

Tritium Purification, Storage and Supply in the Tritium Plant of a Fusion Reactor

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The techniques used at the JET tritium plant for separation of torus exhaust gas and impurity processing are discussed in view of the requirements for the tritium plant of a next generation fusion reactor. A chemical method to convert tritiated hydrocarbons and water with a liberation of molecular hydrogen is presented. The efficiency and safety of uranium hydride containers used at JET for pumping, storage and supply of tritium and deuterium are analyzed and the possible use of non-radioactive intermetallic alloys for metal hydride storage systems is investigated.

1. Introduction

The JET tritium plant (Active Gas Handling System, AGHS) has recently been commissioned with trace tritium - a mixture of $\approx 30\text{TBq}$ of tritium diluted in D_2 , left over from JET's first tritium experiment in 1991 [1]. In the near future, the operating inventory will be built up in steps (up to 20g T_2 in 1996/1997) to the full operating inventory of up to 90g T_2 in 1998/1999. The JET tokamak, with its integral fuel recycling system, is expected to provide a sound database for the tritium processing requirements of ITER, operating with a burning D-T plasma.

This paper concentrates on torus exhaust gas processing which is currently a key point of the fuel cycle and on the operation and performance of JET uranium hydride containers. As an alternative, several candidate intermetallic alloys for tritium storage and supply are discussed. The paper presents results of experiments at JET, at the Russian Research Institute of Inorganic Materials and at the Karlsruhe Research Center (Germany).

2. Basic features of tritium processing on JET and ITER

The main features of the JET tritium plant is its batch mode of operation: during torus

operation, deuterium, tritium and impurities are pumped by cryocondensation pumps in the torus and neutral beam injection system. These primary pumps may be regenerated at the end of each operation day. The released gas mixtures are transferred to cryocondensation panels in the AGHS Cryogenic Forevacuum (CF) system [2]. The CF system separates the pumped gases into three "clean" product streams:

- helium (which is accumulated on charcoal cryosorption panels);
- hydrogen isotopes mixture;
- impurities.

Helium and impurities are collected in a buffer tank for further processing, hydrogen isotopes are transferred to the Intermediate Storage (IS) uranium containers (U-container) and subsequently fed to either of the two isotope separation systems (ISS), ie Gas Chromatography (GC) or Cryogenic Distillation (CD). Separated D_2 and T_2 are stored in Product Storage (PS) uranium containers and supplied to the torus and neutral beam injectors.

Impurities released from the torus cryopumps during daily regeneration to 77K consist mainly of methane and ethane. However, recent experience has shown that regeneration of the cryopumps to room temperature, or higher, after extended operation for one to two weeks leads to the release of substantial

amounts of higher hydrocarbons (see Fig. 1) which had been cryocondensed on the 77K baffles of the cryopump. The JET AGHS Impurity Processing (IP) system can cope with these species as well as with the previously expected methane and ethane.

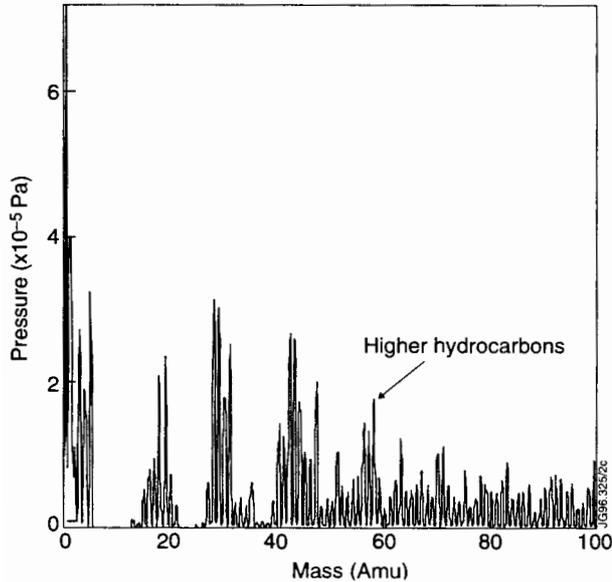


Fig.1. Mass spectrum of gases released from JET divertor cryopump regenerated at $\approx 300\text{K}$

The tritium plant for ITER will have to process exhaust gas quasi-continuously at a flow rate of ≈ 200 times the throughput of JET. The ITER design [3,4] considers a once-through operation mode for torus exhaust gas separation and impurity processing. Gas separation is to be performed by a front-end palladium permeator. Most of the permeated clean D-T fuel will be directly recycled to the torus, while a bypass stream is isotopically separated to adjust isotopic balance as required for optimum torus operation. For impurity processing, a number of advanced techniques are currently under development [3]. The once-through mode of operation presently selected for ITER impurity processing requires a substantially higher efficiency than the process used at JET, where impurity batches can be recirculated until the necessary detritiation factor is achieved.

3. Torus exhaust gas processing

To minimize releases of tritium to the environment for both JET and ITER, a gas detritiation factor (DF_{tot}) in the order of $\approx 1 \cdot 10^8$ is desirable. DF_{tot} is the product of detritiation factors of primary fuel clean-up (DF_{FCU}), the impurity processing (DF_{IP}) and the final exhaust detritiation system (DF_{ED}). For the currently adopted nominal composition of plasma exhaust gas (total impurity content $< 10\%$, including $\leq 4.5\%$ methane and $\leq 0.5\%$ water), detritiation factors DF_{FCU} and DF_{IP} in the order of ≈ 10 and $\approx 10^5$, would be acceptable.

The primary purpose of the fuel clean-up (FCU) system is the purification of hydrogen isotopes to a purity acceptable for the isotope separation system or direct recycling to the plasma. Purification of torus exhaust gas by the ITER front-end palladium permeator should easily achieve impurity concentrations of $\leq 1\text{ppm}$ in the permeated hydrogen stream, with an impurity "waste" stream containing less than 1% hydrogen.

Tests of the JET method of cryogenic separation of gas mixtures have shown that this system separated D_2 from the methane (8.9%) and helium (2.2%) giving a D_2 purity of $\geq 99.95\%$. Final purification of hydrogen isotopes takes place in the U-containers in the IS system.

3.1 Impurity processing in the current JET AGHS

Hydrocarbons and traces of molecular hydrogen are converted into water (with oxygen added as required) on a palladium catalyst at temperatures up to 770K. The water is collected in a cold trap operating at 160K. The gas mixture is recirculated until the desired DF_{IP} is achieved. Subsequently, after stacking the detritiated waste gas through the Exhaust Detritiation (ED) system, the water from the cold trap is passed through modules filled with iron powder at $\approx 823\text{K}$. The resulting molecular hydrogen is absorbed in U-containers together with residual water not decomposed on iron. Oxidized iron can be regenerated with tritium-free H_2 . Tritium

contaminated water resulting from regeneration is retained in the ED molecular sieve adsorbers for disposal as tritiated water. The option of further reprocessing in a CECE (combined electrolysis catalytic exchange) type water detritiation system is being considered. Trace tritium commissioning with 500appm tritium concentration showed high efficiency of the recombiner/cold trap system, with residual activity of $<0.1\text{GBq/m}^3$ in the 2m^3 buffer tank after processing a mixture of tritiated methane, hydrogen and water. The residual tritium concentration in water produced during reduction of FeO_x with trace tritium hydrogen corresponds to $\approx 0.5\%$ of the processed tritium inventory, giving a $\text{DFIP} \geq 200$, which is acceptable for the next tritium phase of JET.

Evaluation of a new technique for impurity reprocessing in the JET tritium plant. We are presently investigating advanced techniques to improve the performance of the JET AGHS IP system, with the following desirable features:

- capability to process impurities of widely variable flow rates and composition;
- capability to process "abnormal" batches, such as the recently found higher hydrocarbons;
- long service life without component replacement;
- avoidance of cumbersome on-line and real-time chemical analysis equipment for process control.

The key point of this technique is the introduction of a palladium permeator and a new catalytic reactor.

Investigation of palladium membrane. Pd-membrane ageing caused by the accumulation of ^3He , which is a product of ^3H decay, and poisoning by several gases are investigated for membranes used for commercial Pd-permeators. Samples of commercial tubes of Pd-alloy (containing 15%Ag, 0.6% Pt, 0.6% Ru, 1.0% Au and 0.2% Al additionally to Pd) have been exposed [5] to D:T=1:1 mixture at temperature of 620K and gas pressure of 500-900kPa. Tensile properties, the dimension of

He bubbles, crystal lattice parameters of an initial sample and samples exposed for extended periods of time have been measured. Electron microscopy, X-ray analysis and positron annihilation spectroscopy have been also used for characterization of exposed samples. Details and results of the experiments are given in [5]. Positron annihilation analysis of the samples shows a tendency for increase in concentration and size of defects of the crystal lattice with extended exposure of the membrane in tritium atmosphere. Nevertheless, no obvious effect of ^3He (^3He atomic concentration is calculated to be 470appm, average size and concentration of bubbles are 3nm and $\approx 2 \cdot 10^{16}\text{cm}^{-3}$) on mechanical properties and crystal lattice parameters is detected after 2000 hours of exposure.

The Pd-membrane (Pd-25%Ag alloy) throughput decreased to 50% of its initial value after ≈ 150 hours of operation at 723K with He containing from 1 to 50kPa of CH_4 and increased again up to 60% after ≈ 50 hours of operation with mixtures containing additionally H_2O , CO, CO_2 . This effect means that the efficiency of long term operation of Pd-membrane depends on the chemical composition of the gas mixture. It was found that the operation of the Pd-membrane for several months with gas mixtures containing higher hydrocarbons and at temperatures $>673\text{K}$ can lead to irreversible chemical contamination of the membrane surface and a decrease in its permeability for hydrogen.

Possibility of HITEX reactor application in JET tritium plant. A schematic layout of HITEX operation in a batch regime is shown in Fig.2. Gas species are mixed with protium in a buffer tank (BT) and then circulated through a catalytic reactor (CR) and a palladium permeator (P). Molecular hydrogen removed from the mixture by the Pd-permeator is sent to ISS. The tritium depleted product of the ISS is returned to the catalytic reactor inlet.

The rate of carbon deposit on Nikki 111-catalyst was measured to be 2.5w.%/h at CR temperature of 820K. The rate of volume rise of growing carbon deposit was found to be $\approx 5[\%(\text{volume rise})/\text{w.}\%(\text{C})]$. This considerable increase in the volume of Nikki 111 during carbon deposition needs a very large void in CR for the achievement of high carbon deposition.

The structure of carbon deposited on Ni depends on the initial composition of the catalyst and the operation conditions of hydrocarbon cracking. Highest bulk density has carbon deposited in graphite form [8]. To allow carbon to be deposited in the form of graphite, the original Nikki 111-catalyst has been modified. Several modifications of Nikki 111 were tested. Results of the tests presented in Table 1 show that there is a possibility for optimization of Ni-reactant modification.

Table 1. Results of the test of CH₄ cracking on modified Ni-catalyst

Parameter	Nikki 111	Modified Ni-catalyst		
		No.1	No.2	No.3
Rate of C deposit ^a), w.%/h	2.5	≈ 20	≈ 80	≈ 40
Bulk density of C deposit ^b), kg(C)/m ³ (C)	--	7000	9000	4000

a) Rate of CH₄ cracking was measured by thermogravimetric analysis (sample of $\approx 0.1\text{g}$, $T=770\text{K}$, $P_{\text{CH}_4}=100\text{kPa}$). Maximum carbon depositions of 1200w.%, 3600w.% and 1800w.% are achieved experimentally for reactants of modification No.1, No.2 and No.3, respectively.

b) Error of measurement is of $\approx 50\%$ because of very small sample.

The removal of CH₄ and H₂O from gas mixtures is investigated with a Ni-reactant No.1. As an example, experimental kinetic curves are shown in Fig.3. Methane is more difficult to remove than water. The rate of H₂O removal rises with increasing of partial pressure of water vapor. The rate of CH₄

removal is found to be independent of CH₄ pressure in the range between 1 and 100kPa. Rate of carbon deposition is a function of the amount of carbon deposited on Ni. The maximum rate of 24w.%/h was observed at carbon deposition of $\approx 400\text{w.}\%$. The decrease in catalytic activity of Ni-reactant at carbon deposit $>400\text{w.}\%$ is caused by diminution of concentration of Ni-particles penetrated into the carbon deposit from the "mother" reactant [12]. The rate of the reactant volume rise is measured to be $\approx 0.4[\%(\text{volume rise})/\text{w.}\%(\text{C})]$. Efficiency (F) of impurity removal from gas mixture circulating through a CR was estimated by fitting the experimental kinetic curves to the equation:

$$\ln(P_0/P) = (G \cdot F \cdot t)/V \quad (2)$$

which is a solution of differential kinetic equation:

$$V \cdot (dP/dt) = -G \cdot (P - P_{\text{out}}) = -G \cdot [P \cdot (1 - F) \cdot P] \quad (3)$$

(where V is the volume of BT; P₀, P are the initial and actual pressures of impurity in BT; P_{out} is the pressure of impurity at the CR outlet; G is the gas flow rate).

Fig.3 shows a comparison of the kinetic curves of CH₄ removal from helium for Nikki 111 and Ni-reactant No.1. Simple calculations show that to provide the same efficiency of CH₄ removal, the amount of Ni-reactant No.1 required is ≈ 5 times less than that for Nikki 111.

The simulation of impurity processing for the JET tritium plant has been carried out with helium containing 50vol.% of CH₄ and 5.5vol.% of H₂O. Three different modes of IP operation are considered (see Table 2). Mode 1 presents single stage process of circulation of a gas mixture of 100kPa initial pressure through CR and Pd-permeator until $DF=1 \cdot 10^5$ is achieved. Mode 2 is a double stage process with gas pressures of 100kPa in the stage I and 500kPa in the stage II which should provide $DF_I=10^3$ and $DF_{II}=100$. Mode 3 is a double stage ($DF_I=DF_{II}=500$) process with initial

pressures of 200kPa. In mode 3, impurities remaining after stage I are diluted by a factor ≈ 100 via injection of additional amounts of CH_4 and H_2O .

It is clear from Fig.4 that a long time is

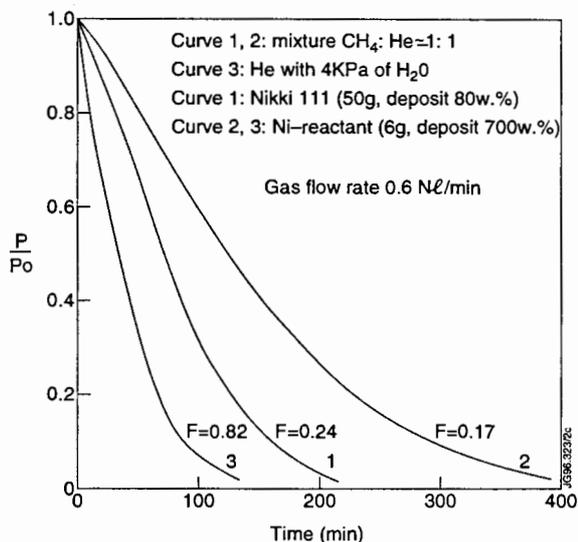


Fig.3. Kinetic curves of CH_4 and H_2O removal from helium in the integral loop (initial gas pressure is 100kPa, temperature of the Pd-permeator is 720K, temperature of CR is 870K)

required for impurities to be removed at low pressure when the driving forces of chemical reactions and hydrogen diffusion in the Pd-membrane are low. Gas compression at stage II (see mode 2 in Table 2) allows reducing duration of processing by $\approx 50\%$ but still requires operation with low residual pressure of methane and water. Injection of additional

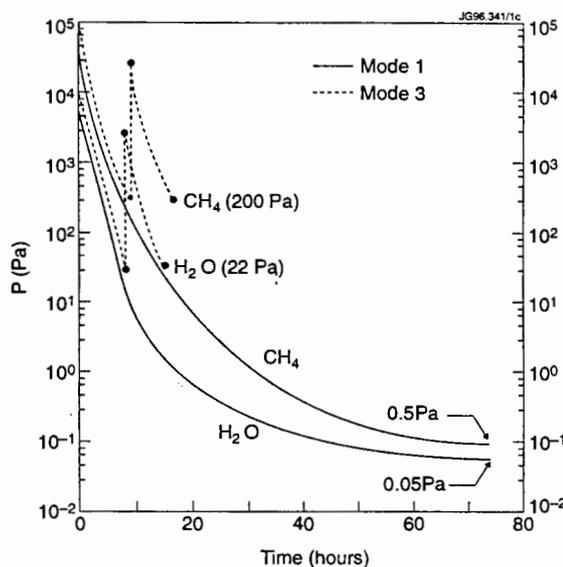


Fig. 4. Time dependence of CH_4 and H_2O pressures in helium for mode 1 and mode 3 of IP operation

Table 2. Comparison of different modes of impurity processing^{a)}.

Parameter ^{b)}	Chemical reactor			HITEX reactor
	Mode 1	Mode 2	Mode 3	
P_i^0 (Pa)	$5.55 \cdot 10^4$	$5.55 \cdot 10^4$ (278)	$1.1 \cdot 10^5$ ($2.2 \cdot 10^4$)	-
P_i^f (Pa)	0.55	55.5 (2.8)	222 (222)	-
Duration (hours)	>70	≈ 40	≤ 20	≈ 30
V_{Q2} (mol)	5	5	6	93.5

a) Impurities are removed from 100 moles of exhaust gas containing 4.5 mol. of methane, 0.5 mol of water and 4.5 mol of inert gases need to be detritiated;

b) P_i^0 , P_i^f are initial and final total pressures of methane and water (values for second stage are given in brackets); V_{Q2} is the amount of hydrogen isotopes generated during impurity processing as feed flow for isotope separation.

amounts of CH_4 and H_2O at the beginning of stage II (see mode 3 in Table 2) considerably decreases duration of processing and allows avoidance of operation at impurity low pressure. The amount of hydrogen isotopes to be isotopically separated after impurity processing is increased only by 20%. This value is much less than 1770% for a HITEX reactor. Mode 3 of chemical impurity

processing meets JET requirements and can be a choice for further development.

4. Tritium storage, supply and pumping by metal tritides

4.1 Investigation of JET uranium containers

Uranium is in general use as hydride-forming material for tritium handling in tritium facilities. Uranium containers are used in the JET tritium plant for hydrogen isotope storage, supply and pumping. Modern uranium containers have secondary (evacuated) and often tertiary (evacuated or filled with an inert gas) containments, software (SW) and hardwired (HW) interlock protection against overheating/overpressurizing; valves and interconnecting lines placed in an inert gas atmosphere box. JET uranium containers additionally have cooling systems with forced N₂ flow, large heat capacity (5.7 kJ/K), increased thermal conductivity and quick response on uranium temperature rise due to Ni-fins placed in uranium powder layers and brazed to the primary containment wall [13].

To transfer hydrogen isotopes, U-containers need to be heated to >770K to provide the required pressure of ≥120kPa. Tritium permeation loss from the primary containment has been measured (with D₂) as 3.8scc/h for a container operated at 770K with a gas pressure of 100kPa. This value corresponds to hydrogen permeability through SS316 of 4.2·10⁻⁶ scc/m·s·kPa^{1/2} and agrees well with known values. The pumping capacity (10bar·L) of the currently used getter pump (C500) filled with getter material ST707 allows >2000 hours of continuous pumping of hydrogen permeated from the primary containment of the U-container. The residual pressure in the secondary containment of the uranium container, continuously evacuated with a getter pump, rises from ≈10⁻⁵ Pa by ≈2 orders of magnitude when the container is operated at high temperature. The low gas conductance through the pump inlet filter at low gas pressure limits the hydrogen transport rate from the containment to the getter material. Therefore, the temperature of the secondary containment wall (in JET, the secondary containment is placed in an additional evacuated containment) rises to achieve a steady state temperature 50-70K below the temperature of the primary containment. Due to the high activation energy of hydrogen diffusion through stainless steel, tritium permeation loss from primary containment

should decrease rapidly with decreasing temperature. At 570K permeation rate would be ≈3% of the value measured for 770K.

The ability of uranium containers to operate as high vacuum hydrogen pumps is limited because of exothermicity of reaction of hydrogen sorption by uranium. Time dependencies of temperature of primary containment of a JET tritium storage container filled with 4.5 moles of U and continuously cooled down with N₂ flow (110NL/min) during D₂ sorption from a 100L tank is shown in Fig.5. During the container loading, the uranium temperature increases by 70K. About 1 hour is required to pump 95NL of D₂ by a U-container of 150NL capacity. To achieve a low temperature of uranium and high vacuum, the U-container needs to be continuously cooled down with N₂ flow for several hours. To evacuate hydrogen from a volume to low residual pressure and absorb it by uranium

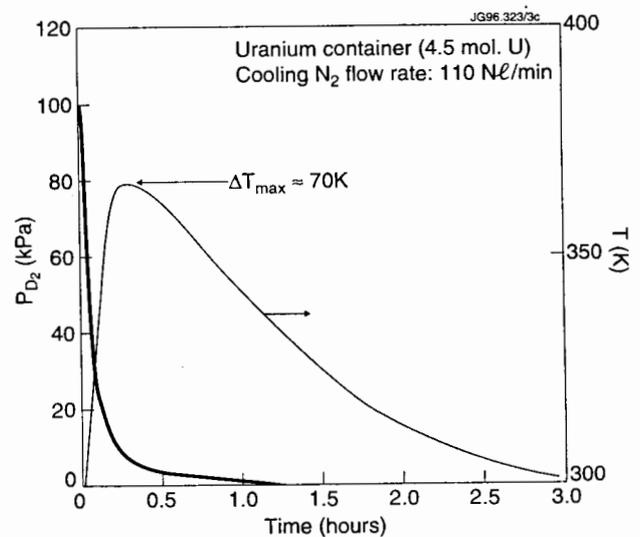


Fig. 5. Kinetic curve of D₂ absorption from 100L tank

more quickly, gas could be pumped and compressed at the U-container inlet via a molecular drag pump or a small (≈15m³/h) Normetex pump.

The vacuum isolation of the primary and secondary containments of JET U-containers allow them be maintained under quasi-adiabatic conditions for some period of time. This fact is used [14] for "in-situ"

measurement of container tritium inventory. The calibration curve, ie the time derivative of the container temperature (K/s) versus tritium loading (g), is a linear function $y=ax$ with $a=3.98 \cdot 10^{-5}$ K/s.g(T_2). Error of measurement decreases from $\pm 15\%$ to $\pm 0.3\%$, with tritium loading increasing from 1 to 30g of T_2 .

Hydride containers for tritium handling should satisfy a criteria of inherent safety in accidental situations. Experimental investigations [15,16,17] of U-containers in air ingress accidents have been carried out with containers of low capacity and have shown contradictory results giving either high temperature rise and even partial damage of container or negligible uranium temperature rise. Analysis of the conditions and results of the experimental tests allows the conclusions: fine uranium powder can start a reaction with air (both with O_2 and N_2) even at room temperature; hydrogen presence in a primary containment prevents a reaction with air due to hydrogen blanketing.

Safety of a JET uranium container in accidental situations has been evaluated [18] using results of thermogravimetric analysis of the reactions of uranium powder with air and N_2 . Modern U-containers, as mentioned above, are very well protected against any air ingress accident. U-containers of high capacity (with a large amount of uranium placed into a primary containment of small volume) designed for a fusion tritium plant could be considered as safe even for the very unlikely event of air ingress into evacuated container (no hydrogen blanketing can take place). Argon, existing in air with 1% concentration, will start blanketing uranium powder and moderate a reaction with incoming air after absorption <100 NL of air. The resulting highest temperature of uranium should not exceed the SW/HW protection temperature of the container. A high temperature rise could be expected only in an event of air flowing through uranium powder with low flow rate. Such an accident, as estimated [18], could lead to a local temperature rise up to 1400K. This value agrees well with experimental result [15].

The disadvantage of uranium as a tritium storage material is that even during normal operation of U-container its temperature can be higher than the temperature ($T_{N_2}^*$) sufficient to sustain a self-accelerating reaction with N_2 , which is estimated [18] to be ≈ 770 K. Analysis of the JET U-container in an N_2 ingress accident shows [18] that the uranium temperature could not exceed the SW/HW protection temperature plus 80K.

The advantage of uranium as a hydrogen isotope pumping material, ie its low equilibrium sorption pressure, is difficult to be utilized in a fusion tritium plant. On the other hand, its high reactivity even towards N_2 and the temperature of tritide thermodissociation need special precautions for hydride container operation. If other hydride-forming metals/alloys could show properties better than those of uranium, they could be considered as candidates to replace uranium in the tritium plant.

4.2 Hydride-forming alloys for tritium storage, supply and pumping

To be an alternative to uranium, a hydride-forming material should have the following desirable properties:

- $T_{des} < T_{N_2}^*$;
- stability against hydrogen-induced disproportionation at the highest operation temperature (T_{des});
- $T_{des} \leq 573$ K (to provide low tritium permeation loss from primary containment during tritium supply);
- suitable hydrogen sorption/desorption characteristics.

No hydride-forming metal meets to these requirements. The results of investigations of several hydride-forming intermetallic alloys are given briefly below. Kinetic curves of reaction of uranium and several alloys in powder form with air measured [19] by thermogravimetric analysis with a heating rate of 5K/min are presented in Fig.6. Experimental kinetic data have been fitted to different kinetic equations [20] of gas-solid

reactions. Probable mechanisms of the reactions have been estimated and kinetic parameters have been found. As an example, reaction of air with ZrCo powder (at $T > 673\text{K}$) can be described by an equation of nucleation with two-dimensional growth with activation energy of 15kJ/mol .

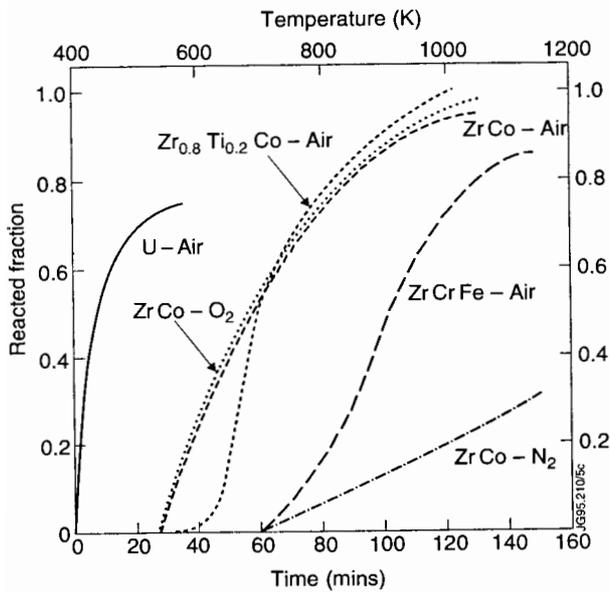


Fig. 6. Kinetic curves of uranium and alloys reactions with air, O_2 and N_2

Uranium clearly shows the highest reactivity towards air. ZrCo starts reacting with air at a temperature higher than that of uranium and LaNi_3Mn_2 but reaction occurs also with high rate. The difference in pyrophoricity of the alloys ZrCo, LaNi_3Mn_2 , $\text{Zr}_{0.8}\text{Ti}_{0.2}\text{Co}$ found at relatively low temperatures becomes negligible [19] at $T \geq 770\text{K}$ because of high values of activation energy (33kJ/mol for LaNi_3Mn_2 , 86kJ/mol for ZrCo at $T < 673\text{K}$). For all tested alloys (see Table 3) $T_{\text{N}_2}^*$ is found [18] to be $> T_{\text{des}}$.

The deterioration of the sorption properties of alloys via their disproportionation, i.e. intrinsic deconstruction of alloy into a metal hydride and a pure metal or a new alloy, leads to loss of the thermodynamic sorption/desorption properties and the ability to absorb hydrogen reversibly. Disproportionation of alloys occurs during their contact with hydrogen at a pressure higher than equilibrium pressure at an elevated temperature. Kinetic curves for disproportionation of several alloys are shown

in Fig.6 for a hydrogen pressure of 100kPa and a temperature of T_{des} . ZrCo does degrade within $\approx 1\text{hour}$, LaNi_3Mn_2 disproportionates more slowly but also at a visible rate. The stability of ZrCo towards disproportionation can be improved by partial substitution of Zr with Ti. The alloy ZrCrFe demonstrates no disproportionation during long term exposure at T_{des} . Completion of the reaction needs $> 100\text{hours}$ at temperature $> 300\text{K}$ higher than T_{des} .

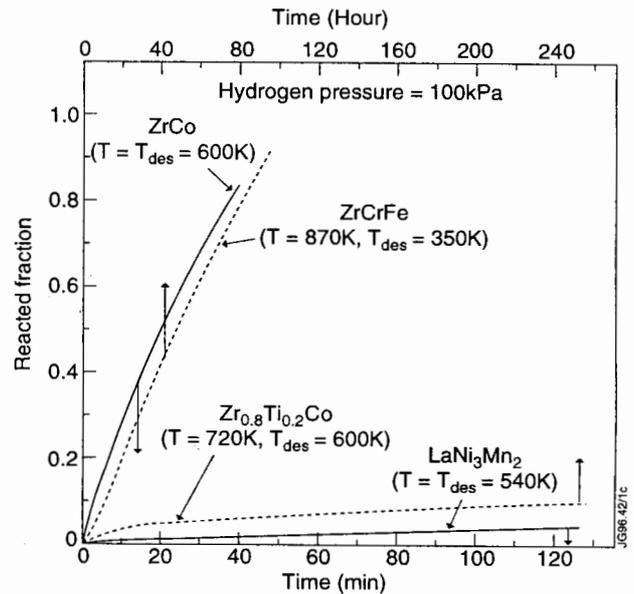


Fig. 7. Kinetic curves of intermetallic alloys disproportionation

4.3 An intercomparison of tritium storage materials

Analysis of several properties of some hydride-forming alloys given in Table 3 shows that the application of some alloys can increase the safety of hydride containers for tritium handling. Alloys, like ZrCrFe, $\text{Zr}_{0.8}\text{Ti}_{0.2}\text{Co}$ which show stability against disproportionation and relatively low T_{des} at pressures $\geq 100\text{kPa}$, could be considered as materials for tritium supply. Alloys, like $\text{Zr}_{0.8}\text{Ti}_{0.2}\text{Co}$, could be also used for tritium pumping.

5. Summary

The impurity detritiation sub-system of a tritium plant for a fusion reactor is currently a key point of fusion tritium technology.

Table 3. Comparison of hydride-forming metals and alloys for tritium storage, supply and pumping.

Metal/ Alloy	P _{sorp} (Pa)	T _{des} (K)	Reaction rate at T=T _{des}	
			of dispropor- tionation	with N ₂
Ti	6E-13	≈800	no	no
U	<1E-3	710	no	fast
LaNi ₃ Mn ₂	4	550	slow	no
ZrCo	4E-3	690	fast	no
Zr _{0.8} Ti _{0.2} Co	4	600	no	no
ZrCrFe	11E3	560	no	no

*) P_{sorp} is given for T=300K; T_{des} is given for P_{H2}=100kPa.

Selection and efficiency of operation of impurity processing techniques depends very much on the chemical composition of the fusion reactor exhaust gas which is an unknown variable. A technique based on the chemical conversion of gas species with the liberation of molecular hydrogen seems to be attractive because is not very dependent on impurities composition and does not require a powerful isotope separation system. Development of this technique for the JET tritium plant will be continued after torus exhaust gas composition is further evaluated during near future JET D-T experiments.

Hydride containers filled with uranium are used successfully within the JET tritium plant for hydrogen isotope storage, pumping and supply. However, their safety could be increased if uranium is replaced with hydride-forming alloys which have more suitable properties. The selection of an appropriate alloy would require optimization for every application and is not currently being considered.

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