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REFERENCE TANDEM MIRROR FUSION BREEDER

REACTOR SAFETY ASSESSMENT

I. MAYA, C. G. HOOT, C. P. C. WONG, K. R. SCHULTZ,

GA TECHNOLOGIES, INC.
P. O. BOX 85608
SAN DIEGO, CA 92138

J. K. GARNER, S. J. BRADBURY, W. G. STEELE,
AND D. H. BERMALD

TRW
TRW ENERGY DEVELOPMENT GROUP
ONE SPACE PARK
REDONDO BEACH, CA 90278

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PREFACE

This report represents Volume 2 of 6 volumes directed towards the resolution of special topics associated with a reference fission-suppressed tandem mirror fusion breeder reactor design developed during 1981-1982 (D. H. Berwald, et. al., "Fission-Suppressed Hybrid Reactor - The Fusion Breeder," Lawrence Livermore National Laboratory, UCID-19638, 1982. A complete list of the special topics reports follows:

- Volume 1: Liquid Metal MHD Pressure Drop Effects In The Packed Bed Blanket by T. J. McCarville, D. H. Berwald, and C. P. C. Wong.
- Volume 2: Reactor Safety Assessment by I. Maya, C. G. Hoot, C. P. C. Wong, K. R. Schultz, J. K. Garner, S. J. Bradbury, W. G. Steele, and D. H. Berwald.
- Volume 3: Beryllium Lifetime Assessment by L. G. Miller, J. M. Beeston, B. L. Harris, and C. P. C. Wong.
- Volume 4: Structural Analysis by G. Orient, R. A. Westmann, and N. Ghoniem.
- Volume 5: Neutronics Issues and Optimization by J. D. Lee.
- Volume 6: Materials Compatibility Issues and Experimental Results by J. H. DeVan and P. Tortorelli.

Since compiling this report on the reactor safety prospects for the fusion breeder, a new safety-related concern, has been identified. Namely, during a thermal transient with the beryllium pebbles in place, volumetric growth of the pebbles leading to structural deformation and failure is a concern. This phenomenon, identical to that experienced in post-irradiation

annealing measurements of beryllium swelling due to internal helium gas formation at the lower irradiation temperature, should be studied in detail to determine the relevant time constants and its full implications.

The most likely implication is that either coolant flow must be maintained in the module (via the primary coolant loop or the safety coolant loop) or the dump valve must open, allowing the fuel to be removed from the module and cooled in the dump tank. Situations in which the dump valve fails to open and fuel remains in the blanket with no coolant or no coolant flow can now be expected to result in a loss of module integrity. An additional coolant line at the bottom of the module may be required to provide valve redundancy to address a loss of primary coolant flow event in which the dump valve fails to open, but the safety coolant system remains operable. In all, the beryllium growth concern is expected to be addressable via the design of redundant safety systems.

Another issue identified in the report is fission product release to the primary coolant via recoil from the unclad fuel. The extent and implications of this issue are unclear and design fixes (e.g., thin coatings) have been proposed. Since completion of the report, we have also identified a 1969 ORNL study, "Unclad-Metal Breeder Reactor (UMBR) For Desalting Or Power" (R. P. Hammond, et al., ORNL-4203), which addresses the same issues. The reactor proposed by the ORNL investigators would have 100 times the fission burnup and 10 times the fission product release estimated for the fusion breeder. The ORNL study concluded that fission products in the loop are a concern, but can be controlled by cold trapping and other methods.

Finally, it should be recognized that the fusion breeder design and its safety analysis are at an early stage of development. At this point, the component failure probabilities and consequences are estimates based upon nuclear plant operating experience related to fission reactors. Similarly, it is clear that additional failure paths have yet to be identified. Nevertheless, this study has been very useful in beginning to develop the design philosophy and safety logic for the fusion breeder.

D. H. Berwald
September, 1984

ABSTRACT

The safety features of the reference fission suppressed fusion breeder reactor are presented. These include redundancy and overcapacity in primary coolant system components to minimize failure probability, an improved valve location logic to provide for failed component isolation, and double-walled coolant piping and steel guard vessel protection to further limit the extent of any leak. In addition to the primary coolant and decay heat removal system, reactor safety systems also include an independent shield cooling system, the module safety/fuel transfer coolant system, an auxiliary first wall cooling system, a passive dump tank cooling system based on the use of heat pipes, and several lithium fire suppression systems. Safety system specifications are justified based on the results of thermal analysis, event tree construction, consequence calculations, and risk analysis. The result is a reactor design concept with an acceptably low probability of a major radioactivity release. Dose consequences of maximum credible accidents appear to be below 10CFR100 regulatory limits.

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A OVERVIEW OF REACTOR SAFETY ASSESSMENT

A.1 Introduction

This volume presents an assessment of the safety features of the reference fission-suppressed fusion breeder reactor. It builds upon safety analyses performed previously (1-3) and assesses the reactor safety prospects of the fusion breeder using improved thermal transient models, updated data estimates, and additional analysis. Following the philosophy established in the earlier references, this effort's focus was on blanket safety since this is the location of the largest radioactive hazard. Specifically, the analysis was concentrated on off-normal events that could result in mobilization of the blanket and/or primary loop radioactivity, and on the design of systems to prevent such occurrences. The design and assessment of safety systems to address other parts of the plant (e.g., the end cell afterheat, radioactivity in the fuel handling system, magnet safety) has not been considered, but the issues appear to be tractable.

The approach taken to accomplish this most recent safety assessment consisted of the following:

- Perform the thermal analyses of the reference blanket under a variety of safety related transient conditions (Section B).
- Reconsider the effectiveness of the shield cooling system and investigate an auxiliary first wall cooling system (using cover gas flow through the plasma chamber) (Section B).
- Perform additional thermal analysis to investigate the attractiveness of a passive dump tank cooling system using heat pipes and air heat exchangers (Section B).
- Assess fission-product level in the primary coolant (Section C).

- Use the techniques of probabilistic risk assessment (PRA) in conjunction with the results of the thermal analysis to evaluate the risk of potential accident sequences (Section D).

On the basis of the above tasks, the requirements for and the designs of specific safety systems can be assessed by reviewing component logic and identifying the major risk contributors. The conclusions and recommendations for future design work are presented in Section D. The remainder of Section A presents a brief summary of the reactor safety systems and the principal failure modes which were considered.

A.2 Reactor Safety Logic

The systems required for routine operation of the fusion breeder which additionally serve a reactor safety function are:

- The primary coolant and decay afterheat removal systems.
- The independent shield cooling system.
- The independent module safety/fuel transfer cooling system.

Figure 1, a schematic of the primary loop, includes redundancy and overcapacity in the primary coolant system components to minimize failure probability, an improved valve location logic to provide for failed component isolation, and double-walled coolant piping to further limit the extent of any leak. The shield cooling system, for which a detailed mechanical design has not yet been attempted, provides the necessary function of shield cooling during normal operation and is the first backup to the primary loop during most off-normal events. The module safety/fuel transfer cooling system, shown in Figure 2, is normally used to cool the fuel handling circuit during the transfer of thorium fuel from the module to the fuel reprocessing plant. During off-normal events, it provides the primary coolant loop for the dump tank. Both of the above coolant systems utilize liquid lithium, but are

* (REQUIRED NUMBER OF
COMPONENTS/NUMBER OF
COMPONENTS AVAILABLE)

Fig. 1. Primary coolant and decay afterheat removal system,
showing updated components

Fig. 2. Fuel handling (dump) system and module safety coolant system

independent of the primary loop. That is, there are no shared pipes (except the module itself in the case of the module safety/fuel transfer cooling system) and no shared pumps, heat exchangers, or heat sinks so that common mode failures are minimized.

Several additional dedicated safety systems have been incorporated which are not required for routine operation. These include the following:

- The auxiliary first wall cooling system.
- A passive heat pipe cooling system for the dump tank.
- Several lithium fire control and suppression systems.
- Miscellaneous safety systems, including systems to remove tritium and other radioactive products from the reactor and containment building atmospheres.

The auxiliary first wall cooling system would circulate the reactor building cover gas (helium) through the vacuum vessel to cool a limited number of failed blanket modules under certain conditions (see later discussion). The passive heat pipe cooling system for the dump tank provides a backup to the module safety/fuel transfer cooling system discussed above.

There are several lithium fire control and suppression systems which can be invoked. A partial list follows:

- An inert gas environment in the reactor building.
- Steel guard vessels connected to deep, narrow sumps with sloped surfaces in the reactor building floor below the individual blanket modules and the lithium piping.
- Steel liners for concrete structure in the reactor building.
- A sacrificial material between the steel and concrete.
- Steel balls and hollow graphite microspheres in or injected into potential spillage areas.
- Active lithium pool surface cooling.

- Active chemical fire fighting methods.

Lithium fires can increase the overall risk if they lead to new accident pathways or more severe consequences for existing accident pathways. For the purpose of the following fusion breeder safety assessment, we have assumed that, through the use of the above systems, lithium fires would not be a major risk contributor.

In this assessment, we have concentrated upon decay afterheat as the primary mechanism for radioactivity release. Safety system specifications were justified based upon the results of thermal analysis, event tree construction, consequence calculations, and risk analysis. The goal is a reactor design, with very low probabilities of major radioactivity release, whose dose consequences for maximum credible accidents are below 10CFR100 regulator limits.

On the basis of the thermal analysis and the safety analysis of reference 1, the design of the safety systems was improved by reviewing component logic and identifying the major risk contributors using the techniques of probabilistic risk assessment. Four accident types were analyzed:

- Loss of all primary coolant pumps or heat exchangers
- Failure of inlet/outlet piping or the cold or hot manifold
- Failure of the module coolant distribution piping
- First wall failure

Event trees were constructed, probabilities assigned, and consequences calculated as previously (2,3).

In the event of loss of single primary loop components such as a single pump or heat exchanger, the design redundancy and overcapacity allow for continued operation at full load after momentary shutdown or power reduction to isolate the failure and allow the remaining units to pick up the load. The valving arrangement shown in Fig. 1 allows for the necessary isolation and reconfiguration. Alternately, the reactor can be shut down without any loss

B. SAFETY SYSTEMS MODELING

Updated and improved thermal models of safety systems identified in the 1982 study of a reference tandem mirror fusion breeder(1) are presented in this section. Analysis for recently proposed safety systems (e.g., auxiliary first wall cooling system, passive heat pipe cooling system for the dump tank) are also presented.

B.1. Auxiliary First Wall and Shield Cooling Systems

The capabilities of shield and/or auxiliary first wall cooling systems to remove decay afterheat with and without the beryllium/thorium fuel in place were studied. The shield is assumed to be cooled using an isolated liquid metal coolant circuit. The auxiliary first wall cooling system is assumed to consist of a large fan(s) located at the end of the central cell (or in the end cell) which, in the event of an accident, would circulate the reactor building cover gas through the central cell to cool the first wall. As use of the latter system could, in many cases, result in a loss of several grams of tritium to containment, such use is reserved for relatively rare and off-normal events rather than routine operation such as blanket changeout.

B.1.a. Heat Transfer Models

Loss of coolant and loss of coolant flow accidents have been analyzed for variations in the ability to cool the first wall as well as variations in the emissivities of steel and beryllium. The heat transfer analysis were performed using SINDA (4) with one-dimensional cylindrical models similar to those used in the 1982 reference design report (1). The fuel in place thermal model uses 17 zones, as pictured in Figure 3, and the empty reactor model uses 9 zones, as in Figure 4. The empty reactor model is assumed to be radiation dominated, and the fuel in place model uses empirical methods taken from Kunni and Suzuki (5), and Baddour and Yoon (6).

Heat transfer in a packed bed containing a helium cover gas static (i.e., loss of coolant, fuel in place) is treated as an effective thermal conductivity, K_e , of the bed. The effective thermal conductivity is assumed

Fig. 3. Blanket Thermal Model With Fuel In Place (Case 3).

Fig. 4. Blanket Thermal Model With Fuel Removed.

to be made up of the sum of conduction through the gas and through contact between bed particles, K_c , and a radiation term, K_r . Baddour and Yoon (6) indicate that:

$$k_c = 4.444 K_f^{.7927} K_s^{.2073}$$

where

$$\begin{aligned} K_f &= \text{the conductivity of the fluid (helium)} \\ &= 1.645E-4T + .148 \end{aligned}$$

$$\begin{aligned} K_s &= \text{the conductivity of the solid (beryllium)} \\ &= .8 (-.0806T + 172) \end{aligned}$$

over the range of K_f/K_s relevant to the hybrid blanket. Kunni and Suzuki (5) show that the effective thermal conductivity, K_c , drops off near the walls of a packed bed. Their data indicate that the conductivity near the wall, K_{cw} , is given by:

$$K_{cw} = 2.081 K_f^{.8228} K_s^{.2073}$$

in the near wall region (within one ball radius of the wall) for relevant values of K_f/K_s . The radiation term, K_r , is not assumed to change near the wall and was taken from Baddour and Yoon as:

$$K_r = 1.52 \times 10^{-5} (\epsilon d T^3)^{.7} K_s^{.3}$$

where

$$\begin{aligned} \epsilon &= \text{the emissivity of beryllium} \\ d &= \text{the ball diameter} \end{aligned}$$

thus we assume that:

$$\begin{aligned} K_e &= K_c + K_r && \text{away from the walls} \\ K_e &= K_{cw} + K_r && \text{within one ball radius of the walls} \end{aligned}$$

Heat generation in the bred fuel is dominated by ^{233}Th , ^{233}Pa , and fission products and was taken directly from the 1982 reference design study (1). While the afterheat generation in HT-9 structure can be ignored with the fuel in place, first wall cooling requirements for the empty reactor are dominated by HT-9 afterheat values. The HT-9 structure decay heat was estimated using results from work by Blink and Lasche (7,8) for an HT-9 first wall in an inertial confinement fusion reaction. In this analysis, the same decay rate was used throughout the blanket, but the initial heat generation (W/cm^3 in HT-9) was assumed to drop off in proportion to heat generation with the plasma on. The following expression was used:

$$P_0(r) = P_0 \exp - 11.5 (r - r_0)$$

where

$$\begin{aligned} r &= \text{radial position} \\ r_0 &= \text{first wall radius} = 1.5 \text{ m} \\ P_0 &= \text{decay power at first wall at time } 0.0 \end{aligned}$$

The magnitude of the peak afterheat generation in the first wall at reactor shutdown is less certain than the decay rates (Table 1) and the spatial attenuation. Our approach has been to consider conservatively large values, 0.25 and 0.5 W/cc. Based on linear scaling with first wall neutron loading, the MARS LiPb blanket (9,10) would result in an initial first wall afterheat power generation of 0.13 W/cc at the reference hybrid blanket wall loading of 1.3 MW/m^2 . The ICF blanket studied by Blink and Lasche (7) would produce approximately 0.25 W/cc of afterheat at 1.3 MW/m^2 . The results presented in the following section indicate that afterheat cooling system configurations will not change with lower afterheat values.

TABLE 1. Principal Isotopes Contributing to
Afterheat in HT-9.

Isotope	Fraction At Shutdown	Half Life
^{28}Al	0.02	2.24 min
^{52}V	0.08	3.76 min
^{54}Mn	0.16	312.5 d
^{55}Fe	0.005	2.7 y
^{56}Mn	0.74	2.58 h

B.1.b Auxiliary First Wall Cooling System

An auxiliary first wall cooling system could be used during four possible coolant accidents:

- Case 1. Loss of coolant flow with fuel in place,
- Case 2. Loss of coolant flow with fuel removed,
- Case 3. Loss of coolant with fuel in place, and
- Case 4. Loss of coolant with fuel removed.

Cases 1 and 2 were shown to present no problems in the 1982 report (1) and were not analyzed here. Specifically, conduction in the lithium allows good temperature control using either a separately cooled shield or an auxiliary first wall coolant system. Cases 3 and 4 have been re-analyzed including first wall cooling, updated afterheat generation rates in the steel, and updated empirical formula for packed bed heat transfer (Case 3 only).

In the analyses of Cases 3 and 4, helium has been chosen as the inert cover gas for the reactor building atmosphere. Helium has superior heat transfer characteristics (advantageous both in the packed bed and first wall

auxiliary cooling system), does not activate and is already present in large quantities in the cryogenic magnet cooling system. The auxiliary first wall cooling system consists of a large fan (or fans) mounted in one or both of the end cells with vacuum doors (for inlet and exhaust) in the opposite end cell(s). During an accident the doors are opened, the fan turned on to blow the helium cover gas from the building atmosphere through the central cell to cool the first wall, and the tritium bearing cryo-panels are isolated from the gas flow.

Our results indicate that the building atmosphere cannot be used to cool the entire inventory (Case 3) for very long because it would heat up too much, causing damage to equipment in the building. Calculations indicate that, due to decay of the afterheat, removing about 35 MW of the initial 175 MW of afterheat in the fuel at shutdown will be adequate to control structure and fuel temperatures. However, assuming 10^5 m^3 of helium in the building, 35 MW would heat it adiabatically at 0.5°C/s , resulting in unacceptable temperatures for electronic equipment in a few minutes. If the central cell helium velocity were 44 m/s (to keep the blanket structure below 900°C as discussed below) the helium would heat up 122°C per pass at 35 MW, possibly damaging equipment near the exit. However, this type of auxiliary first wall cooling system is suitable for accidents with fuel in place in only one or a few modules, and for accidents with the fuel successfully removed (Case 4).

Figures 5 through 7 show blanket temperatures for Case 3 (loss of coolant with fuel in place). The envisioned accident would involve mechanical damage to a single blanket module, causing the lithium to drain out (uncovering the fuel) and possibly preventing dumping the fuel to the dump tank. Figure 5 shows blanket temperatures as a function of time for a loss of coolant accident with fuel in place with both the shield maintained at 300°C and the first wall cooled. These results indicate that a vigorous first wall cooling system could prevent melting and further mechanical damage to the module. Blowing helium down the central cell at 44 m/s would keep the hottest part of the major structure, the intermediate wall containing the fuel, below 900°C . (The HT-9 structure is expected to have enough residual strength at 900°C to support the static blanket loads).

Fig. 5. Case 3, Loss of Coolant With Fuel in Place. Variations in Helium Velocity.

Fig. 6. Case 3, Loss of Coolant With Fuel in Place. Variations in Emissivity.

Fig. 7. Case 3, Loss of Coolant With Fuel in Place.
Only One of Two Cooling Systems.

A 44 m/s helium velocity would require roughly 90 kw (120 hp) for a 50% efficient fan. A 3 cm beryllium sphere would experience a 0.1 g acceleration in such a "wind". The hottest point in the blanket would reach about 965°C, but this may be tolerable since the small amount of structure in the region may not be required to support the weight of the fuel. The actively cooled first wall remains below 272°C after initial cooling and maintains excellent physical integrity.

Figure 5 also shows the effect of lowering helium velocity from 44 m/s to 10 m/s on intermediate wall and hottest node temperatures. The intermediate wall remains below 900°C for 10 hours in this case.

Emissivity of the beryllium and steel were taken to be 0.25 in both cases. Emissivity of steel and beryllium after wetting and corrosion by lithium is uncertain, but is expected to be between 0.25 and 0.5 (1,9). Parameterizations of emissivity indicate a weak dependence for beryllium, since conduction dominates heat transfer in the packed bed, and a stronger dependence for steel emissivity, since radiation dominates heat transfer to the first wall and shield. Figure 6 shows the intermediate wall temperatures for various emissivities with helium cooling at 10 m/s. Ten m/s was chosen because it illustrates parameterization of the emissivities adequately and to show that it is sufficient to cool the structure at the optimistic emissivity values. A steel emissivity of 0.5 would keep the first baffle (Zone 2) below 870°C.

Figure 7 gives hottest blanket temperatures with the first wall cooled only ($v = 10$ m/s), and the shield cooled only, with steel and beryllium emissivities of 0.25. The first wall auxiliary cooling allows more time before damage occurs; eight hours to 900°C versus three hours for the shield cooled only case. The shield is modeled as a constant temperature (300°C) node at the back of the blanket. These results show that both first wall and shield cooling systems are required to control temperatures in a Case 3 accident.

An auxiliary first wall cooling system for Case 4, loss of coolant with fuel removed, indicates adequate cooling with a helium velocity of less than 0.2 m/s. Results for Case 4 are shown in Figure 8 for auxiliary first wall cooling only (an adiabatic boundary condition was taken at the shield). These indicate that a 10 m/s helium velocity will control first wall temperatures at

Fig. 8. Case 4, Empty Reactor. Variations in First wall Decay Heat Levels.

operational levels ($\sim 500^{\circ}\text{C}$) with a low steel emissivity of 0.25, and that a helium velocity of only 0.2 m/s is sufficient to keep the reactor below 700°C , allowing reuse. Note that the higher velocities (> 10 m/s) required for Case 3 would give very good temperature control in a Case 4 accident, even without shield cooling. Also, recall that the HT-9 initial first wall afterheat value used of 0.25 W/cm^3 is probably conservative.

B.1.c Shield Cooling System

The shield is assumed to be cooled with a separate cooling system which maintains the back of the blanket at 300°C during accident conditions and at 500°C (the blanket coolant outlet temperature) during normal operations. This approach was pursued because, during normal operation, a 200°C ΔT and good thermal contact between the shield and blanket is not feasible. The proposed remedy is to operate the shield at $\sim 500^{\circ}\text{C}$ during normal operation and cool it to $\sim 300^{\circ}\text{C}$ during an accident. Candidate shield coolants, which are chemically compatible with lithium, include lead-lithium, sodium, potassium and lithium itself.

Another possibility is to operate the shield at $\sim 500^{\circ}\text{C}$ and use it as a safety system only during loss of flow accidents (Cases 1 and 2) when the lithium conductivity provides sufficient thermal contact. The auxiliary first wall cooling system can then provide cooling in Case 4 accidents (Figure 8) and time to respond in Case 3 accidents (Figure 7). Conversely, the 1982 reference design report (1) and current calculations (Figure 9) indicate that if the back of the blanket can be kept at 300°C by cooling the shield, a first wall cooling system is not required in Case 4.

Recent design iterations have resulted in a revised lithium outlet temperature of 420°C for the reference design. In Case 3 accidents, a 420°C shield with good thermal contact will raise blanket temperatures $20\text{-}40^{\circ}\text{C}$, requiring slightly higher auxiliary first wall helium velocities if the steel emissivity is low. Figure 9 shows that a 420°C shield temperature will keep the first wall below $\sim 655^{\circ}\text{C}$ even with pessimistic values for steel emissivity (0.25) and initial afterheat (0.5 W/cc) in Case 4 accidents. Since the 500°C shield temperature is adequate in Cases 1 and 2, the 420°C operating temperature will allow the shield to be kept at one temperature during operation and all four accident scenarios.

Fig. 9. Case 4, Empty Reactor. Variations in Emissivities.

B.1.d Conclusions Concerning Auxiliary First Wall and Shield Cooling System.

The preceding discussion suggests several capabilities for both the shield and first wall cooling systems, as shown in Table 2. The shield can be designed to keep the back of the blanket at $\sim 300^{\circ}\text{C}$ (or 420°C) during accidents, making it useful in the more troublesome Cases 3 and 4 (loss of coolant) or it can be used only in Cases 1 and 2 (loss of flow) where thermal contact is assured and the shield temperature can be higher. The first wall cooling system (in conjunction with shield cooling) can be made adequate for Case 3 accidents. However, Case 3 accidents will require both the 300°C (or 420°C) shield and the higher velocity first wall cooling system. Since it appears that Case 4 can be handled with either the 300°C (420°C) shield system or a modest first wall cooling system, the possibility of a Case 3 accident is the main motivation for supplying both systems. Two systems would also provide redundancy in Cases 2 and 4.

TABLE 2. Safety System Requirements For Various Accident Scenarios.

	Shield Cooling Alone	Auxiliary First Wall Cooling Alone	Both Systems
Case 1, LOFA with fuel	500°C OK*	Not adequate	--
Case 2, LOFA without fuel	500°C OK	(Not analyzed - appears feasible)	--
Case 3, LOCA with fuel	Not adequate (3 hrs @ 300°C)	Not adequate (8 hrs @ 10 m/s)**	300°C *** and 44 m/s ($\epsilon = .25$)
Case 4, LOCA without fuel	420°C OK	0.2 m/s	--

* shield temperature

** helium velocity in central cell

*** will work at 420°C shield with higher helium velocity

B.2 Passive Dump Tank Cooling Systems

B.2.a Introduction.

During an off-normal event that disables other safety systems, the reactor fuel inventory would be gravity dumped to tanks beneath the reactor and the fuel afterheat would be removed at the dump tank, with a passive heat removal system being most attractive for such a scenario. In our investigation of passive cooling systems for the dump tanks, thermal capacitance, natural convection cooling loops, and heat pipes were considered. Earlier studies indicated that thermal capacitance requires prohibitively large volumes and will not continue to work indefinitely. *Natural convection loops or heat pipes can cool the dump tanks adequately, and are similar in many respects.*

For either cooling arrangement, the selection of a working fluid is compatible with lithium is a primary concern. In a convection loop, freezing must be prevented at all times, in all parts of the loop. Mercury has been identified as meeting the requirements of not freezing and not reacting with lithium. The heat pipe contains only a small amount of liquid that remains in the dump tank when cool and freezing does not plug the system. Since the heat of the fuel entering the dump tank can, if necessary, melt the heat pipe coolant and start the pipe working. Thus the heat pipe could use other working fluids, such as sodium, that are not liquid at outside air temperatures.

The heat pipe system will also require much less working fluid than the convection loop. A 35 MW mercury convection loop would require about 60 m³ of mercury and a mercury heat pipe system would require about 12 m³. The identified disadvantages of the convection loop system are not great, and further study may disclose advantages. However, the heat pipe system was chosen for this study.

Outside air heat exchangers have been chosen for final heat rejection. Cooling towers are active systems, and cooling ponds or other water systems are difficult to configure with heat pipes and convection loops. Gravity assisted heat pipes (or convection loops) require that the water be above the dump tank. Horizontal heat pipes are feasible, but the heat pipe operating temperature of 500-600°C will make heat exchange with water more complex. Air

heat exchangers will be large compared with a water system, but result in simple and reliable passive cooling systems.

B.2.b. Heat Pipe Cooling System.

Heat pipes have been investigated for similar applications for cooling the fast flux test facility (FFTF) spent fuel (11) and for cooling LWR containment buildings during severe accidents (12). Vertical gravity assisted heat pipes with mercury, sodium and/or potassium working fluid are suitable for this application. Figure 10 shows the fuel temperature in the dump tank as a function of time assuming heat removal at various constant rates by heat pipes. The dump tank is assumed adiabatic otherwise and only the heat capacity of the thorium, beryllium and lithium is included. A 35 MW heat removal system is adequate for an initial heat generation rate of 175 MW. (Fuel afterheat is taken from Reference 1). If less than the entire reactor inventory is involved, a proportionately smaller system is required.

Vertical gravity assisted heat pipes have critical lengths dependent on the wick type, wick structure, and the working fluid. Ranken and Lundberg (13) give a critical length of 12.7 m for sodium in a knurled surface heat pipe. They also tested a spiral trough system for vertical heat pipes which feeds the area of wick (knurled surface) immediately below it, thus allowing heat pipes of indefinite height. Because of the high heat fluxes in the evaporators and condensers of heat pipes, heat exchange between pipes is very good, making it quite practical to use two or more pipes in series to reach the required height. It may be desirable to use two heat pipes in series in the fusion breeder to eliminate the direct path from the activated fuel to the environment. However, the single heat pipe would have to break at both ends for a radioactive release to occur. A single, 35 m long, knurled inner surface, spiral trough heat pipe system is investigated here. Mercury, sodium, and potassium working fluids are compared.

Reference 11 investigates mercury heat pipes operating at relatively low temperature (240-340°C) for FFTF spent fuel cooling. They tested 7.3 m long by 5.1 cm o.d. pipes to 27.8 kw (2.27 kw/cm² through the vapor passage). Kemme et.al., (11) also proposed larger heat pipes (17 cm o.d.) operating at 340 kw (1.22 kw/cm²) with a knurled inner surface for the wick. A mercury

Fig. 10. Fuel Temperatures at Various Heat Removal Rates.

heat pipe operated at higher temperature (500-550°C) is expected to be much more efficient. For the mercury pipe we have baselined 15 cm o.d. and 250 kw (1.6 kw/cm^2).

Sodium and potassium appear to perform similarly to mercury. Reference 11 indicates 2.37 kw/cm^2 for potassium at 640°C and 2.0 kw/cm^2 for sodium at 650°C. The heat carrying capability of heat pipes is very temperature dependent, with highest fluxes occurring toward the upper end of the operating temperature range. Reference 12 shows a slope of approximately $.016 \text{ kw/cm}^2\text{°C}$ for mercury at 250-310°C. Given adequate chemical compatibility, extrapolating to 500°C yields 4.9 kw/cm^2 . Higher heat fluxes have been attained; Holman (14) lists 25.1 kw/cm^2 for mercury at 360°C. Mercury appears to be the most desirable working fluid for this application since it is operating at the upper end of its temperature range (190-500°C) rather than the lower end for sodium (550-900°C) and potassium (400-800°C). It does not react with air or water.

One hundred forty 15 cm o.d. mercury heat pipes operating at 500-550°C and carrying 250 kw each will remove the necessary 35 MW from the dump tanks. This system is comparable in size to that proposed in Reference 12 for LWR accidents. They proposed 100 11.3 cm o.d. water heat pipes removing 250 kw each for a total of 25 MW. The required mercury inventory is 12 m^3 or 155,000 Kg (\$1.3M as of July 1983) (15). If the mercury temperature were reduced by 200°C to avoid chemical compatibility issues, the number of heat pipes would increase ~ 3 fold.

Heat pipe spacing in the dump tank is difficult because of the conduction drop in the thorium-lithium-beryllium mix and the high heat fluxes obtained in the heat pipe evaporator. Rough calculations indicate that 140 pipes in a 4 m deep x 2 m x 107 m dump tank (or tanks) result in a 1.25 m heat pipe spacing and an approximate 900°C peak temperature neglecting convection. Reference 12 proposes angling the evaporator from the vertical, which could be used to greatly increase the evaporator surface area in the dump tank if necessary.

B.2.c Air Heat Exchangers.

Exchanging heat with the outside air at the condenser ends of the heat pipes works fairly well because of the large (~ 470°C) temperature difference. A ten meter high condenser will require 45 vertical fins 20 cm

wide (by 10 m high) to remain at 500°C in no wind at 35°C air temperature (fin efficiency of 54%). Reference 12 uses 11.35 m condensers with horizontal fins and assumes a 3 m/s wind velocity. The vertical fins are inefficient in wind but will cool adequately by natural convection ($h = 5.83 \text{ W/m}^2\text{°C}$) in a no wind condition. The inner condenser surface operates at $< 6 \text{ W/cm}^2$. The 140 heat pipe condensers would have an outer diameter of 55 cm (including fins) and would rise 10 m above the reactor building roof. Making the condensers slightly shorter may be possible, but fin efficiency drops off very rapidly with length and the trade off does not appear to be favorable. It is concluded that, although further optimization is required, a passive dump tank heat removal system is possible with today's technology.

C PRIMARY COOLANT ACTIVITY

The radioactivity in the primary coolant of nuclear reactors under normal operating conditions is important since it provides the source term for any leakage to the containment and eventually to the environment.

C.1 Regulatory Considerations

The allowable dose rates to the public at the Exclusion Area Boundary (EAB) from normal operation are set by 10CFR50 Appendix I as shown in Table 3. These limits are 5 mrem/yr whole body dose and 15 mrem/yr to any organ (for which ^{131}I is usually limiting because of the importance of the infant thyroid dose). Less restrictive limits in 10CFR20 apply to occupational exposure to workers within the plant as noted in the table. These exposure limits and the dose rate determine the amount of time a worker can spend in maintenance operations.

To translate these limits for normal operation in mrem/yr into maximum allowed circulating activity levels requires the specification of the primary circuit leak rate, the containment vent or leak rate, site meteorology, and the containment filter efficiencies. For example, for the low enrichment uranium fueled 2240 MW(t) HTGR (Ref. 16), the limit is 14,000 Ci of ^{88}Kr in the design activity mix. This assumes a PCRV leak rate of 0.01%/day, containment purge rate of 0.5 vol/hr, filter efficiencies of 95% for iodine and 99% for particulates, and a site with an annual average EAB dilution factor of $8 \times 10^{-6} \text{ s/m}^3$.

The plateau activities in the primary circuit impose limitations on the amount of time a person can spend on maintenance operations as (see Table 3). The limit per person is 5 rem/yr from 10CFR20. In addition, restrictions based on industrial practice have been recommended (Ref. 16). Typically, these limit the dose rate for occupational exposure to 100 mrem/hr, and the occupational population dose to 200 man-rem/yr/GWe.

Typical primary circuit activity levels observed in operating HTGR, LWR, and LMFBR plants are given in Table 4. It is seen that noble gas fission

TABLE 3 Radiation Exposure Constraints for Primary Circuit Activity

	Personnel Exposure	Offsite Public Exposure
Circulating noble gases (principally ^{88}Kr and ^{133}Xe)	100 mrem/week/person, whole body external dose, 10CFR20	5 mrem/yr, whole body exposure, Appendix I 10CFR50
Circulating iodines (^{131}I , ^{132}I , ^{133}I)	MPCs limited per Reg. guide 8.8 and 10CFR20	15 mrem/yr, infant milk chain thyroid dose 10CFR50, Appendix I
Plateout activity (Cs, Ag, I, etc.)	5 rem/yr/person, maintenance dose rates <100 mrem/hr, collective exposure <200 man-rem/yr/GWe	Not applicable

TABLE 4 Typical Primary Circuit Activity Levels [Ci/MW(t)]

	HTGR	LWR	LMFBR
Circulating			
^{88}Kr	0.1	0.02	N/A
^{133}Xe	0.02	0.14	N/A
^{131}I	$<7 \times 10^{-4}$	8×10^{-3}	4×10^{-5}
^{24}Na	--	--	9000
^{54}Mn	--	1×10^{-15}	N/A
^{60}Co	--	3×10^{-15}	N/A
^{137}Cs	--	2×10^{-3}	2×10^{-3}
Plateout			
^{131}I	4×10^{-2}	N/A	N/A
^{137}Cs	2×10^{-3}	N/A	N/A
^{90}Sr	5×10^{-4}	N/A	N/A

N/A = not available.

products such as ^{88}Kr and ^{133}Xe are the main activities in HTGRs and LWRs, while the activation of the sodium coolant to ^{24}Na gives the dominant activity in LMFBRs. For this comparison, values from literature for the Fort St. Vrain HTGR, the Ginna PWR, and the Phenix LMFBR were used. Table 5 compares the design levels of circulating activity for an HTGR (Fort St. Vrain), a PWR (standardized reference design large PWR), and an LMFBR (Clinch River Breeder Reactor) plant. Note that the design levels are typically much higher than the achieved levels. Nevertheless, these levels reflect values for which the designers believe the regulatory requirements could be met.

C.2 Analysis of the Fission Product Inventory in the Primary Coolant.

The very low burnup ($\sim 400 \text{ MWD/MTM}$) thorium fuel is in direct contact with the lithium coolant in the reference fusion breeder blanket (1). The fuel form is displayed in Figure 11. The thorium clip dimensions are as follows:

- beryllium pebble radius - 1.5 cm
- snap-ring width - 1.0 cm
- snap-ring thickness - 0.075 cm

For the calculations to follow, the thorium surface area is about 20.4 cm^2 and its volume is about 0.71 cm^3 . Since the fuel and lithium are in direct contact, recoil, knockout, erosion and corrosion can transport fission products from the fuel to the coolant. Erosion and corrosion of the activated structure and tritium breeding contribute to the activity of the lithium coolant and, though mass transfer, activated products can deposit in the heat exchanger and other primary loop components.

In this section, recoil and knockout are examined, a comparison is made to design allowables used for the Clinch River Breeder Reactor Plant (CRBRP), and control methods are considered. The erosion and corrosion of thorium by liquid lithium has not been assessed.

TABLE 5 Primary Coolant Design Activity Levels (Ci)

	Fort St. Vrain [842 MW(t)]	Standardized PWR [3400 MW(t)]	CRBRP [975 MW(t)]
^{88}Kr	4020	50	--
^{133}Xe	695	4495	--
^{131}I	5.3	67	3×10^4
^{137}Cs	5×10^{-4}	4.5	5×10^4
^{90}Sr	1×10^{-3}	2×10^{-3}	44
^{24}Na	--	--	2×10^7

Fig. 11. Beryllium Pebble/Thorium Snap Ring Fuel Form for the Reference Fusion Breeder.

C.2.a Recoil and Knockout.

The release mechanisms examined here are recoil and knockout. These are temperature independent release processes associated with the kinetic energy of a fission fragment. If the newly formed fragment intersects a surface of the solid before its kinetic energy is depleted, it is released to the coolant. This release process is termed direct recoil, and for typical fission fragments, can occur within a surface layer thickness comparable to the range of the fragment ($\sim 10^{-3}$ cm).

Knockout occurs as a result of the displacement cascade occurring in the matrix atoms due to the exchange of kinetic energy from the fragment to the matrix. One fission fragment can have around 30 primary collisions with the matrix. Each of these displaced matrix atoms can lead to numerous secondary and higher order atoms (~ 1000) being displaced. If these displaced matrix atoms are situated close enough to a surface, they can be released into the coolant. The higher order knock-ons dominate this release mechanism. Due to their greatly reduced kinetic energy, this process is effective only within 5×10^{-7} cm of the surface.

Direct recoil can account for the release of fission products directly into the primary coolant without requiring thermal diffusion influences. The knockout process can result in the release of any type of atom present either in the matrix or in an interstitial site. For the fuel form described here, knockout was found to be negligible when compared to recoil for the introduction of activated product into the coolant.

Release Fraction from Recoil. A simple approximation for the fission fragment flux (I_i) emerging from the unclad thorium is

$$I_i = \frac{P_i \mu}{4} \frac{1}{\text{cm}^2 \cdot \text{sec}}$$

where i indicates a particular species, μ is the average range and P_i is the volumetric source of type i fission fragments. P_i is equivalent to the fission yield ($y_i F$) of species i .

The total release rate is the particle flux times the thorium surface area, $I_i A_s$. The total volumetric generation rate for fission fragment species i is $y_i F V$, where V is the volume of thorium. Thus, the fraction (f) of fission products that are released directly to the primary coolant due to recoil is

$$f = \frac{y_i F \mu A_s / 4}{y_i F V} = \frac{\mu A_s}{4V}$$

where the surface area to volume ratio is noticeably important. For the thorium metal clip;

$$f = \frac{(10^{-3}) (20.4)}{4 (0.71)} = 0.007$$

or, 0.7% of the fission products are directly released to the lithium coolant due to recoil. By comparison, the knockout fraction is estimated to be 4×10^{-10} .

Activity Content in Coolant Due to Recoil and Knockout. Both the actinides (e.g., ^{233}Th and ^{233}Pa) and fission products can contribute to the coolant activity, but the fission product inventory will dominate due to recoil mechanisms. Therefore, a good upper bound estimation of the activity content in the coolant is 0.7% of the fission product activity in the thorium clip. The individual fission product species were considered in the 1981 Tandem-Mirror Hybrid Feasibility Study (2) of a gas-cooled blanket which closely resembles the blanket design considered here. Table 6 lists the reported fission product activities. The activity source term in the lithium coolant associated with fission product release from the unclad thorium is estimated to be

$$2.3 \times 0.007 \times 1.7 \times 10^8 \text{ Ci} = 2.7 \times 10^6 \text{ Ci}$$

where the 2.3 factor is the ratio of end-of-cycle fission powers in the 1982 and 1981 designs.

Table 6. Major Contributions to Fission Product BHP
for the 1981 Helium Cooled Blanket Design

Radionuclide		Activity (Ci)		BHP (Km ³)		Half-Life
Sr	89	5.7	+06	1.9	+07	54 d
	90	4.4	+04	1.5	+06	28 y
	91	2.3	+07	2.5	+06	9.7 h
Y	91	6.8	+06	6.8	+06	58 d
Zr	95	6.6	+06	6.6	+06	65 d
Mo	99	2.4	+07	3.4	+06	67 h
I	131	1.1	+07	1.1	+08	8.1 d
	133	2.6	+07	6.4	+07	2.1 h
	135	2.2	+07	2.2	+07	6.7 h
Cs	134	1.4	+06	3.5	+06	2.3 y
Ba	140	2.0	+07	2.0	+07	13 d
La	140	2.0	+07	5.1	+06	40 h
Ce	144	1.5	+06	7.7	+06	285 d
TOTAL		1.7	+08	2.7	+08	
TOTAL/module		7.0	+06	1.1	+07	
TOTAL/kg Annual Fissile Fuel Production		2.5	+04	4.1	+04	

C.2.b Design Allowable Coolant Activities

Given that there will be a potentially large source of fission product activity in the primary coolant, is this acceptable for licensing considerations? As discussed earlier, this question is very difficult and complicated to answer. The NRC regulations are directed toward the maximum permissible exposure of workers and the general public to radioactivity. They do not provide quantitative guidelines for reactor designers. A more logical example to follow, the CRBRP safety report, does provide some perspective. The CRBRP thresholds of concern for fission products to be included in the evaluation are:

1. A cumulative inventory in the coolant of greater than 0.1 percent (based on the fast fission data for Pu-239).
2. An isotopic release resulting in approximately 10 or more curies of activity at ten days after shutdown for one of the following sources:
 - a. the cold trap,
 - b. the reactor plant primary coolant system surfaces,
 - c. the primary coolant.
3. The release of significant quantities of noble gas fission products with half lives in excess of ten minutes.

The CRBRP calculation estimates 10^5 Ci of fission products in the primary sodium coolant. The Na^{24} activity is 2×10^7 Ci with a 15-hour half life.

The fusion breeder blanket will have a higher fission product activity in the primary coolant than that calculated for the CRBRP (2.7×10^6 Ci vs. 10^5 Ci of fission products), but a lower overall coolant activity. Considering only the recoil and source term, the fusion breeder meets the CRBRP design goal of .01 Ci/gm of primary coolant, or 6.4×10^6 Ci.

in decay heat removal capability. The guard vessels surrounding the pipe, pump, and heat exchanger components prevent leakage of the lithium coolant. For a major component loss (e.g., all primary coolant pumps or heat exchangers), the thermal analysis has shown that the shield cooling system adequately removes the afterheat, even with the fuel within the modules. Thus, loss of flow events do not require fuel dump with an operational shield cooling system. Should the latter system fail, a fuel dump is necessary. Cooling is then accomplished with the module safety coolant (fuel handling) system shown in Fig. 2. Further backup is provided by the dump tank passive cooling system (see Section B.2).

In the event of an inlet/outlet piping rupture, the main concern is limiting the extent of the leakage. The main coolant inlet/outlet pipes which service the inlet/outlet manifolds can be above the level of the blanket module to prevent a complete loss of coolant via a primary loop failure external to the blanket. Also, the valving configuration provides multiple opportunities to isolate a leak. Nonetheless, a pipe failure is assumed to result in a direct radioactivity leakage into the containment. Considering just the fission product inventory expected in the primary coolant, as discussed in Section C, and the resultant dose consequences to the public, as presented in Section D.2, double-walled piping may be attractive where feasible. Total coolant radioactivity release events initiated by piping failures require the additional failures of two of three cooling systems, thus making these very low probability events.

In the event of a structural failure within the module boundary (e.g., the first wall), the module content are dumped to permit the shield and/or auxiliary first wall cooling system to remove decay afterheat in the blanket structure. Should the dump valve fail, both the shield and auxiliary first wall cooling systems would be required for adequate heat removal. Thus, the minimum consequence of a loss of module integrity is a primary coolant radioactivity leak. The maximum consequence is the release of the inventory of one module to the secondary containment.

C.2.c. Potential Concerns and Control Mechanisms

The above analysis neglects any reduction due to cold trapping or plating. This reduction is important and should be accounted for. Cold trapping is effective to ~ 1 ppm levels and can be enhanced by isotopic dilution. Fission products will also tend to plate out in cooler parts of the loop. This will reduce the coolant inventory, but can complicate maintenance. Plate out rates will be higher if cold trapping is not used.

Additional concerns facing the use of unclad thorium are corrosion and erosion. If the thorium thickness is reduced due to chemical corrosion or erosion from rubbing, additional fission products and actinides will be released to the coolant. These source terms could easily dominate those from recoil and knockout. Corrosion and erosion can be controlled with a suitable coating. Recoil and knockout mechanisms occur only within 10^{-3} cm of the thorium surface and can also be controlled by a surface coating ~ 30 μm thick with $\sim 95\%$ integrity.

Back diffusion may also lower the activity source term calculated here. Recoil fission products are desposited in a very thin boundary layer where the chemical activity level of individual species could far exceed that of the thorium and/or structure.

Finally, in an accident sequence, the thorium clip may experience very high temperatures. The mobility of certain fission products (noteably noble gases and cesium) will reach values consistent with almost total release to the coolant.

In summary, recoil and knockout in the unclad fuel can potentially result in a fission product inventory in the breeder primary coolant exceeding design specifications, but not limits, established for CRBRP. The beneficial effects of cold trapping within the loop should be assessed. Experimental data is required to evaluate the impact of corrosion and erosion of the fuel on primary coolant activity. A potential design solution to limit both recoil/knockout and corrosion/erosion activation sources is to develop a thin coating. A better assessment of recoil redeposition on the thorium snap-ring and in the loop is needed.

D PROBABILISTIC RISK ASSESSMENT

A qualitative Failure Mode and Effects Analysis (FMEA) of the reference fusion breeder primary cooling system was performed in Reference 1. In the present study, that analysis was extended to a limited probabilistic risk assessment (PRA) at the system/component level. The PRA was performed to assess compliance with regulatory single-failure criteria and to evaluate the probabilities and consequences, and thus the relative risks, of postulated sequences of events. The goal of the effort was neither completeness or the absolute risk indicator. Rather, the goal of this effort was to identify the accidents with the greatest relative risk which should receive major attention in future safety efforts. The event trees considered are presented in Section D.1, followed by the consequence analysis in Section D.2. The conclusions and recommendations of the PRA are presented in Section D.3.

D.1 Event Tree Construction

In this section, event trees are developed for failures of the primary cooling system shown in Figure 1 and described in more detail in Reference 1. The primary coolant system components and interfaces, including the new systems described in Section B above, are listed in Table 7. The procedure followed in constructing the event trees was to individually consider the failure of each component and then determine subsequent corrective action until the accident sequence is adequately resolved or terminates in a radioactive release. Branch probabilities were assigned based on Reference 2 and 3. For event sequences resulting in a radioactive release, consequence analyses were performed and are presented in Section D.2.

During the process of constructing the event trees, it was determined that initiating events that result in failures of single components, e.g., pumps or primary heat exchangers, or single wall failures in the piping to or from the pumps and heat exchangers do not result in lithium spills or radioactive releases. Redundancy in the cooling circuitry allows for adequate

TABLE 7 Primary Coolant System Components and Interfaces

System components

Cold manifold
Inlet piping
Inlet distribution plena
Modules
Module outlet piping
Module outlet plena
Outlet piping
Hot manifold
Primary coolant pumps
Common manifold
Primary Li-to-Na heat exchangers
Piping to and from pumps and heat exchangers (double-walled)
Guard vessels
Valves

System interfaces

Fuel changeout and module safety/fuel transfer coolant systems
Shield cooling systems
Auxiliary first wall cooling system
Passive dump tank cooling systems
Electric power systems
Secondary heat removal systems
Lithium fire control and suppression systems
Control room via detection and instrumentation systems
Maintenance

cooling. Furthermore, the valving arrangement, double-walled piping and guard vessels, require a minimum of two additional failures in order to result in failure of the safety systems to protect the public. One or more of these can be of arbitrary origin but, ultimately, there must be a failure of the containment building. In addition to the extremely low probabilities, the radiological consequences of such a sequence of failures would be extremely small, as discussed in the next section. Failure of the common manifold would result in a limited spill and radioactive release. However, assuming a single additional failure, the redundant valving arrangement would again limit the magnitude of the spill for the postulated event. Its radiological consequences would be very small, even with failure of the secondary containment. Thus, the single failure criterion is satisfied for the above class of initiating events.

The event trees associated with the above scenarios are not very interesting and are not presented. The event trees with significant consequences and/or interesting characteristics are those that lead to the loss of all primary coolant pumps or heat exchangers, the inlet/outlet piping or cold/hot manifolds, the module inlet/outlet distribution plenum, or the first wall. These are presented in Figs. 12 through 15 and are discussed below. Note that none of the event trees include the passive dumptank cooling system and, therefore, the results are in many cases, conservative. They are, nevertheless, encouraging.

Loss of All Primary Coolant Pumps or Heat Exchangers (Figure 12). These events could be initiated via loss of power to all the pumps or loss of the secondary coolant systems. In both cases, the resultant sequence of events is the same. Assuming instantaneous detection and plasma shutdown, the blanket decay heat load can be picked up by the shield cooling system, as shown in the thermal analysis of Section B above, resulting in no damage or radioactive release. If the shield cooling system subsequently fails, the first wall auxiliary coolant system can be activated but after several hours the fuel dump valves must be opened and the contents of the modules must be transported to the dump tank. The safety/fuel management system coolant inlet valves are

LOSS OF ALL PRIMARY COOLANT PUMPS OR IHXS	SHIELD COOLING SYSTEM FUNCTIONS	DUMP VALVES OPEN	SAFETY COOLANT INLET VALVES OPEN	SAFETY COOLANT SYSTEM FUNCTIONS	SECONDARY CONTAINMENT FUNCTIONS	BRANCH ID	PROBABILITY (Per Year)	CONSEQUENCE (Rem)
0.1/y	~1					A1	0.1	0
	10 ⁻³	~1	~1	~1		A2	10 ⁻⁴	0
				10 ⁻³	~1	A3	10 ⁻⁷	9×10 ⁻⁴
					10 ⁻⁴	A4	e(10 ⁻¹¹)	40
			10 ⁻³		~1	A5	10 ⁻⁷	9×10 ⁻⁴
					10 ⁻⁴	A6	e(10 ⁻¹¹)	40
		10 ⁻³			~1	A7	10 ⁻⁷	0.3
					10 ⁻⁴	A8	e(10 ⁻¹¹)	5000

NOTE: BACKUP DIESEL GENERATORS WITH A FAILURE PROBABILITY OF 3×10^{-2} WOULD MAKE ALL ABOVE EVENTS INCONCEIVABLE ($< 10^{-9}/\text{yr}$)

Fig. 12 Event tree for loss of all primary coolant pumps or heat exchangers.

Fig. 13. Event tree for failure of the inlet/outlet piping or cold or hot manifold

Fig. 14. Event tree for module inlet/outlet distribution plenum failure

Fig. 15. Event tree for first wall failure

then opened and the safety coolant system provides adequate module cooling through the fuel handling system piping. If the safety coolant system fails, the passive dump tank cooling system (i.e., the heat pipes) have an independent cooling capability. If the passive system also fails, the modules fail and the activation product inventory of the reactor is assumed to be released to the secondary containment. Failure of the safety coolant inlet valves followed by the failure of the passive system (not shown in Figure 12) results in a similar occurrence but is called out separately. In a worst case scenario in which the fuel dump valves fail to open, the entire radioactive inventory of the reactor is released to the secondary containment. Radioactive release consequences can then occur with or without secondary containment failure, as calculated in Section D.2 below.

Inlet/Outlet Piping or Cold or Hot Manifold Failure (Figure 13). This accident is assumed to be initiated by a structural failure. Following the detection of the failure, the affected section is isolated by closure of valve adjacent to the leakage. In the worst possible scenario, in which the break has resulted in a complete loss of primary coolant flow to the entire reactor, the event tree follows the same sequence of events as in the previous tree, starting with the shield cooling system function. Without loss of cooling, the consequences of this event tree are a small coolant leak (10% of the inventory assumed), and the release of its associated inventory. Note that the redundant valving arrangement and piping elevation makes it inconceivable that leakage allow the entire reactor lithium inventory to be drained.

Module Inlet/Outlet Distribution Plenum Failure (Figure 14). This event is initiated by a structural failure at the module level. Since the failure could occur at any elevation within the distribution plenum, it is desirable to isolate the module, dump the fuel, and allow the shield system to remove the structural heat load. The safety/fuel transfer coolant system provides backup cooling as described previously with additional backup provided by the auxiliary first-wall cooling system (not shown in top half of Figure 14).

If the fuel dump valve fails to open, then both the shield cooling system and the auxiliary first wall cooling system must function to provide adequate module cooling, since the safety/fuel transfer coolant system is inoperative. The consequences of this event are reduced over the previous tree, since only one module is affected.

First Wall Failure (Figure 15). The frequency of "plasma dumps" in tandem mirror reactors is, as yet, unknown, but it has been estimated that such dumps would introduce an ~ 200 psi pressure transient in the liquid metal coolant. As the blanket would be designed not to fail during such a transient, the design basis first wall failure assumed in the present analysis is "random" structural failure of the first wall (assumed probability $10^{-2}/\text{yr}$). The failure quenches the plasma and initiates fuel dumping and isolation of the failed module. A failed first wall can provide a short circuit path for through-the-module cooling via the safety coolant system. However, cooling of the blanket structure can be accomplished via successful operation of either the shield or auxiliary first wall cooling systems. The dumped fuel would be cooled via a module bypass in the safety/fuel transfer coolant circuit or via the passive dump tank coolant. If the fuel dump valve does not open, then both the shield and auxiliary first wall cooling systems are required. The maximum release in the event of failure of either system is the total radioactive inventory of one module.

Branch Probabilities (Table 8). The probabilities used in the present study were obtained from References 17, 18, and 19 and are similar to those used in References 2 and 3. Valve failure probability was increased to 10^{-3} per demand to account for all failures to change state. First wall structural failure was increased to $10^{-2}/\text{yr}$ to account for its complexity. Safety coolant, shield, and auxiliary first wall cooling system failure probabilities were based on those for partly diverse systems (Reference 19).

D.2. Consequence Analysis

Having developed the event trees and branch probabilities of primary cooling system failures in the preceding section, the dose consequences of the postulated radioactive releases are presented in this section. The

TABLE 8 Failure Probabilities

Event Description	Failure Probability ^a
Loss of all primary coolant pumps or heat exchangers	0.1/y
Inlet/outlet piping or cold or hot manifold failure	10 ⁻⁴ /y
Module inlet/outlet distribution failures	10 ⁻⁴ /y
First wall failure	10 ⁻² /y
Valve failure to change state	10 ⁻³
Shield cooling system failure	10 ⁻³
Safety/fuel transfer coolant system failure	10 ⁻³
Auxiliary first wall cooling system failure	10 ⁻³
Secondary containment failure	10 ⁻⁴
Plasma shutdown failure	10 ⁻⁵
Detection system failure	10 ⁻⁴
Diesel generator failure to start on demand	3×10 ⁻²

^aFailure probabilities are on a per demand basis except where indicated.

radioactive inventory is presented first, followed by the consequence analysis.

Radioactive Inventory. The major sources of radioactivity in the fusion breeder are actinides, fission products, activation products, and tritium. The activities and biological hazard potential in air (BHP) contributed from each of these sources, listed by radionuclide, are presented in Tables 9 through 11. Table 12 presents a summary by source.

The inventories of the actinides ^{232}Th , ^{233}Pa , and ^{233}U were explicitly calculated for the reference tandem mirror blanket in Ref. 1. The inventories of the remaining actinides were then obtained by scaling of the inventories of the similar Ref. 2 gas-cooled blanket. The ^{233}Th and ^{231}Pa inventories were scaled proportionally to the ^{232}Th inventory, and the ^{232}U and ^{228}Th inventories were scaled linearly with the ^{233}U inventory. As before, the actinide radioactive hazard is initially dominated by ^{233}Pa . Following decay of the ^{233}Pa (in the time scale of months) the actinide hazard is reduced by approximately two orders of magnitude and is dominated by ^{231}Pa , ^{232}U and its daughters.

The fission product inventory for the present blanket was obtained by linearly scaling the Ref. 2 inventories with the fission power. The present blanket produces 1728 MW of fission power versus 461 MW with the previous blanket, thus the fission product inventory is approximately four times higher. Since the activation products are mainly located at the first wall, this radionuclide source term was scaled linearly with wall loading (i.e., 1.27 MW/m^2 versus the previous 2 MW/m^2). The analysis also accounts for the present ferritic steel versus the previous austenitic 316 stainless steel first wall. The releasable tritium inventory was assumed to be 10^7 Ci.

Although beryllium is a toxic substance, its overall toxicity is many orders of magnitude lower than that of the radioactive products.

Consequence Analysis. The calculation of the radiological consequences of the release of the inventories given above is the last step in computing

TABLE 9 Major Contributors to Actinide BHP for the Reference Tandem Mirror Breeder Blanket Reactor

Radionuclide	Activity (Ci)		BHP (km ³)		Half-Life
²²⁸ Th	2.6	+02	1.3	+06	1.91 y
²³² Th	3.2	+01	3.2	+04	1.4×10 ¹⁰ y
²³³ Th	3.8	+10	9.9	+08	22 m
²³¹ Pa	2.4	+03	6.1	+07	3.3×10 ⁴ y
²³³ Pa	2.1	+10	3.4	+09	27 d
²³² U	2.1	+4	2.3	+07	72 y
²³³ U	2.2	+4	5.5	+04	1.6×10 ⁵ y
Total	5.9	+10	4.5	+09	--

TABLE 10 Major Contributors to Fission Product BHP
for the Reference Tandem Mirror Breeder
Reactor Blanket

Radionuclide	Activity (Ci)		BHP (km ³)		Half-Life
⁸⁸ Kr	4.8	+07	2.4	+06	2.8 h
⁸⁹ Sr	2.1	+07	7.1	+07	54 d
⁹⁰ Sr	1.6	+05	5.6	+06	28 y
⁹¹ Sr	8.6	+07	9.4	+06	9.7 h
⁹¹ Y	2.5	+07	2.6	+07	58 d
⁹⁵ Zr	2.5	+07	2.5	+07	65 d
⁹⁹ Mb	9.0	+07	1.3	+07	67 h
¹³¹ I	4.1	+07	4.1	+08	8.1 d
¹³³ I	9.7	+07	2.4	+08	2.1 h
¹³⁵ I	8.2	+07	8.6	+07	6.7 h
¹³³ Xe	9.5	+07	3.2	+05	5.3 d
¹³⁵ Xe	9.5	+07	9.5	+05	9.1 h
¹³⁴ Cs	5.2	+06	1.3	+07	2.3 y
¹³⁷ Cs	1.7	+05	3.4	+05	30 y
¹⁴⁰ Ba	7.5	+07	7.5	+07	13 d
¹⁴⁰ La	7.5	+07	1.9	7	40 h
¹⁴⁴ Ce	<u>5.6</u>	<u>+06</u>	<u>2.9</u>	<u>+07</u>	285 d
Total	6.4	+08	1.0	+09	--

TABLE 11 Major Contributors to Structural Activation
Product BHP for the Reference Tandem Mirror
Breeder Reactor

Radionuclide	Activity (Ci)		BHP (km ³)		Half-Life
⁵⁴ Mn	2.5	+07	2.5	+06	300 d
⁵⁶ Mn	1.3	+08	7.0	+07	2.6 h
⁵⁵ Fe	<u>2.9</u>	<u>+08</u>	<u>9.5</u>	<u>+07</u>	2.9 y
Total	4.5	+08	4.2	+07	--

TABLE 12 Total Radioactive Inventory and
BHP for the Reference Tandem
Mirror Breeder Reactor Blanket

Radioactivity Source	Activity (Ci)		BHP (km ³)	
Actinides	5.9	+10	4.5	+09
Fission products	6.4	+08	1.0	+09
Activation products	4.5	+08	4.2	+07
Tritium	<u>1.0</u>	<u>+07</u>	<u>5.0</u>	<u>+04</u>
Total	6.0	+10	5.5	+09

risk. Following a blanket heatup event, the radioactivity present is released to the secondary containment. The radionuclide behavior in such situations is a major uncertainty. Potential chemical reactions, product volatilities, plateout and other deposition and escape mechanisms present a major problem in the estimate of consequences. The failure of an engineered safety feature does not necessarily result in an escape. For the purposes of this study, the release fractions were chosen to be consistent with those employed in LWR siting events. These include a 100% release of the noble gases and tritium, 25% release of the halogens, and 1% of all remaining material. The results obtained from the consequence analysis, though potentially overestimating the consequences, are useful in identifying the major consequence contributors.

The computer code TDAC (20) was used to calculate consequence. The radiation path after release from the containment involves meteorological assumptions and intake pathways such as inhalation and ingestion, ground contamination, and surface deposition. This analysis assumed direct airborne plume submersion and prompt inhalation and estimated external whole body exposure and internal inhalation-dose commitment to the organs, thyroid, bone, and lung. A detailed summary of the reference conditions, parameters, and assumptions used in the analysis is found in Table 13. An effective dose equivalent accounting for the total risk attributable to the exposures to all the organs irradiated was used in presenting the results. The weight assigned to the exposure of each organ was derived from the ICRP-26 recommended values and is shown in Table 14. These risk factors are based upon the likelihood of inducing fatal malignant disease, acute effects, and genetic defects.

Tables 15 and 16 present the results of the dose consequence calculation. In both functional and failed containment and cleanup system scenarios, over 80% of the effective dose is attributed to the fission products, mainly in external whole body gamma exposure and inhalation thyroid and bone doses. Approximately 16 percent of the effective dose is contributed by the actinides, essentially from the ^{233}Pa bone and lung dose. Less than one percent of the effective dose is from the activation products, and the tritium contribution is negligible.

TABLE 13 Tandem Mirror Fusion Breeder Reactor Consequence Analysis
List of Parameter Selections and Assumptions

Parameter	Assumption
INITIAL RADIOACTIVITY INVENTORY PRESENT IN THE REACTOR AT $t = 0$	See Tables 9 through 11
RADIONUCLIDE RELEASE FROM BLANKET MATERIAL	Fraction of initial inventory released to secondary containment atmosphere:
Release Fraction	
Material:	
Tritium	100%
Noble gases	100%
Halogens	25%
Others (includes bred fuel)	1%
Release mode to primary or secondary containment	Instantaneous release of inventory fraction to the primary (where applicable) or secondary containment atmosphere at time = 0
PRIMARY AND SECONDARY REACTOR CONTAINMENT	
Intact containment scenarios	Leak rate of 0.1% vol/day (secondary). Containment integrity maintained for the duration of the accident.
Nonisolated or failed containment scenarios.	Leak rate of 40% vol/hour for the duration of the accident.
Engineered containment atmosphere cleanup system characteristics	Recirculation cleanup system assumed operational at initiation of the event for all scenarios involving intact containments, system is assumed to fail for all nonisolated scenarios. Flow rate of 100% vol/hr assumed. Filter efficiencies of 0% for noble gases and 99% for all other materials were employed. Tritium cleanup flow rate - 30%/hr.
Natural deposition mechanisms (i.e., fallout and plateout)	Not modeled.

TABLE 13 (Continued)

Parameter	Assumption
REFERENCE SITE METEOROLOGY CONDITIONS ASSUMED	
Atmospheric dispersion factor, at 2500 meters X/Q (sec/m ³) ^a	8.0×10^{-6} .
DOSE COMPUTATION	
Breathing rate assumed for duration of accident	2.32×10^{-4} m ³ /s.
Inhalation dose commitment effectiveness	Reference 21 adult inhalation values used. (Note: Ref. 21 has increased the tritium inhalation dose commitment factors by 50% to account for transpiration pathway. Also, activation product effectiveness developed using Appendix B of Ref. 20 information.)
Wind speed assumed	1.0 m/s.
Decay and buildup enroute to dose receptor	Considered.
Exposure pathways examined	Prompt inhalation and submersion in the passing plume only.
Dose computation computer program	TDAC/E1 (Ref. 20).

^aBasis of atmospheric dispersion estimate, median atmosphere dispersion estimate employed in HTGR analysis, consistent with U.S. average site.

TABLE 14 Dose-Equivalent Risk Weighting Factors
Used in Calculating Effective Dose

body	0.76
id	0.04
	0.04
	0.16

TABLE 15 Dose Consequence from Total Blanket Involvement Events:
Containment Intact, Cleanup Functional

	Integrated 30-Day Dose (Rem) ^a at 2500 m				
	External Whole Body Gamma	Inhalation			Effective Dose Equivalent ^a (Rem)
		Thyroid	Bone	Lung	
Accident type					
Containment intact, cleanup functional	0.14	2.1	0.06	0.65	0.3
Dose contributor, %					
Fission products, total	99.9	100.0	46.1	8.8	83.2
88Kr	19.9				
89Sr			11.6		
91Y			11.1		
131I		57.7			
133I		34.3			
133Xe	47.7				
135Xe	21.7				
Actinides, total		0.0	52.7	90.2	16.5
231Pa			16.1		
233Pa			33.5	89.4	
Activation products, total	0.1	0.0	1.2	1.0	0.3
Tritium	0.0	0.02	0.7	0.06	0.04
Regulatory limits (10CFR100) ^b	25	300	150 ^c	150 ^c	60

^aDose-equivalent risk weighting factors for effective dose: whole body - 0.76,
thyroid - 0.04, bone - 0.04, lung - 0.16.

^bReference 23.

^cInferred from 10CFR100 intent.

TABLE 16 Dose Consequence from Total Blanket Involvement Events:
Containment and Cleanup Failed

Accident type	Integrated 30-Day Dose (Rem) at 2500 m				
	External Whole Body Gamma	Inhalation			Effective Dose Equivalent ^a (Rem)
		Thyroid	Bone	Lung	
Containment and cleanup failed	280	4.8×10^4	1400	1.5×10^4	4600
Dose contributor, %					
Fission products, total	99.1	100.0	46.1	8.9	82.6
88Kr	35.7				
89Sr	12.6		11.5		
91Y			11.1		
131I		59.0			
133I		33.7			
133Xe	17.0				
135Xe	14.5				
Actinides, total	0.0	0.0	52.5	90.1	16.5
231Pa			16.1		
233Pa			33.4	89.5	
Activation products, total	0.9	0.0	1.2	1.0	0.9
Tritium	0.0	0.0	0.2	0.0	0.01
Regulatory limits (10CFR100) ^b	25	300	150 ^c	150 ^c	60

^aDose-equivalent risk weighting factors for effective dose: whole body - 0.76, thyroid - 0.04, bone - 0.04, lung - 0.16.

^bReference 23.

^cInferred from 10CFR100 intent.

The major organ dose for total blanket involvement accidents is to the thyroid from ^{131}I and ^{133}I . Whole body gamma, thyroid, and lung dose contribute about equally to the effective (health effect producing) dose equivalent for accidents with intact containment and cleanup operating, while thyroid and lung dose contribute about equally to the effective dose equivalent for accidents with both containment and cleanup failed. Fission products contribute nearly all the whole body gamma and thyroid dose, while actinides are the major lung and bone dose contributors. Accidents with intact containment and cleanup operational meet the 10CFR.10 regulatory limits (23), while those that fail these systems do not. Therefore, the probability of accident sequences that include failure of both systems must be kept below credible levels by design.

D.3 Conclusions and Design Implications

The relative risks of the accident sequences investigated are plotted in Fig. 16. Three event branches with significantly greater risk can be identified. The two highest are Event A7, loss of all primary coolant pumps or heat exchangers followed by shield coolant system failure and dump tank failure, and Event D1, primary coolant leakage via first wall failure. These are closely followed by Event B1, primary coolant leakage via failure in the inlet/outlet piping or cold or hot manifold.

Event A7 is a low probability-high consequence event versus the other two which have low consequence but a higher probability, as noted in Fig. 12, the inclusion of diesel generators in the fusion breeder plant design would result in a failure probability considered inconceivable in safety analysis. It is important to note that in the absence of fully developed systems on which to calculate failure probabilities, the data used in this comparative analysis could, instead, serve as design goals.

The design of the first wall represents the result of the resolution of a number of technical constraints - structural, thermal-hydraulic, neutronic,

Fig. 16 Relative risk of the accident sequences of Figures 12 to 15

etc. The magnitude of the risk of a first wall failure indicates that reliability should also be a major consideration. In addition, once a spill via first wall failure has occurred internal to the vacuum chamber, the design of the chamber itself should be such as to serve as an additional barrier to the release of the coolant to the secondary containment. Note that damage to the vacuum chamber was not included in the present assessment.

The high relative magnitude of the risk of Event B1, essentially only a coolant leak to containment, indicates the need to consider methods of controlling this release pathway. Alternatives are to employ double-walled piping and cladding the fuel or otherwise limiting the fission product inventory in the coolant. It is noted that although primary piping failure is uncommon in present nuclear reactors, piping failures could be more extensive in fusion breeder reactors. A better understanding of the fission product inventory in the coolant is needed to better understand this event.

In the relative sense, the fertile-dilute fission-suppressed fusion breeder design offers two advantages over fast-fission blankets. First, the lower radioactive inventory results in lower consequences in maximum blanket involvement events. Second, the lower afterheat rate contributes to the favorable thermal response during loss of primary cooling events. A continuous spectrum of blanket radioactive inventories and afterheat rates span the two concepts. In both cases, a mobile fuel can provide for multiple inherent approaches for redundancy and diversity in cooling. This can, in turn, result in low accident probabilities for high consequence events.

Finally, it is noted that lithium fires and beryllium volatility and toxicity were not considered rigorously in this analysis. These would warrant analysis in future safety work. Similarly, the effects of common mode failures should be included as the design of the fusion breeder reactor attains maturity.

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