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ABSTRACT

The fusion technology development program for tritium in the U.S. is centered around the Tritium Systems Test Assembly (TSTA) at Los Alamos National Laboratory. Objectives of this project are to develop and demonstrate the fuel cycle for processing the reactor exhaust gas (unburned deuterium and tritium plus impurities), and the necessary personnel and environmental protection systems for the next generation of fusion devices. The TSTA is a full-scale system for an INTOR/ITER sized machine. That is, TSTA has the capacity to process tritium in a closed loop mode at the rate of 1 kg per day, requiring a tritium inventory of about 100 g. The TSTA program also interacts with all other tritium-related fusion technology programs in the U.S. and all major programs abroad. This report is a summary of the results and interactions of the TSTA program since a previous summary was published[1] and an overview of related tritium programs.

INTEGRATED LOOP OPERATIONS

In September 1986, a major five-day, round-the-clock operation of the TSTA integrated processing loop was carried out. The goals of the run were achieved as planned, and included the following accomplishments:

- integrated flow processing through all process systems except the compound cryopumps;
- the removal of impurities (up to 7% nitrogen) by the fuel cleanup system (FCU), though without continuous impurity addition or on-line regeneration and recovery of the captured impurities;
- the addition of 20 g of tritium to the flow loop, raising the in-process inventory to 50 g;
- the verification of improved flow control measures added to the loop;
- the production and analysis of tritium gas of 99.93% purity in the cryogenic distillation columns of the isotope separation system (ISS); and
- the training and use of personnel in different operating assignments to broaden staff experience and versatility.

The September run was followed by a similar five-day, round-the-clock operation in December 1986. Goals and accomplishments of this run were the following:

- further development and improvement of flow control in the loop;
- elimination of unwanted interactions between the isotope separation system and effluent gas detritiation system;
- measurement of long time constants (several hours) for the isotope separation system to reach steady state after control changes are made; and
- changing from our previous two 12-hour shifts to three 8-hour shifts, with the night shift being a two-man shift making minimal process changes.

The next major operation of the integrated process loop occurred in June 1987. The goals of this run were the following:

- to increase the in-process inventory of tritium to 100 g;
- to demonstrate the successful removal of helium-3 from tritium decay by two techniques--gettering out the tritium on uranium beds before dumping the helium, and stripping out the tritium in the distillation columns before dumping the helium;
- to provide training for personnel including two new operators;
- to produce and analyze high purity tritium (>99.9%) in the ISS; and
- to verify long-term, continuous addition of impurities (about 1% N_2 and 0.1% CH_4) and their successful removal by the cold (77 K) molecular sieve adsorption beds in the fuel cleanup system.

The inventory was raised to 90 g and all goals except the last two were achieved before operations were concluded prematurely by a broken shaft on the commercial cryogenic refrigerator used to provide cooling to the cryogenic distillation columns. An orderly shutdown under off-normal conditions, without doses to personnel or releases to the environment, was achieved following the unexpected failure in off-the-shelf technology unrelated to tritium handling.

Following a repair and upgrade of the refrigerator by a factory representative, round-the-clock operations were resumed in July, for a period of 5 days, to add to the 4 days before the refrigerator failure. The tritium inventory was increased to 102 g, and the last two run goals were achieved at this time. In particular, a tritium sample of 99.98% purity was prepared in the ISS.

Impurities were added to the fuel stream continuously for 52 hours and removed by the FCU to below the limit of detection of Raman spectroscopy (low ppm levels). The impurities added were a mixture of N_2/CH_4 in 9/1 ratio and comprising 1% of the D-T fuel stream flow. The effectiveness of impurity removal was further confirmed by the lack of any plugging in the cryogenic still that continuously processed the outlet flow from the FCU. On-line regeneration and processing of the captured impurities was not a part of the July round-the-cock operation.

The most recent operation of the integrated processing loop took place in February 1988. The chief new feature of this run was the regeneration of captured impurities from the molecular sieve adsorption beds and the processing of this gas through the steps of catalytic oxidation, trapping the resulting water in a freezer at 160 k, and eventual reaction of the recovered water with hot (750 K) uranium to recover D-T for reuse.

PERFORMANCE TESTS AND MAINTENANCE OF THE ISS

In addition to the integrated processing tests described above, a number of tests have been made using only one process at a time (with the necessary auxiliary systems). Individual columns of the four-column isotope separation system were operated in 5-day, round-the-clock runs in October and December of 1987. The purpose of these tests was to measure fundamental design parameters, such as liquid holdup in columns and the height equivalent to a theoretical plate (HETP). These results are separately reported at this conference.[2]

Repairs were made on instrumentation external to the columns, but inside the vacuum jacket. This was done successfully by lowering the vacuum jacket. Contamination inside the jacket was negligible.

TRITIUM TESTS OF NEW COMPONENTS

Four new components of potential benefit to the fusion program were evaluated for performance and tritium compatibility in individual testing. The components, and organizations with whom TSTA collaborated on each, were:

1. a commercial zironium-iron getter material for detritiation of glovebox atmosphere (Ontario Hydro Research Division);
2. a piezoelectric valve for fuel gas injection at the Tokamak Fusion Test Reactor (TFTR) at Princeton;
3. a ceramic electrolysis cell for recovering D-T in the reprocessing of plasma exhaust gas (Japan Atomic Energy Research Institute); and
4. a palladium-alloy membrane diffuser for purification of plasma exhaust gases (JAERI).

All four proved to be attractive and tritium-compatible components for their intended uses. Detailed results are available in the literature.[3,4,5]

TESTS IN THE EXPERIMENTAL CONTAMINATION STUDIES LABORATORY (XCS)

Associated with the TSTA is a small laboratory dedicated to studies on the contamination and decontamination of equipment, surfaces, and atmospheres exposed to tritium. In collaboration with outside programs, two areas of work have been started since our last reporting.

1. With personnel from TFTR and the Idaho National Engineering Laboratory (INEL), studies are in progress on the effect of catalyst temperature and moisture pre-loading in the driers on the efficiency of a gas detritiation system.

2. With personnel from the Joint European Torus (JET), studies are in progress on the contamination effects and residual contamination of remote maintenance tools (welders and cutters) after use in the torus. Both studies are incomplete at this writing.

JAPAN ATOMIC ENERGY RESEARCH INSTITUTE JOINS TSTA

In June 1987, an international collaborative agreement was signed by the Japan Atomic Energy Research Institute (JAERI) and the U.S. Department of Energy. This agreement,
Annex IV to the U.S./Japan Agreement on Fusion Energy, calls for the joint funding and joint operation of TSTA by DOE and JAERI for the next five years (till 1992), thereby doubling the size of the program. Under the agreement, JAERI will attach a four-person staff to TSTA during the five-year life of the collaboration. The JAERI staff arrived at TSTA in mid June, 1987, in time to participate in the integrated loop operations of last June.

NEAR TERM AND FAR TERM NEW INITIATIVES AT TSTA

Currently under test at TSTA is a tritium pellet injector. The device, built and operated by Oak Ridge National Laboratory (ORNL), is a basic injector designed to verify the design principles in tritium service. In particular, tests will prove the workability of the injector in the presence of the radiation, heat of decay, and product of decay (^{3}He) from tritium. Details are given elsewhere at this conference[6].

Since the inception of TSTA in 1976, long term plans have included the eventual addition of a breeding blanket interface at the facility. In 1987, the initial steps to define and develop this process interface were taken in collaboration with blanket design experts at Argonne National Laboratory[7].

The work includes examining several leading blanket concepts (liquid lithium, solid lithium oxides, and aqueous lithium salt) to define the composition and flowrate of the tritium-bearing product stream from the tritium recovery process in the blanket. In a fusion power reactor, this fluid product stream, after appropriate initial processing, will join the plasma exhaust stream to make up the full reactor fueling stream.

The long term goal is to install the appropriate initial processing technology at TSTA and to demonstrate breeding blanket product processing in conjunction with plasma exhaust gas processing. The definition of stream compositions, flowrates, and processing technologies is proceeding now. The schedule for process installation at TSTA is uncertain, though at least several years away, and may depend on developments with the International Thermonuclear Experimental Reactor (ITER).

TRITIUM DESIGNS FOR REACTORS, CONCEPTUAL AND PLANNED

The experience of TSTA and its personnel in handling tritium and in operating the integrated plasma exhaust gas processes is used in tritium system designs provided by TSTA staff for a variety of U.S. fusion reactors, both conceptual study reactors and actual planned reactors. Conceptual reactors to which TSTA staff contributed designs of tritium systems have been the TITAN Reversed Field Pinch Conceptual Reactor Study[8] and TIBER II.

TSTA staff is working on the Compact Ignition Tokamak (CIT), scheduled for operation at Princeton University beginning in 1996. TSTA personnel have prime responsibility for the design, procurement, installation, and commissioning of the tritium fuel systems (except the pellet injector from ORNL), the tritium monitoring systems, and the detritiation and safety systems[9].

Many tritium system designs for the U.S. ITER are also products of TSTA experience and personnel. In some technology areas, notably water detritiation, the major design effort has been contributed by the Canadian Fusion Fuels Technology Project.

TRITIUM AT THE TOKAMAK FUSION TEST REACTOR

The Tokamak Fusion Test Reactor (TFTR) at Princeton University is well along in preparations for the introduction of tritium into the torus in 1990. At this conference, three papers on these preparations will be presented.[10, 11, 12]

TFTR has worked cooperatively with TSTA over several years to obtain additional support for the tritium preparations. TSTA's role has been primarily in three areas--1) system review and consulting in tritium handling processes and techniques, 2) on-the-job training of TFTR tritium workers on tritium-containing systems at TSTA, and 3) the previously-mentioned performance tests and experiments on the piezoelectric gas injection valve and the gas detritiation system.

FURTHER INTERNATIONAL COLLABORATIONS

In addition to the very large collaboration with JAERI, TSTA has a number of small collaborations with the fusion programs at the Joint European Torus (JET) and the Canadian Fusion Fuels Technology Project (CFFTP).

With JET, TSTA has done experiments with remote handling tools (cutters and welders) to measure the tritium releases and tool contamination that result from their use in a simulated, torus-like tritium environment. TSTA also has done process design and review of hydrogen isotope separation by cryogenic distillation at JET in exchange for personnel expenses.[13]

With technical staff from Ontario Hydro in Canada, experiments with tritium have been done at TSTA using commercial getters of potential use in detritiating glovebox atmospheres. This work has been previously reported.[illegible]

OTHER U.S. FUSION TRITIUM PROGRAMS AND COLLABORATIONS

Activities at three other U.S. DOE laboratories tie directly to the fusion tritium program. These are:

- the fusion safety program at Idaho National Engineering Laboratory (INEL);
- the materials program at Sandia National Laboratory (SNL); and
- the breeding blanket program at Argonne National Laboratory (ANL).

Relevant activities at INEL[14] include safety contributions to the major conceptual reactor design studies and to ITER and CIT; experimental studies on ion implantation relevant to the plasma first wall; studies in collaboration with TSTA on the pyrophoricity of uranium and alternative getter materials (e.g. ZrCo and $LaNi_5$) and on efficiencies of gas detritiation systems; and development of the safety analysis code, TMAP, for use with tritium facilities.

Activities at SNL[15, 16, 17, 18] include the tritium plasma experiment that includes collaboration with INEL on ion implantation studies, and contributions on materials for fusion applications, especially first wall and other reactor materials. A large effort has gone into estimating the tritium inventory expected in the graphite tiles that line the TFTR torus and estimating tritium absorption on metallic materials in the TFTR neutral beam boxes. Other work has included studies on in-situ plasma discharge cleaning to reduce in-vessel tritium inventories for CIT, and studies for ITER and CIT on tritium permeation and inventories.

Activities at ANL include the work previously discussed on defining blanket product streams and compositions in collaboration with TSTA. General blanket work also includes contributions to all the conceptual reactor design studies (including ITER) and limited experimental work on tritium release from blankets and tritium management in blankets.

CONCLUSION

The fusion program in the U.S. in tritium research, development, and demonstration has its focus at the integrated fusion fuel processing facility at TSTA. The overall program is multi-faceted, including preparations for tritium introduction at TFTR in 1990, and programs in safety, materials interactions, and blankets. The program is characterized by increasingly closer ties among its elements within the U.S. and with programs abroad.

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