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ITER SYSTEM STUDY - SAFETY ASPECTS

S. J. PIET
Idaho National Engineering Laboratory
EG&G Idaho, Inc., P.O. Box 1625,
Idaho Falls, ID, 83415, U.S.
1-208-526-9542

H. IIDA, Y. SEKI
JAERI
Naka Fusion Research Establishment
Ibaraki-ken 311-02, Japan
292-95-3111

J. RAEDER
NET
Max Planck Institute,
Garching bei Muenchen, F.R.G.
49-89-3299-4214

L. N. TOPILSKI
Kurchatov Institute,
Ploshchad' Akademika Kurchatova 46
Moscow 123182, USSR
7-095-196-16-06

ABSTRACT

One of the tasks of the ITER Systems Analysis Group has been safety and environmental analyses. The safety analyses show that radiological doses from operational effluents and accidents are consistent with anticipated regulatory dose limits. To improve safety further and to meet the ambitious goal of "passive safety", further efforts to reduce inventories of tritium and activation products are proposed. Most major safety issues are now associated with Plasma Facing Components (PFC).

INTRODUCTION

There are two principal objectives of ITER safety studies. First is to show that ITER can be designed and operated safely, satisfying anticipated regulatory requirements. Second is to see how future fusion machines, including ITER, can be improved further to benefit from the potential advantages of fusion reactors. These potential advantages include the absences of radioactive products from the fusion reaction itself and of criticality accidents.

We examined national regulations in the various ITER countries. We formulated a "conservative" average of these requirements. The major anticipated regulatory dose limits are as follows:

1. Doses due to normal operation effluents must stay below 0.1 mSv/yr (10 mrem/yr).
2. Doses calculated for any accident of regulatory concern must stay below 100 mSv (10 rem).

To move toward the potential advantages of fusion energy, we formulated more ambitious targets for further improvement. The most important one is as follows:

3. A strong effort based on passive features should be made to keep doses below 100 mSv (10 rem), which are calculated under very low probability assumptions and do not take credit for confinement by the buildings.

ASSESSMENT OF NORMAL OPERATION EFFLUENTS

The assessment of tritium and activation products effluents is summarized in terms of public doses (μ Sv/yr) from atmospheric release and dispersion:

	QF=1	QF=2
Tritium via active ventilation system	3	6
Tritium release during maintenance	1	3
Steel/water corrosion products	1	1
Niobium/water corrosion products	1	1
Total atmospheric effluent dose	6	11

The two values for each pathway result from different values for the tritium quality factor, QF. In most countries, the present regulatory QF for tritium is 1. In the USSR, the present regulatory QF is 2. Also, recent experimental evidence points toward QF=2 for tritium as the better technical position.

Besides atmospheric effluents, there will also be liquid effluents. We believe that these are not likely to dominate. Thus, we expect the total dose from effluents to be well below the anticipated regulatory requirement of 100 μ Sv/yr.

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ACCIDENT ASSESSMENTS

Radioactivity Inventories

Estimates of tritium inventories in Table 1 show significant uncertainties. Major contributions come from tritium Fuel Cycle processing systems, carbon armor and its dust in the Physics Phase, and beryllium multiplier in blanket in the Technology Phase. Increase of the tritium burn fraction together with improved PFC materials would significantly reduce tritium processing system inventories.

There are some uncertainties in Table 1 that are more than an order of magnitude. The downward uncertainty of 300 for the primary vacuum pumps reflects the potential improvement if turbopumps could be used instead of cryopumps. The upward uncertainty of 50 for fuel processing mainly reflects concern that the maximum mobility fraction could be much

higher than the reference 1%. Lack of data causes the order of magnitude uncertainty for PFC dust. For the Technology Phase tungsten PFC, the two order of magnitude uncertainty reflects the lack of knowledge of tritium behavior and possible trapping in irradiated sprayed tungsten. The 200 upward uncertainty for beryllium multiplier is caused by the concern that the maximum mobility fraction could be much higher than 1% if accident temperatures were high enough to trigger swelling in the beryllium. The 20 downward uncertainty for beryllium multiplier reflects the idea to remove say 90% of the tritium produced in beryllium during operation. Each of these uncertainties implies needs for research.

The total induced activity is about 4×10^{19} Bq (1 GCi). Most is not prone to mobilization, except for the corrosion products, in-vessel dust caused by erosion and plasma

Table 1. Tritium Inventory Estimates

Inventory	----- Physics Phase -----			----- Technology Phase -----		
	Inventory (g)	Mobility Fraction	Uncertainty Factors ^a	Inventory (g)	Mobility Fraction	Uncertainty Factors ^a
Primary vacuum pumps	150	100%	2 / 300	150	100%	2 / 300
Backing vacuum pumps	487	100%	2 / 30	487	100%	2 / 30
Fuel processing	210	1%	50 / 2	210	1%	50 / 2
Isotope separation	430	100%	2 / 3	430	100%	2 / 3
Atmosphere processing	30	1%	2 / 20	30	1%	2 / 20
Water detritiation	40	10%	2 / 8	40	10%	2 / 8
Pellet injector	160	100%	2 / 7	160	100%	2 / 7
Gas puffing	100	100%	2 / 8	100	100%	2 / 8
Neutral beam lines	10	100%	2 / 2	10	100%	2 / 2
PFC surfaces	975	100%	2 / 8	220	10%	200 / 200
Cold PFC dust	1000	100%	20 / 20	100	100%	20 / 20
Divertor water coolant	35	30%	2 / 2	35	30%	2 / 2
First wall water coolant	35	30%	2 / 2	35	30%	2 / 2
Blanket water coolant	1	30%	2 / 2	1	30%	2 / 2
Solid breeder	150	10%	2 / 100	14	10%	2 / 5
Beryllium multiplier	20	1%	200 / 6	1200	1%	200 / 20
Breeder tritium recovery	50	100%	2 / 5	160	100%	2 / 20
Solid waste recovery	180	10%	2 / 4	180	10%	2 / 4
Carbon waste processing	1000	10%	2 / 20	100	10%	2 / 20
Long-term fuel storage	600	1%	2 / 4	600	1%	2 / 4
Total on-site inventory	5700			4300		

a. Uncertainty factors always include a factor of 2 for uncertainty in dose calculations. The first factor is the upward uncertainty. The second factor is the downward uncertainty.

disruptions, and oxidized and mobilized refractory metals such as tungsten. Tungsten, which is one of the candidate armor materials for the Technology Phase, will be highly activated on its surface and may mobilize in some accidents. Thus, it is not a desirable material from the safety point of view.

Selection of Accidents for Analysis

The effort to examine accident scenarios led to a list of those system, components, and initiators that have to be analyzed first. The listed events have been grouped together and investigated in the following four areas:

- o In-vessel safety
- o Ex-vessel safety
- o Tritium system safety
- o Magnet safety

In-vessel Accidents:

Analyses have been conducted for LOCA (Loss of Coolant Accident) and LOVA (Loss of Vacuum Accident). Possible consequences of pipe or header break inside vacuum chamber are overpressure in the vacuum chamber, chemical reactions between in-vessel components and steam, and hydrogen production as a result of chemical reactions. The overpressure depends on the temperature of the water. During tokamak operation, water is under 150 °C, which does not lead to vacuum chamber pressures in excess of the design overpressure of 200 kPa (100 kPa above ambient).

However, there is a separate issue of bakeout/conditioning requirements of PFC materials. After major exposure to air or water, long bakeout of the PFC is required at 350 °C (carbon) or 180 °C (metals). After air or water accumulation or after major plasma disruptions, PFC conditioning is needed at up to 250 °C (carbon) or 180 °C (metals). At present, bakeout/conditioning above 150 °C is not done via hot water. Instead, some other method is to be used, e.g., hot helium for the infrequent 350 °C bakeout.

Water above 150 °C appears to present overpressure concerns for both the vacuum chamber and the building confinement. If high temperature is used, overpressure due to LOCA during conditioning (250 °C) becomes more than 2 MPa, which is far beyond the design value. Overpressure would be relieved into a dedicated part of the reactor building before it reaches the design value of the vacuum vessel. Blowdown into the reactor building causes spread of tritium and activation products residing in the vacuum vessel.

Hydrogen production resulting from carbon-steam reactions is estimated to be several kg, implying low probability of hydrogen explosion even if air ingress follows. This estimate is valid for a carbon temperature of 1000 °C for 90% of the first wall area, 1500 - 1800 °C for the rest. However, in some accidents, higher temperatures may be reached, e.g., a LOFA (loss of flow accident) preceding the LOCA. In this case, larger hydrogen production can be expected. Also, the final type of PFC carbon selected may increase the predicted hydrogen produced. For these reasons, the design prudently has inert gas around the plasma vacuum boundary, preventing air ingress into the vacuum vessel and eliminating the possibility of a hydrogen explosion.

Analysis of loss of vacuum accidents shows that the reaction between air and carbon is self-terminating mainly because of continuous cooling.

Ex-vessel Accidents

LOCA and LOFA outside the vacuum vessel have been analyzed. Ex-vessel LOCA, for example, a pipe break at pressurizer connection in the divertor cooling system during plasma burn leads to loss of heat transfer in the divertor. This requires almost immediate plasma shutdown to avoid major damage of in-vessel divertor pipes. If major damage occurred in the vacuum chamber, radioactive inventories could leak through the failed cooling system into the reactor building. Without plasma burning, decay heat during an ex-vessel LOCA may cause a gradual temperature increase. The maximum temperature of the divertor plate is estimated to be 700 ± 100 °C. If this is followed by a LOVA causing air/steam ingress, tungsten and rhenium activation products will oxidize and become volatile.

Overpressure in the reactor building, which is the final confinement barrier, can also be caused by an ex-vessel LOCA. The maximum overpressure is estimated to exceed the design pressure of 130 kPa (30 kPa above ambient) for an ex-vessel LOCA during conditioning if high temperature water is used. For a large overpressure, a pressure relief is necessary through a vent/filter system to keep the integrity of the reactor building confinement. Most of the tritium in HTO form and the activation products in the form of aerosols can be trapped by filter pressure relief, but tritium in HT form cannot without some active capture mechanism. During pressurization of the reactor building, some uncontrolled leakage from the reactor building could result at 1 % volume/day.

Tritium System Accidents

Tritium system safety technology is being developed in existing test facilities. The basic safety approach uses multiple confinement barriers. The ITER approach uses primary and secondary confinements in addition to the final containment by the confinement building. The primary confinement is the process equipment and piping.

To prevent a large tritium release, the following safety approach is essential. Hydrogen inventories in storage beds and cryogenic components, such as isotope separation columns, are limited to low levels. Rupture disc and suppression volume are provided to keep pressure increase low enough even in case of accident. Oxygen concentration is controlled to avoid explosive reactions in chemical processing components, such as a catalytic reactor in the fuel clean-up system. Interlock and scram systems employ fail-safe design, such as fail-off heater and fail-off gas circulation pumps.

Secondary confinement will be glove boxes for small component and rooms for large components. Both are filled with inert gas to avoid hydrogen combustion.

Although basic safety technology already exists for tritium, careful design and construction are necessary since the ITER tritium system is more complicated than existing test facilities.

Magnet Accidents

The magnet system does not confine significant radioactivity inventories, but the large energy inventory is a potential accident initiator. Without careful design, accidents could propagate to the vacuum chamber, which confines significant amounts of radioactivity. Therefore, the magnet system must be designed to confine any damage. The toroidal field (TF) magnet system appears to be the primary concern. The capacity for the poloidal field (PF) system to interact with an external system is less, and there is substantial design flexibility to prevent such interaction. Fault conditions of the TF coil system have been investigated, and the most severe ones have been identified. Depending on design details, some could cause large deformations or movement of inner coil legs. They include (a) a short circuit in one coil with fast discharge of the others and (b) a break of a current lead to one coil, internal discharge of that coil, and failure to discharge the others.

In the first case, the current in the shorted coil is driven to a maximum of about

2.6 times its operational value, while current in the other coils decreases to near zero. Structural analysis shows that the coil case stress remains below the yield values but that friction does not maintain the wedging in the central vault. The shorted coil tends to adopt a circular form. There is some possibility of its contacting the vertical structure of the central solenoid. However, the solenoid is capable to withstand the extra pressure and any damage would be confined to the magnet system.

In the second case, the current in one coil will reach zero while all others remain fully charged. The vault compression tends to squeeze the faulted coil outwards from the vault, with the movement restrained by friction and shear keys. If slip occurs, the faulted coil can contact the thermal shield and vacuum vessel. In the present design, a friction factor of 0.2 would be enough to prevent slip. Further detailed analyses are required to confirm that slip cannot happen. If necessary, vault rigidity can be improved by design modification or by welding vault elements together.

Quench propagation with failure of fast coil discharge has been analyzed. The quenched coil can be heavily damaged, but the coil case still retains its integrity and damage is contained within the coil.

The maximum helium inventory in one coil is about 8 m³ (PF coil 6). In the event of cooling line break, it is possible to expel this amount of helium in to the cryostat and pressurize cryostat to 30 kPa. This does not appear to be a problem.

One of the remaining concerns is damage of the plasma vacuum boundary by arcs, e.g., the vacuum pump duct. Careful design for preventing this damage should be done in the upcoming Engineering Design Activity (EDA).

Consequences of Major Accidents

Table 2 describes examples of upper bound accident scenarios and their dose consequences. These are selected from the accidents that lead to release of large amounts of radioactivity into the reactor building. Consequences are evaluated with conservative assumptions, such as, no function of detritiation system during the first seven days after the accident. Doses are calculated for the maximum exposed individual. Weather and release parameters are assumed to be worst case: class F stability, 1 m/s wind velocity, ground level release. The site boundary is assumed to be 1 km from the release point.

Table 2. Upper Bound Accident Scenarios

IN-VESSEL LOSS OF COOLANT ACCIDENT AND/OR LOSS OF VACUUM ACCIDENT (LOVA) DUE TO IN-LOCA

Disruption, runaway electrons, or earthquake cause major water ingress from PFC coolant systems into the torus. Steam pressurizes torus. PFC temperatures decrease quickly; thus volatility of PFC surfaces is minimal. Releases from PFC dust, tritium in coolants, tritium in near-torus systems, and activated corrosion products.

	<u>Physics phase</u>	<u>Technology phase</u>
Maximum dose with confinement ^a (mSv)	24	11
Maximum dose without confinement ^b (mSv)	1200	900
Uncertainty factor	3x	10x

COMPLETE EX-VESSEL LOSS OF COOLANT ACCIDENT (EX-LOCA) - WITHOUT LOVA

Earthquake, magnet motion, water hammer, disruption, or pipe whip cause several ring headers to fail, spilling coolant from PFC and blanket. Purge stream continues to remove tritium from blanket. Plasma shuts down; but decay heat is not removed. Reference LOCA temperature is 700 ± 100 °C for divertor, 600 ± 100 °C for first wall, lasting for 1 week. Releases from tritium in breeder and beryllium, tritium in coolants, and activated corrosion products. Use of four separate coolant quadrants may make this scenario outside of regulatory concern.

	<u>Physics phase</u>	<u>Technology phase</u>
Maximum dose with confinement ^a (mSv)	0.9	2.3
Maximum dose without confinement ^b (mSv)	45	110
Uncertainty factor	3x	10x

COMPLETE EX-VESSEL LOSS OF COOLANT ACCIDENT (EX-LOCA) - WITH LOVA

Same as above, except that high temperature leads to LOVA, so that air enters the torus. So, in addition to above scenario, this also mobilizes tokamak dust and causes oxidation-driven volatility. Releases from volatilized species, PFC dust, tritium in breeder and beryllium, tritium in near-torus systems, tritium in coolants, and activated corrosion products. Use of four separate coolant quadrants may make this scenario outside of regulatory concern.

	<u>Physics phase</u>	<u>Technology phase</u>
Maximum dose with confinement ^a (mSv)	25	65
Maximum dose without confinement ^b (mSv)	1200	4800
Uncertainty factor	3x	10x

MAJOR TRITIUM PROCESSING SYSTEMS ACCIDENT

Major accident like hydrogen explosion or earthquake in tritium processing systems, fails most tritium processing systems. Releases from tritium process systems. Provisions for inert gas may make this scenario outside regulatory concern.

	<u>Physics phase</u>	<u>Technology phase</u>
Maximum dose with confinement ^a (mSv)	12	14
Maximum dose without confinement ^b (mSv)	600	700
Uncertainty factor	3x	3x

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- a. Doses with confinement correspond to a regulatory safety assessment. In these cases, the credit for radioactivity confinement is conservatively assessed.
- b. Doses without confinement correspond to a passive safety assessment. In these cases, all active and radiological confinement systems are assumed to fail.
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Two types of dose assessments are made. First, "mobility doses" are estimated taking no credit for radioactivity confinement or any active safety systems. These "doses without confinement" therefore reflect a passive safety assessment. These hypothetical doses use the mobility fractions in Table 1 for tritium inventories and similar mobility fractions for activation product inventories.

Second, an assessment from the regulatory standpoint is made. These "doses with confinement" take credit for the radioactivity confinement. We have estimated the maximum confinement release fraction as generally 2%, resulting from 1% volume per day confinement leakage rate and restoration of detritiation and/or negative pressure at 7 days after the accident. Also, we take a factor of 3.8 credit for wind re-entrainment during long duration radioactivity releases. For some cases involving short-lived isotopes, the maximum confinement release fraction is lower, down to 0.5% because of significant decay before release. The results show that the doses at the site boundary will be consistent with the anticipated regulatory requirements.

ESTIMATE OF RADIOACTIVE WASTES

Radioactive wastes from ITER are expected to be larger in volume, but significantly lower in radiotoxicity, than for a light water reactor of similar thermal power.

Other than the divertor plate, the decommissioning waste mass is about 20,000 tonnes, which are mostly steel. Of this waste, 70 - 100% could be disposed by either shallow geological burial or near-surface burial, depending on national criteria, averaging, and time before disposal.

The mass of replaceable parts of divertor is about 100 tonnes/replacement and design capability includes 15 replacements over the lifetime. This waste (likely to be Nb-1Zr or Mo alloy) would likely require deep geological burial.

CONCLUSIONS

Preliminary safety analyses conducted during the CDA indicate that the evolving design is expected to meet anticipated regulations.

The doses from normal operation and maintenance effluents are consistent with anticipated regulatory requirements.

By proper design, the possibility of uncontrollable accident such as explosion of combustible gases and large damage of torus by magnet deformation can be eliminated.

The dominant radiological hazards in ITER are associated with tritium in the Physics Phase and activated tungsten in the PFC in the Technology Phase. Most of the tritium in the vacuum chamber and much oxidized tungsten can be mobilized in the case of postulated severe accidents. However, the analyses show that mobilized tritium and tungsten can be reasonably confined by the ITER building and the site boundary dose is expected to be consistent with anticipated regulatory requirements.

If radioactive confinement by the reactor building is not taken into account in calculating accident doses, the current ITER design still leads to calculated accident doses larger than 100 mSv. To satisfy the ambitious "passive safety" goal, further reduction of inventories of tritium and mobile activation products is needed.

We expect the following to effectively increase ITER safety performance in the EDA:

1. Find and validate ways to increase the tritium burn fraction from present 2.6%.
2. Find ways to passively shutdown the plasma within about 1 s of significant off-normal conditions.
3. Test a wider range of possible PFC materials that allow less tritium inventory in the vacuum vessel and are radiologically less hazardous as well.
4. In current machines, test ways to reduce tokamak erosion dust, so that the dust inventory in ITER stays below about 1 kg.
5. Verify ways for conditioning without use of high temperature water.

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