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DEVELOPMENT OF RELAP5 NODALIZATION FOR IRIS NON-LOCA TRANSIENT ANALYSES

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ABSTRACT

RELAP5/MOD 3.3 code has been selected as the main computer code for performing transient and accident calculations among members of the IRIS consortium. A rather detailed nodalization of the reference IRIS design has been developed with a basic set of protective functions and controls. The Engineered Safety Features of the concept are being also incorporated in the nodalization. This paper discusses some of the features of the developed model and the problems discovered during initial steady state and transient simulations of the nodalization.

1. INTRODUCTION

State-of-the-art computer codes have achieved sufficient maturity and have been successfully used in reactor technology development for safety assessments in the licensing process. However, the introduction of a new reactor and supporting systems poses great challenges to the development and use of suitable analytical tools for the transient analysis of the concept in question. Also, transient analysis computer codes are now commonly used as a design tool for complex projects as well as a tool for safety assessment and licensing.

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. IRIS is being developed by an international consortium led by Westinghouse/BNFL and includes about 20 organizations from all over the world. IRIS is based on proven LWR technology and employs new engineering to implement attractive and innovative features, but without the need for any new technology development, [1, 2].

One of the main characteristics of the design is the integral reactor vessel configuration that enhances safety performance of the IRIS reactor. The integral vessel houses the reactor core and support structures, core barrel, upper internals, control rod guides and drivelines, a pressurizer located in the upper head and eight helical coil steam generators (SG) coupled with eight low-head spool type primary reactor coolant pumps. The SGs deliver superheated steam to the secondary system. The primary coolant pumps are located on top of each SG and circulate primary coolant through the shell side of each steam generator. The large pressurizer located in the upper head portion of the vessel acommodates pressure surges during power changes and heatup of the system. 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, 3, 4, 5].

The IRIS safety features are presented in a companion paper [6]. The IRIS is based on safety-by-design approach and takes maximum advantage of the opportunities offered by the integral reactor vessel 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

IRIS is 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. In addition to the RELAP5 code, different Westinghouse proprietary codes will be used to address specific phenomena (core subchannel analyses and departure from nucleate boiling evaluations, fuel performance, etc…). Also, CFD tools will be used to evaluate mixing effects for some asymmetrical events.

2. 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 [7]. During the following months, several design details have been finalized and different IRIS partners have also performed studies to define the thermal-hydraulic characteristic of the main systems and components. CNEN has been responsible for the development of the design of integral pressurizer, and the Polytechnic of Milan developed the RELAP model for the steam generator modules and the emergency heat removal system. Westinghouse was responsible for core design (both, thermal-hydraulic and neutronic) and for development of preliminary protection system model. 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. The University of Zagreb has been chosen to prepare and maintain a reference nodalization with accompanying documentation [8] based on the inputs from other institutions.

2.1 RELAP5 Nodalization of the IRIS Reactor

The experience gained with the use of the preliminary model, was used to develop a more detailed nodalization that would provide the proper balance between the required level of detail, the challenges posed by integral concept of the IRIS reactor, and the current project needs regarding licensing and design.

The structure of the nodalization is simple (Figure 1) and it is based on the most updated component designs and operational data. While the overall structure is relatively simple and straightforward, the discretization of the components is rather detailed in order to take into account the important phenomena, resulting in 1718 and 1767 volumes and junctions, respectively. A sliced approach has been used in the discretization of the reactor vessel due to the importance of natural circulation in the chosen safety concept. Most of the calculational nodes have a linear size in the range of 0.2 to 0.5 m. The nodalization was prepared so as to maintain the free volume of the system and elevation differences, as well as core and SG heat exchange areas. The assumption of complete mixing of coolant streams leaving steam generators was used; with the provision that special mixing models will be introduced in the nodalization later, based on CFD calculations of downcomer and lower plenum [9].

The IRIS integral reactor coolant system nodalization is divided in the following main regions:

- Lower downcomer
- Lower plenum
- Core/bypass region
- Riser
- Pressurizer
- Pump Suction Plenum (Upper downcomer)
- Reactor coolant pumps (RCP)
- Primary side of SG modules,
- Inactive volume around SG modules,
- Inactive volume inside SG modules,

Each of the eight RCP/SG modules is explicitly modeled. The original idea was to use a lumping based on a 1-1-2-4 approach that was considered sufficient to take into account different transient and accident sequences. It was however decided to use an explicit modeling in order to better address physical phenomena, take into account interaction of SG modules and the EHRS loops (asymmetry due to different length of feed and steam lines) and preclude possible artificial recirculation in parallel loops introduced due to lumping for numerical reasons. The explicit modeling will also make future multi-dimensional treatment and interaction with CFD-like codes easier.

The pumps are described using preliminary homologous curves in first quadrant. Dummy zero head/zero torque curves are provided for second quadrant. The pump coastdown when power is removed is described by table of pump rotation velocity versus time defined according to preliminary design information.

The balance of plant is only partially modeled, and consist of the following regions:

- Feed Lines from the main feed isolation valves (MFIV) to the SG modules
- Secondary side of the SG
- Steam Lines from the SG modules to the main steam isolation valves (MSIV)
- Main feed and steam isolation valves

Two SGs are connected to each feed/steam line. Only one SG is shown in Figure 1, but the actual SG layout is given in Figure 2 for all modules.

Finally, the Engineered Safeguards Features [6] of the plant are also included. The only engineered safety features currently modeled are the emergency heat removal system (EHRS) and the emergency boration tank (EBT). Also, the Refueling Water Storage Tank (RWST) is modeled as the ultimate heat sink for the EHRS heat exchangers.

These systems are sufficient for the analysis of all IRIS non-LOCA transients and accidents.

Since all the other IRIS safety features (automatic depressurization system (ADS); pressure suppression system (PSS); and the long term core makeup system (LTCMS) establish an interaction between the integral reactor coolant system and the containment, the approach that will be used to model them will depend on the evaluation models used to study LOCA events. Currently, a coupling of RELAP (for the integral reactor coolant system and the secondary side) and GOTHIC (for the containment) is being considered [10]. The model of the interfaces between the two codes (and thus between vessel and containment) depends therefore on this coupling, and is not discussed in this paper where the focus is on non-LOCA analsyses.

In various design basis events, breaks in the primary (SBLOCA, steam generator tube rupture) or secondary system (steam or feed system piping failure) need to be modeled. The same modeling approach will be used to represent all the different breaks, and to provide an example a break for a steam system piping failure accident is provided in the base input deck. In order to model breaks two valve components and one sink volume (that for feed and steam line breaks simply represent a boundary condition) are used. The valves are normally closed unless an appropriate trip signal is generated. For modeling of double ended breaks one additional valve is needed to cut the normal flow path. The same break model can be easily used for different accidents by simply reconnecting the break model, providing additional valve in case of double ended break, and redefining break cross section areas.

Two artificial control systems were used during steady state calculation. First keeps pressurizer pressure at required value and second is responsible for definition of initial water level in the pressurizer. There is no other control system modeled in this nodalization.

Boundary conditions (BCs) used in steady state calculation are steam pressure downstream of the MSIVs and feedwater mass flow rate and temperature upstream of the MFIVs. Four equal, independent groups of boundary conditions were used in the model.

The model initial conditions are based on nominal primary pressure and core inlet and outlet temperatures on primary side, the steam downstream pressure, the feed water inlet temperature, and the steam outlet temperature on secondary side.

Initial flows are given for main flow paths based on best estimate flow values. Minor flow paths are initialized at zero mass flow rates. Taking into account large thermal constants of the system, 1000 s is required to attain steady state conditions.

All the main heat structures are included in the model (the total number of heat structures is 1776, with 9553 mesh points). On the primary side; the core, baffle, barrel steel reflector, axial and radial shields, vessel wall, pump casing and some of the internal plates are all modeled to the best detail allowed by the design information available. For the SG module; the tubes, feed water and steam headers, and the inner and outer shrouds are taken into account. For the steam/feed line piping and EHRS piping; all the pipe walls are modeled. The outer surface of the reactor vessel is assumed to be perfectly insulated. This assumption will be modified in a later phase of the project. The structures are approximately initialized to the average temperature of their bounding hydraulic volumes.

A preliminary version of the reactor protection system based on the RELAP5 trip model was implemented. The model is continuously improved based on analysis of preliminary accident sequences results. Control variables are provided for calculation of: transferred power (core, SG, EHRS), fluid mass in all relevant parts of the nodalization, and for some irreversible pressure losses (current numbers of trips and control variables are 225 and 461, respectively).

The core heat source is based on a power versus time table or point kinetics model. The point kinetics input is preliminary, based on limited available IRIS core design. The kinetics data are calculated for three characteristic burnup cycle points, BOC/MOC/EOC. The total scram reactivity is defined together with corresponding control rod insertion characteristics. Axial power shape is based on chopped cosine with Fnz=1.55, or on calculated axial power shapes for BOC/MOC/EOC. Reactivity weighting uses squared chopped cosine or relevant calculated axial power profiles. The radial power profile in fuel rod is flat. The core decay heat is calculated using the ANSI 79 tables + 2σ uncertainty assuming an infinite operation time, and U235 as the only isotope.

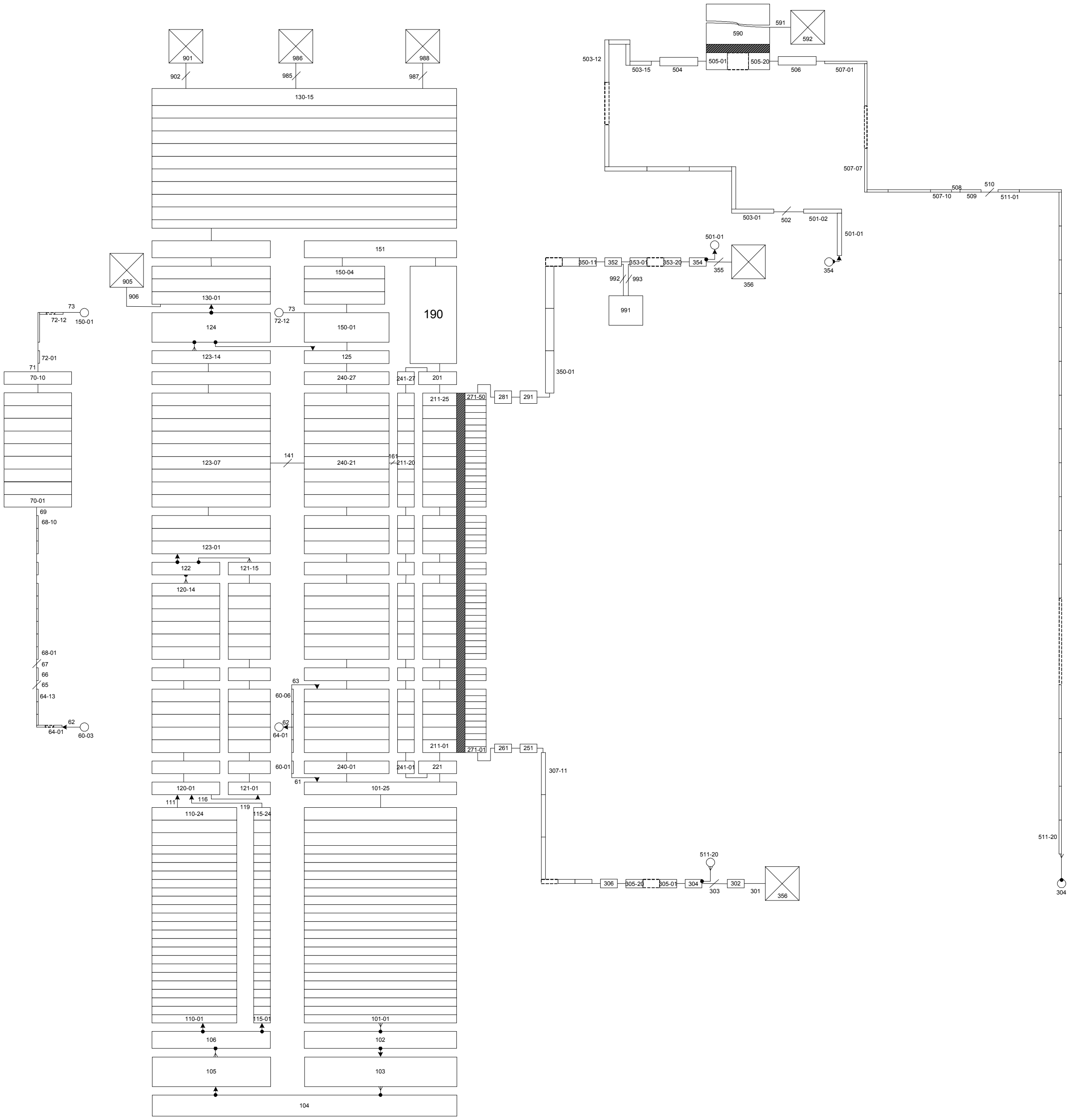

Figure 1. RELAP5/mod3 nodalization of the IRIS reactor

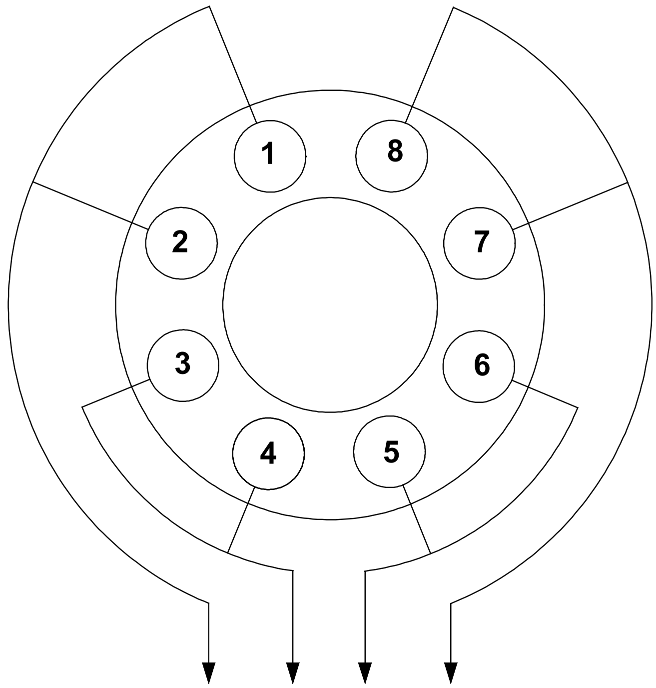

Figure 2. SG modules connection layout

2.2 Steady State Qualification of the IRIS nodalization

A limited steady state qualification of the IRIS RELAP5/MOD3.3 model has been performed. It is usual to compare calculated data to a reference design or to measured data for the steady state qualification, but at this specific stage of the project development, the reference operating full power data are the design values envisaged by the designers. The reference pressure drops were 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.

Usual criteria for quantification of steady state prediction quality have been adopted:

- primary fluid temperature error < 1 ^{o}C
- secondary temperature error < 2 %
- heat structure surface temperature error < 5 %
- pressure error < 1 %
- dp error < 10 % (for main flow paths)
- flow error < 5 % (for main flow paths)
- PRZ level error < 1 %.

The core power is described with control variables that are linked to general tables. This approach allows the flexibility in the power input. For the IRIS to date, the total core power is fixed at 1000 MW. Simple pressurizer pressure and level control are used to reach required primary steady state values. Boundary conditions used on the secondary side are: the

downstream (turbine side) time dependent volume pressure, the upstream (FW side) time dependent volume temperature and pressure, and the inlet FW mass flow rate. Simple initial conditions based on the nominal inlet and outlet core temperatures are applied in the vessel. Within the core, the inactive space in the SGs and vessel annular region more detailed initial conditions from the end of steady state run (1000 s) were used. The upper part of the pressurizer is assumed to be dry saturated steam. Minor flow paths were initialized at zero mass flow. A more detailed initialization could be employed and would include transfer of all mesh point temperatures from the end of a converged steady state run as well as initialization of the volumes using internal energies and local pressures.

Typically a 200 second null transient is sufficient to achieve a stable steady state conditions in most reactor designs. However, in the case of IRIS; the null transient requires up to a 1000 seconds due to the approximate initial conditions and large time constants caused by large water and metal masses within reactor vessel, influence of bypass flow paths associated with SG modules and related inactive space, and currently uncompensated heat losses between pressurizer space and upper downcomer area.

The comparison of the reference IRIS data and the calculated values after 1000 s of RELAP5/mod3.3 steady state run are shown in Table 1., where the agreement of most of the values is acceptable.

The asymmetrical routing of the steam and feed lines discussed in Section 2, causes the presence of four discrete values for most of the secondary variables (SG pressure, steam temperature, transferred power). As can be seen from Figures 3 and 4, this difference in secondary SG pressure and temperature values and transferred power has a small influence on primary side variables in the different SG modules.

The dynamic behavior of the model is satisfactory. Most of the equilibrium values were reached, or their rate of the change was small after first 500 seconds of calculation.

Steady state results are satisfactory for this phase of the project and nodalization has already been successfully used for evaluation of primary pressure drops and influence of steam line pressure drops on the amount of asymmetry on the transferred power for each SG.

Table 1. Comparison between IRIS reference values and calculated steady state data

Parameter	Unit	Reference	Relap5 mod 3.3
Pressurizer pressure	MPa	15.5	15.49
BE vessel flow	kg/s	4707	4697
BE core flow	kg/s	4504	4499
Core inlet temperature	K	565.2	565.25
Core outlet temperature	K	601.5	601.8
SG pressure	MPa	5.8	5.79-5.82
Steam exit temperature	K	590.2	590.2-590.6
Total steam flow	kg/s	502.8	502.8
Dp core	kPa	52.0	53.8
Dp SG1 prim/sec	kPa	72.0/296	71.3/298.4
Total SG power	MW	1001.47	1001.1
RCP head	m	19.1 (18.3-21.3)	19.9

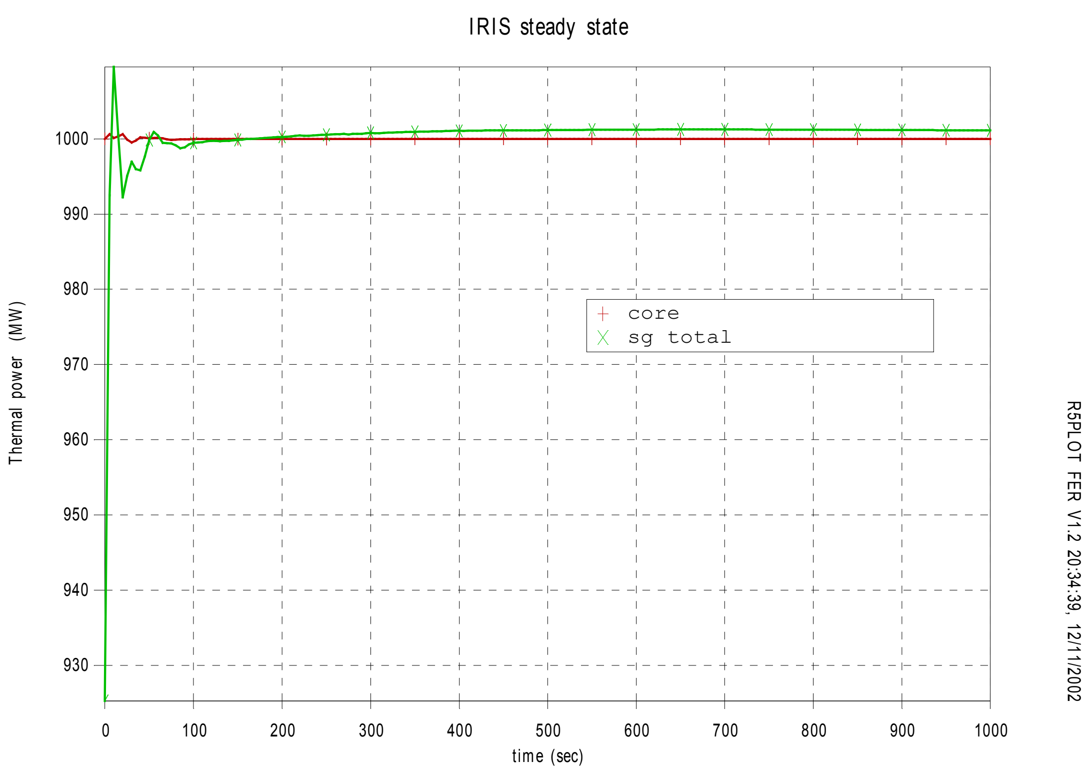

Figure 3. Core and SGs thermal power in steady state run

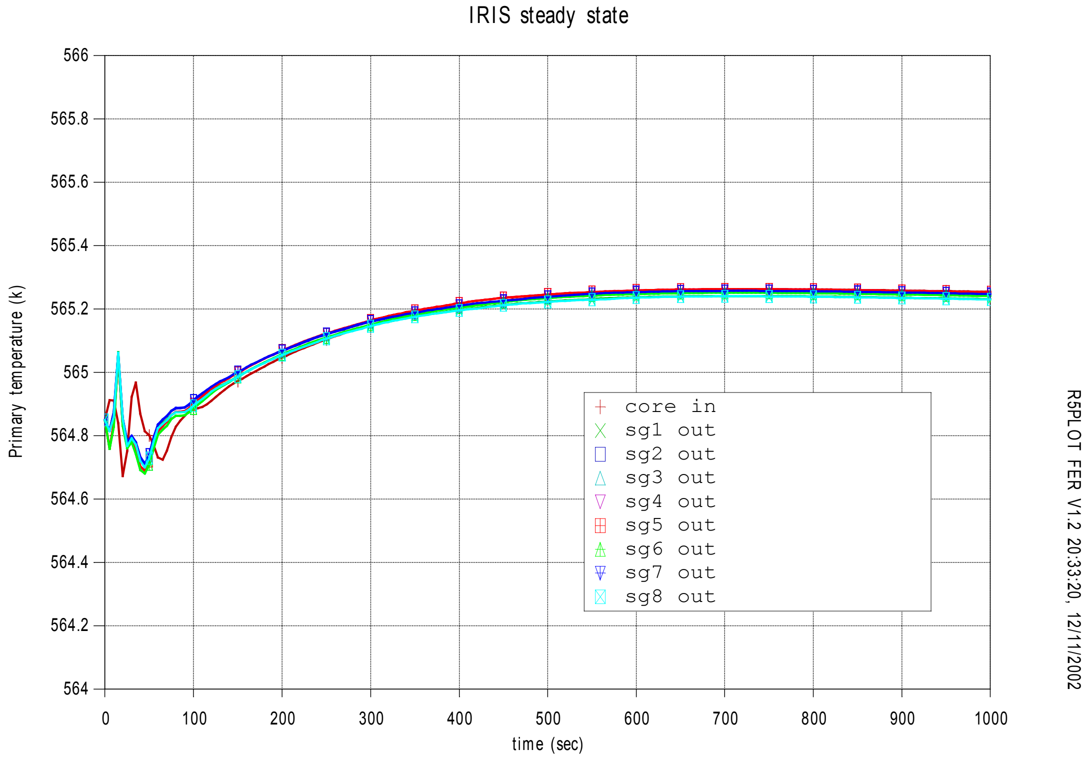

Figure 4. SG outlet and core inlet water temperatures in steady state run

3. LESSONS LEARNED DURING NODALIZATION DEVELOPMENT

The IRIS RELAP5 nodalization was intended to be used to provide a better understanding of IRIS behavior in different transients and accidents, as well as to verify the IRIS response to Safety Analysis Report sequences, and to assist in the initial system design. In order to fulfill these requirements the nodalization has to pass tests of the calculated steady state and transient responses, as well as experimental verifications.

The objective of the preliminary code testing is to detect (together with a review process) errors in the input data and model implementation, to improve the nodalization, to check accident sequences, and to provide examples and prepare guidelines for usage by the "IRIS transient analysis group". It is important to identify possible deficiencies in mathematical models and input data early in the nodalization development.

As a result of steady state and transient code testing, the following problems were discovered and corrected:

- The influence of code and steam table versions on important secondary parameters (steam pressure and temperature, SG thermal power) was assessed. The oscillations in secondary side fluid values are related to the degree of steam superheat. The old RELAP5 versions and old steam tables are too sensitive to the steam superheat at about 50 K of

superheat. The use of RELAP5 mod3.3 and new steam tables was proposed for future calculations, Figure 5.

- The use of cross-flow junctions for connecting the steam and feed lines for two SGs to a common steam and feed line, together with a rather large pressure drop in common part of steam lines; caused a noticable difference in steam generator outlet temperature and pressure, as well as corresponding differences in transferred power and primary temperatures. Therefore, these cross-flow junctions were eliminated. Also, an increase in steam line piping diameters was proposed and accepted by the design team to reduce asymmetrical behaviors.
- Small errors in the temperature value used for initialization of the core heat structures caused errors in the determination of the heat transfer coefficients at steady state. This resulted in an error in the calculation of stored heat at upper core elevations, which had an unrealistic effect on several accident analyses.
- A discontinuity in the post-CHF heat transfer calculation at certain values of steam superheat was discovered in steady state steam generator results, Figure 6.
- A non correct calculation of boron reactivity worth during void formation in the core in some transients was identified and traced back to the wrong implementation of boron feedback in point kinetics equations, Figure 7. The error was corrected in the updated version of the code currently being used in IRIS analyses.
- In the analyses of feed/steam line break events some deficiencies in the current RELAP5 heat transfer model in the special case of condensation with presence of noncondensables was identified. Some code subroutines were not correctly initialized following the detection of non-condensables in some volumes of the system. The error was corrected in the currently used code version

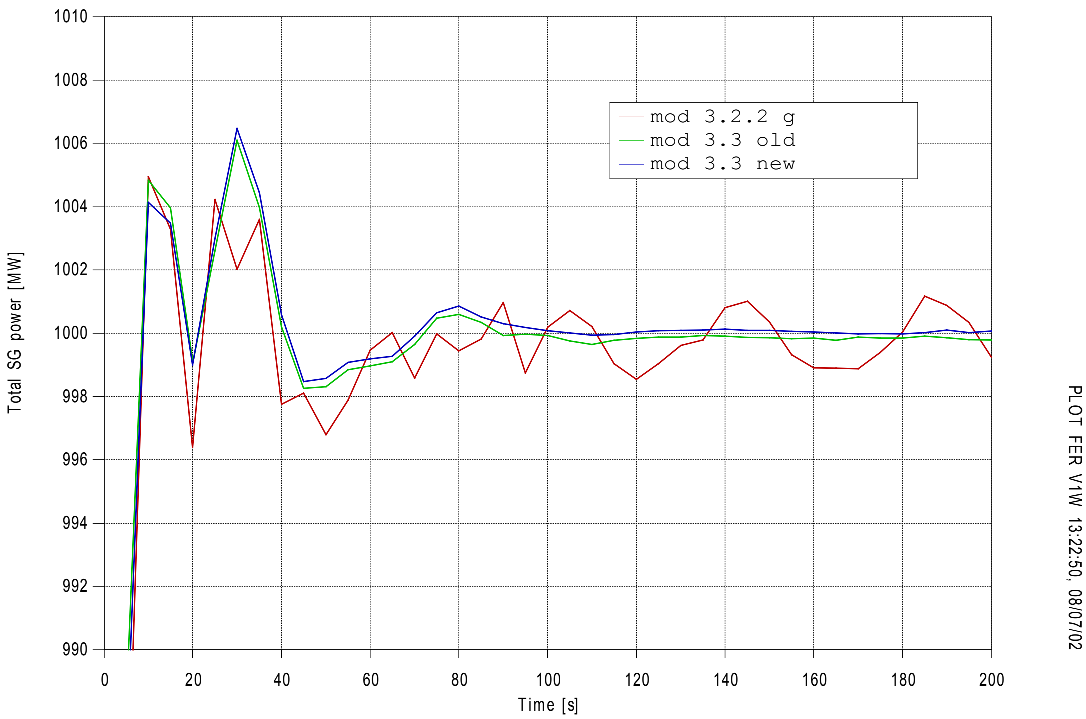

Figure 5. Influence of code and steam table version choice on calculated SG power

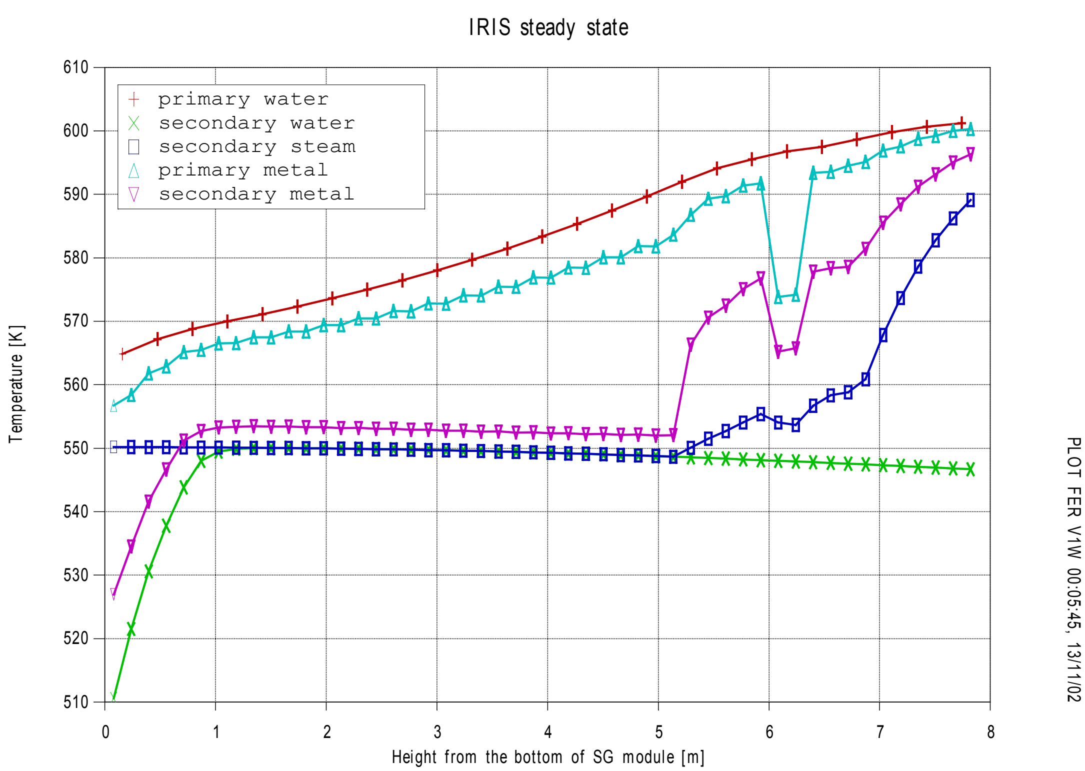

Figure 6. Discontinuity in heat transfer calculation in superheated steam region

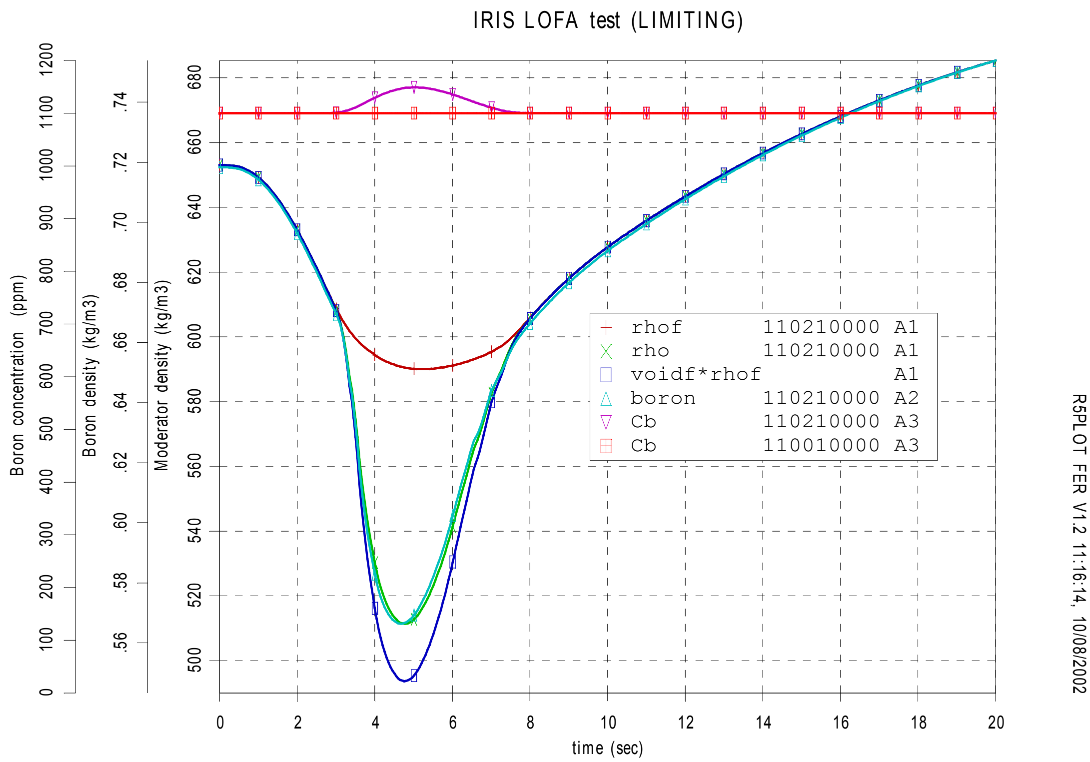

Figure 7. Calculation of boron concentration in case of void presence in RELAP5 volume

4. CONCLUSIONS

An International team consisting of different organizations has developed an analytical model based on the RELAP5/MOD3.3 computer code suitable for the transient analysis of the IRIS. Development of such a model proved to be a demanding process with necessary and frequent interaction between involved institutions.

Preliminary testing of the model has showed that discretization approach is acceptable and that the model produces reasonable steady state and transient results. Some errors were discovered and a few code improvements were implemented. Interfaces between partners were improved. Further improvements and corrections are expected after the next phase of model testing that will be performed by the IRIS consortium. After this phase of testing, the nodalization will be used for preliminary safety analyses of the IRIS design and for verification/determination of control and protection system setpoints.

Planning and execution of the IRIS testing program is essential to obtain experimental data on which to base further developments of the nodalization and for the verification and validation of critical code models. The final goal is to have reliable nodalization and code that is applicable to study of most of the IRIS transient and accident analyses required to license the design.

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