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TRANSIENT ANALYSIS OF THE IRIS REACTOR

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ABSTRACT

An international consortium of industry, laboratory, university and utility establishments, led by Westinghouse, is developing a modular, integral, light water cooled, small to medium power reactor, the International Reactor Innovative and Secure (IRIS). IRIS features innovative, advanced engineering, but it is firmly based on the proven technology of pressurized water reactors (PWR).

Given the large number of organizations involved in the IRIS design, the RELAP5/MOD 3.3 code has been selected as the main system code. A nodalization of the reference IRIS design has been developed with a basic set of protective functions and controls. Engineered Safety Features of the concept are being also implemented, and in particular the Emergency Heat Removal System that is used for safety grade decay heat removal and in the small break LOCA response of IRIS (Large break LOCAs are eliminated in IRIS by the adoption of the Integral layout) This paper discusses developed model and transient behavior of the system for representative transient sequences.

1 INTRODUCTION

Introduction of the new reactor concepts poses great challenge to the development and use of suitable analytical tools for the transient analysis of the concept in question. Transient analyses are commonly used in many aspects of the project and cover safety and design assessment . State-of the-art system codes have achieved sufficient maturity and have been successfully used for many purposes in reactor technology development, e.g. licensing process, design optimization etc.

IRIS (International Reactor Innovative and Secure) is a next generation advancement of the pressurized water reactor (PWR) that addresses the Generation IV goals, i.e. enhanced reliability and safety, and improved economics. It has been selected as an International Near Term Deployable (INTD) reactor, within the Generation IV International Forum activities.

One of the main characteristics of the design is the integral vessel configuration that enhances safety performance of the IRIS reactor. Since IRIS is essentially an Integral Primary System Reactor (IPSR) plant, and as such, with some modifications/improvements, state-of-the-art computer codes can be successfully used for its transient analysis. Therefore, widely used RELAP5/MOD3.3 computer code has been chosen for this purpose. Specific Westinghouse proprietary codes will be used to address specific phenomena (core subchannel analyses and departure from nucleate boiling evaluations, fuel performance,…). CFD tools will be used to evaluate mixing effects for some asymmetrical events.

2 IRIS REACTOR OVERVIEW

IRIS is based on proven LWR technology that will employ new engineering to implement attractive and innovative features, but without the need for any new technology development, [1]. Due to its modularity, the plant size on site could ranges from several hundreds to thousands of electric megawatts. The design is being developed by an international consortium led by Westinghouse/BNFL and comprise of about 20 organizations from all over the world.

2.1 IRIS System Configuration

The integral vessel houses the reactor core and support structures, core barrel, upper internals, control rod guides and drivelines, pressurizer located in the upper head, internal shielding and eight helical coil steam generators (SG) coupled with eight low-head spool type primary reactor coolant pumps, Figure 1. SGs deliver superheated steam (around 40 K) to the secondary system., primary coolant pumps are located on the top of each SG and circulate primary coolant through the shell side of each steam generator. A large pressurizer located in the upper head portion of the vessel yields a good inertance to pressure surges.

Since primary coolant pumps are completely contained inside the vessel, large vessel penetrations are eliminated. The vessel has a height of ~22 m and an outside diameter of ~6.7 m, a size which is still within the state-of-the-art fabrication capabilities. Further details on main reactor components such as, reactor vessel and internals, helical steam generators, and spool pumps can be found, respectively, in Refs.[2-4].

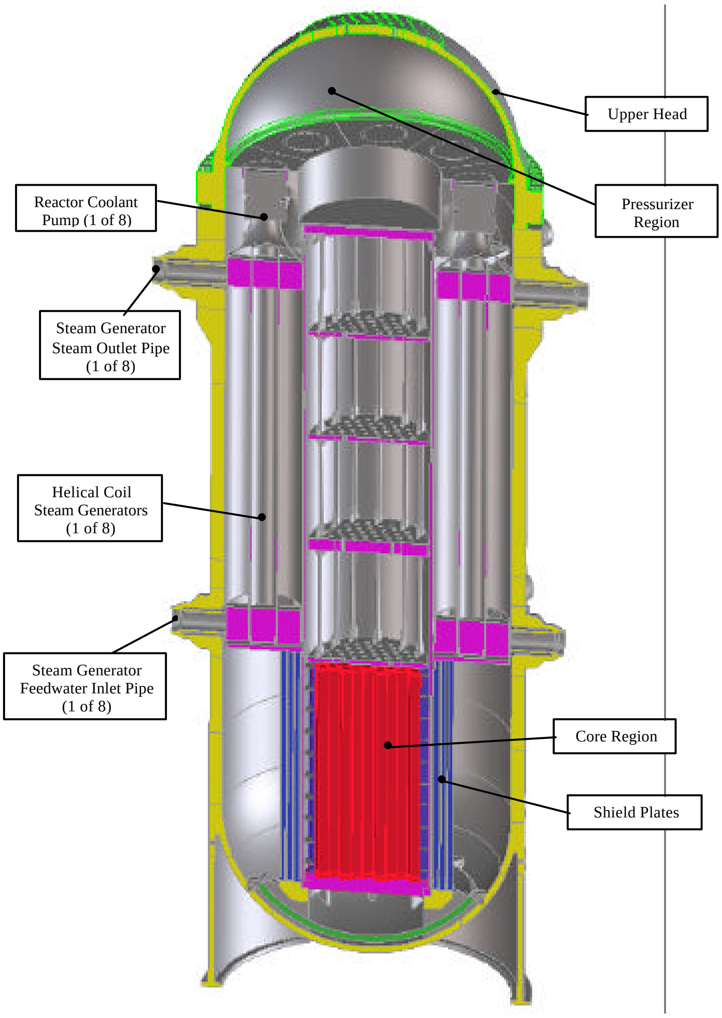

Figure 1 IRIS Configuration

2.2 Inherent Safety Features of the IRIS Reactor

The principles of safety-by-design have been previously reported, [7], [8] and are a consequence of the approach to designing IRIS by taking maximum advantage of the opportunities offered by the integral configuration to:

- Physically eliminate the possibility for some accidents to occur,
- Decrease the probability of occurrence for majority of the other accident scenarios and
- Eliminate/reduce the consequences if an accident actually occurs

The integral configuration and the absence of large vessel penetrations eliminate the possibility for large LOCAs to occur. The low pressure-drop path and the large distance between the thermal center of the steam generators and the core enhance natural circulation.

IRIS Response to small break LOCAs is based on the strong coupling of the reactor and the very compact containment that reduces initial blowdown, and on the depressurization of the system through steam condensation on the passive natural circulation heat removal system connecting the steam generators with heat exchangers located outside the containment. The ultimate result is that during a small-to-medium LOCA the core remains covered for an extended period of time (several days and possibly weeks) without any need for emergency water injection or core makeup.

Suppression pools have the role of limiting the initial containment pressure peak, and they also eventually double the makeup water inventory, acting as a gravity driven source.

3 RELAP5 MATHEMATICAL MODEL OF THE IRIS REACTOR

Development of the first RELAP5 model for IRIS reactor was initiated in year 2001, and the results of this preliminary work are described in [9]. Different IRIS partners also performed studies to define the thermal-hydraulic characteristic and preliminary design of the main systems and components. Examples of this activity are presented in sections 3.1 and 3.2. Input was also provided for the Protection System, the Control System and the Neutronic Feedback and Reactor Kinetics coefficients. The preparation of the IRIS nodalization has been the result of an international effort that involves several organizations: responsibility for different parts of the IRIS thermal-hydraulic design is shared between the partners according to Table 1. The University of Zagreb has been chosen to prepare and maintain reference nodalization with accompanying documentation based on the inputs from other institutions. Developed model is currently under the review by other organizations.

Table 1 Preliminary Thermal-Hydraulic Design for Safety Analyses: Work Breakdown

IRIS System/Component	Responsibility (Organizations)
Primary System and Protection/Control System	
Reactor Pressure Vessel (RPV)	University of Zagreb - FER
Core Thermal Hydraulic Design	Westinghouse - WEC
Pressurizer (PRZ)	Nuclear Energy Commission - CNEN
Reactor Coolant Pumps (RCP)	Washington Group (W-EMD), University of Pisa - UNIPI, WEC
Steam Generators (SG)	ANSALDO, Polytechnic of Milan - POLIMI
Reactor Protection System	WEC
Reactor Control System (RCS)	WEC, CNEN, Ansaldo,
Neutronic Feedback Coefficients	WEC
Balance of Plant	
Secondary System	FER
Safety Systems	
Emergency Heat Removal System (EHRS)	POLIMI, WEC
Automatic Depressurization System (ADS)	WEC
Emergency Boration Tank	WEC, FER
Long Term Core Makeup System	WEC
Other Engineered Safety Features (ESF)	WEC

3.1 IRIS Integral Pressurizer Design Studies

Initial design analyses of the IRIS pressurizer were performed by CNEN using updated first RELAP model, [9]. The simulation was divided in four different phases:

1. Controlled Steady State: It is characterized by the artificial control of the pressure and level in the pressurizer.
2. Free Steady State: It does not uses the artificial controls described in the previous phase and the pressurizer behavior is determined by its own characteristics as dimensions, power of the heaters, thermal insulation and system conditions.
3. Power Variation: During this lap the power delivered by the core and that transferred to the secondary of the steam generator are simultaneously reduced from 100% (1000MW) to 60% (600MW).
4. Stabilization: It was calculated to show the behavior of the pressurizer/system after the power change.

The results showed that the proposed pressurizer design could be able to accommodate, as condition I events, steps and ramps more severe than what is currently being proposed for the system in normal operation. Detailed evaluation of the pressurizer shall be performed using a real comprehensive test set with the model described in section 3.3.

3.2 RELAP5 Analysis of IRIS Emergency Heat Removal System

Polytechnic of Milan developed the RELAP model of the IRIS EHRS, together with the pool condenser dimensioning. This computer model is based on a preliminary design of the IRIS EHRS, which is composed of four identical loops, each being connected with two helical tube steam generators and one pool condenser, all condensers being placed in the same pool (Refueling Water Storage Tank).

In order to verify the EHRS performance a primary side steady boundary condition was reproduced (pressure 15.50 MPa, mass flow 4484 kg/s, average temperature 584.15 K), and then the EHRS was connected and allowed to reach a steady state. The results of the analysis showed satisfactory overall behavior of the EHRS RELAP model. Before EHRS connection stagnant vapor fills the hot leg up to the siphon while stagnant cold liquid is present in the pool condenser and in the return leg, thus the siphon is effective in isolating the EHRS from the plant in normal operation. The pool condenser dynamics appears noding dependent, i.e. its thermal performance improves slightly increasing the number of nodes. The 20 nodes model has been suggested to be implemented in the IRIS RELAP5 model as good compromise between precision and computational time consumption.

3.3 RELAP5 Nodalization of the IRIS Reactor

Experience gained with the use of the preliminary model, [9], was used to develop more detailed model that would balance between required level of detail, challenges posed by integral concept of the IRIS reactor and current project needs regarding licensing and design.

The structure of the nodalization is simple, Figure 2, and takes into account currently available geometrical and operational data. Besides relatively simple nodalization structure, discretization of the components is rather detailed in order to take into account all important phenomena. All the main flow paths are modeled with sufficient detail, with almost all of the minor flow paths. Total number of volumes and junctions is 1398 and 1419, respectively. Sliced approach was used in the discretization of the reactor vessel taking into account importance of natural circulation in chosen safety concept. Most of the calculational nodes have linear size in range 0.2 to 0.5 m. The nodalization was prepared so to maintain free volume of the system and elevation differences (due to importance of natural circulation) as well as core and SG heat exchange areas.

Eight RCP/SG modules are explicitly modeled. Original idea was to use 1-1-2-4 lumping what is enough to take into account different accident sequences and actuation of safety systems. It was decided to use explicit modeling in order to better address physical phenomena, take into account interaction of SG modules and EHRS loops (asymmetry due to different length of feed and steam lines) and preclude possible artificial recirculation in parallel loops introduced due to lumping or numerical reasons. The pumps are described using preliminary homologous curves in first quadrant with dummy zero head/zero torque curves provided for second quadrant. A coastdown characteristic provided by the pump designers is used to follow the pumps coastdown. Outside the reactor vessel and primary system, each feed/steam line (two SGs are connected to each) is modeled up to corresponding isolation valves. The Only safety system presently modeled is EHRS, but FER and Westinghouse are currently updating the model to include all the IRIS Engineered Safety Features. PORV and SV valves are located on top of pressurizer. Their exact position and cross section area are being defined by CNEN.

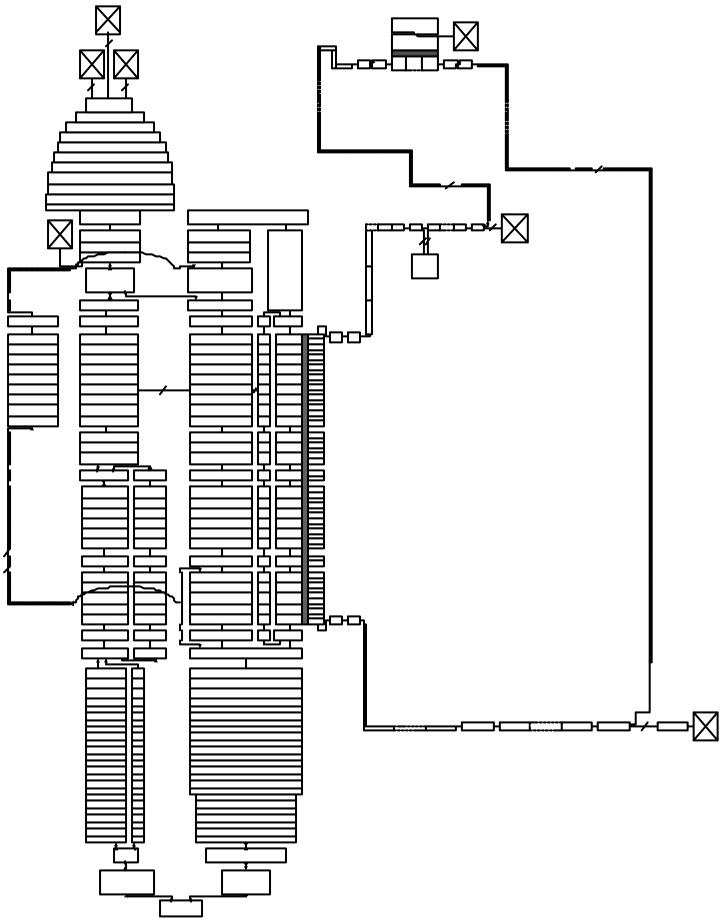

Figure 2: RELAP5/mod3 nodalization of the IRIS reactor

All the main heat structures are included in the model. On primary side that means core, baffle, barrel steel reflector, axial and radial shields, vessel wall, pump casing and some of the internal plates. On the SG side module tubes, collectors and outer shrouds, steam, feedwater and EHRS piping pipe walls are modeled. The model includes EHRS heat exchanger piping and collectors. Outer surface of the vessel is assumed to be insulated.

The model also includes a preliminary version of the basic reactor protection system that will be improved and extended after preliminary accident sequences analysis. Control variables are provided for calculation of: transferred power (core, SG, EHRS), fluid mass in all important parts of the nodalization and for some irreversible pressure losses.

Core heat source is based on power versus time table or point kinetics model. The point kinetics input is preliminary, based on limited available IRIS data.

3.4 Steady State Qualification of the IRIS nodalization

Limited steady state qualification of the IRIS RELAP5/MOD3.3 model has been performed. It is usual to compare calculated data to reference design or measured data in frame of the steady state qualification, but in this specific stage of the project development, reference operating full power data were envisaged by the designers. Taking into account early phase some of the data are still missing, like flows of the minor flow paths inside RPV and are not covered by this qualification. Reference pressure drops are given only for core and SGs as well as estimation of required pump head. The RELAP model will be used to estimate the importance of other minor pressure drops in the system.

Usually 200 s are used for steady state qualification, but in this case duration of steady state run was 1000 s in order to take into account large time constants associated to large water, low velocities and large metal masses that are present in IRIS design.

The comparison of the reference IRIS data and the calculated values after 1000 s of RELAP5/MOD3.3 steady state run is shown in Table 2. Agreement for most of the parameters is acceptable. Calculated irreversible core pressure drop is larger than reference due to the recent change from 15x15 to 17x17 fuel assembly (FA) design (the reference value for pump head is estimation based on core 15x15 FA design). Most of the equilibrium values were reached within first 100 s of calculation. Rate of the change after first 200 s is slow and the values are approximately constant. Steady state results are satisfactory for this phase of the project and nodalization was successfully used for evaluation of primary pressure drops and influence of steam line pressure drops on transferred power.

Table 2 Comparison between IRIS reference values and calculated steady state data

Parameter	Unit	Reference	Relap5 mod 3.3
Pressurizer pressure	MPa	15.5	15.56
BE vessel flow	kg/s	4707	4702
BE core flow	kg/s	4504	4503.6
Core inlet temperature	K	565.2	564.7
Core outlet temperature	K	601.5	601.4
SG pressure	MPa	5.8	5.79-5.82
Steam exit temperature	K	590.2	589.6-590.2
Total steam flow	kg/s	502.8	502.8
Δp core	kPa	52.0	53.8
Δp SG1 prim/sec	kPa	72.0/296	70.7/294.4
Core power	MW	1000.0	1000.0
Total SG power	MW	1001.47	1000.3
RCP head	m	19.1 (18.3-21.3)	19.8

4 THERMAL-HYDRAULIC ANALYSIS OF THE IRIS REACTOR

4.1 Scope of the Preliminary Transient Analysis

Scope of the preliminary transient analysis of the IRIS reactor is bounded by three main objectives:

1. **Understanding of IRIS behavior in different transients and accidents.** For most accidents, IRIS is similar to current PWRs, and therefore the same evaluation models may be used for the accident analysis. Merit parameters (DNBR, system pressure/temperature, RV water inventory) are practically the same as for current PWRs. In this frame one representative event is selected for each category of accidents. Then, selection of parameters and assumptions that have to be verified are deduced based on the single failure criterion and the degree of “conservatism” used/achieved for IRIS. Due to a “best-estimate” nature of the RELAP5 code and integral design of the IRIS, “conservatism” in this sense is defined by an appropriate selection of initiating conditions and sequence definition. Event is then studied at different times during life (BOC, EOC and MOC) to consider the full range of possible core and plant conditions (boron concentration can be a major influence). Results for key merit parameters are compared to acquire confidence in model response and confirm that “conservatisms” used are appropriate. Finally, response of the IRIS RELAP model is rationalized in comparison to current PWR experience and AP1000/AP600 results.

2. **Verify IRIS response to Safety Analysis Report sequences.** Meeting of the NRC acceptance criteria for different transients and accidents is evaluated by examining the safety response of IRIS and the effect of "Safety By Design". In this frame, all typical PWR accident sequences included in NRC Standard Review Plan (SRP) are to be studied. For each sequence, compliance to regulations by IRIS must be verified. However, one must bear in mind that at this stage of the project, the IRIS safety analysis is not a Safety Analysis Report (SAR). Many design details are still missing, the plant RELAP model is still in early development stage, and is not and can not be verified and validated due to a lack of data. Also, RELAP code requires different assumptions than Westinghouse codes that are typically used in current plants SARs. Therefore, accident sequences that will target this objective have been defined largely on the basis of PWR experience, trying to maximize the probability that proposed sequences will be acceptable to NRC. To summarize, the objective is to develop a comprehensive set of analyses, with a degree of "conservatism" that is expected to bound effective plant design limits, and to provide confidence that all requirements will be met once the real plant SAR will be developed.
3. **Assist in the system design.** Use of preliminary safety analysis and sensitivity studies play important role in the process of specification of the "final" design requirements. In particular, these preliminary analyses will play an important role in the definition/sizing of the safety systems and in the definition of the protection system.. Given the differences between IRIS and AP1000, both in overall plant layout and in safety systems functions/definition, a sufficiently wide spectrum of analyses is expected to acquire a sufficient confidence that the sizing of safety systems is appropriate

4.2 Preliminary Results of the Transient Analysis

Considering current status of the nodalization development, a set of transient sequences has been chosen for the nodalization testing purpose. The objective of the testing was to detect errors in the model (together with review process), improve nodalization, check accident sequences and provide examples and prepare guidelines for usage in the "IRIS transient analysis group". In this frame following transients/accidents were utilized.

Arbitrary sequence

Arbitrary sequence is essentially reactor trip with forced operation of different number of EHRS loops. This analysis verified proper implementation and minor modifications of the POLIMI model of the EHRS. Capability of the EHRS to remove decay heat is visible from . The case with 4 EHRS was interrupted at 400 s to prevent the actuation of the Emergency Boration Tank that was not yet implemented in the model.

Long term behavior

This test of the model consists of the introduction of reactor scram with one EHRS loop active for the duration of 3 hours. This fictitious sequence has been chosen to check long term heat removal capabilities. At the same time, the sequence has been used in Reactor SG Connections (RSC) design support: 8 RSC connections, i.e. openings between the riser and the steam generators enabling direct flow paths without passing through the pumps, were modeled to enable long term natural circulation. Figure 4 shows long term behavior with introduction of the RSC connections: natural circulation is maintained for the extended period.

Loss of Flow Accident (LOFA)

The first sequence that was discussed and analyzed has been a complete loss of flow accident, and in this frame four cases were analyzed regarding possible neutronic feedback – BOC, MOC, EOC and a limiting with most adverse neutronic coefficients regarding core heat-up, defined in agreement with typical Westinghouse procedures for the analysis of this event. The results showed a typical PWR behavior for system flow rate, neutronic power and thermal heat flux (Figures 5-7). These and other plant data were provided to Westinghouse for input to subchannel analysis code to calculate departure from nucleate boiling ratios in the core. The test was useful from the error detection side and three major errors were discovered: treatment of boron coefficient (due to some difference in Westinghouse procedures and code input definition), inappropriate initialization of core heat structure and a non correct input for the temperature average calculation in the fuel rod.

Also, this case allowed to establish a better interface between FER and Westinghouse in the management of the necessary interfaces.

Feed Line Break (FLB)

FLB test primary intention was the verification of trip logic and analysis of a typical event of decrease in heat removal. However, detailed investigation of the accident sequence led to many sensitivity cases: influence of EHRS loop 1 status on cooling capabilities of EHRS loop 2 was analyzed, limiting point kinetic data were investigated, as well as some assumptions in accident sequence. Sensitivity studies cover full and partial EHRS activation, traveling time for isolation valves, definition of EHRS actuation sequence, choice of signal to trip turbine, trip of the RCP pumps, influence of different scram curves depending on RCP status and usage of the alternative scram on high pressurizer pressure. Regarding neutronic coefficients, BOC/MOC/EOC and limiting cases were considered.

The second goal of this analysis was the definition of the safe shutdown sequence for IRIS: as the results confirmed, the capability of the secondary side to remove heat is reduced rapidly due to termination of the feedwater flow. The water inventory of the faulted SGs is depleted within first 15 s. Initial power removed by all SGs is similar. After decrease of water inventory in faulted SGs other remove some heat due to temperature difference between primary and secondary fluid.

The large primary side heat sink prevents an excessive heat up while the EHRS are actuated on an appropriate signal. Drainage of EHRS lines proved to be most challenging for time step size of the calculation. The differences between cases are mainly due to different nuclear source. Heatup of the coolant in the core decreases fission power so moderator feedback coefficient has major influence in this period on the model response.

An effective removal of decay heat is established when the EHRS initiate to operate in natural circulation. Assumption about RCPs availability has no big influence in the beginning of the transient due to loss of secondary sink, but a loss of offsite is assumed as a consequence of reactor trip and subsequent turbine trip as per NRC guidelines. Analysis of this event is still under revision, and in particular heat transfer modes in the EHRS are being verified to confirm an appropriate RELAP prediction of the system performance in different conditions.

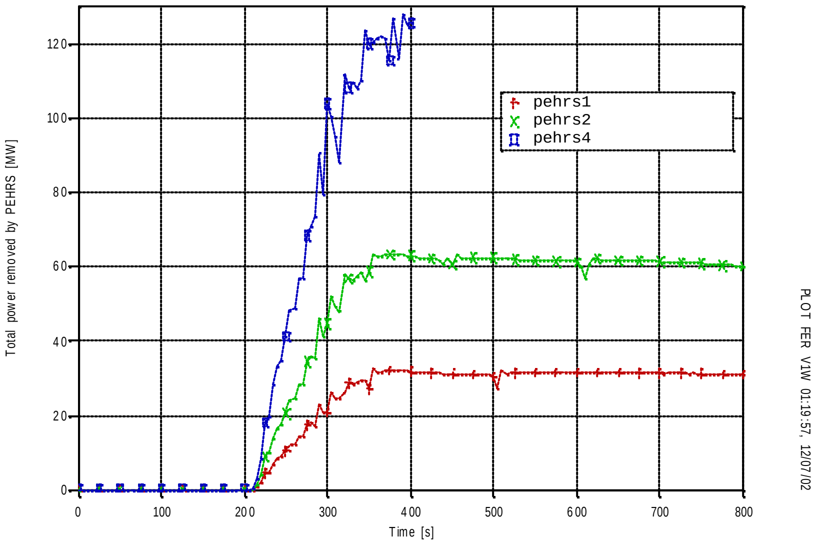

Figure 3. EHRS power – Arbitrary accident sequence

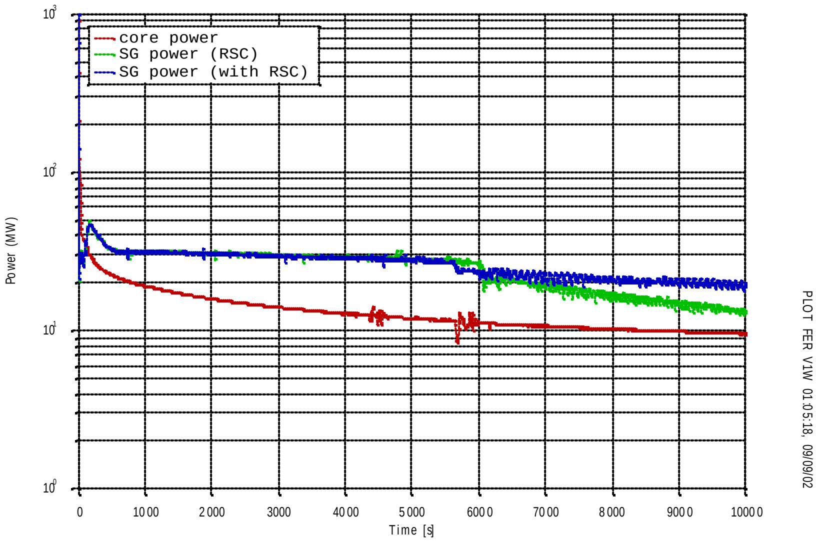

Figure 4. Long term behavior – Transferred power in the core and SGs

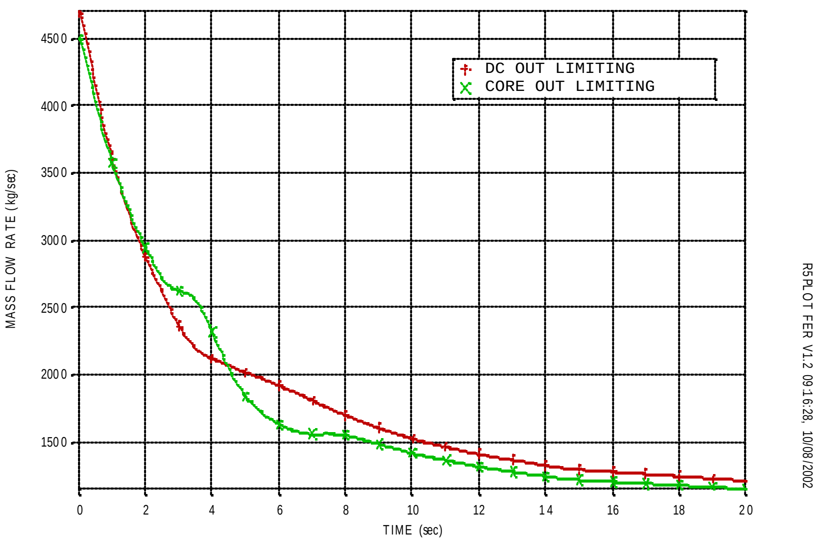

Figure 5. LOFA – RCS mass flows

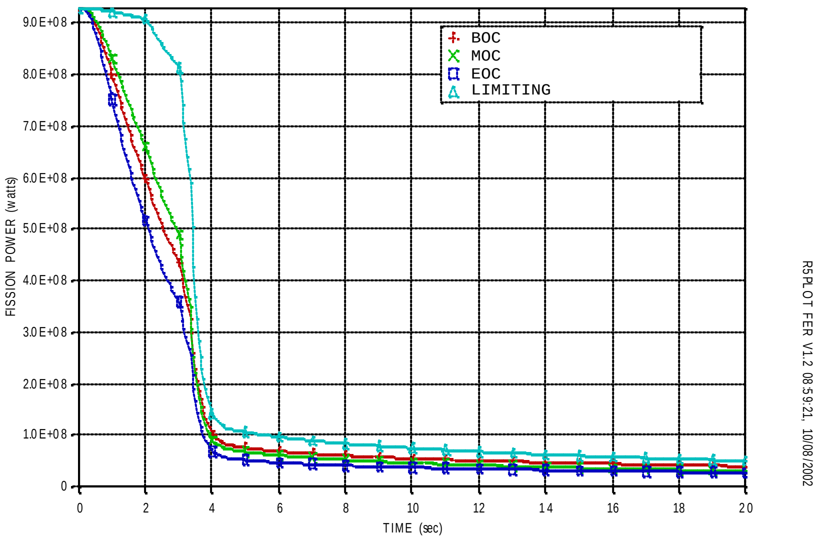

Figure 6. LOFA - Fission power

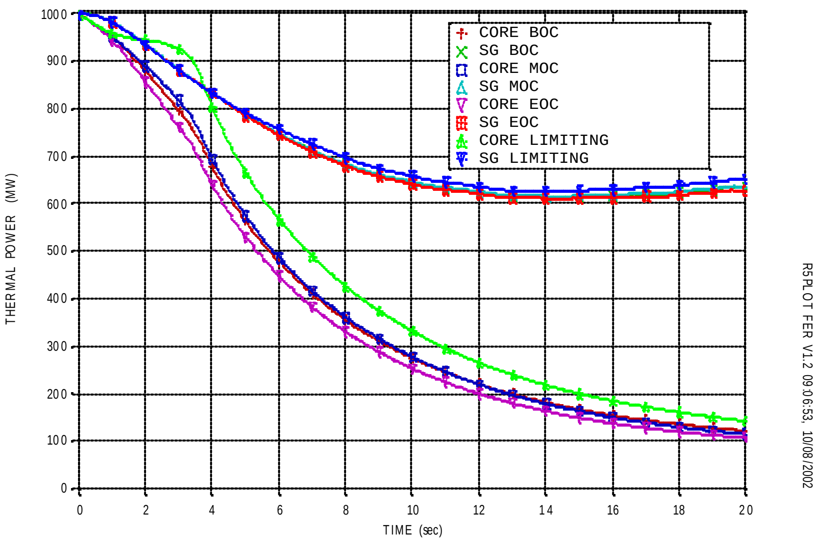

Figure 7. LOFA – Thermal power

5 CONCLUSIONS

IRIS is a PWR, based on existing proven technology, and with significantly improved safety features. An International team consisting of different organizations has developed an analytical model suitable for the transient analysis on the basis of the RELAP5/MOD3.3 computer code. Development of such a model proved to be rather demanding process with necessary and frequent interaction between involved institutions. Nodalization is prepared using best available data and experience in RELAP5 safety analyses.

Preliminary testing of the model has showed that discretization approach is acceptable and that the model produces reasonable results. Some errors were discovered and few improvements implemented. Interfaces between partners were improved. Two accident categories were covered in this phase of testing. Further improvements and corrections are expected after the next phase of testing that will be performed inside IRIS consortium. After this phase of testing, the nodalization will be used for preliminary safety analyses of IRIS system and for determination of control and protection systems setpoints.

Planning and execution of the experimental program, is essential for the further development of the nodalization and crucial model verification and validation. Final goal is to have reliable nodalization applicable to study most of accidents.

ACKNOWLEDGMENTS

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REFERENCES

[1] M. D. Carelli, B. Petrovic, "IRIS – Generation IV Advanced Light Water Reactor for Countries with Small and Medium Electricity Grids", Proc. 4th International Conference of Croatian Nuclear Society, June 16-20, 2002, Dubrovnik, Croatia, Paper IL-2.

[2] C. Lombardi, E. Padovani, A. Cammi, J. M. Collado, R. Santoro, J. Barnes, "Pressure Vessel And Internals of the International Reactor Innovative And Secure (IRIS)," Proc. 10th International Conference on Nuclear Engineering (ICONE-10), April 14-18, 2002, Arlington, VA, USA, Paper ICONE10-22523.

[3] L. Cinotti, M. Bruzzone, N. Meda, G. Corsini, C. V. Lombardi, M. Ricotti, L. E. Conway, "Steam Generator of the International Reactor Innovative And Secure (IRIS)," *Proc. 10th International Conference on Nuclear Engineering (ICONE-10)*, April 14-18, 2002, Arlington, VA, USA, Paper ICONE10-22570

[4] J. M. Kujawski, D. M. Kitch, L. E. Conway, "The IRIS Spool-Type Reactor Coolant Pump," *Proc. 10th International Conference on Nuclear Engineering (ICONE-10)*, April 14-18, 2002, Arlington, VA, USA, Paper ICONE10-22572

[5] M. D. Carelli, L. E. Conway, G. L. Fiorini, C. V. Lombardi, M. E. Ricotti, L. Oriani, F. Berra and N. E. Todreas, "Safety by Design: A New Approach to Accident Management in the IRIS Reactor," IAEA-SR-218-36, IAEA Intl. Seminar on Status and Prospects of Small and Medium Sized Reactors, Cairo, Egypt, May 27-31, 2001.

[6] L. E. Conway, C. V. Lombardi, M. Ricotti and L. Oriani, "Simplified Safety and Containment Systems for the IRIS Reactor," Proc. 9th International Conference on Nuclear Engineering, Nice, France, April 8-12, 2001.

[7] M. D. Carelli, L. E. Conway, G. L. Fiorini, C. V. Lombardi, M. E. Ricotti, L. Oriani, F. Berra and N. E. Todreas, "Safety by Design: A New Approach to Accident Management in the IRIS Reactor," IAEA-SR-218-36, IAEA Intl. Seminar on Status and Prospects of Small and Medium Sized Reactors, Cairo, Egypt, May 27-31, 2001.

[8] L. E. Conway, C. V. Lombardi, M. Ricotti and L. Oriani, "Simplified Safety and Containment Systems for the IRIS Reactor," Proc. 9th International Conference on Nuclear Engineering, Nice, France, April 8-12, 2001.

[9] M. E. Ricotti, A. Cammi, A. Cioncolini, A. Cipollaro, F. Oriolo, C. Lombardi, L. E. Conway, A. C. Barroso, "Preliminary Safety Analysis of the IRIS Reactor," *Proc. 10th International Conference on Nuclear Engineering (ICONE-10)*, April 14-18, 2002, Arlington, VA, USA, Paper ICONE10-22398