5.7 Plasma Fueling and Pumping

5.7.1 Introduction

Tritium pellet injection will be utilized on FIRE for efficient tritium fueling and to optimize the density profile for high fusion power. Conventional pellet injectors coupled with a guide tube system to launch pellets into the plasma from the high field side, low field side, and vertically will be provided for fueling along with gas puffing for plasma edge density control. Recent experiments on ASDEX Upgrade and DIII-D indicate that these pellets will penetrate a sufficient distance into FIRE plasmas to provide peaked profiles. About 1 -2 x 10^{21} tritons/s are required to sustain the plasma density in FIRE which is a modest extrapolation of existing pellet injection technology. About 0.1 g of tritium must be injected during each 10 s pulse. The tritium and deuterium will be exhausted into the divertor. The double null divertor will have 16 cryogenic pumps located near the divertor chamber to provide the required high pumping speed of 375 torr-l/s The tritium from the regenerated cryopumps will be directed to gas holding tanks and fed into a cryogenic distillation system that will separate hydrogen isotopes and purify the tritium for return to the fueling system.

The plasma fueling system design for FIRE is based on previous designs for CIT, BPX and ITER as well as more recent developments and plasma physics results in the area of pellet launch from multiple locations relative to the magnetic axis. The goal is to produce a flexible fueling system that would require minimum change in the progression from FIRE to ITER or a fusion demonstration (DEMO) plant. In the past, tokamaks have generally used gas puffing for establishing and maintaining the plasma density. With this method, the sources of plasma particles are located at the plasma surface. There is general consensus, however, that gas puffing alone will not be sufficient to fuel the next generation of large, long-pulse fusion devices with thick, dense, scrape-off layers, and that core fueling, where the particle sources are located well inside the plasma edge, will be necessary.

A pellet fueling system (PFS) is provided for core fueling and a gas fueling system (GFS) for edge fueling. The FIRE fueling system provides plasma fueling from all sources (D, T, impurity gases) at a rate of 200 torr-liter/s for 20 s to support all fueling functions. The fuel rate to replace the D-T ions consumed by the fusion reaction is quite modest, about 2 torr-liter/s for a fusion power of 200 MW; the resulting burn fraction is thus only 1% of the steady-state fueling rate. Such low burn fractions result in large vacuum pumping and fuel processing systems with associated tritium inventories and were not anticipated in early (1,2) and even more recent (3) fusion power plant assessments, which had burn fractions in the 10-40 % range. The low burn fraction is only partially due to the finite fueling efficiency (see next section). The fueling system (4,5) must also maintain the required plasma density (near the empirical Greenwald density limit), establish a density gradient for plasma particle (especially helium ash) flow to the edge, and also supply hydrogenic edge fueling for increased scrape-off layer flow for optimum divertor operation. Still another function is to inject impurity gases at lower flow rates (25 torr-l/s or less) for divertor plasma radiative cooling and wall conditioning. Finally, the plasma fueling system provides for plasma discharge termination on demand via massive gas puffing or injection of cryogenic mass via pellets or liquid jets. A concept called isotopic fueling (6) can be used to improve the tritium burn fraction above the nominal 1% level described above by frugal use of tritium fuel to those functions only related to the fusion burn and using deuterium for edge fueling. This can reduce in-vessel tritium inventories and the required tritium-breeding ratio for fusion reactors.

The tritium inventory inside the FIRE vacuum vessel is a major consideration. Due to the large retention of tritium observed in carbon plasma facing components and in co-deposited carbon layers on TFTR and JET, carbon PFCs will not be allowed in FIRE. The divertor plates will be tungsten and the first wall protection will use Be tiles.

5.7.2 Fueling Efficiency

The fueling efficiency of tokamaks has been studied since the early 1980's. For gas fueling, the determination of fueling efficiency of short pulse tokamaks has been difficult to quantify because of an outgassing source of hydrogenic fuel from the plasma facing components that can be of the same magnitude as the external gas fueling. Pellet fueling is easier to quantify in terms of fueling efficiency due to the rapid deposition of the fuel (100's of µs) and its deposition beyond the last closed flux surface, which avoids most atomic physics complications in fuel transport to the plasma. In contemporary tokamaks, fueling provides the required density level for a particular plasma experiment. There is incentive to maximize the tritium plasma fueling efficiency due to the cost and safety implications of a large tritium throughput and the complexity of reprocessing large torus exhaust gas loads (6). Fueling efficiency of gas and pellet injection are summarized in Table 5.7.1 and Figure 5.7.1

Device	Gas Fueling Efficiency (%)	Pellet Fueling Efficiency (%)	Remarks
ASDEX	20	30-100	high density
PDX	10-15		high density
Tore Supra	1	30-100	ergodic divertor for gas fuelling
JET	2-10	20-90	active divertor
TFTR	15		low density DT
ASDEX-U		8-50	
DIII-D	10	40-100	active divertor

Table 5.7.1. Tokamak fueling efficiency.

(7,8). Generally, in diverted discharges of the larger tokamaks, gas fueling efficiency is in the range 1-10 % and pellet fueling efficiency is an order of magnitude larger. Recent results from ASDEX-Upgrade (9) are also shown in Figure 5.7.1 which compares the penetration and fueling

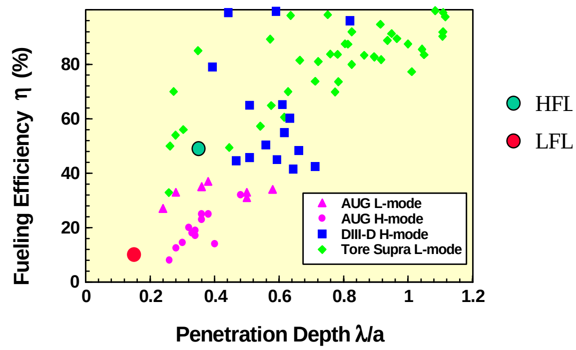


Figure 5.7.1. Pellet fueling efficiency for several experiments. *(Also shown are recent results from high field launch (HFL) and low field launch (LFL) on ASDEX Upgrade.)*

efficiency of pellets launched into the same plasma conditions from the high magnetic field side and low magnetic field side; improvements in pellet penetration and fueling efficiency for high field launch are substantial.

Deuterium pellet injection from four different poloidal locations has been used in experiments on the DIII-D tokamak (10, 11, 12) to investigate several aspects of plasma confinement and density control (see Figure 5.7.2). Pellets can be injected in four locations: outside midplane, vertically inside the major radius, inside launch at ~ 45 degree angle and inside midplane. Pellets injected from the outer horizontal midplane (low field side) show a large discrepancy in the measured fueling efficiency and mass deposition profiles from pellet ablation theory, while the penetration depth compares favorably with theory. An apparent outward displacement of the deposited pellet mass is observed and hypothesized to occur from ∇B and curvature induced drift effects. Injection of pellets inside the magnetic axis from a vertical port and inner wall ports using curved guide tubes has been employed on DIII-D to investigate these effects.

The resulting density profiles show pellet mass deposition well inside the expected penetration

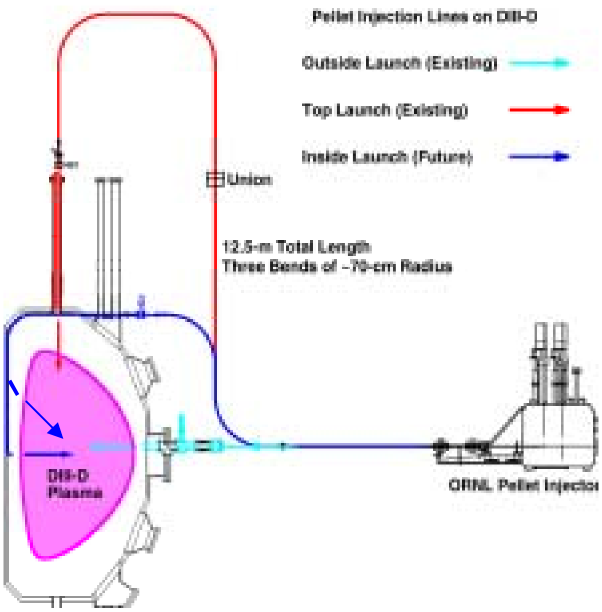


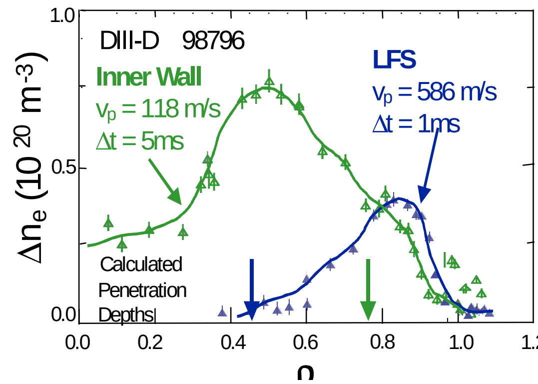
Figure 5.7.2. Pellet launch locations on DIII-D.

radius, suggesting that a drift of the pellet ablatant is occurring toward the low field side (LFS) edge of the plasma (Figure 5.7.3). The formation of highly peaked density profiles with pellets injected from the high field side is possible at higher heating power than is possible from pellets injected from the low field side.

On FIRE, pellet injection will be possible from the outside midplane, vertically and from the inside lower quadrant aimed towards the plasma center. This will be accomplished by three sets of guidetubes.

Recently, there has been interest in repetitive impurity pellets or impurity gas puffing to foster enhanced radiation in the outer plasma and divertor regions and large ("killer") pellets for a controlled, preemptive plasma shutdown in anticipation of a major disruption or vertical displacement event (VDE). These systems typically operate at room temperature or higher cryogenic temperatures, but require similar technology for pellet feed and acceleration as are used on H/D/T pellet fueling systems. Major issues for impurity pellet injection include development of pellet production and feed hardware optimized for the pellet material (i.e.

Figure 5.7.**3**. Plasma density increase from a pellet launched from the inner wall (high field side) launch compared with launch from the outer wall (low field side).



lithium, carbon, nitrogen, argon) and, for killer pellet injectors, high reliability for a single large pellet or liquid jet on demand. Impurity pellet injection systems (typically small lithium or carbon pellets) have been developed for wall conditioning and plasma diagnostics.

5.7.3 FIRE Fueling System Overview

The FIRE fueling system will use a combination of gas puffing and pellet injection to achieve and maintain burning plasmas. This combination will provide a flexible fueling source with D-T pellets penetrating beyond the separatrix to sustain the burning fusion plasma and deuterium-rich gas fueling the edge region to meet divertor requirements in a process called isotopic fueling (6). The isotopic fueling concept was developed to allow independent control of the plasma deuterium and tritium density profiles which can lead to reduced (by factors of 2-4) tritium inventory in plasma facing components. The higher tritium burn fraction allows a significant reduction in tritium gas flows into and out of the vacuum vessel and, for fusion reactors, implies lower required tritium breeding ratios. The fueling system includes; a conventional gas puffing system, using all-metal electromagnetic dosing valves, (four toroidal stations at two poloidal locations at each divertor level), and a pellet injection system.

The FIRE pellet fueling system (PFS) design includes a long pulse pneumatic pellet injector capable of injecting D-T or tritium. It will be a repeating pneumatic injector using an extruder-based hydrogenic feed system. It will be configured to inject pellets using propellant gas for pellet acceleration (up to 1.5 km/s pellets) or a mechanical punch accelerator (up to 100 m/s for pellet injection into curved guidetubes for vertical or high field side launch) or a combination of these two drivers. The mechanical punch operating alone or with a small amount of propellant gas would reduce considerably the need for differential pumping of the pellet injection line and the reprocessing requirements for propellant gas. The PFS comprises a pneumatic pellet injector with three separate pellet extruders/guns, installed in a containment area in the basement below the torus. The PFS and GFS manifolds are also located in a basement area below the FIRE torus. Pellet injection will be possible via curved guidetubes from the outside midplane, vertically and from the inside lower quadrant aimed towards the plasma center. This will be accomplished by three sets of guidetubes. The pellets will be injected to the high magnetic field side of the machine through a curved flight tube routed through the lower divertor region. The hydrogenic feed for the injector is provided by a conventional linear piston hydrogen extruder (sized for a 20 s supply of pellets) or by a continuously rotating screw extruder. Deuterium and tritium pellets up to 10 mm in size have been extruded at rates up to 0.26 grams/sec (for short pulses only); this pellet size and feed rate is sufficient for fueling fusion reactors at the gigawatt power level. Table 5.7.2 below shows preliminary parameters for the FIRE hydrogenic fueling system.

Parameter	Gas Fueling System	Pellet Fueling System	Remarks
Design fueling rate	200 torr-l/s for 20 s	200 torr-l/s for 20 s	Torus pumping capacity is 375 torr-l/s
Operational fuel rate	100-175 torr-l/s	100-25 torr-l/s	Isotopic fueling
Normal fuel isotope	D (95-99%) T,H (5-1%)	T (40-99 %) D(60-1%)	D-rich in edge, T-rich in core
Impurity fuel rate	25 torr-l/s	TBD (prefer gas for impurity injection)	25 torr-l/s reduces DT fuel rate due to fixed pumping capacity
Impurity species	Ne, Ar, N_2, other?	TBD	TBD
Rapid shutdown system	Massive gas puff	“killer” pellet or liquid D jet	For disruption/VDE mitigation
Pellet sizes (cyl. diameter)	N/A	3, 4, 4 mm	3 mm for density rampup, 4 mm for flat-top

Table 5.7.2. Preliminary FIRE fueling system parameters.

5.7.4 FIRE Fueling System R&D

The screw extruder concept has been demonstrated by a Russian Federation prototype system which ran for 1-hour pulses using hydrogen feed gas producing ~2 mm extrudant.

This needs to be extrapolated to deuterium and tritium feed and larger pellet sizes using this technology or variants such as gear or double-screw extruders. ITER-scale (10-mm) pure tritium and D-T pellets have been extruded with a piston-type linear extruder and accelerated to about 1 km/s (see Figure 5.7.4) in the Tritium-Proof-of-Principle Phase II (TPOP-II) experiment at the Tritium Systems Test Assembly (13)

The technology to deliver intact pellets at the highest possible speeds around curved surfaces (guide tubes) is under development (11, 12). This is a complex issue and depends on the pellet speed and temperature (strength) as well as the guide-tube radius of curvature, its diameter relative to the pellet size, and its cross-sectional shape. The speed dependence of penetration for high field side or vertical launched pellets is not known.

5.7.5 FIRE Pumping System

The current baseline design is a set of refrigerated duct D-T cryocondensation/diffusion pumps backed by turbo/drag pumps. This system is designed to pump in both the free-molecular and viscous flow regimes. Water is pumped on the inside diameter of the 160 mm diameter by 1 meter long 30 K entrance duct which connects the divertor to the cryocondensation pump. Other impurity gases and hydrogen are pumped by cryocondensation on a 1/2" O.D. x 0.035 wall stainless steel tubing coil refrigerated by liquid helium. The 2 torr-l/s helium gas produced by the D-T fusion reaction is compressed by viscous drag in the entrance duct by a factor of up to 100. The compressed helium gas is carried from the cryopump to a turbo/drag pump located outside the biological shield through the divertor duct. The design D-T throughput is 200 torr-liter/s for the 20 s pulse length. The partial pressures prior to a discharge are 10^{-7} torr for fuel gases (H, D, T) and 10^{-9} torr for impurities. There will be a total of 16 cryopumps with 8 each top and bottom (at alternate divertor ports) located 1 meter into the pump duct from the double-null divertor. The duct behind the cryopump will be constructed with transverse optically opaque shielding baffles which will allow 200 l/s helium gas conductance per port to the turbo/drag pumps located outside the biological shield. There are no moving parts inside the torus.

A layout of the cryopumps is shown in Figure 5.7.5. The cryopumps are designed to have a low mass and active helium gas cooling. Between shots the helium flow will be stopped to allow the pumps to regenerate into the compound turbo/drag pumps. This will limit the tritium

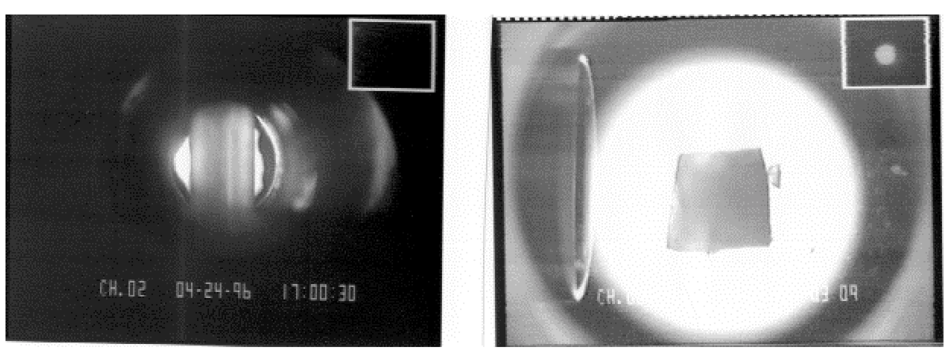


Figure 5.7.4. Pure tritium extrusion and pellet.

contained on the cryopumps to less than 1 gram for a 20 sec. discharge. Gas will be returned to the tritium system where it will be processed through an impurity removal step and a cryogenic distillation system that will separate the hydrogen isotopes and purify the deuterium and tritium for return to the fueling system.

The cryogenic cooling requirement for the 16 pumps for the design pumping rate of 375 torr-l/s and the nuclear heating loading which is estimated at 0.03 watt/cm^3 at the proposed cryopump location 1 meter from the divertor is 3 watts per pump. The liquid helium cooling rate required during a shot is 64 l/h for the 16 pumps.

The maximum divertor pressure during the pulse is ~0.02 torr. At this pressure and the design duct size the Knudsen number is 0.01 so the gas transport is in the viscous flow regime dominated by gas-gas interaction. In this case the minority gas species such as helium and impurities will be carried by viscous drag to the cryopump. This effect can be used to achieve a helium compression of 100 in the inlet duct so that the required helium pumping speed can be reduced and still maintain a high effective helium pumping speed at the divertor.

During the tokamak discharge the effective pumping speed for 375 torr-l/s flow is 2,000 l/s per duct (32,000 l/s total) at the divertor for D-T, He, and impurities. After the shot the pumps will be warmed and regenerated. The 4,000 torr-l of D-T pumped during the shot will raise the 18 m^3 torus chamber to 0.2 torr. The pumping time constant for the 16 turbo-drag pumps with 3,200 l/s combined speed will be 6 seconds. Between discharges, with the cryopumps warmed to 100K, the turbo/drag pump set will have a speed of 3,200 l/s for all gases. Prior to the discharge, with the pumps cold, in the free molecular flow regime, the pump set will have an effective speed of 16,000 l/s for D_2, 6,400 l/s for air, 46,000 l/s for water vapor, and 3,200 l/s for helium.

The sixteen evacuation locations will be grouped in four sets of four. Each set will have its own cryogenic control system. Liquid helium will flow in series through the four cryocondensation pumps and will go through a heat exchanger to completely flash it before it is sent in parallel through the four cooled ducts. The four turbo/drag pumps in the group will be backed by a single 3.3 l/s scroll pump that is backed with a metal diaphragm pump.

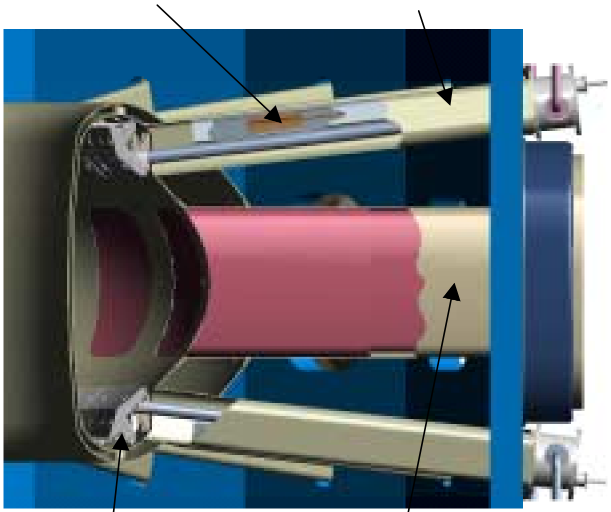


Figure 5.7.5. Elevation view of FIRE torus showing divertor duct and cryopump.

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5.8 Tritium System Requirements

5.8.1 Introduction

The tritium system is a key system for FIRE operations, as well as providing development information and operating experience for the fusion program. The development of a safe, low-inventory tritium system is an important project goal.

5.8.2 Tritium Injection Requirements

Pellet injection will be the primary plasma system for fueling the core of the FIRE plasma utilizing both high speed pellets and guided lower speed pellets. Gas injection systems will also be provided for edge fueling. The total number of tritons in the nominal FIRE plasma is:

$N_e = \langle n_e \rangle V_p /2 \sim 2.25 \times 10^{20}\ m^{-3} \times 18\ m^3 \sim 5 \times 10^{21}$ tritons

The particle confinement in tokamaks is described by $D \sim \chi$, or $\tau_p \sim \tau_E$ which translates into $\tau_p \sim 0.5$ to 0.8 s in FIRE. A fueling rate of ~ 0.5 - 1 x 10^{22} particles/s would be required to sustain the density with zero recycling. The standard assumption for FIRE and ITER-RC is that $\tau_{He} \sim 5\ \tau_E$ which suggests an 80% recycling of helium. We make the additional reasonable assumption that the same recycling applies to the hydrogenic species. Therefore, a net rate ~ 0.1 - 0.2 x 10^22 particles/sec would be required to sustain the nominal FIRE plasma.

In present experiments with outer mid-plane pellet fueling, the efficiency is low ~20%. FIRE will be employing vertical launch of high-speed pellets aimed inside the magnetic axis or slower pellets guided by tubes to near the inside mid-plane. The injection geometry will be updated, as more information becomes available from ongoing experiments. A pellet fueling efficiency of 50% is assumed for FIRE. The gross tritium fueling rate for the plasma core is then ~ 0.2 - 0.4 x 10^{22} particles/s.

5.8.3 Requirements for Potential Pulse Sequences

The total number of injected tritons required for various scenarios involving 10 second long pulses is:

2 - 4 x 10^{22} T/pulse ; 0.8 - 1.6 kCi (~0.1g) / pulse; 10 s pulse

2 - 4 x 10^{23} T/day ; 8 - 16 kCi (~1g)/day; 10 pulses/day

1 - 2 x 10^{24} T/week; 40 - 80 kCi(~5g)/week; 50 pulses/week

(where 2.09 x 10^{19} T atoms = 1 Ci, 10^4 Ci = 1 g, pulses are 10 seconds long)

The total number of DT pulses in FIRE is limited to < 5TJ of fusion energy, or 2,500 pulses at 200 MW for 10 s, or a tritium fueling throughput of 2 - 4 MCi. There will also be many partial power pulses that will consume tritium while not producing optimal fusion power, therefore the lifetime throughput of tritium is assumed to be increased by ~ 5 to a total throughput of 10 - 20 MCi. Assuming that this program takes place over 5 years would require a tritium throughput capability of 2 - 4 MCi/year.

The fractional tritium burn-up of ~ 5% does not affect these estimates significantly.

5.8.4 Tritium Retention and Inventory

The provisional limit for the tritium site inventory has been set at ≤30 g (~0.3 MCi). According to DOE STD 1027, FIRE would be classified as a Low Hazard Nuclear Facility. Similar to the TFTR tritium management strategy, the maximum tritium in any on-site tritium system is set at 15 g, 1/2 the site limit. The D-T experiments on TFTR and JET have shown that the use of carbon plasma facing components produced an effective tritium retention rate of ~ 40%. Assuming the annual tritium throughput of 2 - 4 MiCi, this level of retention would cause the FIRE system limit to be exceeded in 1 to 2 months. Therefore, the use of carbon will not be allowed in the FIRE vacuum vessel. The initial materials for plasma facing components and divertor plates will be Be and W.

Cryogenic pumps will be installed in the behind the divertors of FIRE to provide adequate pumping during the pulse. Essentially all of the injected tritium will end up on the cryopumps. The schedule for regenerating the cryopumps will be determined to maintain the tritium inventory < 30g. The tritium inventory for several regeneration schedules is:

1. weekly regeneration: < 5g of tritium on the cryopumps would be transferred to the tritium handling system. The tritium separation systems described below would be able to easily separate the tritium from the deuterium and other exhaust gases in < 2 days, so that tritium could be ready for experiments the following week. Need to estimate the number of deuterium only shots in a run sequence to estimate the total number of torr-liters of gas on the cryopumps to see if this is a reasonable sequence.

2. daily regeneration: < 1g inventory on the pumps, probably not worth the hassle of the regeneration procedure.

3. monthly regeneration: < 20g of tritium on the cryopumps if running continuously with tritium. This level is also expected to satisfy the explosive limits.

A monthly regeneration of the divertor cryopumps would fit naturally with the anticipated experimental schedule.

5.8.5 Tritium Systems for FIRE

The tritium systems will be similar to those used successfully at TFTR, and will include Tritium Storage and Delivery, plasma exhaust cleanup, tritium purification system (for reprocessing the on-site inventory), appropriate room air cleanup systems, tritium exhaust gas processing systems, and tritium monitoring for process control and personnel protection. The block diagram for the tritium system is shown in Fig 5.8.5-1.

The FIRE tritium delivery system will be capable of supplying tritium with a purity > 98%. Tritium will be received from a DOE supplier in hydride transport vessels (HTVs) in quantities up to 25 grams. Tritium inventory will be loaded into the tritium storage and delivery system (TSDS) and will be available upon demand (within 6 hours of when required). The FIRE TSDS will be capable of supplying quantities of tritium up to 3 kCi per pulse via direct gas injection. This capability could be upgraded for the long pulse (~ 40s) pulses in the advanced tokamak phase.

FIRE exhaust gas will be collected in a plasma exhaust tank where it will be stored until processed by the on-site tritium purification system (cryogenic distillation). On-site tritium processing will separate non hydrogen isotopes from the plasma exhaust effluent and cryogenically separate tritium from deuterium and protium, thus producing tritium with a purity of > 98 % purity. Plasma exhaust processing will require ~ 24 hours to be recycled back to the tritium storage and delivery system.

The on-site tritium purification system for FIRE will have a resident tritium inventory of ~ 10 grams of tritium with a throughput of (up to) 50 kCi (5 grams) /

day. ITER had planned to reprocess 164 g of tritium during the 40 minute cycle period for an ITER pulse[1].

Tritium residual gases (in glove boxes and in other small volumes) will be processed, oxidized, and deposited on disposal molecular sieve beds for disposal at an off-site facility.

5.8.6 Options to the Tritium Inventory

The tritium inventory has been set at 30g (~0.3 MCi), to allow sufficient operational flexibility without introducing additional restrictions. However, there is the potential for reducing the inventory to even lower levels. If a tritium reprocessing system is used which is able to recycle the working tritium on a daily basis, then the daily working inventory is = 20 kCi (2g).

As noted above, ITER was planning on reprocessing tritium at the rate of >4 g/minute. If FIRE had a system capable of processing 1g/120 minutes, then the working inventory could be reduced by an order of magnitude to 2kCi(0.2g). The main contributions to the inventory would now be in residual holdup in various systems including the vacuum vessel. There should be a follow-up study to look at the minimum tritium inventory case.

[1] D. K. Murdoch, "Tritium Inventory Issues for Future Reactors"; Choices, Parameters, Limits. Proc., SOFT 1998

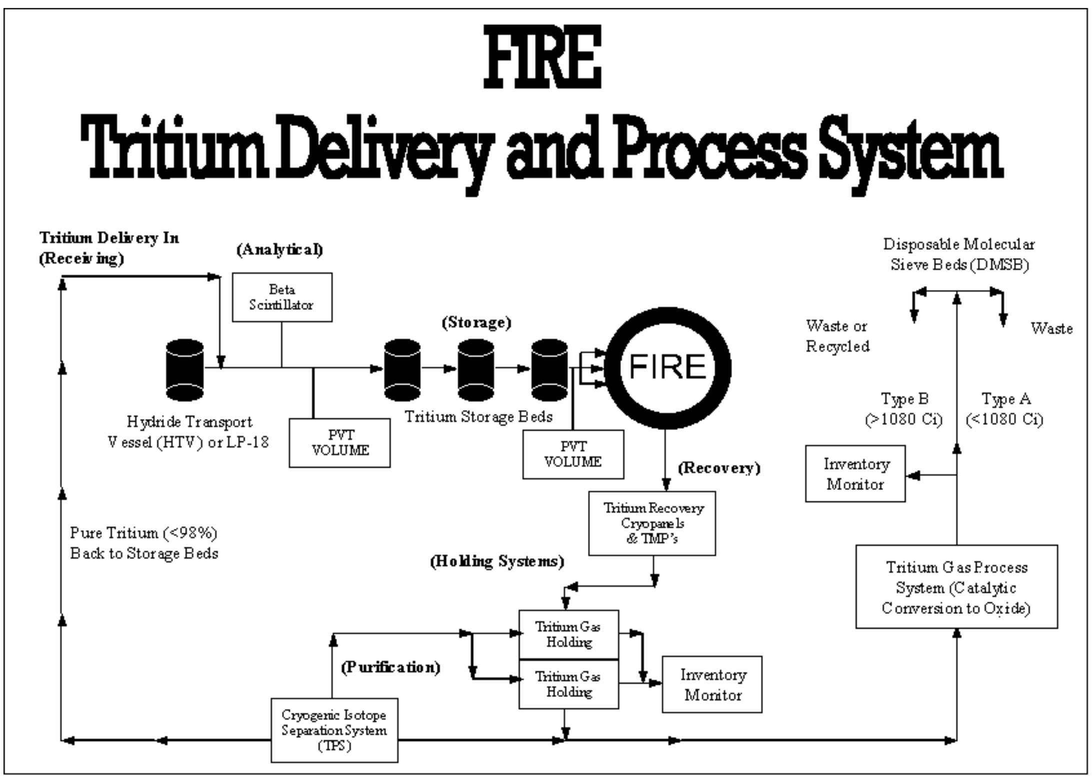

Fig. 5.8.5-1 Tritium System Piping and Instrumentation Diagram (P&ID) for FIRE