TECHNICAL REPORT

Design Study of Swimming Pool Type
Tokamak Reactor (SPTR)

Kiyoshi SAKO, Tatsuzo TONE, Yasushi SEKI, Hiromasa IIDA, Akio MINATO\textsuperscript{1},
Hiroki SAKAMOTO\textsuperscript{2}, Takashi YAMAMOTO\textsuperscript{3} and Kazunori KITAMURA\textsuperscript{4}

Japan Atomic Energy Research Institute*  

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In order to relieve the difficulties of repair and maintenance and to make the reactor size compact, a concept of tokamak reactor which is installed in a water pool has been proposed. Preliminary design study of the concept was carried out. As the result of this study the following advantages over conventional tokamak reactors are shown: The size of TF coil can be considerably reduced while retaining sufficient space for repair and maintenance because a solid shield is eliminated. Disassembling and reassembling of vacuum vessel seems to be done with realistic remote handling technique. The problem caused by radiation streaming can be considerably eased. Radioactive waste disposal is reduced considerably because a solid shield is eliminated.

KEYWORDS: tokamak fusion reactor, swimming pool fusion reactor, design study, experimental reactors, blanket structure, tritium breeding, shielding, repair, maintenance

I. INTRODUCTION

Tokamak reactor components such as divertor plates, first wall and blanket vessels, which are placed near the plasma, are expected to have a low reliability since they receive large and cyclic heat loads, electromagnetic forces and high radiation fluence. Their replacements should be planned in the normal operational scheme. A reactor concept to meet the requirement for easy replacement is proposed in this paper.

Configuration of a tokamak reactor is complex and its components will be highly activated by neutron irradiation. Therefore repair and maintenance of the reactor involves many difficult problems and the selection of their scheme affects greatly the reactor overall design. Figure 1 shows two conventional repairing schemes (Types A and B) and a new one (Type C) which we propose in this paper. In Type A, toroidal field (TF) magnets and shield module are withdrawn together with blanket module when a blanket module requires replacement. While this type has an advantage of small sized TF magnets, it also has a disadvantage that it requires disassembling of TF magnets which causes even more difficult problems. In Type B, blanket and shield modules are taken out in straight motion through the opening between TF magnets which stay in place. The handling technique required for this type is relatively simple. However, its great
disadvantage is that the size of TF magnets becomes large in order to obtain sufficient space between the magnets. Since poloidal field (PF) magnets will be located farther from the plasma when large sized TF magnets are used, required electrical current in PF magnets becomes large. The large current increases both the overturning force acting on TF magnets and the capacity of power supply system. Then the overall reactor system becomes large, meaning greater capital investment.

![Type A, Type B, Type C](image)

*Fig. 1 Various concepts of reactor repair and maintenance*

The allowable toroidal field ripple in the plasma is said to be about ±1%. When the bore of the TF magnets is reduced, the number of magnets must be increased. As a result the space between the magnets become too small for the withdrawal of blanket and shield modules. One solution for such situation is to divide the module into 2 or 3 sectors. In the INTOR Phase One design, such module division is considered to be not feasible because moving some sectors not only in the radial direction but also in the toroidal direction seemed too difficult. Such difficulty is true for the heavy and bulky shield but when the module is small and light we feel the module division scheme is feasible.

We have conducted design studies of tokamak fusion reactors, mainly the JXFR (JAERI Experimental Fusion Reactor) (1) and recently participated in the design work of INTOR (International Tokamak Reactor) (2)(3). In those design studies we tried hard to establish a promising repair and maintenance scheme of a tokamak reactor. We employed Type A in the design work of the JXFR. In the INTOR Phase Zero we studied Type A scheme as a reference design and Type B with small toroidal magnets as an alternative in which sectored blanket modules are withdrawn from removable hatches located on the upper part of shielding structure(4). The alternative concept had difficulties in the transferring operation of a blanket module inside a solid shield. In the INTOR Phase One design Type B with large toroidal magnet was selected and studied in detail. The reason for this selection is that the first priority is placed on the simplification of reactor repairing scheme although the reactor size becomes large.

As well as the difficulties of developing fully remote and sophisticated handling techniques required for the repair and maintenance design, countermeasures are necessary for the accidents during the repairing process: for instance, a module run off from the guide track or sticking of a remote handling machine at some awkward location. We consider this problem to be crucial for a repairing design. On top of this problem additional shielding for radiation streaming through various openings on the shield modules makes
reactor configuration very complex.

A new concept of tokamak fusion reactor is thought to be necessary for solving these problems suffered by the conventional concepts. Since the large and heavy solid shielding structure for superconducting magnets is the cause of most of those problems, we think the reactor concept using water instead of solid as radiation shielding is promising\(^{(5)}\). Therefore we conducted a preliminary design study of this concept which is Type C in Fig. 1.

II. REACTOR CONCEPT

Concept of the reactor building and the layout of major components are shown in Fig. 2. The NBI (Neutral Beam Injector), RF (Radio Frequency) heating devices and main vacuum pumps are placed in the room around the pool whereas reactor modules and magnets are submerged in the pool. Connecting ducts are installed between the vacuum vessel and the components such as vacuum pumps, NBI's or RF heating devices.

The upper part of the pool wall can be removed when large sized components are transferred into or out of the reactor building. This makes possible the effective use of the upper space of the building. Installed also on the reactor room is an overhead
500 t crane with manipulators for reactor maintenance. The amount of water in the pool is about 8,000 t. The water is purified continuously.

Vertical and horizontal cross sections of the reactor are shown in Figs. 3 and 4. As shown in Table 1 this reactor is intended for an experimental reactor having nearly the same parameters as the INTOR-J(2).

Table 1 Major design parameters

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fusion power</td>
<td>400 MW</td>
</tr>
<tr>
<td>Plasma major radius</td>
<td>5.3 m</td>
</tr>
<tr>
<td>Plasma minor radius</td>
<td>1.1 m</td>
</tr>
<tr>
<td>Plasma chamber radius</td>
<td>1.3 m</td>
</tr>
<tr>
<td>$B_t$ on axis</td>
<td>5.2 T</td>
</tr>
<tr>
<td>Plasma current</td>
<td>4 MA</td>
</tr>
<tr>
<td>Tritium breeding ratio</td>
<td>~1.0</td>
</tr>
</tbody>
</table>

One of the design targets of this reactor is to achieve the tritium breeding ratio of more than unity.
1. Blanket and Vacuum Vessel

Table 2 shows the design parameters of blanket. The reactor module consists of a part of vacuum vessel, divertor plates and blanket modules. As shown in Fig. 4 there are two types of reactor modules, e.g. Types A and B. The reactor modules are joined together with dielectric bolts. Vacuum seal welding will be made between the modules. Type A module which has the port for NBI or ducts for evacuation is placed at the opening of the TF magnets and simply taken out radially when it requires replacements. Type B module is moved in the toroidal direction before it is withdrawn radially.

The inner blanket is installed in contact to the vacuum vessel while the outer blanket is fixed to the vessel with the support legs providing space between them for helium ash exhaust. Divertor plates are installed at the top and bottom parts of the vacuum vessel. Since heat deposition rate in the vessel near the divertor plate is expected to be high because of neutron streaming through the divertor throat, additional cooling will be necessary in this part of the vessel. Other part of the vessel can be cooled by natural convection of the pool water.

The first wall is an assembly of stainless steel pipes cooled by heavy water. Tube in shell type structure is employed for tritium breeding blanket. Figure 5 shows the horizontal cross section of an outboard blanket module. Lithium oxide (Li\(_2\)O) is employed as tritium breeding material and lead as neutron multiplier. These materials are cooled by water in tubes distributed in the blanket vessel. Helium gas of 1 kgf/cm\(^2\) is flowed in the Li\(_2\)O region in order to recover tritium produced in the blanket.

The divertor plates are made of assemblies of co-axial tubes so as to minimize the electromagnetic force acting on it in case of plasma disruption. The tubes are made of copper and cooled by pressurized water.

The vacuum vessel consists of two types of segments corresponding to the reactor module Types A and B as shown in Fig. 4. Vacuum vessel segments are joined with flanges. Bellows are installed on the center line of the vacuum vessel sector in order to

![](image)

**Fig. 5** Plane view of blanket module
obtain enough loop resistance in the toroidal direction. The part of the vacuum vessel behind the divertor is cooled actively because the nuclear heating is large. The rest of the vacuum vessel is cooled only by water hence the thickness of the vessel is restricted by the temperature gradient due to nuclear heating. For the present stress analysis, the thickness is selected to be 80 mm. With this thickness, even the inboard part of the vacuum vessel where the nuclear heating rate is the largest can be cooled by the pool water.

The total heat generations in the vacuum vessel and the pool water are 11.0 and 2.5 MW, respectively. Therefore, heat loss into the pool water is 13.5 MW. To remove this heat effectively and to prevent water boiling at the surface of the vacuum vessel, inlets of ducts which lead to a heat exchanger placed outside of the pool, are installed between the vacuum vessel and the cryostat.

2. Superconducting Magnet

The Nb$_3$Sn conductor stabilized by Cu is employed for TF magnets. The number of TF magnets is 16. To support the overturning force caused by interaction between vertical magnetic field and the current in the TF magnet itself, the TF magnets are fixed tightly at their top and bottom parts and connected rigidly to each other at the outboard part of the torus by the so-called shear panels. Support structure of TF magnets becomes smaller and lighter by using such shear panels.

Because of the existence of 8 shear panels, the opening for the reactor module extraction is allowed on the 8 magnet windows.

The NbTi conductor stabilized by Cu is employed for PF magnets. The outermost PF magnet is lowered to the bottom of the pool when reactor modules are removed through TF magnet windows.

The configuration of the cryostat of the superconducting magnets is shown in Fig. 3. All superconducting magnets are contained in one huge cryostat. Bellows are installed in the poloidal direction on the cryostat just inside the TF magnets in order to obtain enough loop resistance.

Preliminary stress analysis shows that the primary stresses in the TF and PF magnets are below the allowable limit. Although the cyclic stresses are also estimated to be below the allowable limit according to the ASME Boiler and Pressure Vessel Code Sec. III(6), detailed analysis considering fracture mechanics is necessary.

III. Stress Analysis of Vacuum Vessel

The vacuum vessel is D-shaped toroidal shell consisting of segments joined with flanges. Hydraulic pressure which varies with the water depth acts on the outer surface of the vessel as the external pressure. A stress analysis has been conducted to evaluate whether the vacuum vessel has enough structural strength to endure the hydraulic pressure: A finite element structural analysis code “SAP-V” is employed to calculate the stress. Shell element is adopted for vacuum vessel and three-dimensional beam element is adopted for the flange. The structural material is Type 316 stainless steel. The temperature distribution in the vacuum vessel is not considered in the present analysis.

Figure 6 shows the poloidal distributions of the stress intensities at the outside surface. The maximum of the stress intensity is 1.35 kgf/mm$^2$ on the midplane at the outboard section.

The results obtained from the above stress analysis were evaluated with ASME Code
Sec. III(6). The maximum of the stress intensity corresponding to $P_m$ is 1.24 kgf/mm² and that corresponding to $P_L+P_b$ is 1.35 kgf/mm². The design stress intensity $S_m$ of Type 316 SS at 150°C is about 14 kgf/mm². Each stress intensity is lower than $S_m$ and 1.5 $S_m$, respectively ($P_m<S_m$, $P_L+P_b<1.5S_m$). However, the behavior of the D-shaped vacuum vessel under external pressure is not uniform and the compressible stress through the thickness occurs locally. In such case, buckling or instability problems should be evaluated. In addition, the following quantities should be evaluated in the future; (1) thermal stress by nuclear heating, (2) stress and buckling under magnetic force with plasma disruption and (3) strength of the bellows used for the vessel under hydrostatic pressure, thermal load and magnetic force.

IV. NEUTRONICS AND SHIELDING

As regards the shielding design, SPTR has several distinct merits as follows: The shielding for the penetrations in the blanket and shield such as NBI ports, diagnostic ports and coolant pipings may be greatly simplified. The wall and roof thickness of the reactor room may be reduced significantly because water will serve also as the biological shield. As water fills any void gaps between blanket, shield and superconducting TF magnets, total space required between the first wall and TF magnet may be reduced. In addition, the quantity of radioactive waste will be reduced considerably because the major portion of activating shield will be replaced by less activating water.

Tritium breeding, nuclear heating, radiation damage and induced activity dose of the reactor were evaluated using one-dimensional ANISN calculations(7). The Inboard Model of the reactor employed in the calculations is shown in Fig. 7. The Inboard Model consists of a cylindrical plasma surrounded by the reactor components such as the blanket and magnet on the inboard side, inside the major radius. The Outboard Model, includes the reactor components on the outboard side. A coupled 42-group neutron and 21-group $\gamma$-ray cross section set GICX40(8) was used. For induced activation dose calculation, THIDA system(9) was used.

The tritium breeding ratio ($T_{in}$) for the Inboard Model and that ($T_{out}$) for the Outboard Model are calculated. Taking account of the area allocated for the divertor and NBI ports in Fig. 3, the tritium breeding ratio of the whole reactor is assumed to be given as

$$T_{total} = 0.2 T_{in} + 0.6 T_{out}.$$
Tritium breeding ratio contributions from the \(^{6}\text{Li}(n, a)\text{T}\) and \(^{7}\text{Li}(n, n'\alpha)\text{T}\) reactions are given in Table 3. The \(T_{\text{total}}\) value is calculated to be nearly 1.0.

Neutron and \(\gamma\)-ray fluxes in the Inboard Model is shown in Fig. 8. The first wall neutron loading is 1 MW/m\(^2\).

The flux of neutrons with energy greater than 0.1 MeV decreases about six orders of magnitude from the first wall to the surface of the superconducting TF magnet. The irradiation condition and damage rates on the surface of TF magnet are shown in Table 4. Since the shielding capability of water is lower than that of stainless steel, the required net thickness of bulk shield and blanket becomes greater in the case of SPTR compared to conventional reactor. However, as water can fill the gaps between blanket, shield and TF magnets, the overall distance between the first wall and TF magnet is about the same for SPTR and conventional one. All the calculated values for the radiation damage of TF magnets in the table satisfies the design criteria adopted from the IAEA INTOR Zero Phase Report\(^{(3)}\). Since the material thickness in the Outboard Model is 50 cm thicker than the Inboard Model, there will be no problem in the outboard side.

![Fig. 8 Neutron and \(\gamma\)-ray flux distribution in inboard blanket, water and SCM](image)

**Table 3** Tritium production in SPTR blankets

<table>
<thead>
<tr>
<th>Reaction</th>
<th>(T_{\text{in}})</th>
<th>(T_{\text{out}})</th>
<th>(T_{\text{total}})</th>
</tr>
</thead>
<tbody>
<tr>
<td>(^{6}\text{Li}(n, a)\text{T})</td>
<td>1.116</td>
<td>1.233</td>
<td>0.963</td>
</tr>
<tr>
<td>(^{7}\text{Li}(n, n'\alpha)\text{T})</td>
<td>0.032</td>
<td>0.041</td>
<td>0.031</td>
</tr>
<tr>
<td><strong>Total</strong></td>
<td>1.148</td>
<td>1.274</td>
<td>0.994</td>
</tr>
</tbody>
</table>

**Table 4** Irradiation conditions on inboard surface of TF magnet of SPTR

<table>
<thead>
<tr>
<th>Items(^{1})</th>
<th>Tentative design criteria(^{11})</th>
<th>Calculated values</th>
</tr>
</thead>
<tbody>
<tr>
<td>Maximum dpa in Cu (dpa/yr)</td>
<td>(5 \times 10^{-5})</td>
<td>(1.0 \times 10^{-5})</td>
</tr>
<tr>
<td>Maximum neutron (E_n &gt; 0.2) MeV</td>
<td>(2 \times 10^{16})</td>
<td>(4.9 \times 10^{16})</td>
</tr>
<tr>
<td>Maximum nuclear heating (\text{W/cm}^2)</td>
<td>(10^{-5})</td>
<td>(3.4 \times 10^{-5})</td>
</tr>
<tr>
<td>Maximum epoxy dose (rad)</td>
<td>(3 \times 10^{9})</td>
<td>(4.1 \times 10^{8})</td>
</tr>
</tbody>
</table>

\(^{1}\) Availability and the reactor lifetime are assumed to be 20% and 10 yr, respectively.

\(^{11}\) From the INTOR Zero Phase Report (8) Table XVI-1.
Nuclear heating rate in the Inboard Model is shown in Fig. 9. The maximum heating rate is 13.5 W/cc at the surface of the first wall. Total nuclear heat deposition in the blanket and water is calculated to be 14.2 MeV per source neutron, which is lower than an expected value. This is likely to be caused by the low nuclear heating rate in the Pb region due to not taking account of Bremsstrahlung contribution in the γ-ray kerma factor calculations(10).

Activation dose rate distribution in the Inboard Model 1 d after shutdown after 1 yr of continuous operation is shown in Fig. 10. The γ-ray dose rate at the first wall is $4 \times 10^6$ rem/h. The development of fully remote repair and maintenance operations under this high dose rate is required. The maximum dose rate outside the Pb shield is 10.6 rem/h at the surface of the magnet cryo-vessel. This dose level still necessitates remote maintenance operation.

When all the stainless steel structures are replaced by Al structures, and the Cu stabilizer in the superconducting magnets is replaced by Al, the dose rate from the induced activity will be reduced by about $10^{-4}$. Even then the blanket will have to be remotely handled but the maximum dose rate will be reduced to the level experienced in the present day nuclear fission related facilities.

V. REPAIR AND MAINTENANCE

Among the components in the reactor module the divertor plate is expected to have the shortest life followed by the first wall and blanket. However in this design the exclusive exchange scheme of the divertor plate or blanket is not adopted.
This is because the exclusive withdrawal of divertor plate with other components in place is not necessary in the SPTR since a whole reactor module can be replaced far more easily than in conventional type reactors. Easy replacement of the blanket is the basic requirement for the experimental reactor. Failed divertor plates or blankets will be repaired in hot cells in a reactor site. By employing this type of maintenance scheme the configuration of the reactor module is simplified.

Weights of the individual reactor components are listed in Table 5. The SPTR has enough space for repair and maintenance in comparison with the conventional type reactor, and this is a significant merit for repair and maintenance of a reactor. This reactor concept has also another advantage of not requiring the removal of heavy shield structure. In this chapter, the repairing procedures of reactor structure and the backup system for the accident during repair and maintenance are described.

Repair and maintenance procedures of reactor module are divided into minor repair and major repair. The minor repair means that one of the reactor modules is replaced when a component in the reactor module fails. The major repair means the replacement of all modules, for instance, when the stage I operation is changed into the stage II and the entire vacuum vessel needs to be replaced. The minor repair is carried out after draining the pool water but a major repair can be done either within the pool or after draining the water. The major repairing procedure is basically carried out by repeating that of minor repairs. The procedure of minor repair is described below and illustrated in Fig. 11.

<table>
<thead>
<tr>
<th>Table 5 Weight of major reactor components (including support structures)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Vacuum vessel (including first wall, blanket and divertor)</td>
</tr>
<tr>
<td>TF magnet</td>
</tr>
<tr>
<td>PF magnet (central coils)</td>
</tr>
<tr>
<td>PF magnet (large radius coils)</td>
</tr>
<tr>
<td>Cryostat</td>
</tr>
<tr>
<td>Total</td>
</tr>
</tbody>
</table>

Fig. 11 Procedure of minor repair
(1) Drain the pool water and dry up the pool, (1).
(2) Remove the outermost PF magnet to a pit at the bottom by a crane and take off NBI ports or vacuum ducts, (2).
(3) Release connecting bolts of the failed blanket module which includes the divertor plates, first wall, blanket structures and vacuum vessel.
(4) Cut seal welded joints between blanket modules and cooling pipes by an automatic cutter.
(5) Withdraw the failed blanket module is located in front of TF magnet windows simply in the radial direction by a special machine (3).
(6) Remove the failed blanket module to a cooling pond by the crane, (4).
(7) In case of failed blanket module which located at the rear of a shear panel between the TF magnets, rotate it first in the torus direction, and then withdraw it in the radial direction.
(8) Install a new blanket module and fix it with the setting bolts. (The failed blanket module is repaired during next operation.)
(9) Make seal welding between the blanket modules with an automatic welding machine and inspect the welded joint.
(10) Join the blanket modules with connecting bolts through flanges of vacuum vessel.
(11) Reinstall the outermost PF magnet, NBI ports and vacuum ducts.
(12) After the vacuum test and baking fill the pool water.

Note: Figures in parenthesis at the end of each step description correspond to the step number shown in Fig. 11.

The divertor is not necessary to be replaced periodically. As a small leak which causes the shutdown of the plasma can be detected, the divertor is replaced only at the breakdown. The repair to be carried out after draining the pool water has been investigated. On the other hand, the repair can be done within the pool in a manner similar to the exchange of the fuel assembly in Light Water Reactor. Such repair method has a merit of allowing personnel access into the reactor room. Some of the uncertainties for this method are; the feasibility of the welding in water and whether the tritium leaks in the pool water when the vacuum vessel is filled with the pool water, can be accommodated. Another repair method considered is to set the machines to release the connecting bolts in water and then to cut the welding part after draining.

If the initial failure of the SCM (superconducting magnets), which may occur during the first stage experiment using hydrogen plasma, is fully taken care of, the SCM will very seldom break thereafter. Such thinking has been accepted in many tokamak designs. In this design also, the repair of SCM during D-T burning stage is not considered. Even if the repair of SCM should become necessary, as the repair of the reactor components other than SCM is easy, the repair of the SCM should also be easier than the conventional type reactor.

It may be possible that the manipulator or the structure to be moved, sticks and ceases to move by an incident such as an earthquake during repair and maintenance. On such occasions, resuming the repairing procedure should be very difficult in the case of the conventional type reactor because the handling space is usually very limited.

On the other hand, SPTR without the shielding structures generally has enough space for TV camera to approach failed parts and to obtain good view. The weight and size of the structure to handle is relatively small. Therefore, the recovery operation of
removing stuck objects can be carried out. Moreover, even if the recovery should be impossible, the demolition or decommission of the reactor is easier and more safe. The handling of the waste structures which is lighter is easier compared to conventional reactors. In addition, a man can enter the room and conduct the setting of machines from above the water level of the pool. The monitoring by TV cameras during repairing operation is required. The radiation dose rate at the first wall 1 d after shutdown is \(4 \times 10^4\) rem/h and that at the outer surface of the vacuum vessel is \(\sim 10^3\) rem/h. Since the allowable dose rate of the semiconductor is \(10^4\sim 10^5\) rem\(^{(11)}\), the monitoring at the outer surface is possible without any special shielding. But, a radiation shield is required to monitor at the first wall.

**VI. DISCUSSIONS AND CONCLUSIONS**

A preliminary design study of SPTR is conducted and the following possible advantages over conventional type reactors are found:

1. The size of TF magnet bore can be considerably reduced because a solid shield is eliminated.
2. Since the distances between plasma and PF coils become small, the required electric current in the PF magnets and hence the overturning force acting on the TF magnet, is reduced. This reduced force eases the support structure in the TF magnets.
3. The reduction of electric current in the PF magnets also makes the required capacity of electric power supply smaller.
4. Although the required technological level for the remote maintenance operations is much higher than presently available, realistic and feasible operations seem to be attainable due to the following reasons; the weight of volume of the structures to be handled are relatively small and there is enough space between the TF magnets and the vessel for remote maintenance operations.
5. If the radioactive structures get stuck by accident during a repairing procedure, the countermeasure to cope with the accidental situation will be obtainable because of the radiation protection by the water.
6. The problem caused by radiation streaming can be considerably eased.
7. Radioactive waste is greatly reduced because of the elimination of the solid shield structure.
8. Thick building roof is not necessary with regard to skyshine by shielding effect of neutrons by the pool water during reactor operation. Only biological shield of \(\gamma\)-ray from activated blanket modules during reactor disassembling has to be considered in deciding the roof thickness.
9. An SPTR is particularly suitable as the next step reactor because the vacuum vessel and blanket can be easily replaced\(^{(12)}\). Advantages of this concept can be also applied to a power reactor\(^{(13)}\).

We have studied for about 1 yr the SPTR but so far no critical problem inherent in this concept is found. The following problems and accidents are considered and evaluated:

1. The stress on the magnet cryostat and plasma vacuum vessel due to water pressure is permissible.
2. The corrosion problems of the components by submerging reactor in water including those due to electrochemical effect are also found not to be critical.
(3) The leak rate of tritium into the pool water through the vacuum vessel and bellows during the reactor operation is roughly estimated. The permeation rate is estimated to be less than 10 mCi/d.

(4) In case of small failure of the vacuum vessel, the pool water flows into the vessel and some of it become vaporized reacting with hot blanket, first wall or divertor plates. The vapor and water in the vessel will be highly contaminated with tritium if the gate valve of the main vacuum pump (Cryo pump) fails to close. The vapor and water in the vessel can be extracted from the vessel to the storage tank. Purification can be done with reasonable capacity since it is not necessary to purify them in a short time. In the course of these processes small amount of tritium can leak out into the pool water. In any event it is possible to prevent a large amount of tritium leakage into the pool water.

(5) The large amount of tritium leakage into the pool water will be considered as a design basis accident. A part of tritium will be trapped in the pool water reducing tritium release into the air of reactor room, hence reducing the release out of the reactor site. From the standpoint of public safety, the SPTR is preferable to a conventional reactor.

(6) Repairing procedure in which the reactor modules are disassembled in the pool water is also conceivable. In this case personnel access to the reactor room is allowed.

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