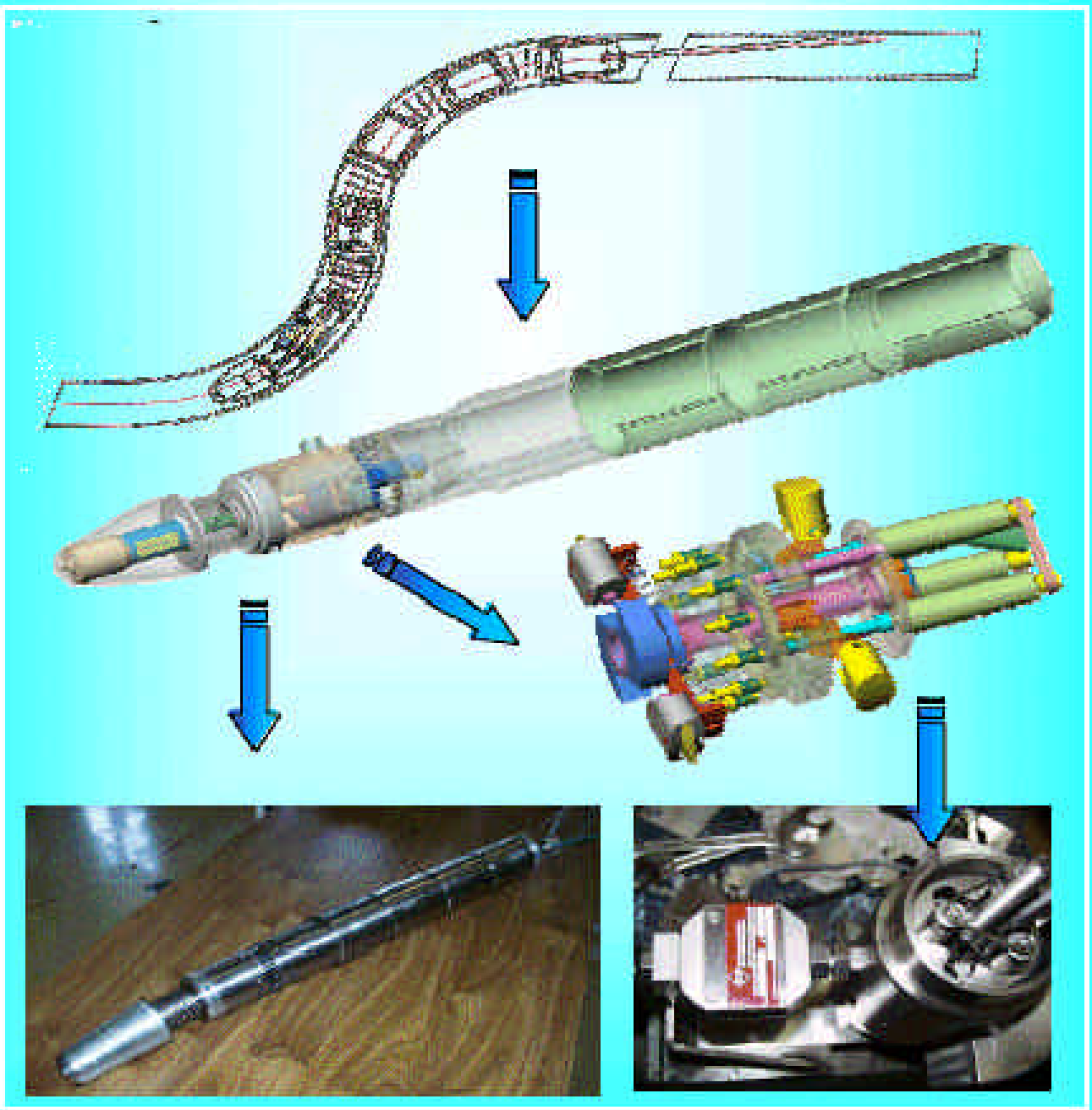
FUSION TECHNOLOGY

Annual Report of the Association EURATOM/CEA 1999

Compiled by : Ph. MAGAUD



ASSOCIATION CEA/EURATOM
DSM/DRFC

SEA 1-14

Task Title : CONTRIBUTION TO SSSR Hydrogen Hazard Assessment

INTRODUCTION

The main purpose of this work is to define a safety criterion concerning hydrogen hazard for each ITER vessel, which presents a potential risk of hydrogen combustion. This safety criterion is a maximal hydrogen amount under which the hydrogen hazard can be considered as negligible.

In this way the principal object is to pass critical judgment on the current safety criterion of 10 kg hydrogen, which is today taken into account to exclude all potential combustion phenomena. This amount had been obtained considering the particular case of an uniform air–hydrogen mixture inside the vacuum vessel (lower volume) with a lower flammability limit equal to 4 per cent.

In this work the potential hydrogen hazard is evaluated as a function of the geometric data and the thermodynamic conditions for the worst reference accident scenario considering a given room. The study is divided into three main parts. First the significant parameters for hydrogen hazard are described. The worst thermodynamic conditions are after that determined for each main ITER vessel. A safety criterion is finally deduced from the precedent data.

1999 ACTIVITIES

SIGNIFICANT PARAMETERS FOR HYDROGEN HAZARD

The deflagration of an air-hydrogen mixture and its main consequence (overpressure) depend strongly on:

- The location of the ignition point,
- The stratification of hydrogen products,
- The possible local over-concentrations,
- The presence of diluent gas,
- The initial thermodynamic conditions,
- The turbulent nature of the atmosphere,
- The presence of steel particles.

To determine the lower flammability limit LFL of an air-hydrogen mixture we follow in this work a conservative approach. The unpleasant parameters like temperature that are symbolized by a negative dependence factor are taken into account in a realistic way. The favorable components like steam presence are neglected (nil contribution).

Table 1 : Dependence Factors for Upward Propagation

PARAMETER	DEPENDENCE FACTOR	CONSERVATIVE APPROACH
Turbulence	$\cong 0$	0
Pressure	$\cong 0$	0
Steam Presence	+ 0.2381	0
Temperature	$T < 640\ °C \Rightarrow -0.000050$	-0.000050
	$T > 640\ °C \Rightarrow -0.000065$	-0.000065

$$T < 640\ °C \Rightarrow LFL_{upward}\ (\text{Turbulence}, \%\ H_2O, P, T) = 0.037 - 5.0 \times 10^{-5}\ (T\ (K) - 373)$$
$$T > 640\ °C \Rightarrow LFL_{upward}\ (\text{Turbulence}, \%\ H_2O, P, T) = 0.050 - 6.5 \times 10^{-5}\ (T\ (K) - 298)$$

METHODOLOGY AND SAFETY CRITERIA

The methodology used to assess the hydrogen hazard in ITER machine is based on a systematic approach, which comprises four main steps. First we select the compartments or vessels, which present together a low volume and a high probability to contain an air-hydrogen mixture during reference accidents. Second we examine all reference accidents, but we only focus on the hydrogen location and the atmosphere temperature.

Third we determine on the basis of these input data the lower flammability limit for upward propagation according to the temperature field. At last we deduce the safety criterion for each taken into account vessel.

A first analyze has led to select four main ITER vessels: the vacuum vessel, the vacuum vessel pressure suppression tank, the cryostat, and the primary heat transfer system.

The **Table 2** shows that the definition of an unique hydrogen safety criterion is not pertinent considering the large diversity of volumes and thermodynamic conditions.

We must also note that each precedent hydrogen safety criterion is based on a well mixed hydrogen-air mixture. This hypothesis does not take into account the probable hydrogen over-concentration phenomena, more particularly nearby the vapor generators in the heat transfer system vaults.

Table 2 : Safety Criteria for Hydrogen Hazard in ITER Machine

LOCATION	WORST SCENARIO	CATEGORY	T (K)	LFL	H2 (kg)
Upper Primary Heat Transfer System Vessel	**Ex-Vessel LOCA (inside the HTS vault) ⇨ In-Vessel LOCA** A direct connection between the torus and the HTS is established. RA : Large Ex-Vessel Coolant Leak	IV	373	3.7	**25.9**
Upper + Lower Primary Heat Transfer System Vessel	**Ex-Vessel LOCA (inside the HTS vault) ⇨ In-Vessel LOCA** A direct connection between the torus and the HTS is established. RA : Large Ex-Vessel Coolant Leak	IV	373	3.7	**52.9**
Vacuum Vessel Pressure Suppression Tank	**Reference Accidental Scenarios which lead to H2 production** VV pressure > 0.2 MPa ⇨ rupture of disks between VV and VVPST. RA : Multiple FW Pipe Break	IV	353	3.8	**1.0-2.4 [1]**
Cryostat Vessel	**Cryostat Air Ingress + Hydrogen Leak** An undetected air ingress is possible because the pumping of cold surfaces maintain the vacuum (small leak). The sublimation of frozen air during a warming up may induce the formation of an air-H2 mixture.	IV	310	4.0	**62.9**
Vacuum Vessel	**Ex-Vessel LOCA (OB/LIM modules) ⇨ In-Vessel LOCA** The passive shutdown is assumed to be accompanied by a disruption that produces an in-vessel failure of the uncooled OB/LIM modules. RA : OB/LIM Ex-Vessel LOCA with Failure of FPS	V	523 [2]	3.0	**5.3**

[1] According to the vacuum vessel pressure suppression tank atmosphere volume, which may range from 370 to 900 m^3
[2] The maximum temperature is about 800 K, but the transient period is relatively short (500 seconds)

MAIN RESULTS

The simplest method to assess the mechanical loads of an hydrogen-air mixture deflagration in a closed room is based on the adiabatic isochoric complete combustion AICC approach. The term AICC defines the assumption used in relating conservation of mass and energy to the combustion process. It must be noted that the peak pressure is only a function of the initial temperature, pressure and chemical composition.

The AICC pressure is simply the absolute pressure that results when all the chemical energy of the hydrogen-air-steam mixture in the containment atmosphere has been combusted.

In the particular case of ITER machine, we must be aware that the adiabatic isochoric complete combustion peak pressure is not necessary a major value.

As a matter of fact many parameters may influence the real value of the peak pressure like the location of the ignition point, the probable stratification of hydrogen products and possible local over-concentrations, the turbulent nature of the atmosphere, the presence of steel or copper particles that present a catalytic effect for the combustion reaction kinetic and may double the overpressure.

CASE STUDY

To quantify the relative influence of each parameter on the peak pressure, we are just taking into account an elementary scenario e.g. an OB/LIM ex-vessel loss of coolant accident with failure of fusion power shutdown. Applying a safety factor of two on the beryllium-steam reaction rate, we obtain a maximum hydrogen amount in the vacuum vessel due to reaction between beryllium and steam equal to 8.4 kg. Adding the 0.16 kg of hydrogen produced from beryllium dust, the total amount of hydrogen in the vacuum vessel is equal to about 8.6 kg.

A deflagration phenomenon in the vacuum vessel may admit two main consequences: a brusque temperature increase of the vacuum vessel atmosphere due to the heat released by the combustion, and a static pressure peak that may lead to a containment integrity loss. The method used to calculate temperature increase and pressure peak in the vacuum vessel after a deflagration phenomenon, is based on three main hypotheses. First hydrogen, air and steam are considered to be perfect gas. Second air and steam occupy the same volume and are temperature balanced. Third the convection loss (heat exchanges with cold structures) are neglected.

As shown in **Table 3** the 2 bar criterion in the vacuum vessel is exceeded in every case studied.

But we must highlight that the initial thermodynamic state before the accident (initial temperature) in the vacuum vessel has little influence on the final thermodynamic state (final pressure). At last we must emphasize that the together presence of steam and steel particles inside the vacuum vessel constitutes a problematic configuration. A conservative approach about the water vapor pressure and the consideration of a probable steel particles presence in the vacuum vessel may lead to a pressure greater than 5 bar (corrective pressure factor due to steel particles equal to 2.0). In particular cases, the vacuum vessel safety criterion of 5 bar could be exceeded.

Table 3 : Case Studies – Final Thermodynamic State after an Hydrogen Combustion

CASE	T_i (K)	P_{H20} (bar)	H_2 %	T_f (K)	P_f (bar)
1A	393	×	3.6	923	2.30
1B	393	0.02	3.5	918	2.41
2A	473	×	3.6	1000	2.07
2B	473	0.05	3.5	980	2.21

CONCLUSIONS

Further to the present study we can remark that the hydrogen hazard in ITER-FDR installation can not be approached by an unique safety criterion. Only a case by case approach may integrate the large diversity of volumes, and of thermodynamic conditions. To conclude it seems that four axes must be favored in the future:

- the precise definition of a settled design for the ITER installation,
- the exact determination of hydrogen products/location for each reference scenarios,
- the characterization of stratification phenomena (TONUS) and particles effects,
- the elaboration of a really case by case approach for the hydrogen hazard in ITER.

REPORTS AND PUBLICATIONS

[1]. Determination of safety criteria for the hydrogen combustion hazard in ITER machine - E. Bachellerie, F. Arnould , TA-127886/A

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SEA 4-4

Task Title : PLANT SAFETY ASSESSMENT Code PAXITR creating a data set

INTRODUCTION

PAXITR is a thermohydraulic computer code for spot calculations, designed to calculate the pressure rise in containment in conjunction with computation of depressurization of one or several pressurized vessels in this containment.

This program is an adapted version of the PAX code derived from the combination of two computer codes:

EXPRESS examines only the "depressurization" phase of the pressurized vessels and provides the mass flow and enthalpy flow rates required for the other code,

PAC calculates the evolution of pressure and temperature in the containment, based on the transient information provided by EXPRESS.

The PAXITR computer code (like EXPRESS and PAC) is written in Fortran. It uses most of the EXPRESS subroutines. However, the part involving the containment pressure taken from PAC has been completely modified.

1999 ACTIVITIES

The 1999 activities consisted of making a new version of the PAX code dedicated to fusion studies (called PAXITR), and making the input manual in reference [1].

The main evolutions of the code are the following:

- New Heat transfer coefficient correlation for the condensation without uncondensable gas (Nusselt Law),
- New Heat transfer coefficient correlation for the Pool boiling on the wet surfaces of the vessel,
- Extraction of the correlations dedicated to nuclear propulsion (confidential Technicatome correlations).

CONCLUSION

The PAXITR code is available for the fusion studies since july 1999.

REFERENCES

[1] PAXITR CODE CREATING A DATA SET
TA 126154 Ind A

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SEA 5-2

Task Title : COHERENT SYSTEM OF CODES FOR THE ITER SAFETY ANALYSIS (SEA 3-1) VALIDATION OF COMPUTER CODES AND MODELS (SEA 5-2)

INTRODUCTION

The main goal of 1999 has been to finalize the moving from Isas-1 to Isas-2, with updating of the slave codes and of the input decks already used for coupling cases.

The developments of Isas are now controlled by a quality assurance way.

1999 ACTIVITIES

SUPPORT TO THE ISAS USERS

Introducing of the new ISAS2.0 kernel

The moving from Isas1 to Isas2 has been done by a full rewriting of the Isas kernel in C language, and introducing the concept of an Isas component, where a component is a code written in FORTRAN, C, C++, or OCAML language, which communicates with Isas across an API interface.

An Isas component can be structured as an OCAML object, where a class defines a calculation server, and associated methods define the available functions of this server.

The Gibiane command language has been also replaced by OCAML, an object oriented language developed by INRIA.

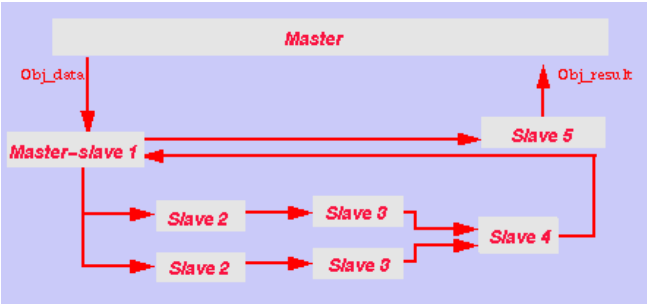
The user can choose between two ways for working with Isas2 :

- Keeping unchanged the old slave codes, by using the compatible version of Isas-2. In this case, he just needs to translate from Gibiane to OCAML the input decks, and not any recompilation of codes is needed.

- Acceding to the full Isas functionality's, by introducing slight changes in the communication functions and recompile the slave codes with the new Isas libraries.

The most interesting new features of Isas-2 are :

- direct communication between slave codes and delegation of tasks :



Potential way for a communication network between slave codes

- full encapsulation of PVM routines and better control of the virtual machine built,

- introduction of a trace facility,

- powerful functional abilities,

- introduction of plotting,

- class and objects mechanisms.

Definition of each Fusion code as an object with associated methods

This task consists to define each Fusion code as a reusable Isas component, put differently to create an OCAML object with associated methods.
The methods have been mainly defined to get :

- a time control on the code,

- a data exchange with the code by formatting variables and lists of variables concerned,

- some utilities as partial printing, calculation functions.

Methods have been validated for Athena, Intra, Naua and Safaly, and in all cases, the data exchanges potentialities are the same than for Isas-1 versions.

Moving input decks and validation

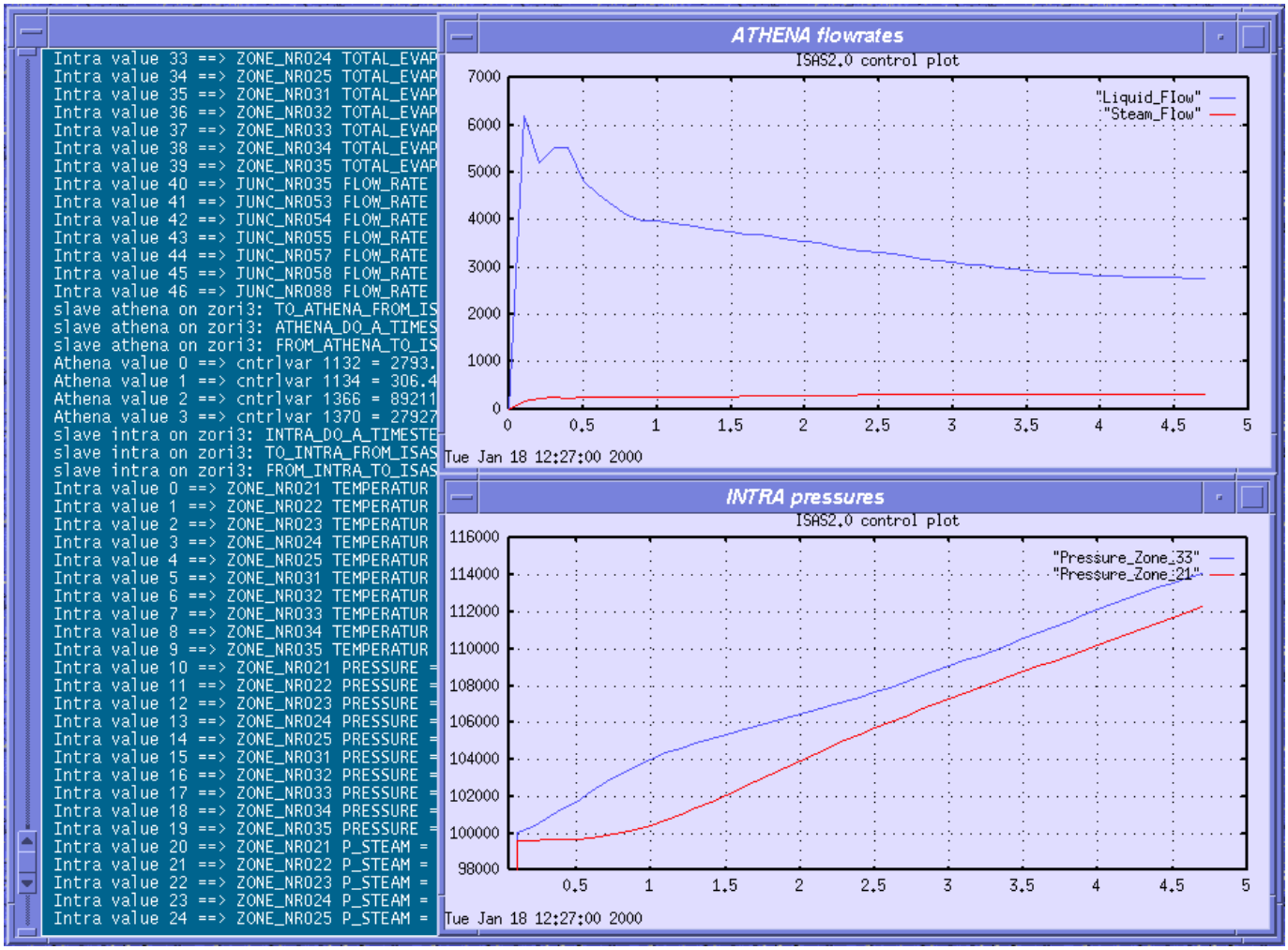
The goal of this task has been to write the news input decks for coupling runs in OCAML language to have a comparison with the existent coupling scenarios.

A strong improvement of the concision and of the robustness has been obtained for input decks, mainly by introducing functions for intermediate calculations.

Plotting

An on line connection with gnuplot, a free software chosen for his potentialities in real time working, is now available

The user now can get graphical outputs either during a run, or after a run.



Example of a graphic control for a coupling run with Athena and Intra

REPORT AND PUBICATIONS

Two reports have been provided :

The first report [1] describes the Isas2.0 kernel functionality's and the use of the interface functions to integrate a slave code in Isas.

The second report [2] is a condensed overview of Isas2.0 presentation and applications used as a technical support for a training organized on January 20,21 at Garching.

[1] Updating of the Isas user's guide : version 2.0
SYSCO/DIR/RT/99-050/A

[2] Isas 2.0 overview
SYSCO/DIR/RT/00-002/A

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SEA 5-31

Task Title : VALIDATION OF COMPUTER CODES AND MODELS Thermalhydraulic codes validation and benchmark

INTRODUCTION

In the frame of fusion task SEA5.3, CEA must carry out a set of experimental tests to validate accident simulation codes used in fusion safety assessments. In this respect, the EVITA experimental program devoted to condensation phenomena on cold surfaces, has been set up. The experimental conditions are now well defined and these data will be used as references for pre-test calculations with the codes to be validated. The first tests are scheduled to be performed by the beginning of year 2000.

Also in the frame of this task it was decided to perform a benchmark exercise for the computer codes used in the safety analysis of fusion accidents. The definition of the 7 cases for the tests and the calculations by the 5 involved associations were performed during 1999.

1999 ACTIVITIES

DEFINITION OF THE EVITA TEST PROGRAM

During the 2000 test program, the tests objectives will be to first obtain the operating conditions and then to perform the condensation experiments. In a second step, different operating conditions and injection of other components (air, helium) will be scheduled.

The nominal conditions for the cryogenic test will be **:**

- initial vacuum : 10^{-3} mbar
- inlet steam temperature : 165°C
- inlet steam flow rate : 0.0015 kg/s
- cryogenic surface temperature : 80°K

Then different vapor flow rates will be tested. For each of the tested vapor flow rates, the nitrogen flow will be adjusted to produce three experimental cases: (1) complete absorption of the injected vapor energy and (2) incomplete absorption of the vapor energy at two separate nitrogen flow rates less than the value in case (1). In case (2), the vessel will pressurize at a rate determined by the nitrogen flow rates used. The nitrogen flow rates in case (2) might be factors of 1/2, and ~ 0 of the nominal nitrogen flow rate used in case (1). This last situation has particular significance because it is the one most similar to the real-life situation (the nitrogen cooling capacity being low in comparison to the energy of the injected vapor).

A preliminary test matrix is presented in table I, below.

Table I : Preliminary test matrix for cryogenic experiments

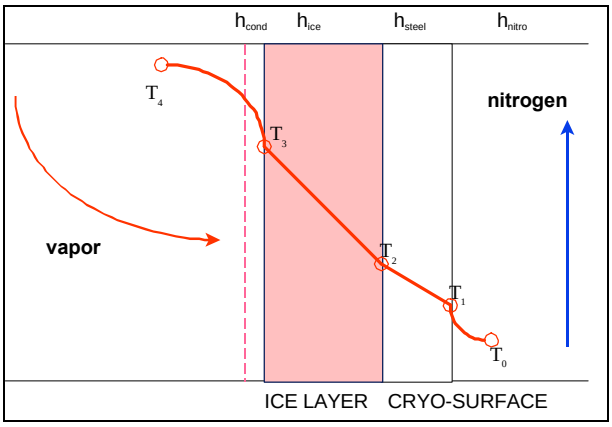
vapor temperature : 165°C		**NOMINAL**			
vapor flow rate (kg/s)	0.0007	**0.0015**	0.003	0.005	0.007
input power (KW)	2	**4.2**	8	15	20
nitrogen	2	**4.2**	8	15	20
power removal	1.3	**2.8**	5.3	9.9	13.2
capacity (KW)	≈ 0	**» 0**	≈ 0	≈ 0	≈ 0

COMPUTER MODELING OF THE EVITA EXPERIMENTS

A computer program, EVITA-30, has been written to model the thermal-hydraulic responses of the EVITA experiment.

This FORTRAN program predicts (1) growth of the ice layer on the condenser surface, (2) surface temperature of the developing ice layer, and (3) pressure response of the vacuum chamber. EVITA-30 modeling capability includes the following scenarios: (a) rapid pressurization of the vacuum chamber when the injected vapor power exceeds the liquid nitrogen power, and (b) steady state conditions after slight pressurization when injected power equals nitrogen power.

These modeling results are used to suggest the optimum regime of study for the EVITA experiments which are to be performed at the DER/STPI facility.

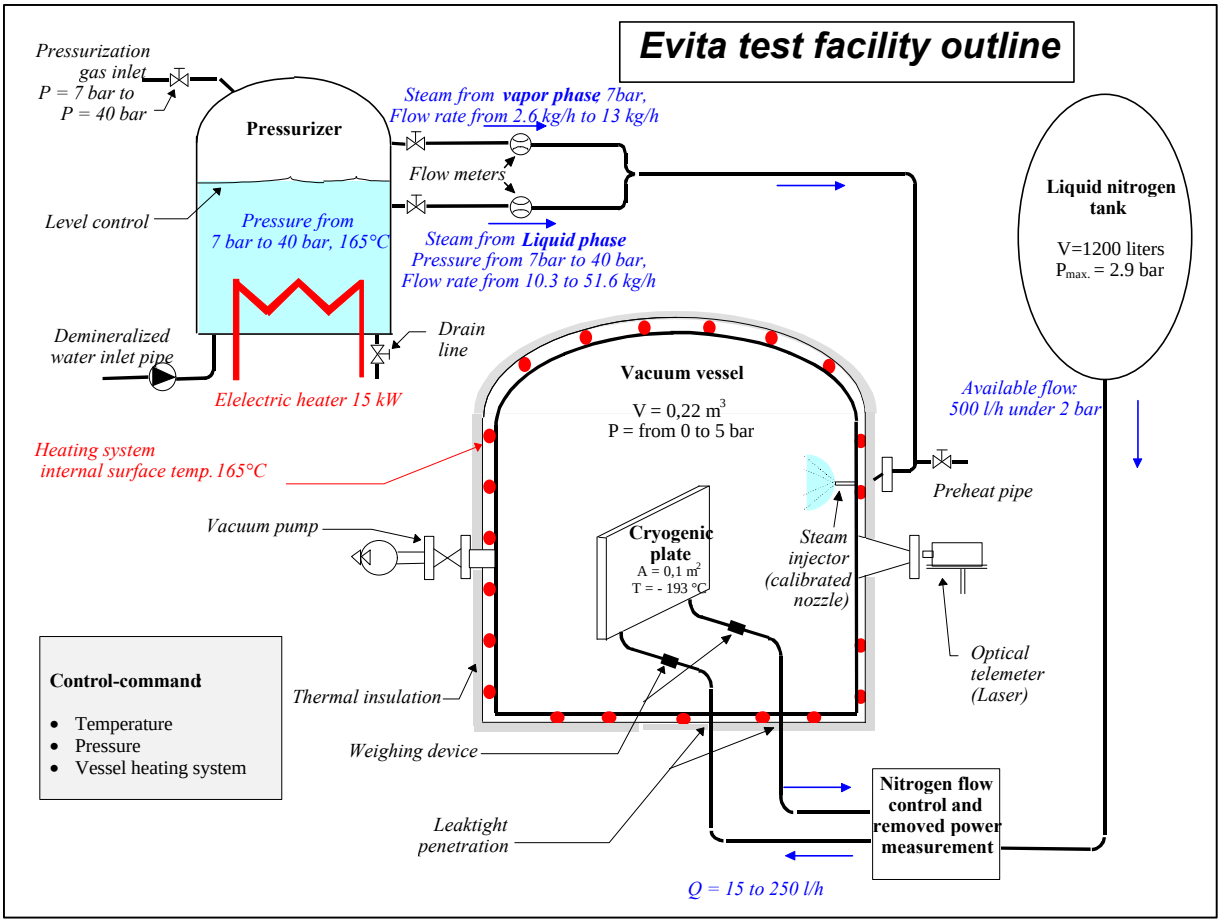


EVITA : Schematic of analytical model

MANUFACTURING AND PROCUREMENT OF THE FACILITY

In 1999 the definition and the manufacturing of the different components of the facility have been performed. The procurement and the assembly were done during the last days of 1999 and only the cryogenic tank was to be procured and installed in the very beginning of 2000.

The figure hereafter show the outline of the facility which include : a pressurizer to provide hot subcooled water or saturated steam, the vacuum vessel itself where experiments will be conducted and the cryogenic system which allows to circulate nitrogen in the condensation plate located inside the vacuum vessel.



BENCHMARK OF COMPUTER CODES USED IN FUSION

In the frame of the EU task N° SEA5, « Validation of computer codes and models » it has been decided to perform a «code benchmark » related to phenomena that were not covered by the previous JCT defined benchmark.

These phenomena are essentially :

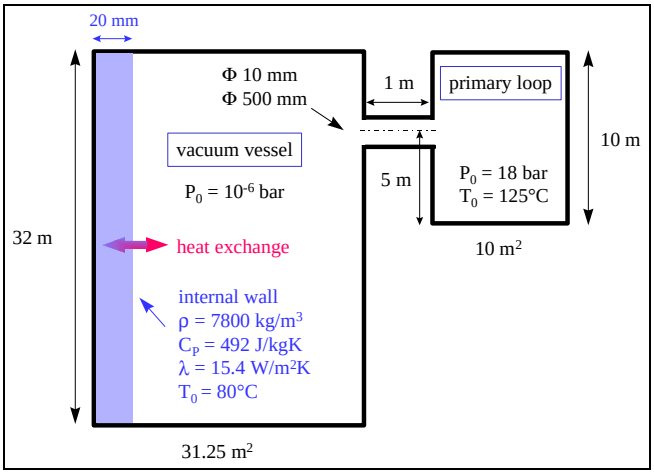
- the break critical flow,
- the influence of downstream (vessel) pressure on the break flow,
- the coupling of a discharge tank.

The codes involved in this benchmark are listed in the table II , below.

Table II : The different codes participating at the benchmark assessment

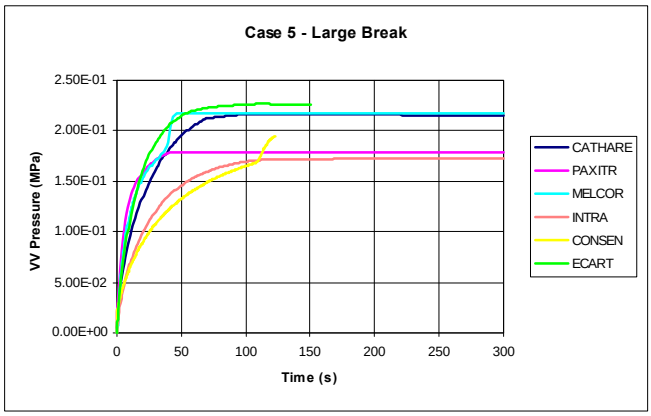
code	organisation
MAGS	FZK - Karlsruhe
MELCOR	JCT - Naka
INTRA	NFR - Studsvik
CONSEN	ENEA
ATHENA + INTRA (with ISAS)	ENEA
PAX	CEA - Technicatome
CATHARE + CONTAIN (with ISAS)	CEA

Seven different test cases were defined and, as an example, the figure below shows the conditions defined for benchmark test case # 5.



Benchmark definition : test case #6

In 1999, the results have been compiled and plotted together, see figure hereafter and showed that some more analysis had to be performed in order to issue recommendations for code users and code assessment.



Benchmark results : test case #5

CONCLUSIONS

The validation and verification of computer codes and models is a very important task in the process of the safety assessment of a nuclear facility.

This task covers the main topics of such process : experimental works and codes benchmarking. The analysis of code results and discrepancies will be performed in 2000 and will allow to better understand the codes accuracy's and limitations. The experimental results that will be obtained on the EVITA facility should also help to better understand codes results as well as phenomenological behavior of fusion plant components under accidental conditions.

REPORTS AND PUBLICATIONS

[1] « EVITA kick-off meeting Garching - June, 1999» C. Girard - CRR DER/SERSI/LEFS 99/50086

[2] « EVITA review meeting - Nov., 1999» C. Girard - CRR DER/SERSI/LEFS 99/50150

[3] « EVITA -1999 test program» C. Girard, T.D. Marshall - NT DER/SERSI/LEFS 99/5005

[4] « Etudes de dimensionnement d'EVITA pour les essais cryogéniques» C. Girard, D. Chiarradia - NT DER/SERSI/LEFS 99/5011

[5] « EVITA-30 : Computer Modeling of the EVITA experiment», T.D. Marshall, C. Girard - NT DER/SERSI/LEFS 99/5031

[6] « Cahier des spécifications techniques d'une enceinte d'essais et d'une plaque cryogénique - Etude et réalisation», Tétard, Baud, Chiaradia - NT DER/STPI/LCFI 99/016

[7] « Cahier des spécifications techniques pour un générateur de vapeur - Etude et réalisation», D. Chiaradia - NT DER/STPI/LCFI 99/030

[8] « Benchmark kick-off meeting Garching - June, 1999» C. Girard - CRR DER/SERSI/LEFS 99/50087

[9] « Fusion code benchmark meeting - Nov., 1999» C. Girard - CRR DER/SERSI/LEFS 99/50149

[10] «Definition and requirements of the fusion code benchmark » P. Sardain - Letter DER/SERSI/LECC 99/40148

[11] « Fusion code benchmark : results and analysis of Cathare and contain calculations » G. Mignot - NT DER/SERSI/LECC 99/4059

[12] « Fusion code benchmark 1999 - Synthesis of the results» P. Sardain- NT DER/SERSI/LECC 99/4067

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SEA 5-32

Task Title : VALIDATION OF COMPUTER CODES AND MODELS ITER benchmark reference case calculations of in vessel LOCA's with the PAX code

INTRODUCTION

In the frame of validation and verification of the codes used for ITER safety demonstration, a benchmark is under progress between the different parties of the project in order to compare calculation tests achieved by different pressurisation codes.

The aim of this paper is to present the results of the ITER reference case calculations that have been done by TECHNICATOME in the frame of the ITER safety studies programme. These calculations have been obtained with the code PAX.

These different cases are representative of Loss of Coolant Accidents inside the vacuum vessel (in-vessel LOCA's).

The main goal of the present work is to evaluate the critical flow at the break, the effects of downstream pressure on the break and the different effects on the modelling of a discharge tank.

These tests results will be presented during the CEA meeting in November 99.

These studies have been conducted under ISO 9 001 Quality Assurance procedures.

1999 ACTIVITIES

The objectives of the Benchmark are to investigate the pressurisation rate, the critical flow rate, and the counter pressure effect.
The PAX code is a safety studies code used mainly to evaluate both :

- the depressurisation of a containment (cooling loop for example) and,
- the resulting pressurisation of a vessel receiving the fluid from the depressurisation.

Safety Authorities acknowledge this code as a tool for pressurisation calculation and safety reports support.

The main parameters that can be taken into account are :

- for the containment (here the boiler) :
  * thermodynamic conditions,
  * thermal inertia of the containment structures,
  * pressure losses in the piping to the break.

The flow at the break can be evaluated through a correlation developed under a specific experimental programme.

- for the vessel :
  * initial thermodynamic conditions,
  * temperature profile in the structures (made of different material slices),
  * rupture disks (or scrubbers) and expansion containment.

The main results that can be provided are :

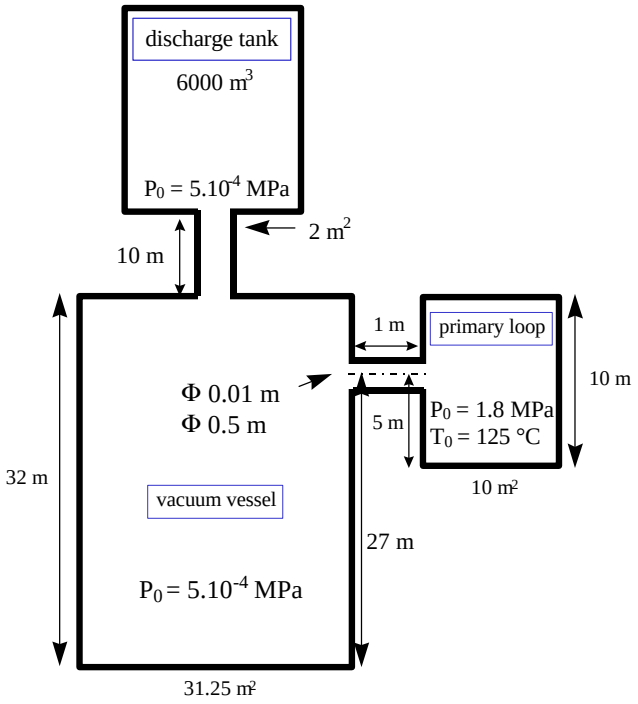
- for the containment :
  * thermodynamic conditions,
  * temperature of the containment structures,
  * flow at the break.

TESTS PARAMETERS

The different cases of the ITER benchmark are the following :

- Case 1 : Reference (input flow and enthalpy imposed : 10 kg/s- 2700 kJ/kg),
- Case 2 : critical flow (liquid) for small and large break,
- Case 3 : critical flow (vapour) for small and large break,
- Case 4 : calculation of the upstream transient for small and large break,
- Case 5 : calculation of the upstream transient and downstream pressure effect for small and large break,
- Case 6 : calculation of the upstream transient and downstream pressure effect with condensation,
- Case 7 : calculation of the upstream and downstream transient with discharge tank modelling.

A schematic modelisation used for the case n°7 (most complete test) of the benchmark is presented on figure 1. Pressurized water is flowing through a small nozzle (water injector Φ=0.01m and Φ=0.5m) into a vacuum vessel. An isolation valve may allow the connection of the vessel to a blow-down tank in case of overpressurisation of the vessel (the valve is set to open at 1 bar).



CONCLUSION

The benchmark test results showed strong discrepancies between the different codes especially on the estimation of the critical mass flow rate.

This was not the case during the last ICE experiment, because the mass flow rate was given as a data.

Several working groups were composed during the meeting in order to analyse the different physical models, this work is actually under progress.

REFERENCES

[1] ITER BENCHMARK REFERENCE CASE CALCULATIONS OF IN VESSEL LOCA 'S WTH THE PAX CODE TA130938

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S2.3

Task Title : ALTERNATIVE CONCEPT PLANT MODELS Response of Li-Pb self cooled (TAURO concept) and AHCPB models to loss of helium accidents

INTRODUCTION

In this task, the response of two fusion reactor models to loss of helium accidents is studied. The concerned models are :

- the Lead-Lithium self cooled model, also called ''TAURO'' concept [1],
- the Advanced Helium Coolant Pebble Bed model (AHCPB model) [2].

The Lead-Lithium self cooled model is a fusion reactor concept based on liquid Lithium-Lead blankets with SiC-SiC material structures. Lead-Lithium is cooled in a Li-Pb/Helium heat exchanger.

The AHCPB model consists of Li_4SiO_4 or Li_2TiO_3 solid pebbles blankets cooled by helium circulating in SiC-SiC tubes.

The two fusion reactor concepts are based on direct gas cycle for power conversion.

For the ''TAURO'' concept, the scenario analyzed is a rupture of a pipe of the secondary helium circuit.

For the AHCPB model, a rupture of a pipe of the primary helium circuit is considered.

In order to evaluate the pressure evolution in the containment building during the two scenario, computations with the CONTAIN code [3] are performed. The results of computations are analyzed in particular to verify that containment can sustain pressurization.

More, for the ''TAURO'' concept, a scenario based on a phenomenological approach is proposed to try to describe consequences of high pressure helium ingress in the Lithium-Lead flow in the case of a rupture of heat exchanger tubes.

1999 ACTIVITIES

CONTAIN COMPUTATIONS

For Tauro concept, results show that the time to obtain pressure balance in the helium circuit and the containment volume is very short (4s < t <10s).

The values of the final pressure in the containment reached at the end of the transient vary between 1.9 bar and 2.4 bar depending on the helium circuit capacity. The final pressure in the 3 cases is widely over the design pressure (P_{design} =1.4 bar). For the reference volume (V = 2250 m^3), the final pressure is 2.18 bar.

The maximum break mass flow rates are respectively 2450 kg/s and 5200 kg/s for break sizes D = 0,96 m and D = 1,44 m.

For the break diameter D = 1,2 m, the maximum break mass flowrate is about 3800 kg/s depending on the volume of the helium circuit.

The minimum volume for containment volume which would lead to a maximum pressure of 1.4 bar (design pressure) was calculated and we found that 300000 m^3 would satisfy the pressure condition.

For AHCPB model, the final pressure in the containment remains widely over the design pressure. The values of the final pressure in the containment reached at the end of the transient vary between 2.3 bar and 2.98 bar depending on the helium circuit capacity. For the reference volume (V=2600 m3), the final pressure is 2.61 bar.

The maximum break mass flow rate are respectively 3050 kg/s and 6300 kg/s for break sizes D = 0,96 m and D = 1,44 m. For the break diameter D = 1,2 m, the maximum break mass flowrate is about 4600 kg/s depending on the volume of the helium circuit.

In order to evaluate the influence of the initial temperature of helium in the circuit on the final pressure, computations were performed with different initial temperature values : 300°C, 400°C and 800°C. For the 3 computations, the volume of helium circuit is 2600 m^3, the initial helium pressure is 80 bar and the break diameter is 1.2 m. The computations results give :

T=300°C	P_{final} = 2.76 bar
T=400°C	P_{final} = 2.72 bar
T=800°C	P_{final} = 2.52 bar

The results show that the final pressure decreases when the initial helium temperature increases. This is due to a lower initial quantity of helium in the circuit. In fact, when the initial temperature is higher, the initial mass of helium is lower and leads to a lower final pressure. It is noted that for important variations of helium initial temperature, the variation of final pressure is low.

For AHCPB model, a value of 400000 m^3 was found for the containment capacity to avoid a containment pressurization above 1.4 bar.

HELIUM INGRESS IN Li-Pb (TAURO MODEL)

If a pipe rupture occurs in the Li-Pb/helium heat exchanger, consequences on the primary circuit may be more important. In the case of a break in the heat exchanger, it can be assumed that helium ingress inside the liquid metal flow. The pressure difference between helium and Li-Pb is very important and about 65 bar in the TAURO concept. When helium will be in contact with Li-Pb, the pressure of liquid metal will increase rapidly to a pressure close to the helium pressure. To avoid Li-Pb pressurization and to protect blankets, a safety expansion vessel filled with neutral gas (no chemical reaction) and connected to the Pb-Li circuit is needed. The capacity of this vessel is about 900 m3 from the reference [1].

In fact, when high pressure gas ingress in a low pressure liquid, the pressure of gas decreases very rapidly and leads to the creation of pressure waves [5]. The propagation of these pressure waves depend on different parameters, in particular on the geometry of the liquid capacity (vessel, pipe, container ...). Reflection of pressure waves on walls of the liquid capacity may occur and causes an increase of the local pressure. The intensity of pressure waves can be very important and causes serious damages on structures.

Phenomena involved in the problems of high pressure gas ingress in liquid are very complex. A specific study should be carried out on the TAURO model to evaluate with precision consequences of helium ingress in Li-Pb. This study would need a detailed description of the Li-Pb cooling pipes in order to model shock waves propagation and access what could be the consequences on structural integrity of these pipes.

For others kind of pipe rupture everywhere in the secondary circuit (except in the heat exchanger), consequences on the primary circuit are only an increasing of temperature of Li-Pb. Because the SiC-SiC material structures enables operating in very high temperatures (T>1300 °C), no other safety system except the expansion vessel is needed.

CONCLUSIONS

In this task, the response of two fusion reactor models to loss of helium accidents is studied. The concerned models are :

- the Lead-Lithium self cooled model, also called ''TAURO'' concept ,
- the Advanced Helium Coolant Pebble Bed model (AHCPB model).

The particularity of these two concept is that the blankets are based on SiC-SiC material structures.

In order to evaluate the pressure evolution in the containment building during transient, parametric computations with the CONTAIN code were performed. The results of computations show that for the both models the design pressure of containment is exceeded (maximum pressure ≈ 2.5 bar).

The analysis of the pressure transient for both TAURO and AHCPB concepts show that the value of the final pressure is mainly related to the initial pressure ; it depends also on the helium temperature but the influence is lower than the initial pressure. It was found that if the initial helium pressure was increased, the final pressure in the containment would be increased, but if the helium initial temperature was increased, the final pressure would be decreased.

This study showed clearly that mitigation systems (like rupture disks for instance) were necessary to keep the containment pressure under the design pressure. It was found that a volume of 300000 m^3 and 400000 m3 could maintain the pressure under the design pressure for respectively TAURO and AHCPB models.

For the ''TAURO'' concept, a scenario based on a phenomenological approach is proposed to describe consequences of high pressure helium ingress in the Lithium-Lead flow in the case of a rupture of heat exchanger tubes.

REPORTS AND PUBLICATIONS

[1] « Response of Li-Pb self cooled (TAURO concept) and AHCPB models to loss of helium accidents » G. Mignot NT DER/SERSI/LECC 99/4049b

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S2.5

Task Title : ALTERNATIVE CONCEPT PLANT MODELS
Evaluation of the safety and environmental characteristics of stellarator concepts

INTRODUCTION

The principal goal of this report was to perform a broad preliminary evaluation of the safety and environment characteristics and issues associated with the stellarator design for a fusion power station. It is important to note that any nuclear installation must respect the general safety requirements and recommendations established by the Safety Authorities and Utilities. In order to assess whether the stellarator design can conform to such safety requirements and recommendations, this report models the stellarator's safety aspects through determination and analysis of the principal safety-relevant inventories (i.e., energy, tritium and activation products) and preliminary considerations of the current design for the radioactivity confinement strategy. These safety-relevant inventories calculated for the stellarator design are subsequently compared with those predicted for tokamak designs comparable as fusion output (e.g., ITER and SEAFP). The comparison of the inventories for the two systems allow the first global evaluation, from a safety point of view, on the differences between the two configurations.

1999 ACTIVITIES

THE STELLARATOR CONCEPT

Stellarator configurations are intrinsically three dimensional (3-D) and require more sophisticated plasma theory and engineering than their tokamak counterparts. However, this 3-D aspect, which is particular to the stellarator, provides significant freedom in the plasma's design in order to circumvent many of the problems typically associated with the magnetic confinement scheme for fusion power. Accordingly, the 3-D aspect can be used to optimise the stellarator's configuration in order to obtain the desired fusion performance or for studying particular plasma physics at a minimum cost.

The various stellarator configurations currently available have distinct advantages over other magnetic confinement concepts and offer interesting solutions to many of the challenging problems of tokamak reactors. Such advantages and solutions include: more simplified plasma control, disruption-free operation, possibility for simplified divertor design, high β values, and others.

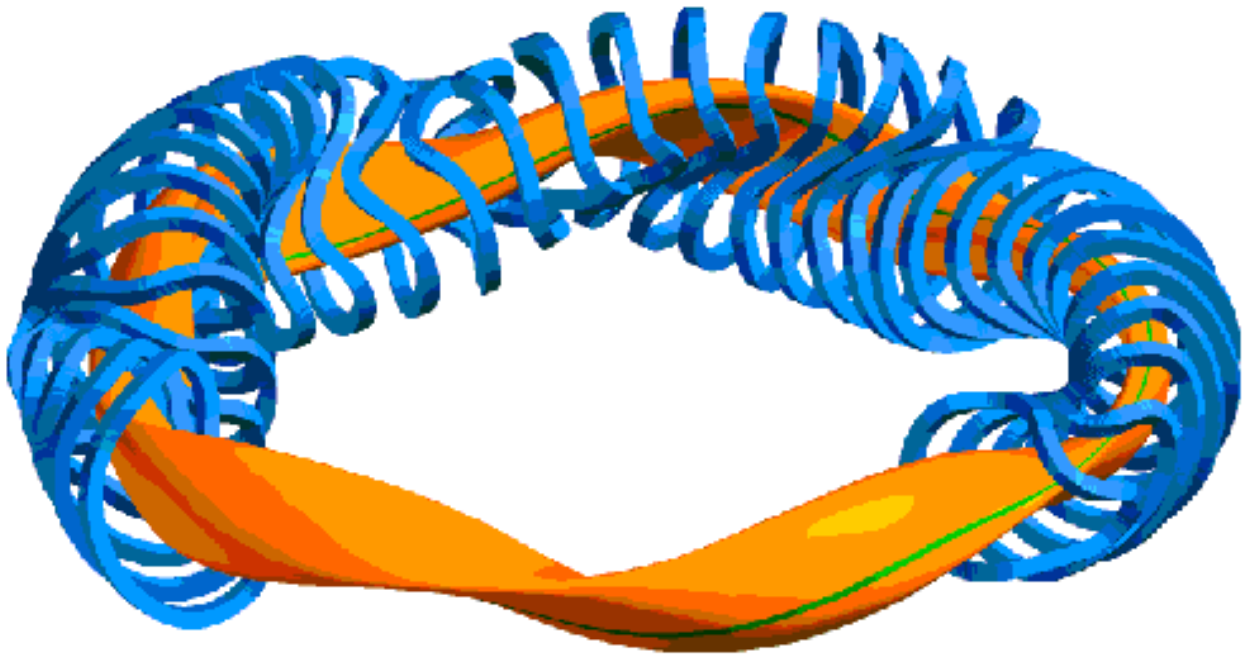

Figure 1 : Magnetic coils, plasma and magnetic axis in W7-X stellarator experiment

These capabilities are complementary to the advances of the symmetric tokamak program and offer new solutions to some of the problems that plague advanced toroidal configurations.

Accordingly, it would be interesting to provide a R&D synergy which combines current technologies of tokamak physics with the demonstrated control and design advantages of stellarators.

Currently there is development work on a new and advanced stellarator design concept. This concept offers interesting prospects concerning the progression towards steady-state fusion reactors. Advanced stellarator designs emphasise concept of modular coils, which circumvents the technical issues associated with the large helical windings of conventional stellarators.

The design criteria of a Helias (Helical Advanced Stellarator) type power reactor, such as Helias Stellarator Reactor (HSR), which, are presented in [1, 2]. The HSR is an upgraded version of the W7-X experiment [3].

The primary concern of the stellarator design is that stellarator power plants may be too large. For comparison of the tokamak and the stellarator designs, the main parameters of the HSR, ITER [4, 5], and SEAFP [6] reactors are listed in Table 1.

Table 1 : Main HSR device parameters versus ITER and SEAFP tokamaks

Parameter	Unit	SEAFP	ITER	HSR stellarator
Total fusion power (P_{fus})	(MW)	3000	1500	3000
Major radius (R)	(m)	9.4	8.14	22
Minor radius (a)	(m)	2.09	2.8	1.8
Aspect ratio (A)	-	4.5	2.9	12.2
Plasma volume	(m^3)	~ 1360*	2350	1407
PFC plasma surface	(m^2)	1400	1523	2600 - 3200
Neutron wall loading (q_n)	(MW/m^2)	2.1	~ 1.0	0.92 ÷ 0.75
Neutron fluence	($MW-y/m^2$)	10	0.3 ÷ 1.0	TBD
Equivalent current (I)	(MA)	10.4	21	3.5
Magnetic field on coils	(T)	12.8	12.5	10
Magnetic field on axis	(T)	7.8	5.7	4.75
Line average density (n)	(m^{-3})	1.62 E+20	1.25 E+20	2.12 E+20
Average plasma β	(%)	2.2	2.3	4.24
Tritium burn-up	(g-T/day)	480	~ 235*	470

*) estimated value.

PRELIMINARY SAFETY EVALUATION OF STELLARATOR

Energy inventories

Energy inventories were taken directly from, or provided by sample calculations, using as main reference the advanced stellarator concept HSR facility [1, 2]. Results for the calculated energy inventories for HSR stellarator device are summarised in Table 2.

Table 2 : HSR device energy inventories

Energy source	Amount of energy	comment
In-vessel fuel (DT)	4.6 GJ	Conceived as DT_power x particle confinement time (613 MW x 7.5 s)
Chemical energy (by reactions in PFC materials)	TBD (arising from W-steam reactions)	W is baseline armour material for FW and Divertor components
Toroidal field coils (N. 50)	100 GJ	
Poloidal field coils	0	HSR stellarator does not foresee poloidal filed
Plasma thermal energy	760 MJ	
Plasma magnetic energy	706.5 MJ	Due to both diamagnetic and Pfirsch-Schlüter currents
Coolant thermal energy (enthalpy)	TBD	The heat transfer system design is currently in progress as well as the choice of the primary coolant.
Total nuclear decay heat	TBD	Decay heat is proportional to the neutron flux [2].

Tritium and activation product inventories

Tritium and activation product inventories were deduced or calculated using simple calculations and the HSR reference. The currently available data on the tritium cycle for HSR are shown in Table 3 [2].

Table 3 : HSR device Tritium data

Tritium cycle parameter	Value	comment
Tritium burn-up (g/d)	470	This value is common for any fusion plant at fusion power of 3000 MW. There is not specific difference with respect to tokamaks.
Breeding production (g-T/d)	520	There is not specific difference with respect to tokamaks. A breeding rate = 1.1 has been considered.
T implantation in PFCs and BBL structures	171	The larger stellarator BBL (V = 2400 m^3) implies that larger amount of coolant material (He or Li-Pb) must be processed. The value will be strongly affected by specific material choice for PFC structure and armours.
Number of Tritons within the plasma volume	1.22 E+23	As a comparison, this value for the ITER plant is 5.8 E+22
Refuelling rate (g-T/h)	293	As a comparison, this value for the SEAFP plant is 400

In regards to the activation product quantification, there is currently a scarce amount of data for the stellarator design.

It is possible to suggest that since stellarators have larger dimensions (and thus larger surfaces under neutron irradiation and larger amounts of materials) than tokamaks, it can be concluded that the potential for activated material production is greater. However, all of the above-mentioned parameters (in particular the neutron fluence) have been fully taken into account.

Inventories comparison to tokamaks

In evaluating the general potential hazard of the stellarator facilities it is important to compare, as precisely as possible, the safety-relevant stellarator inventories with those of comparable (as fusion output) tokamaks. Even if the lack of information does not permit us to achieve exhaustive sets of results, a preliminary evaluation is presented in Tables 4 and 5.

Table 4 : Stellarator – Tokamak energy inventories comparison

Energy source in [GJ]	**ITER** [7]	**SEAFP** [6]	**Stellarator HSR** [2]
In-vessel fuel (DT)	15	325	4.6
Chemical energy (H_2, ozone or reactions in PFC materials by Be, C or W) (a)	8140 (Be), 0.6-0.7 (H_2)	285	TBD
Toroidal field coils	103	180	100
Poloidal field coils	47	≤ 50	0
Plasma thermal energy	1.14	1	0.76
Plasma magnetic energy	1.5	0.3	0.7065
Coolant-water thermal energy (enthalpy)	800 (b)	370 (c)	TBD
Total nuclear decay heat (1 day / 1 week)	512 / 1030	1910 / 6100	TBD (d)

(a) Possible reactions are : C-air, C-steam, Be-H_2O, Be-steam, W-steam. (b) For 1505 m^3 of total coolant inventory. (c) For 2600 m^3 of total coolant inventory. (d) « Decay heat is proportional to the neutron flux. For this reason I expect a smaller decay heat per volume in the stellarator », by Wobig in [8]. This value is, a minima, proportional to the output power (without considering the size aspect).

From Table 4, it can be concluded that the different amounts of stored energies are slightly smaller for the stellarator in comparison to the tokamak. However, globally they are comparable and of the same order of magnitude.

More detailed information is necessary for comparing the chemical energies (e.g. listing of the possible chemical reactions occurring in stellarators), coolant energy (choice of the heat transfer systems : materials, lay-out, etc.) and nuclear decay heat if the stellarator with the equivalent energies of the tokamak.

In regards to the decay heat, it appears that stellarators take an advantage of their larger area of plasma-facing surfaces. These surfaces create an enhanced thermal inertia and therefore produce a smaller temperature rise in the stellarator blanket. However, even with the expected smaller decay heat per volume of the stellarator, the integrated decay heat is probably the same for both reactor concepts.
Table 5 shows the tritium inventories calculated by using best-estimate approximations, for the ITER and SEAFP tokamaks in relation to, where possible, qualitative or quantitative indications of the HSR stellarator.

Table 5 : Stellarator – Tokamak Tritium inventories (in g-T) comparison

Location	**ITER**	**SEAFP**	**Stellarator HSR**
Fuel cycle outside the reactor	830	1100	~ the same
PFC armour	560 (Be)	1000 (Be) or 50 (W)	86 (W) (a)
PFC structure (FW, D, BL)		~ 20	85
Breeder blanket	750 (Be)	20 (with Li-Pb)	34 (with Li-Pb)(b)
coolant (H_2O)	5	40	170 (c)
Total site	1400	~ 2000 (d)	TBD

(a) Tungsten is baseline armour for FW and divertor. (b) Favourite candidate in HSR is Li-Pb breeder blanket. This choice avoids the presence of great amount of Beryllium. (c) Coolant system and water are considered. This high value is perhaps due to the larger stellarator dimensions. Anyway, this parameter is highly depending on the efficiency of the detritiation system. (d) Very conservative assumption.

The qualitative indications on the tritium inventory are as follows.

The tritium inventory in the exhaust and recycling system is approximately the same. The tritium implantation into the first wall, divertor, and blanket structures totals to a larger amount than in a smaller tokamak blanket. The tritium inventory in the PFCs and in the breeding blanket (BBL) is proportional to the area of the first wall and to the volume of the blanket. In addition, larger volumes and smaller particle confinement of the stellarator reactor should imply greater amounts of Tritium that require treatment. Although the tritium inventory for the total site is not yet defined, the above considerations result in a larger expected Tritium inventory for a stellarator than in a tokamak with comparable fusion output.

Radioactivity confinement analysis

The predicted distribution of containment barriers in the HSR stellarator is practically the same as in tokamak devices. In particular, three barriers are expected : Vacuum Vessel (VV), Cryostat Vessel (CV) and reactor building.

The VV, with its toroidal geometric development, is characterised by a shape that varies strongly its poloidal section along the toroidal direction (triangular, elliptical or D-shaped). This is due to the fact that VV must embody the particular helix-like structure of the plasma magnetic axis (Figure 1, for W7-X).

The CV is also predicted to have a toroidal geometric development, but having a rectangular constant poloidal section. The reactor building should have construction characteristics that are analogous to traditional nuclear power plants.

A general overview of some of the geometrical dimensions of the three containment barriers in HSR, SEAFP and ITER, is shown in Table 6.

It is clearly shown in the Table 6 that the stellarator concept has the larger dimensions of the three devices.

This implies a greater difficulty in achieving the needed reliability for the containment barriers (increased failure frequency), due to their size.

In addition, maintenance and inspection operations will became more important, taking into account that larger surfaces have to be treated.

Within this comparison analysis between stellarator and tokamaks facilities other safety important aspects have been considered, such as accident analysis, maintenance, waste and decommissioning.

Table 6 : Stellarator - Tokamak containment strategy comparison

	stellarator HRS			**SEAFP**			**ITER**		
containment barrier	*type*	*Vol (m^3)*	*D max (m)*	*type*	*V (m^3)*	*D max (m)*	*type*	*Vol (m^3)*	*D max (m)*
1st - VV	toroidal (varying Section)	9,230*	44	toroidal (constant Section)	5,160	~ 30	toroidal (constant Section)	3,800	26.4
2nd - CV	toroidal (rectang. Section)	42,000	58	cylindrical	20,000	40	cylindrical	19,000	36
3rd -Building	classical	-	~ 150	classical	-	~ 120	classical	-	~ 120

**) Value estimated from HSR VV dimensions.*

CONCLUSIONS

Comparing stellarators to tokamaks with comparable fusion output, the following considerations, from safety point of view, may be outlined.

RADIOACTIVE INVENTORY

Stellarators have larger dimensions and higher aspect ratio (and thus larger surfaces under neutron irradiation and larger amounts of materials) than tokamaks. This could imply that the potential for activated material production is greater. However, all of the key parameters (in particular the neutron fluence) have been taken into account to obtain well-founded conclusions. Even if the total site tritium inventory is not yet defined, the greater local tritium amounts (i.e. increased hazards for occupational dose exposure) could result in a larger expected Tritium inventory.

ENERGY INVENTORY

Locally stored energy inventories of the stellarator are slightly lesser than in comparable tokamaks. Nevertheless it can be concluded that the energy inventory of the global site is comparable and of the same order of magnitude with respect to tokamaks.

However, even with the expected smaller decay heat per volume of the stellarator, the integrated decay heat is probably the same for both reactor concepts.

CONFINEMENT

Larger stellarator dimensions imply greater difficulty in achieving the needed reliability for the containment barriers (increased failure frequency). The stellarator design predicts many VV and CV penetrations, which are characterised by non-constant poloidal distribution and different cross-section. These penetrations represent a weakness of the VV and CV from a reliability viewpoint. Maintenance could be easier in stellarators due to their larger available spaces. This aspect would imply less dose to the workers, but operations frequency will be higher because larger surfaces have to be controlled and more complex geometry requires diversified operations.

PLASMA CONTROL

Stellarators eliminate (or greatly reduce) the need for externally driven plasma current, which reduces the re-circulating power in the reactor and enables a simplified plasma control system.

Disruptive discharge termination and quench of the plasma current are not observed in stellarators and their exclusion precludes the existence of vertical displacement events.

The relative ease in which plasmas are shutdown is a very impressive characteristic of stellarators. Also, for unscheduled shut-down, neither violent plasma excursions nor thermal shock on PFCs are expected.

WASTE

The radioactive waste in a stellarator is expected to be comparable, under the same conditions of neutron flux and output power, to those in tokamaks.

LIFETIME

Considering that lifetime of FW and blanket segments is determined by the maximum allowable neutron fluence, recent studies conclude that this lifetime in a stellarator may be several factor larger than for tokamak reactors. If divertor lifetime is dependant upon the disruption number, than stellarators will have an additional factor that increases their lifetime over that of tokamaks.

Finally, it is important to stress that a more proven knowledge of some fundamental safety-related parameters (e.g. radioactivity confinement strategy, planned irradiation scenarios, time scales for inventories release, and others) remains necessary in order to perform more meaningful comparisons.

REPORTS AND PUBLICATIONS

[1] « Evaluation of the safety and environmental charcateristics of stellarator concepts » M. Costa, C. Girard NT DER/SERSI/LEFS 99/5043

NOMENCLATURE

BBL	Breeder blanket
beta (b)	plasma pressure at which the plasma can be confined by the magnetic configuration
CV	Cryostat Vessel
FW	First Wall
HSR	Helias Stellarator Reactor
ITER	International Thermonuclear Experimental Reactor
PFC	Plasma Facing Component
SEAFP	Safety and Environmental Assessment of Fusion Power
VV	Vacuum Vessel
W7-X	Wendelstein 7-X experiment

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S3.1

Task Title : OPERATIONAL TRANSIENTS AND THEIR IMPACT ON SAFETY Operational states

INTRODUCTION

In the frame of the long term studies for fusion reactor, most of the accidental situations were related to the normal operating conditions. It has been decided for the complementary studies of the 1999 SEAFP program to assess the possibilities of having to manage accidental situations during other operating states (task CEA S3.1).

In this respect, after having postulated the possible operating states as well as the operational transients of a fusion reactor, the different controls which allow the facility to « track » those transients have been identified. The accidental sequences, based on the failure of those controls or the reach of a setpoint, are then defined and proposed for a detailed analysis (ENEA task S3.2).

1999 ACTIVITIES

DEFINITION OF A FUSION REACTOR OPERATING STATES

Shutdown for blanket/divertor replacement :

In this case the Vacuum Vessel ports are opened, the remote handling tools are active and the boundaries of the first confinement are well beyond the Vacuum Vessel and limited to the working areas located behind the magnets.

The decay heat has to be removed by the primary circuits which are not under maintenance. Those circuits are not necessarily at the nominal conditions (flow, pressure and temperature).

Cold shutdown :

This state can be separated in two cases :

- cold shutdown for repair ; in this case the circuits are at low pressure and the temperature are between 10°c and 70°c . The Vacuum Vessel is closed while the Cryostat can be accessed,

- normal cold shutdown : this state is the first state before start up and also the final state when some components are failed and need maintenance. The cooling loops can be pressurized (~ 20 bars) and the temperature is controlled by a minimum flow in each loops. The primary loops are solid (single phase) there is no vapor in the pressurizer.

Intermediate states :

These intermediate states are those reached between cold and hot shutdown, they are of two types :

- single phase intermediate state (when there is no vapor in the pressurizer),

- double phase intermediate state (once the vapor dome is formed in the pressurizer)

In the second case the circuit is less sensitive to pressure variations which can be controlled by the pressurizer valves.

Hot stand-by :

The nominal values for the cooling loops are reached in this case. The energy is provided by the decay heat and the pumps rotation and is removed by the secondary circuit through the atmosphere or in the condenser.

Normal operation :

In this case we can also distinguish between full power (above 20% of nominal power) and reduced power.

In the case of reduced power a certain number of actions are manually performed by the operator such as fueling increase, opening of main steam valve and feedwater control in the steam generators.

In the transition phase at reduced power where changes from manual to automatic operation has to be performed, it is necessary to define properly the procedures in order to avoid or minimize human errors. Accidental sequences during this phase are certainly covered by sequences at full power but special attention should be paid to these transitions where automatism must be triggered.

DEFINITION OF THE DIFFERENT CONTROLS

Normal operation controls

The possible needed controls for a fusion reactor are defined below :

- fusion power control,

- pressurizer pressure and temperature,

- steam generator water level,

- turbine by-pass system .

Emergency systems controls

The emergency systems are triggered on different high or low signals from control parameters. Others protection systems may be defined as well as their triggering parameters, they could be for a fusion reactor :

- fast plasma shutdown,
- turbine by-pass circuit,
- vacuum vessel pressure suppression system,
- isolation of different areas of the reactor building,
- active cooling of the vacuum vessel,
- magnets system discharge.

DEFINITION OF ACCIDENTAL SEQUENCES IN NORMAL OPERATION

Following the review of the operating conditions, normal transients and possible controls of a nuclear fusion reactor, we can define some scenarios where assessments of the behavior of the plant has to be performed in order to check if safety limits cannot be reached.

During operating states

It is obvious that maintenance states where only one barrier is provided between radioactive products and environment needs more attention.

The maintenance states where the divertor ports or the vacuum vessel ports are opened for replacement of internal components, need a detailed analysis.

The accidental situation in this operating states which is the most significant for external releases is related to the mobilization of the tritium in the Plasma Facing Components and its release. It can be assumed a loss of the inertization with oxygen ingress, a local hot spot on the PFC and consequently a reaction with beryllium which could release an important quantity of tritium.

Two human errors inducing leak of the contaminated cooling fluid can also be envisaged :

- inadvertent opening of the circuit while still pressurized,
- inadvertent opening of an isolation valve during maintenance activity.

An assessment of the consequences of these human error induced sequences could be done. However, we can suppose that an appropriate venting of the rooms under maintenance activity could prevent external releases beyond the second barrier.

During operating transients

These transients are :

- normal start up of the plant,
- instantaneous 10 % decrease of the removal heat capacity on the turbine,
- total loss of the turbine load, opening of the turbine by-pass circuit and decrease of the fusion power to 20% of nominal power.

The assessment of these transients can also be separated in two items :

- verify that an operating transient does not trigger the emergency systems,

- model an accident that could happen during a normal transient.

CONCLUSION

This first approach of the definition of different possible operating states for a fusion reactor shows that detailed studies have to be done for :

- the clear definition of these states,

- the procedures and means to provide to go from one state to the other,

- the clear and accurate definition of the different controls needed to avoid physical parameters reaching the safety limits.

Although not all these information were not available in the SEAFP project, the assessment, for this year, of some normal operating situations can be performed. The scenario identified, once simulated with safety codes (task SEAFP - S3.2), can help give recommendations for the definition of safety studies and safety implementation of operational states in a fusion reactor study.

REPORTS AND PUBLICATIONS

[1] « Definition of accidental situations for different operating states for a fusion reactor » C. Girard NT DER/SERSI/LEFS 99/5012

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S3.2

Task Title : ACCIDENT SEQUENCE ANALYSIS Dimensioning of a pressurizer for a fusion reactor; Requirement analysis and dimensioning model for a turbine bypass system applied to a power generation fusion reactor

INTRODUCTION

The SEAFP fusion reactor is still in a definition phase. There are two different types of SEAFP reactor, one cooled by helium and one cooled by water. These documents only considers the water-cooled variant.

The first document describes the different options that can be taken for the definition of a pressurizer. The main hypothesis is that no regulation on the pressurizer level is supposed, which means that the primary circuit has a constant mass.

The second document describes a turbine bypass system. On a fusion reactor, there are no counter reactions that could equalise the power from the plasma, and the extracted power in the steam generators. That's the reason why a system is needed in order to reduce the effect of a turbine setting off transient.

1999 ACTIVITIES

DIMENSIONING OF A PRESSURIZER :

This document provides the information necessary for the dimensioning of a pressurizer for the SEAFP fusion reactor. Following a description of the role of the pressurizer and its operation, the criteria affecting the dimensioning are discussed, and the internal systems within the pressurizer (pressure regulation systems, subvolume at medium temperature, etc.) are examined.

Finally, the volume of the pressurizer is calculated.

DIMENSIONING A TURBINE BYPASS SYSTEM

This documents proposes a system for the SEAFP reactor to guard against sudden reductions in power demand.

After demonstrating the need for such a system, the document describes a Turbine Bypass System (TBS) used in power generating reactors (PWR) and a solution for the SEAFP reactor based on this TBS is proposed.

CONCLUSION

These two notes are preliminary studies, more data are requested in order to make the precise definition of these two systems.

It shows the need of a turbine bypass system on a fusion reactor.

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S3.3

Task Title : OPERATIONAL TRANSIENTS AND THEIR IMPACT ON SAFETY Confinement strategy

INTRODUCTION

The design of a Power plant is not only driven by the nominal operational states and by the impact of severe accidents protection.

Prevention of more common accidents and easy management of those ones may have a significant impact on the overall design and definition of safety systems.

This issue is being addressed starting from a set of a selected number of operational states and transients.

Following an identification of the operational transients, different analysis were performed in order to assess the consequences of a possible accidents occurring in these situations.

The main controls involved in those transients are related to the pressurizer, the steam generator and the turbine by-pass.

Analysis of a LOFA of the primary circuit occurring at 20 % of Nominal Power, showed that in some circumstances the First Wall melting is obtained before the pressurizer level reaches the high level alarm setpoint (the low flow setpoint being deactivated in this case).

An assessment of the requirements for the turbine by-pass circuit is being performed and give recommendations for the design of auxiliary safety rated circuits.

Following the identification of the different operating states (task S3.1) and possible induced accidental situations, recommendations on how to account for these situations from the operational and safety point of view was performed in this macrotask.

1999 ACTIVITIES

PRESURIZER DESIGN REQUIREMENTS

The variations in temperature and pressure in the primary circuit during the operation of a reactor result in changes in the volume of the primary water.

A reduction in the power demand leads to an increase in the primary temperature and a consequent expansion of the primary water.

These variations in volume may be absorbed in one of two ways :

- by adding and removing water (variable mass operation),
- by storing and returning water (constant mass operation).

In the first case, a separate circuit absorbs the variations in the volume of primary water, in the second case, the variations in volume are absorbed in a storage vessel.

With a variable mass operation the pressurization margin may fall rapidly during some incidents which could lead to damaging the first wall (critical heat flux) as the plasma can’t be stopped immediately.

The constant mass operation seems preferable to the fusion reactor because the transients are very quick.

For the same reason (low pressurization margin), the pressurizer with gas is not adapted : in case of extrusion, there isn’t a water boiling which counteracts the pressure decrease.

The pressurizer design (especially the internal volume) depends on the turbine bypass system and the control and monitoring system (delay between plasma disruption and secondary valve closing, …) and therefore must be done in accordance with the design of theses systems.

The pressure regulation system depends on many parameters :

- the efficiency required (linked with the pressurization margin and maximal pressure increase),
- the worst case transients (for instance, in case of a spray with medium water reservoir, the maximal intrusion flow must be taken into account for the reservoir volume design),
- the operating conditions (like the difference between the pressurizer temperature and the primary temperature, …).

TURBINE BYPASS SYSTEM

Any reduction in the power generated by the turbo-alternator caused, for example by a failure of the turbine, induces an imbalance between the power demand and the power supplied by the primary.

This reduction in power demand will have the following effects :

- the temperature of the primary circuit will rise leading to reduced cooling of the first wall (with a consequent risk of reaching a critical heat flux and damaging the first wall) and increased pressure (as a result of a rise in pressurizer level) with the risk that the pressurizer discharge valves will open.

- the temperature and pressure of the secondary circuit will rise which may lead to the steam generator pressure relief valves opening.

A system is therefore required to protect the plasma chamber cooling from a sudden reduction in turbo-alternator power. This system must be capable of accepting the excess steam when the turbine load falls, thus maintaining a constant rate of steam generation.

CONCLUSIONS

In the macro-task 3 (task S3.1 : Operational states , task S3.2 : Accident sequence analysis and task S3.3 : Confinement strategy), a first tentative effort was made to define what could be the operational transients for a fusion reactor. It was thus necessary to define the different operating states although a lot of issues were still unresolved, particularly for the plasma operation.

These transients are those due to external causes like a total power loss on the network which necessitates to deride rapidly the reactor output power (Turbine Bypass System). There are also network requirements on the instantaneous variation of the power which is usually limited to 10% of the nominal power.

The operational transients related to internal causes are primarily those due to the normal operation of the plant where detailed procedures have to be defined in order to avoid human errors. There are also those transients related to equipment failures whose consequences are not important, but which necessitate an operator action.

Once those transients are defined, an assessment of the transient conditions is performed with the same computer codes used in accident analysis. It was shown, for example, that in some cases the pressurizer level by itself was insufficient for detecting a loss of flow.

This is why it is necessary to define the controls and the means that will ensure the protection of the plant for all the possible states. The most important controls can be defined as the :

- fusion power control,
- primary cooling circuit conditions control,
- steam generator water level,
- turbine by-pass system.

The work performed under this macrotask also gives recommendations for the identification of the design basis for systems that are not directly safety relevant, but whose correct operation is needed to avoid the initiation of safety rated equipment's. The Turbine Bypass System is one of these systems not required by safety authorities, but whose presence allows limiting the use of the primary and secondary relief valves and also to realise the plant's isolation in case of power network failure.

The conclusions showed that special attention has to be paid to those transients even if their uncontrolled consequences are minor and covered by the reference accidents. In fact, the operational transients have no impact on safety provided they are well controlled by operational procedures and adequate controls that must be defined and assessed in the conceptual study.

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S5.21

Task Title : TRITIATED WASTES MANAGEMENT

INTRODUCTION

Steel detritiation remains a crucial element in waste management in the framework of the ITER project. The Nuclear Materials Processing Department at the Nuclear Research Centre of VALDUC is performing detritiation of metallic wastes resulting from the operation or dismantling of glove boxes used in tritium, by means of a melting process (initial activity of about 1000 Ci/t). This experience is important for steel detritiation studies in the framework of waste management for ITER.

The objective of this task was to better understand the phenomena involved during the melting using autoradiography images that permit to assess the distribution of tritium activity in the ingots. In a second time, the optimization of the process using melting with bubbling with argon and hydrogen was studied.

1999 ACTIVITIES

CHARACTERIZATION OF INGOTS...

For the moment, the metallic wastes produced in CEA/DAM VALDUC are characterized by the decontamination factor after melting and the measurement of the degassing rate.

A cooperation agreement has been initiated with the Laboratoire de Métallurgie Structurale at CNRS to study methods of local rather than overall degassing rates measurements as well as tritium activity distribution in ingots and to characterize the distribution of the residual amount of tritium in ingots after detritiation treatment.

Autoradiographic observations were performed in order to assess the detritiation efficiency through the mapping of the tritium distribution and to look for trapping on microstructural heterogeneities within the ingot. Furthermore, the tritium desorption kinetics were quantified in order to relate the rate of tritium outgassing from the specimens with the total tritium quantity measured at the CEA/DAM on the whole ingot.

The samples were provided by the CEA/DAM. They were cut in an austenitic stainless steel ingot with a low speed saw machine, cooled by a VORTEC type system.

Care has been taken during the cutting process to avoid any heating of the samples which may favor partial tritium desorption (maximum temperature 40°C, cutting time : 50 hours).

It must be emphasized that numerous internal voids and some very large cavities sketched on figure 1 were observed on the cross sections of the ingot. The tritium concentration in the stainless steel ingot (58.3 kBq/g in the lower part and 166 kBq/g in the upper part) and the tritium desorption rate from the ingot (8.76 kBq/24h) have been previously characterized by the CEA/DAM.

For the autoradiographic work, each surface of the specimens was polished with grit SiC paper and then carefully rinsed and cleaned in ethanol.

If there is any tritium released during sawing or polishing, it is expected to be homogeneous on all the surface and will not have modified the tritium distribution shown by autoradiography.

After polishing the samples were stored at -20°C in order to limit the tritium desorption.

Then, the polished surface of each specimen was covered with a specific film (Amersham Hyperfilm ^{3}H) used for macroscopic tritium autoradiography observations.

The autoradiographic observations performed on the longitudinal and transverse sections of the ingot are shown on figures 1 and 2.

The macroscopic tritium distribution is quite homogeneous throughout the ingot both in the longitudinal and transverse directions.

On the external surface, two phenomena were evidenced : a marking in the shape of a halo due to a tritium enrichment in the oxide layer and a tritium depletion due to the areas that have already desorbed. The same phenomena are evidenced on the internal surface of some large cavities.

Particular areas with an important amount of tritium were observed. They correspond to small cavities with probably a special composition that permit tritium fixation. This point must be checked in 2000 because addition of elements such as calcium, aluminium, manganese... during melting can be a solution for tritium degassing limitation at the final stage of the process.

Except this, there is however no clear dependance of the tritium ditribution on the macrostructure of this cast alloy.
The tritium distribution observed on the bottom of the ingot (1^{rst} melt) and on its top (2nd melt) is quite similar. However a slight increase in the darkening of the autoradiographs obtained in the top cross section may reflect a larger tritium concentration in this part of the ingot in agreement with the measurements performed at the CEA/DAM.

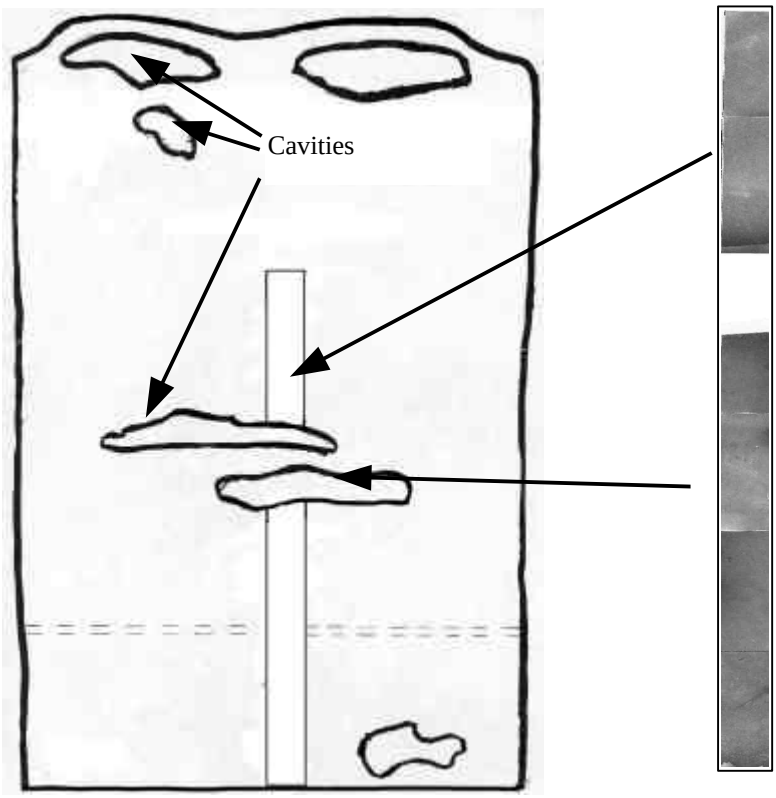

Figure 1 : Autoradiographic observation of the macroscopic distribution of the residual tritium on the longitudinal cross section of the ingot

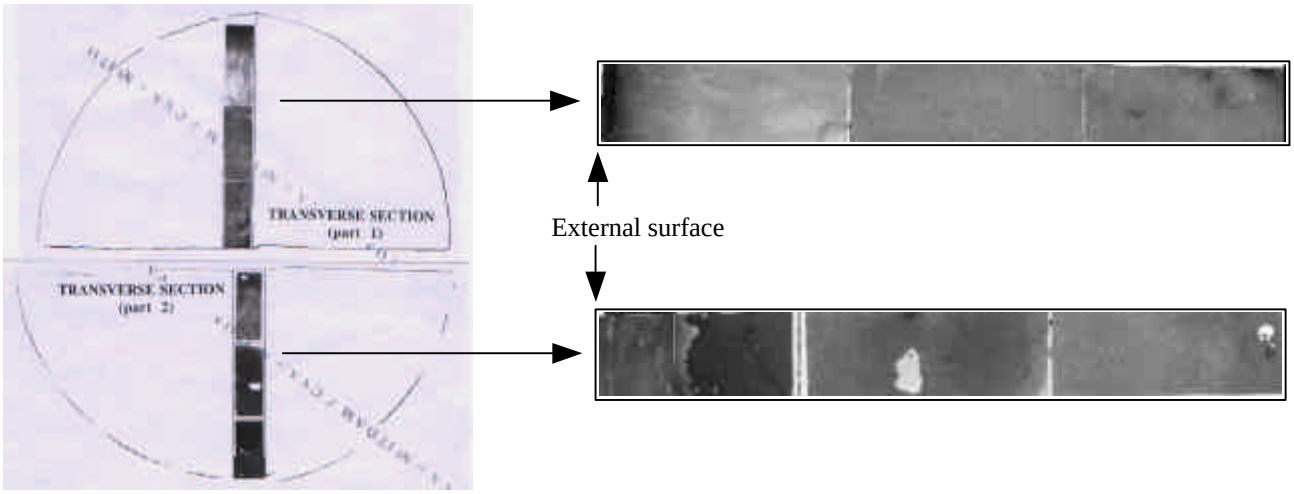

Figure 2 : Autoradiographic observations of the macroscopic distribution of the residual tritium on the transverse cross section of the ingot

An original procedure developed at the CNRS Orsay, based on liquid scintillation counting allows to record the cumulated amount (expressed in dpm) of tritium desorbing from massive specimens as a function of time. The tritium desorption profiles were recorded at room temperature for 1500 min. One of the desorption curves obtained is shown on figure 3. A quasi linear cumulated amount of desorbed tritium-time behaviour was observed. This is a consequence of the very low tritium desorption kinetics from the small massive specimen (about 1,2 cm^3) at room temperature. The mean desorption flow (expressed in $Bq.cm^{-2}.s^{-1}$) can be estimated from the slope of the straight line representative of the activity vs time relationship, knowing the value of the lateral surface of the specimen.

The desorption flow recorded on the small samples yields 8 to 10 10^{-4} $Bq.cm^{-2}.s^{-1}$. These very small values illustrate the very high sensitivity of this technique. The desorption flow recorded on small polished samples was however about 25 times larger than the overall flow computed from the measurements (8.76 kBq/24h) performed at CEA-DAM on the full ingot.

This difference may be explained by a stronger barrier effect of the oxide layer covering the surface of the ingot and by the existence of a tritium depleted zone on the external surface of the ingot as suggested by the autoradiographic observations.

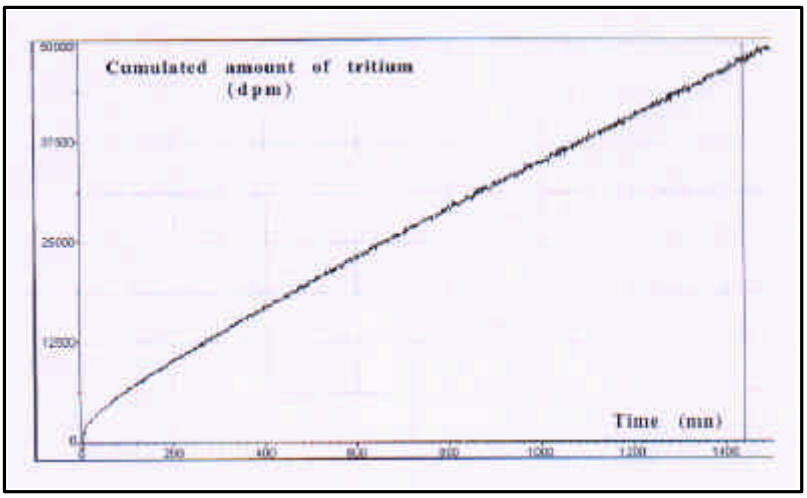

Figure 3 : Tritium desorption kinetics at 20°C from a specimen of detritiated stainless steel used for the autoradiographic observations.

Modelling of melting with bubbling...

With respect to the optimization of the detritiation process by melting, melting under reduced pressure with bubbling appears to be an interesting avenue of investigation. The experience of the IRSID (Metallurgical Research Institute) on decarburization, dehydrogenation, and denitridation can be used for detritiation.

Computational code developed by the IRSID for deshydrogenation was adapted to detritiation. The methods for removing the hydrogen from the metal therefore consists in decreasing the total pressure by using vacuum and reducing the molar fraction of the hydrogen by agitation and purging using gas. In the case of detritiation, it can be interesting to use argon and hydrogen bubbling, argon allowing a better agitation of the metal melted and hydrogen promotting isotopic exchange.

In any case, the modelling of the phenomena thus encountered depends on :

- The thermodynamics of the metal/gas reactions (governed by Sievert's law),
- The elementary kinetics (taking into account the formation of H_2, T_2 and HT),
- The reactor hydrodynamics.

This computational code is now ready to give the calculus limits of melting with argon and hydrogen bubbling, in term of detritiation rate.

CONCLUSIONS

In one hand, tritium autoradiography allows the characterization of the macroscopic distribution of residual tritium in a detritiated ingot of austenitic stainless steel.

With the present detritiation process, the macroscopic distribution of tritium is relatively homogeneous throughout the ingot. This is an important information concerning cooling kinetics that are too rapid to allow a tritium enrichments in the ingot core. Nethertheless, two phenomena are evidenced near the external and internal (large porosities) surfaces : some local tritium enrichments associated with the oxide layer and depletion due to tritium desorption. The differences observed between tritium desorption measurements performed at CEA/DAM and CNRS Orsay show the importance of parameters such as the oxide layer and the depleted zone on the external surface of the ingot.

In the other hand, a computational code has been developped for modelling detritiation by melting with argon and hydrogen bubbling. It will permit to assess the influence of process parameters such as gas composition, gaz flow, mass of melted metal... on detritiation rates.

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