19. TITAN-II SAFETY DESIGN AND RADIOACTIVE-WASTE DISPOSAL

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19.1. INTRODUCTION

Strong emphasis has been given to safety engineering in the TITAN study. Instead of an add-on safety design and analysis task, the safety activity was incorporated into the process of design selection and integration at the beginning of the study. The safety-design objectives of the TITAN-II design are: (1) to satisfy all safety-design criteria as specified by the U. S. Nuclear Regulatory Commission on accidental releases, occupational doses, and routine effluents and (2) to aim for the best possible level of passive safety assurance.

This section presents the safety design and evaluations of the TITAN-II reactor. The safety-design goals for the TITAN reactors are discussed in Section 19.2. To achieve a high level of safety assurance, the TITAN-II fusion power core (FPC) is submerged in a pool of low-temperature, low-pressure water. Various safety scenarios and the impact of the water pool are reviewed in Section 19.3. Detailed analyses of loss-of-flow and loss-of-coolant accidents in the TITAN-II FPC are reported in Section 19.4. Plasma-related accidents and plasma shutdown methods for TITAN-II are similar to those of TITAN-I (Section 13.6) and thus are not repeated here. Section 19.5 describes the radioactive-waste-disposal issues and ratings for the TITAN-II design. A summary of the TITAN-II safety design and analysis is given in Section 19.6.

19.2. SAFETY-DESIGN GOALS

Two main objectives have guided the TITAN safety design: (1) to satisfy all safety-design criteria as specified by the U. S. Nuclear Regulatory Commission (U. S. – NRC) on accidental releases, occupational doses, and routine effluents; and (2) to aim for the best possible level of safety assurance.

Although the accident scenarios and classification systems developed by the U. S. fission industry may not apply directly to fusion reactors, the dose guidelines used by the fission industry will probably either be directly applicable or serve as useful references in defining the radiological safety requirements for fusion-reactor designs. The U. S. – NRC regulations covering fission reactors are described in the Code of Federal Regulations in Sections:

- 10CFR20 – Standards for Protection Against Radiation [1],
- 10CFR50 – Domestic Licensing of Production and Utilization Facilities [2],
- 10CFR100 – Reactor Site Criteria [3],
- 10CFR61 – Licensing Requirements for Land Disposal of Radioactive Waste [1].

Details of the present industry guidelines for satisfying the regulations and the corresponding numerical dose limits are presented in Section 13.2 and are summarized in Tables 13.2-I and 13.2.-II.

Recently, four levels of safety assurance were proposed to facilitate the preliminary evaluation of different designs [4,5]. While these levels are neither precisely defined licensing criteria nor rules for formal safety evaluation, they do provide a relatively simple guide for designers who can use these definitions of different levels of safety to evaluate their designs or to improve on their safety features when appropriate. The following summarize the interpretation of these four levels of safety assurance as suggested by Piet [4] (also see Reference [5]).

Level 1 – "Inherent safety." Safety is assured by inherent mechanisms of release limitation no matter what the accident sequence is. The radioactive inventories and material properties in such a reactor preclude a violation of release limits regardless of the reactor condition.

Level 2 – "Large-scale passive-safety assurance." Safety is assured by passive mechanisms of release limitation as long as severe reconfiguration of large-scale geometry is avoided, and escalation to fatality-producing reconfigurations from less severe initiating events can plausibly be precluded by passive design features. In such a reactor, natural heat-transfer mechanisms suffice to keep temperatures below those needed, given the radioactivity inventory and material properties, to produce a violation of release limits unless the large-scale geometry is badly distorted.

Level 3 – "Small-scale passive-safety assurance." Safety is assured by passive mechanisms of release limitation as long as severe violations of small-scale geometry, such as a large break in a major coolant pipe, are avoided, and escalation to fatality-capable violations from less severe initiating events can plausibly be precluded by passive design features. In such a reactor, sufficiency of natural heat-transfer mechanisms to keep temperatures low enough, given its radioactivity inventories and materials properties, to avoid a violation of release limits can only be assured while the coolant boundary is substantially intact.

Level 4 – "Active safety assurance." There are credible initiating events that can only be prevented from escalating to site-boundary-release limit violations or reconfigurations by means of active safety systems. This is the conventional approach of add-on safety.

The public is adequately protected by all four levels of safety assurance. To understand the meaning of adequate protection of the public, the concept of safety assurance can be further strengthened in the context of probabilistic risk assessment. The risk-based safety goal for TITAN is that fusion accidents would not increase the individual cancer risk of the public by more than 0.1% of the prevailing risk. As a consequence of this goal, a site-boundary whole-body dose limit of 25 rem for accidental release for fission reactors (10CFR100 [3]) has been adopted.

19.3. SAFETY-DESIGN FEATURES

The TITAN-II FPC is cooled by an aqueous lithium-salt solution and therefore the cooling circuit is a pressurized-water system. Furthermore, the primary coolant contains tritium at a high concentration of 50 Ci/kg. A passive safety system is thus required to handle different accident scenarios, to control the potential release of high-pressure primary coolant which contains tritium, and to prevent the release of induced radioactivities in the reactor structural materials even under the conditions of a loss-of-coolant-accident.

Different approaches for passively safe design of fission pressurized-water reactors (PWRs) were reviewed. It was concluded that the most passive approach is the Secure-P (PIUS) design [6] developed in Sweden. The PIUS design approach is to enclose the fission reactor vessel and the primary-loop system into a prestressed concrete vessel filled with cold pressurized water. The hot coolant loop and the cold pool can communicate through upper and lower density locks. One of the key safety features of the PIUS design is the termination of the fission reaction by passively introducing the cold pool of borated water into the fission core through the density locks, when necessary.

Two points of contention for the PIUS design are the stability and reliability of the density locks and the cost of the massive prestressed concrete vessel. These two requirements are unnecessary for a fusion reactor such as TITAN-II, however the safety advantages of a pool design can be fully utilized. Therefore, to achieve a high level of safety assurance, the complete TITAN-II FPC, the pressurized primary-coolant system, and the steam generators are submerged in a pool of low-temperature, low-pressure water as shown in Figure 19.3-1. The cold pool of water acts as a heat sink to dilute the reactor thermal and decay afterheat energy and also eliminates the possibility of releasing tritiated water vapor or other radioactive material to the environment.

The basic sources of thermal energy at reactor shutdown are from the plasma thermal and magnetic energy, the thermal energy of the hot loop, and the induced afterheat power in the FPC structure. In the TITAN-II design, the FPC and the primary-loop circuit are arranged such that during a loss-of-flow accident (LOFA), natural circulation will be developed to remove the blanket afterheat to the secondary loop. In the case of a major coolant-pipe break, the coolant in the hot loop will mix with the pool of water since the complete primary loop is in the pool. With this mixing, the temperature of the pool will only rise slightly because of the much larger volume of the water pool. The pool also acts as a heat sink for the decay afterheat in the structural material of the FPC. In fact, even if the heat transfer from the pool to the surrounding earth is ignored, it would take more than seven weeks for the temperature of the water pool to reach 100 °C. Therefore,

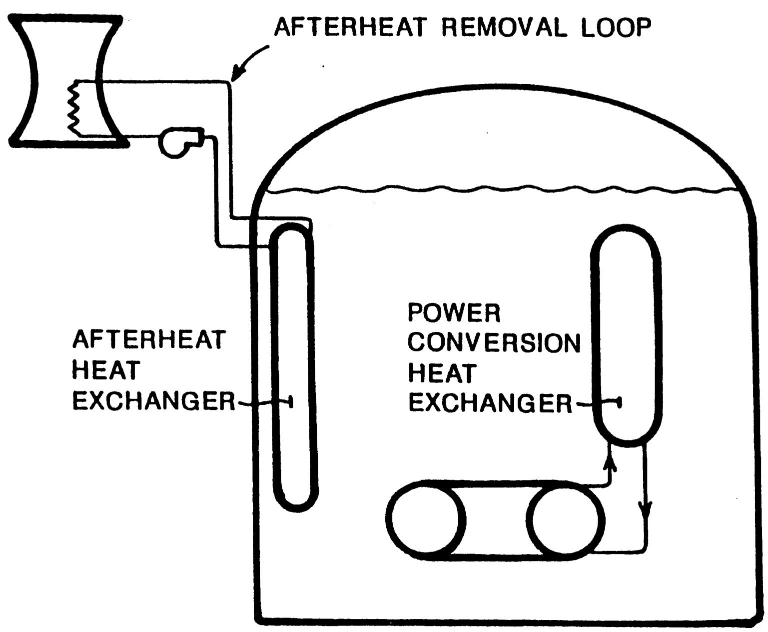

Figure 19.3-1. The TITAN-II "loop-in-pool" configuration.

the potential release of tritium or other radionuclides in the primary-coolant loop to the public is significantly reduced.

19.3.1. Safety Scenarios

Based on the "loop-in-pool" concept of the TITAN-II design, different scenarios for handling normal and off-normal situations were evaluated and are discussed in the following section. The size and operating conditions of the TITAN-II water pool are determined by these analyses.

19.3.1.1. Normal operation

Under normal operation, the FPC is actively cooled by the primary-coolant loop system. In the TITAN-II design, the primary-cooling circuit is not completely insulated from the pool so that the pool can absorb the decay afterheat power in case of LOFA in either the primary circuit or the steam generators. The level of decay afterheat in the TITAN-II FPC is estimated at 34 MW after one full-power year (FPY) of operation at 18 MW/m^2 of neutron wall loading. Therefore, during the normal operation, 34 MW of thermal power is conducted through the steam generator vessel and primary-coolant piping walls to the pool. This power is then removed by separate heat exchangers in the pool.

The pool temperature should be kept as low as possible to maintain an adequate heat-sink capability in the pool in case of an accident. On the other hand, the pool temperature should be reasonably high such that the size of the afterheat-removal heat exchangers in the pool, which are capable of removing the steady power of 34 MW, can be minimized. The exact pool temperature should be determined by detailed design. For the TITAN-II reactor, a pool temperature range of 60 to 70 °C is found to be reasonable based on detailed evaluation of the accident scenarios.

19.3.1.2. Loss of primary-coolant flow

In the accidental condition of the loss of flow in the primary-coolant circuit, plasma operation should be terminated. The steam generators are located above the TITAN-II FPC and the available static head in the primary-coolant loop is more than adequate to establish natural circulation in the primary loop (a static head of one meter is needed for the removal of the afterheat). This afterheat power can then be removed from the primary loop through the steam generators or will be absorbed by the water pool.

19.3.1.3. Loss of primary-coolant pressure

Any sizable leakage in the primary-coolant circuit will result in loss of pressure in the primary-coolant circuit. In this case, the plasma operation should be terminated. The primary coolant will be mixed with the coolant of the cold pool and the pool water temperature will rise. The pool must also absorb the afterheat power from the FPC. The thermal energy in the pool is then removed by the afterheat removal system in the pool. Detailed analysis was performed to estimate the initial temperature rise in the pool after mixing and to find out how fast the pool water temperature would rise should the afterheat-removal system in the pool also fail.

This thermal analysis was performed by using the code TOPAZ [7]. The decay afterheat power as a function of time was provided from neutronics calculations considering key contributing isotopes: ^{56}Mn, ^{55}Fe, ^{54}Mn, ^{60}Co, ^{14}C, ^{32}Si, and ^{185}W. Spatial distributions of the afterheat power from these isotopes were integrated through the entire FPC. This integrated afterheat power as a function of time after shutdown is shown in Figure 19.3-2. The afterheat-power-density distribution in the TITAN-II pool volume itself was also included in the calculation. To simplify the thermal analysis, a cylindrical geometry was used instead of the complicated geometry of the TITAN-II reactor. The water pool has a diameter of 36 m and a height of 31 m. The pool boundary consists of a 0.25-cm-thick steel liner and a 50-cm-thick concrete wall. Thermal conduction to the surrounding earth was considered. The surrounding earth was assumed to extend 100 m both in the radial direction and underneath the pool. Also, to simulate the effect of natural circulation in the pool, a high value for the thermal conductivity of the water in the pool was used.

Several calculations have been performed with different initial and boundary conditions. Figure 19.3-3 shows the estimated temperature rise of the cold pool as a function of time after the accident. These results indicate that the instantaneous mixing of the hot primary loop with the cold pool will cause the pool temperature to rise only 2 °C. This small increase is mainly due to the much larger volume of water in the pool than the volume in the primary loop, and the smaller specific heat capacity of the primary loop containing 6.4 at. % of lithium. Cases 1 and 2 in Figure 19.3-3 correspond to two different initial pool temperatures at 62 and 72 °C. It is estimated that it would take, respectively, more than 116 and 92 days for the water in the pool to reach 100 °C. As

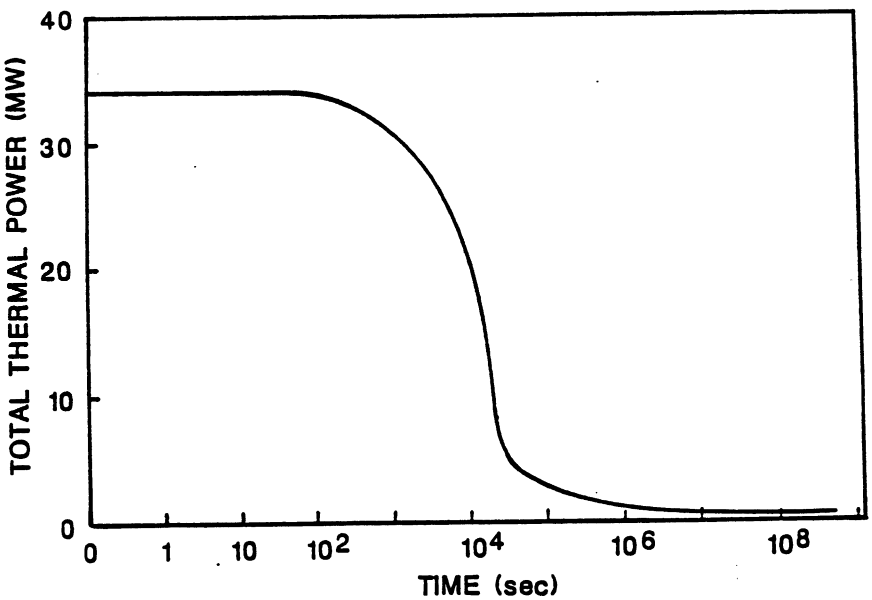

Figure 19.3-2. The integrated afterheat power in TITAN-II FPC as a function of time after shutdown.

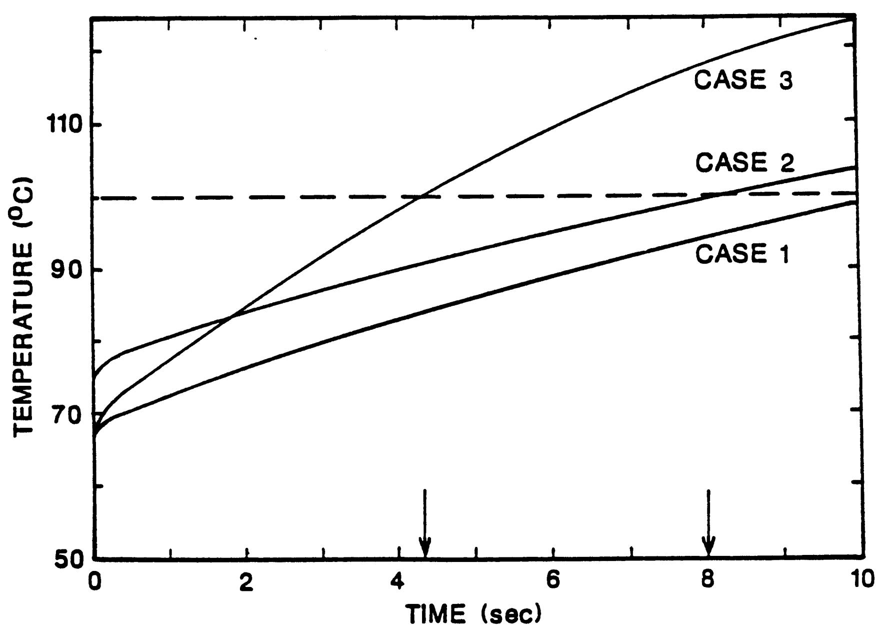

Figure 19.3-3. Estimated temperature of the TITAN-II water pool temperature as a function of time after shutdown.

a comparison, case 3 in Figure 19.3-3 shows that if the pool boundary is thermally insulated, and with an initial pool temperature of 62 °C, it would still take 49 days for the pool water to reach 100 °C. These analyses show that the TITAN-II pool is fully adequate to handle the decay afterheat produced in the FPC even if the pool active afterheat-removal system is not operational, allowing enough time for the recovery or replacement of the afterheat-removal system.

19.3.1.4. Catastrophic failure of primary loop and pool containment

In the catastrophic situation of the failure of the primary loop and the pool containment, the worst accident would be the release of the tritium inventory of the primary loop. At a tritium concentration of 50 Ci/kg in the primary-coolant loop, the complete release of the primary-loop water would amount to 1.37 kg of tritium, which would be higher than the allowable release of ~ 200 g [8] during severe hypothetical accidents (10CFR100) [3]. The complete release of the tritium inventory, however, is a very unlikely event because of the presence of the large pool of water. Also, it is assumed that the induced radioactivities in the structure of the FPC would not be released. This latter assumption is again based on the presence of the large pool of water which would absorb the decay afterheat power.

19.3.1.5. Routine releases

For the TITAN-II design, routine tritium releases and handling of ^{14}C waste are more of a concern than the releases under severe accidents. The potential tritium-leakage rate from the TITAN-II reactor depends on a trade-off between the acceptable level of tritium release and the costs of leakage control and the tritium-recovery system (Section 18). The issues of routine handling of ^{14}C waste are discussed in Section 19.5.

19.3.2. Pipe Break in the Pool

A potential accident for pressurized-water systems is a double-ended rupture of a main coolant line. The escaping jet of the primary coolant (as steam), which may contain radioactive material, will raise the pressure inside the primary containment building and may result in the release of radioactivity to the environment. Another advantage of the TITAN-II water pool surrounding the FPC is the potential to suppress the consequences of a double-ended rupture of the primary-coolant circuit by containing the escaping jet of the primary coolant inside the water pool.

The question arises: If a design-basis accident (DBA) occurs, can the TITAN-II pool prevent a "blow through" of the escaping jet of tritiated water to the surface of the pool? Under a blow-through condition, the pressurized primary coolant can escape directly through the pool of water into the containment building, leading to a high pressure rise and escape of tritium. For our evaluation, the TITAN-II DBA is postulated to occur in the hot leg at a location closest to the water surface in the pool. To prevent the escape of the primary coolant, the minimum pool height above the location of the postulated accident is then identified (Section 19.3.2.2).

19.3.2.1. Fission reactor experience

Energy suppression and fission-product transport in pressure-suppression pools of boiling-water reactors were studied in the early 1970s [9]. A facility was constructed in which steam could be discharged through single or multiple down-comers under the surface of a water pool. In these experiments, the ratio of the maximum visible length of the steam jet to the inside diameter of the steam tube (L/D) was measured as a function of steam flow rate and water temperature. To examine the fission-product transport, iodine was dissolved in the steam.

Pressure-suppression experiments were performed using steam mass-flow rates up to 2.25×10^6 kg/h-m^2 into a degassed or air-saturated pool of water at temperatures up to 60 °C. The maximum inner diameter of the steam tube was about 2 inches. The use of ordinary process water (non-degassed) in the pool resulted in the production of numerous gas (air) bubbles of small diameter (< 1 mm) at a rate which appeared to depend on the steam injection rate. These bubbles are highly undesirable since they trap volatile fission gases (iodine) and transport them to the surface of the pool. Using a degassed pool of water eliminated the production of air bubbles and the character and size of the steam jet could be observed readily. Furthermore, for subsonic steam-flow conditions the L/D ratio is larger in non-degassed than in degassed water (a longer jet of steam). On semilog plots, the observed values of L/D showed a linear relationship with the steam mass-flow rate ($\dot{G}$), with a sharp increase in the value of L/D above sonic flow conditions. The empirical equations for L/D values [9], after conversion to SI units, are

$$\frac{L}{D} = 0.492 \ln \dot{G} - 4.45 \qquad (19.3\text{-}1)$$

for steam mass-flow rates below 8.3510^5 kg/h-m^2, and

$$\frac{L}{D} = 3.18 \ln \dot{G} - 32.57 \tag{19.3-2}$$

for steam mass-flow rates above 8.3510^5 kg/h-m^2. Note that the maximum measured value of the L/D ratio was about 2 in the experiments which were carried out up to a flow rate of 2.2510^6 kg/h-m^2 in degassed water at 60 °C. At that maximum flow rate, Equation 19.3-2 predicts a L/D ratio of 3.85, higher than the observed value of 2 in the experiments.

19.3.2.2. Minimum pool height

The TITAN-II pool is designed to prevent blow throughs. To estimate the flow rate of steam from a coolant-pipe break in TITAN-II, simple empirical correlations for the critical flow rate are used. The critical flow rate of a two-phase mixture is defined as the flow rate at which a drop in the pressure of the discharge plenum for the pipe no longer results in an increase in the flow rate through the pipe. The critical flow depends on the flow regime, *i.e.,* it is a function of the ratio of the length of the discharge pipe to the diameter of the discharge. According to the DBA, the pipe break is to occur as close to the pool surface as possible. This means that the length of the discharge pipe would be much larger than the break diameter. Therefore, the Fauske correlation [10] for large ratios of discharge length to diameter (> 12) is used which results in a maximum steam flow rate of 3.2×10^7 kg/h-m^2 from the 7-MPa TITAN-II coolant loop. Using the sonic-flow steam-jet correlation (Equation 19.3-2), the maximum jet L/D ratio is found to be about 12.3. Thus, for a double-ended rupture of a 0.5-m-diameter hot leg, at least 6 to 7 m of cold (60 °C), fully degassed water is needed above the break to prevent a direct discharge of steam into the containment building.

It should be noted that in the experiments conducted by Stanford and Webster [9], the jet was directed downward, *i.e.,* towards the bottom of the tank. Therefore, applicability of these correlations to a jet pointing towards the top of the tank (as in TITAN-II) may be questioned. But the escaping steam, flowing at supersonic velocities, *does not* form bubbles that are collapsed under the hydrostatic pressure of the pool. Instead, steam is absorbed at an almost perfect sink: the water-jet interface. Because of the nature of the dissipation of the steam jet, it can be argued that the direction of the jet will not drastically affect the L/D ratio of the jet. Upward-pointed steam jets will probably be clipped at the tip and steam bubbles can separate from the top of the jet cone and rise before collapsing.

Experiments are needed to clarify these issues, but preliminary investigations indicate that a pool height of about 10 m above the primary-coolant pipes is sufficient to prevent blow throughs.

19.4. LOSS-OF-FLOW & LOSS-OF-COOLANT ACCIDENTS

Two of the major accidents postulated for the FPC are the loss-of-flow accident (LOFA) and the loss-of-coolant accident (LOCA). Demonstration of the level of safety attainable by the power plant requires in-depth analyses of the response of the FPC to these accidents. Thermal response of the first wall, blanket, and shield of the TITAN-II design to LOFA and LOCA are modeled using a finite-element heat-conduction code. The results of these analyses helped guide the engineering design of the reactor so that the maximum level of safety can be achieved. The principal concern of these analyses is to predict the temperature history and peak temperature of the torus assembly during the accident. Several heat conduction codes are available to model the thermal response of the system during LOCA and LOFA scenarios. Finite-element codes such as ANSYS [11], TACO2D [12], and TOPAZ [7] have sufficient flexibility to handle the time-dependant, nonlinear problem. For the safety analysis of the TITAN-II design, TACO2D is used and analytical checks have been performed, when possible, to verify the results.

19.4.1. Accident Models

An elevation view of TITAN-II is shown in Figure 19.4-1. A cutaway view of the first wall, blanket, and shield is illustrated in Figure 19.4-2 which shows the repetitive nature of the toroidal and radial cross sections. Therefore, a simple geometric model can be used to represent the radial build of the blanket. The use of symmetry conditions reduces the blanket to the configuration shown in Figure 19.4-3. The coarseness of this finite-element mesh appears, at first, to be inconsistent with transient-problem analysis. The problem under study, however, is primarily one of heat capacity and heat flow between components which is mainly governed by the radiation between surfaces and, therefore, is insensitive to the size of the elements within the materials.

The spatial variation of the afterheat at shutdown and the time-dependance of the decay heat in the first wall are shown in Figure 19.4-4. The initial value of the decay heat at shutdown is 12.7 W/cm^3 and the heating rate in the beryllium is zero.

Three accident scenarios have been studied for TITAN-II. The first case is a LOCA without the pool to verify the necessity and/or impact of the low-pressure pool. The

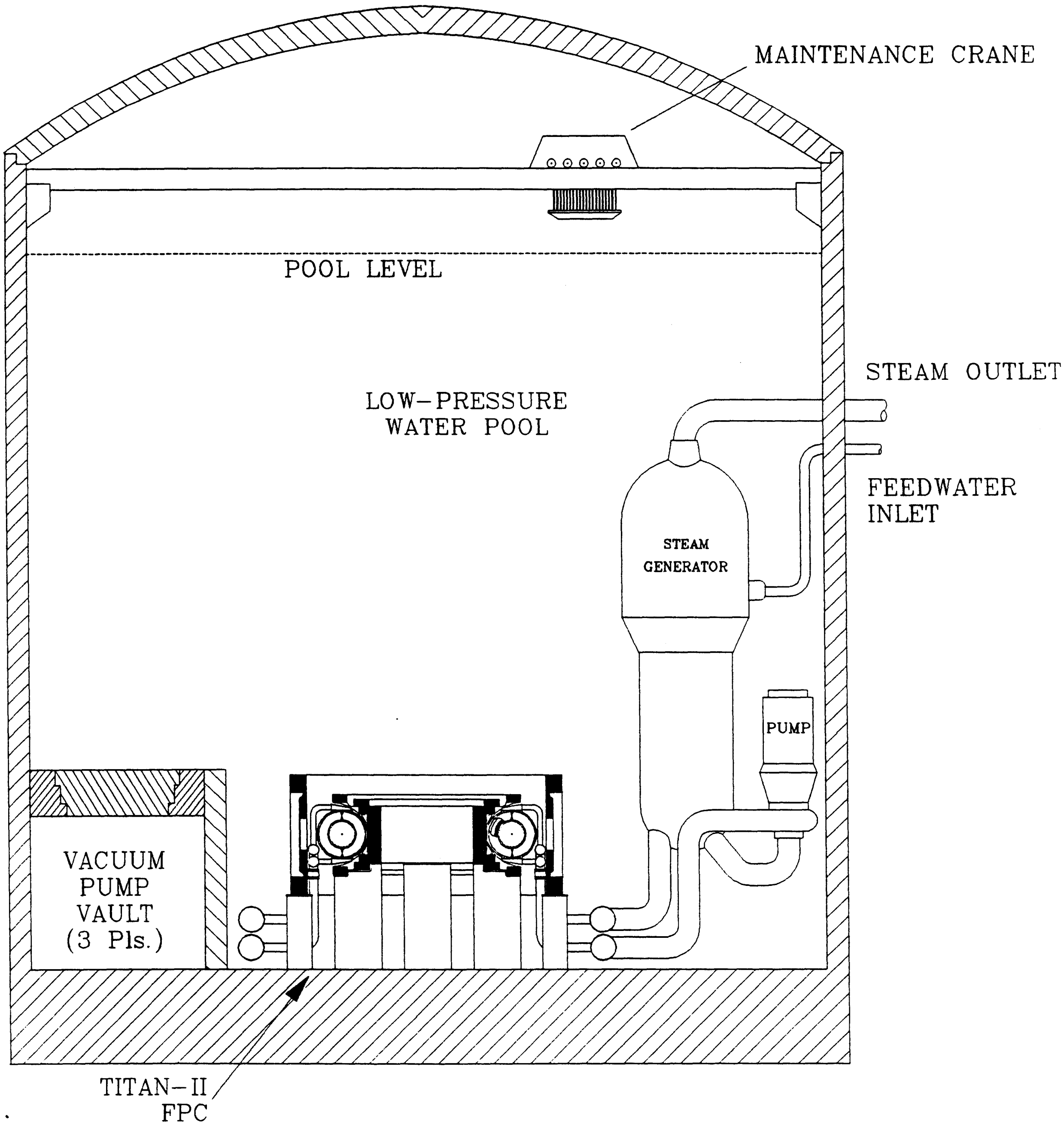

Figure 19.4-1. The elevation view of the TITAN-II fusion power core.

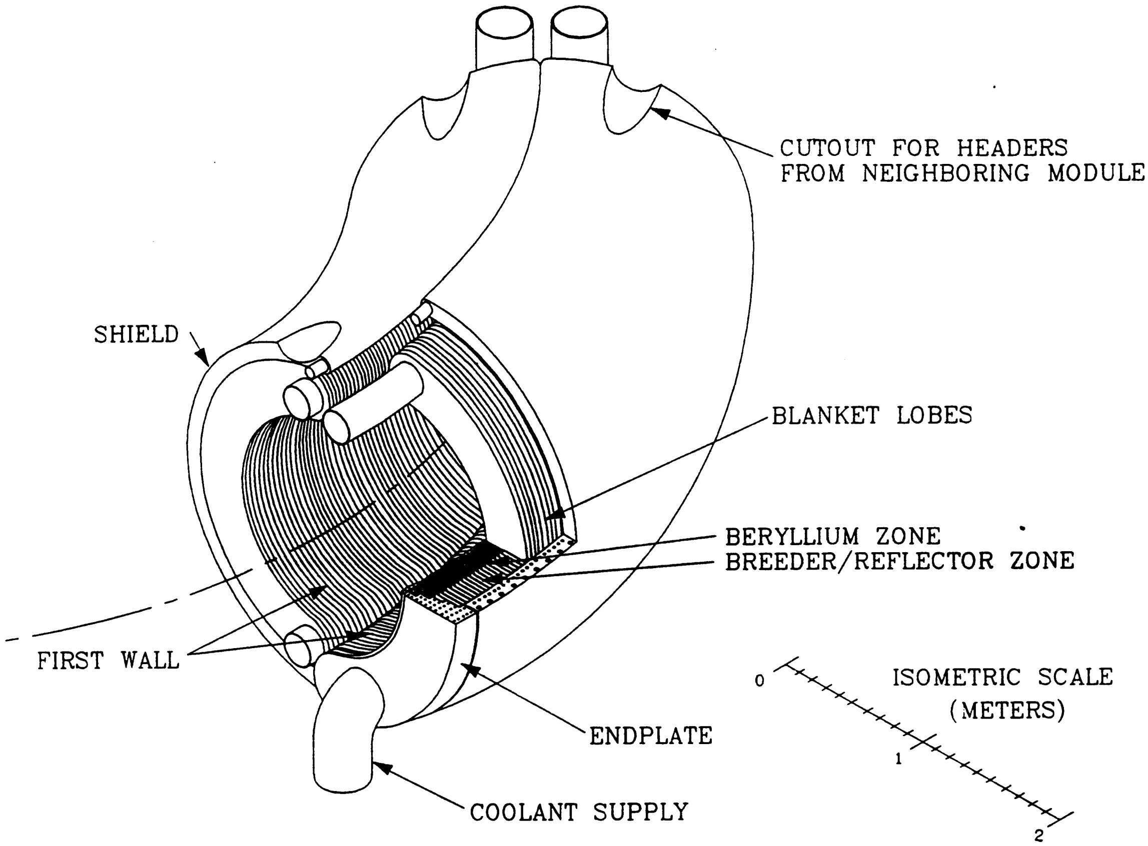

Figure 19.4-2. Cut-away view of the TITAN-II fusion power core.

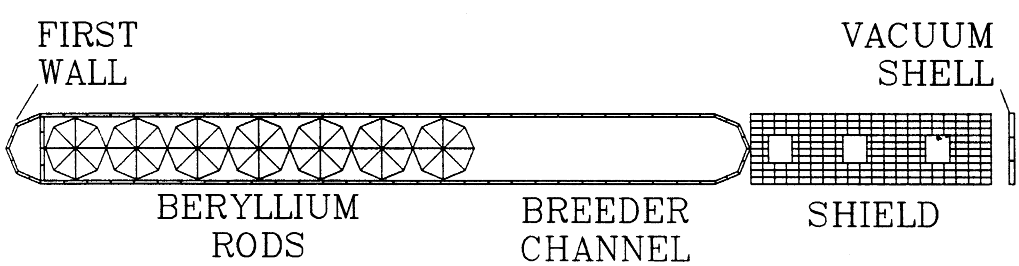

Figure 19.4-3. Finite-element model of the TITAN-II fusion power core used in LOFA and LOCA analysis.

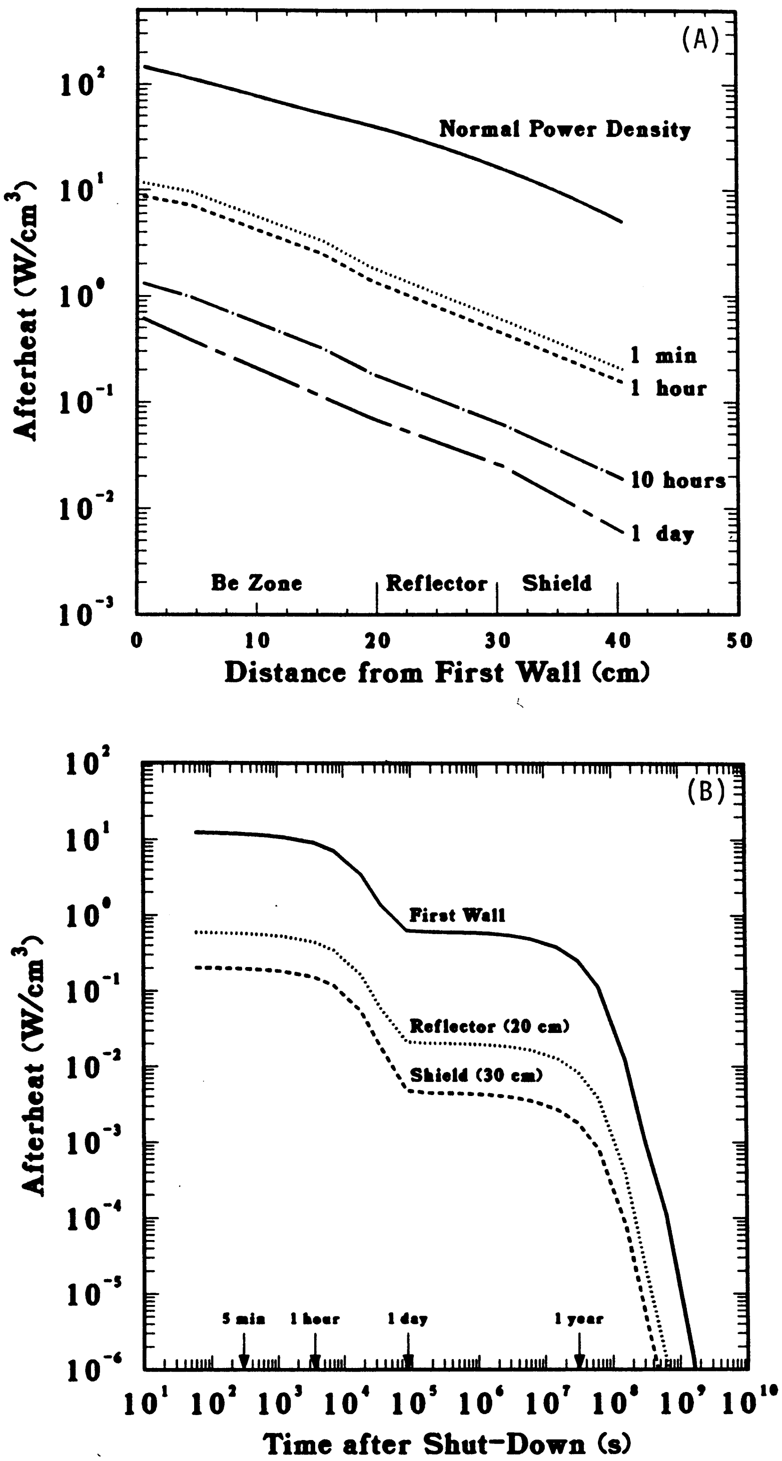

Figure 19.4-4. The level of decay afterheat in the TITAN-II fusion power core at shutdown (A) and as a function of time after shutdown (B).

second case is a LOFA with the pool. The final case is a LOCA with the pool in which the primary-coolant piping is assumed to have ruptured into the low-pressure pool. For all three accident scenarios, the plasma is assumed to have shut down at the instant of the accident initiation. Pump coast down and the time required for primary-loop evacuation will likely negate the effect of a short plasma shutdown period. The effect of a finite plasma shutdown time was not included.

The initial temperature fields are quite different for the above three accident scenarios. Detailed analytical solution of the fluid flow during the transients was beyond the scope of this analysis; however, a qualitative description of the fluid transients is presented and appropriate boundary and initial conditions are used in the numerical analysis. In general, all of the initial temperature fields are based on the maximum coolant temperature (*i.e.*, the coolant exit temperature, 330 °C).

19.4.2. LOCA without the Pool

This is the classic light-water-reactor LOCA, without any means of re-flood. One of the cold legs in the primary loop suffers a guillotine break and the primary loop is emptied after a short drain period. The break is below the level of the torus, therefore no coolant can be trapped inside the FPC. The effect of finite drain time and finite plasma shutdown time are assumed to cancel out. Thus, the initial temperature for the numerical analysis is set at 330 °C. The only heat-removal mechanism is radiation at the back of the vacuum boundary, which is behind the shield. The heat is radiated to the surrounding structure (ohmic-heating coils, coil supports, *etc.*) which are at ambient temperature.

Figure 19.4-5 shows the temperature histories of the first wall and front of the shield as functions of time after shutdown. The peak temperatures are reached after 3.3 hours, quicker than for TITAN-I. The peak temperatures in the first wall and beryllium are 1780 and 1755 °C, respectively, 360 and 471 °C above the melting points for the ferritic steel and beryllium. The necessity of the low-pressure pool is clearly evident from these results.

19.4.3. LOFA with the Pool

This accident can occur, for example, during a loss of off-site power without diesel-generator backup. As before, it is assumed that the heat removal provided by the finite pump coast-down time is sufficient to remove any heat generated by a finite plasma

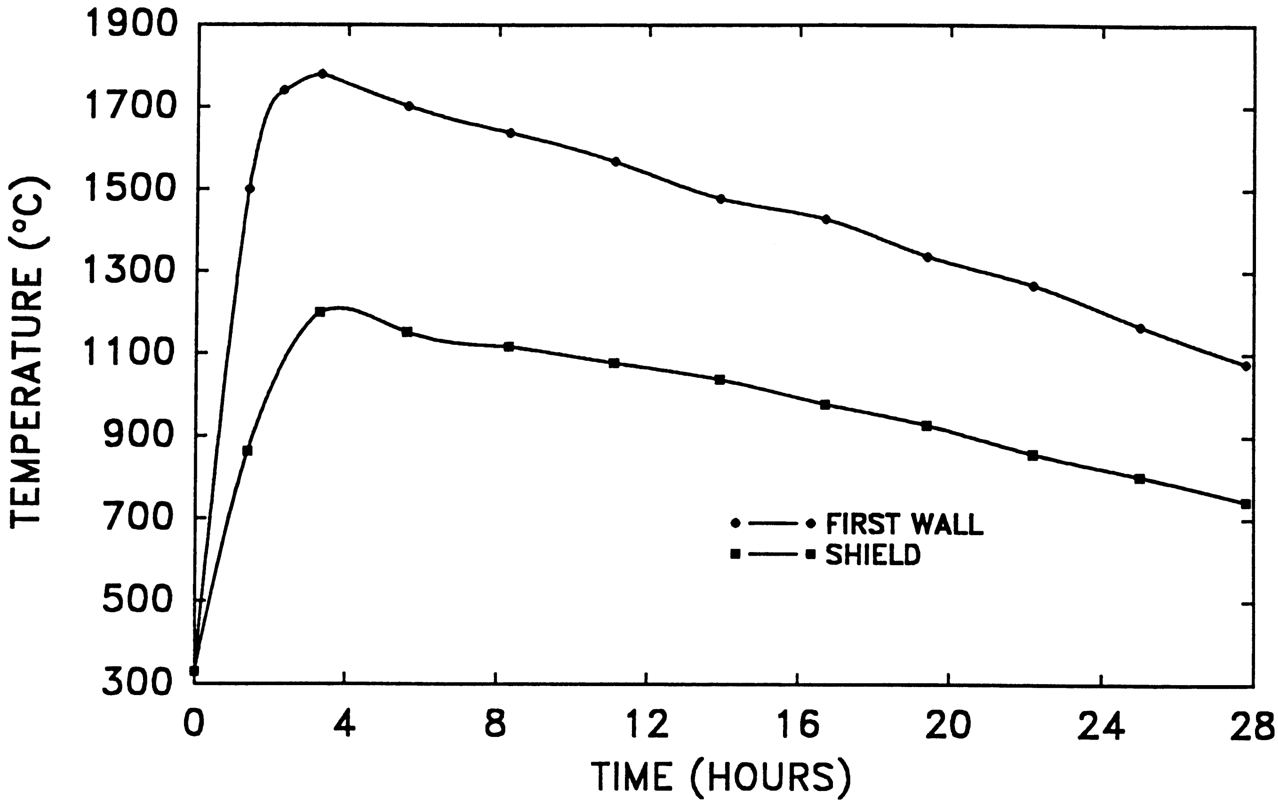

Figure 19.4-5. Thermal response of the TITAN-II FPC to a complete LOCA without the pool as a function of time after the initiation of the accident.

shutdown time. Pump coast-down time is expected to be in excess of one minute. If the plasma shutdown time approaches the pump coast-down time, then this assumption is not valid. In order to model the worst-case accident, it is assumed that no heat removal is provided by the steam generator, and that the pool heat-removal system has also failed. Therefore, the initial temperature is conservatively set to the coolant outlet temperature, 330 °C.

In this scenario, the decay afterheat in the FPC is removed by natural convection of the pool water in contact with the vacuum shell, behind the shield. An average, initial heat-transfer coefficient between the vacuum shell and the pool is estimated to be 0.001 W/cm^2-°C for a horizontal cylinder with a radius of 120 cm. Although the temperature of the water pool rises with time after this accident (Section 19.3.1.3), this increase in pool temperature occurs over a time scale much longer than the torus heat-up period. Thus, the natural circulation in the pool would not be affected during the torus heat-up period.

Natural convection of the primary coolant inside the blanket also removes the decay afterheat which is then lost through the walls of the primary-coolant piping and steam generators to the pool (Section 19.3.1.2). In fact, the normal inlet and outlet temperature conditions of the TITAN-II blanket are sufficient to remove the decay afterheat. The average heat-transfer coefficient for heat removal inside the blanket is estimated at $0.037\,W/cm^2$-°C. The initial bulk temperature of the pressurized, primary coolant is 330 °C and the pool temperature is 60 °C.

Figure 19.4-6 shows the temperature of the TITAN-II FPC as a function of time after the initiation of the accident. For this accident scenario, very little temperature excursion is observed, primarily because of the presence of natural convection within the pool and the primary loop. The first-wall peak temperature of 348 °C is reached after 355 seconds. The TITAN-II reactor appears to be capable of withstanding the loading conditions of this accident scenario.

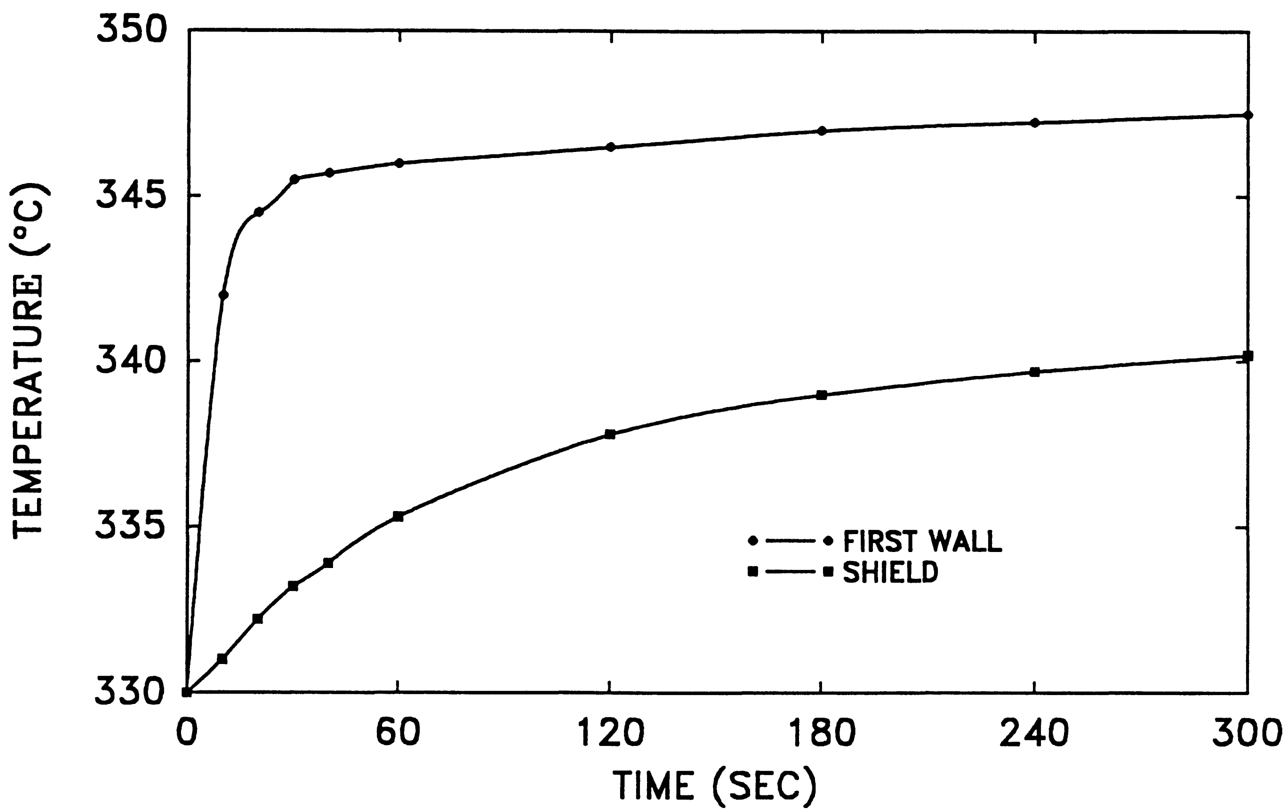

Figure 19.4-6. Thermal response of the TITAN-II FPC to a LOFA with the pool as a function of time after the initiation of the accident.

19.4.4. LOCA with the Pool

This accident scenario is the most complex. The accident is initiated with a guillotine break in the primary cold leg, below the level of the torus. The consequences of this accident have been broken into the following four basic phases.

Phase I: Initial primary-loop de-pressurization. At the onset of the accident, a very rapid (~ 1 s) de-pressurization of the primary loop occurs until the primary-loop pressure reaches the saturation pressure of the primary coolant. For the salt concentration levels used in TITAN-II, the saturation pressure is 5.6 MPa at 330 °C. The time required for this initial de-pressurization from 7 MPa normal operating pressure to 5.6 MPa is very short, on the order of one second, and many factors influence this initial de-pressurization time. For modeling the thermal response of the TITAN-II FPC, the de-pressurization is assumed to be instantaneous. On the other hand, if force calculations were needed, such as required for analyzing a pipe-whip, then a more precise time constant would be necessary.

Phase II: Primary-loop blow down. Following the initial de-pressurization to saturation conditions, a slower de-pressurization takes place until the primary loop and the pool are at equal pressure. Choked flow at the pipe break determines the rate of de-pressurization. As the pressure in the primary loop drops below the saturation pressure of the primary coolant, flashing of the primary coolant occurs and the sudden volume change forces the coolant out of the pipe break (blow-down phase). As with the initial de-pressurization, there are many factors which limit the rate of blow down. Typical design-basis accidents for PWRs will fully de-pressurize in 10 to 20 seconds, provided that no emergency core-cooling system is engaged.

For accident analysis of the TITAN-II FPC, the pipe break is assumed to be at the lowest point of the primary loop (*i.e.,* the worst case accident) since any steam that forms inside the primary piping is trapped because of the buoyancy force. At a pressure of 5.6 MPa, the change in specific volume of the primary coolant from liquid to steam is 27 times. Thus, only about 4% of the primary fluid needs to be vaporized to fill the entire primary loop with a steam bubble. At lower pressures the relative change in specific volume would be even greater. Therefore, it is conservatively assumed that at the end of the blow-down phase, the entire primary loop will be filled with 330 °C steam (operating conditions).

The blow-down phase is assumed to last 20 seconds. It is assumed that during this phase, the heat transfer in the primary loop is linearly decreasing to zero to reflect that the trapped steam does not have any heat capacity or natural circulation. The steam does provide a weak conduction path for heat flow from the steel to the beryllium rods. At the end of the 20 seconds blow-down phase, thermal radiation between internal structures takes place. Initially this radiation is quite small because the temperatures are low. But, as the temperature of the FPC structure increases, the radiation path plays an important role in the removal of the decay afterheat.

Phase III: Primary-loop re-flood. During the re-flood phase, heat is lost from the primary loop (steam) to the surrounding pool and the steam trapped in the primary loop begins to condense. This occurs at the break interface and through the pipe walls. As the steam condenses, its specific volume decreases and the primary-loop pressure decreases. As the pressure drops, the pool water is forced into the primary loop until all of the steam has condensed. The condensation rate depends on a great number of variables; for this analysis, it is assumed that this phase would last 5 minutes. Virtually any condensation rate can be designed into the system simply by adding insulation to the piping (decreased rate of condensation), or by exposing more primary piping to the pool water (increased rate of condensation).

During the re-flood phase, the TITAN-II reactor torus is conservatively assumed to be filled with steam until all of the primary loop is filled with pool water. However, it is assumed that the steam temperature decreases linearly with time from 330 to 150 °C, which is the saturation temperature of pure water at 0.5 MPa. Since the TITAN-II reactor torus is located at very nearly the lowest point in the primary loop, it will experience re-flood conditions sooner than the remainder of the primary loop, an effect not accounted for here.

Phase IV: Natural circulation. The final phase of the design-basis accident is the onset of natural circulation. The TITAN-II reactor is configured so that natural circulation of liquid water is effective in removing the initial decay afterheat with minimal elevation of the steam generator above the reactor torus (Section 19.3.1.3). For this analysis, natural circulation is assumed to be fully developed upon the completion of the primary-loop re-flood phase.

Many assumptions, mostly conservative, have been made for the above four-phase accident scenario. One major assumption is that the steam generator has no influence on the accident scenario. In reality, the steam generator would cool to the temperature of the condenser coolant (nominally 20 °C) and would increase the rate of condensation inside the primary loop. The location of the torus relative to the pipe break is also important. In a realistic design, the torus should be at the lowest point in the primary system. This reduces the amount of piping below the torus and also decreases the volume of piping that needs to be re-flooded before pool water reaches the torus. Any pipe that breaks above the torus will not trap steam inside the FPC, and the reactor torus will, therefore, refill immediately following the blow down.

Thermal response of the TITAN-II fusion power core to this accident scenario is shown in Figure 19.4-7. The peak temperature of the FPC is 732 °C which is 688 °C below the melting point of the ferritic steel. The peak beryllium temperature is 481 °C which is 802 °C below its melting point. Figure 19.4-7 shows that the initial temperature rise in the first wall is linear in time at a rate of about 2.4 °C/s. At the end of the re-flood period (after 300 s), the rate of temperature increase in the first wall is reduced to about 0.62 °C/s. In this analysis, the blanket re-flood is modeled as a step change in the heat-transfer coefficient to account for the natural convection in the primary loop. This step change occurs 300 seconds after the initiation of the accident and accounts for the sudden drop in the first-wall temperature.

The selection of a 300-s re-flood time is somewhat arbitrary since, by design, a wide range of re-flood times can be chosen. Therefore, thermal response of the TITAN-II FPC is also examined for a case that does not include re-flood, *i.e.*, the steam bubble remains in the primary loop and only the radiation inside the blanket to the back of the vacuum boundary governs the heat transport. The result of this analysis is shown in Figure 19.4-7. For this scenario, the first-wall temperature peaks at 1721 °C after 10,500 s (2.9 hours) which is 500 °C *above* the melting point of the ferritic steel. The result of the accident with no re-flood can be used as a design curve to predict the maximum allowable re-flood time. For example, if the design goal is to limit the peak temperature to two-thirds of the melting point ($2/3\,T_m = 1129\,\mathrm{K} = 856\,^\circ\mathrm{C}$), then re-flood must begin no later than seven minutes after the accident initiation. If, on the other hand, the design limit is the melting point of the steel, then the maximum time to re-flood is about 47 minutes.

19.4.5. Discussion

The necessity of the low-pressure pool is clearly evident from the results of the LOCA-without-pool scenario. Peak temperatures well in excess of the melting point of the

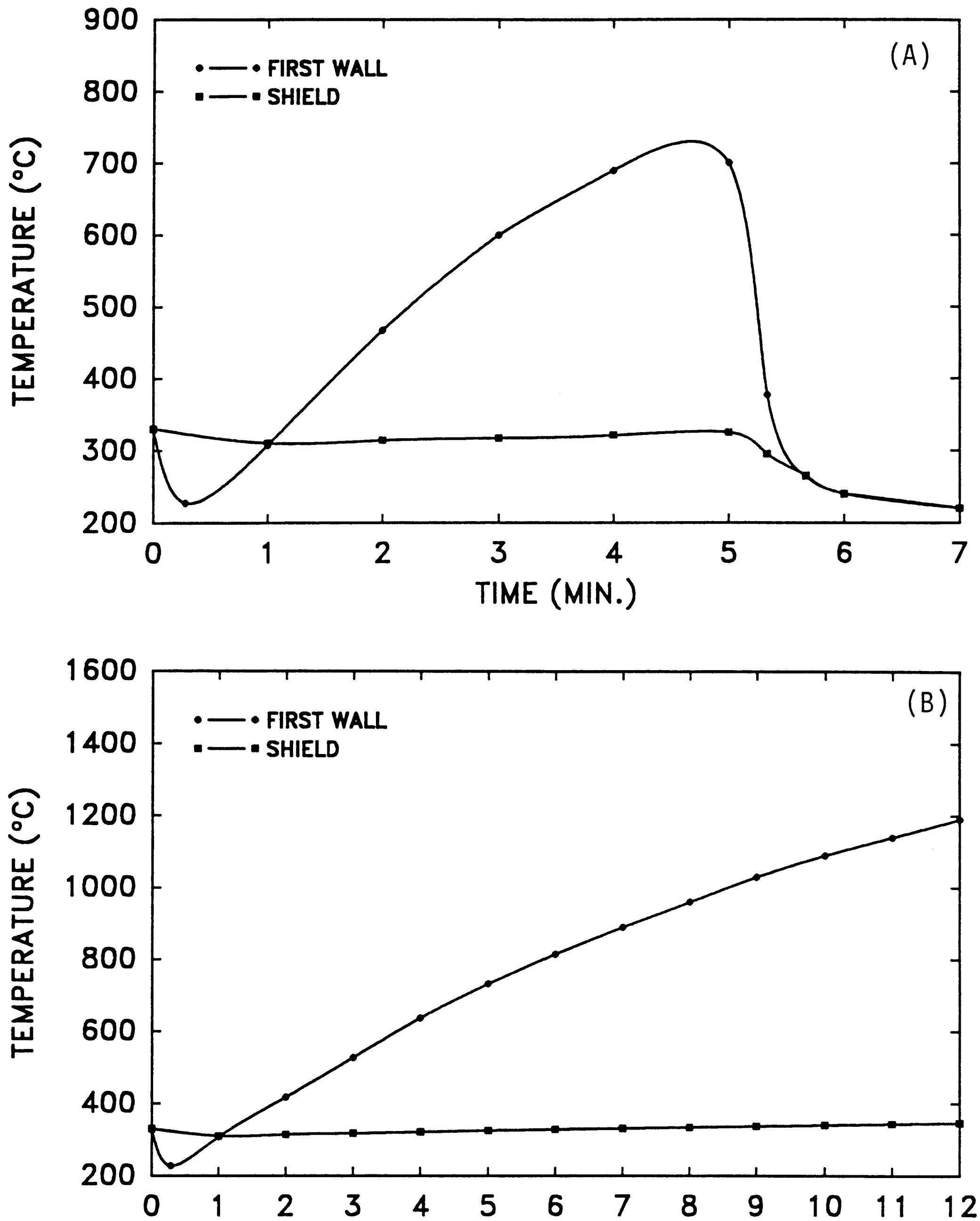

Figure 19.4-7. The thermal response of the TITAN-II FPC to a LOCA with the low-pressure pool as a function of time after the initiation of the accident with a re-flood time of 300 s (A) and with no re-flood (B).

structure are predicted, thus leading to damage to the FPC and potential release of radioactivity. The consequences of a LOFA with pool are minimal, primarily because of the natural circulation in the primary loop and the large heat capacity of the pool. By design, the primary loop will have a heat loss rate to the pool equal to the initial decay afterheat, thereby ensuring adequate heat removal.

A LOCA with pool can have severe temperature excursions if the design of the primary loop does not account for this type of accident. Specifically, the primary-coolant piping must allow for fast condensation of the trapped steam so that re-flood will occur within the desired time period. A re-flood time of approximately 10 minutes will prevent the temperature of FPC structural material from exceeding two-thirds of its melting point, most likely preventing any structural damage. The internal pressure of the blanket during this accident is low ($\sim 0.35\,\mathrm{MPa}$) and thermal-creep rupture should not pose a problem if peak temperatures are below the two-thirds limit.

The LOCA-with-pool scenario is the worst credible accident envisioned. If the torus is at the lowest point in the system, then a trapped steam bubble is impossible and the LOCA cannot occur. This design criteria is chosen for TITAN-II; however, a small length of primary pipe is necessary below the torus to help the development of natural convection.

19.5. RADIOACTIVE-WASTE DISPOSAL

19.5.1. Radioactive-Waste-Disposal Issues

The classification of nuclear waste for disposal is given under the Code of Federal Regulations, specifically 10CFR61 [1]. Four waste classes have been defined: Class A (segregated waste), Class B (stable waste), Class C (intruder waste), and geologic waste. The first three classes of waste are eligible for near-surface burial, while the last class needs deep geologic burial. Radionuclides with half-lives of less than five years will decay by at least six orders of magnitude in 100 years after disposal. These radionuclides can be reasonably managed to meet either Class-A or Class-B disposal requirements. For long-lived radionuclides with half-lives greater than 100 years, however, it will be difficult to meet either Class-A or Class-B disposal requirements solely by radioactive decay to reduce the activity level. To qualify as Class-C or better nuclear waste, the nuclear components in a fusion reactor should minimize the quantity of their alloying and/or impurity elements that would produce long-lived radionuclides.

The limiting-specific activities for near-surface (Class A, B, and C) disposal of nuclear waste were also specified in 10CFR61 regulations for the following radionuclides: ^{59}Ni, ^{94}Nb, ^{99}Tc, ^{129}I, ^{90}Sr, ^{137}Cs, and alpha-emitting transuranic nuclides with half-lives greater than five years. These limiting-specific activities were given primarily for radionuclides relevant to present-day applications such as fission-based nuclear reactors. Many radionuclides with half-lives greater than five years, such as ^{42}Ar (half-life 33 y), ^{108m}Ag (127 y), and ^{186m}Re (2×10^5 y), may be produced by fusion reactors in the elements constituting the structural alloy, the divertor collecting plates, and impurities in the structural alloys. However, the limiting-specific activities for near-surface disposal of nuclear waste containing these nuclides are not available in 10CFR61.

Evaluations of limiting-specific activities for fusion-relevant nuclides were made based on 10CFR61 values. A complete work was recently performed by Fetter [14] providing limiting-specific activities for near-surface disposal of all radionuclides with atomic numbers less than 84. These evaluations, consistent in methodology with the 10CFR61 regulations, were used in the waste-disposal analysis of the TITAN reactors.

19.5.2. Radioactive-Waste-Disposal Ratings of TITAN-II Reactor

The neutron spectra at the first wall, blanket breeder zone, shield, and toroidal-field (TF) coil components of the TITAN-II design are given in Table 19.5-I. In general, a factor of 30 reduction in both total- and fast-neutron fluxes is provided by the TITAN-II blanket and shield.

The neutron fluxes calculated for the reference TITAN-II reactor were used as the input to the activation calculation code, REAC [15]. Table 19.5-II summarizes the maximum allowable concentration levels of niobium, molybdenum, silver, terbium, iridium, and tungsten in various TITAN-II reactor components to qualify for Class-C near-surface-burial waste. Note that the elements given here are primarily those elements identified in Section 13.7 as the most undesirable, appearing as either alloying elements or impurities in the structural alloys. As shown in Table 19.5-II, the allowable concentration levels for Nb, Mo, Ag, Tb, and Ir impurities are, respectively, 0.6, 2.6, 0.33, 0.2, and 0.05 appm in the reduced-activation ferritic-steel structural alloy of the TITAN-II FPC. These levels are, in general, factors of 2 to 10 lower than the allowable concentration of the same elements in the TITAN-I FPC components. This is mainly caused by the softer neutron spectrum in TITAN-II (because of the presence of better neutron-moderation materials, beryllium and water) which results in a higher neutron-capture reaction rate in the TITAN-II FPC components.

Table 19.5-I.

NEUTRON SPECTRUM IN THE TITAN-II FPC

Component$^{(a)}$	Neutron Flux (n/cm^2)	Flux-Reduction Factor$^{(b)}$	Fast-Neutron Fraction$^{(c)}$
First wall & Be zone	5.3×10^{15}	–	59%
Breeder zone	8.0×10^{14}	7.9	54%
Hot shield	3.8×10^{14}	1.9	61%
TF coils	1.8×10^{14}	2.3	57%

(a) Flux values at the side of component facing the plasma.

(b) Defined as the ratio of flux in the front of the component to that in the back.

(c) For $E_n \geq 0.1$ MeV.

The first-wall, blanket, and shield components of the TITAN-II reactor are all integrated in a one-piece lobe design and are all replaced every year. Therefore, one may estimate the allowable concentration levels of the impurity elements by averaging over all components in the lobe. The maximum allowable impurity concentration in the "averaged" TITAN-II FPC are shown in Table 19.5-II and are more easily met than the more restricted levels in the first wall and the blanket Be zone. Comparing the maximum allowable concentration to the expected nominal impurity level (Table 19.5-II), it appears that the concentration limits for all these impurity elements, except niobium and terbium, are readily achievable when the average-limiting concentration levels are imposed. Careful impurity control processes are necessary for Nb and Tb when the structural alloy is fabricated.

The reduced-activation, ferritic steel 9-C used as structural material for the TITAN-II reactor contains tungsten as one of the important alloying elements replacing molybdenum which is an undesirable element for Class-C waste disposal. However, the tungsten content should also be controlled because of the production of a second-step reaction-daughter radionuclide, ^{186m}Re (with a half-life of 200,000 years). The allowable concentration levels of tungsten in the TITAN-II FPC components are also shown in Table 19.5-II.

The "averaged" allowable concentration level of tungsten is 11.0%, more than two orders of magnitude larger than the present tungsten level in the reduced-activation ferritic steels (0.89%).

The waste-disposal ratings of the TITAN-II FPC with nominal level of impurities and 0.89% of tungsten in the presently available reduced-activation ferritic steels (present case), and for controlled impurity levels (controlled case) are presented in Table 19.5-III. These waste-disposal ratings were estimated for the irradiated blanket-lobe assembly averaged after 1 FPY at 18 MW/m^2 of neutron wall loading. It can be seen that with

Table 19.5-II.

MAXIMUM CONCENTRATION LEVELS OF KEY IMPURITIES IN TITAN-II REACTOR COMPONENTS TO QUALIFY AS CLASS-C WASTE[(a)]

Element	Major Nuclide (Activity Limit)	Maximum Concentration in Component[(b)]				Nominal
		FW & Be Zone	Breeder Zone	Shield	Average	Level[(c)]
Nb	^{94}Nb (0.2 Ci/m^3)	0.6	1.7	7.7	2.4	0.1%
Mo	^{99}Tc (0.2 Ci/m^3)	2.6	53	588	22	1.0%
	^{94}Nb (0.2 Ci/m^3)					
Ag	^{108m}Ag (3 Ci/m^3)	0.33	0.83	4.2	1.3	1
Tb	^{158}Tb (4 Ci/m^3)	0.2	1.0	2.4	0.85	5
Ir	^{192m}Ir (2 Ci/m^3)	0.05	0.09	0.2	0.13	5
W	^{186m}Re (9 Ci/m^3)	1.9%	9.1%	116.0%	11.0%	0.9%

(a) Based on operation at 18 MW/m^2 of neutron wall loading for 1 FPY.
Note that a conservative lifetime fluence value of 15 $MW\,y/m^2$ is used for the TITAN-II reference design (0.8 FPY at 18 MW/m^2).

(b) All concentrations in appm expect those noted in atomic percentage.

(c) From Reference [16].

Table 19.5-III.

WASTE-DISPOSAL RATINGS FOR THE "AVERAGED" TITAN-II BLANKET[a]

Element	Present Case: Nominal Level[c] (appm)[b]	Present Case: Class-C Rating	Controlled Case: Controlled Level (appm)[b]	Controlled Case: Class-C Rating
Nb	0.1%	8.33	1[d]	0.42
Mo	1.0%	0.27	6[d]	0.30
Ag	1	0.054	0.07	0.054
Tb	5	1.06	0.1[d]	0.10
Ir	5	0.0077	0.001	0.0077
W	0.9%[e]	0.081	0.9%	0.081
TOTAL		9.78		0.96

(*a*) Based on operation at 18 MW/m^2 of neutron wall loading for 1 FPY. Note that a conservative lifetime fluence value of 15 MW y/m^2 is used for the TITAN-II reference design (0.8 FPY at 18 MW/m^2).

(*b*) All concentrations in appm expect those noted in atomic percentage.

(*c*) From Reference [16].

(*d*) Controlled levels lower than impurity levels in ferritic steel.

(*e*) Present tungsten content in the reduced-activation ferritic steel.

nominal impurity levels (present case), a Class-C waste-disposal rating of 9.78 is estimated for the TITAN-II FPC, with main contributing elements Nb (85%), Tb (11.0%), and Mo (2.8%). These elements should be controlled to levels below those presently available in the structural alloy. In the controlled case of Table 19.5-III, we have presented a possible case for the impurity control: Nb (from 20 to 1 appm), and Tb (from 0.9 to 0.1 appm). The resulting waste-disposal rating is about 1.0 for this controlled case.

Assuming that the structural alloy meets all required levels of impurity and alloying elements as shown in the controlled case in Table 19.5-III, estimates are made the TITAN-II reactor materials and related waste quantities for Class-C disposal. The divertor-shield coverage is taken as 13% in the TITAN-II design, identical to the TITAN-I design. The results are presented in Table 19.5-IV. The annual replacement mass of TITAN-II FPC is estimated at about 71 tonne/FPY (9.1 m^3), assuming that the entire blanket lobe and the divertor shield are replaced every FPY. Note that a conservative lifetime fluence of 15MW y/m^2 is used for the TITAN-II reference design (0.8 FPY at 18 MW/m^2). In addition, Table 19.5-IV is for a modified design with a 0.03-m shield and a 0.17-m blanket breeder zone, rather than the 0.1-m shield and 0.1-m blanket breeder zone of the reference design. The reduced shield thickness in this design decreases the annual replacement mass by about 50 tonne/FPY and also satisfies the structural-design aspects of the blanket lobe. The penalty for this modified design is a 1.5% reduction in the blanket energy multiplication.

The lifetime (30 FPY) components in the TITAN-II design are the TF, ohmic-heating (OH), and equilibrium-field (EF) coils, and the shield for the EF coils. The averaged annual replacement mass from the disposal of these lifetime components is also given in Table 19.5-IV and is estimated to be about 25 tonne/FPY (3.1 m^3). Thus, the averaged annual replacement mass of the entire TITAN-II reactor, both 1-FPY and lifetime components, is about 96 tonne/FPY, or about 12% of the total mass of the TITAN-II FPC and is similar to that of the TITAN-I design.

In addition to the Class-C waste disposal materials and quantities discussed above, the TITAN-II divertor plates are fabricated with W-26Re alloy as in the TITAN-I design (Section 13.7). The annual replacement mass of this non-Class-C waste is about 0.35 tonne/FPY, about 0.4% of the total annual replacement mass. Because of the nitrate salt dissolved in the aqueous-solution coolant, the TITAN-II reactor is also producing ^{14}C from ^{14}N (n,p) reactions. The annual production rate of ^{14}C is about 5.2×10^4 Ci. Using the present 10CFR61 regulations, where the allowable concentration of ^{14}C for Class-C disposal is 8 Ci/m^3 and if ^{14}C remains in the aqueous-solution coolant, the coolant should be replaced at a rate of 7×10^3 tonne/FPY (6.5×10^3 m^3). The replacement mass of the

Table 19.5-IV.

SUMMARY OF TITAN-II REACTOR MATERIALS AND RELATED WASTE QUANTITIES FOR CLASS-C WASTE DISPOSAL[(a)]

Component	Material	Lifetime (FPY)[(a)]	Volume (m^3)	Weight (tonne)	Annual Replacement Mass (tonne/FPY)
First wall	Ferritic steel (9-C)	1	0.26	2.0	2.0
Be zone	Ferritic steel (9-C)	1	2.5	19.7	19.7
Breeder zone	Ferritic steel (9-C)	1	2.0	15.3	15.3
Shield	Ferritic steel (9-C)	1	3.9	30.5	30.5
TF coils	Modified steel		0.54	4.8	0.08
	Copper		3.8	34.0	1.13
	Spinel		0.54	2.2	0.08
	TOTAL	30	4.9	41.0	1.39
OH coils	Modified steel		5.4	49.	1.63
	Copper		38.2	342.	11.4
	Spinel		5.4	23.	0.77
	TOTAL	30	49.0	414.	13.8
EF-coils shield	Modified steel	30	5.6	50.	1.7
Divertor shield	Ferritic steel	1	0.48	3.78	3.78
TOTAL CLASS-C WASTE (lifetime)			334.	2643.	88.1

(*a*) Based on operation at 18 MW/m^2 of neutron wall loading for 1 FPY. Note that a conservative lifetime fluence value of 15 MW y/m^2 is used for the TITAN-II reference design (0.8 FPY at 18 MW/m^2).

coolant can be reduced to about 80 tonne/FPY, if Fetter's evaluation [14] is used as the limiting value (700 Ci/m^3). Because of the large quantities of aqueous solution to be disposed of annually and uncertainties in the transport of the ^{14}C isotope in the primary loop, extraction of the ^{14}C activity from the coolant and disposal of the concentrated quantity as non-Class-C waste should be considered.

19.6. SUMMARY AND CONCLUSIONS

Strong emphasis has been given to safety engineering in the TITAN study. Instead of an add-on safety design and analysis task, the safety activity was incorporated into the process of design selection and integration at the beginning of the study. This approach was projected to enhance the potential of attaining the design goals of design simplicity, passive safety, high availability, and low cost of electricity.

The safety-design objectives of the TITAN-I design are: (1) to satisfy all safety-design criteria as specified by the U. S. Nuclear Regulatory Commission on accidental releases, occupational doses, and routine effluents; and (2) to aim for the best possible level of safety assurance.

The key safety feature of the TITAN-II design is the low-pressure, low-temperature water pool that surrounds the FPC and the entire primary-coolant system. Detailed safety analyses have been performed which show that the TITAN-II pool can contain the thermal and afterheat energy of the FPC and will remain at a low enough temperature so that tritium or other radioactive material in the primary-coolant system will not be released. Therefore, the public safety is assured by maintaining the integrity of the water pool. Since the water-pool structure can be considered a large-scale geometry, the TITAN-II design can be rated as a level-2 of safety assurance design [5,4]. The potential safety concerns are the control of routine tritium releases and the handling of ^{14}C waste.

Plasma-accident scenarios need to be further evaluated as the physics behavior of RFPs becomes better understood. Preliminary results indicate that passive safety features can be incorporated into the design so that the accidental release of plasma and magnetic energies can be distributed without leading to major releases of radioactivity. Activities in this area need to be continued, especially for high-power-density devices. It should be pointed out that for the TITAN-II design, plasma-related accidents are of concern from the consideration of investment protection and would have minimum impact on public safety. This characteristic is again, a result of the presence of the large pool of water that allows the passive protection of the public.

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