MEETING REPORTS

SUMMARY OF THE AQUEOUS LITHIUM SALT BLANKET WORKSHOP, TORONTO, ONTARIO, CANADA, FEBRUARY 6-9, 1989

INTRODUCTION

A 3-day workshop on the aqueous lithium salt blanket (ALSB) concept was held under the U.S./Canada agreement for cooperation in fusion technology. There were ~30 participants from Canada, the United States, Europe, and Japan. The purpose of the workshop was to review recent work, develop a consensus on design values or approaches where possible, and to discuss research and development (R&D) needs and plans. This report reflects major points presented and discussed. It is expected that many of the technical presentations will be published separately as journal articles.

DESIGN AND PROPERTY DATA BASE

The status of the U.S. and European aqueous salt blanket designs for the International Thermonuclear Experimental Reactor (ITER) and the Next European Torus (NET) was briefly summarized. The U.S. ITER concept is a high-breeding-ratio blanket with substantial beryllium and a LiOH-solution-cooled poloidal cylinder inside a water-and-steel blanket module. The separate cylinder provides good strength at 2-MPa salt solution pressure. The present NET design is based on adding LiOH or $LiNO_3$ salt to a water-cooled shielding blanket consisting of thick steel plates with poloidal coolant channels. This configuration has a breeding ratio of 0.4 to 0.6. In both cases, the coolant temperature is <100°C, and the pressure is set to keep radiolytic gases dissolved in solution.

The total volume of salt solution determines the inventory of blanket tritium and possibly also the dissolved gases. Most of this volume is in the external loop outside the blanket. Two estimates of this external volume were presented that, when scaled to the U.S. ITER design, differed by a factor of 2 to 3. This volume needs to be estimated more accurately.

The material properties data base was reported for LiOH and $LiNO_3$. This includes both literature surveys and some newly measured properties, including hydrogen and oxygen gas solubility. The data base appears to be sufficient for now, although measurements on thermal conductivity would be useful.

CHEMISTRY

Experimental results were reported in several key areas. In general, these results indicate that the blanket designs presented at the beginning of the workshop are based on corrosion and radiolysis assumptions that are consistent with the experimental observations.

Experimental results were presented on radiolysis of aqueous solutions of LiOH, $LiNO_3$, and Li_2SO_4. Both gamma- and neutron-induced radiolysis experiments were performed in closed glass capsules to ITER-relevant doses, but at up to 5% ITER dose rates. The in-reactor yields (in molecules per 100 electron volts) were $G(H_2) = 0.8$ and $G(O_2) = 0.2$ for 4.7 M LiOH; $G(H_2) = 0.2$, $G(O_2) = 1.7$, and $G(NO_2^-) = 2.4$ for 4.4 M $LiNO_3$; and $G(H_2) = 1.6$ and $G(O_2) = 0.18$ for 2.2 M Li_2SO_4.

The gas yield values indicated that radiolytic gases produced in LiOH under ITER conditions could be kept dissolved by operating at system pressures of ~2 MPa. In addition, computer simulations of radiolysis suggested that a small amount of dissolved H_2 might be sufficient to suppress radiolytic gas production. However, it was cautioned that these simulations should be verified experimentally. It was also suggested that radiolysis might be exploited as an option for direct tritium extraction. Future experiments under consideration include a reactor in-pile loop at Studiecentrum voor Kernenergie/Centre d'Étude de l'Énergie Nucléaire (Belgium). Such an experiment, with ITER-relevant doses and dose rates in a flowing loop, is considered critical to the ASLB concept.

General corrosion results were presented for salt solutions of LiOH, $LiNO_3$, $LiNO_2$, and Li_2SO_4 at both ITER (<100°C) and high temperatures (250°C). The LiOH solutions exhibited the highest corrosion rates while the $LiNO_3$ solutions exhibited the lowest corrosion rates, at the lower temperatures. For the LiOH solutions, preliminary measurements of general corrosion rates of Type 316 or 316L stainless steel were reported as 5 to 50 μm/yr at 90 to 95°C and <1 μm/yr at 65°C. It was suggested that a corrosion rate <10 μm/yr should be the target value and that the experimental program should define operating conditions under which this could be achieved. A major loop test is under construction at the Catholic University of Leuven (Belgium).

Results were presented on localized corrosion in both the LiOH and $LiNO_3$ systems. Experiments were performed at both ITER (<100°C) and high temperatures (250°C). For temperatures <100°C, these preliminary results indicated no stress corrosion cracking for the LiOH system, although some evidence of intergranular attack was noted. Conflicting results were presented for $LiNO_3$ at ITER temperature. Some experiments noted no localized effects while other experiments observed pitting. At the same time, it was noted that the pitting was eliminated when small amounts (1 g/100 cm^3) of LiOH were added to the $LiNO_3$ solutions. The need for continued experiments on localized corrosion was emphasized. In addition, design features such as welds should be included in the experimental program.

In the presence of a transverse magnetic field, the flow of the aqueous salt solution will induce a potential gradient in the solution. For the current U.S. ITER design (modified such that the cross-field channel widths are <1 cm everywhere), the magnitude of the induced potential is calculated to be in the 0.1- to 0.3-V range. While it was agreed that this induced potential would affect the electrochemical characteristics of the salt solution, the impact of this effect could not be predicted without experiments. At present, Rensselaer Polytechnic Institute is planning such an experiment.

During the discussions of this session, it was agreed that the chemistry state of the salt solution would have a major impact on corrosion and radiolysis. This state would itself be affected by interactions between these phenomena and with the external recombiner and purification system. For example, the addition of dissolved H_2 to try to suppress radiolysis in LiOH would also decrease corrosion rates and affect stress corrosion cracking. To find a solution chemistry state with acceptable corrosion (including electrolysis) and radiolysis was agreed to be the key R&D requirement for the salt solution blanket. Previous experiments have been primarily focused on corrosion or radiolysis rates alone. Future experiments must also consider the interactions among these phenomena.

TRITIUM AND AUXILIARY SYSTEMS

In this session, out-of-core systems required for the ALSB were reviewed. These systems are shown in block diagram form in Fig. 1.

A recombiner may be necessary to treat the full flow of salt solution exiting the blanket in order to recombine any

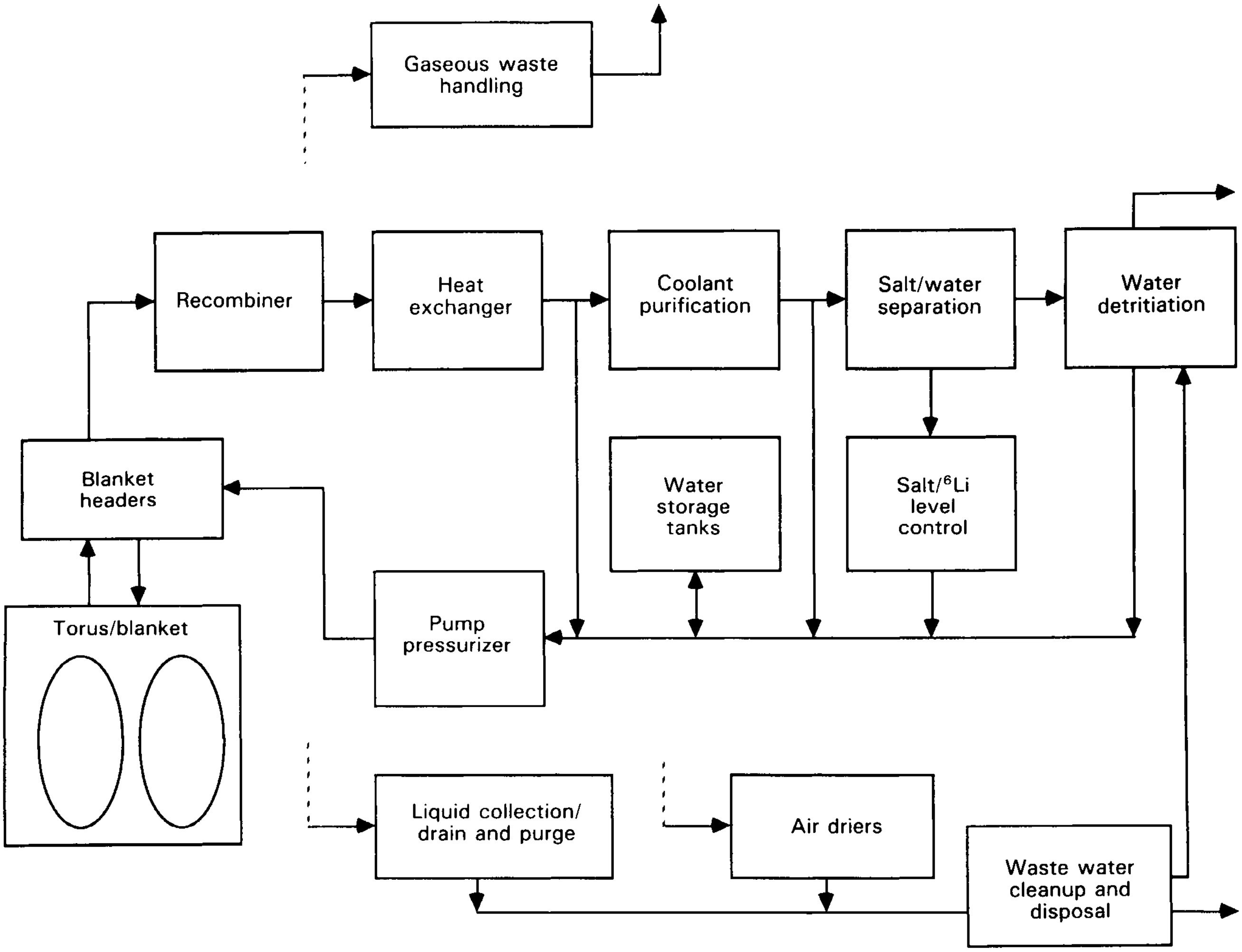

Fig. 1. Schematic of ALSB external loop and major subsystems.

H_2 and O_2 produced by radiolysis. For LiOH, the gas entering the recombiner will likely be hydrogen-rich, since oxygen is removed from the circulating solution by corrosion-related reactions. Design alternatives involving both gas- and liquid-phase recombination appear to require acceptable quantities of commercially available wetproof catalyst, costing in the range of U.S. \$100 000 to \$150 000. Gas-phase recombination could be used if radiolysis products were allowed to come out of solution, while liquid phase would be used with the preferred approach in which the radiolytically produced gases are maintained in solution by the system pressure.

Uncertainties exist concerning the impact of the salt solution on the catalyst lifetime and performance, decomposition of H_2O_2 (produced by radiolysis) on the catalyst, and the possible presence of catalyst poisons such as copper. Design and safety issues include the flow resistance for the liquid-phase recombiner, which may be sharply increased by deposition of circulating "crud" and the potential for platinum from the catalyst to be spread throughout the cooling loop, leading to the possibility of explosive recombination if it deposits where a gas pocket could occur. While none of these were viewed as critical to feasibility, a pilot-scale test using relevant system parameters was strongly recommended.

Sources of impurities and possible removal methods for coolant purification on the main salt solution flow included degassing (depressurization), filtration, and ion exchange. Tritium extraction, involving a sidestream following the main purification flow, includes processing steps that also remove impurities. An ion-exchange material involving lithiated resins was proposed to minimize the loss of lithium from the salt solution. Resin cost based on a 5.5-m^3 bed that might be expected to have a 1-yr lifetime was estimated to be of the order of U.S. \$60 000. Behavior of the ion-exchange resin in the salt solutions at these high salt concentrations is unproven, although the resin is expected to function based on elementary chemical considerations and manufacturer's recommendation. Performance and lifetime in salt solution should be measured. It was noted that when impurity compositions and quantities are not known, the purification system design becomes more complex, leading to uncertainties in reliability and increases in cost. Given the uncertainty in the impurity constituents in the flow and the potentially large quantities of corrosion products involved, testing under representative process conditions appears to be an essential step to provide the necessary design data. This point was given further emphasis during the discussion of safety aspects, where it was pointed out that the anticipated levels of activated corrosion products could be expected to increase the risk of plant contamination and reduce the feasibility of hands-on maintenance.

Salt/water separation based on flash evaporation and/or desalination evaporators was discussed. The only issue of note raised was the uncertainty in the effectiveness of each evaporator stage (and thus in the required number of stages) in limiting carryover of impurities, particularly if foaming occurs. It was noted that foaming is likely in LiOH, especially in the presence of organic impurities. There is a substantial base of industrial knowledge in the area of salt/water separation, and the workshop participants agreed that the logical next step would be to carry out an industrial study on design feasibility. The possibility of using membrane separation was considered, but there do not seem to be sufficient performance, safety, or economic incentives to pursue alternatives to evaporators.

Water detritiation based on water distillation, vapor-phase catalytic exchange, and cryogenic distillation of hydrogen isotopes was accepted as the reference process for tritium extraction. An alternative based on separation of the radiolytic gases produced by blanket irradiation was considered, but the uncertainties of this method (namely, the extent of radiolytic decomposition, the fraction of tritium present in the gases, and the process dependence on reactor operation) led to the conclusion that this approach was not currently an appropriate design alternative.

It was noted that the water detritiation step was a part of the overall isotopic separation scheme for next-step fusion engineering machines and that many of the components were required for other tritium recovery duties. Thus, the cost estimate presented (U.S. \$60 million, including engineering, buildings, and installation) should not be viewed as fully chargeable against the blanket system. Part of the cost presented (U.S. \$1.5 million) was attributed to meeting containment requirements given the tritiated water levels. It was noted that licensing requirements and design practices vary from country to country and that this cost might consequently be higher. To properly design and cost the system, it was recommended that detailed tabulation of overall isotopic separation design requirements be carried out in the ITER context. In particular, the need to detritiate coolant flows other than the salt solution and end-of-life coolant detritiation should be considered.

TRITIUM CONTROL AND SAFETY

Worker exposure, routine losses to the environment, and accidents were considered for tritium and activated corrosion products. Canada deuterium uranium (CANDU) fission reactor experience with internal (tritium) and external (gamma) dose was presented. As each new CANDU station was designed, improvements in tritium control were implemented. At present, tritium uptake accounts for ~25% of average worker dose. There is little correlation between tritium concentration in the moderator or primary coolant and worker uptake. This is because of hazard isolation, leaktightness technology improvements, confinement dryers installed in generally accessible areas, and use of plastic suits in rooms with high atmospheric tritium concentrations.

The transport of activated corrosion products was noted as a major concern for worker exposure. It was estimated that an ALSB could have more circulating corrosion products than a nonsalt water blanket. Assuming 10 μm/yr corrosion rates and 3-mm-thick walls and using transport models based on pressurized water reactors, it was reported that fields on coolant piping could be too high for hands-on maintenance (~1.5 rem/h at 1 m) even with ion-exchange purification systems in place.

For ITER, tritium emissions resulting from aqueous salt systems were estimated to be 1 to 24 Ci/day for 10 Ci/kg water, depending on the assumed loss from leakage and spills and the extent of recovery. Two approaches to estimating recovery were presented. One suggestion was to use the water recovery experience of a CANDU power station, which is ~95% recovery of all escaped heavy water. The second method is to directly estimate all leakage paths in ITER/NET, including drier performance.

Dryer technology is proven and simple in applications of chronic detritiation of water-vapor-contaminated rooms. Dried rooms contribute very little to overall station emissions. For example, at the Bruce CANDU station, only ~3% of station tritium emissions came from dried confinement areas. Most came from generally accessible equipment areas with low tritium-in-air levels, where once-through ventilation is used. In these stations, there is no strong incentive to reduce component water leakage in dried areas – or to put additional contaminated rooms on dryers – if the station emission targets are being met. For normal operation, dryer sizing is determined by the rate of air leakage into the dried area, the tritiated coolant leakage rate, and the acceptable tritium-in-air levels. The room size is important for emergency cleanup. For the ITER/NET systems handling tritiated coolant, it was suggested that one air change per day would be a practical leaktightness.

The total water loss from leakage and spills was estimated to be 0.2 to 2 kg/h. Better documentation of the CANDU water leakage experience was noted as an area that would allow better predictions, although it would not include any effects due to high salt concentrations. Assuming 0.2 kg/h total water leakage at up to 30 Ci/kg into dried rooms, six 1.9 m^3/s (4000 ft^3/min) CANDU-type driers were suggested as adequate.

Scoping analyses were presented for accident involving internal blanket breeder tube failure, failure of breeder tube leading to first-wall rupture, and loss of coolant. Preliminary results indicate that breeder tube break accidents, with loss of building confinement, may have acceptable off-site doses. It was reported that off-site doses from a loss-of-coolant accident may approach the 10-rem ITER target for passive safety, with the assumed high release fractions of corrosion products dominating the dose. In all cases, loss of the tokamak boundary integrity would considerably worsen the accident consequences because of the possible release of tokamak dust, first-wall tritium, volatilized first-wall steel, and hydrogen from steam/graphite reactions. The important factors in determining accident consequences are the tritium concentrations, water evaporation rates, and the release of circulating and deposited activation products. The modeling of the accident scenarios, release fractions, and accident frequency is not well characterized at this time.

Hydrogen safety was also discussed with respect to Ontario Hydro's experience in the design of the Tritium Removal Facility. The desire to release and disperse hydrogen before it can accumulate and detonate works against the desire to contain tritium releases. Industrial experience is that 30 to 40% of all hydrogen leaks result in fires (mostly) or explosions. Fusion facilities handling hydrogen must be designed for hydrogen safety: physical separation of systems with large hydrogen inventory from systems with large radioactivity inventory and safety design features to minimize the likelihood of a hydrogen release leading to a detonation (e.g., adequate room volume to prevent flammable mixtures forming).

CONCLUSIONS

This workshop was the first occasion in which all the major active participants in this area – in all technical disciplines – were brought together to review both their own technical areas, as well as its relation to the other tasks in supporting the ALSB concept. Several new experimental and analytic results were presented. Based on the workshop presentations and discussion, it appears that the ITER reference design (4 *M* LiOH operating at <100°C) is feasible. At the same time, several areas were identified where future work is required. On the experimental side, the key area is system chemistry. On the analytic side, the key area is safety. Some of the issues will be addressed within the context of current plans, but additional effort will be required.

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SUMMARY OF THE U.S.-JAPAN WORKSHOP ON EVALUATION OF IMPURITY PELLET INJECTION FOR ALPHA DIAGNOSTICS, SAN DIEGO, CALIFORNIA, JANUARY 16-17, 1989

INTRODUCTION

The U.S.-Japan Workshop on Evaluation of Impurity Pellet Injection for Alpha Diagnostics was held on January 16-17, 1989, at General Atomics (GA) in San Diego, California. The first day was devoted to presentations from the participants describing their relevant work. The second day was a roundtable discussion of the important issues and future plans. The meeting agenda is given in Table I, and a list of attendees is given in Table II.

The purpose of this workshop was to evaluate progress and discuss future plans for work on developing a diagnostic of fast confined alpha particles in a fusion plasma using impurity pellet injection. Methods based on charge-exchange (CX), nuclear, and X-ray excitation interactions of the alphas with the pellet cloud were discussed.

Pellet Penetration

All the proposed methods share the problem of getting a suitable pellet far enough into a hot, dense, burning plasma. For example, a carbon pellet, sized to cause only a 5% volume-averaged electron density perturbation, must be injected at ~10 km/s to penetrate near the center of the Compact Ignition Tokamak (CIT) at $n_{e_0} = 7 \times 10^{14}$ cm^{-3}, $T_{e_0} = 25$ keV. This result is based on a pellet ablation model developed by Parks et al.[1] Presentations were made on two possible pellet injection methods. K. Sato described the Nagoya University Institute of Plasma Physics (IPP) work on accelerating pellets using laser ablation. This approach may lead to high velocities; J. Osher mentioned that Lawrence Livermore National Laboratory (LLNL) had accidently accelerated an ammonia pellet to 8 km/s in a 1-kJ laser heating experiment. Unfortunately, the work in Japan was part of the Reacting Plasma Project (R-Project) and has stopped due to lack of funding. Osher described the work on electric guns at LLNL where kapton ($C_{22}H_{11}N_2O_5$) projectiles have been accelerated to >20 to 25 km/s. Adapting this technology into a tokamak impurity pellet injector requires testing the gun's