IAEA
International Atomic Energy Agency

Integral Pressurized Water Reactor Simulator Manual

VIENNA, 2017

TRAINING COURSE SERIES 65

INTEGRAL PRESSURIZED WATER REACTOR SIMULATOR MANUAL

The following States are Members of the International Atomic Energy Agency:

AFGHANISTAN
ALBANIA
ALGERIA
ANGOLA
ANTIGUA AND BARBUDA
ARGENTINA
ARMENIA
AUSTRALIA
AUSTRIA
AZERBAIJAN
BAHAMAS
BAHRAIN
BANGLADESH
BARBADOS
BELARUS
BELGIUM
BELIZE
BENIN
BOLIVIA, PLURINATIONAL STATE OF
BOSNIA AND HERZEGOVINA
BOTSWANA
BRAZIL
BRUNEI DARUSSALAM
BULGARIA
BURKINA FASO
BURUNDI
CAMBODIA
CAMEROON
CANADA
CENTRAL AFRICAN REPUBLIC
CHAD
CHILE
CHINA
COLOMBIA
CONGO
COSTA RICA
CÔTE D'IVOIRE
CROATIA
CUBA
CYPRUS
CZECH REPUBLIC
DEMOCRATIC REPUBLIC OF THE CONGO
DENMARK
DJIBOUTI
DOMINICA
DOMINICAN REPUBLIC
ECUADOR
EGYPT
EL SALVADOR
ERITREA
ESTONIA
ETHIOPIA
FIJI
FINLAND
FRANCE
GABON
GEORGIA
GERMANY
GHANA
GREECE
GUATEMALA
GUYANA
HAITI
HOLY SEE
HONDURAS
HUNGARY
ICELAND
INDIA
INDONESIA
IRAN, ISLAMIC REPUBLIC OF
IRAQ
IRELAND
ISRAEL
ITALY
JAMAICA
JAPAN
JORDAN
KAZAKHSTAN
KENYA
KOREA, REPUBLIC OF
KUWAIT
KYRGYZSTAN
LAO PEOPLE'S DEMOCRATIC REPUBLIC
LATVIA
LEBANON
LESOTHO
LIBERIA
LIBYA
LIECHTENSTEIN
LITHUANIA
LUXEMBOURG
MADAGASCAR
MALAWI
MALAYSIA
MALI
MALTA
MARSHALL ISLANDS
MAURITANIA
MAURITIUS
MEXICO
MONACO
MONGOLIA
MONTENEGRO
MOROCCO
MOZAMBIQUE
MYANMAR
NAMIBIA
NEPAL
NETHERLANDS
NEW ZEALAND
NICARAGUA
NIGER
NIGERIA
NORWAY
OMAN
PAKISTAN
PALAU
PANAMA
PAPUA NEW GUINEA
PARAGUAY
PERU
PHILIPPINES
POLAND
PORTUGAL
QATAR
REPUBLIC OF MOLDOVA
ROMANIA
RUSSIAN FEDERATION
RWANDA
SAN MARINO
SAUDI ARABIA
SENEGAL
SERBIA
SEYCHELLES
SIERRA LEONE
SINGAPORE
SLOVAKIA
SLOVENIA
SOUTH AFRICA
SPAIN
SRI LANKA
SUDAN
SWAZILAND
SWEDEN
SWITZERLAND
SYRIAN ARAB REPUBLIC
TAJIKISTAN
THAILAND
THE FORMER YUGOSLAV REPUBLIC OF MACEDONIA
TOGO
TRINIDAD AND TOBAGO
TUNISIA
TURKEY
TURKMENISTAN
UGANDA
UKRAINE
UNITED ARAB EMIRATES
UNITED KINGDOM OF GREAT BRITAIN AND NORTHERN IRELAND
UNITED REPUBLIC OF TANZANIA
UNITED STATES OF AMERICA
URUGUAY
UZBEKISTAN
VANUATU
VENEZUELA, BOLIVARIAN REPUBLIC OF
VIET NAM
YEMEN
ZAMBIA
ZIMBABWE

The Agency's Statute was approved on 23 October 1956 by the Conference on the Statute of the IAEA held at United Nations Headquarters, New York; it entered into force on 29 July 1957. The Headquarters of the Agency are situated in Vienna. Its principal objective is "to accelerate and enlarge the contribution of atomic energy to peace, health and prosperity throughout the world''.

TRAINING COURSE SERIES No. 65

INTEGRAL PRESSURIZED WATER REACTOR SIMULATOR MANUAL

INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA, 2017

COPYRIGHT NOTICE

For further information on this publication, please contact:

Nuclear Power Technology Development Section
International Atomic Energy Agency
Vienna International Centre
PO Box 100
1400 Vienna, Austria
Email: Official.Mail@iaea.org

INTEGRAL PRESSURIZED WATER REACTOR SIMULATOR MANUAL
IAEA, VIENNA, 2017
IAEA-TCS-65
ISSN 1018–5518

Printed by the IAEA in Austria
June 2017

FOREWORD

The IAEA has established a programme in nuclear reactor simulation computer programs to assist Member States in educating and training nuclear professionals in the operation and behaviour of nuclear power plants. The simulators enable hands-on training for nuclear professionals involved in teaching and training topics such as nuclear power plant design, safety, technology, simulation and operations. The simpler design of the simulators allows professionals to grasp the fundamentals without becoming lost in the details of a more complex full scope simulator. The objective is to provide, for various reactor types, insight and practice in reactor operational characteristics and their response to perturbations and accident situations.

Together with management and development, the IAEA regularly produces training material and provides training courses to assist professionals in Member States to understand the simulators and their associated technologies. This training is more cost effective and less time consuming than alternative methods and is highly suitable for Member States with limited resources.

This publication provides detailed explanations of the theoretical concepts that the simulator users have to know to gain a comprehensive understanding of the physics and technology of integral pressurized water reactors. This publication provides explanations of each of the simulator screens and various controls that a user can monitor and modify. A complete description of all the simulator features is also provided. A detailed set of exercises is provided in the Exercise Handbook accompanying this publication. The publications can be used both independently and together, and they serve as educational and training material for IAEA training courses and workshops. This publication provides comprehensive reference material to support human capacity building in all Member States improving their expertise in education and training related activities for small modular reactor technology.

The IAEA gratefully acknowledges the contribution of I. Parrado (Spain) to the preparation of this publication. The IAEA officer responsible for this publication was C. Batra of the Division of Nuclear Power.

EDITORIAL NOTE

This publication has been prepared from the original material as submitted by the contributors and has not been edited by the editorial staff of the IAEA. The views expressed remain the responsibility of the contributors and do not necessarily reflect those of the IAEA or the governments of its Member States.

Neither the IAEA nor its Member States assume any responsibility for consequences which may arise from the use of this publication. This publication does not address questions of responsibility, legal or otherwise, for acts or omissions on the part of any person.

The use of particular designations of countries or territories does not imply any judgement by the publisher, the IAEA, as to the legal status of such countries or territories, of their authorities and institutions or of the delimitation of their boundaries.

The mention of names of specific companies or products (whether or not indicated as registered) does not imply any intention to infringe proprietary rights, nor should it be construed as an endorsement or recommendation on the part of the IAEA.

The IAEA has no responsibility for the persistence or accuracy of URLs for external or third party Internet web sites referred to in this publication and does not guarantee that any content on such web sites is, or will remain, accurate or appropriate.

CONTENTS

1. INTRODUCTION 1

1.1. BACKGROUND 1
1.2. OBJECTIVES 1
1.3. SCOPE 2
1.4. STRUCTURE 2

2. GENERAL INSTRUCTIONS 2

2.1. INSTALLATION 2
2.2. START UP 3
2.3. INITIALIZATION 3

3. IPWR DESCRIPTION 4

3.1. GENERAL SMR DESCRIPTION 4
3.2. SYSTEMS SIMULATED 5
3.2.1. Reactor coolant system and reactor core (RCS) 5
3.2.2. Main steam system 9
3.2.3. Feedwater system (FWS) 10
3.2.4. Turbine system (TUR) 11
3.2.5. Generator system (GEN) 12
3.2.6. Condenser system (CNR) 14
3.2.7. Circulating water system 14
3.2.8. Containment building system 15
3.2.9. Automatic depressurization system (ADS) 17
3.2.10. Containment cooling system 18
3.2.11. Gravity injection system (GIS) 19
3.2.12. Pressure injection system (PIS) 20
3.2.13. Passive decay heat removal system (PDHR) 21
3.2.14. Protection and control system (PCS) 22
3.3. SYSTEMS NOT SIMULATED 31
3.3.1. Shutdown Cooling System (SCS) 31
3.3.2. Containment vent 31
3.3.3. Generator auxiliaries 31
3.3.4. Turbine auxiliaries 32
3.3.5. Off-site power, diesel and batteries lights 32

4. MAIN FEATURES OF THE SIMULATOR 32

4.1. SIMULATOR OPERATION 32
4.1.1. Operating modes 32
4.1.2. Core cooling 32
4.1.3. Circulating water cooling 33
4.1.4. Setpoints 33
4.2. CONFIGURATION MENU 33
4.2.1. New page 34
4.2.2. Trends 34
4.2.3. Export to excel 36
4.2.4. User manuals 36
4.2.5. Initial conditions 37

4.2.6. Backtrack 38
4.2.7. Speed control 41
4.2.8. Controlled parameters 42
4.2.9. Generic malfunctions 42
4.2.10. Specific malfunctions 43
4.2.11. Restore layout 45
4.3. LIST OF IPWR-SMR DISPLAY SCREENS 45
4.3.1. Home display 45
4.3.2. Overview display 46
4.3.3. Core display 48
4.3.4. System display 49
4.3.5. Trips display 52
4.3.6. PZR pressure control 53
4.3.7. PZR level control 54
4.3.8. Feedwater control 55
4.3.9. Turbine control 56
4.3.10. Turbine bypass control 57
4.3.11. Rod position control 58

5. MATHEMATICAL MODELS USED 59

5.1. NEUTRONIC CORE 59
5.1.1. Neutronic equations 59
5.1.2. Neutronic calculations using space-time separation 60
5.1.3. Calculation of coupling coefficients 62
5.1.4. One dimensional amplitude calculation 63
5.1.5. Three dimensional shape calculation 63
5.1.6. Renormalization 65
5.1.7. Calculation of 1-d reactivity 67
5.1.8. Fuel temperature (Doppler) reactivity 67
5.1.9. Boron reactivity 69
5.1.10. Xenon concentration reactivity 69
5.1.11. Samarium concentration reactivity 69
5.1.12. Control rods reactivity 70
5.1.13. Moderator void reactivity 70
5.1.14. Calculation of 1-d reactivity 71
5.1.15. Startup source and photoneutrons 72
5.1.16. Xenon and iodine 73
5.1.17. Promethium and Samarium 73
5.1.18. Decay heat calculation 74
5.2. REACTOR COOLANT (RCS), BOP (MSS, FWS, TUR) AND CONTAINMENT (CBS, PDHR) HYDRAULICS 75
5.2.1. Balance equations 77
5.2.2. Constitutive correlations 81
5.3. SAFETY SYSTEMS (PIS/GIS/CCS) AND CIRCULATING WATER (CWS) 84
5.3.1. Description of hydraulic network resolution 85
5.3.2. Description of heat transfer between node and wall 88
5.4. ELECTRICAL SYSTEM (GEN) 93
5.4.1. Description of electrical network resolution 93
5.5. LOGICS (PCS) 94

APPENDIX 97

REFERENCES 101

ABBREVIATIONS 103

CONTRIBUTORS TO DRAFTING AND REVIEW 105

1. INTRODUCTION

1.1.BACKGROUND

The International Atomic Energy Agency (IAEA) has a suite of nuclear reactor simulation computer programs to assist its Member States in education and training. The objective is to provide, for a variety of advanced reactor types, insight and practice in-reactor operational characteristics and their response to perturbations and accident situations. To achieve this, the IAEA arranges for the supply and development of a suite of basic principles nuclear power plant simulators which are available for free to Member States upon request and are intended for educational purposes. The IAEA also provides associated training material, sponsors training courses and workshops, and distributes the documentation and computer programs.

The use of basic principles simulators to aid in teaching complex system interactions can considerably improve students' comprehension and retention of engineering course materials. In addition, the use of simulators on nuclear fundamentals type training courses can greatly add to trainees' understanding of reactor operation and the role of various systems.

The IAEA's existing suite of basic principles simulators are currently based on a variety of large scale, water cooled nuclear reactor technologies. The basic principles simulator helps in demonstrating the essential physical processes as well as fundamental system engineering concepts. This type of simulator also serves training objectives such as providing an overview of plant behaviour or a basic understanding of the normal as well as abnormal operation modes.

Such simulators may consist of complete primary and secondary circuits, sometimes with a reduced number of loops or redundancies. The scope of simulation focusses on the main systems where auxiliary or supporting systems may be neglected. The control room or panels very often have a fundamentally different design in comparison with conventional control room design.

The IAEA's suite of simulators is used on personal computers to develop understanding of the various reactor designs as well as their operational characteristics. They are intended for nuclear professionals involved in teaching and training on topics related to nuclear power plant design, safety, technology, simulation and operations; and could also be used, in some cases, for a broad audience of both technical and non-technical personnel as introductory educational tool. The simulators are not expected to produce accurate results but do demonstrate realistic trends and transients in response to changes made by the user.

In recent years the IAEA has seen an increase in the participation of its Member States in its programme for the technology development of small modular reactors. Various designs are under development in several Member States. A large number of the designs that are in development are light water cooled and moderated small integral pressurized water reactors (iPWRs). Common features of iPWR designs include modularity, passive safety systems for core and containment cooling, and integrated design — where most or all primary components are located inside the reactor vessel.

1.2.OBJECTIVES

The IAEA suite of basic principles simulators for nuclear power plants are used for educational purposes on personal computers to aid in the understanding of reactor plant fundamentals, operational characteristics and various approaches to reactor designs. Each simulator is supplemented by a simulator manual that serves the dual objective of providing all the necessary

information about the simulator and its underlying concepts as well as a consolidated training material.

The objective of this publication is also to support the understanding of the iPWR simulator and serve as a training material for training courses. An accompanying exercise handbook (Training Course Series No. 65) contains practical examples and exercises.

1.3.SCOPE

This manual consists of course material for training courses and workshops on Small Modular Reactor (SMR), of the iPWR type. Users of this simulator could run various normal operations as well as possible malfunctions in this type of reactor, thus gaining insight and understanding of the design and operational characteristics of iPWR power plant systems in normal and transient situations.

The iPWR system description, plant operation like plant load manoeuvring, turbine and reactor trips and recovery are all covered here, in addition to plant responses to malfunction events. Some malfunction events lead to reactor or turbine trip; other serious malfunctions (station black out) lead to accident situations, causing actuation of the passive core cooling safety systems.

It should be mentioned that the equipment and processes modelled in the simulator represent realistic thermal hydraulic characteristics. However, for the purpose of the educational simulator, there are necessary simplifications and assumptions made in the models, which may not reflect any specific vendor´s design and performance.

1.4.STRUCTURE

The user manual is divided into five main sections. The first section is basic introduction, providing idea about the scope and objectives of the document. The second section explains the installation, start-up and initialization of the simulator. The third section explains in detail the characteristic and design of the iPWR plant type used to make the simulator, along with detailed description of each system simulator. A brief description of systems not simulated, but important, is also provide. The fourth section describes the main features of the simulator explaining all the display screens and configuration menus. The fifth section discusses various mathematical models used in order to simulate the phenomena properly.

2. GENERAL INSTRUCTIONS

2.1.INSTALLATION

The simulator software is available as an executable installer. The installer will install the application needed to enable the user to run the iPWR-SMR simulator.

There are two installer releases: 64 bits CPU and 32 bits CPU.

The user should double click on the installer application and wait until the process is finished. The time required will depend on the user´s CPU and the applications already on the computer (for example installer will install.NET framework 4.6 if it is not present).

2.2.START UP

Once the simulator has been installed, a shortcut will be created on the user´s desktop. Then follow these steps:

(a) Double click the launcher (shortcut on the desktop);
(b) Once the simulation load is finished, the iPWR-SMR simulator will start automatically (Fig. 1);
(c) By default, initial condition 1 (IC#1): 100%, beginning of life (BOL) and NC (natural circulation) is loaded;
(d) Simulator is then ready for operation.

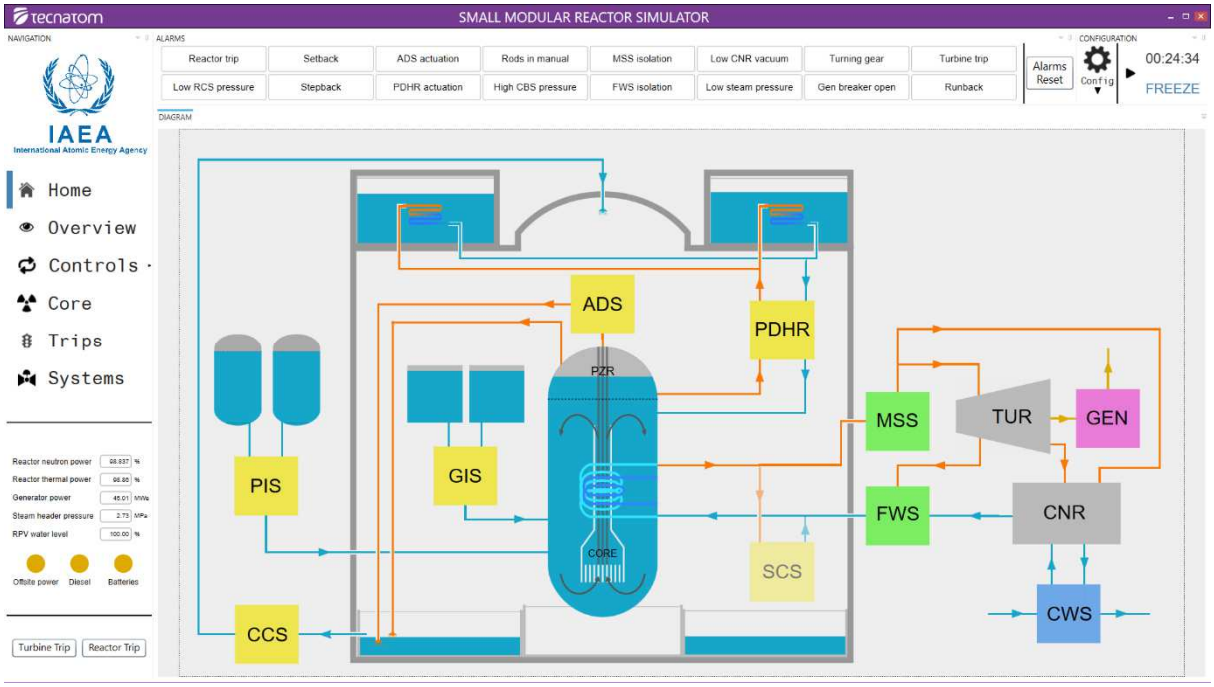


FIG. 1. Simulator overview.

2.3.INITIALIZATION

By default, IC#1 is loaded; however, the user can load any of the other ICs available (see Section 4.2.5).

Following is the set of available initial conditions:

(a) IC#1: 100% BOL (Beginning of life), NC (Natural circulation);
(b) IC#2: 100% MOL (Middle of life), NC (Natural circulation);
(c) IC#3: 100% EOL (End of life), NC (Natural circulation);
(d) IC#4: 100% BOL (Beginning of life), FC (Forced circulation);
(e) IC#5: 100% MOL (Middle of life), FC (Forced circulation);
(f) IC#6: 100% EOL (End of life), FC (Forced circulation);
(g) IC#7: 0% BOL, NC. Before withdrawing rods;

(h) IC#8: 0% NC. Hot reactor critical condition Keff =1;

(i) IC#9: 0% NC, Hot zero power at the point of adding heat (POAH);

(j) IC#10: 0% BOL, FC. Before withdrawing rods;

(k) IC#11: 0% FC. Hot reactor critical condition Keff =1;

(l) IC#12: 0% FC, Hot zero power at the point of adding heat (POAH).

3. IPWR DESCRIPTION

3.1.GENERAL SMR DESCRIPTION

There is continuing global interest in the development of SMR designs and technologies. One variety of reactors that is being developed in several countries is the small integral pressurized water reactor (iPWR). In this design, primary circuit components are placed within the reactor pressure vessel, eliminating the need for primary circuit pipework, with the intention of enhancing safety and reliability (Fig. 2).

With the establishment of nuclear power generation since the 1950s, the size of reactor units has grown from 60 MW(e) to more than 1600 MW(e). At the same time, there have been many smaller power reactors built for naval use (up to 190 MW(th) and as neutron sources. The IAEA defines 'small' as under 300 MW(e), and up to about 700 MW(e) as 'medium'. Together, they are now referred to by the IAEA as SMRs. However, 'SMR' is used more commonly as an acronym for 'small modular reactor', designed for serial construction and to involve a large nuclear power plant.

SMRs are defined as nuclear reactors generally under 300 MW(e), designed with modular technology using module factory construction, pursuing economies of series production and short construction times.

Generally, modern small reactors for power generation, and especially SMRs, are expected to have greater simplicity of design, economy of series production largely in factories, short construction times and reduced siting costs. Most are also designed for a high level of passive or inherent safety in the event of malfunction. Also, many are designed to be emplaced below ground level [1].

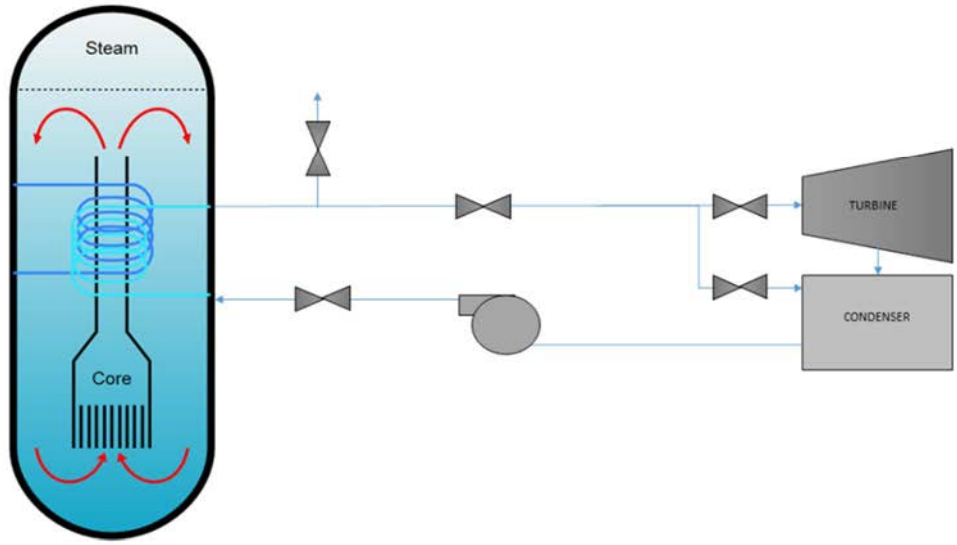


FIG. 2. iPWR sketch.

3.2.SYSTEMS SIMULATED

This iPWR simulator has been designed to represent the primary and balance of plant (BOP) behaviours. In order to simulate the whole operation, the following systems have been simulated:

(a) Reactor coolant system and reactor core (RCS);
(b) Main steam system (MSS);
(c) Feedwater system (FWS);
(d) Turbine system (TUR);
(e) Generator system (GEN);
(f) Condenser system (CNR);
(g) Circulating water system (CWS);
(h) Containment building system (CBS);
(i) Automatic depressurization system (ADS);
(j) Containment cooling system (CCS);
(k) Gravity driven water injection system (GIS);
(l) Pressure injection system (PIS);
(m) Passive decay heat removal system (PDHR);
(n) Protection and control system (PCS).

3.2.1. Reactor coolant system and reactor core (RCS)

Reactor coolant system is design to produce 150 MW(th) using UO_2, 4.95% Fuel enrichment. The core design uses 24 standard 17 × 17 fuel assemblies. The user could choose one of the two methods of core coolant circulation (natural or forced).

The RCS, along with the reactor control and protection systems, is designed to keep the reactor coolant under temperature, pressure and flow conditions suitable to protect the core, in order not to suffer any damage, and prevent high power, high temperature or low pressure conditions of the reactor resulting in a departure from nucleate boiling ratio (DNBR).

The system is the second barrier to fission products, the first being the fuel clad.

The stainless steel reactor pressure vessel (80.78 m^3) contains the reactor core, steam generator and pressurizer (8.078 m^3). The layout of the components inside the vessel is such that natural circulation is guaranteed, due to the heat sink (SG) being at a higher elevation than the heat source (Rx core). However, in this simulator, user could choose the method of core cooling, natural or forced, selecting one or another IC from the graphical user interface (GUI).

The simulator incorporates adjustments for both designs (natural or forced). In both cases, thermal power is the same (Fig. 3 and Fig. 4).

Relatively cold water enters the core at this base (255ºC). This water is then heated and flows up through the plenum, the core outlet temperature is around 320ºC. Saturation temperature at 15.5 MPa is 344.8ºC, so the reactor coolant is subcooled. It then flows down over the steam generator's secondary side tubes, in which water boils to form steam. SG acts as a separating barrier between the primary and secondary. SG consists of two helical-tube located within the reactor pressure vessel at a suitable height above the core.

As standard, coolant flow (424Kg/s) is driven round the reactor pressure vessel (RPV) due to buoyancy induced natural circulation resulting from temperature differentials in the water. If a forced circulation configuration is selected, the coolant flow will be driven by four horizontal pumps installed within the RPV above the core.

The reactor core consists of 24 assemblies of standard 17 × 17 design fuel assemblies for a total of 6336 fuel rods. The heated length is approximately 1.35 m and the equivalent diameter is approximately 1.2 m.

Reactivity control is made using the rods and boron, so reactor coolant water acts as a neutron moderator, reflector and boric acid solution as well.

The control rods are organized into two groups: a control group and a shutdown group. The control group is used during normal plant operation to control reactivity. The control rods are gravity actuated so that they will automatically drop into the core upon loss of power or scram events. Its function is to provide sufficient shutdown margin under any operating conditions. In this simulator, there are one shutdown bank and two control banks (rod worth aprox 1042 pcm). There is no bank overlap (usual in other nuclear power plant technologies) in order to simplify the simulator understanding.

The pressurizer is integral to the RPV, separated from the main RPV volume by a baffle plate. PZR is designed to accommodate any volume changes and limit the pressure changes in the RCS due to temperature variation during all operation modes. The pressurizer serves as the primary means of RPV pressure control and is designed to maintain constant reactor coolant pressure during operation (15.5 MPa).In the PZR, water and steam are kept in equilibrium under saturation conditions with the help of electrical heaters and sprays.

A bank of heaters is installed above the pressurizer baffle plate. Coolant pressure is increase by applying electrical power to these heaters (900KW). Coolant pressure is reducing by spraying

water into the pressurizer from spray nozzles which are fed by the chemical and volume control system. If spray is not enough for reducing pressure, there are two relief and two safety valves to prevent that RCS pressure exceeds the design value (17.05MPa).

The spray flow is designed to avoid the relief valve opening when there is a load rejection of 10%, with control rods in auto mode.

Each relief valve has a motorized valve in series with it in order to prevent a LOCA through the relief valve if it fails opened.

Besides, close to relief and safety valves, there are three ADS valves designed to reduce pressure in case of LOCA.

The pressurizer has a level control as well. The set point control changes during operation (and according to vessel average temperature) in order to accommodate eventual power transients, and it is controlled by charge and discharge flows (Chemical and volume control system).

RCS has connections with the safety systems: PIS and GIS which floods the RPV at certain reactor pressure and ADS which depressurizes the primary system in case of LOCA.

RCS has level and pressure instrumentation in PZR which is used for PZR level and pressure control. In-reactor vessel, there are pressure and level instrumentations as well. Besides, there are flow transmitters in charge/letdown/sprays lines and temperature transmitters at inlet/outlet reactor core and subcooling margin. Neutron power and variation in neutron power are also measured [2, 3].

PCS will generate reactor trip (rods at the bottom) when certain set point is reached in order to prevent the release of radioactivity. Following are the setpoints for reactor trip:

(a) Low pressure upper plenum (P < 11.0 MPa);
(b) Low level upper plenum (L < 5.0%);
(c) Low flow downcomer (Q < Q=F(P));
(d) High core outlet temperature (T < 340.0ºC);
(e) High reactor neutron flux (Flux < 120%);
(f) High log rate (SUR > 2. dpm);
(g) High pressure upper plenum (P > 16.4 MPa);
(h) FW pumps trip;
(i) ADS actuation;
(j) Seismic Event;
(k) Manual scram.

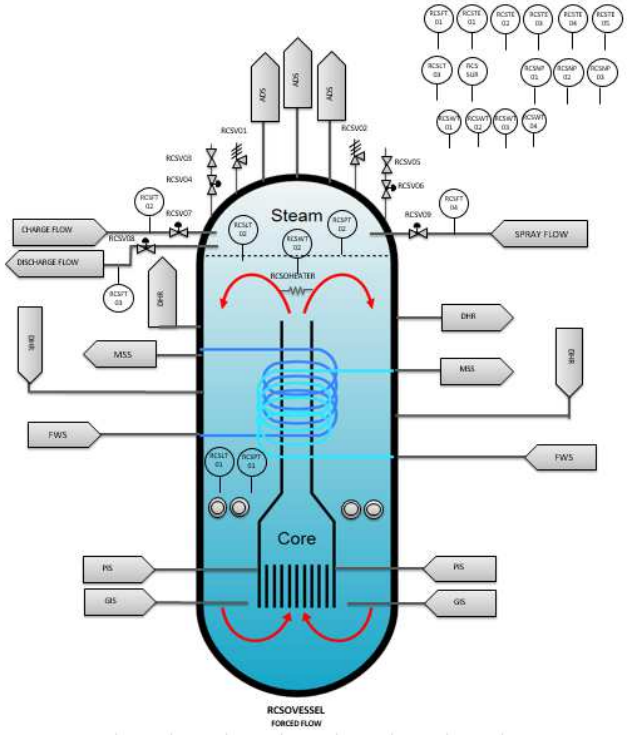


FIG. 3. Forced circulation.

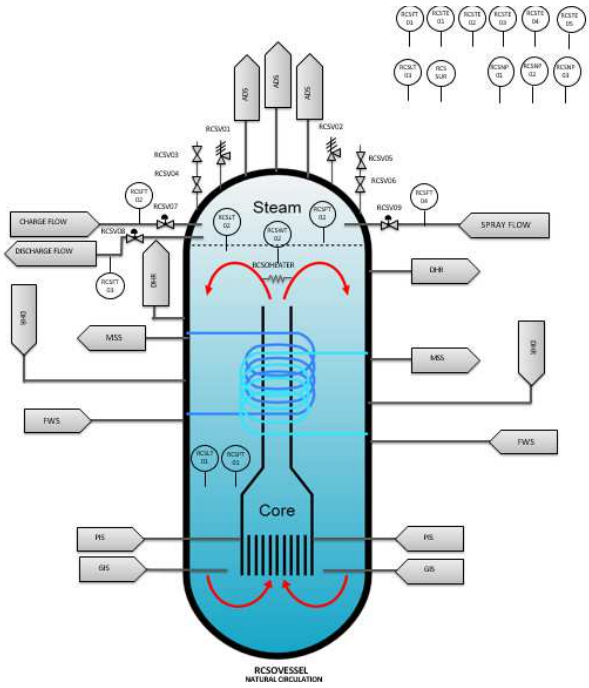


FIG. 4.Natural circulation.

3.2.2. Main steam system

The main steam supply system supplies steam from the steam generators to the turbine generator set in suitable quantities and adequate quality.

There is just a steam generator (SG) with two helical-tubes located within the reactor pressure vessel at a suitable height above the core. Each helical-tube has 506 tubes, Ø 16 mm and a tube thickness of 0.9 mm. They are made of thermally treated Inconel90.

Cold feedwater enters the secondary side tubes at the base of the SG and superheated steam is collected from its top. SG acts as a barrier between the primary and secondary circuits.

Superheated steam (around 30ºC) leaves the steam generator through an isolation valve on each SG steam line (Ø 200 mm each) which close by protection signal. At full power, MSS flow is 77Kg/s at 2.7MPa.

The steam lines join in a manifold (Ø 400 mm) and the steam then passes through a main steam control valve and isolation valve (Ø 400 mm) into a one stage pressure steam turbine. PCS controls the position of these valves.

The function of the isolation valves is to interrupt the flow of steam to the turbine. The function of the control valves is to regulate the steam flow to the turbine and to regulate this flow during turbine run-up (from 5rpm to synchronization speed 3600rpm). The isolation and control valves are installed in the steam boxes.

A turbine bypass line (Ø 300 mm) is fitted to allow steam to pass straight from the SG to the condenser via a control valve and isolation valve. This is used to allow reactor cooling to be maintained in the event of a turbine trip (so there is no reactor trip with turbine trip), load rejection and during start up/programmed shutdown, controlling temperature excess in RCS or pressure excess in MSS depending on the main steam bypass mode chosen. PCS controls the position of these valves. The turbine bypass has a capacity of 100%.

There is also a steam discharge which allows steam to be dumped to atmosphere via steam release valves and safety valves. The safety set point is higher than the relief set point.

Relief/safety valves open when turbine bypass is not capable of absorbing the steam produced by the reactor so they have enough capacity to maintain the reactor core cooled if the bypass is not available and there has been a reactor trip. Safety valves are mounted in order to prevent MSS pressure exceeding the design value (Fig. 5).

Turbine power is proportional to the turbine admission pressure, so a pressure transmitter is placed over admission chamber in order to allow the turbine control calculation.

Flow, pressure and temperature transmitters are mounted in each line and in the steam header. Besides, flow and valve control position of turbine line and turbine bypass line are also there.

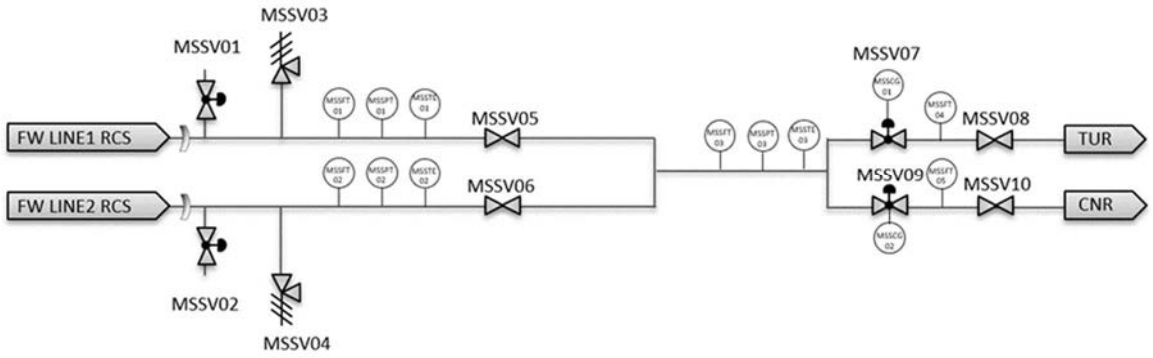


FIG. 5. Main steam system.

3.2.3. Feedwater system (FWS)

Feedwater system (FWS) provides a flow of treated and preheated water to the steam generators under the necessary conditions, in an amount equivalent to that converted into steam inside the SGs.

The condensate from condenser is pumped through condensate pumps (out of the simulator scope) and preheated in three heat exchangers, which preheat the water with steam extractions from the turbine, in order to increase the efficiency of the plant.

Then, condensate is pumped back to SGs by two variable feedwater pumps (they are installed in parallel to provide redundancy). Each pump has a capacity of 100%. FW pumps trip generates reactor scram and FW pumps trip if there is low steam header pressure (<1MPa) or low condenser level (10%). FW flow (77Kg/s) enter the only SG via two FW lines (two helical-tube) at approximately 173ºC.

Each helical-tube has 506 tubes, Ø 16 mm and a thickness of 0.9 mm.

A feedwater control valve and an adjacent isolation valve control the feed into each helical-tube of the only SG. These valves are mounted on pipes Ø 150 mm. Isolation valves will close when FW pumps trip or there is injection signal.

PCS calculates these valves position and FW pumps speed demand. This control tries to maintain equilibrium between the two helical loops and takes into account the pressure difference between MS header and FW header with reheated steam.

There are valves at the inlet/outlet of heat exchanger 1 and valves at inlet/outlet of the heat exchanger 2 and 3 group, in order to isolate them. Besides, there is one valve per each extraction in order to isolate the FW heat exchangers in case of a tube rupture inside the heat exchanger. In this case, the heat exchanger affected should be isolated and its bypass valve opened. There is a bypass valve for heat exchanger 1 while heat exchangers 2 and 3 share the bypass valve (Fig. 6). When this happens, FW flow won´t be reheated and will enter the SG cooler. This effect will provoke an increase in neutron power due to the over cooling.

When turbine trip takes place, feedwater pumps continue feeding the SGs.

Core cooling is made by FWS/MSS during operation at power or by PDHR in the case of loss of heat sink.

There is a storage tank which adds water to the secondary side in case there is a feedwater or main steam loss, for example opening of a steam relief valve. This tank is capable of adding water during 2 hours in case of opening of steam relief valve.

Instrumentation consists of: Flow transmitters at the entrance of each SG and temperature transmitters at the entrance/exit of each heat exchanger.

Feedwater pumps have a control for manually starting/stopping and for auto mode. If a feedwater pumps stops, the other one starts automatically if there is no malfunction activated.

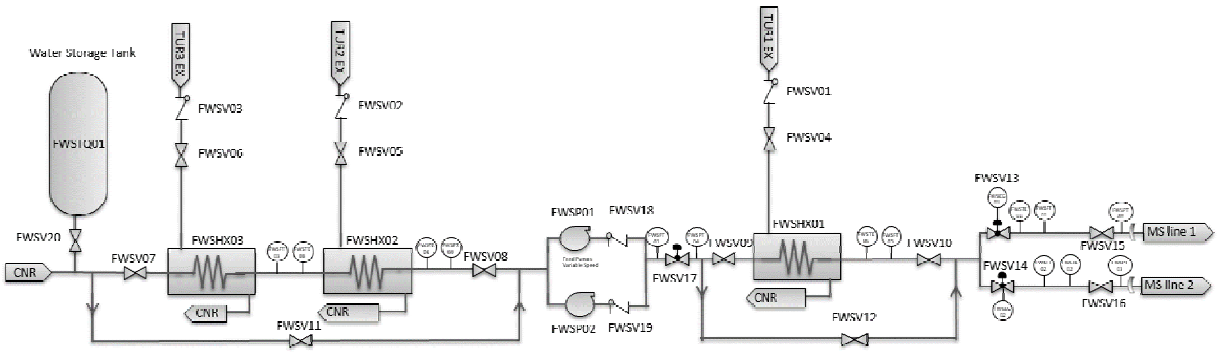


FIG. 6. Feedwater system.

3.2.4. Turbine system (TUR)

Turbine provides the alternator rotor with the mechanical power required to satisfy the load demands of the grid, maintaining a frequency of 60 HZ.

The turbine is designed to drive the generator, the maximum guaranteed output power is 45 MW(e) at a speed of 3600 rpm.

The thermal energy in water-steam is represented by its enthalpy. The objective of a turbine is to use this enthalpy to produce mechanical energy. The greatest transformation of enthalpy into mechanical work is achieved by means of isentropic expansion. In practice, this expansion is impossible to achieve, although the design of the turbine must be such that the expansion of steam in its interior tends to be isentropic.

The characteristic pressure-load curve of the SG and the pressure drop occurring in the steam/generator-turbine interconnecting pipe supply the admission pressure, which is the initial basic item of data required for the turbine design. The admission pressure is approximately proportional to the generated power (2.7MPa at 100%).

The second item of data that is taken into account during turbine design is the pressure loss at the turbine regulating valves during full load operation (nominal flow).

The turbine is a one stage pressure steam turbine. Steam is extracted from the turbine stages at three positions to preheat the feedwater in order to increase the efficiency of the plant. Once steam has been expanded in the turbine, it is discharged into condenser where it will be condensed by the circulating water (Fig. 7).

PCS use the first stage pressure transmitter to establish desired primary average temperature that will control the rod position when turbine leading mode is selected.

Turbine valves are described in MSS (Section 3.2.2).

PCS calculates a trip signal for turbine when certain parameters are reached. Following are the parameters that cause turbine trip:

(a) Low Vacuum at CNR (P > 254.0 mmHg);
(b) Reactor Scram;
(c) Reactor power less than 4%;
(d) Manual turbine trip;
(e) Low Oil Pressure (out of simulator's scope);
(f) Grid inestabilities (out of simulator's scope).

There are transmitters for turbine speed, turbine load demand, turbine load rate and first stage pressure.

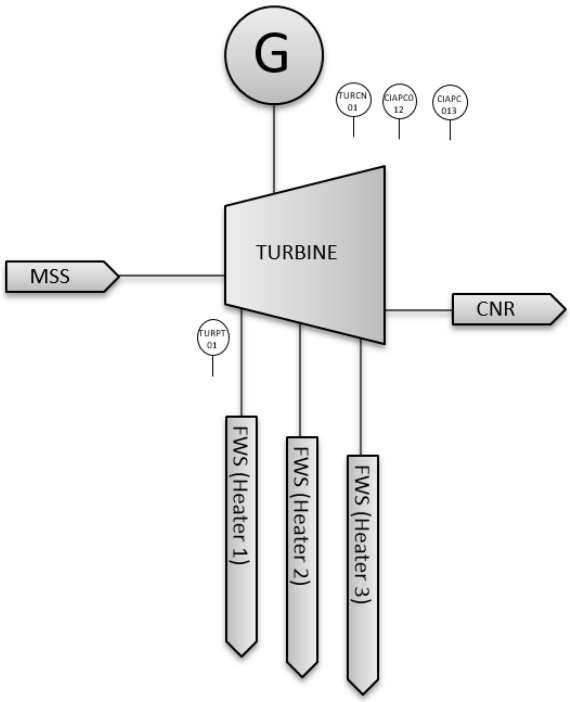


FIG. 7. Turbine system.

3.2.5. Generator system (GEN)

The generator converts the mechanical energy provided by the turbine into electricity. The GEN is designed with a production capacity of 45 MW(e) and a frequency of 60Hz. GEN is a self-exciting three phase AC.

The energy transformation processes that take place at a nuclear power plant begin with the production of thermal energy from the fission reactions occurring in the reactor core.

Subsequently, the thermal energy (SG steam) is transformed into mechanical energy by expanding the steam in the turbine.

Finally, this mechanical energy is transformed into electrical energy in the Main Generator (GEN). This electrical energy is distributed to the different consumption centres via a step up

transformer and part of it is used for electricity supply at this plant itself. If there is loss of off-site power, diesel generator will startup and the plant will be fed up using them. In case of Station Black Out (loss of AC electrical power), just some plant components will be fed up using the batteries, such us PCS.

Main Generator is made up of the following:

(a) Shell, stator, rotor and bearings;
(b) Excitation (out of simulator's scope);
(c) Generator auxliary systems (out of simulator's scope).

The shaft rotation and excitation generates the required magnetic field to induce the electromotive force needed to generate electric power.

When the generator is synchronized to the net, the main breaker has to be closed. When there is a turbine trip, the main breaker opens (Fig. 8).

There is a transmitter that measures the generator load (MW).

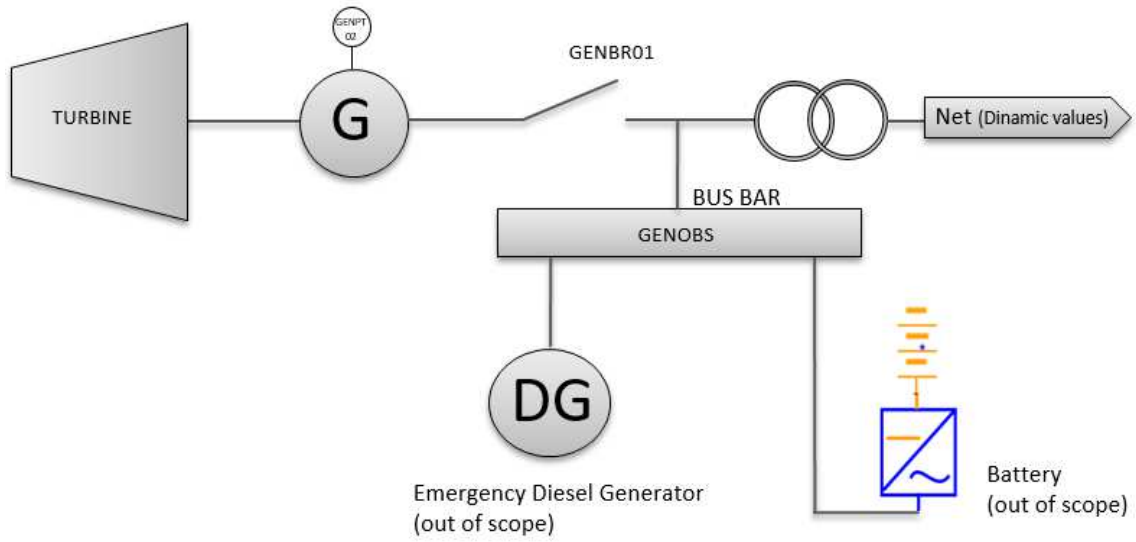


FIG. 8. Generator system.

3.2.6. Condenser system (CNR)

Condenser consists of a shell and a set of tubes for the circulating water. The exhaust steam from the turbine enters the shell where it is cooled and converted into condensate (liquid) by flowing over the tubes.

Condenser acts as the heat sink for the plant's secondary cycle. CNR transfers heat to the circulating water running through the inside of the tubes and acting as a coolant.

The condenser collects the flow from the turbine, the flow from the steam lines through the main steam bypass and the flow from the drains of the heat exchanger chain (Fig. 9). Condensate pumps suck from CNR (out of the simulator scope).

The condenser works under vacuum (47.83 mmHg) in order to maintain the steam pressure of water under the atmospheric pressure, improving the thermodynamic performance of the cycle.

CNR has a pressure and level transmitter.

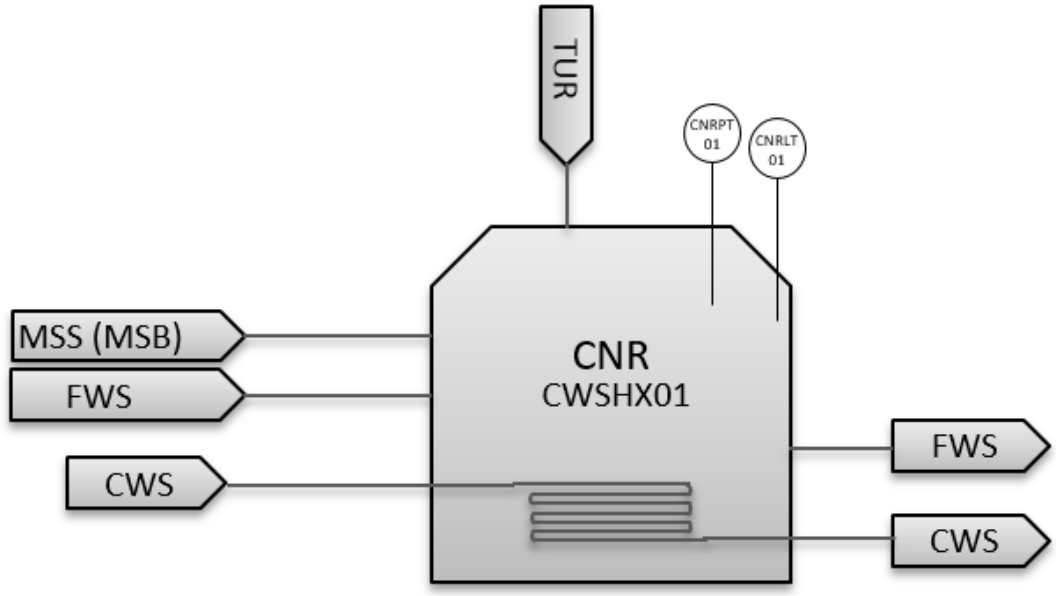


FIG. 9. Condenser system.

3.2.7. Circulating water system

Once the steam exits the turbine and enters to the condenser, it is cooled by the circulating water where steam condensates.

The functionalities of the CWS system are:

(a) To supply to the condenser the amount of water needed to extract the maximum thermal load produced by the condensation of the steam from the turbine and;

(b) To maintain the vacuum in the condenser.

The CWS system has 2 different operating modes. First one (open loop) consists of taking and discharging water from and to the sea while second one (close loop) recirculates water through the tube condenser using a refrigeration tower (Fig. 10). In both operating modes the system has 2 circulating pumps, each having a capacity of 50% of the total flow. In the recirculation mode the refrigeration tower refrigerates the water flow to a suitable temperature to be carried again to the condenser.

A cooling tower is a heat rejection device which rejects waste heat to the atmosphere through the cooling of a water stream. Cooling tower uses the evaporation of water to remove process heat and cool the working fluid to near the wet-bulb air temperature (parameter that could be changed by user).

CWS is at atmosphere pressure while there is vacuum in condenser. In case, tube rupture takes place, water flows into CNR from CWS.

The shutdown of one or more CWS pumps will generate a partial or total loss of vacuum in the condenser.

CWS has transmitters of temperature at the entrance and exit of condenser and a flow transmitter.

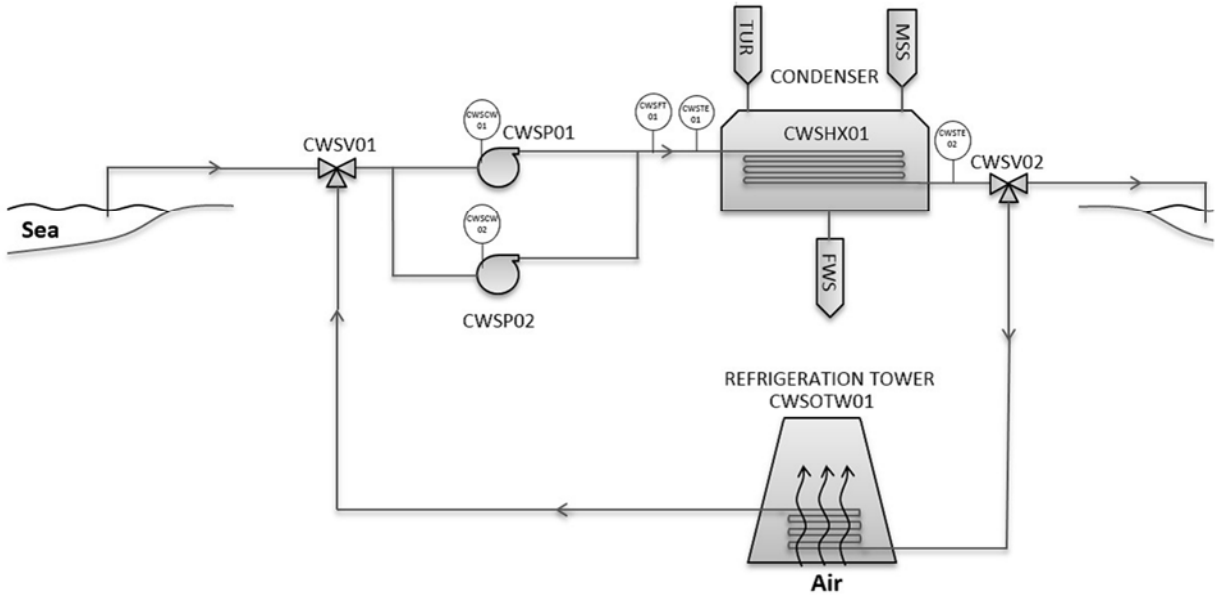


FIG. 10. Circulating water system.

3.2.8. Containment building system

The containment building is an enclosure building that houses among other systems the reactor vessel, the cooling and part of the safety systems. This containment is an air containment (other SMR designs have a water containment).

The structure of the containment building of the reactor is a cylindrical concrete structure with an embedded steel liner (Height = 25.3 m, Radius=12 m, Volume = 11445 m^3). The primary containment is of pressure-suppression type with two major compartments: a drywell and wet well. The drywell includes the volume that surrounds the reactor pressure vessel. A partition

floor and cylindrical wall separate the drywell from the wet well. The lower part of wet well volume is filled with water that works as the condensation pool (576.0 m^3) (PZR relief and safety valves and ADS valves discharge here), and the upper part is a gas chamber.

CBS contains two pools at the upper part which work as condensers for the Decay Heat Removal System (PDHR) (Fig. 11).

Any water leaking from the RPV itself will be collected within a cavity in the containment that surrounds the lower section of the RPV. This will flood over time, providing cooling for the RPV. Any water that boils to steam within the containment will condense on the steel liner of the containment. In case of sump overflowing, this water will go to suppression pool.

A core spray system is used to reduce containment pressure and temperature. This is an active system, with water being pumped from the containment´s suppression pool (CCS).

The functionalities of the CBS system are:

(a) To keep within acceptable limits the leakage rates of radioactive materials to the environment, both in normal operation and in any of the design basic accidents;
(b) To provide a radiation shielding of the reactor and the cooling system. It isolates the reactor cooling system and other safety systems from the external environment conditions, and;
(c) To provide last barrier in the retention of radioactive products.

The containment isolation consists of a set of valves installed in the pipelines that pass through the wall of the containment building so that these lines can be isolated. It is designed to prevent the escape of radioactive fluids through the pipelines that close the containment building during an accident, especially loss of reactor cooling accident.

Containment vacuum is achieved through the use of ventilation systems (out of simulators' scope).

CBS has level and temperature instrumentation in each pool, suppression pool and DHR condensers. The air part of CBS has pressure and temperature transmitters as well.

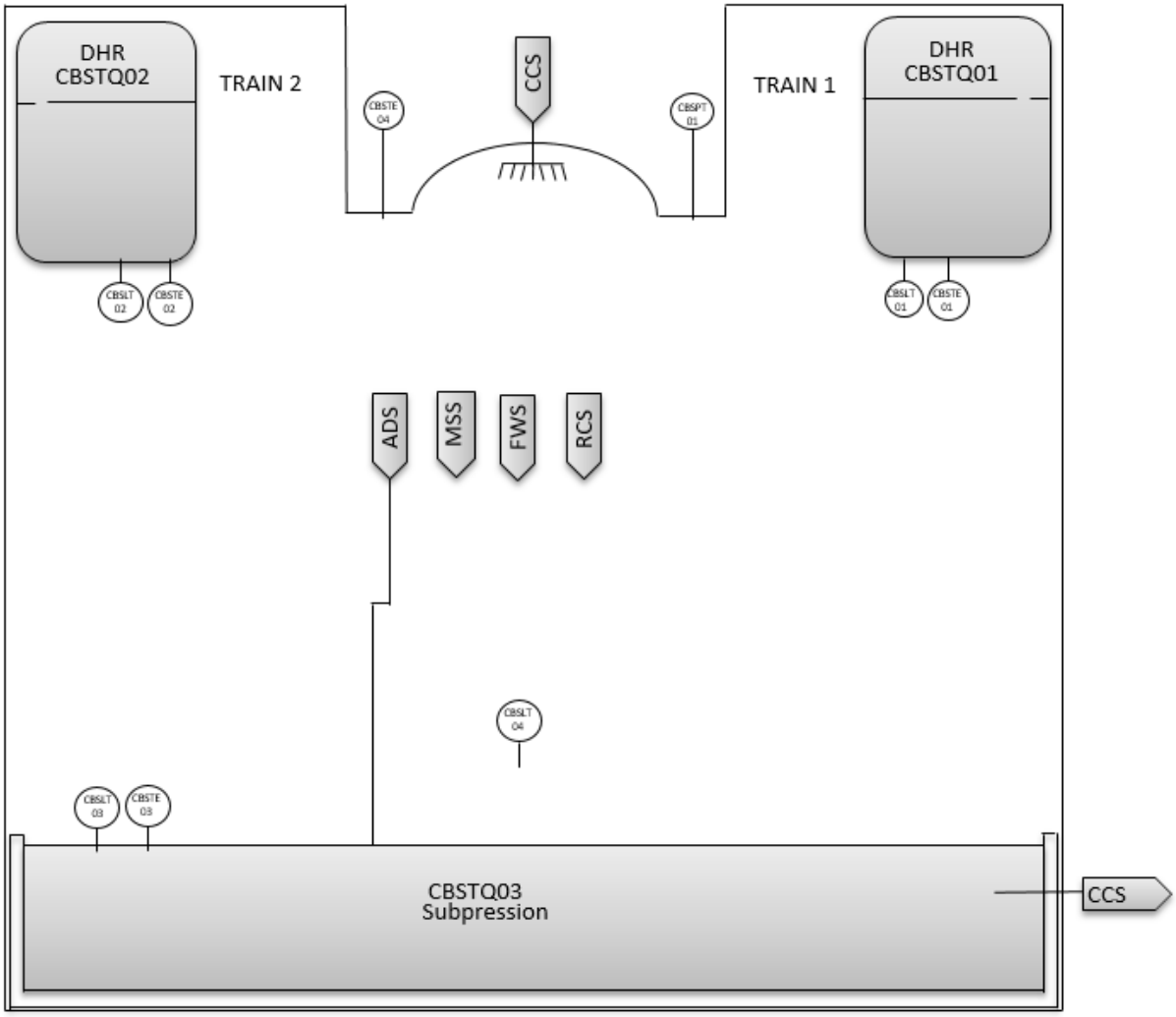


FIG. 11. Containment Building system.

3.2.9. Automatic depressurization system (ADS)

The ADS is initiated by opening three valves connected to the primary system. Through these valves the steam is released into the containment building (suppression pool) where it condensates.

ADS is a safety system along with GIS, CCS, PIS and DHR.

The functionality of the ADS is to lower the reactor pressure by releasing steam to the suppression pool (containment building, see section 3.2.8) where it condenses. This allows PIS or gravity feed water sources (GIS) to supply water to the reactor in order to keep the core covered (for example in case of LOCA). ADS design criteria is to lower RCS pressure to avoid core void and, consequently, core damage (Fig. 12).

Three pressure relief valves provide redundancy, with each being able to provide 100% of the require pressure relief capacity (Ø 101.6 mm).

PCS calculates the ADS valve position which open one by one and need around 5 minutes to open completely in order to avoid the core void production.

ADS valves will open if there is an ADS signal.

There are flow transmitters downstream ADS valves.

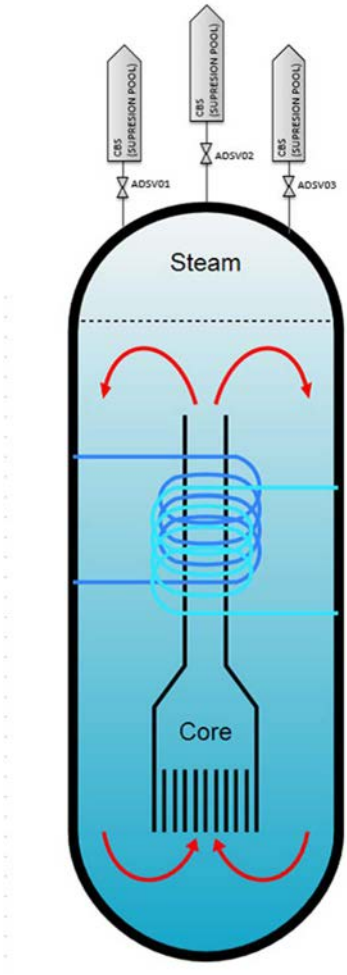


FIG. 12. Automatic Depressurization system.

3.2.10. Containment cooling system

The CCS system is a safety system. It comes into operation automatically when pressure is very high in the containment building (0.019MPa). Borated water solution (2500 ppm) from the CBS system is sprayed into the containment building via spray nozzles.

CCS is a safety system along with GIS, ADS, PIS and DHR.

CCS is an active system, with water being pumped from the containment´s suppression pool to the upper part via the spray nozzles. There are two independent trains, each with a pump of 100% (Fig. 13).

PCS calculates the pumps and discharge valves status demand which start/open when containment pressure is bigger than 0.019 MPa above atmosphere.

The functionalities of the CCS system are:

(a) To reduce the pressure in the containment building to a value less than the design pressure in case of LOCA or steam line break or feedwater line break inside the containment, and;
(b) To reduce the concentration of iodine in the atmosphere of the containment building after an accident (out of simulator scope).

Instrumentation consists on flow transmitters, one per train.

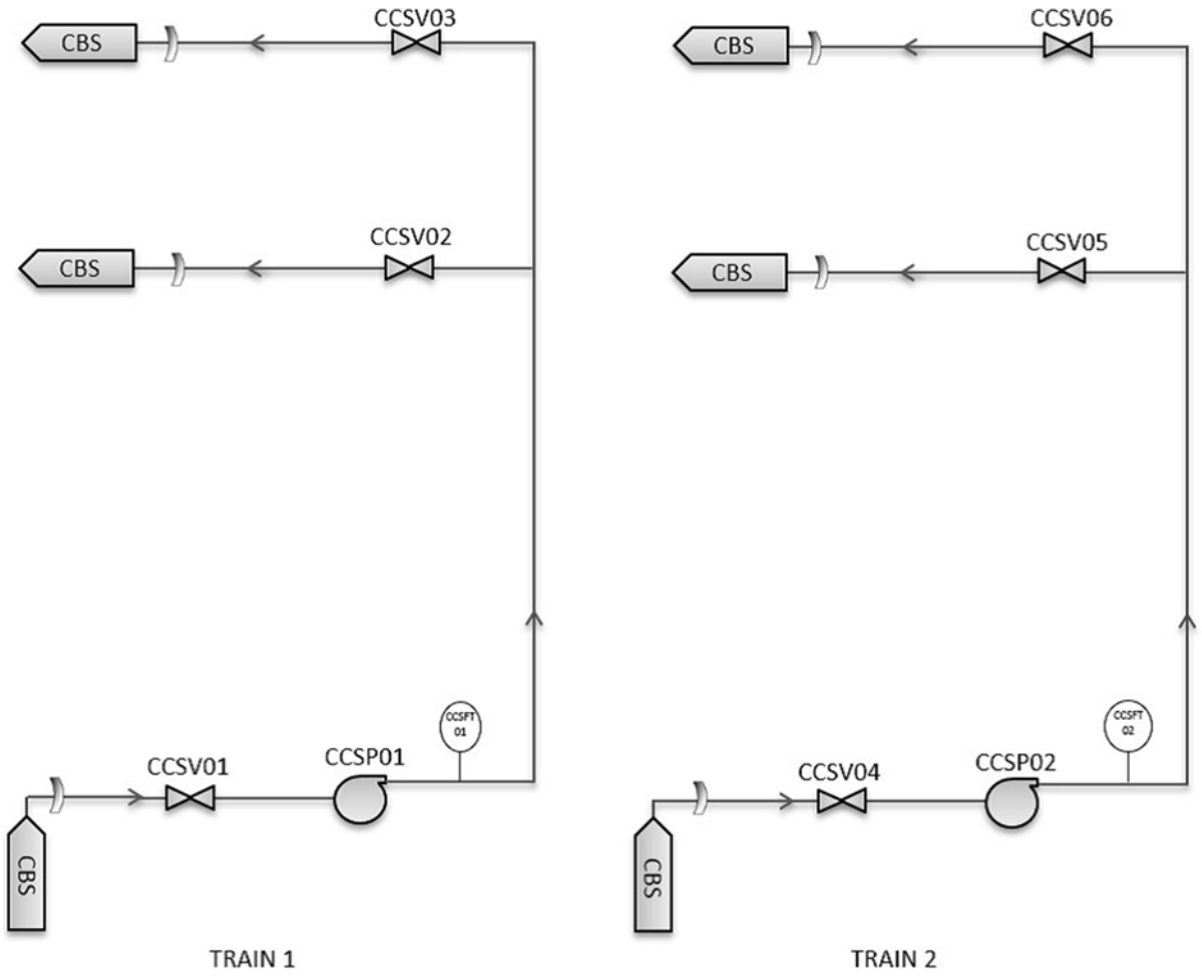


FIG. 13. Containment cooling system.

3.2.11. Gravity injection system (GIS)

GIS system is a passive water injection system. Once pressure decreases in the reactor vessel, the system comes into operation by gravity. The system consists of two tanks with borated water (2500 ppm) connected to the reactor vessel (two trains).

GIS is a safety system along with PIS, CCS, ADS and DHR.

The functionality of the GIS system is to inject water to the reactor vessel to reduce the pressure in the primary system and ensure shutdown margin. Each of the two tanks is connected to the reactor vessel (above 20 m) by one piping line with 2 valves. One of the valves is a check valve which opens when pressure in the tank is bigger than the reactor vessel pressure (P<0.5MPa, dynamic valve) and the second one is a motorized valve which just closes when plant goes to shut down in order to avoid the tank discharge in the RCS when it is not required..

This line goes from the lower part of the tank to a position below the reactor core level. When the pressure in the primary system decreases to gravity, the check valves open automatically and the borated water from the tank drains into the primary system by gravity (Fig. 14).

The tanks volume along with PIS tanks are designed to keep the core flooded under LOCA condition for a long term [4].

GIS has two flow transmitters.

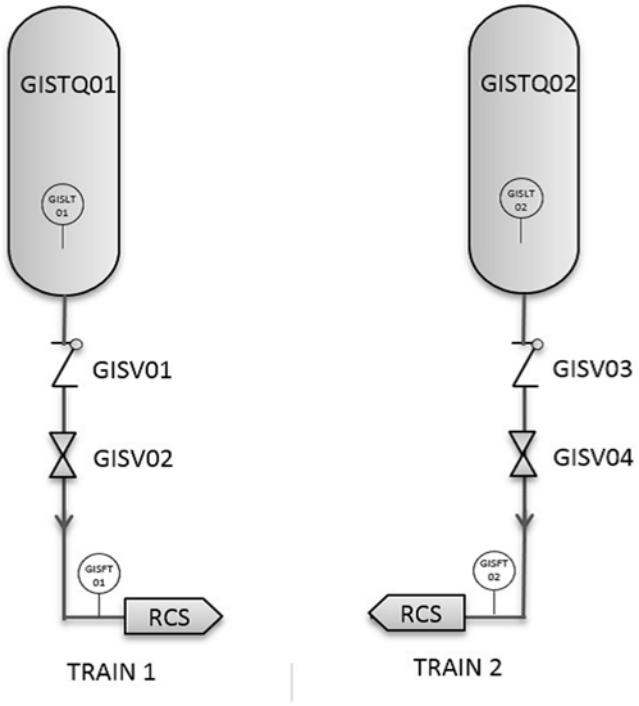


FIG. 14. Gravity Injection system.

3.2.12. Pressure injection system (PIS)

PIS is a passive injection system. It comes into operation when the pressure in the reactor vessel is below 5.0 MPa. It consists of two trains with a borated water tank (2500 ppm) connected to the reactor vessel. The tanks are pressurized at 5.0 MPa.

PIS is a safety system along with GIS, CCS, ADS and DHR.

Once the ADS has depressurized the RCS up to 5.0 MPa, PIS passively starts to injected borated water into the RPV. The storage tanks (two independent trains) for this water are pressurized to only flood the RPV when system pressure drops below a specified pressure. This water injection maintains the water level within the RPV above the top of the core.

Each of the two tanks is connected to the reactor vessel by one piping line with 2 valves. One of the valves is a check valve which opens when pressure in the tank is greater than the reactor vessel pressure (dynamic valve) and the second one is a motorized valve which just closes when plant goes to shut down in order to avoid the tank discharge in the RCS when it is not required (Fig. 15).

The tanks volume along with GIS tanks is designed to keep the core flooded under LOCA condition for a long term.

There are two flow transmitters, each per train.

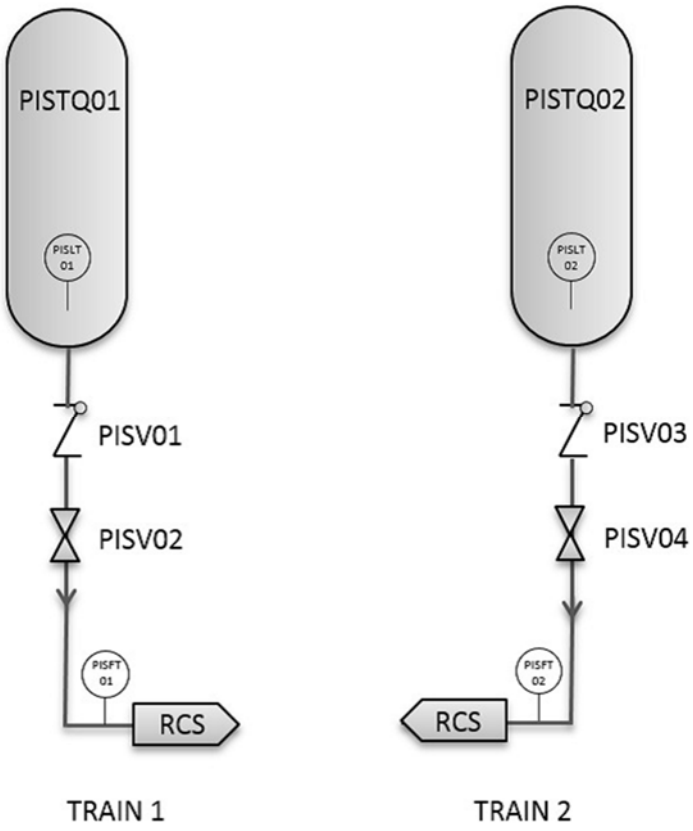


FIG. 15. Low pressure injection system.

3.2.13. Passive decay heat removal system (PDHR)

The PDHR system operates by condensing steam from the primary system in emergency condensers. These condensers are located in a pool inside the containment building and over the reactor vessel in order to have natural circulation.

Two independent PDHR trains remove heat from the core through the establishment of natural circulation loops, with decay heat removal pools working as heat sinks (Fig. 16).

During normal operation, FWS/MSS are the systems which cool the reactor core but in case of losing them, PDHR is the system responsible of cooling.

The reactor vessel is connected to decay heat removal condensers within dedicated pools inside the containment. Upon activation of the decay heat removal system, the inlet/outlet valves open

and FWS and MSS are isolated. Water drains from the condenser tubes into the reactor vessel and draws steam into the condensers. Here, it transfers its heat to the water pool. This establishes a natural circulation loop.

Pools must have enough volume (148.5 m^3) in order to maintain for a minimum of 7 days the natural circulation and takes the reactor to a temperature lower than the temperature of zero load. If tubes are not cooled enough by this decay heat removal pool, there will be neither delta temperature in the tubes nor natural circulation.

Each train of this decay heat removal system has sufficient capacity to remove the decay heat after a shutdown from full power operation.

PDHR pool has a vent in order to lower pressure in case of overpressure transients (The vents simulation is out of simulator's scope).

PCS calculates the valves position which opens when there is a PDHR Signal.

There are two flow transmitters, one per train.

Shutdown Cooling System (SCS) is an active system (with pumps). This system (out of simulator's scope) would be used when plant is taken to cold shutdown under normal operation.

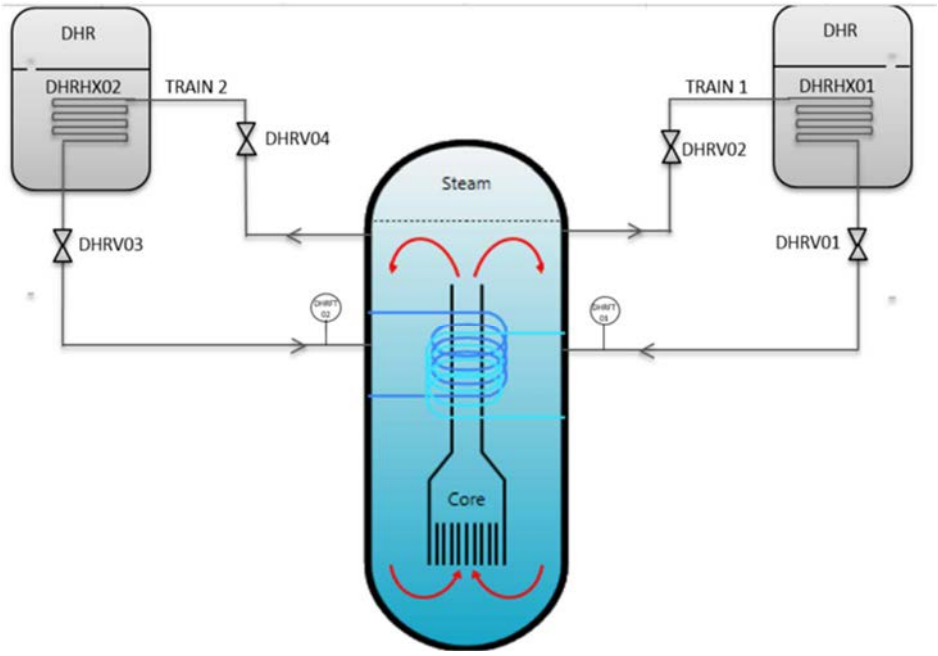


FIG. 16. Passive decay heat removal.

3.2.14. Protection and control system (PCS)

The function of the protection system is to protect the three barriers between the fuel and the public: fuel cladding, reactor coolant vessel and containment. The Control system function is to provide the changes needed to keep the operating parameters.

PCS includes the control loops required for controlling the reactor and a reactor protection system which causes the reactor or turbine trip or stepbacks/setbacks when plant parameters exceed certain setpoints.

The purpose of the protection system is to provide automatic protection against unsafe operation conditions during steady state operation and power transients. Its functions are:

(a) To generate proper trip, interlock and safety signal;
(b) To measure the process parameters in a continuous fashion during all operation modes, and;
(c) To keep certain parameters and systems within their operating range.

The protection signals are:

(a) Reactor and turbine trip (Fig. 28 and Fig. 29);
(b) Reactor setback, reactor stepback and turbine runback (Fig. 31 and Fig. 32);
(c) Safety signal: ADS actuation and PDHR actuation (Fig. 30).

The purpose of the control system is to fulfil the following functions:

(a) To establish and maintain the balance of power between the primary and secondary systems during steady state plant operations;
(b) To restrict transient operating conditions with a view to prevent plant trip and re-establishing steady state plant operation;
(c) To provide the operator with the necessary information by means of the monitoring instrumentation in order to ensure understanding of the conditions of the process at each moment in time.

The control systems are:

(a) Pressurizer Pressure Control (RCS);
(b) Pressurizer Level Control (RCS);
(c) Feedwater Control (FWS);
(d) Turbine Control (MSS-TUR);
(e) Main Steam Bypass Control (MSS);
(f) Rods Control (RCS).

Pressurizer level control: The set point control changes during operation (and according to vessel average temperature) in order to accommodate eventual power transients, and it is controlled by charge and discharge flows (chemical and volume control system) (Fig. 17 and Fig. 25).

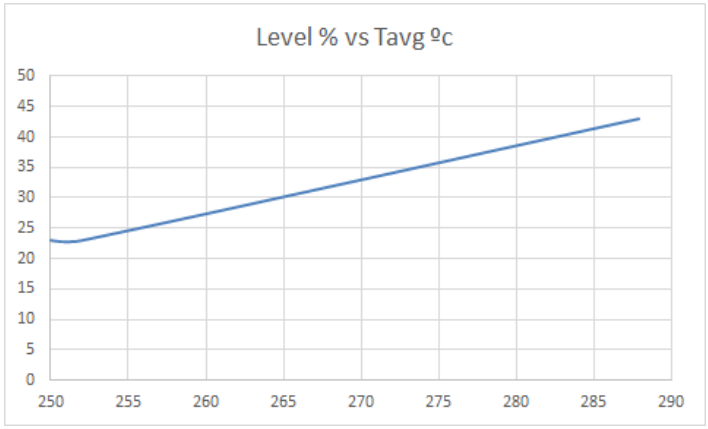


FIG. 17. Level program.

Pressurizer pressure control: This control is designed to maintain constant reactor coolant pressure during operation (15.5 MPa). In order to get this goal, pressure is controlled by a bank of heaters, spray valve and relief valves (Fig. 18 and Fig. 24).

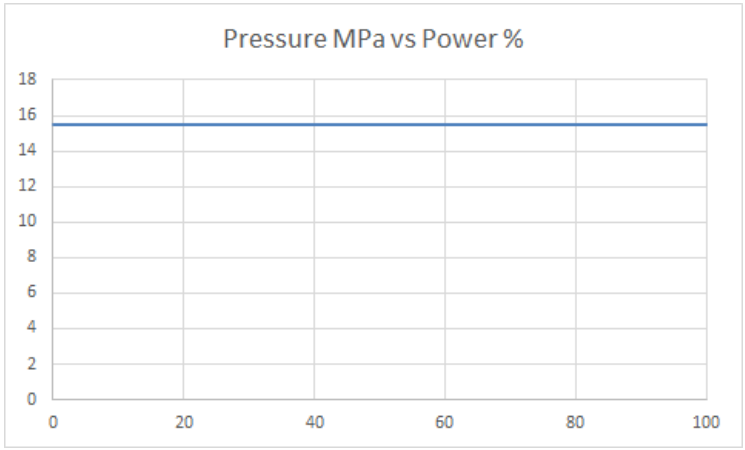


FIG. 18. Pressure program.

Feedwater control: This control tries to maintain equilibrium between the two helical loops of SG and takes into account the pressure difference between MS header and FW header with reheated steam (Fig. 19 and Fig. 27).

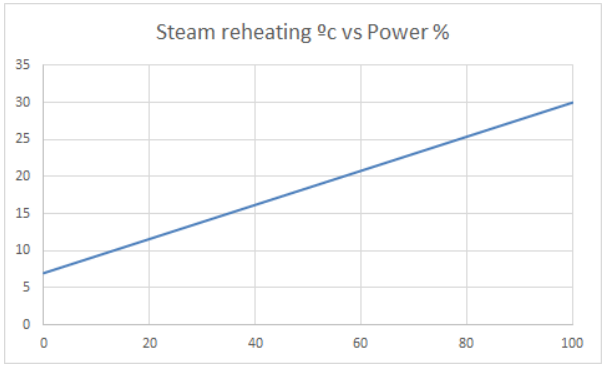


FIG. 19. Feedwater program.

Turbine control: This control tries to maintain the load required by regulating the steam flow to the turbine during power operation if simulator is in turbine leading mode or tries to keep steam pressure if simulator is in-reactor leading mode (Fig. 20 and Fig. 23).

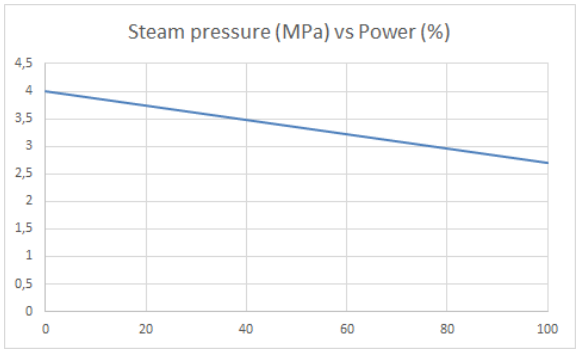


FIG. 20. Steam pressure program.

Main steam bypass control: This control tries to maintain average temperature in RCS or steam pressure in MSS depending on the chosen operation mode (Fig. 21 and Fig. 26).

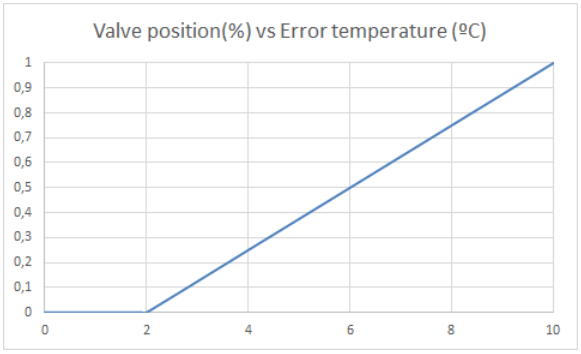


FIG. 21. Bypass program, turbine leading mode.

Rods control: This control moves the rods in order to maintain average temperature if turbine leading mode is selected. If reactor leading mode is chosen, this control moves the rods in order to keep the reactor power demanded (Fig. 22).

Alarms belong to PCS as well. Each of the protection signals has an alarm associated.

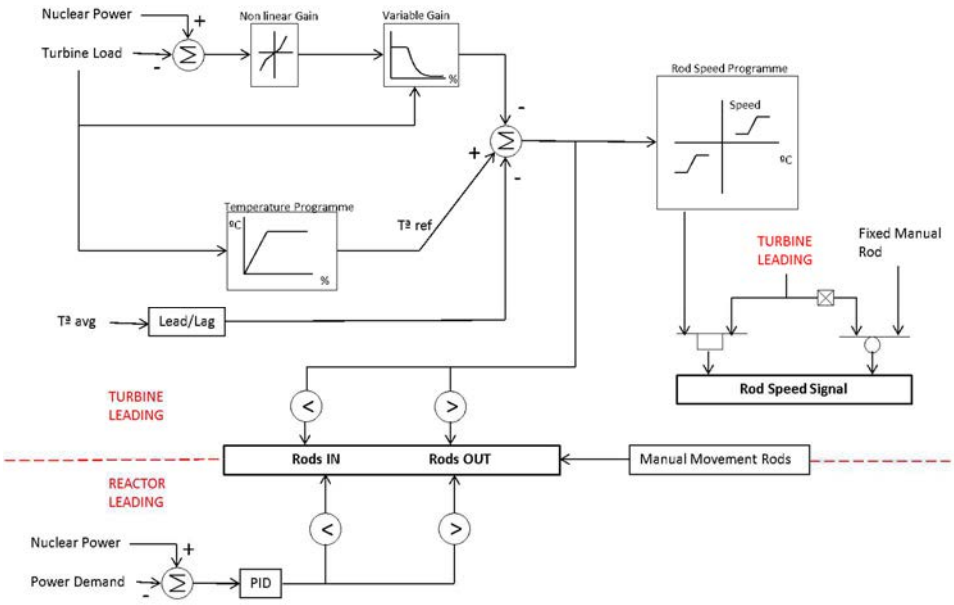


FIG. 22. Rods control.

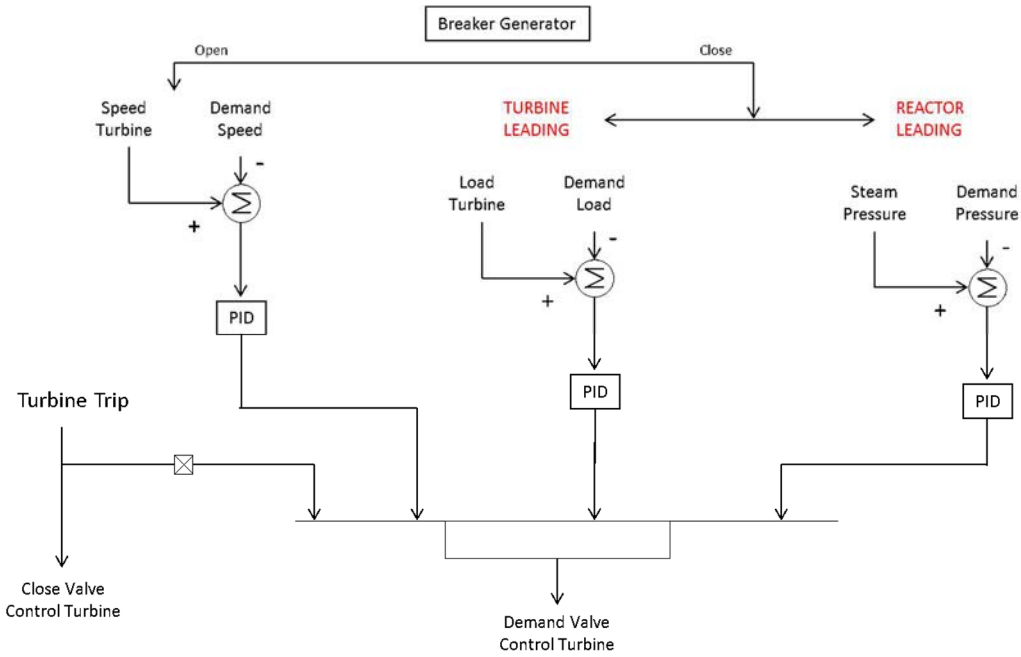


FIG. 23. Turbine control.

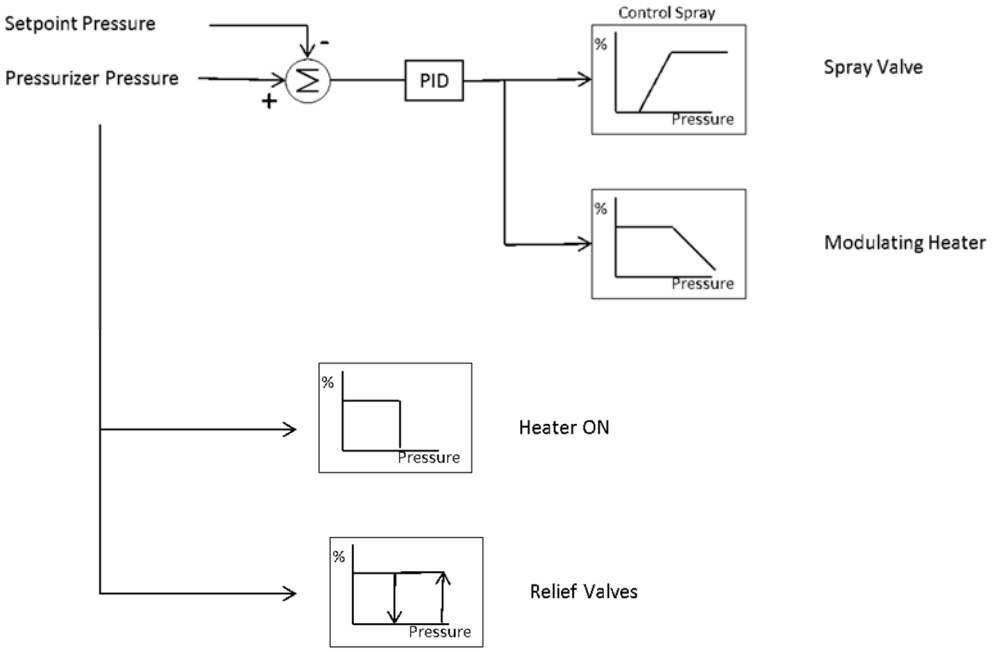


FIG. 24. Pressure control.

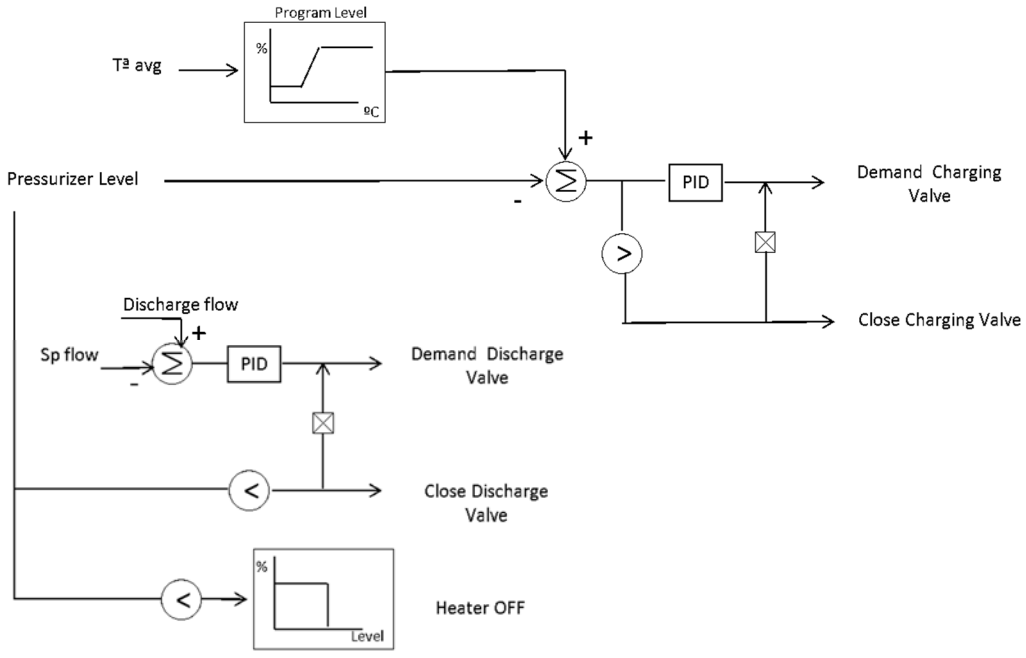


FIG. 25. Level control.

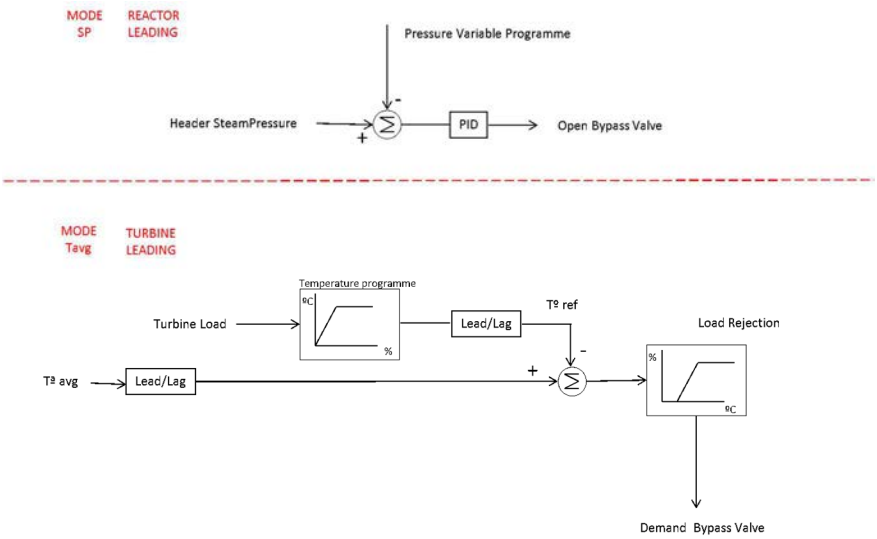


FIG. 26. Bypass control.

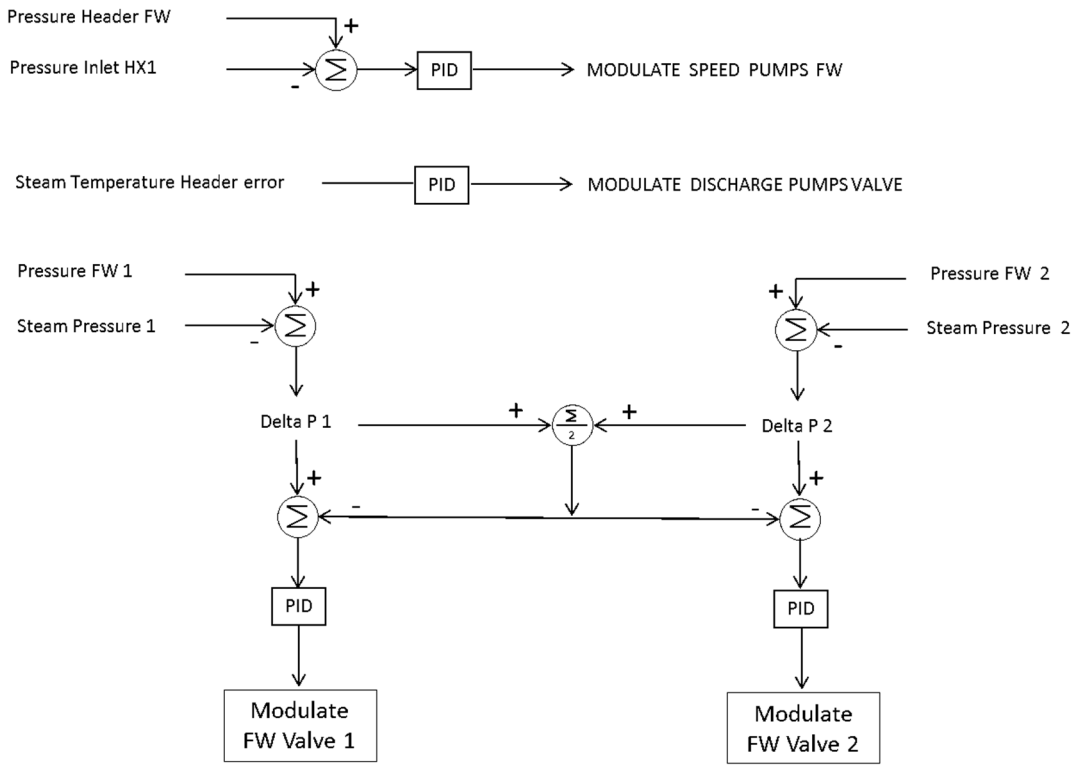


FIG. 27. Feedwater control.

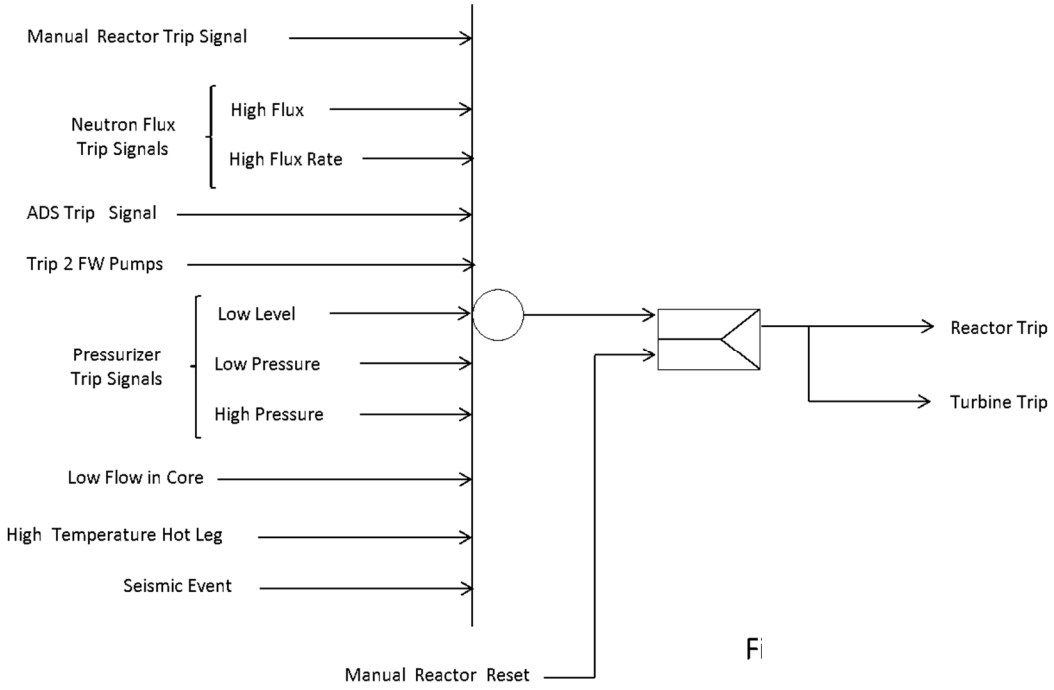


FIG. 28. Reactor trip signals.

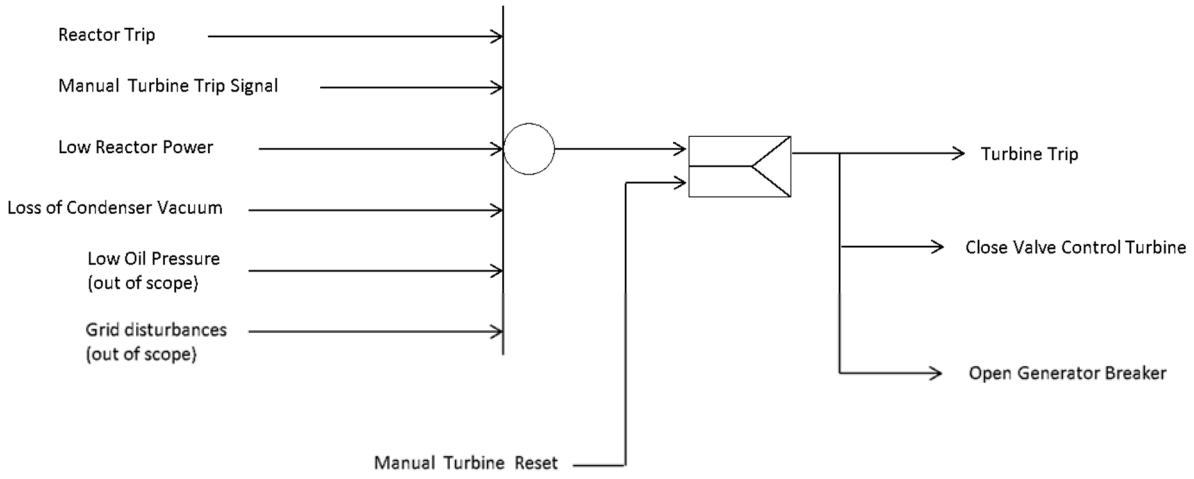


FIG. 29. Turbine trip signals.

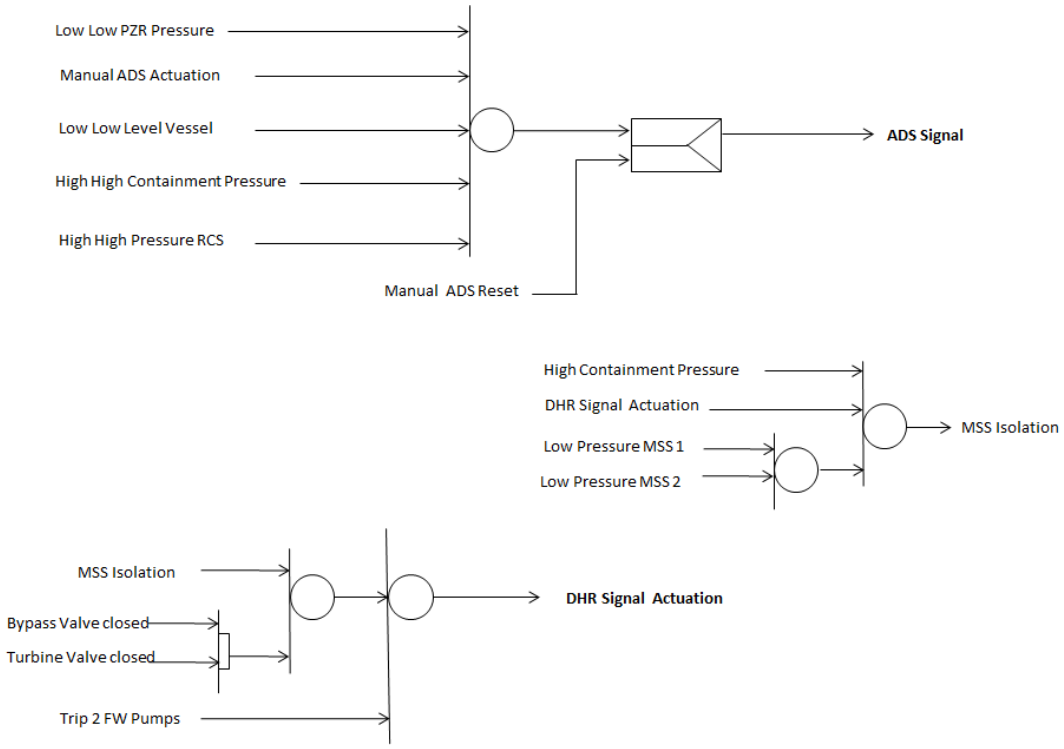


FIG. 30. Safety actuation signals.

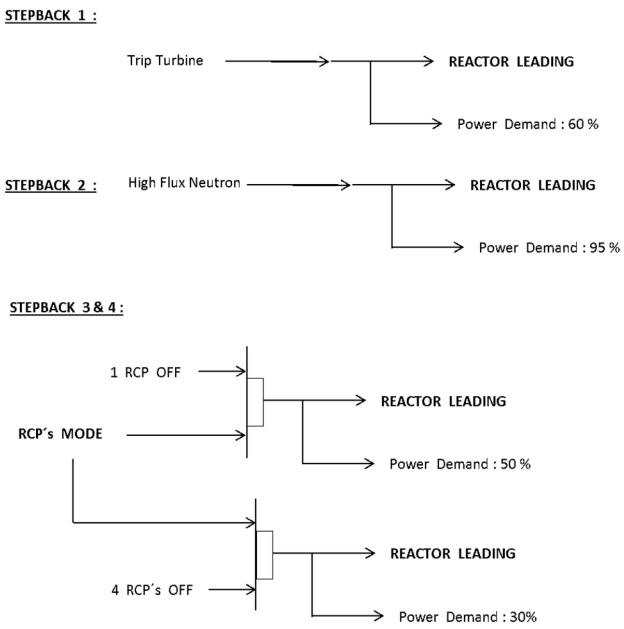


FIG. 31. Stepback Mode.

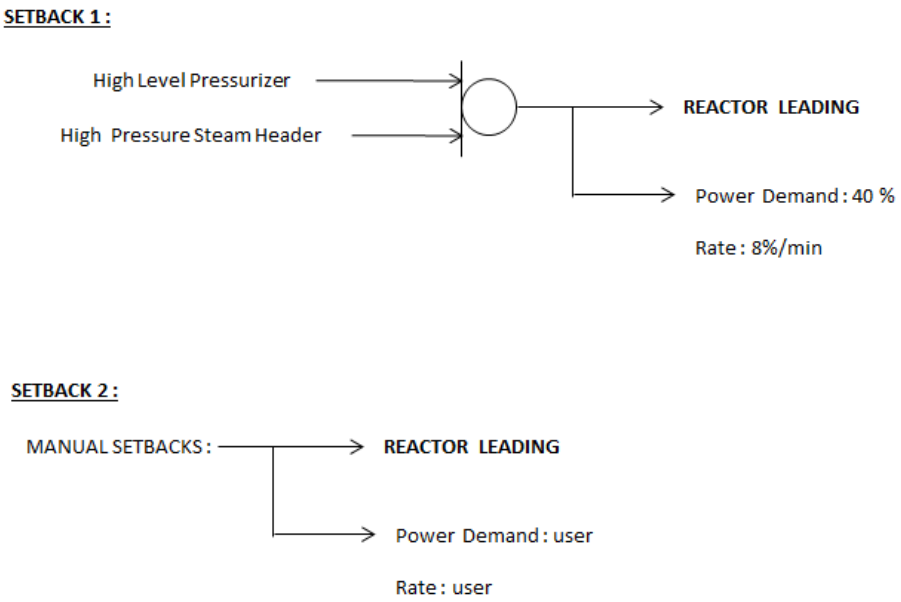


FIG. 32. Setback mode.

3.3.SYSTEMS NOT SIMULATED

The following systems are out of simulator's scope or have a partial scope but they have been introduced in the manual for a better understanding of the user:

(a) Shutdown Cooling System;
(b) Containment venting;
(c) Generator auxiliaries;
(d) Turbine auxiliaries;
(e) Off-site power, Diesel and Batteries ligths.

3.3.1. Shutdown Cooling System (SCS)

SCS would be the system for removing residual heat under normal operation. This system would be an active system and consist of one pump and a heat exchanger cooled by component cooling system.

3.3.2. Containment vent

Each DHR pool has a vent valve that would vent to atmosphere in case pool pressure reaches a set point.

3.3.3. Generator auxiliaries

Generator auxiliary systems such as oil, water and air cooling for coils, windings are out of the simulator's scope.

In order to generate the required magnetic field to induce the electromotive force needed to generate electric power, the shaft rotation and excitation are required. However, excitation simulation is out of the simulator's scope.

3.3.4. Turbine auxiliaries

Turbine auxiliary systems such as low oil pressure system (for turbine bearing cooling) and high oil pressure system (for electrohydraulic fluid) are out of the simulator's scope.

Grid instabilities and turbine vibrations are out of scope as well.

3.3.5. Off-site power, diesel and batteries lights

The simulation of off-site power, diesel and batteries lights is just logic for good understanding of Station Blackout (SBO).

4. MAIN FEATURES OF THE SIMULATOR

4.1.SIMULATOR OPERATION

iPWR-SMR simulator has mainly four characteristics:

(a) Two plant operating modes: Turbine leading mode and reactor leading mode;
(b) Possibility of chosing between natural circulation or forced circulation for core cooling;
(c) Possibility of chosing between close or open loop for circulating water;
(d) Possibility of changing certain setpoints.

4.1.1. Operating modes

iPWR-SMR simulator has two plant operating modes: Turbine leading mode and reactor leading mode.

When turbine leading mode is selected (this is the normal operation of a typical PWR):

(a) Turbine valve (MSSV07) controls generator power (MW(e));
(b) Turbine bypass valve (MSSV08) controls average temperature;
(c) Rods control average temperature.

However, when reactor leading mode is selected (this is the normal operation of a typical BWR):

(a) Turbine valve (MSSV07) controls steam header pressure (MPa);
(b) Turbine bypass valve (MSSV08) controls steam header pressure (MPa);
(c) Rods control difference between reactor power and power demand.

4.1.2. Core cooling

iPWR-SMR simulator has two core cooling options: Natural circulation (no Reactor coolant pumps (RCPs)) or forced circulation (4 RCPs).

The selection of having RCPs or not is made through a selection of appropriate initial condition. If user wants natural circulation, he/she must load an IC with no pumps. On the contrary, if the user wants forced circulation, he/she must load an IC with pumps.

First case will show a reactor vessel with no pumps in display and second case will show a reactor vessel with 4 RCPs above the core.

4.1.3. Circulating water cooling

iPWR-SMR simulator has two circulating water cooling option: Open loop and close loop.

There is a 3-way valve in order to change from one cooling mode to another. Open loop will take the water from an open water body shown as 'SMR lake' in the simulator, whereas the close loop will be connected to a cooling/refrigerating tower.

4.1.4. Setpoints

iPWR-SMR simulator allows user to change some setpoints, for example reactor trips or ADS actuation setpoints. PZR pressure and level control can also be changed by the user.

4.2.CONFIGURATION MENU

Configuration menu is next to alarm system and have the following functionality (Fig. 33):

FIG. 33. Configuration menu.

(a) RUN/FREEZE button and simulation time;
(b) Config dropdown menu contains (Fig. 34):
  (i) New page;
  (ii) Trends;
  (iii) Export to excel;
  (iv) Help;
  (v) Initial conditions;
  (vi) Backtrack;
  (vii) Speed control;
  (viii) Controlled parameters;
  (ix) Generic malfunctios;
  (i) Specific malfunctions.

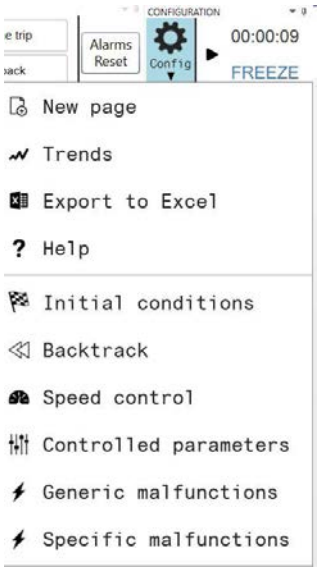


FIG. 34. Configuration menu.

4.2.1. New page

This function allows the user to open a new set of displays on another monitor.

4.2.2. Trends

This function allows displaying any number of trend windows. Inside each window, it is possible to display several trend areas. At the same time, it is possible to display several curves inside each trend area (Fig. 35).

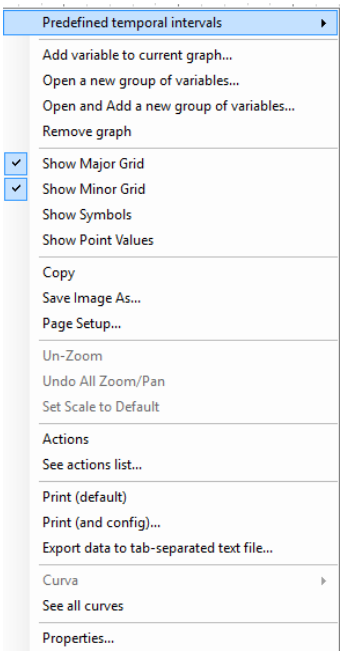


FIG. 35. Graphic trend menu.

4.2.2.1.Predefined temporal intervals submenu

It allows changing the graphic temporal interval. Several temporal intervals are predefined, but the user can also manually select a different interval (Fig. 36).

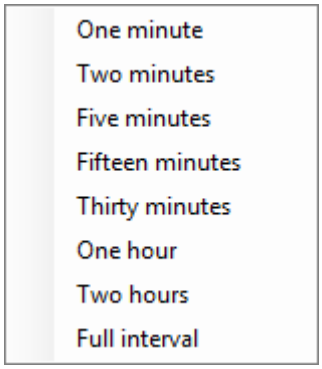


FIG. 36. Graphic trend time.

4.2.2.2.Add variable addition to current graph

It allows the addition of a new variable to the active trend group (Fig. 37).

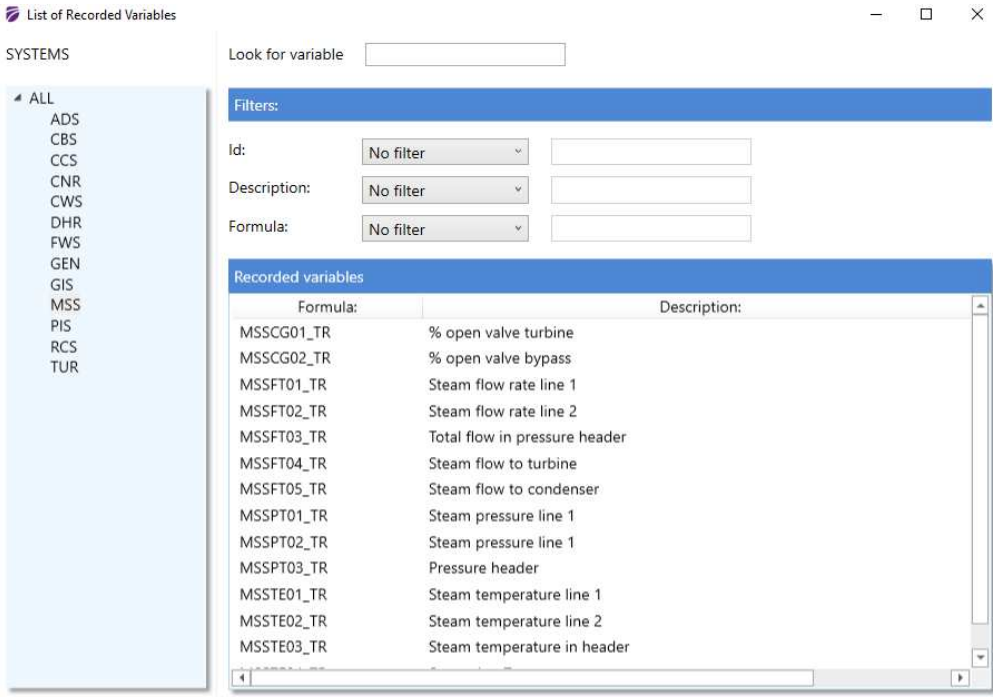


FIG. 37. Trend graphic variables.

The user has to select the variable to be added to the trend graph by clicking the left mouse button on the required variable, dragging and dropping it to the trend graph, without releasing the click.

The variable will be added to the trend group and the scale will be automatically adjusted (unless indicated otherwise in the trend properties).

4.2.2.3. Open a new group of variables

It allows loading a trend group from a file.

4.2.2.4. Remove graphs

It deletes the active trend group. A confirmation is requested before the deletion.

4.2.2.5. Show major grid

It enables or disables the display of the major trend grid.

4.2.2.6. Show minor grid

It enables or disables the display of the minor trend grid.

4.2.2.7. Show symbols

It enables or disables the display of symbols on the curves.

4.2.2.8. Show point values

When this option is active, it allows the user to show a point value in a tooltip when placing the cursor on the graphic.

4.2.2.9. Copy

It copies an image of the active trend to the clipboard.

4.2.2.10. Save image as

It saves an image of the active trend in a file.

4.2.2.11. Page set-up

It allows the configuration of the page layout (landscape or portrait, margins…) to the printing.

4.2.2.12. Print

It prints the graphic.

4.2.2.13. Print (and config)

It opens the print configuration window.

4.2.3. Export to excel

User is able to record all the simulator variables in an excel format. Microsoft excel must be pre-installed.

4.2.4. User manuals

User manuals (theory or exercise) will open.

4.2.5. Initial conditions

An IC is a group of values of the variables needed to initialize the mathematical models of the simulator for a specific state of the plant. Before starting any simulation session, the simulator must be initialized in a specific starting point.

This function includes two types of operations:

(a) Initialize the simulation with an Initial Condition: LOAD IC function;
(b) Save an Initial Condition for a later load: SAVE IC function.

4.2.5.1. Interface description

When accessing to functions related to initial conditions, a window will be displayed with a list of all the initial conditions available.

The initial conditions window is divided into four sections (Fig. 38):

(a) Control section;
(b) Load IC/Save IC;
(c) LOADED IC with the information of the last initial condition loaded;
(d) IC SUMMARY with a list of the initial conditions.

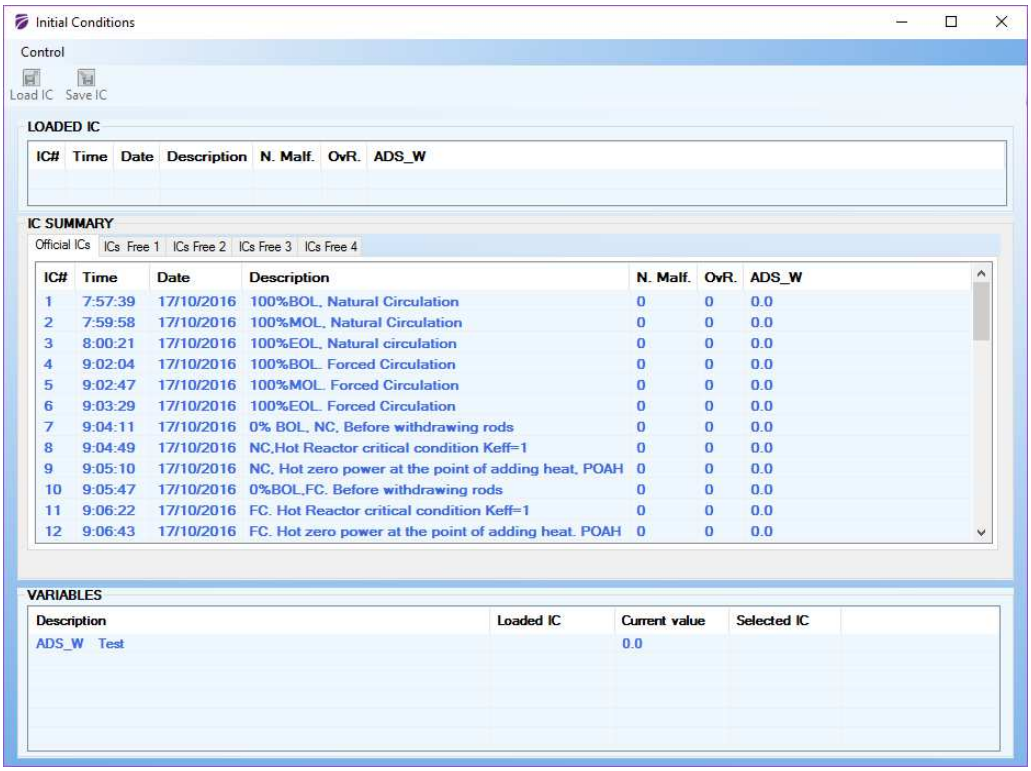


FIG. 38. Initial Conditions view.

4.2.5.2. Control section

It contains a button Control that pops up the following options:

(a) Reset IC: It is the same as load IC;

(b) Reset Photo IC: Functionality out of scope;
(c) Snapshot IC: It is the same as save IC;
(d) Delete IC: It removes an IC (Delete IC is only enabled when there is an initial condition selected in the ICs list);
(e) Refresh IC List: It refreshes the list of initial conditions if no text in ICs appears;
(f) Passwords: It allows modifying the passwords for the groups of the initial conditions;
(g) Variables: Functionality out of scope;
(h) Save scenario data: Functionality out of scope;
(i) Exit: It closes the initial conditions window.

4.2.5.3. Tool bar

It contains two buttons:

(a) Load IC;
(b) Save IC.

These buttons are only enabled when there is an initial condition selected.

4.2.5.4. Selecting an initial condition and loading

User has to select Config menu → Initial conditions → IC SUMMARY → Select one IC → Load IC.

4.2.5.5. Save an initial condition

User has to select Config menu → Initial conditions → IC SUMMARY → Select one IC which is available → Save IC → Introduce the description of the new initial condition in the window displayed.

The IC saved with its description will appear in the list of ICs.

4.2.5.6. Deleting an initial condition

Select an initial condition → Select Control → Delete IC.

4.2.6. Backtrack

These functions allow restarting the simulation to some of the available previous states (backtracks).

A backtrack is a copy of the state of the simulation variables in a specific moment of the simulation session. The backtracks are stored in files inside a specific directory in the same way as the initial conditions.

The environment allows to select the time period between two consecutive backtracks and also to specify the number of them to preserve, defining the maximum time available for the backtrack function. The recording interval is one minute by default, but it is possible to change this time during the session by modifying the option interval between backtracks.

It has a button Control that is displayed with the following options:

(a) Reset BKTK;
(b) Replay;
(c) Exit.

The two firsts option are only enabled when there is a snapshot selected. It also contains three quick access buttons to the following options (Fig. 39):

(a) Freeze simulation;
(b) Reset BKTK;
(c) Replay.

The functions Reset BKTK and replay read the previous states selected by the operator and restart the simulation to that state:

(a) Reset BKTK: the operator continues the session as if he had loaded an initial condition;
(b) Function replay: it is possible to load sequentially the snapshots saved from the state selected, the colour of the row changes as the snapshot is being executed in the list of backtracks.

When selecting any of the functions, the window Backtrack Summary will be displayed. It is divided into three sections.

(a) Main menu with the different actions of the function: Control, Freeze Sim, reset BKTK, replay and exit. Reset BKTK and replay are only enabled if there is a snapshot selected;
(b) Backtracks area: it displays the available snapshots grouped by initial condition:
    (i) The whole list of backtracks available are displayed in this area;
    (ii) They are grouped in different tabs, one for each initial condition loaded during the simulation session. For each backtrack, its identification number is displayed (or order occupied in the circular data file).
(c) Backtrack interval area: it contains controls for changing the interval between records. The current value (seconds) of the backtrack interval and an entry to modify this value appears at the bottom of the window. Besides, there is a button to apply the change;

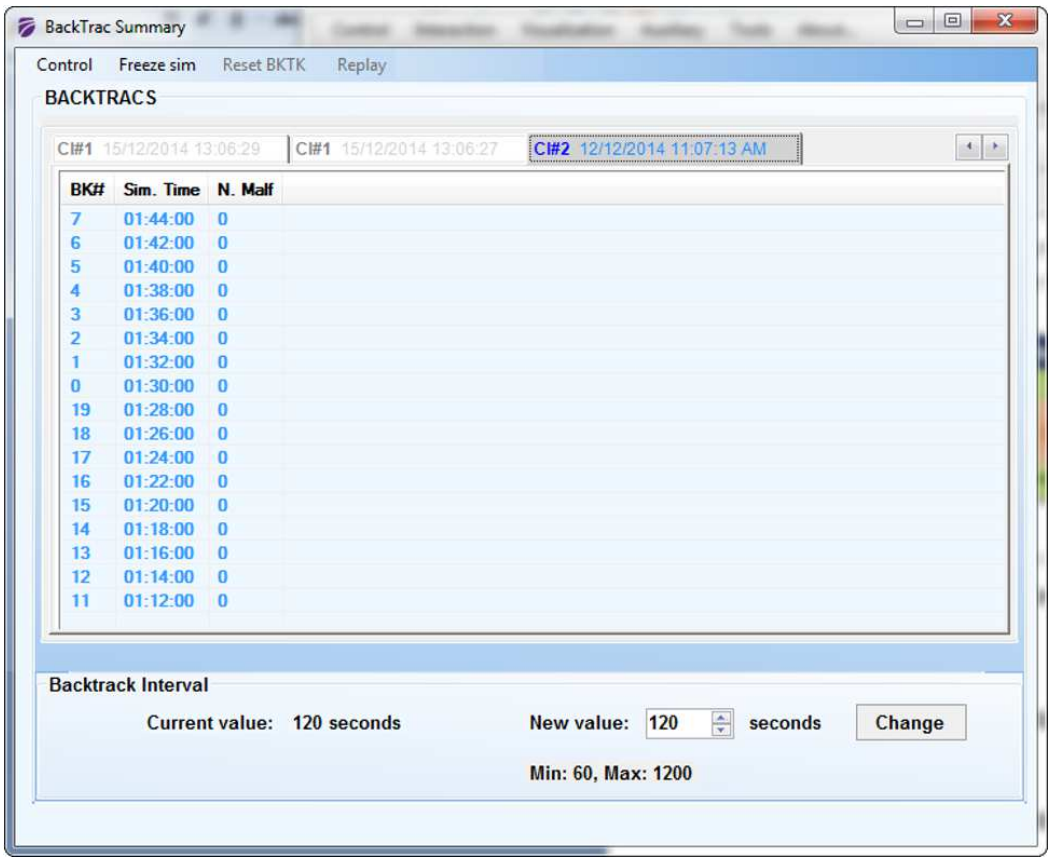


FIG. 39. Backtrack window.

(d) Replay window: if a backtrack record has been selected and the user presses the option Replay, a window will appears. This window allows controlling the replay actions (Fig. 40).

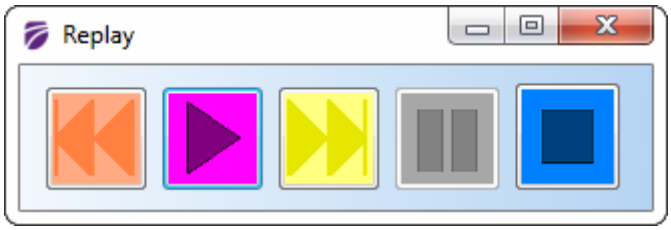


FIG. 40. Bactrack buttons.

The window contains five icons:

 . Backward  . Play  . Forward

 . Pause  . Stop

Freeze the simulation: select the quick access button Freeze Sim from the main menu. The simulation state changes to FREEZE;

(a) Select an initial condition tab: select the desired tab to access different backtracks;
(b) Select a backtrack: click on the records list, notice the change of colour and click on Reset BKTK;
(c) Go back to a previous record: select a backtrack → Reset BKTK in the main menu of the window Backtrack Summary. The sequence to load the initial condition is started. The state of the simulations changes to FROZEN;
(d) Repeat a set of previous states: select a backtrack record, click on the replay button in the window Backtrack Summary. Select the Forward or Backward button. In the records list appears marked the following record to the one selected. The simulation will read the records saved from the selected record. The state of the simulation changes to REPEATING. Select the button Stop and the simulation state will change to FREEZE;
(e) Changing the backtrack interval. Select the interval time between backtracks (minimum value is 60 seconds), press the change button to apply the new value.

4.2.7. Speed control

This function allows the user to change the simulation speed. There are three different modes (Fig. 41):

(a) Real time: Simulation is running in normal mode. One second of simulation is equivalent to one second of real process;
(b) Slow time: Simulation runs slower than real time. User has several options to slowdown the simulation: from 2 to 10 times;
(c) Fast time: Simulation runs faster than real time. User can select between 2 options: make the simulation twice faster or make it as fast as the user´s computer can.

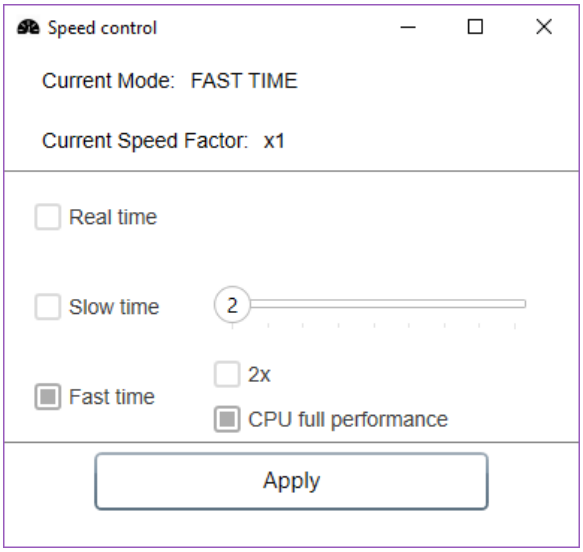


FIG. 41. Speed control window.

If the performance of the user´s computer is not enough, it may happen that if 2x option is selected, some simulation modules will end (in this case, user will have to load an initial condition or backtrack). For this reason, it is convenient that before selecting 2x option, user

selects CPU full performance and read in the number that appears in the current speed factor. If this number is lower than 2x, user should not select this option.

4.2.8. Controlled parameters

This function allows the operator to modify the value of specific parameters as the ambient temperature, tanks level etc.

The window showing the controlled parameters consists of two sections (Fig. 42):

(a) Systems tree: It allows filtering the list of controlled parameters according to the system;
(b) Controlled parameters list: If user wants to change any controlled parameter, he/she must roll the mouse over the value, change it and then push apply.

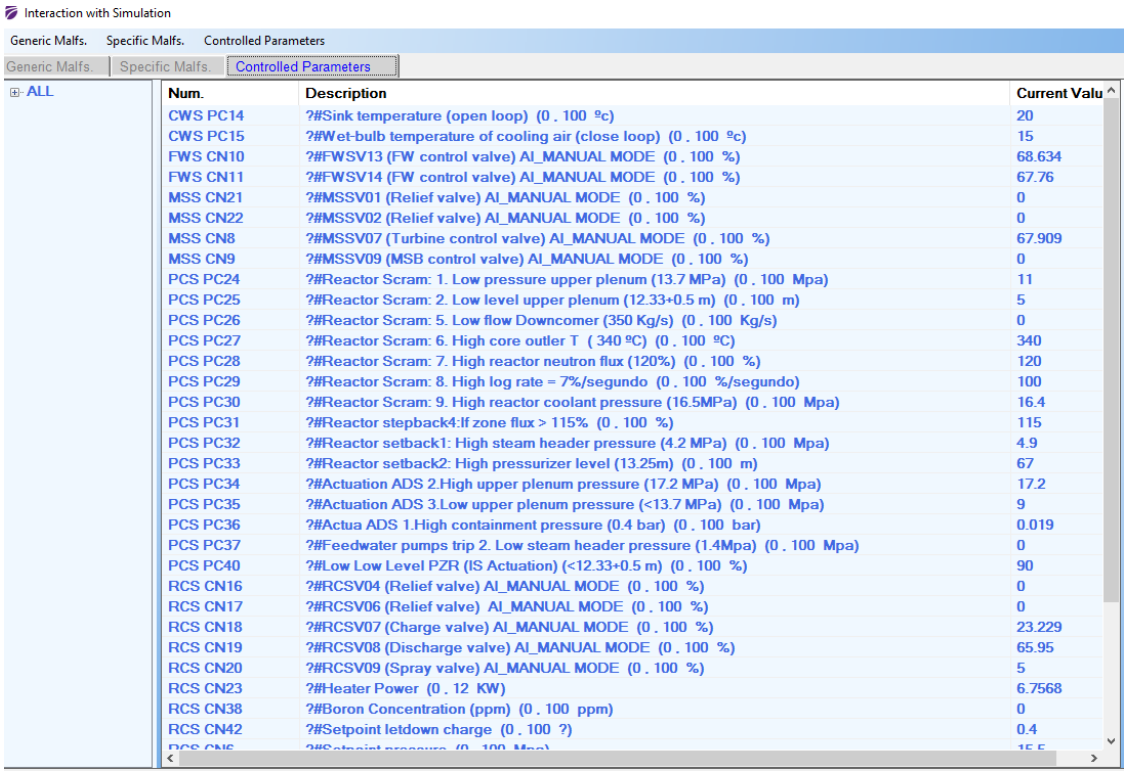


FIG. 42. Control parameters window.

4.2.9. Generic malfunctions

The user interface for generic malfunctions consists of two windows: one to display the malfunctions list, the second one to define the activation parameters of the malfunctions (Fig. 43).

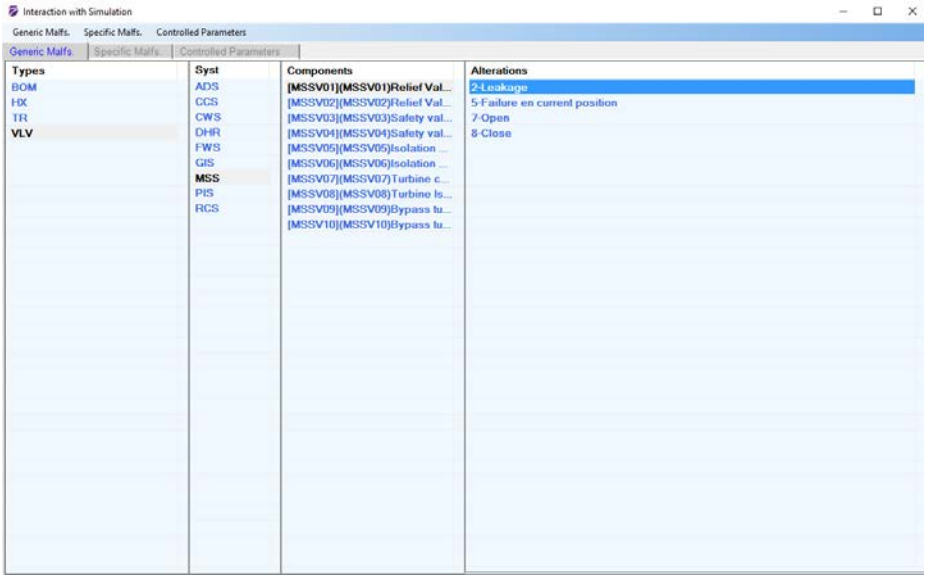


FIG. 43. Generic malfunctions.

User has to select the malfunction required and the following window will appear (Fig. 44):

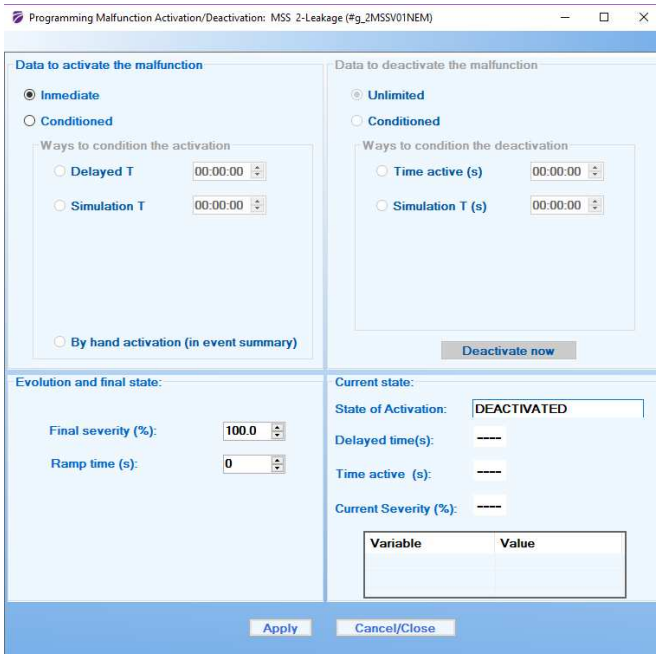


FIG. 44. Example of generic malfunction.

Valve leaks or heat exchanger breaks have severity, user can break tubes just a little bit (severity =1%) or more (severity = 100%). There are other malfunctions like valve opening etc. that do not have the severity feature.

4.2.10. Specific malfunctions

Specific malfunctions can be loaded from configuration menu or from the display pages.

If user loads the malfunction from configuration menu the following window will appear (Fig. 45):

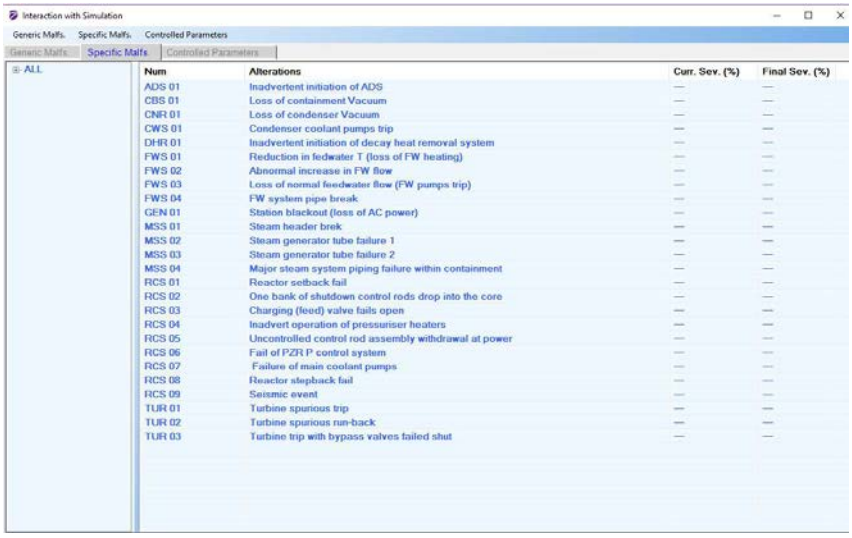


FIG. 45. Specific malfunction.

User has to select the malfunction required and the following window will appear (Fig. 46):

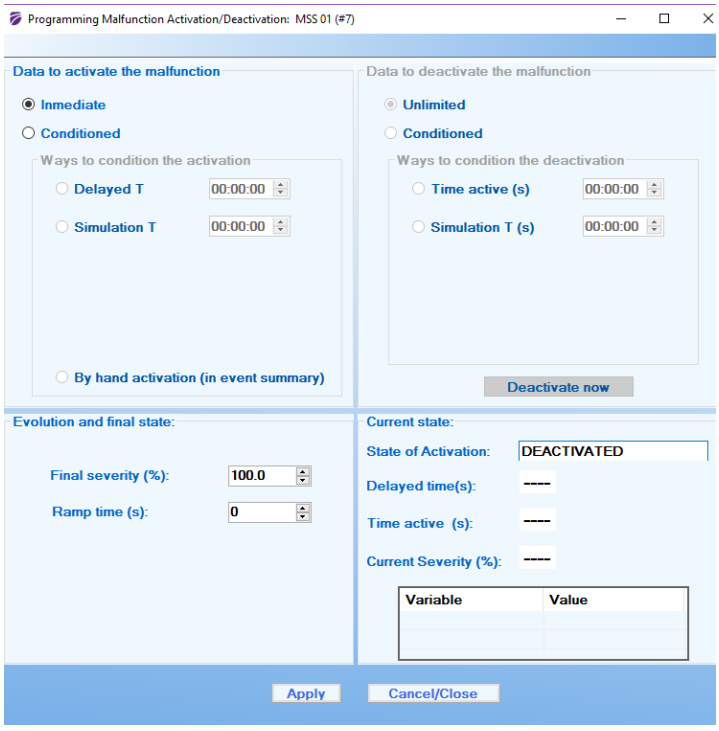


FIG. 46. Example of specific malfunction.

Pipe breaks have severity, user can break a pipe just a little bit (severity =1%) or totally (severity = 100%). There are other malfunctions like inadvertent actuation of ADS etc. that do not have severity feature.

4.2.11. Restore layout

If user changes the appearance of the application, user can return to the initial configuration just pressing Ctrl+Alt+r.

4.3.LIST OF IPWR-SMR DISPLAY SCREENS

The GUI of this simulator has been designed to ensure that the operational interfaces consider human limitations and to make the best use of human abilities, especially due to the complexity of nuclear power simulator operation.

The entire interface has been designed applying the state of-the-art in human factors engineering (HFE) principles and methods. The design is based on international standards, in compliance with the recommendations defined in human-system interface design review guidelines NUREG-0700, Rev.2, Chapters 1 and 2. These chapters contain criteria to allocate the information in the displays and criteria related to the interaction between the users and the system.

The iPWR-SMR simulator is made up of 11 interactive display screens or pages. All of these screens have the same information at the top and left of the displays, as follows:

(a) Top of the screen contains configuration menu and 16 alarms and a reset button. These indicate important status changes in plant parameters.
(b) Left of the screen is the navigation area and it shows:
    (i) The values of the following major plant parameters:
        — Reactor neutron power (%);
        — Reactor thermal power (%);
        — Generator power (MW(e));
        — Steam header pressure (MPa);
        — RPV water level (%).
    (ii) The state of the off-site power, diesel and batteries availability;
    (iii) The bottom left hand corner allows the initiation of two major plant events:
        — Reactor trip;
        — Turbine trip.
    (iv) Display list:
        — Home;
        — Overview;
        — Controls;
        — Core;
        — Trips;
        — Systems.

4.3.1. Home display

This screen appears once the simulator starts-up. This display shows the general layout of the plant with arrows indicating the flow path (Fig. 47).

When one system of home display is selected, the screen changes to system display. System display is divided into three areas:

(a) First one for RCS, ADS, PDHR, GIS, PIS and CCS;

(b) Second one for MSS, TUR, GEN, CNR;
(c) Third one for CWS.

For example if user selects MSS from Home page, the part of the system display that appears will be the second. On contrary, if users select ADS, the part of the system display that appears will be the first.

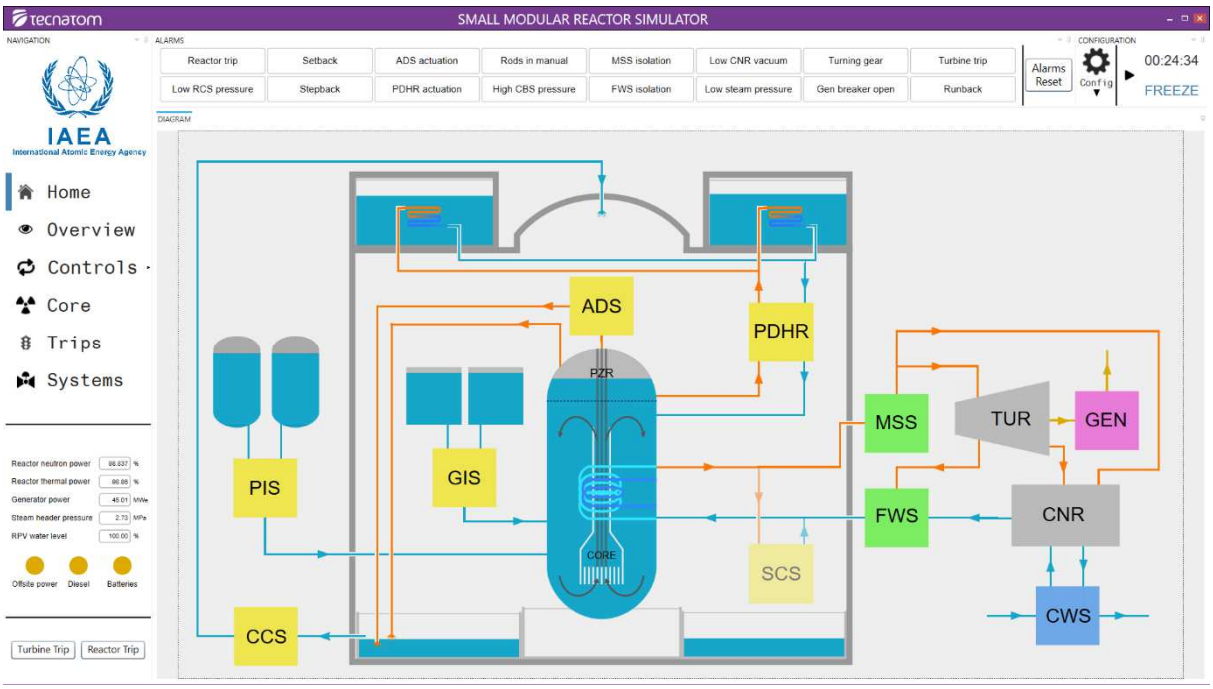


FIG. 47. Home display.

4.3.2. Overview display

This display shows the general layout of the plant and the main parameters of every system. This display is just for monitoring the plant behaviour (Fig. 48).

Each system has a dynamic text that changes with the plant status as shown in TABLE 1:

TABLE 1. DYNAMIC TEXTS

System	State_1	State_2
RCS	POWER	TRIP
MSS	NOT ISOLATED	ISOLATED
FWS	NOT ISOLATED	ISOLATED
GEN	SYNC	NO SYNC
CWS	OPEN LOOP	CLOSE LOOP
CNR	VACUUM	NO VACUUM
CBS	NORMAL	EMERGENCY
PIS	OFF	EMERGENCY
GIS	OFF	EMERGENCY
ADS	OFF	EMERGENCY
CCS	OFF	EMERGENCY
PDGR	OFF	ON
TUR	SYNC	DECELERATING
TUR	RUN-UP	TURNING GEAR

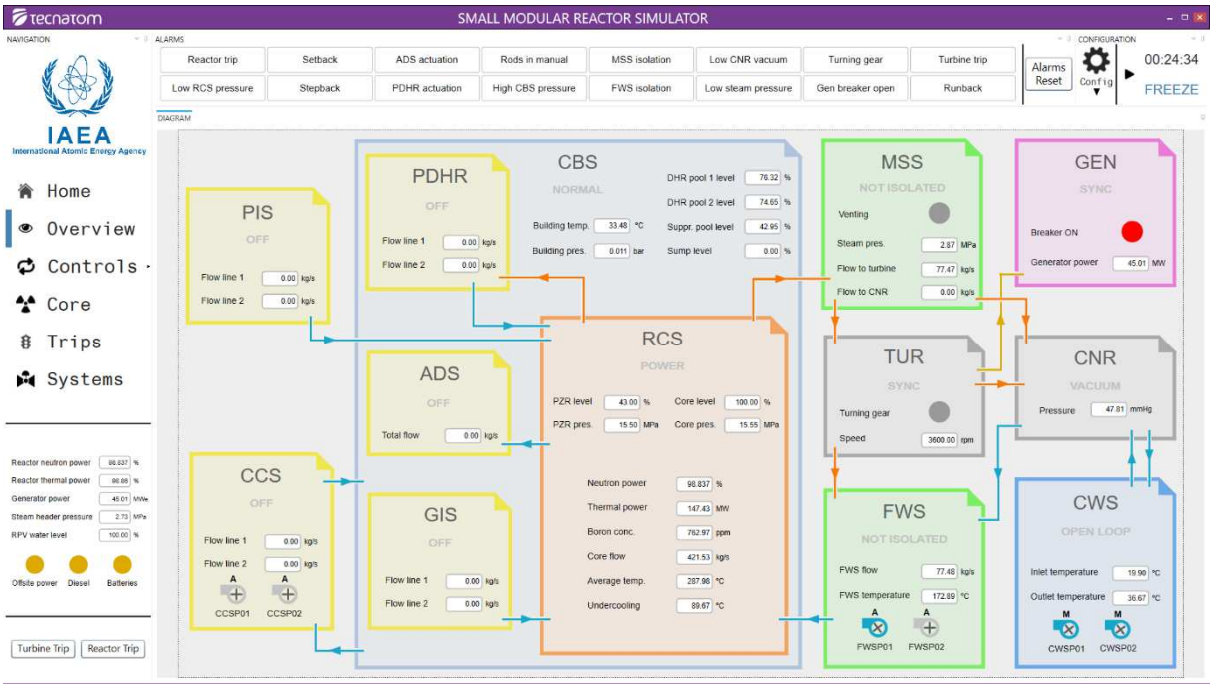


FIG. 48. Overview display.

4.3.3. Core display

This display shows neutronic and thermal hydraulic variables from the core. It is divided into two areas (Fig. 49):

(a) The right side:
  (i) Boron concentration can be modified by user, boration or dilution:
    — If user writes a boron concentration lower than actual (dilute) but selects boration in selector, system will not do anything;
    — If user writes a boron concentration greater than actual (borate) but selects dilution in selector, system will not do anything.
  (ii) Reactivity values: Control rods, fuel temperature, moderator temperature, boron, xenon and total;
  (iii) Fuel temperatures.

(b) The left side:
  (i) User should choose one variable from the dropdown list and one height (z) from the core height. Once these two options have been selected, the variable result will be shown in the core plane (xy) selected;
  (ii) Besides, the scale limitis can be modified from relative to absolute;
  (iii) In the core height, the axial flux is shown;
  (iv) ΔI graph represents the difference between nuclear power in upper half of the core and lower half.

$$\Delta I = \frac{Is - Ii}{I100} \quad (1)$$

where

ΔI is Delta I;
Is is upper half of the core nuclear power;

Ii is lower half of the core nuclear power;
and I100 is power at 100% full power.

ΔI should be controlled as follows:

(i) If ΔI is at the left side of the target band, rods are excessively inserted. This means that rods depress neutron flux in the upper half of the core and force a high flux to exist in the lower half. To make ΔI return to the target band, borate the RCS to make rods withdraw from the core;

(ii) If ΔI is at the right side of the target band, rods are excessively withdrawn. This means that a higher flux exists in the upper half of the core compared to the lower half. To make ΔI return to the target band, dilute the RCS to make rods insert into the core.

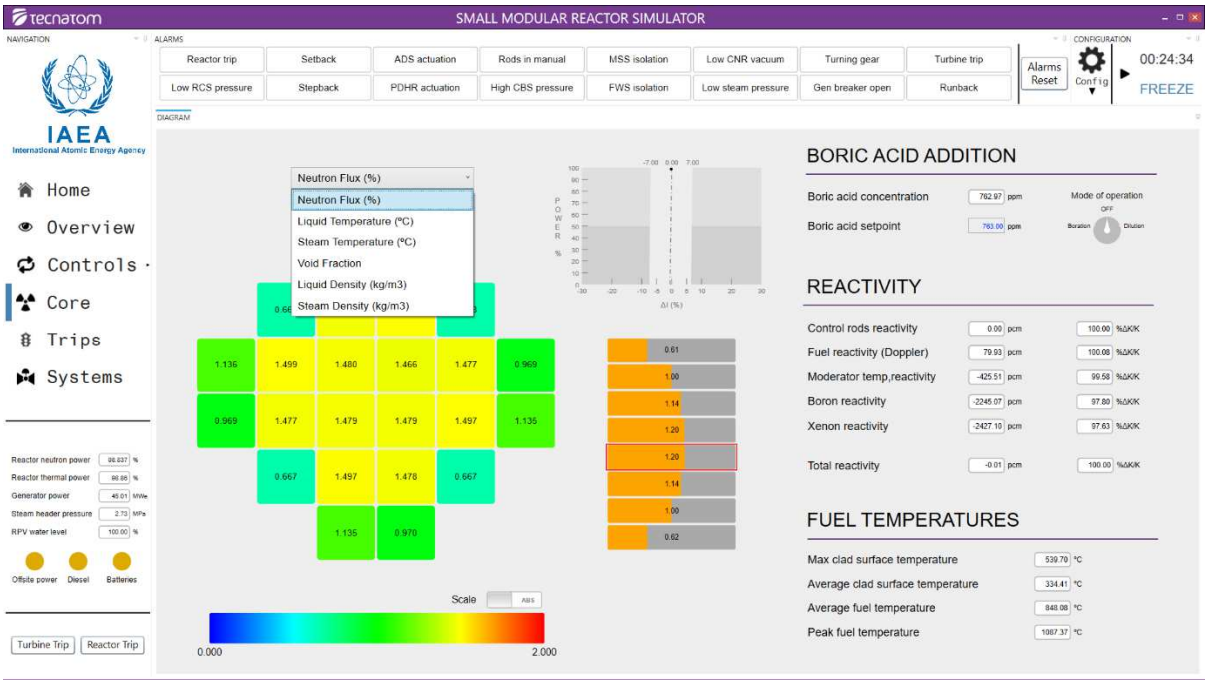


FIG. 49. Core display.

4.3.4. System display

This display shows every plant system and all the system components in just one page. User can scroll from primary and safety systems area to circulating water part using the scroll bar at the bottom (Fig. 51, Fig. 52 and Fig. 53).

Any part of the system display can also be reached from home page or just clicking system display from the display list.

Components can be operated by clicking over them. A pop-up menu will appear, allowing opening/closing a valve, starting/stopping a pump or changing the controller from manual to auto etc. showing:

(a) At the top of each component, there is an A, if component is in auto, or an M, if component is in manual;
(b) If a component is closed or stopped, its colour will be grey;

(c) If a component is opened or running, its colour will be the same as the pipe colour over it:
   (i) Blue – Water;
   (ii) Orange – Steam;
   (iii) Yellow – Tension.
(d) If a component is malfunctioned, its colour will be red;
(e) If there is no power, colour will be white (SBO).

There are two types of readouts:

(a) Readouts just for monitoring; 

(b) Readouts that a user can change value. 

Malfunctions can be introduced from GUI pages by just clicking over the malfunction symbol. Once the MF has been chosen, a description about what will take place will appear as pop-up menu (Fig. 50).

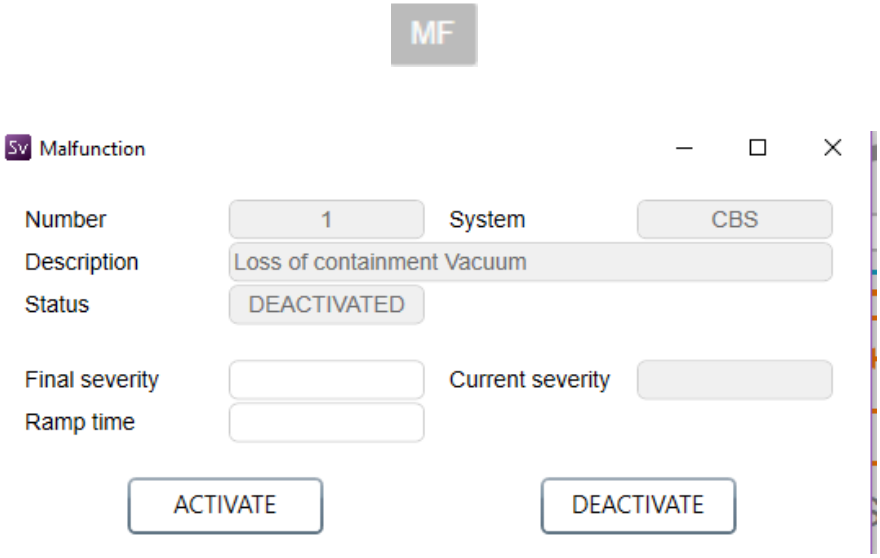


FIG. 50. Example of malfunction.

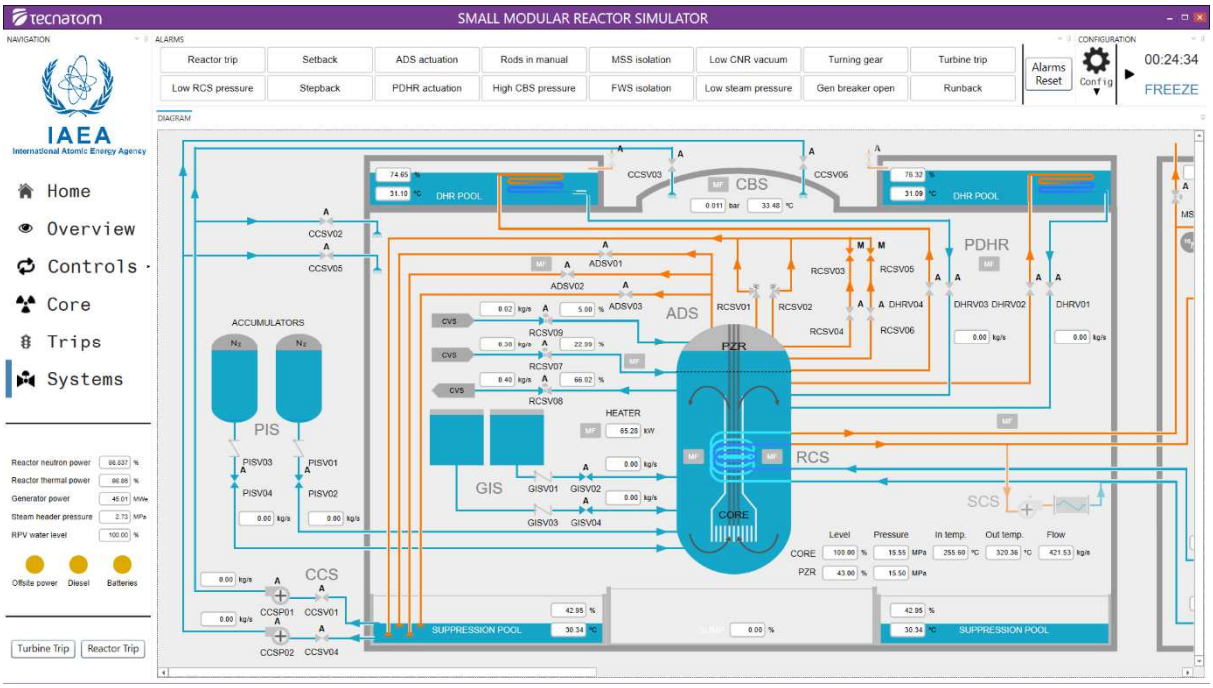


FIG. 51. System display – primary side.

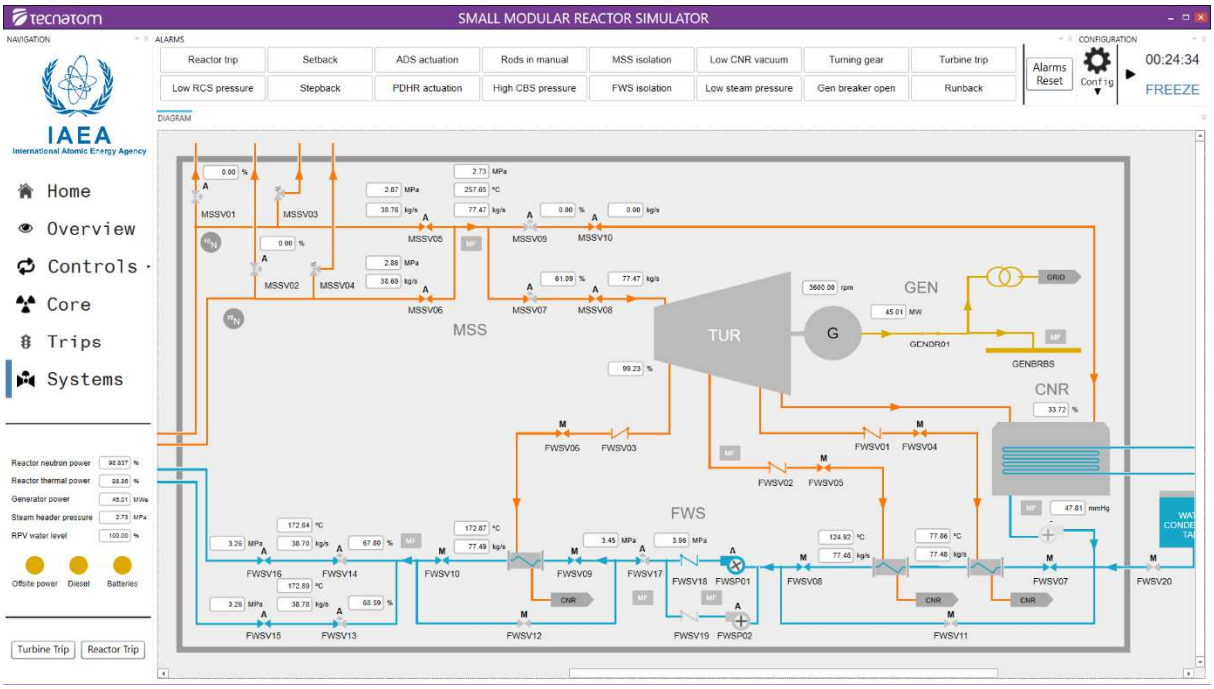


FIG. 52. System display – secondary side.

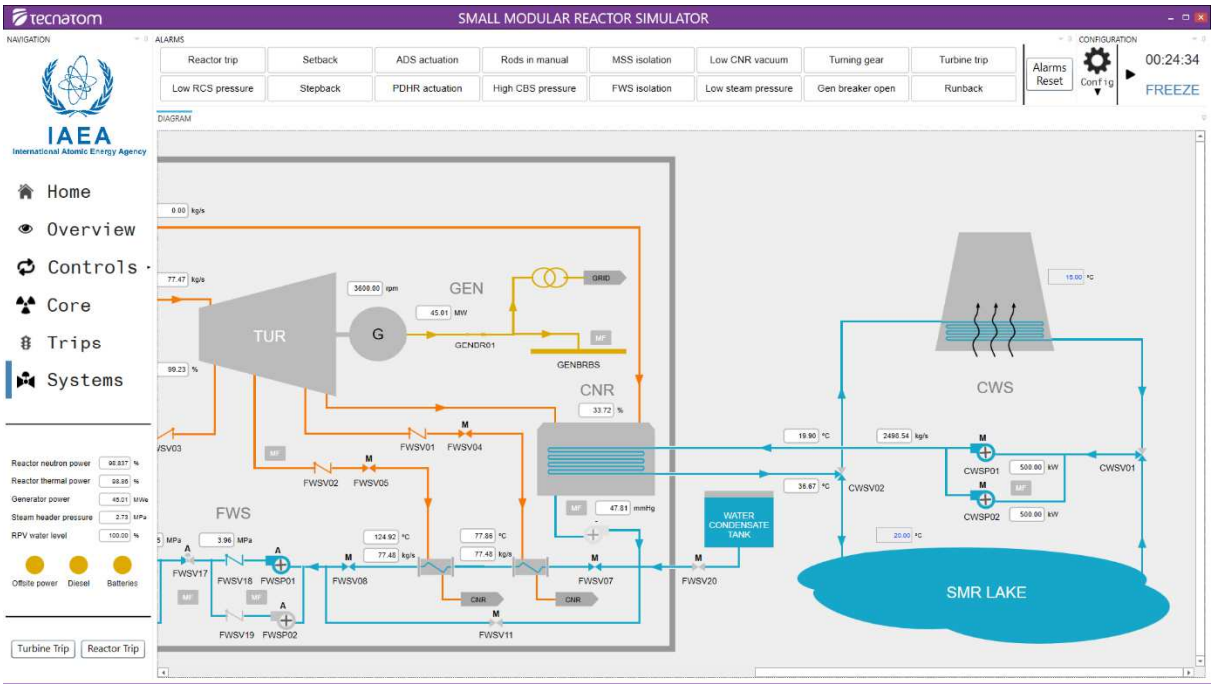


FIG. 53. System display – tertiary side.

4.3.5. Trips display

This display shows reactor trip, turbine trip, reactor setback, reactor stepback, ADS actuation and PDHR actuation causes (Fig. 54).

Next to each cause, there is the real value of the variable and the set point at which the action takes place. Setpoints can be modified by user.

Example: If there is reactor trip due to low pressure in the upper plenum at 11.0 MPa, user can modified this value (between some limits) by clicking in the area and inserting the desired value.

The plant shall be able to reduce reactor power when certain setpoints will be exceeded (in these cases a full reactor trip is not necessary):

(a) Reactor stepback: reduction of reactor power in a large step;
(b) Reactor setback: ramping of reactor power at a fixed rate to a setback target.

User can perform the following malfunctions from here:

(a) Reactor setback fail;
(b) Reactor stepback fail;
(c) Turbine spurious runback.

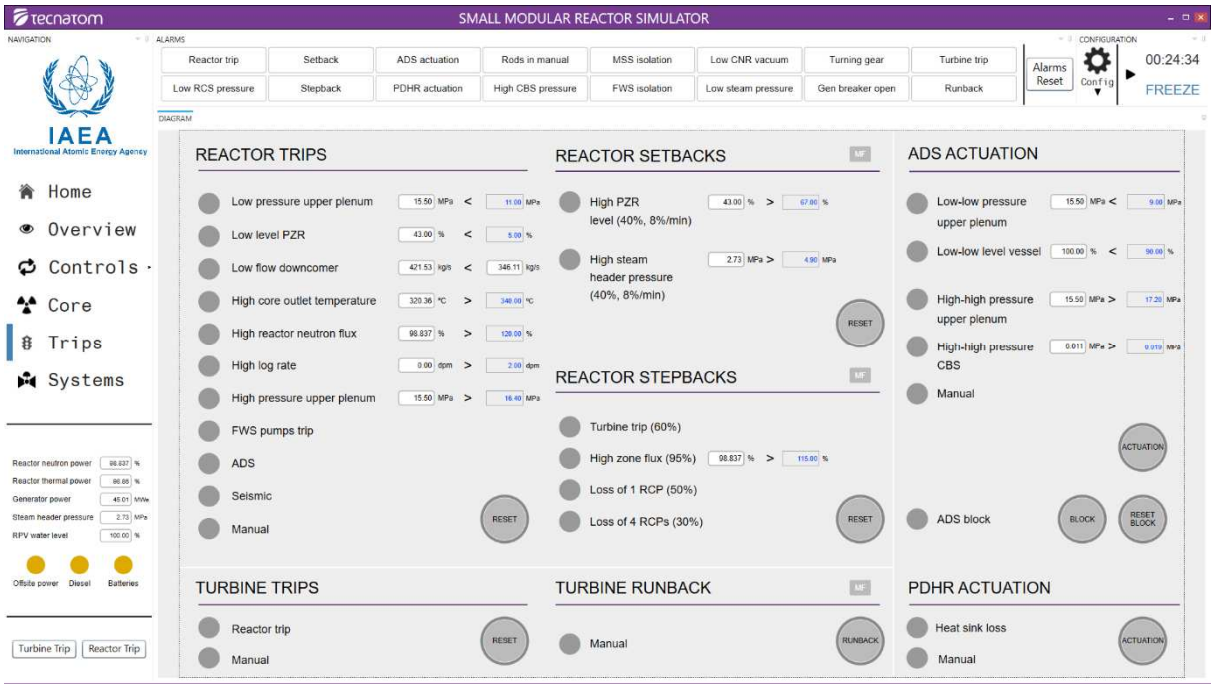


FIG. 54. Trips display.

4.3.6. PZR pressure control

This screen shows the parameters relevant to controlling the coolant pressure. In order to control pressure, this system requires a spray valve, heaters, two relief valves and two safety valves (Fig. 56).

Heaters and spray valve have a controller (PID) which is normally in auto and whose set point is 15.5 MPa. This set point can be change by user, using the pop-up menus (Fig. 55).

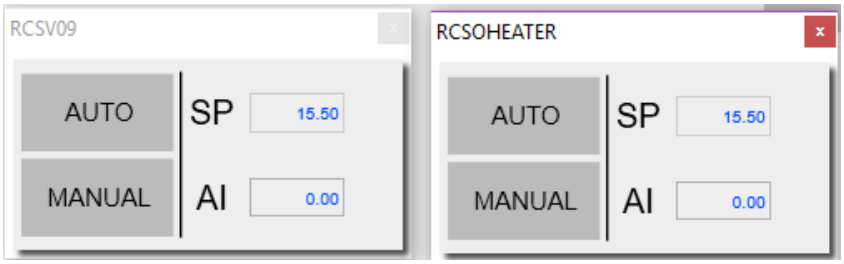


FIG. 55. Spray and heater controller.

User can change the control to manual, by clicking manual button and introducing the valve position (%) required in AI part;

(a) User can perform the malfunction of failure of PZR control system;
(b) PZR pressure vs reference pressure is plotted in 10 minutes interval;
(c) Heaters, spray and relief valves can be operated from this display;
(d) Pressure setpoints are shown in TABLE 2.

TABLE 2. PRESSURE SETPOINTS

Component	set point
Heaters	P <15.5 MPa
Spray	P >15.5 MPa
Relief	P>16.7 MPa
Safety	P>17.05 MPa

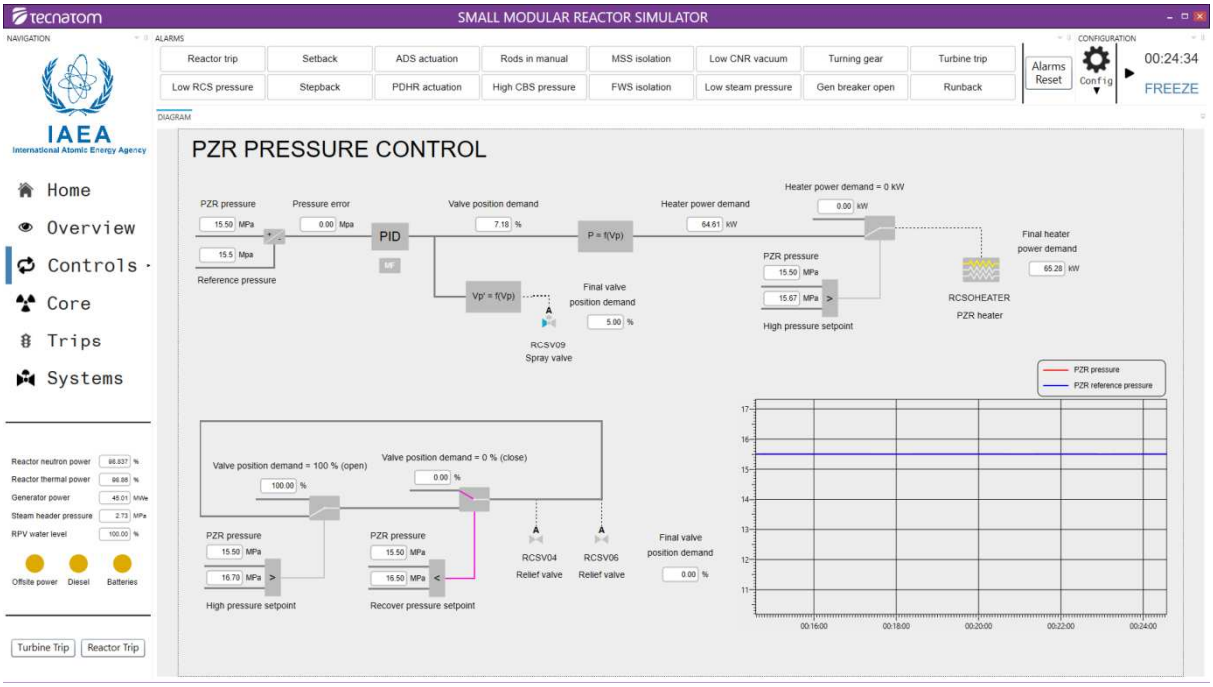


FIG. 56. PZR pressure control display.

4.3.7. PZR level control

This screen shows the parameters relevant to the control of the inventory. Inventory control is achieved by controlling pressurizer level (Fig. 57).

(a) Pressurizer level is normally under automatic control with set point being ramped as a function of reactor power and the expected shrinks and swells from the corresponding temperature changes;
(b) Level control may be transferred to manual and the set point can then be controlled manually;
(c) Charging valve (RCSV07) is normally in auto position and it controls that the PZR level follows the program;
(d) Letdown valve (RCSV08) is normally in auto position and it controls a fixed flow of 0.4 Kg/s;
(e) There are two graphs with 10 minutes interval:
    (i) First (above): PZR level vs reference level;
    (ii) Second (below): Charge and letdown flow.

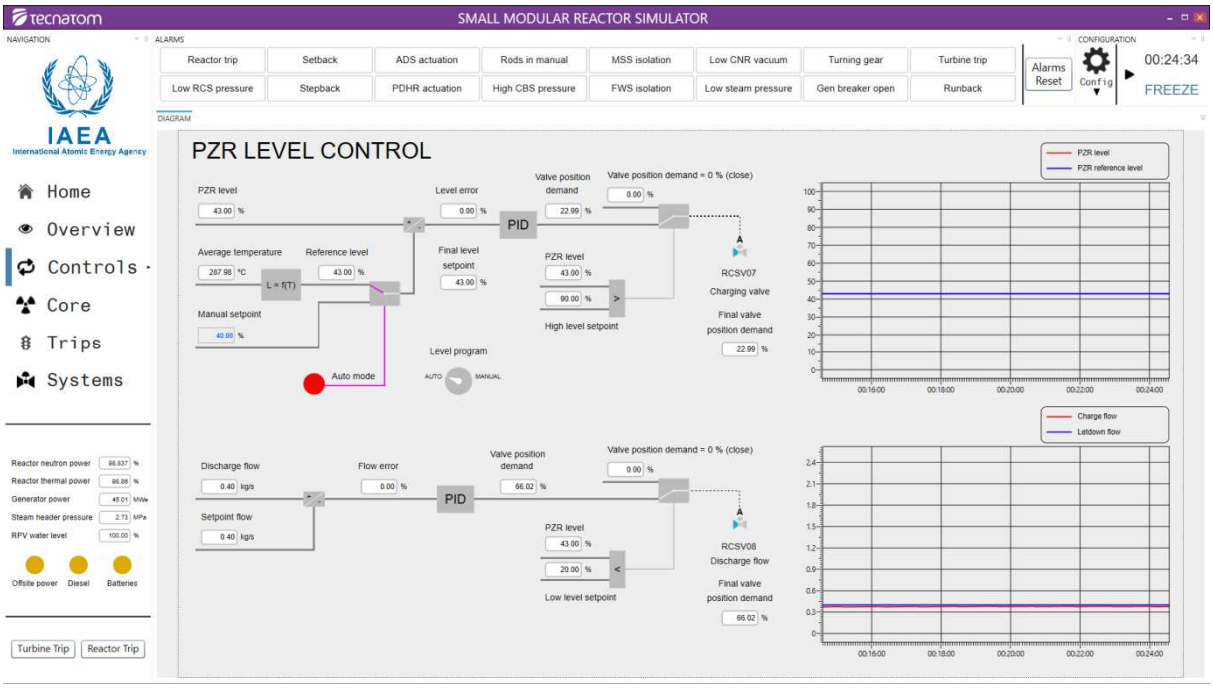


FIG. 57. PZR level control display.

4.3.8. Feedwater control

This screen shows the main parameters and controls associated with feedwater control (Fig. 58). Feedwater control is made through, feedwater pump speed and FWSV17 and FWSV13/14 valves position.

(a) Pumps speed ensures a proper pressure drop in FWSV17;
(b) FWSV17 ensures a proper steam reheating;
(c) FWSV13/14 ensure same pressure drop in each helicoid;
(d) Pumps and FWSV13/14 can be operated from this display;
(e) There are two graphs with 10 minutes interval:
    (i) First (above): Pumps speed;
    (ii) Second (below): Steam and water flows of each line.

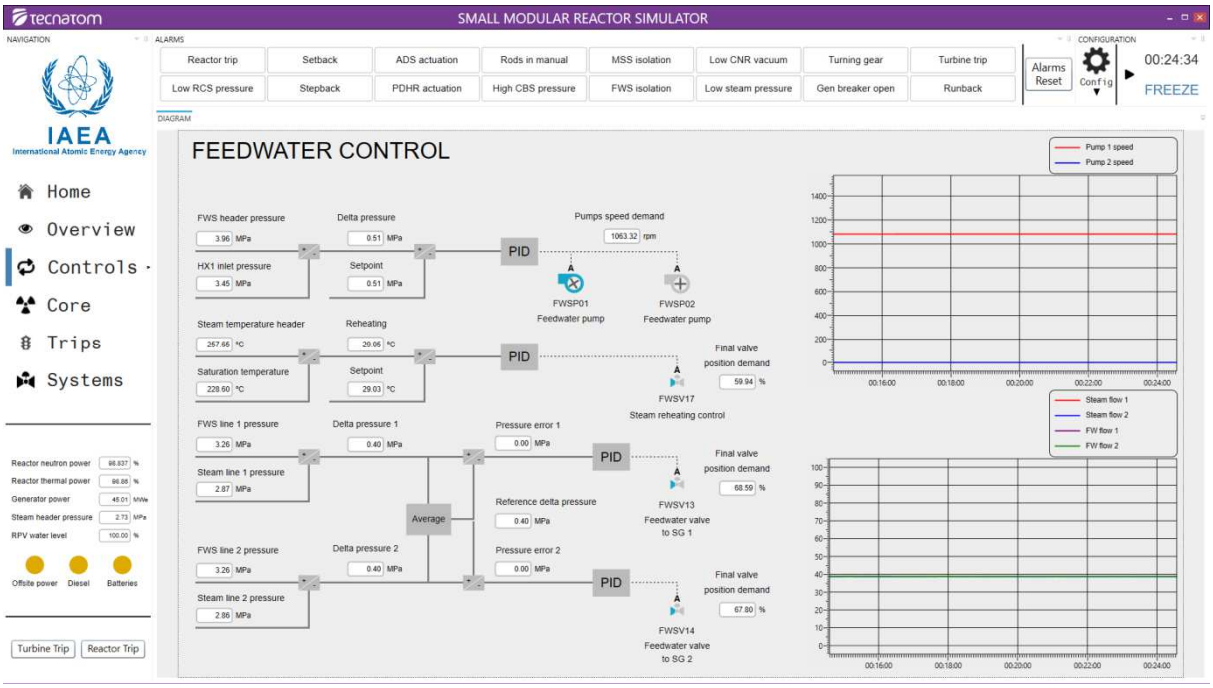


FIG. 58. Feedwater control display.

4.3.9. Turbine control

This screen shows the main parameters and controls associated with turbine (Fig. 59).

Turbine valve (MSSV07) can be controlled in three ways:

(a) If plant mode selector (it is located in rod position control display) is in turbine leading mode and generator breaker is closed, turbine valve controls turbine load (MW(e));
(b) If plant mode selector (it is located in rod position control display) is in-reactor leading mode and generator breaker is closed, turbine valve controls steam header pressure (MPa);
(c) If generator breaker is opened, turbine valve controls turbine speed (rpm).

User can perform the following actions:

(a) The turbine run-up manoeuvres and synchronization;
(b) Actuate two malfunctions:
    (i) Turbine spurious trip;
    (ii) Turbine trip with bypass valves failed close.
(c) Program load reduction introducing MW and MW/min, if turbine leading mode is selected;
(d) There are two graphs with 10 minutes interval:
    (i) First (above): Turbine speed;
    (ii) Second (below): Turbine power.

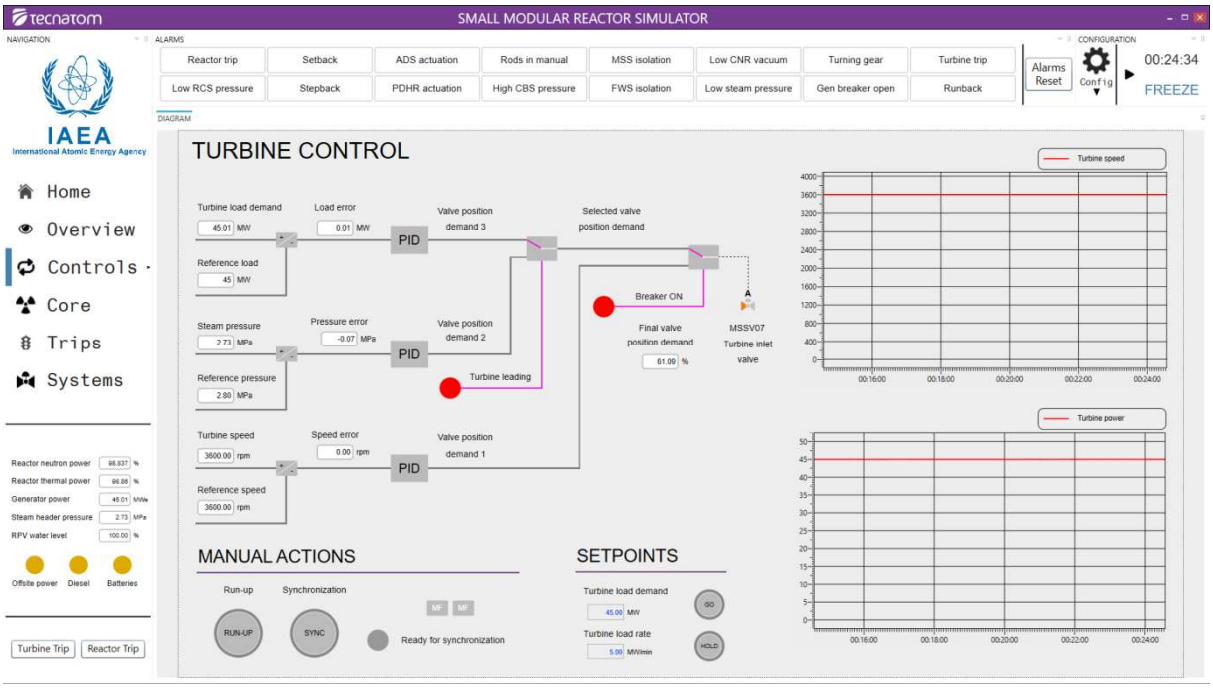


FIG. 59. Turbine control display.

4.3.10. Turbine bypass control

This screen shows the main parameters and controls associated with turbine bypass (Fig. 60).

Turbine bypass valve (MSSV09) can be controlled by two ways:

(a) If plant mode is in turbine leading, bypass valve controls average temperature, or;
(b) If plant mode is in-reactor leading mode, bypass valve controls steam pressure.

There are two graphs with 10 minutes interval:

(a) First (above): Average temperature vs reference temperature;
(b) Second (below): Steam header pressure.

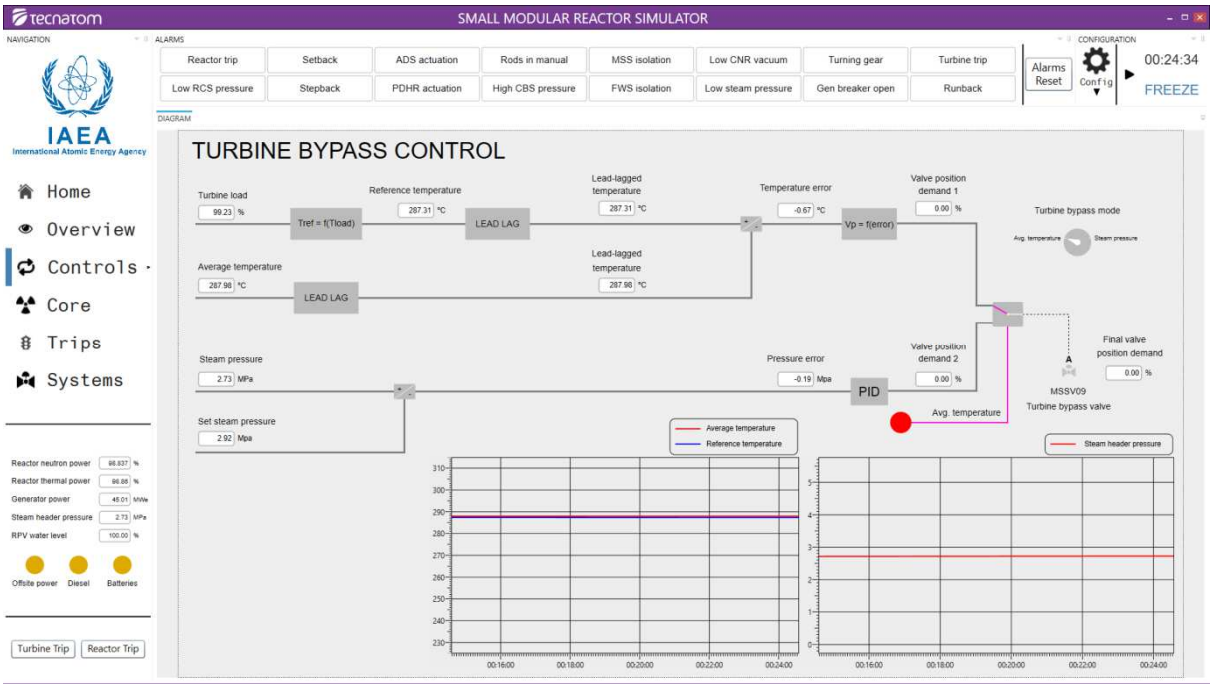


FIG. 60. Bypass turbine control display.

4.3.11. Rod position control

This screen shows the control rod movement (Fig. 61).

There are three banks of rods. One bank for shutdown (bank A) and two banks for control (bank B and C).

Rods can be controlled in two ways:

(a) If plant mode selector is in turbine leading mode, rods control average temperature;
(b) If plant mode selector is in-reactor leading mode, rods control difference between reactor power and power demand.

User can perform the following actions:

(a) Rods in auto or manual:
  (i) If rods are in manual, user should select the bank that he/she wants to move using the bank selection and then pushing in (insert) or out (withdraw);
  (ii) If rods in auto, user should do nothing.
(b) Actuate two malfunctions:
  (i) Uncontrolled control rod assembly withdrawal at power.
  (ii) One bank of shutdown control rods drop into the core.
(c) Perform reactor power reduction introducing % and %/min if reactor leading mode is selected.

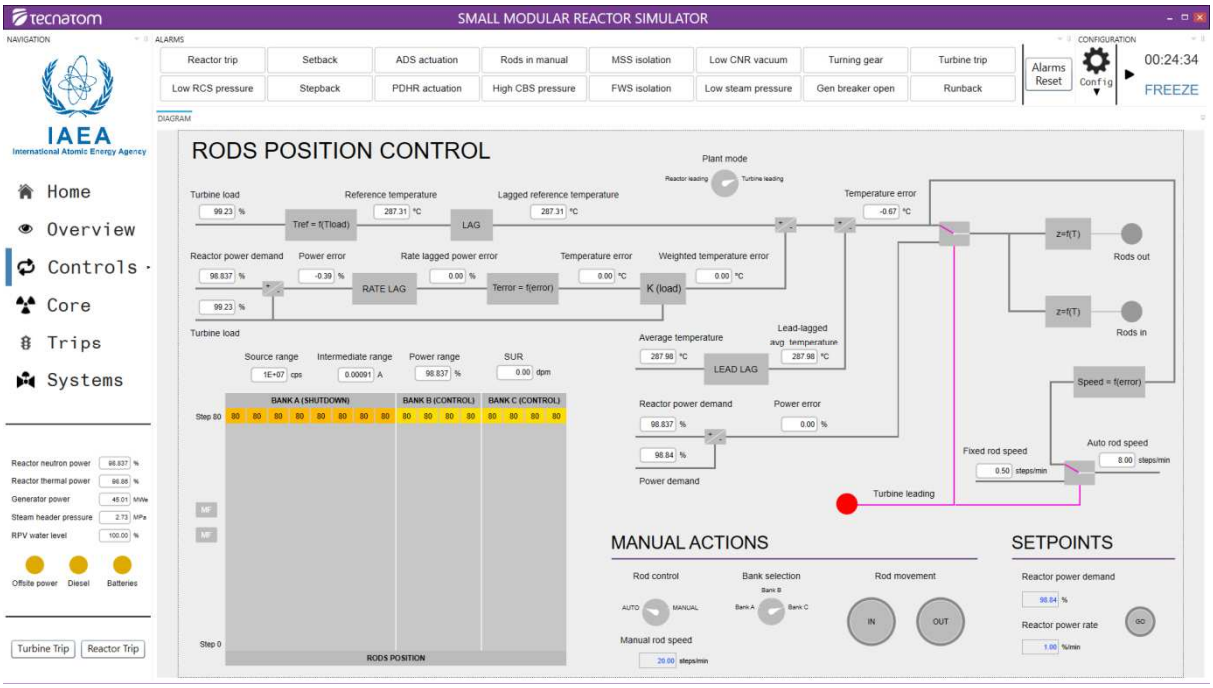


FIG. 61. Rod position display.

5. MATHEMATICAL MODELS USED

5.1.NEUTRONIC CORE

5.1.1. Neutronic equations

NEMO_RT is the neutronic code used to develop the neutronic calculations of this iPWR-SMR simulator.

The neutronic model is a 3-D model based on a modified one-group diffusion theory with nodal feedback on coolant temperature and density, fuel temperature (Doppler), xenon and samarium, boron, voids and the presence of control assemblies.

The neutronic solution is based on first principle processes like absorption, leakage, internodal neutron transport, neutron sources and fission. It models the transport of neutrons among coupled mesh regions, the fission reaction and the production of fission products which lead to delayed neutron emission (delayed neutron precursors).

The method employed is a variant of the method of space-time separation and features one dimensional (1-D) and three dimensional (3-D). The objective of this method is to determine a space and time dependent distribution of the power production. This is represented by the term P(r,t). The derivation begins by separating P(r,t) into an axial and time dependent amplitude function T(z,t), and a space and time dependent normalized shape function S(r,t):

$$P(\vec{r},t) = T(z,t) \times S(\vec{r},t) \tag{2}$$

The 3-D normalized shape function S(r,t) shows a slow variation over time but its numerical solution is very high time consuming. The 1-D amplitude function T(z,t) changes faster over time but is rather low time consuming.

This makes it possible to improve the calculation efficiency by solving the 3-D shape function at a lower frequency than the 1-D amplitude function.

The 1-D calculations are performed at a given high frequency (typically 10 Hz) to provide a smooth and stable modelling of the dynamic conditions in the core and to quickly update the response of the instrumentation. The 3-D calculations, on turn, are performed at a lower frequency (typically 1 Hz) to take into account the 3-D spatial effects of xenon distribution, control rods and thermal hydraulic feedback.

The model has been formulated so that rapid transients are simulated within the constraints of real time simulation while maintaining high degree of fidelity. To do it, certain processes are represented by means of parallel models.

When implemented in the simulator software, the neutronic model is coupled to the thermal hydraulic (TH) models for a real time simulation code (i.e. TRAC-RT, TECNATOM proprietary). In this way, the TH code internal point kinetic model is replaced by NEMO, which determines the total core power level and the 3-D power distribution in the TH model fuel rods as a function of time. On turn, the THs model calculates the resulting axial and radial temperature profiles in the fuel rods, the heat flux through the rod surface and the coolant TH conditions. Parameters required for the reactivity feedback (coolant temperature and density, fuel temperature, boron concentration) are, in this way, supplied by the TH code [5].

5.1.2. Neutronic calculations using space-time separation

The neutronic model determines the core power level and 3-D power distribution as a function of time. It models the transport of neutrons among the mesh regions, the fission reaction and the production of fission products which lead to delayed neutron emission. The method employed is a variant of the method of space-time separation and features one dimensional and three dimensional calculations.

The principle of space-time separation is easy to understand, although its implementation can be algebraically complex. The objective of the calculation is to determine a space and time dependent distribution of the power production. This is represented by the term P(r,t).

The derivation begins by separating P(r,t) into an axial and time dependent amplitude function T(z,t), and a space and time dependent normalized shape function S(r,t):

$$P(\vec{r}, t) = T(z, t) \times S(\vec{r}, t) \tag{3}$$

By maintaining the x-y spatial average of S constant as a function of both z and t, the separation is unambiguous. There is no approximation in the above equation.

The 3-D normalized shape function S shows a slow variation over time but its numerical solution is very high time consuming. The 1-D amplitude function T changes faster over time but is rather low time consuming.

This makes it possible to improve the calculation efficiency by solving the 3-D shape function at a lower frequency than the 1-D amplitude function.

Derivation of the neutronic model proceeds by expressing P as a mesh region integral quantity:

$$P_L(t) = \int_{\text{mesh region L}} P(\vec{r}, t) \times dV \tag{4}$$

A mesh region version of S is derived from the space-time separation assumption:

$$S_L(t) = \frac{P_L(t)}{T_K(t)} \tag{5}$$

for the axial mesh interval K.

The simplified differential equations for $P_L(t)$ and the delayed neutron sources can be written as follows:

$$\Lambda \frac{\partial P_L}{\partial t} = (\rho_{\infty L} - \beta)P_L + \sum_{M \neq L} W_{ML} P_M - \sum_{M \neq L} W_{LM} P_L - l^H_L P_L - l^V_L P_L + \sum_i C_{iL} + S^B_L \tag{6}$$

$$\frac{\partial C_{iL}}{\partial t} = \lambda_i (\beta_i P_L - C_{iL}) \tag{7}$$

where the approximation that neutron production is proportional to energy production have been made. The symbols in the above equation are defined as follows:

P_L is energy production rate in mesh region L (fission rate times total neutrons per fission in mesh region L times the conversion factor from neutron production to energy production);

Λ is prompt neutron generation time;

β is fraction of neutrons which appear as delayed neutrons;

β_i is fraction of delayed neutrons from precursor group I;

$P_{\infty L}$ is local reactivity in mesh region L, ignoring leakage;

C_{iL} is delayed neutron production rate from precursor group i in mesh region L times the conversion factor from neutron production to energy production;

λ_i is decay constant of the delayed neutron precursor group I;

S^B_L is production rate of neutrons from non-fission sources in mesh region L times the conversion factor from neutron production to energy production;

l^H_L is probability that a neutron produced in region L will escape from the

core horizontally;

l^V_L is probability that a neutron produced in region L will escape from the core vertically;

W_{ML} is coupling coefficient, i.e. coefficient of P_M in the expression for the net neutron current from node M to node L; and the expression for neutron current is:

$$J_{ML} = W_{ML} P_M - W_{LM} P_L \quad (8)$$

where J_{ML} is in units of neutrons per second times the conversion factor from neutron production to energy production

The reactivity in a reactor may be defined as the fraction by which prompt and delayed neutron production from fission exceeds neutron absorption and leakage. Since the local reactivity ignores leakage, the local reactivity is the fraction by which fission production exceeds the absorption in a mesh region.

5.1.3. Calculation of coupling coefficients

The leakage fractions l^H_L and l^V_L are obtained directly from calculation runs during the core design.

The coupling coefficients are calculated during initialization.

For horizontal coupling, if L is a mesh region in assembly I,

$$W_{LM} = W^H_I \quad (9)$$

is assumed.

Also, for vertical coupling from nodes L in plane K,

$$W_{LM} = W^V_K \quad (10)$$

is assumed.

These coefficients are calculated by substituting PL and $\rho\infty L$ from a steady state reference solution (data obtained from the fuel vendor's design code) into the steady state form of equation.

For the horizontal coupling, the resulting equations are then summed over the axial direction z. It results in a system of n equations of rank n-1, with unknown W^H_I. To make the system non-singular and to correct for the fact that the leakage fractions may not be obtained from the reference solution used to derive W_{LM}, a new unknown, which corrects the global leakage, and a new equation, which constrains the average coupling coefficient, are added. A similar calculation is used to derive the vertical coupling coefficients.

Once the coupling coefficients and corrected leakage fractions are obtained, an adjustment to the correlation for $\rho_{\infty L}$ is calculated for each mesh region. This causes the power distribution to match the reference solution for the conditions assumed in the reference calculation (usually, a hot full power steady state).

5.1.4. One dimensional amplitude calculation

The equations for T, the 1-D amplitude function, are derived by summing the above equations over all mesh regions in the axial layer K and dividing by the sum of the normalized shape value:

$$\Lambda \frac{\partial T_K}{\partial t} = \left(\rho_K - l_K^V - \beta \right) T_K + \sum_{K' \neq K} \left(W_{K'}^V T_{K'} - W_K^V T_K \right) + \sum_i C_{i,K} + S_K^B \tag{11}$$

$$\frac{\partial C_{i,K}}{\partial t} = \lambda_i \left(\beta_i T_K - C_{i,K} \right) \tag{12}$$

where some of the resulting sums have been replaced with the symbols ρ_K, $C_{i,K}$ and S_K^B.

Axial reactivity distribution is:

$$\rho_K = \frac{\sum_{L \in K} \left(\rho_{\infty K} - l_L^H \right) S_L}{\sum_{L \in K} S_L} \tag{13}$$

(The notation $L \in K$ means nodes L which are at elevation K)

The axial distribution of the delayed neutron production rates is:

$$C_{i,K} = \frac{\sum_{L \in K} C_{i,L}}{\sum_{L \in K} S_L} \tag{14}$$

and the axial distribution of the non-fission neutron sources is:

$$S_K^B = \frac{\sum_{L \in K} S_L^B}{\sum_{L \in K} S_L} \tag{15}$$

No approximations have been made in reducing the 3-D equations to 1-D kinetics.

Several approximations are made at this point. The derivatives are replaced by implicit differences in the program and, in practice, the above sums will be approximate unless a simultaneous 3-D calculation of S is available. The shape function from the most recent shape calculation is used, but it is several time steps out of date. The 1-D kinetic equations are discretized over time intervals corresponding to the calculation time step size. Shorter time intervals are used during periods of high reactivity. The 3-D calculations are run at a lower frequency.

5.1.5. Three dimensional shape calculation

The 3-D shape calculation is derived from equation (6), by substituting the space-time separation of P given in equation (3). It results in a completely equivalent differential equation for the shape function:

$$\Lambda\left(\frac{\partial S_L}{\partial t}+S_L\frac{\partial \ln(T_K)}{\partial t}\right)=\left(\rho_{\infty L}-l_L^V-l_L^H-\beta\right)S_L+\sum_{M\neq L}\left(\frac{W_{ML}T_{K'}}{T_K}-W_{LM}S_L\right)+\frac{\sum_i C_{i,L}+S_L^B}{T_K} \quad (16)$$

where '$T_{K'}$' is the amplitude value of the axial neighbour node and is equal to TK for neighbours at the same elevation.

Several approximations are made at this point.

First, it is assumed that

$$\frac{\partial \ln(\mathrm{T_K})}{\partial \mathrm{t}} \quad (17)$$

can be estimated from the two most recent 1-D kinetic time steps.

Second, the remaining time derivative is differenced implicitly. Thus, the time derivative is expressed as:

$$\Lambda\,\frac{\partial S_L}{\partial t}=\frac{\Lambda}{\Delta t^{3D}}\left(S_L(t)-S_L(t-\Delta t^{3D})\right) \quad (18)$$

where Δt^{3D} is the time interval between shape calculations and where all other terms in equation (15) are evaluated at time t.

Third, it is assumed that the mixture of delayed neutron precursors is in the same proportions everywhere in a given axial layer and that this mixture can be calculated by the 1-D kinetic model. It makes it possible to maintain a single total delayed neutron source term for each 3-D mesh region instead of the usual six ones.

The 1-D kinetic calculation supplies two factors that are needed to update the local delayed neutron source:

(a) for delayed neutron precursors which were in the core at the time of the last 3-D calculation, '$f_{1,K}$' is the fraction by which their neutron production rate has fallen;

(b) for delayed neutron precursors produced during the interval between 3-D calculations, '$f_{2,K}$' is the final production rate of delayed neutrons divided by the final fission neutron production rate.

These two factors are easily evaluated as a by-product of the core average 1-D calculation:

$$f_{1,K}=\frac{\sum_i C_{i,K}\left(t-\Delta t^{3D}\right)e^{-\lambda_i\Delta t^{3D}}}{\sum_i C_{i,K}\left(t-\Delta t^{3D}\right)} \quad (19)$$

$$f_{2,K} = \frac{\sum_i \lambda_i \beta_i \int_{t-\Delta t^{3D}}^{t} T_K(t') e^{-\lambda_i (t-t')} dt'}{T_K(t)} \tag{20}$$

In equations 13 and 14, the values of $C_{i,K}$ and T_K are obtained from the 1-D kinetic model.

These two factors are used to derive the following expression for the local delayed neutron source:

$$\mathrm{C_L(t) = \sum_i C_{i,L}(t) = C_L(t - \Delta t^{3D}) f_{1,K} + T_K(t) S_L(t) f_{2,K}} \tag{21}$$

The above approximations permit a fully implicit system of equations for the local delayed neutron source and the shape S, and also permit long time steps to be taken in most circumstances.

5.1.6. Renormalization

When defining the space-time separation, it was noted that the shape at each elevation is kept normalized. It was also noted that approximations for the reactivity had to be used in the 1-D kinetics.

After a shape calculation is performed, the approximations in both the 1-D kinetics and in setting up the shape calculation result in a final shape that is not normalized. Since the shape was solved with a fixed but approximate value of, any increase (or decrease) in total power due to 3-D effects not seen in 1-D kinetics would be seen as an increase (or decrease) in the magnitude of the shape. It is thus assumed that the 1-D kinetics did not calculate the best value of T_K.

When S_L is renormalized, T_K is adjusted inversely so the product

$$T_K \times \sum_{L \in K} S_L \tag{22}$$

does not change. Of course, if T_K must be adjusted, then the delayed neutron precursor production rate $\beta i T_K$ has implicitly been changed and the core wide delayed neutron source must also be adjusted.

If we let <S> be the specified (but arbitrary) sum of S_L, a normalization factor r_K rK may be computed as:

$$r_K = \frac{\sum_{L \in K} S_L}{\langle S \rangle} \tag{23}$$

This is applied to T_K and S_L as follows:

$$\mathrm{S_L = \frac{S_L}{r_K}} \tag{24}$$

and

$$T_L = \frac{T_L}{r_K} \tag{25}$$

The core wide delayed neutron source at each axial elevation for each delayed neutron precursor group 'i'' is recalculated using:

$$C_{i,K}(t) = C_{i,K}(t - \Delta t^{3D})\, e^{-\lambda_i \Delta t^{3D}} + r_K \left(C_{i,K}(t) - C_{i,K}(t - \Delta t^{3D}) \right) e^{-\lambda_i \Delta t^{3D}} \tag{26}$$

which is equivalent to multiply multiplying the delayed neutron precursor production rate by r_K for the entire interval between (t - Δt) and t.

No renormalization of the local delayed neutron source is required because the product of T_K and S_L, which appears in equation (21), is unchanged by renormalization.

In summary, the reactor core model does represent the first principle processes of absorption, leakage, internodal transport and fission. The basic equations for these processes have been consistently implemented for both the 1-D and 3-D neutronics. One reasonable assumption in deriving this monoenergetic model is that the neutrons per fission and energy per fission are in constant proportion. Beyond that, the model is derived from the one energy group diffusion theory. No assumption is made about flux to fission rate since the model deals exclusively in reaction rate (cross-section times flux). It is noted that the cross-sections are not constant. In this 1-group core model, the cross-sections have been transformed into ρ∞ and coupling coefficients. The ρ∞ is functionalized on numerous variables (moderator temperature, void fraction, xenon and samarium concentration, control rod position, fuel temperature and initial burn-ups).

Like almost all 3-D neutronic methods, other than Monte Carlo and sophisticated transport theory calculations, only nearest mesh neighbours are explicitly coupled to each node. It is recognized that some neutrons do migrate several assemblies through the core. This effect is accounted for in the generation of the coupling coefficients. Thus, the coupling coefficients do not predict the actual fraction of neutrons born in mesh A which are absorbed in neighbour mesh B. They are chosen to achieve the correct power distribution and the proper eigenvalue separation of various power distribution modes (e.g., rodded vs. unrodded, axial stability, etc.).

As stated previously, a single total delayed neutron source is maintained in each 3-D mesh region. The time dependent behaviour (fractional decay per time step and production per fission) of this source is determined in the 1-D neutronics based on a six-group formulation. Hence both the 3-D and core average delayed neutron source quite accurately follow the expected temporal behaviour.

Similarly, the 1-D kinetic model uses an eleven group formulation for decay heat. These eleven groups are divided into three major groups for purposes of determining the time integration and for use in the 3-D calculation. The three major groups are coupled to the eleven groups by an effective decay constant computed for each major group.

5.1.7. Calculation of 1-d reactivity

This calculation determines the reactivity-infinity (ρ_∞) for each neutronic node. A value of $\rho_{\infty L}$ is assigned to each neutronic node L based on the following formula:

$$\rho_{\infty L} = \rho_{\infty L}^{0} + \rho_{\infty J}^{DOP} + \rho_{\infty J}^{MOD} + \rho_{\infty J}^{BOR} + \rho_{\infty J}^{XE} + \rho_{\infty J}^{SM} + \rho_{\infty J}^{RCC} + \rho_{\infty J}^{VOID} \quad (27)$$

where

L is neutronic node;

J is thermal hydraulic node in which neutron node L resides;

$\rho_{\infty L}$ is node reactivity-infinity in absolute reactivity units ($\Delta K/K$);

$\rho_{\infty L}^{0}$ is base ρ_∞ due to local burnup, fuel type and tuning of initial conditions;

$\rho_{\infty J}^{DOP}$ is Doppler contribution to node reactivity;

$\rho_{\infty J}^{MOD}$ is moderator temperature contribution to node reactivity;

$\rho_{\infty J}^{BOR}$ is boron contribution to node reactivity;

$\rho_{\infty J}^{XE}$ is xenon contribution to node reactivity;

$\rho_{\infty J}^{SM}$ is samarium contribution to node reactivity;

$\rho_{\infty L}^{RCC}$ is control rod contribution to node reactivity;

$\rho_{\infty J}^{VOID}$ is void contribution to node reactivity.

The K-infinity (K_∞) for each node L is therefore:

$$K_{\infty L} = \frac{1}{(1-\rho_{\infty L})} \quad (28)$$

The base term $\rho_{\infty L}$ is a tunable constant chosen to give a selected power distribution at a particular burnup time point. It is calculated during the initialization process, which only needs to be performed when developing initial conditions for hot full power at BOL, MOL and EOL.

5.1.8. Fuel temperature (Doppler) reactivity

Reactivity feedback as a result of fuel temperature (Doppler Effect) is given by the following expression:

$$\rho_F = C_{DOP}\left(\sqrt{T_F + 460} - \sqrt{T_{ref} + 460}\right) \quad (29)$$

where

ρ_F is reactivity feedback due to fuel temperature;

C_{DOP} is reactivity feedback coefficient due to fuel temperature;

T_F is fuel temperature;

T_{ref} is reference fuel temperature; this temperature will remain constant throughout the simulation;

460 is factor to convert Fahrenheit to Rankine degrees (absolute temperature).

The reactivity calculated in this way is corrected by means of a multiplier allowing permanent and/or temporary changes to be made to this reactivity, this being useful for both the readjustment of the model and the activation of malfunctions.

Reactivity feedback as a result of moderator temperature, without considering the presence of soluble boron, is given by the following expression:

$$\rho_M(T_M) = \int_{T_{ref}}^{T_M} \gamma_M(T)dT \tag{30}$$

where

ρ_M is reactivity feedback due to moderator temperature, without consideration for the presence of soluble boron;

T_M is moderator temperature;

T_{ref} is reference moderator temperature; this temperature will remain constant throughout the simulation;

T is moderator temperature; integration variable;

dT is differential of T;

$\gamma_M(T)$ is reactivity feedback coefficient due to moderator temperature, at moderator temperature T.

It has been assumed that the moderator temperature reactivity feedback coefficient, $\gamma_M(T)$, varies with moderator temperature T in accordance with a second degree polynomial function:

$$\gamma_M(T) = a + bT + cT^2 \tag{31}$$

where

T is moderator temperature;

a, b, c are coefficients remaining constant throughout the simulation, and adjustable via common REBLOC (NEMO source file).

With this assumption, the above equation becomes:

$$\rho_M(T) = \left[aT + \frac{1}{2} bT^2 + \frac{1}{3} cT^3 \right] - \left[aT_{ref} + \frac{1}{2} bT_{ref}^2 + \frac{1}{3} cT_{ref}^3 \right] \quad (32)$$

The reactivity calculated in this way is corrected by means of a multiplier allowing permanent and/or temporary changes to be made to this reactivity, this being useful for both the readjustment of the model and the activation of malfunctions.

5.1.9. Boron reactivity

Reactivity feedback as a result of soluble boron concentration is given by the following expression:

$$\rho_B = \frac{\gamma_B C_B (1 - \alpha)}{SPV} \quad (33)$$

where

ρ_B is reactivity feedback due to soluble boron concentration;

γ_B is reactivity feedback coefficient due to soluble boron concentration;

C_B is soluble boron concentration;

α is void fraction (soluble boron is only in liquid water);

SPV is liquid water specific volume.

The reactivity calculated in this way is corrected by means of a multiplier allowing permanent and/or temporary changes to be made to this reactivity, this being useful for both the readjustment of the model and the activation of malfunctions.

5.1.10. Xenon concentration reactivity

Reactivity feedback as a result of xenon concentration is given by the following expression:

$$\rho_{Xe} = \gamma_{Xe} C_{Xe} \quad (34)$$

where

ρ_{Xe} is reactivity feedback due to xenon concentration;

γ_{Xe} is reactivity feedback coefficient due to xenon concentration;

C_{Xe} is xenon concentration.

The reactivity calculated in this way is corrected by means of a multiplier allowing permanent and/or temporary changes to be made to this reactivity, this being useful for both the readjustment of the model and the activation of malfunctions.

5.1.11. Samarium concentration reactivity

Reactivity feedback as a result of samarium concentration is given by the following expression:

$$\rho_{Sm} = \gamma_{Sm} C_{Sm} \tag{35}$$

where

ρ_{Sm} is reactivity feedback due to samarium concentration;

γ_{Sm} is reactivity feedback coefficient due to samarium concentration;

C_{Sm} is samarium concentration.

The reactivity calculated in this way is corrected by means of a multiplier allowing permanent and/or temporary changes to be made to this reactivity, this being useful for both the readjustment of the model and the activation of malfunctions.

5.1.12. Control rods reactivity

Reactivity feedback as a result of control rods insertion in the core is given by the following expression:

$$\rho_R = \gamma_R \times f_R \times f_S \tag{36}$$

where

ρ_R is reactivity feedback due to control rod insertion;

γ_R is reactivity feedback coefficient due to control rod insertion;

f_R is fraction of node that is rodded;

f_S is scalloping correction factor.

A specific neutronic node may be unrodded, partially rodded or fully rodded.

If the node is unrodded, then $\rho_R = 0$.

If the node is partially rodded, then $\rho_R = \gamma_R \times f_R \times f_S$, where f_R accounts for the fact that the rod is not completed inserted in the node, and f_S is a correction factor that de-linearize the node reactivity as a function of rod insertion.

If the node is fully rodded, then $f_R = f_S = 1$; and then $\rho_R = \gamma_R$.

5.1.13. Moderator void reactivity

Reactivity feedback as a result of moderator voids is given by the following expression:

$$\rho_v(\alpha) = \int_{\alpha_{ref}}^{\alpha} \gamma_v(\alpha')d\alpha' \tag{37}$$

where

ρ_v is reactivity feedback due to moderator voids;

α is moderator void fraction;

α_{ref} is reference moderator void fraction; this void fraction will remain constant throughout the simulation;

α' is moderator void fraction; integration variable;

$d\alpha'$ is differential of α';

$\gamma_v(\alpha')$ is reactivity feedback coefficient due to moderator voids, at void fraction α'.

It has been assumed that the moderator void reactivity feedback coefficient, $\gamma_v(\alpha)$, varies with moderator void fraction α in accordance with a second degree polynomial function:

$$\gamma_v(\alpha) = A + B\alpha + C\alpha^2 \tag{38}$$

where

α is moderator void fraction;

a, b, c are coefficients remaining constant throughout the simulation, and adjustable via common block REBLOC (NEMO source file).

With this assumption, the above equation becomes:

$$\rho_v(\alpha) = \left[A\alpha + \frac{1}{2}B\alpha^2 + \frac{1}{3}C\alpha^3\right] - \left[A\alpha_{ref} + \frac{1}{2}B\alpha_{ref}^2 + \frac{1}{3}C\alpha_{ref}^3\right] \tag{39}$$

The reactivity calculated in this way is corrected by means of a multiplier allowing permanent and/or temporary changes to be made to this reactivity, this being useful for both the readjustment of the model and the activation of malfunctions.

5.1.14. Calculation of 1-d reactivity

The calculation of the 1-D neutronics reactivity uses the same correlations than those for the 3-D local reactivity. Since the xenon and samarium variation occur over a long time period relative to the 1-D solution, their respective contributions are included upon the completion of the 3-D neutronic solution.

The axial ρ_∞ for each 1-D neutronic node is based on the following formula:

$$\rho_{\infty K} = \rho_{\infty K}^{0} + \Delta\rho_{\infty K}^{DOP} + \Delta\rho_{\infty K}^{MOD} + \Delta\rho_{\infty K}^{BOR} + \Delta\rho_{\infty K}^{RCC} + \Delta\rho_{\infty K}^{VOID} \tag{40}$$

The units are the same as in the 3-D reactivity calculation

where

K is axial neutronic node;

$\rho_{\infty K}$ is axial node reactivity-infinity;

$\rho_{\infty K}^{0}$ is base ρ_{∞} updated at each 3-D calculation;

$\Delta\rho_{\infty K}^{DOP}$ is Doppler contribution to node reactivity since the last 3-D calculation;

$\Delta\rho_{\infty K}^{MOD}$ is moderator temperature contribution to node reactivity since the last 3-D calculation;

$\Delta\rho_{\infty K}^{BOR}$ is boron contribution to node reactivity since the last 3-D contribution;

$\Delta\rho_{\infty K}^{RCC}$ is control rod contribution to node reactivity since the last 3-D calculation;

$\Delta\rho_{\infty K}^{VOID}$ is void contribution to node reactivity since the last 3-D calculation.

The base term, $\rho_{\infty K}^{0}$, is updated at each 3-D time step calculation.

5.1.15. Startup source and photoneutrons

Photoneutrons are the principle neutron source supporting subcritical multiplication from 15 minutes to 6 hours after a reactor shutdown. Other sources which are nearly constant dominate after 6 hours.

The photoneutrons are generated by fission product released gamma radiation absorbed in deuterium. The physical equation for photoneutron and startup sources for a pure D_2O coolant is:

$$\frac{\partial S_i^{PN}}{\partial t} = \lambda_i^{PN}\left(\frac{\gamma_i^{PN} P^{FISS}}{k^{FISS}} - S_i^{PN}\right) \tag{41}$$

where

i is 1 of 9 photoneutron fission product groups;

S_i^{PN} is pure D_2O photoneutron source from group I;

λ_i^{PN} is group i photoneutron decay constant;

γ_i^{PN} is yield of photoneutrons per fission;

k^{FISS} is energy per fission;

P^{FISS} is total core power;

Δt_{PN} is photoneutron time step.

The above equation is integrated over time steps of Δt_{PN} = 30 seconds on a core average basis.

5.1.16. Xenon and iodine

The objective of this calculation is to compute the 3-D xenon and Iodine concentrations for use in the local K_∞ calculation. The physical equations for xenon and Iodine in a TH node K are:

$$\frac{\partial I_K}{\partial t} = \gamma^I R_K^{FISS} - \lambda^I I_K \tag{42}$$

and

$$\frac{\partial Xe_K}{\partial t} = \gamma^{Xe} R_K^{FISS} + \lambda^I I_K - \left(\lambda^{Xe} + A_K^{Xe}\right) Xe_K \tag{43}$$

where

K	is thermal hydraulic node;
I_K / Xe_K	is Iodine / xenon concentration;
γ^I / γ^{Xe}	is fission yield for I^{135} / Xe^{135};
R_K^{FISS}	is fission rate;
λ^I / λ^{Xe}	is decay constant for I^{135} / Xe^{135};
A_K^{Xe}	is absorption rate for Xe^{135}.

5.1.17. Promethium and Samarium

The objective of this calculation is to compute the 3-D Samarium and Promethium concentrations for use in the local K_∞ calculation. The governing physical equations for Samarium and Promethium concentrations in a THs node K are:

$$\frac{\partial Pm_K}{\partial t} = \gamma^{Pm} R_K^{FISS} - \lambda^{Pm} Pm_K \tag{44}$$

and

$$\frac{\partial Sm_K}{\partial t} = \lambda^{Pm} Pm_K - A_K^{Sm} Sm_K \tag{45}$$

where

K	is thermal hydraulic node;
Pm_K / Sm_K	is Promethium / samarium concentration;
γ^{Pm} / γ^{Sm}	is fission yield for Pm^{139} / Sm^{139};

R_K^{FISS} is fission rate;

λ^{Pm} is decay constant for Pm^{139};

A_K^{Sm} is absorption rate for Sm^{139}.

and where the absorption constant is:

$$A_K^{Sm} = K^{Sm} R_K^{FISS} \tag{46}$$

and the constant

$$K^{Sm} = \frac{\sigma_a^{Sm}}{\sum^{FISS}} \tag{47}$$

is assumed to be spatially uniform.

5.1.18. Decay heat calculation

This calculation computes the following:

(a) Fission power in each TH node;
(b) Local and total core decay heat;
(c) Local and total core sensible power.

In the 1-D neutronic model, a core average decay heat model is used which uses eleven decay heat groups ('g'). The time integration method varies depending on the decay constant of the particular group. The eleven groups are divided into 3 major groups: A, B, and C for purposes of determining the time integration and for use in the 3-D decay heat calculation.

In the 3-D calculation, only the major groups A, B and C are modelled. They are coupled to the eleven group core average model by effective decay constants computed for each major group. These are assumed to be uniform over the core.

The groups in major group C have half-lives varying from 77 days to 29 years. For purposes of the simulation, they are assumed to be constant heat sources for each burnup time point. They are also assumed to be uniformly distributed over the core. This assumption is acceptable because group C is less than 0.2% of the nominal core power.

The governing physical equation for all decay heat groups is:

$$\frac{\partial E_g^{DK}}{\partial t} = \gamma_g^{DK} P^{FISS} - \lambda_g^{DK} E_g^{DK} \tag{48}$$

where

E_g^{DK} is energy stored in decay heat group 'g';

γ_g^{DK} is group 'g' yield expressed as a fraction of the total fission energy release;

P^{FISS} is total fission power including energy being stored as decay;

λ_g^{DK} is decay constant for decay heat group 'g'.

This equation is expressed on a total core basis, but the same equation with appropriate subscripts applies to each TH node.

The method of integrating the time derivative is defined for each major decay group. The calculation is done in two parts: one part is carried out each time the 1-D model is performed, the other part is performed as part of the 3-D thermal hydraulic calculation.

5.2.REACTOR COOLANT (RCS), BOP (MSS, FWS, TUR) AND CONTAINMENT (CBS, PDHR) HYDRAULICS

TRAC_RT is the code used to develop the THs of the reactor core, main steam, feedwater, turbine and containment systems.

TRAC_RT is an advanced version of the TRAC 'best estimate' TH code series for simulation of transients in light water reactors (LWR), adapted by TECNATOM for being implemented into nuclear power plant training and engineering simulators. To accomplish these training and engineering functions, TRAC_RT combines the major features of engineering best estimate codes along with those of real time simulation codes for training purposes [6].

TRAC_RT major features are:

(a) Non-homogeneous and non-equilibrium model for the two phase flow system. TRAC_RT is based on a non homogeneous, non equilibrium two fluid model, which uses a separate description of the steam and liquid phases allowing unequal velocities and temperatures. The six basic governing equations are the conservation of mass, momentum and energy for each phase;

(b) Boron and non-condensable models. The steam phase may be a mixture of steam and air, in which case an extra equation for air mass conservation is added. The liquid may contain dissolved boron, and then other extra equation for boron mass conservation is solved;

(c) Shear and heat transfer at steam-liquid interfaces and wall surfaces dependent on flow regime. Additional constitutive correlations are formulated for interfacial shear and heat transfer, and for the heat transfer and friction between fluids and structure walls. These correlations are determined on a flow regime map, covering all possible flow regimes and wall conditions (Fig. 62).

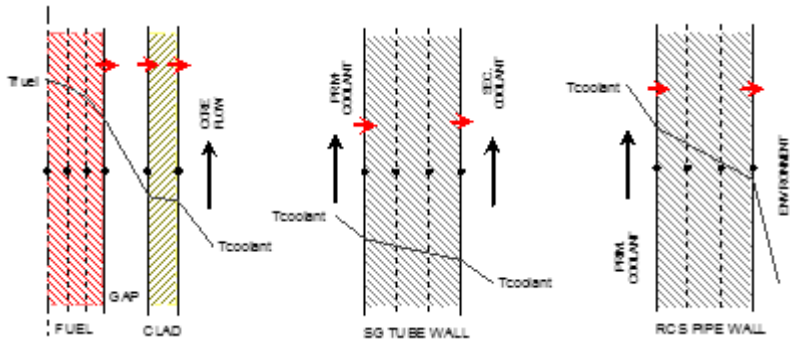


FIG. 62. One dimensional heat conduction.

(d) One dimensional heat conduction model in fuel rods and structures. A one dimensional heat conduction model is used in fuel rods and other structures, such as pipe walls and reactor vessel structures. The convective heat transfer from the surfaces to the fluid covers all major heat transfer regimes. Energy generation due to fission, decay heat and metal-water reaction is accounted for;

(e) Three dimensional vessel model. TRAC_RT has the capability of representing a three dimensional model of the reactor vessel by means of the provided VESSEL component. A three dimensional flow calculation can be used within this component to allow an accurate calculation of the complex multidimensional flow patterns inside the reactor vessel that are important in determining accident behaviour (Fig. 63);

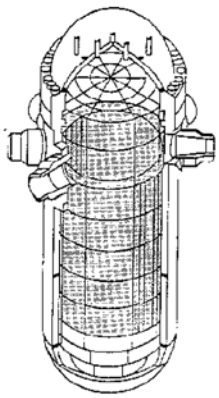


FIG. 63. 3-D Vessel model.

(f) Implicit numerical scheme. TRAC_RT uses fast implicit numerical methods to integrate the two phase flow mass and energy equations for both one dimensional and three dimensional components. The fully implicit integration of the mass and energy equations allows for better energy balances and more accurate solutions for large time steps, is inherently stable and allows the use of time step sizes in excess of the material Courant limit. In this way, the time step size is only limited by accuracy considerations. An implicit coupling is also utilized for the heat transfer between surfaces and the hydraulics.

5.2.1. Balance equations

The basic form of conservation equation for 1-D flows across a duct or channel differential volume, dV is:

$$\frac{\partial \Phi}{\partial t} = - \frac{\partial}{\partial x} (V \, . \, \Phi) + \Gamma \tag{49}$$

where Ø is the physical magnitude which is conserved and V is the flow velocity. The member on the left in the above equation represents the time rate of change of magnitude Ø in dV. The first term of the member on the right is the net rate of flow of magnitude Ø out of the volume dV through the inlet and outlet surfaces (convection or flux term). Finally, T represents the magnitude Ø generation rate (source terms)

TRAC_RT solves the conservation equations for mass, momentum and energy for the gas and liquid phases. The gas phase can be a mixture of steam and air, while no air is assumed to be dissolved in the liquid. The air and steam in the gas phase is assumed to be perfectly mixed and thus have the same velocity and temperature. Furthermore, Dalton's law applies to the partial steam and non-condensable gas pressures (P = Psteam + Pa). The liquid phase can have dissolved boron. The concentration of boron is assumed to be small enough that the liquid properties and the hydraulic solution are not affected. The boron transport by the liquid phase is treated similarly to the non-condensable transport by the gas phase (boron velocity and temperature are the same than those for the liquid). Consequently a total of eight conservation equations are needed: four mass equations for the steam, mixture, air and boron; two momentum equations for the gas and liquid and two energy equations for the gas and mixture.

5.2.1.1. Mass balance equations

The equations expressing conservation of masses in the channel differential volume dV considering the potential addition or removal of mass (source terms) along with mass transfer between the phases (evaporation or condensation) are:

Gas mass:

$$\frac{\partial}{\partial t}(\alpha \rho_g) = - \frac{\partial}{\partial x}(\alpha \rho_g V_g) + \Gamma_g + M_g \tag{50}$$

Mixture mass:

$$\frac{\partial}{\partial t}((1 - \alpha) \rho_\ell + \alpha \rho_g) = - \frac{\partial}{\partial x}((1 - \alpha) \rho_\ell V_\ell + \alpha \rho_g V_g) + M_m \tag{51}$$

Non-condensable mass:

$$\frac{\partial}{\partial t}(\alpha \rho_a) = - \frac{\partial}{\partial x}(\alpha \rho_a V_g) + M_a \tag{52}$$

Boron mass:

$$\frac{\partial}{\partial t}(m_B) = -\frac{\partial}{\partial x}(m_B V_\ell) + M_b \tag{53}$$

5.2.1.2. Momentum balance equations

The momentum conservation equations are based on Newton's second law, which determines that the forces acting on each phase can be equated to the rate of change of momentum of that phase. In TRAC_RT hydrodynamic model, the pressure (P) gradient stress, wall shear forces (F_w), interfacial shear forces ($f_{\ell g}$) and gravity body force (g) are considered.

Gas momentum:

$$\frac{\partial}{\partial t}V_g = -V_g\frac{\partial V_g}{\partial x} - \frac{k\rho_c}{\alpha\rho_g}\left[\frac{\partial}{\partial t}V_R + V_d\frac{\partial V_R}{\partial x}\right]$$

$$-\frac{1}{\rho_g}\frac{\partial P}{\partial x} - \frac{1}{\rho_g}F_w - \frac{1}{\alpha\rho_g}f_{\ell g} - g \tag{54}$$

Liquid momentum:

$$\frac{\partial}{\partial t}V_\ell = -V_\ell\frac{\partial V_\ell}{\partial x} + \frac{k\rho_c}{(1-\alpha)\rho_\ell}\left[\frac{\partial}{\partial t}V_R + V_d\frac{\partial V_R}{\partial x}\right]$$

$$-\frac{1}{\rho_\ell}\frac{\partial P}{\partial x} - \frac{1}{\rho_\ell}F_w + \frac{1}{(1-\alpha)\rho_\ell}f_{\ell g} - g \tag{55}$$

where

$$V_R = V_g - V_\ell \text{ (relative velocity)} \tag{56}$$

$$V_d = (1-\alpha).V_g - \alpha.V_\ell \text{ (dispersed phase velocity)} \tag{57}$$

5.2.1.3. Energy balance equations

The energy conservation equations determine that the variation in the gas and mixture enthalpy equals the net sum of all the enthalpy and heat additions per unit time. In TRAC_RT model, the wall-gas and wall-liquid heat transfer rates (q_{wg} and $q_{w\ell}$), the interfacial heat transfer rates (interface-gas q_{ig} and interface-liquid q_{il}) and the contribution from the interfacial mass transfer rates are considered. Viscous dissipation and surface tension are neglected

Gas energy:

$$\frac{\partial}{\partial t}(\alpha \rho_g e_g) + P\frac{\partial \alpha}{\partial t} = -\frac{\partial}{\partial x}(\alpha \rho_g e_g V_g) - P\frac{\partial}{\partial x}(\alpha V_g) + q_{wg} + q_{ig} + q_{dg} + \Gamma_g h_g \quad (58)$$

Mixture energy (note than different velocities affect the gas and liquid associated convective terms):

$$\frac{\partial}{\partial t}[(1-\alpha)\rho_\ell e_\ell + \alpha \rho_g e_g] = -P\frac{\partial}{\partial x}((1-\alpha)V_\ell + \alpha V_g)$$

$$-\frac{\partial}{\partial x}[(1-\alpha)\rho_\ell e_\ell V_\ell + \alpha \rho_g e_g V_g] + q_{w\ell} + q_{wg} + g_{dl} + q_{dg} \quad (59)$$

The following is a general list of TH and thermodynamic parameter symbols used herein (SI units are used):

α is void fraction;

α_{trans} is transition void fraction;

CB is boron concentration (boron mass per unit mass of liquid);

Γ_g is vapour mass generation rate;

Γ_l is liquid mass generation rate, $\Gamma_l = -\Gamma_g$;

e_g is gas specific internal energy;

e_l is liquid specific internal energy;

Fw is wall force per unit volume;

f_{lg} is interfacial friction force per unit volume;

g is gravity acceleration;

k is virtual mass coefficient;

$$k = 0.5\,\alpha\frac{1+2\alpha}{1-\alpha} \quad \text{for} \quad 0 \le \alpha \le \alpha_{trans}$$

$$k = 0.5\,(1-\alpha)\frac{3-2\alpha}{\alpha} \quad \text{for} \quad \alpha_{trans} < \alpha \le 1 \quad (60)$$

M_a is non-condensable mass source;

M_b is boron mass source;

M_g is gas mass source;

M_m is mixture mass source;

P is pressure;

q_{dg} is power deposited directly into gas (without heat conduction processes);

q_{dl} is power deposited directly into liquid;

q_{ig} is interfacial heat transfer rate to gas;

q_{il} is interfacial heat transfer rate to liquid;

q_{wg} is wall-gas heat transfer rate;

q_{wl} is wall-liquid heat transfer rate;

ρ_a is non-condensable density;

ρ_c is continuous phase density:

$$\rho_c = \rho l \text{ for } 0 < \alpha < \alpha \text{ trans (liquid continuous);}$$

$$\rho_c = \rho v \text{ for } \alpha \text{ trans} < \alpha < 1 \text{ (vapour continuous);} \quad (61)$$

ρ_g is gas density;

ρ_l is liquid density;

t is time;

V_g is gas velocity;

V_L is liquid velocity;

V_R is relative velocity, $V_R = V_g - V_L$;

x is axial distance;

and subscripts

c is continuous phase;

d is dispersed phase;

g is gas;

i is interphase;

l is liquid;

w is wall.

5.2.2. Constitutive correlations

The field or balance equations (conservation equations for mass, momentum and energy) require certain auxiliary or constitutive equations to effect closure. Those are the thermodynamic or state equations for each phase along with an extensive set of basic models, consisting of constitutive correlations for shear and heat transfer at the walls and at the gas-liquid interface and net vapourization rate, that is, these correlations express the rates of exchange of mass, momentum and energy between each phase and its surroundings. The constitutive correlations are calculated consistently dependent on the flow regime and the heat transfer regime.

5.2.2.1. Flow regime map

The flow regimes considered are defined in the flow regime map which determines the existing flow pattern in a given condition based on the wall condition (wetted or dry) and the void fraction. In this way, the flow regimes addressed are given in TABLE 3.

TABLE 3. FLOW REGIME MAP

		Wall condition	
Void fraction		**Wetted**	**Dry**
Vapour continuous flow	1.0 (vapour single phase)		Single phase vapour
	$\alpha_t + 0.1 < \alpha < 1.0$	Annular	Dispersed Droplet
Transition	$\alpha_t < \alpha < \alpha_t + 0.1$	Transition	Transition
Liquid continuous flow	$0.0 < \alpha < \alpha_t$	Bubbly/churn	Inverted annular
	0.0 (liquid single phase)	Single phase liquid	

Shear and heat transfer treatment at vapour-liquid interfaces and wall surfaces vary depending on the existing flow regime. As it is showed, the flow regime map is divided into two major flow regimes: the liquid continuous flow regime and the vapour continuous flow regime. A transition flow regime is also defined. The liquid continuous flow regime includes the single phase liquid flow, bubbly/churn flow and inverted annular flow, while the vapour continuous flow regime includes the dispersed annular, droplet flow and single phase vapour flow.

A transition regime of 0.1 assures a smooth transition in the constitutive correlations and avoids discontinuities, which tend to cause numerical problems. The transition void fraction,

$\forall t$, represents the transition between churn and annular flow, and is calculated as a function of liquid and vapour densities and Reynolds number.

Transition between bubbly/churn and inverted annular flow takes place when the liquid in the film (or entrained droplets) can be lifted by the vapour flow relative to the liquid velocity in the bubbly/churn flow regime. Transition between annular flow and dispersed droplet flow is given by the onset of entrainment. For low vapour flow, annular flow will exist and, as the vapour flow is increased, more and more entrainment will occur, causing a gradual transition to droplet flow.

5.2.2.2. Interfacial shear

Interfacial shear term (f_{lg}: Drag force per unit volume between phases) is calculated, in agreement with Ishii's drift flux model and J.G.M. Andersen model for the equivalence between the drift flux and two fluid model. Depending on the existing void fraction, bubbly/churn, transition or annular/drop flow regimes are considered. The drag force per unit volume between phases is calculated in the form:

$$f_{lg} = c_i \mid v_r \mid v_r \tag{62}$$

where c_i is interfacial drag coefficient;

v_r is relative velocity.

Interfacial drag coefficient and relative velocity are functions of:

(a) Flow regime (bubbly/churn, annular, droplet);
(b) Void fraction;
(c) Properties of steam and liquid.

5.2.2.3. Interfacial heat transfer

In order to calculate the interfacial heat transfer rates (q_{il}, q_{ig}), the associated interfacial areas and heat transfer coefficients are calculated, depending on the existing flow regime : bubbly/churn flow, transition or annular/drop. The calculation is based on the assumption that liquid-vapour interface is always at saturation temperature corresponding to partial steam pressure. Thus, liquid-vapour interface heat transfer rates are expressed as:

$$q_{il} = A_i . h_{il} . (T_l - T_{sat}) \quad q_{ig} = A_i . h_{ig} . (T_g - T_{sat}) \tag{63}$$

where A_i is interfacial area per unit volume (depends on the flow regime);

h_{il}, h_{ig}, are convective HTC's (depend on the flow regime)(Same flow regime map that for the interfacial shear model).

Interfacial mass exchange $э_g$ is then evaluated as:

$$\Gamma_g = \frac{q_{il} + q_{ig}}{h_g - h_l} \quad \begin{cases} \Gamma_g > 0 : \text{evaporation} \\ \Gamma_g < 0 : \text{condensation} \end{cases} \tag{64}$$

5.2.2.4. Wall friction and local pressure losses

Pressure drop associated terms (F_w) are calculated as:

$$F_w = 2\frac{f_1}{D_h}\frac{G^2}{\rho_1}\phi_l^2 \quad + \quad \frac{1}{2}\frac{G^2}{\rho_1}K\,\phi_{\text{hom}}^2 \qquad (65)$$

$$[wall \quad friction] \qquad [local \quad losses]$$

$$\text{Wall friction}\begin{cases} F_{wl} = (1-\alpha)\,F_w \\ F_{wg} = \alpha\,F_w \end{cases} \qquad (66)$$

Where f_1 is single phase friction factor (Moody)

Φ_l^2 is two phase multiplier (Chisholm)

Two phase flow frictional pressure drop is thus correlated by means of a two phase multiplier which relates the overall frictional pressure drop to a reference frictional pressure drop for an equivalent single phase flow. Regarding the local pressure losses, loss coefficients (K) are input and tuned by the user based on the geometry of the flow and reference stable condition data.

5.2.2.5. Water heat transfer

Wall-gas and wall-liquid heat transfer rates (q_{wg}, q_{wl}) are accounted for in the standard way, that is:

$$q_{wg} = [\text{wall-gas heat transfer area}] * h_{wg} * (T_w - T_g) \qquad (67)$$

$$q_{wl} = [\text{wall-liquid heat transfer area}] * h_{wl} * (T_w - T_l) \qquad (68)$$

The heat transfer areas represent an actual estimate of the total wall surface wetted by each phase, whereas the wall to liquid and wall to gas heat transfer coefficients (h_{wg}, h_{wl}) are based on different heat transfer correlations from the literature, depending on the considered heat transfer regime. Based on the wall and local TH conditions, the wall heat transfer is divided into several heat transfer regimes given in TABLE 4.

TABLE 4. WALL HEAT TRANSFER

	Wall condition boiling transition			
	NO (wet wall)		**YES (dry wall)**	
Flow regime	**Tw < Tsat**	**Tsat < Tw < T_{CHF}**	**T_{CHF} < Tw < Tmin**	**Tmin < Tw**
$\alpha = 0$	Liquid convection	Liquid convection		
$0 < \alpha < \alpha_t$	Liquid convection	Subcooled / nucleate Boiling	Transition boiling	Film boiling
$\alpha_t < \alpha < 1$	Condensation	Forced convection Vapourization	Transition boiling	Film boiling / dispersed flow
$\alpha = 1$			Vapour convection	Vapour convection

5.2.2.6. Structures 1-d heat conduction

Heat conduction equation is solved for:

(a) Fuel rods (CHANNEL component);
(b) Pipe walls (1-D components);
(c) Outer wall and internal heat slabs (3-D VESSEL component).

Component wall and fuel rod heat conduction equation (cylindrical coordinates)

$$\rho C_p \frac{\partial T}{\partial t} = \frac{1}{r}\frac{\partial}{\partial r}\left[r k \frac{\partial T}{\partial t} + q''' \right]$$

$$q''' : \text{heat generation rate (input, kinetics, metal-water reaction)} \tag{69}$$

5.3.SAFETY SYSTEMS (PIS/GIS/CCS) AND CIRCULATING WATER (CWS)

TEAM_FLOW is the tool used for modelling PIS/GIS/CCS and CWS hydraulics.

TEAM_FLOW is a tool designed for the modelling of hydraulic systems that is especially aimed at real time simulation. In addition to obtaining system flows and pressures, it allows

for the calculation of temperatures and of the transport of substances dissolved and in suspension.

5.3.1. Description of hydraulic network resolution

The following basic equations are used as a basis for the resolution of a hydraulic network:

(a) Continuity equation (or conservation of mass);

(b) Conservation of momentum equation;

(c) Energy conservation equation.

To these basic equations are applied the following simplifications typical of hydraulic networks:

(a) Non-compressible flow, i.e. $\frac{\partial M}{\partial P} = 0$, in each node there is compliance with $\sum F_i = 0$;

(b) No consideration is given to the effects of density change as a result of temperature change. There is no accumulation of mass;

(c) One dimensional flow; the volumetric effects are neglected. The properties vary in one direction; i.e., for a section perpendicular to flow all the properties remain constant, but these may vary in module in any other section perpendicular to the fluid. In formulating piping, cylindrical symmetry will be assumed at all times;

(d) Single phase flow; the fluid occurs in a single phase and there is no phase change;

(e) Steady state flow; the properties do not vary over time. In other words, given a set of bounding conditions, the properties of the system are constant;

(f) Shown below is a closed duct representing a node with its two connections. The figure shows the direction of variation of the properties (one dimensional flow), overall length L, the area of the section oriented transversally to the direction of flow and the speed v of the fluid (Fig. 64).

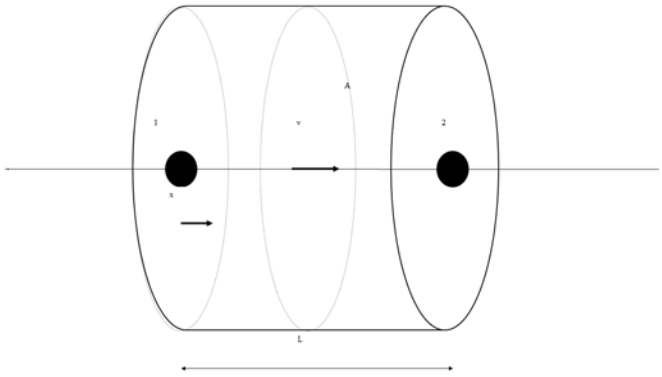


FIG. 634. Team_Flow cell.

In the closed duct shown in the figure, the mass flow at the inlet is equal to the mass flow at the outlet, due to application of the equation of continuity:

$$\frac{dF}{dx} = 0 \Rightarrow F_1 = F_2 \tag{70}$$

By applying the energy conservation equation integrally over the duct of length L, the following is obtained:

$$P_1 - P_2 = \rho \cdot g \cdot (Z_2 - Z_1) + \rho \cdot g \cdot H_L + \rho \cdot \frac{dv}{dt} \cdot L \tag{71}$$

The term $\rho \cdot g \cdot H_L$ represents the load losses due to friction in the piping, changes in cross-section, changes in direction. This term may generally be expressed as being proportional to speed squared.

$$\rho \cdot g \cdot H_L = k \cdot v^2 \xrightarrow{F=\rho \cdot v \cdot A} \rho \cdot g \cdot H_L = \frac{F^2}{a^2} \tag{72}$$

where

a is admittance of the duct;

ρ is fluid density;

Z_i is elevation of point i.

The resulting equation in terms of flow is as follows:

$$P_1 - P_2 = \rho \cdot g \cdot (Z_2 - Z_1) + \frac{F^2}{a^2} + \frac{L}{A} \cdot \frac{dF}{dt} \tag{73}$$

where

L is length of duct;

A is area of passage of flow of duct.

The following results from applying the steady state flow condition:

$$P_1 - P_2 = \rho \cdot g \cdot (Z_2 - Z_1) + \frac{F^2}{a^2} \tag{74}$$

From where it is possible to solve the mass flow trivially in square root terms.

$$F = a \cdot \sqrt{P_1 - P_2 - \rho \cdot g \cdot (Z_2 - Z_1)} \tag{75}$$

For resolution this has to be linearized due to its being a second order equation.

The linearization would give the following:

$$F = a_L \cdot \left(P_1 - P_2 - \rho \cdot g \cdot \left(Z_2 - Z_1\right)\right) \tag{76}$$

where

$$a_L = \frac{a}{\left(\sqrt{P_1^{'} - P_2^{'} - \rho \cdot g \cdot \left(Z_2 - Z_1\right)}\right)} \tag{77}$$

a_L is Linearized admittance;

$P_i^{'}$ is Pressure at point i in the previous time step.

A hydraulic network may be broken down into a set of ducts, to each of which may be applied the aforementioned equation. The system resulting from applying this equation to each of the ducts in the network, along with application of the continuity equation at the joints between ducts, allows the pressures and flows of the network to be obtained simultaneously.

The mass balance for each node i in the network will be as follows:

$$\sum_j F_{ij} + F_i^{ext} = 0 \tag{78}$$

Where:

F_i^{ext} is flows external to node I;

F_{ij} is flow from node i to node j.

In general, this flow will depend on the pressure and density of nodes i and j, as well as on the admittance of this duct. As has been seen in the previous paragraph, this function will be non-linear, although it may be linearized in the previous terms.

Consequently, the following general equation may be written for F_{ij} :

$$F_{ij} = \alpha_{ij} \cdot \Delta\Gamma_{ij} + \beta_{ij}$$

$$\Delta\Gamma_{ij} = \Delta P_{ij} + \Delta G_{ij} = P_i - P_j + \rho \cdot g \cdot (Z_i - Z_j) \tag{79}$$

Where P_i and P_j are the pressures of nodes i and j.

If the pressure of node j corresponds to a bounding condition, the following is obtained:

$$F_{ii} = \alpha_{ii} \cdot \Delta\Gamma_i + \beta_{ii} \tag{80}$$

and by applying the mass balance condition for each of the nodes, the following is obtained:

$$\left(\sum_j \alpha_{ii}\right) p_i - \sum_{j \neq i} (\alpha_{ij} p_j) = -\sum_j \beta_{ij} - F_i^{ext} \tag{81}$$

Evidently, the flow from node i to node j is equal to and of the opposite sign from the flow from node j to node i:

$$F_{ji} = -F_{ij} = \alpha_{ji}(p_j - p_i) + \beta_{ji} = \alpha_{ij}(p_j - p_i) - \beta_{ij} \tag{82}$$

for any node i and j, from where the following may be deduced:

$$\alpha_{ji} = \alpha_{ij} \tag{83}$$

$$\beta_{ji} = -\beta_{ij} \tag{84}$$

It may be appreciated, therefore, that we are dealing with a system of equations with a symmetrical, real and positive matrix:

$$\mathrm{A} \cdot \mathrm{P} = \mathrm{B} \tag{85}$$

Where, P is the vector of the unknown pressures of the system

This system is resolved using a standard Cholesky(*) factorization method.

Once the pressures are known, the calculation of mass flows between nodes may be accomplished by applying equations (79) (80)

5.3.2. Description of heat transfer between node and wall

The temperatures are calculated at the nodes. The calculation of the temperature of each node is accomplished via a process of node enthalpy balancing.

The calculation of the coefficients of heat transfer between the node and its wall and between the wall and the exterior is explained below. Also indicated is how the calculation of wall temperature is performed.

5.3.2.1. Coefficient of heat transfer between node and wall

Two regimes of transfer between the fluid and the wall have been contemplated: natural convection and forced convection.

If heat transfer is by forced convection:

$$H_{FORZ} = \frac{k_{FORZ} \cdot \mathrm{Re}^{0.8}}{D_i} \tag{86}$$

If heat transfer is by natural convection:

$$H_{NAT} = k_{NAT} \cdot \left(\left| \frac{T - T_{PARED}}{D_i} \right| \right)^{1/4} \tag{87}$$

The following is calculated:

$$H = \frac{1}{\max\left(H_{NAT}; H_{FORZ}\right)} + X_{PARED} \cdot k_{ESP} \tag{88}$$

Finally, the coefficient of heat transfer between the node and the wall is obtained:

$$h_{NW} = \frac{A_{PARED}}{H + H^i_{SUCIO} \cdot f^i_{SUCIO}} \tag{89}$$

If this coefficient is known, it is possible to determine the heat flux exchanged between the fluid crossing the node and the wall:

$$Q_{PARED} = h_{NW} \cdot \left(T_{PARED} - T\right) \tag{90}$$

where

H_{FORZ} is the coefficient of heat transfer in the case of forced transfer;

k_{FORZ} is the constant of heat transfer by forced convection;

H_{NAT} is the coefficient of heat transfer in the case of natural transfer;

k_{NAT} is the constant of heat transfer by natural convection;

X_{PARED} is thickness of wall;

k_{ESP} is constant quantifying the effect of wall thickness on coefficient of heat transfer;

H^i_{SUCIO} is coefficient of heat transfer by soiling (additional resistance to heat transfer);

f^i_{SUCIO} is factor affecting effect of soiling on heat transfer, in terms of performance time.

5.3.2.2. Coefficient of heat transfer between wall and exterior

In this case a distinction will be made between two initial cases, depending on whether the fluid enveloping the node is water (liquid) or air (gaseous).

If the fluid is a liquid, a calculation analogous to that described above will have to be performed:

If heat transfer is by forced convection:

$$H_{FORZ} = \frac{k_{FORZ} \cdot \mathrm{Re}^{0.8}}{D_e} \tag{91}$$

If heat transfer is by natural convection:

$$H_{NAT} = k_{NAT} \cdot \left(\left| \frac{T - T_{PARED}}{D_e} \right| \right)^{1/4} \tag{92}$$

The following is calculated:

$$H = \frac{1}{\max\left(H_{NAT}; H_{FORZ}\right)} + X_{PARED} \cdot k_{ESP} \tag{93}$$

If it is gaseous, the case of forced convection is not considered and the calculation is simplified, the equation being left as follows:

$$H = KL_{gas} + X_{PARED} \cdot k_{ESP} \tag{94}$$

The coefficient of heat transfer between the wall and the exterior is finally obtained:

$$h_{WO} = \frac{A_{PARED}}{H + H^{e}_{SUCIO} \cdot f^{e}_{SUCIO}} \tag{95}$$

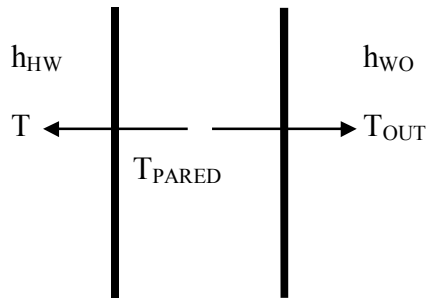
By knowing this coefficient, it is possible to determine the heat flux exchanged between the wall and the exterior:

$$Q_{EXT} = h_{WO} \cdot \left(T_{PARED} - T_{OUT}\right) \tag{96}$$

where

KL_{gas} is constant of heat transfer in the case of natural convection and a gaseous fluid.

5.3.2.3. Calculation of wall temperature



The balance of energy in the wall is considered:

$$Q_{PARED} = -(Q_1 \cdot \Delta t + Q_2 \cdot \Delta t) \tag{97}$$

$$C_E \cdot \Delta T_{PARED} = -(h_{HW} \cdot (T_{PARED} - T) \cdot \Delta t + h_{WO} \cdot (T_{PARED} - T_{OUT}) \cdot \Delta t) \tag{98}$$

The following is obtained:

$$\Delta T_{PARED} = T_{PARED} - T^*_{PARED} \tag{99}$$

Replacing and solving the temperature of the wall:

$$T_{PARED} = \frac{T^*_{PARED} \cdot C_E + \Delta t \cdot (T \cdot h_{HW} + T_{OUT} \cdot h_{WO})}{C_E + \Delta t \cdot (h_{HW} + h_{WO})} \tag{100}$$

where

T_{PARED} is wall temperature;

T^*_{PARED} is wall temperature in previous time step;

T_{OUT} is exterior temperature;

C_E is specific heat coefficient of wall;

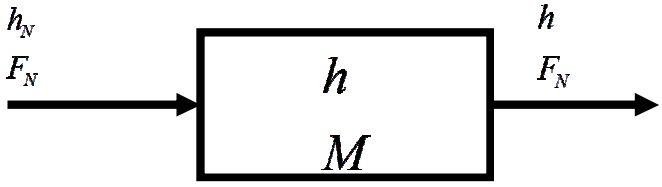
h_{HW} is coefficient of film heat transfer between fluid and wall;

h_{WO} is coefficient of film heat transfer between wall and exterior;

Δt is time step;

5.3.2.4. Temperature network calculation

In each node an enthalpy balance is performed as shown in the sketch below:



$$M \frac{dh}{dt} = -F_N \cdot (h - h_N) \tag{101}$$

Reordering:

$$\frac{dh}{h - h_N} = -\frac{F_N \cdot dt}{M} \tag{102}$$

Integrated in the time step:

$$\ln\left(\frac{h-h_N}{h^*-h_N}\right)=-\frac{F_N\cdot\Delta t}{M} \tag{103}$$

By applying exponentials to the two members of the equation we obtain the following:

$$\frac{h-h_N}{h^*-h_N}=e^{-\frac{F_N\cdot\Delta t}{M}} \tag{104}$$

The following term is known as the transport factor: $e^{-\frac{\Delta t\cdot F_N}{M}}$.

By solving the following may be written:

$$h=h_N+e^{-\frac{\Delta t\cdot F_N}{M}}\cdot\left(h^*-h_N\right) \tag{105}$$

Generalizing, the following expression reflects the calculation of the specific enthalpy in a generic node, taking into account the transfer of energy from the wall and the external energy contributions made to the node:

$$h=h_N+e^{-\frac{\Delta t\cdot F_N}{M}}\cdot\left(h^*-h_N\right)+\left(\frac{Q_{Pared}+Q_{Ext}}{M}\cdot\Delta t\right) \tag{106}$$

Terms used:

F_N mass flow across the node, calculating $F_N = \Sigma F_i \quad (F_i > 0)$

M mass of node.

Δt time step.

h_N specific enthalpy of flows entering node

$$\begin{aligned} F_N &= \Sigma F_i \\ h_N &= \frac{\sum F_i\cdot h_i}{F_N} \quad (F_i>0) \end{aligned} \tag{107}$$

h specific enthalpy of node.

h^* specific enthalpy of node in previous time step.

Q_{Pared} heat flux exchanged between node and wall

Q_{Ext} heat flux contributed from exterior.

The specific enthalpy in each node having been calculated, the temperature will be as follows:

$$T = \frac{h}{C_{pl}} \tag{108}$$

T temperature in node.

C_{pl} specific heat of fluid.

5.4.ELECTRICAL SYSTEM (GEN)

TEAM_ELECTRIC is the tool used for modelling GEN behaviour.

TEAM_ELECTRIC is a tool designed for the modelling of electrical systems that is especially aimed at simulation in actual time. In addition to providing the intensities and voltages of the network, it allows for the calculation of active, reactive and apparent power, as well as frequencies.

5.4.1. Description of electrical network resolution

In principle, and once the topology of the electrical circuit and the values of impedance of the connections between nodes are known, there are various alternatives for calculation of the voltage and intensity values (iterative methods, algebraic methods, etc.).

The algebraic method is considered to be more appropriate for both its robustness and its time intensity. On the basis of a set of vectors of voltages and currents between nodes (connections), this method makes it possible to gain insight into the rest of voltages and currents by applying the algebraic relations that link the voltages and currents at all the connections.

The algebraic method must obtain the solution by converting the network into a matrix. In this way, by means of matrix calculation it is possible to determine the electrical variables. Evidently, the Kirchoff laws and equivalent Norton circuits must be used.

Applying Ohm's law:

$$\text{Intensity} = \text{Admittances} * \text{Nodal voltage} \tag{109}$$

And in matrix form:

$$[\mathbf{I}] = [\mathbf{Y}] * [\mathbf{V}] \tag{110}$$

Evidently, the calculation must be performed to obtain first the voltages at the nodes (busses/bars) and subsequently, and depending on these, calculate the rest of the electrical variables (Fig. 65).

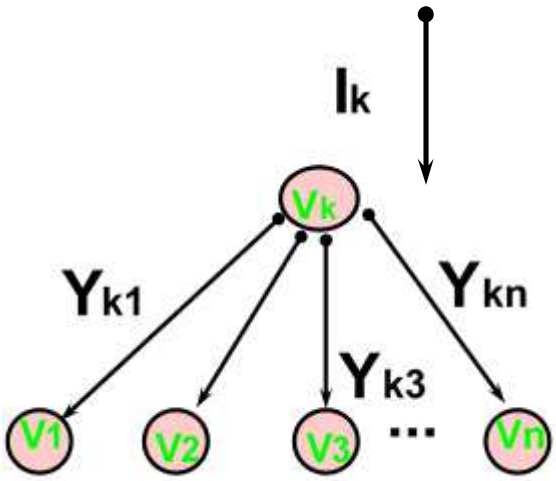


FIG. 65. Voltages-Currents calculation.

$$\mathbf{I}_k = \sum_{n=1}^{N} \mathbf{Y}_{kn} * \mathbf{V}_n \tag{111}$$

Resolving to obtain the voltage:

$$[\mathbf{V}] = [\mathbf{Y}]^{-1} * [\mathbf{I}] \tag{112}$$

The solution for intensities (active and reactive) between nodes:

$$\mathbf{I}_{nk} = (\mathbf{V}_n - \mathbf{V}_k) * \mathbf{Y}_{nk} \tag{113}$$

The solution for powers (active and reactive):

$$\mathbf{S}_{nk} = \mathbf{V}_n * \mathbf{I}^*_{nk} \tag{114}$$

The solution for sequences: where I^* is the complex conjugate

$$\mathbf{F}_k = \frac{\sum_{s=1}^{NS} \mathbf{P}_s * \mathbf{F}_s}{\sum_{s=1}^{NS} \mathbf{P}_s} \tag{115}$$

5.5.LOGICS (PCS)

TEAM_LOGIC is the tool used for modelling logics of valves, pumps, controllers and PCS.

TEAM_LOGIC is a tool designed for the modelling of any logic. Tool has different graphic icons that represent a specific calculation in Fortran, for example a sum, a multiplication,

comparison... The connection among the icons (equations) create the logic of the valve/pump… the icons (equations) are solved in sequential way.

APPENDIX

A.1. SUMMARY OF THERMAL HYDRAULIC VARIABLES

The summary of thermal hydraulics variables is shown in TABLE 5

TABLE 5. THERMAL HYDRAULIC VARIABLES

Thermal hydraulic variable	Value	Units
Reactor neutron power	100.0	%
Thermal power	150.0	MW
Generator power	45.0	MW(e)
Primary pressure	15.5	MPa
Pressurizer level	43.0	%
Core level	100.0	%
Core flow	422.0	Kg/s
Inlet core temperature	255.51	ºC
Outlet core temperature	320.36	ºC
Average core temperature	288.0	ºC
Steam header pressure	2.72	MPa
Steam reheating	29.0	ºC
Steam flow	77.0	Kg/s
Feedwater flow	77.0	Kg/s
Feedwater temperature	173.0	ºC
Turbine speed	3600.0	rpm
Condenser vacuum	47.81	mmHg
Lake temperature	20.0	ºC
Containment pressure	0.011	MPa
Containment temperature	33.4	ºC

A.2. LIST OF MALFUNCTIONS

A list of malfunctions is provided in Table 6.

TABLE 6. SPECIFIC MALFUNCTION

Number	System	Specific Malfunction
1	ADS	Inadvertent initiation of ADS
2	CBS	Loss of containment vacuum
3	CNR	Loss of condenser vacuum
4	CWS	Condenser coolant pumps trip
5	DHR	Inadvertent initiation of decay heat removal system
6	FWS	Reduction in feedwater temperature (loss of FW heating)
7		Abnormal increase in FW flow
8		Loss of normal feedwater flow (FW pumps trip)
9		Feedwater system pipe break
10	GEN	Station blackout (loss of AC power)
11	MSS	Steam header break
12		Steam generator tube failure 1
13		Steam generator tube failure 2
14		Major steam system piping failure within containment
15	RCS	Reactor setback fail
16		One bank of shutdown control rods drop into the core
17		Charging (feed) valve fails open
18		Inadvertent operation of pressurizer heaters
19		Uncontrolled control rod assembly withdrawal at power
20		Fail of PZR control system
21		Failure of main coolant pumps
22		Reactor stepback fail
23		Seismic event
24	TUR	Turbine spurious trip
25		Turbine spurious runback
26		Turbine trip with bypass valves failed closed

A.3 LIST OF SETPOINTS

The list of setpoints is given in Table 7.

TABLE 7. LIST OF SETPOINTS

System	set point	Value
RCS	Heaters	P <15.5 MPA
	Spray	P >15.5 MPA
	Relief RCS valve	P>16.7 MPA
	Safety RCS valve	P>17.05MPA
	Alarm low RCS pressure	P<12.5MPA
PCS-Trips	Low pressure upper plenum	P <11. MPA
	Low level PZR	L<5%
	Low flow downcomer	W<347Kg/s
	High core outlet temperature	T>340ºC
	High reactor neutron core flux	Flux>120%
	High log rate	SUR>2dpm
	High pressure upper plenum	P>16.4
PCS-Setbacks	High PZR level	L>67%
	High steam header pressure	P>4.9 MPA
PCS-Stepbacks	High zone flux	Flux>115%
ADS	Low low pressure upper plenum	P <9 MPA
	Low low level vessel	L<90%
	High-high pressure upper plenum	P>17.2MPA
	High-high pressure CBS	P>0.019MPA
CBS	Alarm high CBS pressure	P>0.012MPA
CCS	CCS starts at CBS pressure	P>0.019MPA
PIS	Pressure set point	P<5MPA
GIS	Pressure set point	P<0.5MPA
CNR	Alarm low CNR vacuum	P>100 mmHg
	Loss of CNR vacuum	P>254 mmHg
MSS	Relief RCS valve	P>5.3 MPA
	Safety RCS valve	P>5.7 MPA

REFERENCES

[1] WORLD NUCLEAR ASSOCIATION, Small nuclear power reactor, World Nuclear Association, London (2016).

[2] NUSCALE POWER, L. L. C, NuScale Plant Design Overview, RP-1114-9375 Rev0, NuScale Power, Oregon (2014).

[3] INTERNATIONAL ATOMIC ENERGY AGENCY, Advances in Small Modular Reactor Technology Developments, A supplement to: IAEA Advanced Reactors Information System (ARIS), IAEA, Vienna (2014).

[4] CNEA, History, current status and future projections of CAREM project, I&C Workshop, Bariloche (2007).

[5] TECNATOM, "NEMO_RT, 3-D Neutronic code for light water reactors" TECNATOM (2011).

[6] TECNATOM, "TRAC_RT, Real time simulation code for light water reactors general description", TECNATOM (2011).

ABBREVIATIONS

ADS	AUTOMATIC DESPRESSURIZATION SYSTEM.
AI	ANALOGICAL INPUT
BOL	BEGINNING OF LIFE
BOP	BALANCE OF PLANT
BWR	BOILING WATER REACTOR
CBS	CONTAINMENT BUILDING SYSTEM.
CCS	CONTAINMENT COOLING SYSTEM.
CNR	CONDENSER SYSTEM.
CWS	CIRCULATING WATER SYSTEM.
DHR	DECAY HEAT REMOVAL
DNBR	DEPARTURE FROM NUCLEAR BOILING RATIO
EOL	END OF LIFE
FWS	FEEDWATER SYSTEM
FC	FORCE CIRCULATION
GEN	GENERATOR SYSTEM
GIS	GRAVITY INJECTION SYSTEM
GUI	GRAPHIC USER INTERFACE
HFE	HUMAN FACTORS ENGINEERING
IAEA	INTERNATIONAL ATOMIC ENERGY AGENCY
IC	INITIAL CONDITION
iPWR	INTEGRAL PRESSURIZED WATER COOLED REACTOR
LWR	LIGTH WATER REACTOR
LOCA	LOSS OF COOLANT ACCIDENT
MOL	MEDIUM OF LIFE

MSS	MAIN STEAM SYSTEM
NC	NATURAL CIRCULATION
PCS	PROTECTION AND CONTROL SYSTEM.
PDHR	PASSIVE DECAY HEAT REMOVAL SYSTEM.
PIS	PRESSURE INJECTION SYSTEM
POAH	POINT OF ADDING HEAT
PWR	PRESSURIZED WATER REACTOR
PZR	PRESSURIZER
RCS	REACTOR COOLANT SYSTEM.
RPV	REACTOR PRESSURE VESSEL
SBO	STATION BLACKOUT
SCS	SHUTDOWN COOLING SYSTEM
SG	STEAM GENERATOR
SMR	SMALL MODULAR REACTOR
SUR	STARTUP RATE
TUR	TURBINE SYSTEM

CONTRIBUTORS TO DRAFTING AND REVIEW

Abarca, D.	TECNATOM, Spain
Batra, C.	International Atomic Energy Agency
De la Torre, R.	TECNATOM, Spain
Fernandez, D.	TECNATOM, Spain
Fuentes, A.	TECNATOM, Spain
Lenhardt, G.	TECNATOM, Spain
López, J.	TECNATOM, Spain
Maroto, E.	TECNATOM, Spain
Painter, C.	International Atomic Energy Agency
Parrado, I.	TECNATOM, Spain
Ruiz, J.A.	TECNATOM, Spain
Sancho, A.	TECNATOM, Spain
Subki, M.H.	International Atomic Energy Agency
Villalba, S.	TECNATOM, Spain

IAEA
International Atomic Energy Agency

No. 25

ORDERING LOCALLY

In the following countries, IAEA priced publications may be purchased from the sources listed below or from major local booksellers.

Orders for unpriced publications should be made directly to the IAEA. The contact details are given at the end of this list.

CANADA

Renouf Publishing Co. Ltd

22-1010 Polytek Street, Ottawa, ON K1J 9J1, CANADA
Telephone: +1 613 745 2665 • Fax: +1 643 745 7660
Email: order@renoufbooks.com • Web site: www.renoufbooks.com

Bernan / Rowman & Littlefield

15200 NBN Way, Blue Ridge Summit, PA 17214, USA
Tel: +1 800 462 6420 • Fax: +1 800 338 4550
Email: orders@rowman.com Web site: www.rowman.com/bernan

CZECH REPUBLIC

Suweco CZ, s.r.o.

Sestupná 153/11, 162 00 Prague 6, CZECH REPUBLIC
Telephone: +420 242 459 205 • Fax: +420 284 821 646
Email: nakup@suweco.cz • Web site: www.suweco.cz

FRANCE

Form-Edit

5 rue Janssen, PO Box 25, 75921 Paris CEDEX, FRANCE
Telephone: +33 1 42 01 49 49 • Fax: +33 1 42 01 90 90
Email: formedit@formedit.fr • Web site: www.form-edit.com

GERMANY

Goethe Buchhandlung Teubig GmbH

Schweitzer Fachinformationen
Willstätterstrasse 15, 40549 Düsseldorf, GERMANY
Telephone: +49 (0) 211 49 874 015 • Fax: +49 (0) 211 49 874 28
Email: kundenbetreuung.goethe@schweitzer-online.de • Web site: www.goethebuch.de

INDIA

Allied Publishers

1st Floor, Dubash House, 15, J.N. Heredi Marg, Ballard Estate, Mumbai 400001, INDIA
Telephone: +91 22 4212 6930/31/69 • Fax: +91 22 2261 7928
Email: alliedpl@vsnl.com • Web site: www.alliedpublishers.com

Bookwell

3/79 Nirankari, Delhi 110009, INDIA
Telephone: +91 11 2760 1283/4536
Email: bkwell@nde.vsnl.net.in • Web site: www.bookwellindia.com

ITALY

Libreria Scientifica "AEIOU"

Via Vincenzo Maria Coronelli 6, 20146 Milan, ITALY
Telephone: +39 02 48 95 45 52 • Fax: +39 02 48 95 45 48
Email: info@libreriaaeiou.eu • Web site: www.libreriaaeiou.eu

JAPAN

Maruzen-Yushodo Co., Ltd

10-10 Yotsuyasakamachi, Shinjuku-ku, Tokyo 160-0002, JAPAN
Telephone: +81 3 4335 9312 • Fax: +81 3 4335 9364
Email: bookimport@maruzen.co.jp • Web site: www.maruzen.co.jp

RUSSIAN FEDERATION

Scientific and Engineering Centre for Nuclear and Radiation Safety

107140, Moscow, Malaya Krasnoselskaya st. 2/8, bld. 5, RUSSIAN FEDERATION
Telephone: +7 499 264 00 03 • Fax: +7 499 264 28 59
Email: secnrs@secnrs.ru • Web site: www.secnrs.ru

UNITED STATES OF AMERICA

Bernan / Rowman & Littlefield

15200 NBN Way, Blue Ridge Summit, PA 17214, USA
Tel: +1 800 462 6420 • Fax: +1 800 338 4550
Email: orders@rowman.com • Web site: www.rowman.com/bernan

Renouf Publishing Co. Ltd

812 Proctor Avenue, Ogdensburg, NY 13669-2205, USA
Telephone: +1 888 551 7470 • Fax: +1 888 551 7471
Email: orders@renoufbooks.com • Web site: www.renoufbooks.com

Orders for both priced and unpriced publications may be addressed directly to:

Marketing and Sales Unit
International Atomic Energy Agency
Vienna International Centre, PO Box 100, 1400 Vienna, Austria
Telephone: +43 1 2600 22529 or 22530 • Fax: +43 1 2600 29302 or +43 1 26007 22529
Email: sales.publications@iaea.org • Web site: www.iaea.org/books

ISSN 1018-5518