactor designs under contract with Shell.

Interactive Simulators
Several interactive simulators were created for the purpose of training for reactor physics students, reactor operators, etc.

Pressurized Water Reactors

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. KEMA Nuclear (the Netherlands) 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.

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. 1).

- 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].

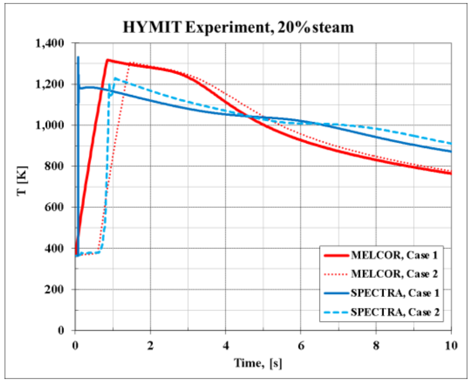


Fig. 1

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

KCB Borssele SB LOCA Analyses

SB-LOCA analysis have been performed for the Borssele NPP. The Borssele NPP is a 485 MWe, two-loop PWR operated by the Dutch utility EPZ. The analysis were part of the 10-yearly safety review. The results were submitted to the Licensing Authorities. A TRAC-P model (Fig. 2) has been developed and analysis were performed for a wide range of break sizes with both best-estimate and bounding conditions. In addition a comprehensive overview and assessment of the loop-seal clearance issue for the Borssele NPP has been provided.

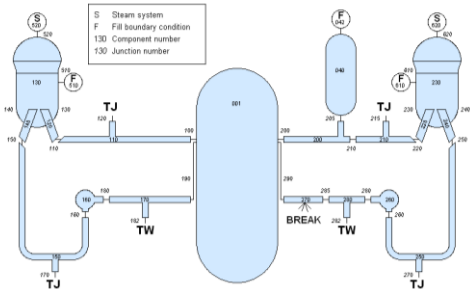


Fig. 2

KCB Borssele PSA Analyses

Severe accident analyses for the Borssele PWR were performed for the PSA level 2. The project was a part of an extensive update of the entire PSA for Borssele. NRG (Netherlands) prepared a complete input deck for the calculations with the Severe Accident code MELCOR 2.1 (Fig. 3). In total approximately 50 scenarios have been calculated, including source term calculations. Several plant damage states were analyzed. Both power and shutdown states were assessed. A spent fuel pool accident scenario was included.

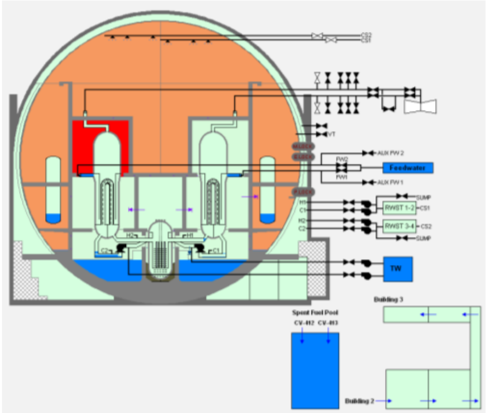


Fig. 3

KCB Hydrogen Analyses

Hydrogen distribution and flammability was studied for three selected scenarios using the MELCOR code in order to assess the performance of the hydrogen recombiners in the containment (Fig. 4). The results of the calculation(s) were used for detailed CFD-analyses.

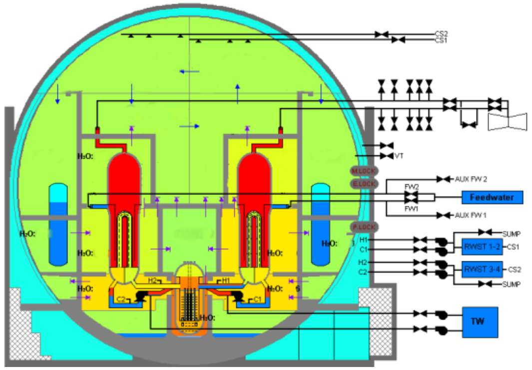


Fig. 4

TRACTEBEL

TRACTEBEL contracted NRG to help with validation calculations of a new version of RELAP5 against a number of experimental facilities. M. Stempniewicz joined temporarily the RELAP5 team at TRACTEBEL and performed analyses of ROSA-LSTF (ISP26), and ISP27-Bethsy (ISP-27) and Thermal Hydraulic Test Facility. Results were submitted to the Licensing Authorities.

Westinghouse SMR

The Westinghouse Small Modular Reactor (SMR) is an 800 MW_{th} integral pressurized water reactor (Fig. 5). Its design does not require large loop piping, with the benefit of eliminating the occurrence of large break loss of coolant accidents (LOCA). The Westinghouse SMR achieves a high level of safety by relying on passive safety systems which utilize gravity, natural circulation, passive heat sinks, and stored potential energy. NRG performed a full-scope analysis of the Westinghouse SMR response to a design basis accident scenario involving a Direct Vessel Injection (DVI) line break. Analyses were performed with the thermal-hydraulic system codes: SPECTRA, MELCOR, and RELAP5 and compared to the results obtained by Westinghouse WCOBRA/TRAC-TF2 code system. The results show the safety behavior of the plant in the unlikely event of the postulated DVI break.

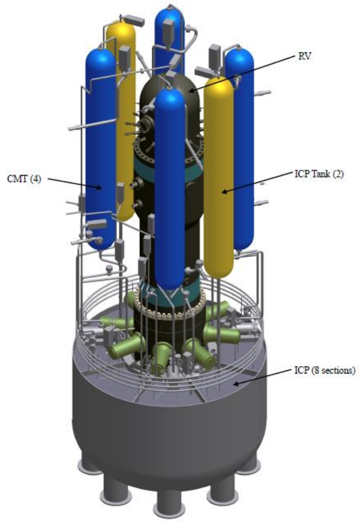


Fig. 5

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 SARNET2 project such a Generic Containment nodalization (Fig. 6), based on a German PWR (1300 MW_e), was defined. The benchmark consisted of 3 steps, with increasing complexity:

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

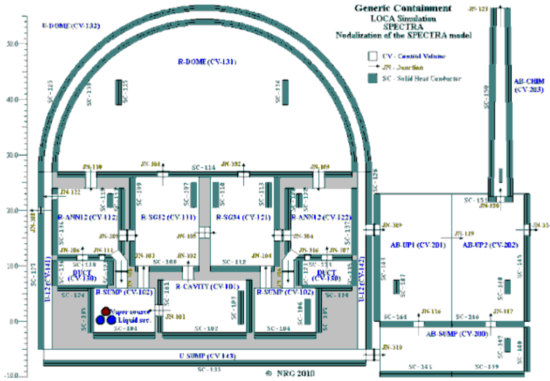


Fig. 6

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 Daiichi 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 SA codes (ASTEC, MELCOR, ATHLET-CD, RELAP/SCDAPSIM, ICARE/CATHARE, SPECTRA) [Coindreau et al., 2017]. The code to code comparison 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. A new oxidation breakaway model was developed (Fig. 7) and implemented into SPECTRA. The model was checked against experiments and results of other codes (Fig. 8). Results are described in [Stem, 2016a] and [Stem, 2016b].

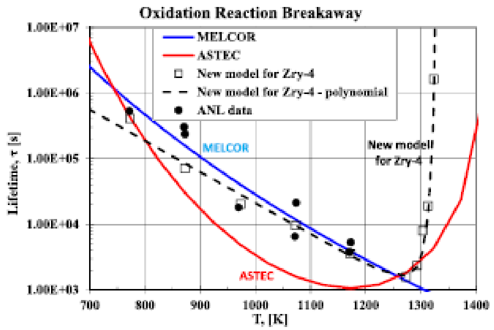


Fig. 7

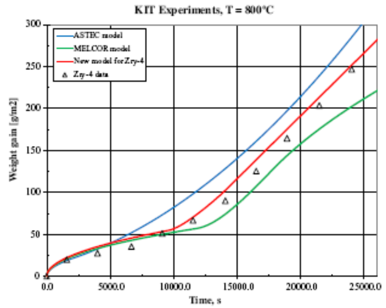


Fig. 8

Analysis of a Generic Spent Fuel Pool

NRG performed analyses of a generic Spent Fuel Pool with four system codes (MELCOR, RELAP, SPECTRA, TRACE) and one CFD code (CFX). Conclusions:

- Modeling of SFP in CFD codes is difficult because of phenomena like air oxidation and decay heat generation. Appropriate models are readily available in the STH codes.
- Modeling of SFP in STH codes is difficult because 3-D effects and phenomena like Coanda or Rayleigh-Bernard effect. Models are readily available in the CFD codes.

Boiling Water Reactors

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 WAVCO code. 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. 9) 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 wall and structures.

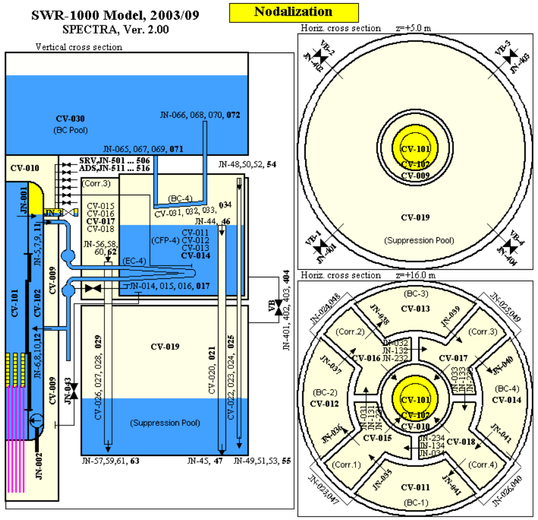
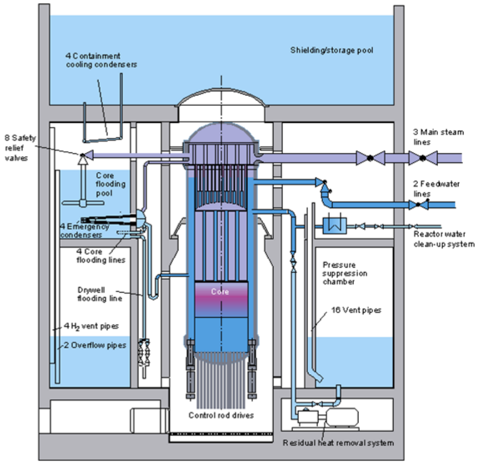


Fig. 9

KERENA (Up-scaled SWR-1000)

KERENA is a 1300 MW_e boiling water reactor design with a number of passive safety features (Fig. 10). Essentially it is an up-scaled version of SWR-1000, designed by AREVA-Siemens. Severe accident analyses for the AREVA designed KERENA were performed of specified accident scenarios for inclusion in a level-2 PSA. NRG prepared the MELCOR KERENA input deck and for performing the calculations. 19 scenarios were analyzed. Source term calculations and source term uncertainty analysis was performed. MELCOR version 1.8.6 was used. Both power and shutdown states were assessed.

Fig. 10



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. 11) 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".

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

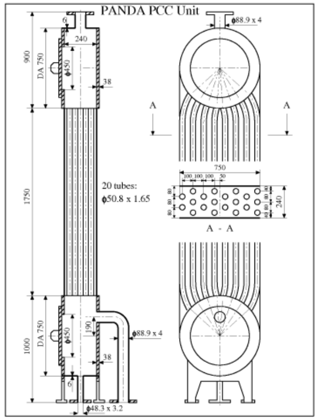


Fig. 11

Comparison of code calculations and measured data showed that all codes under-predicted the PCC efficiency (fraction of steam condensed) [Stem, 2000] (Fig. 12). To explain this, 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, 2017].

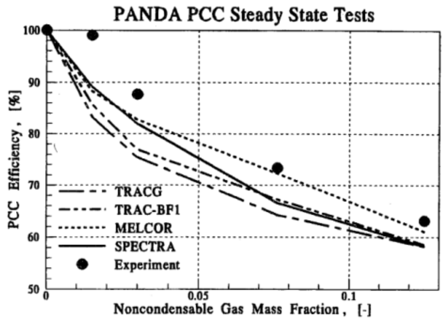


Fig. 12

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. 13) 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].

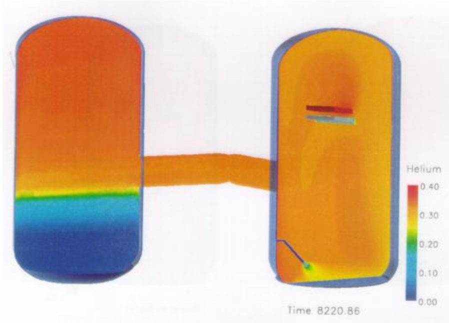


Fig. 13

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. 14). Stratification models applied in the open phase (Fig. 15), 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].

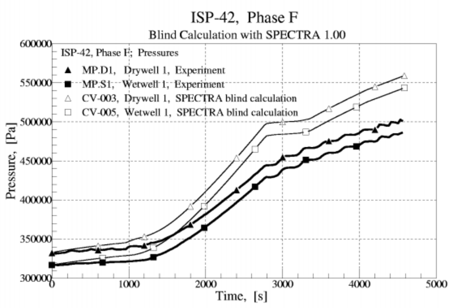


Fig. 14

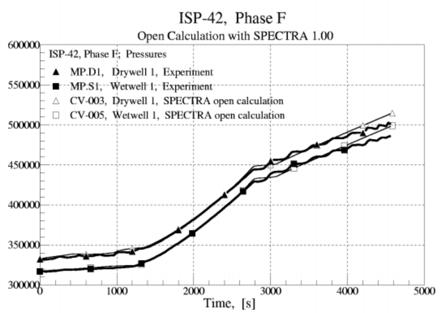


Fig. 15

Dodewaard (Natural convection BWR)

Severe accident analyses for the Dodewaard NPP were performed for the PSA level 2. NRG prepared a complete input deck and for the calculations with the Severe Accident code MELCOR 1.8.3 (Fig. 16). In total approximately 30 scenarios have been calculated, including source term calculations.

Furthermore, an analysis of High Pressure Melt Ejection (HPME) and Direct Containment Heating (DCH) were performed with MELCOR and Contain codes.

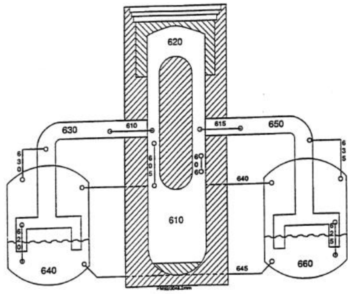


Fig. 16

Oskarshamn

On February 25, 1999, the Swedish Oskarshamn-2 BWR experienced a stability event. A combination of various occurrences culminated in diverging power oscillations, which triggered an automatic reactor scram at high power [Kozlowski et al., 2014].

NRG participated in the OECD/NEA Oskarshamn-2 (O2) BWR Stability Benchmark for Coupled Code Calculations. A model of the O2 plant was created with the TRACE/PARCS code system (Fig. 17). The results of the performed work (Fig. 18) indicate that it was not easy to accurately reproduce BWR instability transients with diverging power oscillations with the TRACE/PARCS code system. The results are very sensitive to the input parameters and boundary conditions (this will be the focus of the OECD/NEA UAM-LWR benchmark).

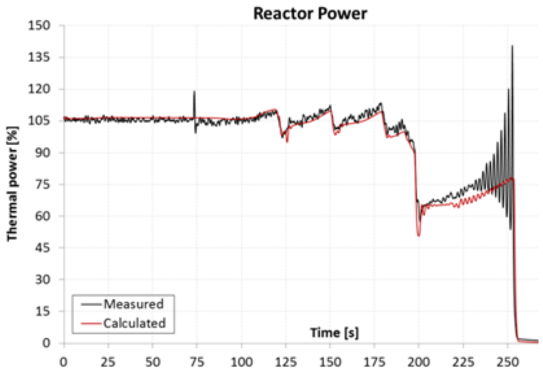


Fig. 17

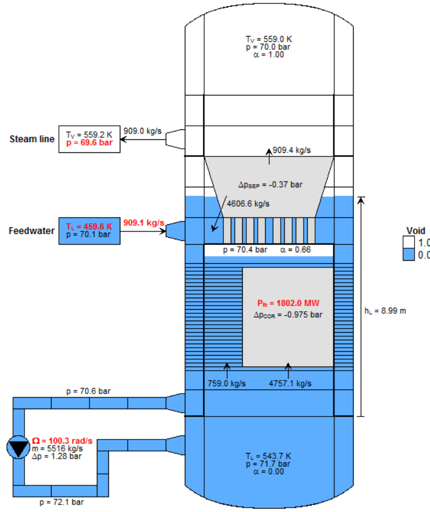


Fig. 18

High Temperature Reactors

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. 19) 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%

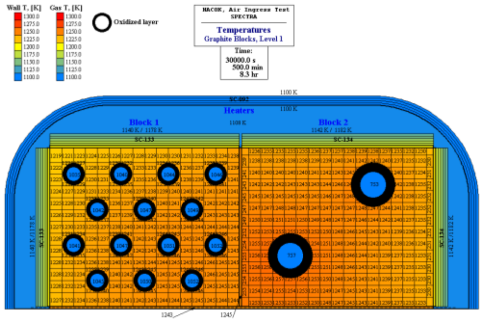


Fig. 19

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. Two well-known resuspension models (Veinstein, Rock'n Roll) and a new model, proposed by Komen and Stempniewicz [Komen-Stem, 2010] were investigated using the experimental data of Reeks and Hall experiments (Fig. 20) (monolayer deposit) [Komen-Stem, 2010].

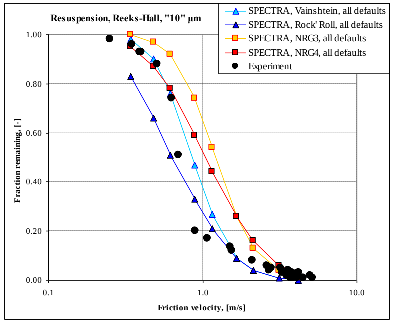


Fig. 20

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 (Fig. 21) (multi-layer deposit) [Komen-Stem, 2010].

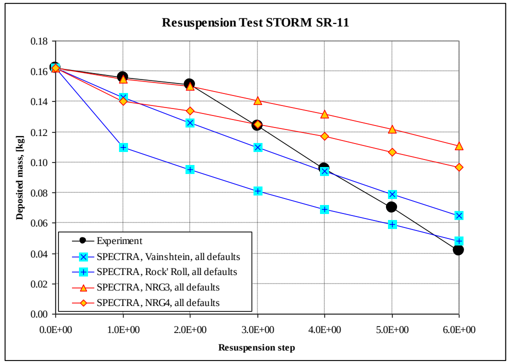


Fig. 21

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. 22), S201. Good agreement with FLOWNEX results was obtained in most cases. Differences were investigated in detail.

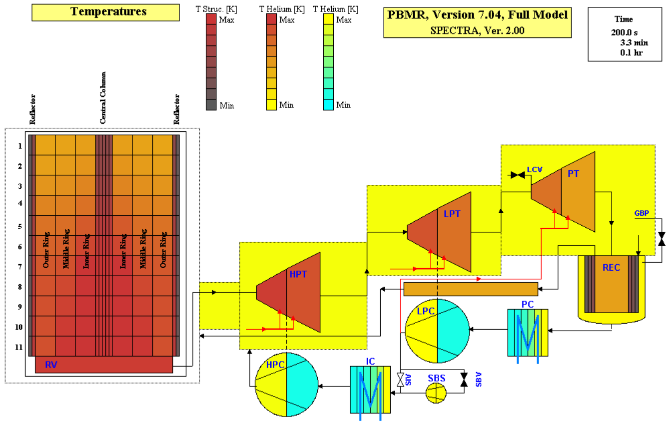


Fig. 22

Design-support calculations of the PBMR Reactor Cavity Cooling System (RCCS) (Fig. 23) 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 RELAP5 results was obtained.

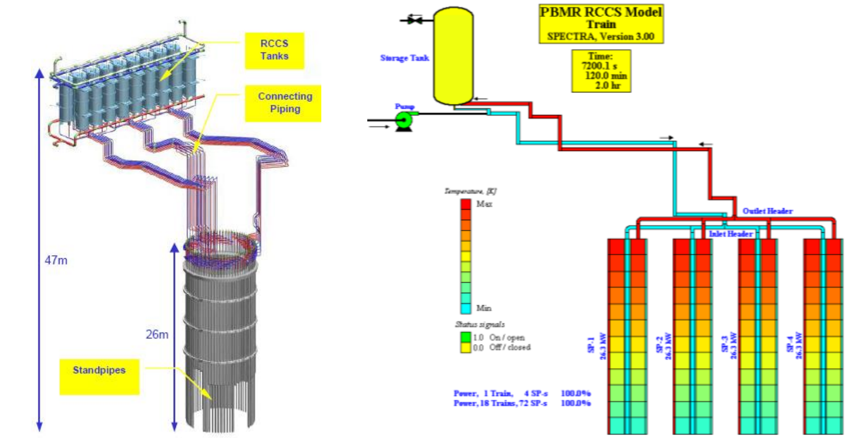


Fig 23

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. 24), coolers, as well as several sub-systems: Core Conditioning System (CCS), Core Barrel Conditioning System (CBCS), Fuel Handling System (FHS).

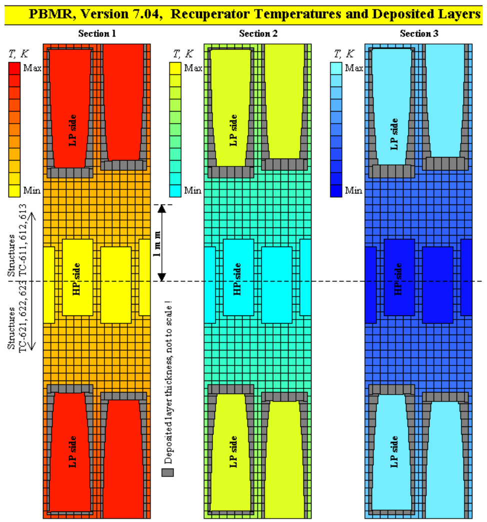


Fig. 24

HTR-PM

The Chinese research institute 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. 25) 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

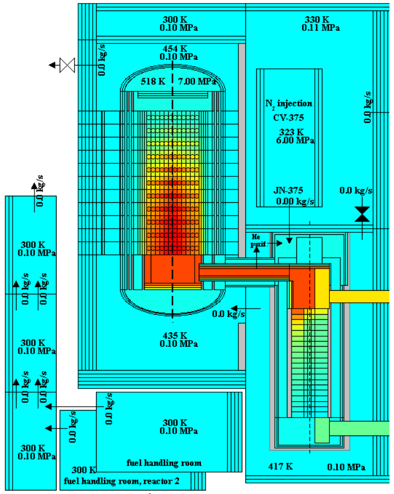


Fig. 25

GEMINI+

Within GEMINI+ a prismatic block HTR was designed. NRG performed DLOFC/PLOFC analyses, both protected and unprotected (Fig. 26) and air ingress scenario (Fig. 27).

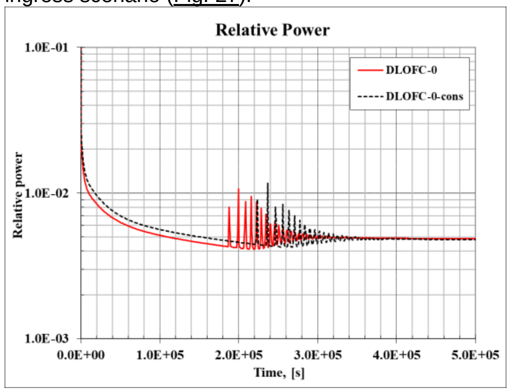


Fig. 26

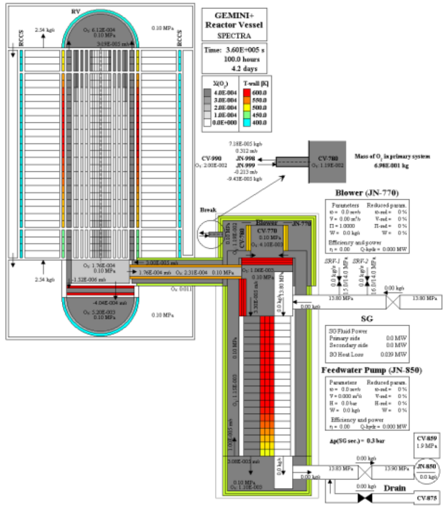


Fig. 27

Liquid Metal Fast Reactors

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. 28), the intermediate heat exchanger (IHX) primary inlet temperatures, IHX secondary outlet temperatures and primary coolant flow rates with measured data provided by Argonne National Laboratory (ANL), the SPECTRA stand-alone model and the multi-scale thermal hydraulic coupled SPECTRA/CFX model (Fig. 29, 30) proved to be able to provide satisfactory results for this benchmark [Stem, et al., 2017c].

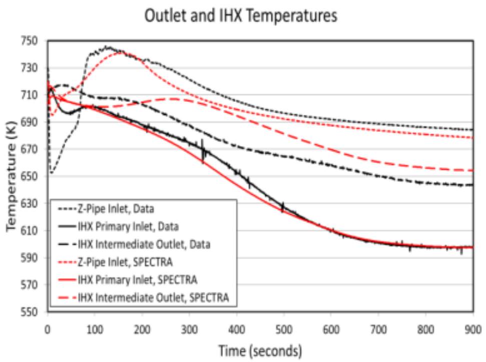


Fig. 28

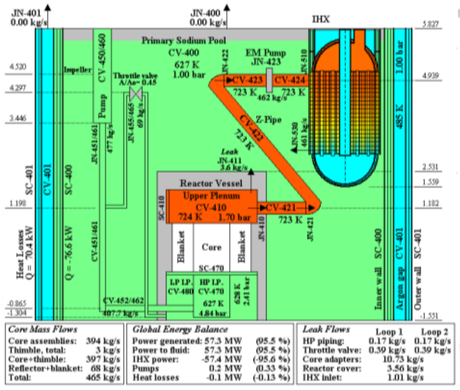


Fig. 29

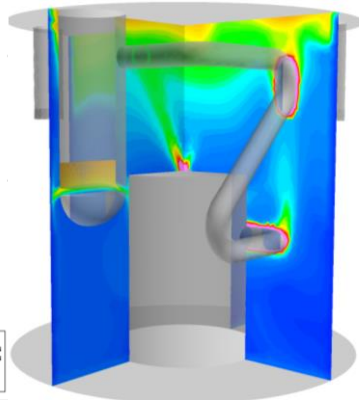


Fig. 30

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. 31, 32).

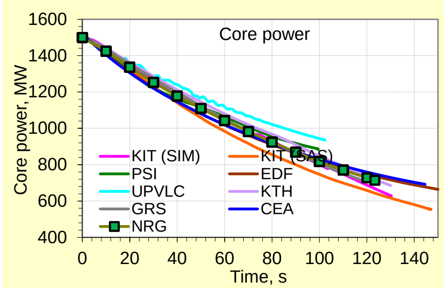


Fig. 31

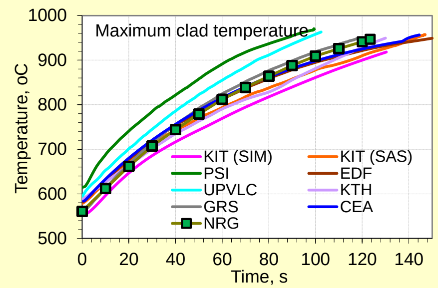


Fig. 32

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. 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 SPECTRA.

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. Eight accident scenarios were analyzed [Stem, 2013].

Molten Salt Reactors

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. 33). The steady state model parameters compared to available data showed good agreement [Stem et al., 2017a].

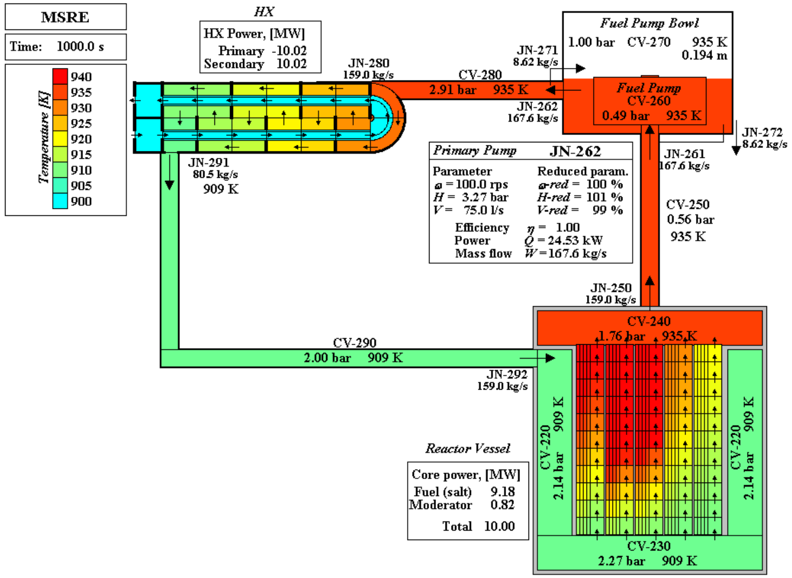


Fig. 33

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.

LUMOS

A molten salt demonstration loop is currently being designed at NRG, with the goal of constructing and operating the loop in the flux field directly next to the core of the High Flux Reactor. Design-support and safety analyses are being performed with the system thermal-hydraulic code SPECTRA (Fig. 34, 35). Results were presented at NURETH conference [Stem, 2017b].

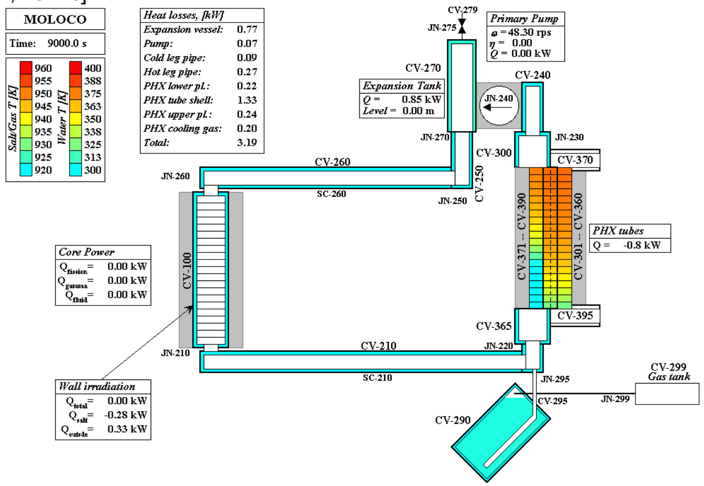


Fig. 34

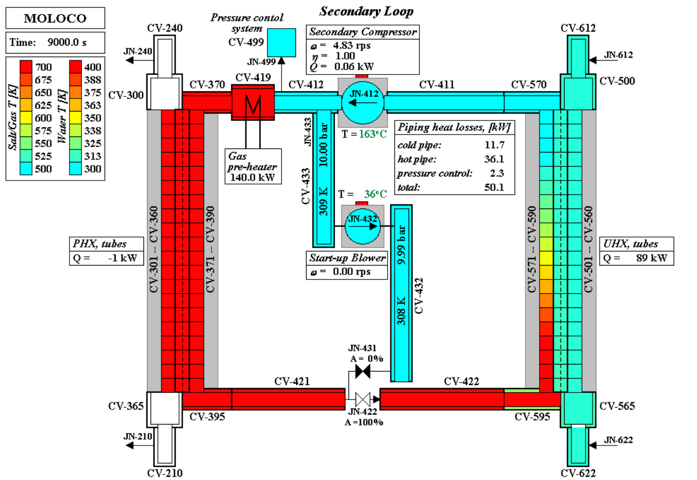


Fig. 35

Research Reactors

HFR (High Flux Reactor, Petten)

Non-LOCA and LOCA analysis have been performed for the High Flux Reactor in Petten. The reactor is a 50 MW^{th} tank in pool type research reactor (Fig. 36) operated by NRG. The analyses were part of the License Renewal Project. The results were accepted by the Licensing Authorities. Models have been developed and analyses have been performed with: RELAP5 and SPECTRA (thermal-hydraulics), COBRA-TF (DNBR and heat flux), MCNP (reactivity), MELCOR (severe accidents). In addition analyses for the Technical Specifications were performed.

The project was consisting of the following stages:

- Selection of Postulated Initiated Events (PIEs)
- Determining acceptance criteria
- Determining initial and boundary conditions
- Validation and verification of models
- Evaluating results of analyses
- Acceptance by authorities

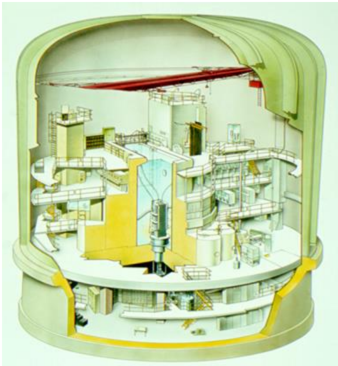


Fig. 36

Severe accident analyses were performed as part of the PSA Level-2. Models were developed and analyses performed with MELCOR (thermal-hydraulics and severe accident progression, Fig.) and MCNP/ORIGEN (radioisotope inventory of the core and the Spent Fuel Pool). Results of the Level-2 analyses were used in the PSA Level-3 (radiological consequences).

HOR (Hoger Onderwijs Reactor), Delft

The HOR reactor (Fig. 37) located at Delft University is a pool-type reactor that operates up to 3 MW_{th} using forced convection, or 750 kW during natural circulation. Safety analyses were performed as part of the license renewal project. A RELAP5 model was developed, including core, primary and secondary system, pool, and reactor protection and control systems. Both LOCA and non-LOCA scenarios (including reactivity transients and ATWS) were analyzed.

In addition, a MELCOR model of the HOR containment was developed for the PSA Level-2. A number of bounding severe accident scenarios were analyzed. Scoping analyses to support the PSA Level-1 modelling were also performed.

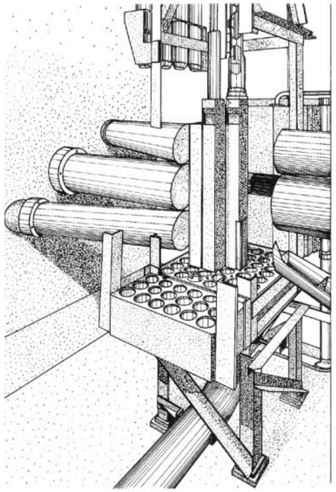


Fig. 37

Code Validation for Boiling in Narrow Channels

Research reactors fuel consists of fuel plates with small gaps (1 - 3 mm) between the plates. In case of loss of forced circulation boiling may occur in the gaps. The thermal-hydraulic codes that are used at NRG for research reactor safety analyses are validated for typical fuel geometries applied in the power reactors, but not for geometries encountered in the research reactors. 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. were used to validate the codes RELAP5 and SPECTRA for boiling in narrow channels. Models were built with SPECTRA and RELAP5 and both codes results were compared to measured data (Fig. 38, 39) [Stem et al., 2016].

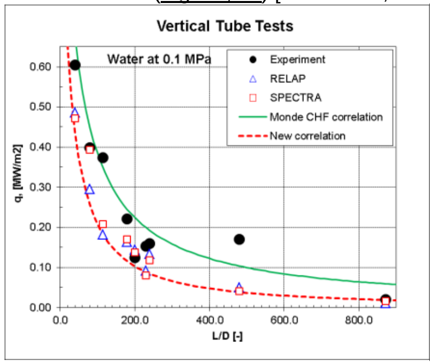


Fig. 38

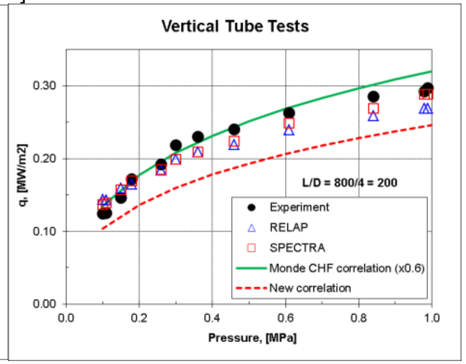


Fig. 39

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. 40). Very similar results were obtained with both codes.

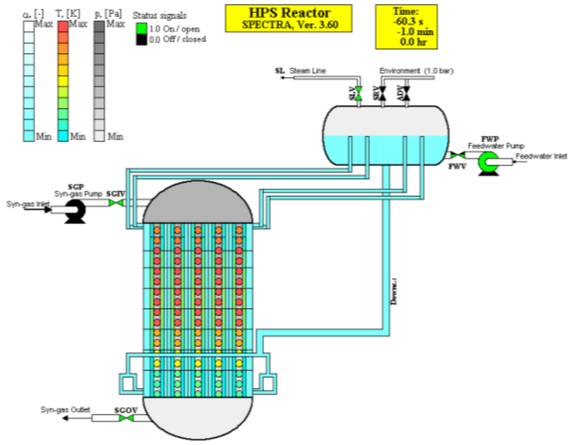


Fig. 40

Interactive Simulators

BWR Simulator (SPECTRA-VISOR)

A simple BWR interactive simulator (Fig. 41) 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.

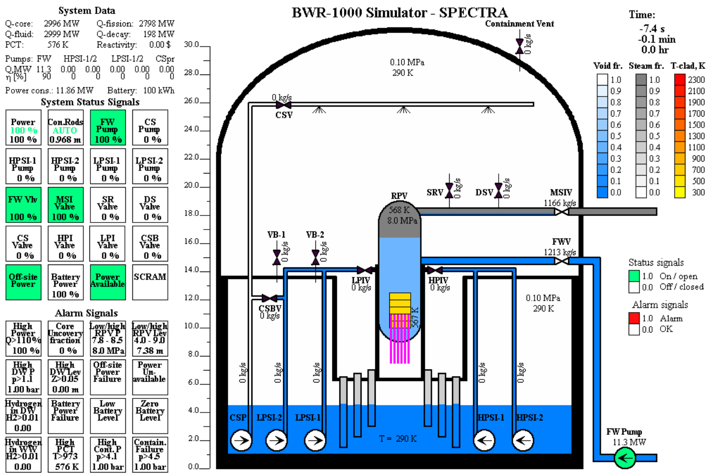


Fig. 41

PWR Simulator (SPECTRA-VISOR)

PWR interactive simulator (Fig. 42, 43)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.

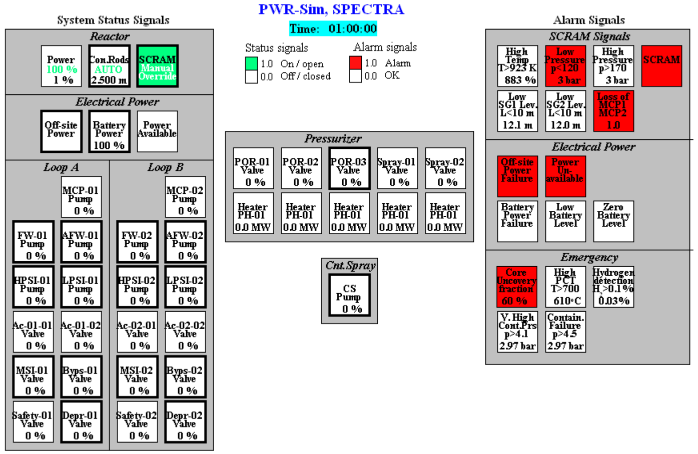


Fig. 42

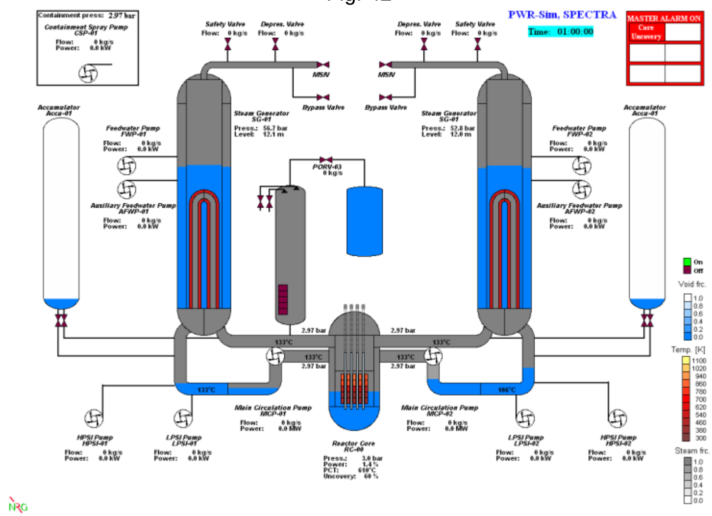


Fig. 43

KCB (PWR) Simulator (MELCOR-VISOR)

An interactive simulator of KCB Borssele nuclear power plant (2-loop PWR) was created as a training tool for the Dutch regulatory body (Fig. 44, 45). Eight different accident scenarios can be simulated where the user may initiate additional failures. Plant mimic screen shows animated liquids, atmosphere, pumps, valves, locks, sprays, control rods and temperature changes.

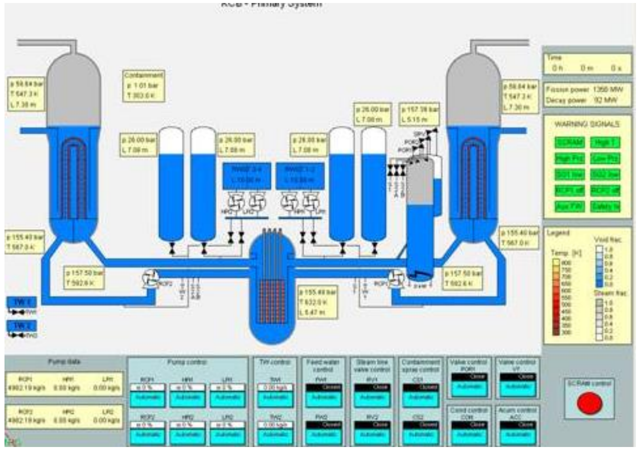


Fig. 44

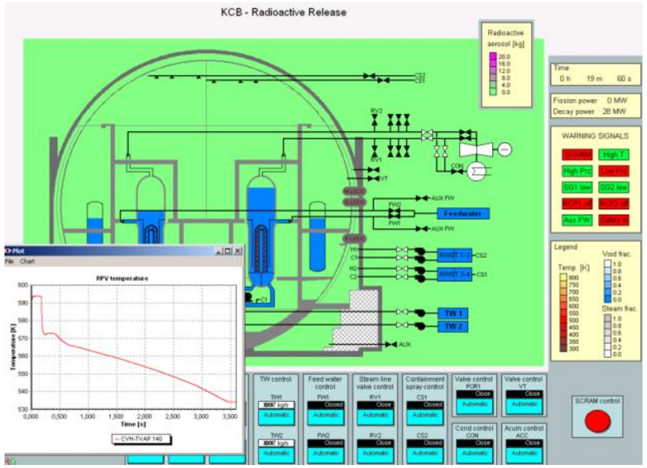


Fig. 45

LFR Simulator (SPECTRA-VISOR)

An interactive simulator of LFR (Fig. 46, 47) 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.

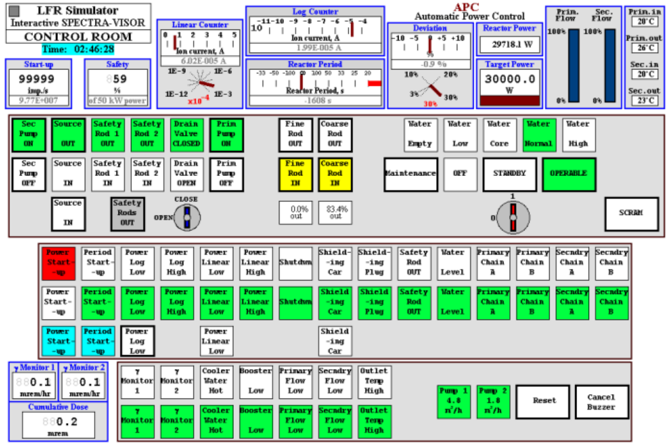


Fig. 46

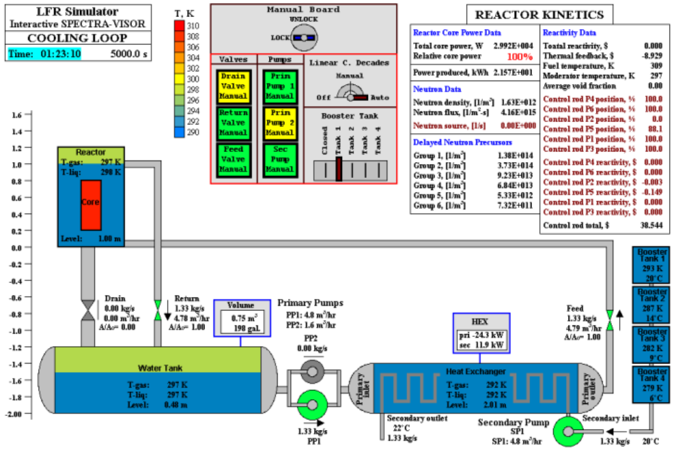


Fig. 47

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