4) How would the ST program address and resolve the following questions?

c. Fusion nuclear operations and tritium: Seems that the goal of a test facility is to develop solutions to these issues. Won't ITER experience produce sufficient knowledge base to operate this facility, including remote handling? Won't such a device need a lot of tritium to conduct meaningful tests? [co-authors: Tom Burgess, Alice Ying/Mohamed Abdou]

One of the research goals of any fusion test facility is to increase machine operational availability. This is an area of research where experience transferred from facility to facility (such as JET to DIIID, and vice-versa.) An availability assessment requires a complete description of the facility components, data on their failure rates and repair times, and scheduled maintenance. A high availability (or duty factor) can be reached if high reliability (or low failure rate) of components within their lifetime span can be assured or a relatively fast replacement performed through a flexible design with easy access. Since the reliability of the major enabling component of a foreseen ST will not be known, all chamber systems, including neutral beam injectors, RF system, divertors, and test blanket modules must allow relatively straightforward replacement via remote handling to minimize the mean-time-to-repair/replace (MTTR) to achieve a high availability [1]. ITER design, construction and operation experience will provide the most up-to-date knowledge base on remote handling of interest to any activated toroidal facilities.

Many remote handling techniques that are being developed for ITER can be used in the ST facility. These includes the transfer cask concept for handling the activated components and test blanket modules (TBM) in ITER [2], in which the mechanical attachments for a fast maintenance scheme as well as remote handling tools such as orbital welding tools, in-bore tools for pipe connections, electrically grounding and cable connections are being developed in Parties [3]. R&Ds are also being conducted on the radiation tolerance assessment of fiber-optics, electronic components to ensure the periodic maintenance operations in a severe nuclear environment, exposing operating tools to estimated total doses at the MGy level and temperatures ranging from 50 to 200°C [4]. These remote handling tools and capability as well as the initial stage of ITER operational experience benefits the ST by providing greater confidence of achieving a higher availability from the start. With these experiences supplemented by additional specific remote handling R&D for the ST [5], a goal average availability of 0.3 may be envisioned with continued improved operational experiences.

However, fusion nuclear operation goes beyond developing remote handling maintenance capability. An objective of such an operation is to develop deep scientific understanding of the failure-initiating mechanisms for in-vessel components such as blankets in an unprecedented fusion complex environment. Such understanding is necessary to inspire the development of innovative methods to minimize failure rates and mitigate failure effects and increase component reliability. Previous analyses have shown that to achieve a DEMO reactor availability goal of ~50%, all reactor major components require reliability (much) greater than 80% [1]. This is particularly ambitious for blanket components, where tremendous operational experiences needed gear towards identification of failure modes and effects, aggressive iterative design/test/fix programs aimed at improving reliability, and the obtainment of failure rate data sufficient to predict MTBF. The US ST fusion program of the ITER era can certainly contribute R&D to establish the reliability knowledge base for an attractive fusion energy source.

Developing, testing and fixing a component concept to an accumulated fluence level of about 6 MW-yr/m^2 is considered essential before the start construction of DEMO [1]. This fluence level corresponds to roughly 40% of the goal lifetime fluence of in-vessel breeding blanket components, and allows its DEMO goal reliability to be demonstrated to a sufficient confidence level prior to designing, constructing and operating full breeding blankets in DEMO. Although it is desired to achieve this goal fluence as soon as possible, it would be adequate for the ST goal of the ITER era achieving a total fluence capability half-of the targeted goal of 3 MW-yr/m^2. This would, however, be more than an order of magnitude progress beyond the accumulated duty factor of major magnetic fusion experiments to date and therefore a reasonable step towards the effective component testing conditions.

Such a fluence level of 3 MW-yr/m^2 can be achieved by operating a ST device with an average neutron wall load of 1 MW/m^2 and fusion power of about 72 MW for an average availability of 0.3 in a time scale of 10 years [6]. This also calls for beginning this operation at the start of the 2nd phase of ITER operation. Preceding this operation it could be a 3-year period of reduced power operation of 36 MW, which attributes to a neutron wall load of about 0.5 MW/m^2 and produces a substantial divertor heat flux, to develop divertor solutions for high heat flux removal capability and mechanisms of radiating the heat load to the first wall area to achieve a similar divertor heat flux level when operated at a higher fusion power.

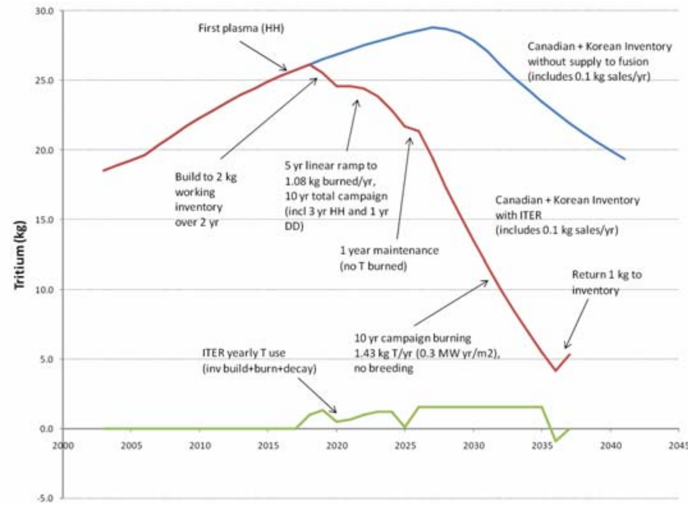

Figure 1 Projections of Canadian + Korean tritium supply and consumption using ITER current schedule. (From Scott Willms [March 2007]).

The amount of tritium required to support this scenario is estimated to be ~ 16.9 kg including a 20% contingency, which accounts for inventory, losses, and a low fractional burn-up. Since it is assumed that the ITER second phase will be conducted and would be given "priority" on the world tritium supply, the ITER second phase will essentially require most of the world tritium supply, except ~ 5 kg as shown in Figure 1 [7]. This implies that the ST goal of the ITER era would require its own breeding blankets and blanket test modules for a substantial portion (80-90%) of tritium production (Not only the option of external supply of tritium using fission reactors is costly and puts fusion's credibility into question, but also 2 kg per year reaches the critical edge of external tritium supply.) The issue of tritium consummation becomes even greater for a larger device with higher fusion power.

Capturing the neutrons from the burning plasma in lithium-bearing breeding materials to regenerate the tritium necessary to fuel the plasma involves many unresolved scientific topics. Data for processes as basic as tritium mass transport, chemical kinetics, and solubility presently have large uncertainties, and the fundamental physics of tritium exchange at liquid metal/metal and liquid metal/gaseous interfaces is not known. Tritium produced in a high-temperature breeder has significant mobility, which makes tritium control, accountancy, and prevention of release to the environment challenging issues. R&D utilizing existing fission and/or out-of-pile facilities to obtain a fundamental database; gain understanding of basic, separate- and some multiple-effect phenomena occurred in both base breeding and advanced test blankets should be as extensive as necessary in order to balance the costs and risks of complex tests in the ST fusion environment. Effective utilization of the ITER device for TBM experiments is also essential to reducing the risk by eliminating earlier life of failure. These experiments are the vehicle for discovering whether the scientific understanding gained from various separate and multiple effects tests is valid in the integrated fusion environment, and validating component level fabrication techniques.

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