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SAFETY OF FUSION REACTORS

Final Report Prepared for
US Energy Research and Development Administration

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PPG-342

October 1977

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Prepared for the
US Energy Research and Development Administration
under ERDA Contract No. E(11-1)-GEN-10, PA26

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ABSTRACT AND SUMMARY

Following a number of first-generation fusion power plant design concepts and their subsequent analysis, a second generation of fusion reactor designs have emerged. These include three conceptual experimental power reactor (EPR) design studies and UWMAK-III, a non-circular Tokamak power reactor design. In this report, the results of an initial examination of potential safety questions for these second-generation reactor designs is presented. It is envisioned that the results of such preliminary studies will aid designers, so that as EPR's move from the conceptual design stage towards actual design, systems to eliminate or mitigate potential hazards can be included.

The initial steps in the examination of plant safety are to identify:

- a) Potential hazards and their magnitude,
- b) Significant accident sequences,
- c) Potential operational safety problems, and
- d) Major gaps in safety-related information.

In this study, these questions are handled within the context of questions relating to tritium, the first wall, the primary and secondary heat removal systems, and the magnet systems.

As most fusion power reactors operate on the deuterium-tritium fuel cycle, the principal hazards of the EPR and UWMAK-III conceptual designs are tritium and induced activity in the first wall and blanket structure. If the EPR designs do indeed have a small tritium inventory (due to the absence of breeding), then first wall activity will be the dominant hazard.

The principal accident paths are difficult to identify because of a lack of detailed design and safety related data. Preliminary examination of these designs do indicate, however, several potentially significant accident sequences which can lead to a release of radio-activity.

Comparison of the EPR and UWMAC-III designs with the goal of listing potential accident initiators and potential plant malfunctions is hampered due to deficiency of information. While this may be attributed to emphasis on other aspect of design by the design groups, it does indicate areas where future work must focus.

The tritium-handling systems and detritiation facilities are inadequately described in the three EPR designs to allow a complete, comparative analysis. In more general terms, the use of detritiation systems to allow rapid access to, and give acceptably low off-normal and steady-state tritium releases from, the reactor components is seen to have potentially adverse safety consequences. Several questions which cannot be answered at this time, will markedly affect the detritiation systems' safety implications. For example, several system failures are outlined which lead to tritium release from the detritiation facility but which arise from otherwise benign events. The safety implications of this cannot be assessed until more is known about the normal operating procedures of a detritiation facility.

The three EPR concepts present significantly different designs for blanket and first wall cooling functions. Clear differences can be seen in terms of materials, coolants, characteristic temperatures, and the existence of more than a single means of heat removal. An analysis

of the safety of these designs during a loss-of-coolant accident (LOCA) has been undertaken to assess whether substantial benefits can be expected from adopting one design philosophy, with particular emphasis on the question of whether multiple first-wall and blanket modules and multiple coolant systems can be expected to afford tangible safety advantages. Analysis of an approximately spherical geometry demonstrates that there is in fact a considerable benefit to both of these options. For example, the peak temperatures during a LOCA in a single module could be stabilized at a temperature sufficiently low to allow times of several minutes or more to detect the failure. Preliminary analysis of the more complex toroidal geometry, while incomplete, demonstrates a similar trend.

Potential safety problems associated with the various primary coolant candidates currently considered for the EPR fusion blanket designs were examined. The basic concern is the possibility of overheating and melting of the first wall and the blanket, induced by a malfunction in the primary coolant system. These accidents include the loss of coolant flow, the loss of heat removal, overpower transients, and the loss of coolant. Following a mechanistic safety analysis for these four types of accident sequences and comparing helium and liquid metal cooling, it was found that helium has a more adverse effect on the first-wall heat-up in the event of a loss of heat-removal or a loss of coolant because of its lack of thermal inertia.

The present safety assessment of EPR system magnets focuses attention upon the superconducting magnets, in particular, stabilizer choices and effects related to the thermodynamic state choice

and mode of operation of the magnet coolant. Major conclusions are that either extrapolations of unknown coolant conditions in conjunction with new superconductor configurations and materials are elements of concern, or that insufficient knowledge is available about large scale extrapolation of principally known pool-cooling cases of small and medium-sized magnets. In all cases, questions arise concerning the optimum use of coolant conditions for stabilizer design, the effect of stratification (for pools), and standby refrigeration requirements for loss of magnet-coolant events. Key factors involved in performing this safety assessment are data for superconductor-coolant interaction obtained in a wide range of the thermodynamic state conditions (within the superconducting phase envelope). In any case, it is recommended to operate the interior magnet coolant at 1 bar or above (though not necessarily above 4 K).

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1. GENERAL SAFETY CONSIDERATIONS

1.1. Introduction

During the past several years, a number of first-generation fusion power plants were designed, and a limited number of safety analyses were performed [1-1,1-2]. Utilizing the experience gained in these endeavors, a second generation of fusion reactor designs has emerged. These include the three conceptual experimental power reactor (EPR) design studies [1-3 to 1-5] and UWMak-III [1-6], a non-circular Tokamak power reactor design. The objective of this report is to summarize the results of an initial examination of potential safety questions for these second-generation reactor designs. It is envisioned that the results of such studies will aid designers, so that as EPR's move from the conceptual design stage towards actual design, systems to eliminate or mitigate potential hazards can be included.

The initial steps in the examination of safety are to identify:

- (a) Potential hazards and their magnitude,
- (b) Significant accident sequences,
- (c) Potential operational safety problems, and
- (d) Major gaps in safety-related information.

In this study, these questions are handled within the context of questions relating to tritium, the first wall, the primary and secondary heat removal systems, and the magnet systems.

In this chapter, general safety considerations are discussed. In the remainder of the report, Chapter 2 deals with tritium, Chapter 3 with

first wall integrity, Chapter 4 with heat removal systems, and Chapter 5 with magnet systems. Each chapter contains its own conclusions.

1.2. Potential Hazards, Significant Accident Sequences, and Potential Operational Safety Problems

In a conceptual fusion power reactor such as UWMAK-III, the principal hazards are tritium, induced activity in the first wall, radioactive corrosion products, and the large quantities of non-radioactive but chemically toxic materials that are used [1-7].

With respect to tritium, the only EPR in which tritium will be bred is the Oak Ridge National Laboratory (ORNL) design [1-5], and there only in one blanket module. It would appear that the major sources of tritium in EPR's are the fueling and injection systems, the divertor and removal systems (as employed), and storage. For UWMAK-III, with a predicted tritium breeding ratio of 1.25, the total tritium inventory is estimated to be 35.8 kg.

The Argonne National Laboratory (ANL) and General Atomic (GA) design teams predict inventories of 1.5 and 4.1 kg, respectively. The ORNL report does not show a total inventory. The following systems are listed for tritium reprocessing:

- Purification of the D-T mixtures for recycling to feed systems
- Tritium storage systems
- Separation of hydrogen-isotopic impurities from D-T mixtures
- Plasma fueling and injector feed systems
- Preparation of wastes containing tritium for disposal

The anticipated through-put of tritium is 2 kg/day. It would appear that

the total tritium inventory for the ORNL design may be as high as 10 kg.

If the tritium inventories are indeed kept low, induced activity in the first walls and structural materials may become the principal hazard. The induced activity in the first wall of the ANL design is 3.5×10^6 Ci/MW(th) or 6.5×10^8 Ci, based on a neutron wall loading of 0.5 MW/m^2 . The ORNL and GA designs call for neutron wall loadings of 0.55 and 0.85 MW/m^2 , respectively. Hence, their induced activities will be of the same order of magnitude.

The UWMAK-III wall loading of $1.0\text{-}3.5 \text{ MW/m}^2$ gives an induced activity of 0.8 Ci/W, or 2.0×10^9 Ci, almost an order of magnitude greater than the EPR designs. This induced activity, at shutdown, is similar to those of UWMAK-I and II, but has different long-term behavior as a result of its different material composition.

In addition to the possibility of release of radioactivity via corrosion, sputtering, and loss of cooling, the occupational exposure may become quite large in the event that remote maintenance, repair, and replacement are impractical under normal and unanticipated conditions. In the EPR's, unlike UWMAK-III, the absence of large quantities of lithium leaves no obvious paths for large-scale first wall volatilization. In the ANL design, pressurized water is used as the blanket coolant. In this case, a loss-of-coolant accident (LOCA) would lead to a release of radioactive corrosion products. For the GA and ORNL designs, the blankets are either totally gas-cooled, or partially so. In the event of a depressurization accident, the afterheat in the stainless steel first wall may cause it to melt, with a subsequent release of radioactivity.

An initial list of significant accident sequences is given in Table 1-1. Failures of heat removal systems, missile generation due to magnet

Table 1-1. Significant Accident Sequences.

-
1. Failure of magnets, with missiles or other effects leading to rupture of the primary system, and possible damage of the containment.
 2. Failure of the He cooling system of the magnets with correlated failures of other systems (magnet, primary system coolant, containment integrity, etc.).
 3. Loss of integrity of primary system outside containment (where applicable).
 4. Loss of isolation between primary and higher pressure secondary system (or tertiary system).
 5. Loss of tritium system integrity at many points in reprocessing and fueling system.
 6. Shipment of tritium to site.
 7. Earthquake-induced failures of primary, helium, and tritium systems.
 8. Relatively higher probability, "smaller" tritium spills, particularly as they may arise from the need for remote replacement capability.
 9. Metal-water reactions (chemical, thermal) where possible.
 10. Fires of various kinds; what to do; how to fight them; possible effects.
 11. Loss of power; mechanical failure of on-site power systems.
 12. Loss-of-blanket coolant accident.
 13. Crane failure (dropping of heavy loads).
-

failure, loss of tritium system integrity, and metal-water reactions are of primary concern.

Some special safety considerations arise in the EPR designs. The ORNL device has lithium in its blankets and borated water in its magnetic shield. Of special concern is the possibility of lithium-water thermal or chemical interactions. The GA design has a SiC liner to protect the first wall. Of special concern here are the effects on the first wall should the liner fail. The lithium hydride (LiH) in the shield of the GA reactor is also of concern. LiH melts at 953 K, and operational as well as off-normal transients may cause shield overheating.

UWMAK-III poses some safety questions not examined in the UWMAK-I (and II) studies [1-1]. These include:

1. The high-pressure secondary heat removal system connected to the primary system and the steam generator: $\text{Li-Na-H}_2\text{O}$,
2. The sodium-water divertor plate cooling system,
3. The proposed gross disposal of a large bulk of highly radioactive blanket structure,
4. The back-up heat removal systems,
5. The large tritium inventory, and
6. The escape of primary heat removal coolant (Li) from the primary containment building.

Operational safety questions arise for the various EPR and UWMAK-III designs. The major question concerns the necessity for frequent remote handling and disassembly of blanket components. Because of the high level of induced activity, "hands on" maintenance may not be possible. The type of pipe connectors, joints, electrical connections, and other service connections will have to be compatible with remote

operations. The failure of such connections and joints may be a common problem for all designs. For example, the primary piping may not be able to meet Seismic-Class I design specifications if non-welded connections are used.

For UWMAK-III the following operational questions arise:

- i. The expansion of the blanket upon heating,
- ii. The inerting of the heat exchanger building, how it is to be maintained, and its pressure rating, and
- iii. The remote handling-type connectors to the coolant systems.

1.3. Identification of Major Gaps in Safety-Related Information

In order to assess the safety of these reactors, and to design engineered safeguards to mitigate or eliminate hazards, detailed information on: (a) potential plant malfunctions, and (b) potential accident initiators, must be examined. Table 1-2 lists potential plant malfunctions for conceptual fusion power reactors. Headings I-VI were taken from the Battelle North-West Laboratory (BNWL) report on fusion reactor safety [1-8]. In this study, the three EPR designs and UWMAK-III are compared to this table.

Examination of Table 1-2 shows that tritium release was discussed in more detail than any other malfunction (though only briefly). In addition, no criteria or design bases were given for the release discussed, and no specific accident was evaluated.

Table 1-3 lists potential initiating mechanisms for accidents in conceptual fusion reactors. The entries (excluding the category "other") were also taken from the BNWL report [1-8]. Again, the three EPR and the UWMAK-III designs were compared to the entries in Table 1-3. Although

Table 1.2. Potential Plant Malfunctions for Conceptual Fusion Reactors.

Malfunction	EPR's			UWMK-III
	ANL	ORNL	GA	
I. Magnet System Malfunction				
1. Magnet Failure	M	M	M	M
II. Cryogenic System Malfunction				
1. Failure of Cryogenic System	M	M	M	M
III. Liquid Metal Reactions				
1. a. Lithium Fire	NA	M	NA	M
b. Lithium Exposure to Concrete	NA	ND	NA	ND or M
2. a. Sodium Fire	NA	M	NA	M
b. Sodium Exposure to Concrete	NA	ND	NA	ND or M
3. a. Potassium Fire	NA	M	NA	NA
b. Potassium Exposure to Concrete	NA	ND	NA	NA
IV. Containment Structural Failure Malfunction				
1. Containment Penetration	M	ND	M	M
a. Internally Generated Missiles	M	ND	M	M
b. Externally Generated Missiles	M	ND	M	M
2. Containment Overpressurization	M, DBG, NAE	ND	ND or M	TB, DAG, NAE
V. Hazardous Material Release				
1. Tritium Release from Storage	TB, NCDBG, NAE	TB, NCDBG, NAE	TB, NCDBG, NAE	TB, NCDGB, NAE
2. Tritium Release from Waste Handling Activities	TB, NCDBG, NAE	TB, NCDBG, NAE	TB, NCDBG, NAE	TB, NCDGB, NAE
VI. Primary Boundary Failures				
1. Severe Overpower Transients	M	ND	ND	ND
2. First Wall Failure	M	M	M	M
3. Failure of Spent Fuel Recovery System	M	TB	TB	M
4. Loss of Primary Cooling	M	ND	ND	M
5. Loss of Secondary Cooling	M	ND	ND	ND
VII. First Wall Volatilization	ND	ND	ND	ND

Note: NA = Not Applicable ND = Not Discussed
M = Mentioned TB = Treated Briefly
TM = Treated Moderately T = Treated
NAE = No Accident Evaluation AE = Accident Evaluated
NCDBG = No Criteria or Design Basis Given DBG = Design Basis Given

Table 1.3. Potential Initiating Mechanisms for Accidents for Conceptual Fusion Reactors.

Initiating Mechanism	EPR's			UMMAK-III
	ANL	ORNL	GA	
<u>Internal to Plant</u>				
1. Pipe Rupture	M	ND	ND	ND
2. Pipe Blockage	M	ND	ND	ND
3. Pump Failure	M	ND	ND	ND
4. Change in Magnetic Field	M	ND	ND	ND
5. Change in Plasma Feed Rate	ND	ND	ND	ND
6. Plasma Instabilities	ND	ND	ND	ND
7. Failure of He Containment Boundary	M	ND	M	M
8. Radiation Damage	M	M	M	TM
9. Loss of Cooling System Serving Cryogenic Devices	M	M	ND	M
10. Steam Generator Tube Failures	M	ND	ND	ND
11. Intermediate Heat Exchanger Tube Failures	NA	ND	NA	ND
12. Human Errors	ND	ND	ND	ND
13. Pipe Whip	ND	ND	ND	ND or M
14. Failure of Power Feedback Control Systems	ND	ND	ND	ND or M
15. Mechanical Yield Strengths Exceeded, due to:				
a. Radiation Degradation	M	M	M	TM, NAE
b. Forces from High Transient Electrical Fields	M	ND	ND	ND
c. Pressure Differences Across Vacuum Boundaries	M	?	?	M
d. Excessive Thermal Cycling	M	M	M	TB
16. Parting of Electrical Conductor	M	M	ND	ND or M
17. Magnet Coil Breakage	M	M	M	M
18. Failure of Pressure Relief Components	ND	ND	ND	ND
19. Overshielding of Pressure Relief Systems	ND	ND	M	ND
20. Dewar System Rupture	M	M	M	M
21. Mercury Vapor Pressures Grossly Exceeding Pump Cell Pressures	?	?	?	?
22. Power Loss	M	ND	M	M
23. Staff Sabotage	M	ND	ND	ND
<u>External to Plant</u>				
1. Earthquake	M	ND	T, NAE	TB
2. Tornado	ND	ND	T, NAE	TB
3. Flood	ND	ND	T, NAE	ND
4. Missile Impact (Plane, Meteor)	M	ND	TB	M
5. Explosions, Blasts, Fires (Nearby Industries, Access Routes)	ND	ND	ND	ND
6. Loss of Offsite Power	M	ND	M	TB
7. Sabotage	ND	ND	ND	ND
<u>Other</u>				
1. Metal-Water Reactions	M	ND	ND	ND
2. ECCS	ND	ND	ND	ND
3. Fire	M	M	M	M

Note: NA = Not Applicable ND = Not Discussed
M = Mentioned TB = Treated Briefly
TM = Treated Moderately T = Treated
NAE = No Accident Evaluation AE = Accident Evaluated
NCDBG = No Criteria or Design Basis Given DBG = Design Basis Given

several items are mentioned or treated briefly, no accidents are evaluated and no design bases or criteria are given. Only the GA design treats external events (e.g., earthquakes, tornados, and floods), but no accident evaluations are given. The UWMAK-III design treats radiation damage and degradation, but again, no accidents are evaluated.

It should be emphasized that these designs are in a conceptual stage and that the time available was used to concentrate on other aspects of the design. These comparisons do, however, point out where future design efforts must focus and which safety-related questions addressed, if a safe plant is to be built.

1.4. Conclusions

As with most fusion power reactors operating on the deuterium-tritium fuel cycle, the principle hazards of the EPR and UWMAK-III designs are tritium and induced activity in the first wall and blanket structure. If the EPR designs do indeed have a small tritium inventory (due to the absence of breeding), then first wall activity will be the dominant hazard.

The principle accident paths at this time are difficult to identify because of a lack of detailed design and safety-related data. Preliminary examination of these designs indicates several potentially significant accident sequences which can lead to a release of radioactivity.

Comparison of the EPR and UWMAK-III designs to lists of potential accident initiators and lists of potential plant malfunctions yields a deficiency of information. While this is attributable to emphasis on other aspects of design by the design groups, they do indicate areas where future work must focus.

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2. TRITIUM SYSTEM SAFETY

2.1. Discussion

One of the features of the tritium system present in an experimental power reactor (EPR), which may be absent in smaller devices, including the Tokamak fusion test reactor (TFTR), is a device for the continuous de-tritiation of the atmosphere in the reactor containment structure. The development of such systems on the scale needed for fusion reactors is in its early stage, so that the designs and considerations presented in the three EPR designs considered [2-1 to 2-3] are often incomplete. This makes a detailed analysis of a specific system impossible, and a comparative assessment of the three EPR's sensibly so. Therefore, the approach was taken here that the de-tritiation system would be examined in only the most generic manner. Where specific design data was needed and available, reference has been made to the Argonne National Laboratory (ANL) EPR design [2-1]. These details are summarized in Table 2-1, and Figures 2-1 and 2-2. Table 2-1 delineates the target tritium handling facility parameters; Figure 2-1 schematically represents the tritium systems within the reactor; and Figure 2-2 shows the isotope separation system.

The details of the operation of a de-tritiation facility remain to be established on the basis of experimental projects, such as the Tritium System Test Assembly (TSTA) at Los Alamos Scientific Laboratory (LASL). In the most general terms, however, a de-tritiation facility will consist of a means of circulating the atmosphere of the reactor hall or other

Table 2-1. Summary of Tritium Handling Facility Parameters in the ANL EPR (from Reference [2-1]).

Parameter	$\frac{B_{TFC}}{B_{max}}$	
	8 T	10 T
General		
Power during Burn (MW)*	200	500
Burn Cycle Duty Factor (%)	75	
Plant Availability Factor (%)	67	
Tritium Burnup (g/day)	26	64
Throughput/Burnup Ratio	50	
Tritium Delivery Rate (g/hour)	60	150
Fuel Cycle Turnaround Time (hours)	4	
Plant Inventory (kg)	0.6	1.5
Annual Tritium Consumption at 50% Duty Factor (kg)	6.4	16
Tritium Inventory Disposition		
Cryosorption Pumps (g)	240†	600†
Getter Beds (g)	240†	600†
Distillation Columns (g)	10†	25†
Fuel Cycle Hardware (g)	10†	25†
Storage (g)	~500†	~1,300†
Anticipated Mean Inventory (g)	600	1,500
Isotopic Enrichment by Cryogenic Distillation		
Number of Columns	4	
Number of Equilibrators	1	
Operating Pressures (Torr)		
Column 1	500	
Columns 2-6	1,000	
Reflux Ratio		
Column 1	20	
Columns 2-6	15	
Feed to Column 1 (g/hour)	~100	~250
Height Equivalent Theoretical Plate (cm)	2.54	
Liquid Holdup (m^3/m^3 of packing)	0.18	
Superficial Vapor Velocity (cm/s)	6.1-30.5	
Number of Theoretical Stages per Column	30	
Maximum Column Diameter (cm)	< 5.1	

*The power during burn depends upon a number of factors involved in the details of the burn pulse. These values are representative of the range of burn cycles studied.

†Maximum value at any single time.

(Table continued on following page)

Table 2-1 (continued)

Parameter	Value
Atmospheric Clean-Up System	
Reactor Hall Volume (m^3)	$\sim 2.8 \times 10^5$
Baseline Tritium Level ($\mu Ci/m^3$)	5
Clean-Up Time (days)	≤ 2
Maximum Decontamination Factor	10^6
Air Circulation Rate at Emergency (m^3/s)	47.2
Volume of Catalyst (m^3)	20
Catalyst Operating Temperature (K)	300
Miscellaneous	
Tritium Level in First Wall Cooling Water (Ci/l)	≤ 2
Tritium Release Rate to Environment (Ci/day)	100
Volume of First Wall Cooling Water (l)	10^5
Fuel Cycle	
Nature of Fuel Processing and Recycle Systems	Nonmetallic Element Removal Debris Removal Isotopic Enrichment Fuel Storage Fuel Delivery

(End of Table 2-1)

Figure 2-1. ANL EPR Fuel Cycle Scenario.

Figure 2-2. ANL EPR Cryogenic Enrichment Scenario.

areas through a bed material which is able to chemically combine with and stabilize tritium. The most extensively developed catalytic materials stabilize tritium by oxidation and hydration into a solid material. In considering the events which could lead to off-normal operation of such a system, one must consider the requisite conditions to be satisfied for successful operation. For example, such a catalytic system will require that the temperature be controlled appropriately. This arises as the result of a competitive demand for rapid de-tritiation, requiring high bed temperatures, and high retentivity for tritium, which will generally require low temperatures. This requirement may be particularly important if the tritium is to be recovered from the beds after an appropriate period of use.

Thus, one may consider an event tree for the de-tritiation of beds, with the top event designated as an abnormal temperature. If the abnormality in temperature is that the temperature is too low, then the result will be that tritium will not be abstracted from the reactor hall at as great a rate as anticipated. This, quite probably, will have the same result as other events which render the de-tritiation beds inoperative, such as a failure of the blowers used to circulate the atmosphere through the beds. It will be presumed that at some point the system designer must cope with the question of such a gross failure of the de-tritiation facility to work. Therefore, we shall not consider this question within the more specialized context of this report.

The second possibility, in which the temperature of the de-tritiation bed becomes excessively high, may occur as a result of a wide range of events. Presumably, even the normal operation during the abnormal spill of a large amount of tritium can cause this top event, since the

uptake of tritium is expected to be exothermic. However, another class of events is of greater concern, and must be considered in detail. If, as described, a large tritium spill occurs, heating of the bed will induce a corresponding reduction in the efficiency of the bed operation. This would be reflected in a decrease in the efficiency of de-tritiation. In other words, there would be a somewhat non-linear character to the ability of the bed to abstract tritium from the reactor hall following a spill. It seems reasonable that this characteristic could be factored into the design of the de-tritiation beds, and might simply require a modest overdesign of the beds. On the other hand, suppose a large spill not of tritium but of deuterium occurred. In the absence of the de-tritiation device, the deuterium spill would be of concern only insofar as it represented a mode by which a fire might be started. In the presence of the de-tritiation system, an additional hazard occurs. Since deuterium and tritium will be indistinguishable to the de-tritiation system (with the possible minor exception of isotope effects on the various rate processes occurring within the bed), the deuterium would also be picked up by the de-tritiation system, and for a sufficiently large spill abnormal bed temperatures might ensue. Under these conditions, the de-tritiation system is able to release tritium due to equilibrium effects, precisely in the fashion in which non-linear adsorption rates might be encountered in a large tritium spill clean-up.

However, under these circumstances the tritium represents a new hazard, not intrinsically associated with the initial event. It is presumably true that a sufficient application of heat to the bed could cause this tritium release, perhaps independently of the source of the heat. It is not possible to estimate the magnitude of the heat required,

since a quantitative design for a de-tritiation device has not been made. However, this represents a potential hazard, and not an atypical one, in which the desire to ameliorate the effects of a severe problem leads one to increase the likelihood of small ones.

To facilitate the assessment of the coupling of those events which do not involve a direct release of tritium to events that do lead to a tritium release from the de-tritiation system, the partial event tree shown in Figure 2-3 has been constructed. A complete analysis would require knowledge of the nature of the initiating event in sufficient detail to predict its effect on containment integrity. This question would be addressed most appropriately through another event tree, which has as its top event the release of tritium to the environment, and which considers as one means of effecting such a release the top event of Figure 2-3. However, for the purposes of this consideration it may be sufficient to posit that some initiating events in Figure 2-3, being similar to those considered previously [2-4] as means of violating the containment, do in fact have the potential to lead to a release of tritium or other hazardous materials beyond the site boundary.

2.2. Conclusion

The use of de-tritiation facilities, while still in a preliminary stage of development, appears to lead to a variety of new routes to the off-normal release of tritium. Due to the possibility of the removal of tritium from the beds, there exist a variety of events which themselves, in the absence of a de-tritiation system, do not lead to the release of tritium beyond the site boundary, but which may both violate the integrity of the containment and bring about the release of tritium from the de-tritiation system indirectly. Suppressing the potential for a large-scale

release of tritium may enhance the probability of a small release, with overall safety consequences which may in principle be either favorable or unfavorable.

2.3. References for Chapter 2

- 2-1. Stacey, W.M., et al., "Tokamak Experimental Power Reactor Conceptual Design," Argonne National Laboratory, ANL/CTR-76-3, August 1976.
- 2-2. "Oak Ridge Tokamak Experimental Power Reactor Study-- 1976," Oak Ridge National Laboratory, ORNL/TM-5573, 1976.
- 2-3. Baker, C. C., "Experimental Fusion Power Reactor Conceptual Design Study," Electric Power Research Institute, EPRI-ER-289, December 1976.
- 2-4. Okrent, D., et al., Nuclear Engineering and Design 39:215, 1976.

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3. FIRST WALL INTEGRITY

3.1. Introduction

The loss of cooling function to the first wall has been identified [3-1,3-2] as a potential hazard in a fusion reactor. If one posits that cooling is lost, but that reactor operation is not terminated, then it is possible to reach conditions leading to rapid failure of the first wall in times of 1 to 100 seconds [3-3], with a design dependence to this time. It may not be possible to cease reactor operations sufficiently rapidly to prevent severe consequences, and all such events may not be detectable. Hence, it may be incumbent on the reactor designer to consider, as design alternatives, means by which this failure time may be significantly lengthened and by which the event may be more readily detected.

In the context of fission reactors, a loss-of-coolant accident is considered to be among the most demanding events with which the designer must cope. For example, elaborate secondary cooling systems are routinely incorporated into fission reactor designs. In the case of fusion reactors, it is possible that a loss-of-coolant accident (LOCA) can lead directly to the significant dispersal of hazardous materials, or can serve as an initiating event for further disruptive events. Historical precedence should make us carefully consider the loss-of-coolant event as a potentially serious one in a fusion reactor. Care should also be exercised not to assume that the solution to this potential problem lies in the same direction as in fission reactors, such as incorporating redundant, auxiliary, and emergency cooling systems. Various design alternatives may either be unsuitable to fission reactors or have additional problems not encountered in fusion reactors. Hence, it is the purpose of

this chapter to review the three experimental power reactor (EPR) designs [3-4 to 3-6] in light of the demands for the maintenance of first wall integrity, evaluate the ways in which the designs differ, and attempt to identify significant advantages inherent in the various designs.

3.2. Review of the EPR Blanket and First Wall Designs

The blanket and first wall characteristics of the three EPR designs reviewed are summarized in part in Tables 3-1 through 3-3 and Figures 3-1 through 3-5.

The Argonne National Laboratory (ANL) EPR design [3-4] is summarized in Tables 3-1 and 3-2 and Figures 3-1 and 3-2. This design consists of an independently cooled first wall structure, shown in Figure 3-1, which is coupled to the blanket structure shown in Figure 3-2. The coolant for the first wall will be water, while helium and water are options considered for the blanket coolant. The first wall, the operating characteristics of which are summarized in Tables 3-1 and 3-2, is constructed of Type 316 stainless steel with a beryllium coating, and operates at below 773 K. The coolant temperature at the first wall varies as a function of position from 313 to 583 K.

The General Atomic (GA) EPR design [3-5] is summarized in Table 3-3 and Figures 3-3 and 3-4. The blanket incorporates a silicon carbide shield operating at 1,723 K, and independent helium-cooled blanket modules (shown in Figures 3-3 and 3-4). Cooling of the silicon carbide liner is primarily by radiation.

The Oak Ridge National Laboratory (ORNL) EPR design is shown in Figure 3-5. The blanket is composed of a few large modules, containing helium-cooled potassium or lithium and appropriate reflector and shield materials. Cooling functions, while not completely described in the

Table 3-1. ANL EPR First-Wall Design Parameters.

Design Description

- Free-standing, stainless steel vacuum wall with rib and spar reinforcing.
- Detachable, water-cooled stainless steel panels to shield vacuum wall from plasma.
- Low-Z coating on plasma-exposed face of coolant panel for high-Z impurity control.

Design Parameters

Vacuum chamber	
Material	316 SS
Design stress (ksi)	10
Major radius (m)	6.25
Minor radius (m)	2.4
Volume (m ³)	711
Wall area (m ²)	592
Wall thickness (cm)	2
Ring and spar (cm)	
Width	5
Depth	11
Ports	
Vacuum (0.95 m diameter)	32
Heating (0.75 m diameter)	16
Experimental (1.5 m diameter)	4
Total port area (m ²)	31
Current breaker	
Material	Cr ₂ O ₃
Form	Coating
Preparation	Chemical bond
Coolant panel	
Material	316 SS
No.	352
Area per panel (m ²)	1-2
Length (m)	1-2
Width (m)	~1
Total panel thickness (cm)	~1
Thickness front wall (cm)	0.5
Low-Z coating	
Material	Beryllium
Thickness (μm)	100-200
Preparation	Plasma spray
Coolant	H ₂ O

Table 3-2. ANL EPR First-Wall Operating Parameters and Design Limits*.

<u>Nominal Operating Conditions</u>	
Capacity factor (%)	50
Operating cycle (s)	5
Startup	35
Burn	5
Shutdown	15
Exhaust and replenishment	
Average power loading during burn (MW/m ²)	0.5
Neutron	0.13
Radiation, conduction, convection	
<u>Operating Parameters</u>	
Stainless steel vacuum wall	
Maximum temperature (°C)	<500
Minimum yield stress at 500°C (ksi)	17
Maximum annual fluence (n/m ²)	6×10^{25}
Neutron Damage (dpa/yr)	2.8
Helium generation (appm/yr)	54
Hydrogen generation (appm/yr)	133
Stainless steel coolant panel	
Maximum temperature (°C)	380
Minimum yield stress at 500°C (ksi)	17
Maximum annual fluence (n/m ²)	6×10^{25}
Neutron Damage (dpa/yr)	2.8
Helium generation (appm/yr)	54
Hydrogen generation (appm/yr)	133
Maximum heat deposition (W/cm ³)	5.8
Maximum ΔT across panel surface (°C)	20
Maximum ΔT through panel face (°C)	
With Argon shutdown	100
Without Argon shutdown	75
Maximum ΔT during burn cycle (°C)	100
Maximum thermal strain range (%)	
Operating cycle	0.14
Burn cycle	0.09
Beryllium coating	
Maximum surface temperature (°C)	407
Helium generation (appm/yr)	780
Hydrogen generation (appm/yr)	13
Maximum erosion rate (um/yr)	30
Water coolant	
Maximum pressure (psi)	3000
Velocity (m/s)	1.6
Inlet temperature — first panel (°C)	40
Exit temperature — eighth panel (°C)	310
Pumping power (MW)	<1
Vacuum wall	
Design life (yr)	10
Integrated neutron wall loading (MW-yr/m ²)	2.5
Yield strength — 10 yr (ksi)	75
Uniform elongation — 10 yr (%)	>1
Radiation swelling — 10 yr (%)	<4
Limiting criterion	Ductility
Coolant panel	
Design life (yr)	5
Total burn cycles — 5 yr	10 ⁶
Fatigue lifetime (yr)	5
Radiation lifetime (yr)	8
Limiting criterion	Thermal fatigue
Low-Z coating	
Design life (yr)	1-5
Limiting criterion	D-T sputtering

* Based on a neutron wall load of 0.5 MW/m² and a plant capacity factor of 50%.

Table 3-3. GA EPR Blanket Design Parameters.

Liner Material	Silicon Carbide
Liner Thickness	1.0 cm
Maximum Liner Temperature	1450°C
Liner Neutron Attenuation Factor	1.04
Blanket Material	Stainless Steel/Graphite
Module Diameter	42 cm
Number of Modules	1900
Blanket Wall Area	493 m ²
Max. Neutron Wall Loading	0.85 MW/m ²
Avg. Neutron Wall Loading	0.67 MW/m ²
Maximum Blanket Temp.	600°C
Coolant	Helium
Pressure	4.80 x 10 ⁶ Pa
Inlet Temp.	325°C
Outlet Temp.	575°C
Flow Rate	234 kg/sec
Blanket Delta P	37,000 Pa
Blanket Pumping Power	2.8 MW
Weight of Single Module	276.7 kg
Weight of Modules	526,000 kg
Weight of Support Structure	93,000 kg
Weight of Graphite Filler	75,000 kg
Weight of SiC Liners	9,000 kg
Total Blanket Weight	703,000 kg
Blanket Neutron Attenuation Factor	5
Heat Deposition (max)	9.7 W/cm ³
Heat Deposition (avg)	3.3 W/cm ³

Figure 3-1. ANL EPR Coolant Panel.

Figure 3-2. ANL EPR Blanket Block.

Figure 3-3. GA Blanket Support Structure.

Figure 3-4. GA Blanket Module Attached to Module Support Ring.

3-10

Figure 3-5. ORNL EPR Reference Design Blanket Segment.

design, appear to be neither redundant nor modular on a scale smaller than that dictated by the sectional construction of the reactor itself.

Clearly, the three designs differ substantially. It is therefore of interest to analyze whether these design differences can lead to a significant difference in the ability of the designs to withstand a partial or complete loss of cooling function. The major differences between the designs seem to be related to the question of whether cooling is provided independently to modular portions of the blanket, and whether the possibility of shifting the thermal load from a primary to a secondary means of heat removal exists. Therefore, a general model for analysis of a LOCA in such a reactor has been developed.

3.3. Model for a LOCA in a Modular Blanket

The analysis of blanket and first wall designs similar to those proposed for the EPR's reviewed above is based on a model which evaluates the steady-state thermal response of a blanket composed of independent modules. Incorporated in this model is both a primary and a secondary means of heat removal to a coolant, and the possibility of the interchange of heat between modules by radiation. It is presumed that the modules are otherwise thermally isolated from each other, since this represents the most conservative assumption about thermal transport capabilities.

The thermal response of a blanket module which can be characterized by a single temperature, T , at the first wall, may be given by the solution of the differential equation

$$\rho C \frac{\partial T}{\partial t} = wkV^2T + Q + B - \bar{B} - U_{eff}(T - T_o), \quad (3-1)$$

where

T = wall temperature,

ρ = density,

C = heat capacity,

w = the wall thickness,

k = the thermal conductivity of the wall,

U_{eff} = an effective heat transfer coefficient to a coolant at a temperature T_o ,

Q = the thermal load on the first wall,

B = the radiation loss = $\epsilon \sigma T^4$, and (3-2)

$\bar{B}$ = the radiative load

$$= \int_S B F \, ds \quad (3-3)$$

where F is the view factor, and the integral is taken over the surface of the first wall. In the following analysis it will be assumed that there is no conduction along the first wall, as stated above. Hence, the term $\nabla^2 T$ in Equation (3-1) can be set to zero. To simulate the consideration of primary and secondary thermal transport capabilities, the heat transfer coefficient from the first wall, U_{eff} , will be given as

$$U_{\text{eff}} = \begin{cases} U_o & \text{for normal local conditions} \\ \alpha U_o & \text{for LOCA conditions} \end{cases} \quad (3-4)$$

Analysis in a Spherical Geometry. While obviously a Tokamak reactor is not even remotely related to a sphere, the solution of the above equations in the case of a spherical reactor is particularly simple and elucidates the basic characteristics of the problem without excessive mathematical complexities. This arises from the fact that the view factor in a sphere is a constant. Hence, let us consider a spherical

reactor, in which some region has undergone a LOCA. Specifically, let the segment of the spherical-shell first wall with polar angle given by $0 < \theta < \theta_0$ be the region in which the LOCA has occurred, and let the rest of the spherical shell, $\theta_0 < \theta < \pi$, have normal cooling function. Qualitatively, if the LOCA occurs in a single module, this corresponds to a first wall and blanket composed of a number of modules, N , given approximately as the ratio of the surface area of the LOCA region to that of the entire shell. Hence, for a shell of radius R , N is given by

$$N = \frac{\int_0^{\theta_0} 2\pi R^2 \sin\theta \, d\theta}{\int_0^{\pi} 2\pi R^2 \sin\theta \, d\theta} = \frac{2}{1 - \cos\theta_0}. \quad (3-5)$$

This is plotted in Figure 3-6. As can be seen from this figure, a blanket composed of a few hundred modules will have a characteristic value of θ_0 of about 10° ; a few thousand modules leads to a θ_0 of about 2° .

The thermal analysis can be performed directly by the evaluation of $\bar{B}$. Since, in a sphere, the thermal load is independent of position, only two temperatures exist; T_1 in the module undergoing a LOCA, and T_2 elsewhere. Hence, from Equation (3-3),

$$\bar{B} = \frac{1}{4\pi R^2} \int_0^{\theta_0} B_1 \, dA + \frac{1}{4\pi R^2} \int_{\theta_0}^{\pi} B_2 \, dA, \quad (3-6)$$

where

$$B_i = \epsilon \sigma T_i^4.$$

Then, since $dA = 2\pi R^2 \sin\theta \, d\theta$,

Figure 3-6. θ_0 as a Function of the Number of Modules.

$$\bar{B} = \epsilon \sigma T_1^4 \sin^2\left(\frac{\theta}{2}\right) + \epsilon \sigma T_2^4 \cos^2\left(\frac{\theta}{2}\right). \quad (3-7)$$

Thus, at steady state and under the assumptions stated above, Equations (3-1), (3-2), (3-4), and (3-7) may be combined to give

$$Q + \epsilon \sigma (T_2^4 - T_1^4) \cos^2\left(\frac{\theta}{2}\right) - \alpha U_o (T_1 - T_o) = 0$$

and (3-8)

$$Q + \epsilon \sigma (T_1^4 - T_2^4) \sin^2\left(\frac{\theta}{2}\right) - U_o (T_2 - T_o) = 0.$$

It is convenient to introduce the following dimensionless parameters:

$$\begin{aligned} \tau_i &= T_i/T_o, \\ \phi &= Q/U_o T_o, \text{ and} \\ \lambda &= Q/\epsilon \sigma T_o^4. \end{aligned} \quad (3-9)$$

Combining Equations (3-8) and (3-9) gives

$$\lambda + (\tau_2^4 - \tau_1^4) \cos^2\left(\frac{\theta}{2}\right) - \frac{\alpha \lambda}{\phi} (\tau_1 - 1)$$

and (3-10)

$$\lambda + (\tau_2^4 - \tau_1^4) \sin^2\left(\frac{\theta}{2}\right) - \frac{\lambda}{\phi} (\tau_2 - 1).$$

The case in which $\alpha = 0$ corresponds to the absence of any secondary heat removal function; the solution of Equation (3-10) then reduces to

$$\tau_1 = \{[1 + \phi \sec^2\left(\frac{\theta}{2}\right)]^4 + \lambda \sec^2\left(\frac{\theta}{2}\right)\}^{1/4}$$

and (3-11)

$$\tau_2 = 1 + \phi \sec^2\left(\frac{\theta}{2}\right).$$

The solution for τ_1 is plotted in Figure 3-7 as a function of λ for various values of ϕ . τ_2 can also be calculated, but since $\tau_1 > \tau_2$ it is always τ_1 which will represent the limitation on the integrity of the first wall. As can be seen from this figure, the effect of θ_o , and hence

Figure 3-7. τ_1 as a Function of λ for Various Values of ϕ .

of N , is small over the range from 1 to 50° (N varying from about 6 to about 13,000).

The effect of a secondary heat removal system can be seen in Figure 3-8. For $\phi = 0.2$, corresponding to a normal first wall temperature of $1.2 T_0$, again only a small effect of θ_0 can be seen. However, the effect of α can be large. Suppose that for the same design, a maximum value of $\tau_1 = 3$ is allowed (e.g., $T_0 \approx 573$ K, and the first wall can withstand $\sim 1,723$ K for considerable periods of time). Then, Figure 3-8 shows that the allowable value of λ increases by a factor of about 5 as α increases from 0 to 0.1; that is, by providing a secondary heat removal system with 10% of the capability of the primary one. For example, suitable designs for the ANL EPR might be developed to utilize the blanket coolant system for cooling the first wall, should the primary first wall coolant system fail.

Analysis in Toroidal Geometry. The corresponding analysis in a Tokamak is far more difficult, since now F is not a constant, and each module has a different temperature. For an exact solution, an array of N non-linear equations with N variables must be solved. While such problems are within the capability of computational techniques, the analysis has been pursued only as far as needed to obtain a semi-quantitative comparison to the spherical case considered above.

The toroidal geometry considered is a special case of a circular torus, as shown in Figure 3-9. Two points, a and b , are taken as being on the toroidal shell with major and minor radii R_0 and r_0 , respectively. Points c and d are on the arc described by the rotation of the minor circle about the toroidal center, O . It is convenient to normalize all dimensions by the minor radius, r_0 . The Cartesian coordinates (x, y, z)

Figure 3-8. τ_1 as a Function of λ for Various Values of α .

Figure 3-9. Toroidal Configuration.

are also shown, with the x-y plane defined by the midplane of the torus and the x- and ϕ -axes coincident. Table 3-4 gives the location of the points a through d in normalized toroidal and Cartesian geometry. The aspect ratio, P, is given as R_0/r_0 .

Table 3-4. Points in the Torus of Figure 3-9.

Point	Toroidal Coordinates	Cartesian Coordinates
a	$(1, \theta_1, 0)$	$((P + \cos \theta_1), 0, \sin \theta_1)$
b	$(1, \theta_2, \phi)$	$(\cos \phi (P + \cos \theta_2), \sin \phi (P + \cos \theta_2), \sin \theta_2)$
c	$(0, 0, 0)$	$(P, 0, 0)$
d	$(0, 0, \phi)$	$(P \cos \phi, P \sin \phi, 0)$

The view factor between two areas, A_1 and A_2 , is given by

$$A_1 F_{12} = A_2 F_{21} = \frac{1}{\pi} \int \frac{\cos \chi_1 \cos \chi_2}{|ab|^2} dA_1 dA_2. \quad (3-12)$$

Direct calculations can then be made of the terms needed in Equation (3-12). From the law of cosines,

$$\frac{\cos \chi_1}{|ab|} = \frac{|ab|^2 + |ac|^2 - |bc|^2}{2|ac||ab|^2}$$

and

$$\frac{\cos \chi_2}{|ab|} = \frac{|ab|^2 + |bd|^2 - |ad|^2}{2|bd||ab|^2}, \quad (3-13)$$

and from Table 3-4,

$$|ac|^2 = |bd|^2 = 1,$$

$$\begin{aligned}
|bc|^2 &= 1 + 2P(P + \cos\theta_2)(1 - \cos\phi), \\
|ad|^2 &= 1 + 2P(P + \cos\theta_1)(1 - \cos\phi), \text{ and} \\
|ab|^2 &= 2(P + \cos\theta_1)(P + \cos\theta_2)(1 - \cos\phi) \\
&\quad + 2[1 - \cos(\theta_2 - \theta_1)].
\end{aligned}
\tag{3-14}$$

From Equations (3-13) and (3-14),

$$\frac{\cos x_1}{|ab|} = \frac{\cos\theta_1(P + \cos\theta_2)(1 - \cos\phi) + [1 - \cos(\theta_2 - \theta_1)]}{2(P + \cos\theta_1)(P + \cos\theta_2)(1 - \cos\phi) + 2[1 - \cos(\theta_2 - \theta_1)]}$$

and

$$\tag{3-15}$$

$$\frac{\cos x_2}{|ab|} = \frac{\cos\theta_2(P + \cos\theta_1)(1 - \cos\phi) + [1 - \cos(\theta_2 - \theta_1)]}{2(P + \cos\theta_1)(P + \cos\theta_2)(1 - \cos\phi) + 2[1 - \cos(\theta_2 - \theta_1)]}.$$

Equations (3-12) and (3-15) can be used to compute the required thermal transport between modules. The fraction of the radiative output from module i intercepted by module j is $A_i F_{ij}$, hence the heat transferred is $A_i F_{ij} B_i$, where $B_i = \epsilon \sigma T_i^4$ and T_i is the temperature of module i . Thus, the thermal steady state for each module requires that

$$A_j Q + \sum_i A_i F_{ij} B_i - A_i B_j - A_j U_{\text{eff}} (T_j - T_0). \tag{3.16}$$

As a first step, one may simply consider what the view factors might dictate with respect to how well radiative transfer can disperse the heat from a module undergoing a LOCA. In particular, the above calculation of F_{ij} presumes that the two modules are in optical line of sight. However, this must be further restricted to optical line of sight through the interior of the torus. This may be considered by computing the equation of the straight line between points a and b of Figure 3-9, and computing whether any other intersections of that line with the toroidal shell exist and lie between points a and b . The equation of a

toroidal shell is given as

$$p = 1 \quad (\text{toroidal coordinates})$$

or

$$x^2 + y^2 + z^2 + p^2 - 2P(x^2 + y^2)^{\frac{1}{2}} = 1$$

(Cartesian coordinates),

where

$$p = r/r_0,$$

$$x = x/r_0,$$

$$y = y/r_0, \text{ and}$$

$$z = z/r_0.$$

The equation of the line between points a and b is given as

$$x = Az + B$$

and

$$y = Cz + D,$$

where

$$A = \frac{\cos\phi(P + \cos\theta_2) - (P + \cos\theta_1)}{\sin\theta_2 - \sin\theta_1},$$

$$B = \frac{\sin\theta_2(P + \cos\theta_1) - \sin\theta_1\cos\phi(P + \cos\theta_2)}{\sin\theta_2 - \sin\theta_1},$$

$$C = \frac{\sin\phi(P + \cos\theta_2)}{\sin\theta_2 - \sin\theta_1}, \text{ and}$$

$$D = \frac{\sin\theta_1\sin\phi(P + \cos\theta_2)}{\sin\theta_2 - \sin\theta_1}.$$

(3-19)

Points lying both on the toroidal shell and on the line defined by points a and b must satisfy Equations (3-17) and (3-18) simultaneously. Such common points will have values of z given by the equation

$$\begin{aligned}
& (1 + A^2 + C^2)z^4 + 4(1 + A^2 + C^2)(AB + CD)z^3 \\
& + [4(AB + CD)^2 + 2(1 + A^2 + C^2)(B^2 + D^2 - p^2 - 1) \\
& - 4p^2]z^2 + 4(AB + CD)(B^2 + D^2 - p^2 - 1)z \\
& + (B^2 + D^2 - p^2 - 1)^2 - 4p^2 = 0.
\end{aligned} \tag{3-20}$$

This equation can be factored by $(z - \sin\theta_1)$ and $(z - \sin\theta_2)$, since these are the solutions corresponding to points a and b. Equation (3-20) then reduces to

$$\begin{aligned}
& z^2 + (2\alpha + \sin\theta_1 + \sin\theta_2)z + (\alpha + \sin\theta_1 + \sin\theta_2)^2 \\
& + 2\beta - \gamma^2 - \sin\theta_1 \sin\theta_2 = 0,
\end{aligned} \tag{3-21}$$

where

$$\begin{aligned}
\alpha &= \frac{2(AB + CD)}{1 + A^2 + C^2}, \\
\beta &= \frac{B^2 + D^2 - p^2 - 1}{1 + A^2 + C^2}, \text{ and} \\
\gamma &= \frac{2p}{1 + A^2 + C^2}.
\end{aligned} \tag{3-22}$$

Then, Equation (3-21) can be solved for the other two solutions, z_3 and z_4 . If z_3 and z_4 are complex conjugates (i.e. if the discriminant of Equation (3-21) is negative), then points a and b are in optical line of sight. If z_3 and z_4 are real, then a and b will be in optical line of sight through the center of the torus only if z_3 and z_4 both lie to one side of points a and b.

The analysis performed assumes that each module has its position defined by equal ranges of the angles θ and ϕ . This causes the area of a module to vary somewhat depending on its location in the torus. For a module defined by the variables

$$\theta = \theta_0 \pm \frac{\Delta\theta}{2}$$

and

(3-23)

$$\phi = \phi_0 \pm \frac{\Delta\phi}{2}$$

The area, A, may be shown to be

$$A = P\Delta\phi\Delta\theta + 2\Delta\phi \cos\theta_0 \sin\left(\frac{\Delta\theta}{2}\right). \quad (3-24)$$

The entire toroidal shell can be defined by $\Delta\theta = \Delta\phi = 2\pi$, hence $A_{\text{torus}} = 4\pi^2 P$. Therefore the normalized area, $\bar{A}$, is

$$\bar{A} = \frac{A}{A_{\text{torus}}} = \frac{\Delta\phi\Delta\theta}{4\pi^2} + \frac{\Delta\phi}{2\pi^2 P} \cos\theta_0 \sin\left(\frac{\Delta\phi}{2}\right). \quad (3-25)$$

Equations (3-15) and (3-25) may then be combined with the information on the existence of an optical line of sight between two points given by Equation (3-21), to compute the view factors in a torus. Such calculations are shown in Figures 3-10 through 3-13 for Tokamaks with aspect ratios of $P = 1.5, 5.0$, and 20.0 , respectively, composed of blankets divided into 24 modules in the poloidal direction and 72 modules in the toroidal direction, and for selected poloidal locations of the module undergoing the LOCA. Qualitatively, the results show that the larger the value of P , the smaller the region over which radiation can disperse the heat radiating from a given module. Comparison of results for fixed P as a function of poloidal location shows that the modules close to the interior of the torus ($\theta = \pi$) have a more restricted region of view than those close to the exterior ($\theta = 0$). This would be expected to cause a LOCA to be more difficult to stabilize in the former case.

These results can be used with Equation (3-16) to solve for a complete temperature profile in the reactor. However, one may note from Figures 3-10 through 3-13 that the portion of the toroidal shell over

[illegible]

Figure 3-10. View Factors in a Torus ($P = 1.5$).

F _{max} = 4.691 E-8	F _{max} = 3.856 E-8	F _{max} = 2.411 E-8	Key
#44444444444444444444UUUUUUUUU J76555544444444444444UUUUUUUUU JB76655554444444444444UUUUUUUUU IE97665554444444444444UUUUUUUUU HFAB76655444444444444433UUUUUUU GE887666544444333333UUUUUUUUUUU FEB9876655443322222UUUUUUUUUUU DOB98766554322100UUUUUUUUUUUUU CCA9876655432100UUUUUUUUUUUUUUU BB987665443100UUUUUUUUUUUUUUUUU AA98876654320UUUUUUUUUUUUUUUUUUU A998876654320UUUUUUUUUUUUUUUUUUU A99887765431UUUUUUUUUUUUUUUUUUU A998876654320UUUUUUUUUUUUUUUUUUU AA98876654320UUUUUUUUUUUUUUUUUUU BB987665442100UUUUUUUUUUUUUUUUU CCA9876655432100UUUUUUUUUUUUUUU DOB98766554322100UUUUUUUUUUUUUUU FEB987665544332222UUUUUUUUUUUUU GE887666554444333333UUUUUUUUUUU HFAB76655444444444444433UUUUUUU IE97665554444444444444UUUUUUUUU JB76655554444444444444UUUUUUUUU *76555544444444444444UUUUUUUUU	*HD987665544444443333UUUUUUUUU JGB8766554444433333UUUUUUUUUUU JE9765544333333222UUUUUUUUUUU IE76543322222222UUUUUUUUUUUUU H743221111111111UUUUUUUUUUUUU #Q0000C90000UUUUUUUUUUUUUUUUU F63100CC90UUUUUUUUUUUUUUUUUUUU D964200UUUUUUUUUUUUUUUUUUUUUUU CA76530UUUUUUUUUUUUUUUUUUUUUUU BA876530UUUUUUUUUUUUUUUUUUUUUUU AS8766430UUUUUUUUUUUUUUUUUUUUU A98765420UUUUUUUUUUUUUUUUUUUUU A987665420UUUUUUUUUUUUUUUUUUUUU A98776E5420UUUUUUUUUUUUUUUUUUU AA9887665442100UUUUUUUUUUUUUUU BBA987765443200UUUUUUUUUUUUUUU CCA9887656544322100UUUUUUUUUUU DDCA98776554433322UUUUUUUUUUUUU DEB9877656554443332UUUUUUUUUUU GFE9877656554444333UUUUUUUUUUU HGCE9877666554444433UUUUUUUUUUU IHC98776565554444443UUUUUUUUUUU JIFF9876566554444443UUUUUUUUUUU JTEA877655544444443UUUUUUUUUUU	*JHEB98766530UUUUUUUUUUUUUUUUU θ = 0 JJHEB98766420UUUUUUUUUUUUUUUUU JIGDA9776541UUUUUUUUUUUUUUUUUUU IHFC98766520UUUUUUUUUUUUUUUUUUU HGEB9876530UUUUUUUUUUUUUUUUUUU GFC9876530UUUUUUUUUUUUUUUUUUUUU FDB876530UUUUUUUUUUUUUUUUUUUUU DC976530UUUUUUUUUUUUUUUUUUUUUUU CA8641UUUUUUUUUUUUUUUUUUUUUUU Poloidal Direction + B9630UUUUUUUUUUUUUUUUUUUUUUUUU A6CUUUUUUUUUUUUUUUUUUUUUUUUUU #UUUUUUUUUUUUUUUUUUUUUUUUUUUUU A6OUUUUUUUUUUUUUUUUUUUUUUUUUU A8630UUUUUUUUUUUUUUUUUUUUUUUUU JY76410UUUUUUUUUUUUUUUUUUUUUUU BA876530UUUUUUUUUUUUUUUUUUUUU CB98765310UUUUUUUUUUUUUUUUUUUUU DCB98765420UUUUUUUUUUUUUUUUUUU FECAB98765420UUUUUUUUUUUUUUUUU GFDB98766531UUUUUUUUUUUUUUUUUUU HGCAB98765420UUUUUUUUUUUUUUUUU IHGD98766430UUUUUUUUUUUUUUUUUUU JIGE98766530UUUUUUUUUUUUUUUUUUU JJFEC98766530UUUUUUUUUUUUUUUUU θ = π	0 = 0.000 ≤ F/F _{max} < 0.010 1 = 0.010 ≤ F/F _{max} < 0.015 2 = 0.015 ≤ F/F _{max} < 0.025 3 = 0.025 ≤ F/F _{max} < 0.040 4 = 0.040 ≤ F/F _{max} < 0.065 5 = 0.065 ≤ F/F _{max} < 0.100 6 = 0.100 ≤ F/F _{max} < 0.200 7 = 0.200 ≤ F/F _{max} < 0.300 8 = 0.300 ≤ F/F _{max} < 0.400 9 = 0.400 ≤ F/F _{max} < 0.500 A = 0.500 ≤ F/F _{max} < 0.550 B = 0.550 ≤ F/F _{max} < 0.600 C = 0.600 ≤ F/F _{max} < 0.650 D = 0.650 ≤ F/F _{max} < 0.700 E = 0.700 ≤ F/F _{max} < 0.750 F = 0.750 ≤ F/F _{max} < 0.800 G = 0.800 ≤ F/F _{max} < 0.850 H = 0.850 ≤ F/F _{max} < 0.900 I = 0.900 ≤ F/F _{max} < 0.850 J = 0.950 ≤ F/F _{max} < 1.000 * = 1.000 = F/F _{max} U = Module unseen # = Reference module
F _{max} = 4.573 E-8	F _{max} = 2.961 E-8	φ = 0 Toroidal Direction + φ = π	
*J76655554444444444444UUUUUUUUU J7655444444444444444444UUUUUUUUU #4444444444444444444444UUUUUUUUU 1754444444444444444444UUUUUUUUU HA765444333333333333UUUUUUUUUUU GC866544333322222UUUUUUUUUUUUU FC9766544322211100UUUUUUUUUUUUU DC8765544321000UUUUUUUUUUUUUUUUU CB9876654320UUUUUUUUUUUUUUUUUUU BA9876654320UUUUUUUUUUUUUUUUUUU AA9877654320UUUUUUUUUUUUUUUUUUU A99877665420UUUUUUUUUUUUUUUUUUU A99877665431UUUUUUUUUUUUUUUUUUU A998876654320UUUUUUUUUUUUUUUUUUU AA98876665431UUUUUUUUUUUUUUUUUUU BB98877665432100UUUUUUUUUUUUUUU CCA98776654332100UUUUUUUUUUUUUUU DOB9877665443322100UUUUUUUUUUUUU FEC98776665444333322UUUUUUUUUUU GFCAB77666554444443333UUUUUUUUU HGC98766665544444444444UUUUUUUUU TGC98766655544444444444UUUUUUUUU JGB87666555544444444444UUUUUUUUU JF97666655555544444444UUUUUUUUU	*JGDAB7765643200UUUUUUUUUUUUUUU JIFC9876554320UUUUUUUUUUUUUUUUU JHEAB76654310UUUUUUUUUUUUUUUUU IGD97665320UUUUUUUUUUUUUUUUUUU HFAB866430UUUUUUUUUUUUUUUUUUUUU GD86531 UUUUUUUUUUUUUUUUUUUUUUU F9641UUUUUUUUUUUUUUUUUUUUUUUUU D61UUUUUUUUUUUUUUUUUUUUUUUUUUU #UUUUUUUUUUUUUUUUUUUUUUUUUUUUU B60UUUUUUUUUUUUUUUUUUUUUUUUUUU A8630UUUUUUUUUUUUUUUUUUUUUUUUU A976410UUUUUUUUUUUUUUUUUUUUUUU A9876420UUUUUUUUUUUUUUUUUUUUUUU A987664300UUUUUUUUUUUUUUUUUUUUU AA9876653200UUUUUUUUUUUUUUUUUUU BA988766543200UUUUUUUUUUUUUUUUU CCA987765432100UUUUUUUUUUUUUUU DDBA987655433210UUUUUUUUUUUUUUU FEDB9877655433210UUUUUUUUUUUUU GFCEA9876654432210UUUUUUUUUUUUU HGFDA9876654433210UUUUUUUUUUUUU IHGD987665443210UUUUUUUUUUUUUUU JIGEB98766544320UUUUUUUUUUUUUUU JIGDA98766543210UUUUUUUUUUUUUUU		

Figure 3-11. View Factors in a Torus ($P = 3.0$)

[illegible]

[illegible]

Figure 3-13. View Factors in a Torus ($P = 20.0$)

which heat is effectively dissipated is on the order of $1/P$. By comparison, for the sphere, heat was dissipated uniformly over the entire shell. Thus, some portion of the toroidal shell will become relatively hot, and re-radiate a significant amount of radiant energy back to the module undergoing the LOCA. Thus, one may conclude that the cooling of a module undergoing a LOCA will be more difficult in a torus than in a sphere, in accord with approximate factor $1/P$. However, the effect of α should be similar in the two cases.

3.4. References for Chapter 3

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4. SAFETY STUDY OF PRIMARY COOLANTS IN EXPERIMENTAL POWER REACTOR FIRST WALL DESIGN

4.1. Introduction

The possibility of overheating and melting of the first wall and blanket in fusion power systems is an important safety concern, since they contain highly toxic radioactive materials which, if released, may endanger the public. Although the experimental power reactors (EPR's) are still in their preliminary design stage, an assessment of the initiating accidents and an analysis of the phenomenology involved would provide designers with a set of design constraints for minimizing the probability of occurrence of accidents and maximizing their confinement.

Since there are several conceptual designs for the EPR, and the designs evolve from day to day, the present safety study attempts to be as generic as possible. Accidents relating to the malfunction of the primary coolant system are the main concern of this chapter. These include the loss-of-coolant flow accident (LOFA), the loss-of-heat removal accident, the transient overpower accident (TOP), and the loss-of-coolant accident (LOCA). While helium appears to have the greatest long-range potential as a fusion reactor coolant, the disadvantages of helium from the point of view of large coolant-channel volume fraction and high pumping power requirements may force the designer to search for other coolant materials. In general, the coolant candidates can be classified into four groups: (a) the liquid metal, (b) the molten salt, (c) the inert gas, and (d) the non-metallic liquid. The thermophysical properties for these four classes of coolant are given in Table 4-1. Each fluid offers certain advantages and disadvantages in pumping power requirement, materials compatibility, and tritium contamination. These

Table 4-1. Thermophysical Properties of Candidate Coolant Materials.

Coolant	Example	Reference Temperature or Pressure (K or MPa)	Density (Kg/m ³)	Specific Heat (J/Kg·K)	Thermal Conductivity (W/m·K)	Viscosity (N·s/m ²)	Melting Point (K)	Boiling Point (K)	Prandtl Number
Liquid Metal	Li	477 K	506	4,187	38.1	5.9 E-4	452	1,591	0.065
Molten Salt	LiBe ₂ F ₄ (Flibe)	873 K	1,940	2,400	0.8	8 E-3	636	---	24
Gas	He	5 MPa	3	5,204	0.1731	3.8 E-5	---	---	0.72
Non-Metallic Liquid	Pressurized Water	6.4 MPa	1,000	5,307	0.574	9.8 E-5	0	280	0.902

aspects are summarized in Table 4-2.

In this chapter, the safety aspects of these coolant candidates in the event of the generic accidents, such as LOFA's, LOCA's, and so forth, are fully explored. General methodologies for analyzing the phenomena are established. In view of the preliminary nature of the designs, the analytical model attempts to be simple, and the results are presented in parametric form so that the designer can easily utilize the present results for future design modification.

4.2. First Wall/Blanket Design for EPR's

In general, the blanket/first wall system (or the so-called primary energy conversion system) of an EPR serves the major purpose of the generation and removal of approximately 100 to 300 MW of sensible heat, and of providing adequate shielding for the magnet.

In the Argonne National Laboratory (ANL) design [4-1], the first wall includes the vacuum vessel, which is made of 1 cm thick stainless steel. A layer of graphite and silicon carbide 1 to 4 mm thick is coated on the inside wall of the vacuum vessel, based on an early version of the design; the later version uses beryllium. The ANL design has considered two alternative methods of first wall cooling: (i) pressurized water at about 13.8 MPa and (ii) pressurized helium at about 5 MPa. The thermal fluid analysis parameters for the ANL EPR preliminary blanket design are shown in Table 4-3.

The General Atomic (GA) first wall design [4-2] is composed of circular silicon carbide plates, which are mounted to the blanket modules and cooled by radiation to the blanket. The liner plates are 42 cm in diameter and 1 cm thick. Behind the first wall is the blanket region,

Table 4-2. Advantages and Disadvantages of Selected Coolant Options for EPR's.

Coolant Material	Principal Advantages	Principal Disadvantages
Liquid Lithium	Excellent heat transfer characteristics Minimum interference with breeding Simple interface with liquid Li blanket	Subject to large MHD effects Potentially large mass transfer rates Agressive, flammable medium Requires refractory metal structure
Molten Salt (e.g., LiF-BeF ₂ LiCl-KCl NaBF ₄ etc.)	Good heat transfer characteristics Tolerable MHD effects More benign than Li Good prospects for tritium removal and containment Simple interface with a molten salt blanket	Probably restricted to use at temperatures above 673 K Subject to MHD-induced electrolysis Radiolysis could enhance corrosion Undesirable effects of constituent elements on neutronics
Helium	Most benign medium No MHD effects Excellent prospects for tritium containment Little interference with breeding Substantial technology base exists	May require substantial pumping power Difficult to interface with liquid Li blanket Requires relatively large-area heat transfer interface
Pressurized Water	Good heat transfer characteristics Low void fraction for channeling Low pressure drop in blanket Neutronically compatible Substantial technology base exists	Difficult to interface with liquid Li blanket Limited overall thermal to electrical efficiency Potential source of large protium backflux

Table 4-3. Thermal Fluid Analysis Parameters for the ANL EPR Preliminary Blanket Design.

Parameter	Coolant	
	Water ⁽¹⁾	Helium ⁽²⁾
Thermal Power Removed ⁽³⁾ (MW)	186 ⁽⁴⁾	186 ⁽⁴⁾
Coolant Inlet Pressure ⁽³⁾ (MPa)	13.8	50
Pressure Drop in Blanket ⁽⁵⁾ (MPa)	< 0.1	~0.1
Pumping Power for Blanket (MW)	< 1	~7
Coolant Inlet Temperature ⁽³⁾ (K)	311	630
Coolant Exit Temperature ⁽³⁾ (K)	575	800
Maximum Blanket Temperature ⁽³⁾ (K)	873	873
Tube Wall Temperature ⁽³⁾ (K)	400	823
Number of Radial Zones ⁽³⁾	6	6
Coolant Velocity	(6)	(6)

- Notes:
- (1) Parameters consistent with PWR technology.
 - (2) Parameters consistent with HTGR technology.
 - (3) Parameters fixed in calculations
 - (4) Based on a conservative overestimate of the neutron wall loading of 0.3 MW/m^2 .
 - (5) Average pressure drop per tube.
 - (6) Incompressible flow was assumed; i.e., the coolant velocity was kept at $< 0.3 \times$ sonic velocity.

which is composed of steel blanket modules and graphite filler pieces. During the plasma burn, the silicon carbide liner receives a peak energy input of 72 MW, of which approximately two-thirds is essentially surface heat flux and the remainder internal heat generation. This energy is radiated to the blanket modules where it is conducted through the metal and transferred by convection to the helium coolant, which flows in holes along the blanket wall. In addition to the radiant energy from the liner, there is an internal heat generation rate of about 240 MW in the blanket modules.

In the early version of the Oak Ridge design [4-3], two separate coolant loops (helium and water) were proposed for the first wall coolant. In the later version, the blanket design was modified. Basically, the blanket is composed of 60 segments, each of which is divided into 3 compartments. The first compartment is the main absorber, filled with potassium. All 60 of these compartments are identical except for the substitution of lithium for potassium in the experimental breeding segments. The second compartment is the reflector. A fraction of the relatively high-energy neutrons which have passed through the liquid metal absorber is reflected back into the metal. This is not important for the non-breeding modules, but it is an essential feature of the tritium-breeding test module. The third blanket compartment is a gamma shield composed of three concentric cylinders of stainless steel. In the annuli between these cylinders is liquid metal. The purpose of the liquid metal is to conduct and convect the heat from the stainless steel cylinders to the cooling tubes. Liquid potassium or lithium was selected as the heat transfer medium because the heat from the stainless steel gamma absorber is easily conducted through the liquid metal to the

coolant tubes.

The first liquid metal compartment has two rows of coolant tubes. Helium at a pressure of 7.1 MPa is circulated through these tubes to remove the heat. One row of tubes is adjacent to the inner first wall but not attached to it. Heat from the inner wall is transferred by conduction to the helium coolant tubes. The second row of tubes is adjacent to the rear wall of the liquid metal compartment but not attached to it. This placement is for the purpose of removing the heat from the liquid metal, together with the heat that is radiated from the graphite reflector located outside the blanket.

The various EPR blanket designs are summarized in Table 4-4.

The UWMAK-III reactor [4-5] is designed to generate 5,000 MW(th), with a blanket and shield design as shown in Figure 4-1. The inner blanket and shield closest to the torus centerline is not designed to breed tritium, but only to extract heat and to protect the superconducting toroidal field magnet. As such, it really constitutes a shield which is divided into hot and cold zones. More than 99% of the energy incident on the inner shield is deposited in the hot zone composed of structure (TZM), boron carbide (B_4C), and helium coolant. The cold inner shield uses Type 316 stainless steel as the structure. In front of the first wall of the inner shield is a 25 cm thick graphite zone which is passive (not actively cooled) and which serves to soften the neutron spectrum incident on the first structural wall. This, in turn, reduces the radiation damage to the structure, and the inner hot shield is designed for the plant lifetime.

The outer blanket is cooled with liquid lithium, which also acts as the breeding material. The overall breeding ratio is 1.25, based on

Table 4-4. Comparisons of Various EPR Designs and UWMAK-III.

	ANL	ORNL	GA	UWMAK-III
Power (MW(th))	186	356 (400)*	303	5,000
Power (MW(e))	56	86	124	1,985
Average Core Power Density (MW/m ³)	2.1	2.5	3.3	1.72
Maximum Core Power Density (MW/m ³)	5.3	10	9.7	--
Surface Heat Flux (MW/m ²)	0.13	0.27 (0.40)*	0.23	3.1
Neutron Fluence (n/cm ² /year)	6×10^{25}	6×10^{22}	3.8×10^{16}	4.2×10^{27}
First Wall and Structure	Detachable stainless steel coolant panel coated with beryllium micro-layer	Stainless steel first wall	Silicon carbide plate liner Stainless steel and carbon blanket	Molybdenum and stainless steel
Primary Coolant	Pressurized water or lithium (Stage II)	Combined water and helium*	Helium	Lithium and helium
Containment Design	Cylindrical 73.2 m dia. 50 m height	---	---	---
Breeding Material	None in Stage I design Liquid Li (10 cm thick) or Li compound in Stage II design	Lithium	None	Lithium
Diverter	None	None	None	TZM foil over Li Frozen Li seal Replace every 19 hr
First Wall Radius (m)	2.4	--	--	--
First Wall Area (m ²)	592	--	493	
Blanket Thickness (cm)	30	51.5	42	--
Blanket Temperature (K)	873	823	873 (first wall) 1,723 (liner)	889-1,271
Coolant Pressure (MPa)	13.7	7.1	4.8	1.3 (Li) 6.9 (He)
Coolant Flow Rate (Kg/s)	121	658	234	1,922 (Li) 675 (He)
Coolant Pumping Power (MW)	1 (water) 7 (He)	--	2.8	2 (Li loop)

*Reference design [4-4]

Figure 4-1. Schematic of the Blanket and Shield Design for UWMAK-III.

toroidal-geometry neutron transport calculations. The structural material in the outer blanket is not protected neutronically and will therefore have to be replaced periodically. The wall lifetime is estimated to be about 2 years. However, space is plentiful on the outer portion of a torus, so that the difficulties involved in blanket module replacement are minimized.

The UWMAK-III design uses a closed power cycle with helium gas turbines. The coolant from the inner blanket goes directly to a helium turbine, while the lithium from the outer blanket goes to an intermediate Li-Na heat exchanger, with the sodium going to a Na-He heat exchanger from which the helium goes to drive the turbines. The sodium in the intermediate loop plays a dual role, isolating the tritium-containing lithium from the power cycle and providing a working fluid for thermal energy storage which can be used during the down part of a Tokamak plasma cycle to keep a continuous flow of hot helium going to the turbines. The diverter is cooled by means of a sodium loop which is coupled with a steam cycle for heat removal. The UWMAK-III design is summarized with the EPR designs in Table 4-4.

4.3. Safety Concerns with Various EPR Blankets

This section outlines several unique safety features that are associated with the current EPR designs. The more generic safety problems are analyzed in the next section.

In the ANL design, the panel is under extreme thermal cyclic loads of 273 K peak-to-peak value at 1 cycle/minute, coupled with the pressure force from the pressurized water coolant. Steel swelling and fatigue failure appear to be potential causes of first wall failure. In addition,

the water coolant is contaminated with radioactive material. A loss-of-coolant accident resulting from pipe rupture would lead to the release of radioactivity to the containment. Further heat-up of the blanket steel would lead to metal-water reactions, where hydrogen could be produced. Hydrogen-burning or detonation would impose a potential threat to the containment structures.

The first wall design is rather complicated for the GA-EPR, and the heat-removal process relies on the combined modes of thermal radiation, forced convection, and conduction. Local melting is more likely, as the first wall is immediately heated by thermal radiation.

In the early version of the combined water- and helium-cooled first wall in the ORNL design, and in the diverter of the UWMAK-III design, the presence of lithium or potassium in the blanket would impose a potential threat of direct contact between the liquid metal and the water. This contact would cause a violent chemical reaction [4-5]. In the later version of the ANL design, as well as the other EPR's and UWMAK-III, depressurization or loss of helium flow would impose safety problems on the first wall and the containment.

4.4. Accident Analyses

The safety limit of a fusion reactor is usually focused on the mobility of the activated first wall materials and the release of tritium. It is usually assumed that any malfunction in the coolant system would lead to plasma quench, which terminates the fusion reaction by means of the heat flux into the first wall. Long-term afterheat effects may necessitate the design of an emergency cooling system. Although the quenching phenomenon is not fully understood at this time (some speculate that

rapid quenching might induce large electromagnetic forces on the first wall), it is always important to know the amount of time available for protective action before the first wall reaches the stage of no recovery. These time parameters include the quench time for plasma and the time required to actuate the auxiliary heat removal systems.

Four classes of accidents are considered. A loss-of-coolant flow accident (LOFA) is usually initiated by pump failure. As the flow coasts down, the heat removal rate from the first wall is degraded. First wall heat-up is a major concern of this class of accident. A loss of heat removal transient may be initiated by a loss of feedwater or a steam break on the secondary side of the heat exchanger. The coolant will be heated up along with the first wall. Pressure built up in the primary coolant system may fail the first wall prior to its melting. A transient overpower accident (TOP) is usually caused by plasma instability. The transient is usually short-lived, and the effect is mainly on the heat-up of the first wall. A loss-of-coolant accident (LOCA) is initiated by pipe rupture, where the primary coolant is leaked to the containment. Depending on the initial coolant pressure and thermal conditions, rapid coolant voiding is a possible event. Both first wall heat-up and containment pressurization are major concerns. A LOCA is usually considered as the most serious accident, and it imposes great demands on the operational mode of the auxiliary heat removal system to mitigate the consequences.

Because of the tight space requirement on the first wall, most designs tend to use a combination of two different coolant systems for the first wall and the blanket cooling. For example, ORNL [4-2] proposes to use separated pressurized water and helium loops for their diverter

cooling. These dual coolant loops complicate the mechanical design of the first wall, and in some cases the coolants may not be compatibly safe. For example, sodium or lithium reacts exothermically with water, generating a pressure pulse [4-6]. Highly-pressurized water in the neighborhood of low-pressure helium will impose a difficult problem for the structural design. Some of the safety concerns of these dual cooling systems will be addressed more thoroughly in the future. At present, we focus our attention on single component coolant systems.

4.4.1. Loss-of-Flow Accidents

In the event of pump or circulator failure, or the loss of electric power, the coolant adjacent to the first wall coasts down. The determination of the coolant flow as a function of time is the first step in the analysis. By equating the retarding force (frictional or magneto-hydrodynamic) in the loop to the change in momentum of the fluid,

$$\frac{L_c \rho_c}{g_c} \frac{du_c}{dt} = - C_{fc} \frac{\rho_c u_c^2}{2g_c}, \quad (4-1)$$

where

L_c = the total length of the loop,

t = time,

C_{fc} = the total pressure loss coefficient of the loop,

u_c = effective flow velocity, and

ρ_c = average fluid density,

Equation (4-1) together with the initial condition $u = u_0$ at $t = 0$ yields:

$$\frac{u_c}{u_0} = \frac{1}{Ku_0 t + 1}, \quad (4-2)$$

where

$$K = C_{fc}/2L_c.$$

The temperature transient at the first wall is given by

$$\rho_f C_{pf} d_f \frac{dT_f}{dt} = q_f''' d_f - h(T_f - T_c), \quad (4-3)$$

where

ρ_f = the density of the first wall material,

C_{pf} = the heat capacity of the first wall,

d_f = the thickness of the first wall,

T_f = first wall temperature,

q_f''' = the volumetric heating rate of the first wall,

T_c = coolant temperature, and

h = the heat transfer coefficient.

During coastdown, h is related to the coastdown velocity by

$$\frac{h}{h_o} = \left(\frac{u_c}{u_o} \right)^{0.8}$$

if the flow is assumed to be turbulent. The solution for Equation (4-3) in the case of a constant T_c (which is fairly close to the actual situation if the natural circulation can deposit the heat to the heat exchanger) is

$$T_f - T_c = \frac{1}{\rho_f C_{pf}} e^{-R(t)} \int_0^t q_f''' e^{R(t)} dt + (\Delta T)_o, \quad (4-4)$$

where

$$R(t) = \frac{5h_o}{\rho_f C_{pf} d_f Ku_o} (Ku_o t + 1)^{0.2}, \text{ and} \quad (4-5)$$

$$(\Delta T)_o = T_{fo} - T_{co} = \text{the initial temperature difference.} \quad (4-6)$$

It is observed that the temperature transient is a function of three

parametric groups:

(i) $\frac{q'''}{\rho_f C_{pf}}$, which characterizes the first wall heat flux,

(ii) $\frac{h_o}{\rho_f C_{pf} d_f}$, which characterizes the heat transfer, and

(iii) Ku_o , which characterizes the fluid dynamic.

Typical values of these three parameters for the liquid-cooled and the gas-cooled fusion reactor designs are shown in Table 4-5. Some of the parameters are based on the current technologies of LWR's, HTGR, and large-scale fusion power designs. Although future EPR's may have different numerical values for these parameters, a careful examination of the parameter $R(t)$ would shed some light on the temperature response of the first wall with different coolants.

Liquid lithium has the highest value of Ku_o , because of the pressure drop due to MHD effects. For the systems analyzed, the value of Ku_o for lithium is two orders of magnitude greater than that for helium and three orders of magnitude higher than that for water.

The numerical values of $R(t)$, as shown in Table 4-5, were calculated for a stainless steel first wall with a thickness of 1 cm. It should be noted that although Ku_o is quite different for gas-cooled and liquid-cooled blankets, the difference in $R(t)$ is much less dominant.

It is also found that the parameter Ku_o has a minor effect on the first wall temperature transient. For example, a decrease of three orders of magnitude in Ku_o only reduces the melting time (the time between the accident initiation and the melting of the first wall) by 15%. In fact, the melting time is more sensitive to the first wall heat flux

Table 4-5. Characteristic Parameters for Loss-of-Flow Accidents.

Parameter	Water-Cooled	Gas-Cooled	Liquid Metal-Cooled
Velocity, u_0 (m/s)	4.8	80	0.035
Pressure Drop, ΔP (MPa)	0.1	0.2	2.7
Ku_0 (s^{-1})	0.21	15.36	2,204
Flow Hydraulic Diameter, D (cm)	1	2.5	1
$(Re)^{0.8}$ (1)	3.56×10^4	1.44×10^4	96
$(Pr)^{0.4}$ (2)	0.96	0.877	0.333
Nu (3)	786	290	0.735
h_0 (4) (W/cm^2)	4.5	0.2	0.28
$R(t)$	$\sim 15t^{0.2}$	$\sim 0.03t^{0.2}$	$\sim 0.0078t^{0.2}$

Notes: (1) Re = Reynolds number.

(2) Pr = Prandtl number.

(3) Nu = Nusselt number.

(4) h_0 = the steady-state heat transfer coefficient.

than to the fluid coastdown parameter Ku_0 (see Table 4-6).

Table 4-6. Comparisons between Lithium-Cooled and Helium-Cooled Blanket Responses under Loss-of-Flow Conditions.

Parameter	Coolant	
	Lithium	Helium
Steady-State Pressure Drop (MPa)	2.76	2.1
Flow Coast-Down Parameter, Ku_0 (s^{-1})	2,204	15.36
Time for First Wall Temperature to Reach the Melting Point for Different First Wall Loadings (s):		
$q = 36 \text{ W/cm}^2$	25	26
$q = 18 \text{ W/cm}^2$	50	52
$q = 3.6 \text{ W/cm}^2$	253	259

4.4.2. Loss of Heat Removal

A sudden reduction of the secondary flow in the heat exchanger causes a reduction in the heat removal capability of the primary coolant. The heat addition and removal rates at the first wall remain balanced, initially. As the temperature of the primary coolant rises, the first wall temperature transient is given by:

$$\rho_f C_{pf} d_f \frac{dT_f}{dt} + h(T_f - T_c) = q''' d_f, \quad (4-7)$$

where the transient coolant temperature, T_c , is given by

$$T_c = (T_i + T_o)/2. \quad (4-8)$$

T_i is the inlet temperature; it is assumed to be equal to the outlet.

temperature, T_o , with a time shift of t_1 due to the elapse time, t_1 , of the flow circuit. Hence,

$$T_i = T_o(t - t_1) \quad \text{for} \quad t - t_1 > 0 \quad (4-9)$$

and

$$T_i = T_o(0) \quad \text{for} \quad t - t_1 \leq 0. \quad (4-10)$$

An energy balance for the flow channel yields

$$T_o(t)(\rho_c C_{pc} u + \frac{hA}{2A_c}) = T_o(t - t_1)(\rho_c C_{pc} u - \frac{hA}{2A_c}) + \frac{hA}{A_c} T_f, \quad (4-11)$$

where

t_1 = the elapse time of the coolant loop,

A = the first wall heat transfer area,

A_c = the flow area,

$T_o(t - t_1)$ = the outlet temperature at time = $(t - t_1)$, and

$T_o(t)$ = the outlet temperature at time = t .

It is found that the first wall temperature transient depends to a large extent on the elapse time, t_1 , of the coolant loop. Typical values of this elapse time for the lithium blanket are two orders of magnitude higher than those for helium blankets. Hence, the helium-cooled blanket will experience a more rapid temperature rise, as shown in Table 4-7.

Table 4-7. Comparison between Lithium-Cooled and Helium-Cooled Blanket Responses under Loss of Heat Removal Conditions.

Parameter	Lithium Coolant	Helium Coolant
Elapse Time (s)	~250	~0.8
First Wall Melting Time (s)	Prolonged	19

4.4.3. Overpower Transient

A sudden increase in the first wall heat flux, as a result of plasma instability, would cause a rise in temperature. For an increase of x% of heat flux on the first wall, the temperature rise is given by

$$(\Delta T) = (\Delta T)_0 (1 + x/100), \quad (4-12)$$

where $(\Delta T)_0 = T_{fo} - T_{co}$ is the initial steady state temperature difference and $(\Delta T) = T_f - T_c$ is the new temperature jump as a result of the increase in heat flux. Hence, the first wall thermal response in the event of an overpower transient is quite independent of the coolant system.

4.4.4. Loss-of-Coolant Accident

This section examines potential safety problems associated with a loss-of-coolant accident (LOCA) in lithium-cooled and gas-cooled fusion reactor blankets. A general approach is developed to identify the major physical neutronic factors determining the course and ultimate disposition of the accident sequence. Although the present analysis is performed for the UWMAK-I and UWMAK-II designs [4-7 to 4-9], the results are generic for EPR and UWMAK-III consideration.

In a liquid-metal cooled reactor, the drainage of lithium creates a void in the blanket which affects the neutron flux and temperature distribution. In non-metallic liquid-cooled or gas-cooled reactors, although the loss of coolant would not affect the neutron flux distribution, it could create overheating problems in other regions of the blanket before the failure of the first wall. In addition, the presence of helium or steam in the containment following a LOCA would pressurize the building, the effects of which must be considered.

During the study of a LOCA, three major questions are of primary concern:

1. How long would it take for the coolant to drain out of the blanket after the LOCA begins?
2. Given that the coolant did drain out, is there enough energy to melt the magnet shield?
3. If the lithium next to the first wall drained out, how long would it take for the first wall to fail?

Liquid-Metal Cooled Blanket. First, let us consider the liquid-metal cooled blanket. The analysis of a LOCA is divided into three parts: hydrodynamics, neutronics, and heat transfer. Under the influence of the magnetic and gravitational fields, as shown in Figure 4-2, the magnetic force tries to retard the flow while the gravitational force tries to accelerate it. At steady state, when these forces are balancing each other, the discharge rate of lithium through the break is given by

$$u_m = \frac{\rho g a^2}{k} / \left(H + H^2 \frac{c}{1 + c} \right), \quad (4-13)$$

where

ρ = lithium density,

a = the radius of the pipe,

$$c = \frac{\sigma_w t_w}{\sigma a},$$

σ_w = the electrical conductivity of the wall,

t_w = wall thickness,

σ = the electrical conductivity of the fluid,

a = pipe radius,

H = Hartmann number, and

k = a correction factor for circular pipe = 1.2.

Figure 4-2. Physical Model of Lithium Flow under the Influence of Magnetic and Gravitational Fields.

Using the parameters of UWMAK-I in Equation (4-13), the lithium surface is found to recede at an approximate rate of 3 mm/s. At this rate, the time required for all of the liquid lithium to drain out of the outer zone is roughly one hour. As the lithium drains out of the blanket, the spatial heating rate will change, since fewer neutrons will be attenuated. It would appear that this would be ample time to allow shutdown of the reactor; however, if the lithium were able to drain out faster, for example in the case of the loss of the magnetic field, the consequences of the increased spatial heating in the blanket and shield would be important.

In order to calculate the thermal response of the blanket and shield to a blanket LOCA, the spatial heating rates must be determined. The spatial heating rate due to high-energy neutron reactions with the nuclei of different materials in the blanket, shield, and magnet regions can be divided into two parts: (a) the neutron reactions themselves, and (b) absorption of secondary gamma radiation produced by the neutron reactions. Heating by neutrons at any point can be expressed as

$$H(r) = \int_0^{\infty} \phi(\vec{r}, E) \sum_j N_j(r) \sigma_{ij}(E) E_{ij}(E) dE \quad (4-14)$$

in $\text{MeV}/\text{cm}^3 \cdot \text{s}$

$\phi(\vec{r}, E)$ = the neutron flux at spatial point r and energy E ,

$N_j(\vec{r})$ = the number density of element j at point r (atom/cm^3),

$\sigma_{ij}(E)$ = the microscopic cross sections of element j , and

$E_{ij}(E)$ = the energy released and deposited locally in element j per reaction i induced by a neutron of energy E .

Equation (4-14) can be written as

$$H(\vec{r}) = \int_0^{\infty} \phi(\vec{r}, E) \sum_{ji} N_j(\vec{r}) K_{ij}(E) dE, \quad (4-15)$$

where

$$K_{ij} = \sigma_{ij}(E) E_{ij}(E), \text{ and} \quad (4-16)$$

$$K_j = \sum_i K_{ij}(E). \quad (4-17)$$

The K_{ij} 's and K_j 's are flux- and energy-independent. K_{ij} is the microscopic kerma (kinetic energy relaxed in materials) factor for reaction i in element j . E_{ij} is the sum of the kinetic energies of the recoil nuclei and charged particles emitted.

The gamma kerma factors are given by

$$K_Y^j = \sigma_{pe}^j + \sigma_{pp}^j (E - 1.02) + \sigma_{ca}^j(E), \quad (4-18)$$

where

K_Y^j = the gamma kerma factor for element j (MeV·cm²/atom),

E = photon energy (MeV),

σ_{pe}^j = the photoelectric microscopic cross-section for element j ,

σ_{ca}^m = the Compton microscopic absorption cross-section for element j ,

σ_{pp}^j = the pair-production microscopic cross-section for element j , and

$$H(\vec{r}) = \int_0^{\infty} \phi(\vec{r}, E) \sum_{cj} N_j(r) K_Y^j(E) dE. \quad (4-19)$$

The gamma kerma factors were generated numerically by a code called SMUG

[4-10]. A code named MACK [4-11] was used on conjunction with the neutron cross-sections obtained from the Engineering Nuclear Data File/B [4-12] to calculate the neutron kerma factors. Examination of Equations (4-15) and (4-18) shows that the nuclear heating is a function of the nuclide densities, neutron and gamma flux spectra, and the neutron and gamma kerma factors.

To determine how the heating rate varies spatially, it is necessary to solve the flux spectrum in the blanket and shield. To do this, a one-dimensional discrete-ordinate neutron and/or gamma transport code called ANISN [4-13] was used. Sixty-six group cross-sections were calculated using a weighting function which took into account the 14 MeV fusion neutron peak by utilizing a code called SUPERTOG [4-12]. It was felt that this many groups were necessary because the energy of the neutrons ranged from 14 MeV to the 1 MeV level.

The results for three different LOCA situations in the UWMAK-I design are shown in Figure 4-3. It should be noted that in the absence of lithium, the heating rate increases by approximately an order of magnitude in portions of the blanket and shield. In applying the heat conduction model to the different blanket and shield regions, the adiabatic temperature rise rates were found for the various zones being voided. Near the front edge of the shield, the melting temperature is attained in approximately 20 minutes. Also, even though rapid shutdown may slow down the rise in temperature, the afterheat remains a potential threat to the blanket's integrity.

From this analysis, it appears that in the event of a LOCA in a liquid-metal cooled reactor, there is ample time to scram the system, but if voiding does occur, the shield may reach the melting point within

Figure 4-3. Spatial Heating Rate in Blanket and Shield.

minutes if it is not scrambled.

Gas-Cooled Blanket. From the point of view of the first wall temperature response, a LOCA for a helium-cooled blanket is essentially the same as a LOFA, since both will lead to the loss of the blanket's cooling ability. Also, since helium has little effect on the neutronics of the blanket system, the spatial heating will be the same in a LOFA or a LOCA. After a LOCA begins in the reactor, the first wall, blanket, and shield will lose their cooling; however, they will continue to be heated from neutron and gamma heating, and from afterheat when the plasma is quenched.

In estimating the rate of depressurization of the helium gas, the coolant loop is modeled as a single volume, V . The thermodynamical properties are assumed to be uniform within the volume. The mass and energy of the helium during the transient are given by

$$\frac{dM}{dt} = -\dot{m}A \quad (4-20)$$

and

$$\frac{dU}{dt} = -\dot{m}Ah + Q_{in}, \quad (4-21)$$

where

M = helium mass = ρV ,

U = internal energy = $MC_v T$,

h = enthalpy = $C_p T$,

ρ = density,

Q_{in} = the energy transferred from the first wall to the coolant,
minus the heat transferred to the steam generator,

C_v = the specific heat at constant volume,

C_p = the specific heat at constant pressure,

T = temperature,

$$\dot{m} = \text{mass flow rate} = k \left[\left(\frac{2}{k+1} \right)^{\frac{k+1}{k-1}} P_0 \right]^{\frac{1}{2}}, \quad (4-22)$$

k = the isentropic gas constant = C_p/C_v , and

P = pressure.

For an adiabatic depressurization, Q_{in} is zero. Equations (4-20) and (4-21), together with the ideal gas law, $P/\rho = RT$, give

$$\rho^* = \frac{n^n}{(n + t^*)^n} \quad (4-23)$$

and

$$P^* = \rho^{*k}, \quad (4-24)$$

where

$$\rho^* = \rho/\rho_0,$$

ρ_0 = the initial helium density,

$$t^* = t/t_0,$$

$$t_0 \equiv \frac{1}{C} \left(\frac{\rho_0}{P_0 g} \right)^{\frac{1}{2}},$$

$$C = \frac{A}{V} \left[k \left(\frac{2}{k+1} \right)^{\frac{k+1}{k-1}} \right]^{\frac{1}{2}},$$

$$n = 2/(k - 1),$$

$$P^* = P/P_0, \text{ and}$$

P_0 = the initial system pressure.

In accordance with the closed-form solution, when the heat transfer to the coolant is neglected, the depressurization process is characterized by a single parameter,

$$t^* = \frac{t}{t_0} = \frac{V}{A} \left[k \left(\frac{2}{k+1} \right)^{\frac{k+1}{k-1}} \left(\frac{\rho_0}{P_0 g} \right) \right]^{\frac{1}{2}} t.$$

Physically, this means that the depressurization is a function of three groups of parameters:

(i) The break area-to-volume ratio,

$$A/V;$$

(ii) The initial density-to-pressure ratio,

$$\rho_0/P_0; \text{ and}$$

(iii) The isentropic gas constant, k .

For the given system, $k = 1.67$ for helium, $P_0 = 20 \text{ MPa}$, $T_0 = 273 \text{ K}$, $V = 962 \text{ m}^3$, and for a break area of 0.046 m^2 , the time required for the pressure to be halved is 10 seconds.

On the other hand, if the steam generator effectively maintains the coolant at a constant temperature, then the depressurization process can be approximated to be an isothermal process, and the depressurization rate is slower (see Figure 4-4).

As the system is depressurized, the first wall starts to heat up.

An energy balance on the first wall would yield

$$C_w \frac{dT_w}{dt} = q_w'' - (T_w - T_c) h_w, \quad (4-25)$$

where

T_w = first wall temperature,

T_c = coolant temperature,

q_w'' = the first wall heat flux,

$C_w = \rho_w C_{pw}$,

ρ_w = the density of the first wall,

Figure 4-4. Transient Blanket Pressure and Density during a LOCA.

C_{pw} = the heat capacity of the first wall, and

h_w = the first wall heat transfer coefficient.

Based on a heat flux of 30 W/cm^2 , initial wall temperature of 873 K, C_w of $1.1 \text{ Cal/cm}^3 \cdot \text{K}$, and melting temperature of 1,811 K, an adiabatic heat-up calculation indicates that it takes about 2 minutes for the first wall to reach the melting temperature. A comparison between the gas-cooled and the liquid-metal cooled blankets for a LOCA is shown in Table 4-8. It can be seen that if the magnet field is maintained throughout

Table 4-8. Comparison between Lithium-Cooled and Helium-Cooled Blanket Responses under Loss of-Coolant Conditions.

Loss of Coolant	Coolant	
	Lithium	Helium
Leakage Velocity (m/s)	$\sim 10 \text{ E-3}$	$\sim 1,600$
Time for Auxiliary Cooling (s)	1,020	120

the accident transient, the time available for auxiliary cooling in lithium-cooled reactors is longer than in helium-cooled reactors.

Containment Pressurization. Unlike a liquid-metal cooled Tokamak, if a LOCA occurs in a helium gas-cooled system, the possibility of the containment becoming over-pressurized must be examined. In the calculation it is postulated that the high pressure helium (50 MPa) is released at high temperature (911 K) into the containment. It is assumed that all of the helium is lost and is dumped adiabatically into the free air containment.

For the initial conditions given in Table 4-9,

Table 4-9. Loss-of-Coolant Accident Initial Conditions.

Parameter	Volume (m ³)	Temperature (K)	Pressure (MPa)
Free Air in Containment	2.04×10^5	293	1.0
Helium Inventory	1.36×10^4	772	50

it is found that the final temperature is 523 K and the final pressure is 3.12 MPa (3.08 atm) inside the containment, due to the compression of the helium gas. This pressure is substantial, and the containment building must be able to withstand it. Also, this calculation was performed for a specific design, and the pressure and/or temperature may be higher in a different design.

4.5. Other Safety Problems

So far, we have concentrated our efforts on the first wall and the blanket. The following safety problems should also be considered.

4.5.1. Shielding

The shielding in an EPR acts as a thermal barrier between the high temperature blanket and the cryogenic toroidal field coil. In the ORNL design, the shield is located outside the blanket; it is composed of stainless steel and lead. The shield is cooled by borated water. In the ANL design, aluminum is used as well as lead and steel. In the GA design, the shielding is divided into two regions. The inboard shield consists of 50% stainless steel and 50% lithium hydride. The outer shield is 70% lead and 30% boron carbide. The shield is cooled by borated water.

A loss of coolant in the shielding may cause the shield to overheat.

The loss of shielding may cause magnet overheat and increase the neutron fluence onto the magnet. There is a possibility that the magnet would go normal.

4.5.2. Cryogenic Liquid Helium Systems

All three EPR designs utilize liquid helium to maintain the superconductivity of the magnet. ANL uses a boiling helium pool. The loss of the cryogenic fluid would impose two effects: (i) the magnet would go normal, and (ii) the containment would be pressurized by the flashing of the liquid helium.

4.5.3. Containment Designs

The three designs have not considered the details of the containment system. The containment should be designed against over-pressurization as a result of the loss of primary coolant (helium or pressurized water), the loss of cryogenic fluid, the loss of lithium (i.e. the prevention of a lithium fire), and/or against the missiles that could be generated by abnormal magnet forces.

4.6. Summary and Conclusions

This section examines the potential safety problems that are associated with the various primary coolant candidates currently considered for the EPR fusion blanket designs. The basic concern is the possibility of overheating and melting of the first wall and the blanket, induced by a malfunction in the primary coolant system. These accidents include the loss of coolant flow, the loss of heat removal, overpower transients, and the loss of coolant. While helium appears to have greater long-range potential as a fusion reactor coolant, the disadvantages of helium from the point of view of large coolant-channel volume fraction and high pumping power requirements may force the designer to search for other

coolant materials.

In general, the coolant candidates for fusion reactors can be classified into four groups: (1) liquid metal, (2) molten salt, (3) inert gas, and (4) non-metallic liquid. Each fluid offers certain advantages and disadvantages in pumping power requirements, materials compatibility, and tritium contamination. The thermophysical properties of these four classes of coolant are quite different, and hence their responses to the accident conditions also vary. Since the current most favored coolant candidates are lithium and helium, these two fluids have received full attention in this chapter.

In a LOFA, the coolant adjacent to the first wall coasts down. The slowdown of the flow is established by balancing the inertial force with the frictional force and the magnetohydrodynamic force. The decay of heat transfer as a result of the coastdown of flow causes the first wall temperature to rise. The temperature transient is found to be a function of a parametric group which combines the effects of flow coastdown and the loss of heat transfer. As shown in Table 4-6, liquid lithium suffers a high pressure drop and will experience a rapid flow coastdown, as reflected in the high value of the parameter Ku_0 . However, the time for the first wall to reach its melting point is not sensitive to Ku_0 . Hence, the responses for liquid-cooled and gas-cooled blankets under a loss-of-flow condition are very similar, provided that the blankets have the same steady-state heat transfer coefficient and surface heat flux.

The loss of heat removal capability in the heat exchanger would first cause the primary coolant temperature to rise. The first wall temperature response depends to a large extent on the elapse time of the coolant loop. The typical value of this elapse time for the lithium

blanket is two orders of magnitude higher than for the gas-cooled blanket. Hence, the helium-cooled blanket will experience a more rapid temperature rise. In addition, depending on the system design, the pressure surge in both systems can be substantial.

Since the steady-state heat transfer coefficients for both liquid- and gas-cooled blankets are almost the same, the first wall thermal responses in the event of an overpower transient are the same for both cases.

If the liquid-cooled blanket loses its coolant, both the neutronics and the magnetic fields could be disturbed. There is an order-of-magnitude increase in the heating rates of certain liquid-metal blanket regions from the steady-state value. Such a disturbance does not occur in the gas-cooled blanket. The leakage rates for these two coolants are also quite different. The times available for auxiliary cooling are also different for both systems.

Hence, the lithium- and helium-cooled blankets have different thermal responses under different accident conditions. This preliminary study shows that helium has a more adverse effect on the first wall heat-up in the event of a loss of heat removal or a loss of coolant.

4.7. References for Chapter 4

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5. MAGNET SAFETY

5.1. General

Within the set of various possibilities which lead to relevant accident sequences in Tokamak experimental power reactors (EPR's), the failure of the magnet system and the surrounding cryostat system may be a significant event. The magnet system stores a considerable amount of energy (see Table 5-1), in particular in the toroidal field (TF) coil assembly. The dewar may undergo serious disturbances such as loss of vacuum, accidental pressurization of the vacuum space, serious deformations, and damage resulting from the sudden application of magnetic field gradients. The latter may be the consequence of a collapse of the field in one or several magnet sections. Furthermore, the dewar may be damaged by arcs which burn holes locally, or by missiles originating from other components of the system (e.g. matter propelled by field gradients).

The magnet itself may be subjected to refrigerator failure. Therefore, standby refrigeration is required, which may store large amounts of energy. When pool cooling is specified by the designer, the large amounts of liquid He⁴ again constitute large energies.

Within the magnet, the quench from the superconducting to the normal state is the most significant event considered by the designer in his adoption of stability criteria (which are safety design criteria). The most significant steps are the sizing and adoption of various normal metal components, and the sizing and adoption of various filaments of type II materials. In the first process, the details of the coolant

Table 5-1. EPR Magnet Parameters.

Design	Nominal Specifications for Toroidal Field Coils					
	Energy (GW)	Number of Coils	Materials	Filament Diameter (microns)	Axial B_T (T)	B_{max} (T)
ANL	30	16	NbTi-Cu	--	4.47	--
ORNL	29	20	Nb ₃ Sn NbTi-Cu	40	4.9	11
GA	16.7	16	NbTi-Cu	200	3.93	7.86

system are the most important. Among various design options are the choice of the thermodynamic state of the coolant (He^4 in general), the choice of the mode of convection, and the choice of wetted area in the conductor-structure configuration. In all cases disturbances may show up which lead to accidents (e.g., loss-of-coolant, loss-of-flow, change of thermodynamic state, deformation of conductor, and deterioration of normal metal properties accidents). Aside from coolant-related issues, there are choices of inductances, the sectioning of magnet packages, the incorporation of structural support elements, the choice of dielectrics, and others. Again, a series of disturbing events may be pertinent. The structure may show fatigue or other failure caused by the forces imposed by the fields. The damage to one particular section of the magnet is not without consequences for the other sections unless adequate protection circuitry is available. The dielectric protection may be altered or damaged. In this context, the change of the thermodynamic state of the coolant may be quite significant when the density of the coolant is lowered. Long radiation exposure may cause significant changes in material properties.

Turning to individual system characteristics of EPR designs, we focus attention primarily upon stabilizer safety design, coolant choices, and related cryostat issues. Some pertinent EPR design choices are listed in Table 5-2.

5.2. Oak Ridge National Laboratory EPR

Several new concepts have been incorporated into the design of the Oak Ridge National Laboratory (ORNL) [5-1] magnet system. For some of them, operational experience appears to be absent or available only for small-size prototype sections. Examples are the long-range behavior of fiber-reinforced insulating matrices for structural members under neutron

irradiation. This point is also of concern in other designs. The most striking aspect is the large magnetic field value of $B_{\max} = 11$ T. This necessitates the use of Nb_3Sn in conjunction with NbTi (the latter for low field regions). It is noted that no large-scale magnet experience is available for the most advanced Nb_3Sn composites. These are certainly in the experimental category as far as use in an EPR is concerned, or for similar large magnets. The need for major engineering extrapolations is evident from the high field specification and other related important cryo-components. Conductor-coolant interactions are still unclear for severe quenches, since the coolant circuit is also a novel element of the design. Supercritical He IV has been chosen. This medium has a rather small heat transfer coefficient at small flow rates. Gravitational convection does not provide large density differences. Consequently, natural convection in a pool arrangement is, by far, not sufficient. For this reason forced convection has been specified. These particular conditions pose questions for the loss-of-flow accident (LOFA), in addition to trouble for loss-of-coolant accidents (LOCA's). Obviously, it is desirable to know about the necessity of coolant back-up systems similar to those of pressurized-water reactor coolant systems.

In summary, it looks as if this type of EPR is quite significantly removed from the frontiers of large magnet technology. Thus, it represents a venture which may be a long-range goal in terms of the original EPR time-table.

5.3. Argonne National Laboratory EPR

Several design features of the Argonne National Laboratory (ANL) [5-2] design are quite similar to the UWMAK designs. However, there are

some distinct differences in the way that the composite conductors are wound and cooled. Although some of the generic safety considerations apply as they do in other Tokamaks, including the UWMAK systems, it is noted that there has been a considerable recognition of the (at present) only approximately known stability in actual magnets. In this context the question of operation at 3 K deserves some attention.

Operation at 3 K in the pool boiling mode does not appear to present serious problems. It is recognized, though, that enhanced refrigeration requirements might be economically unattractive. The refrigeration costs of He⁴ (investment and operation) are, however, presumably only a minor part of the entire system cost. The advantage of 3 K for NbTi is the increase in critical magnetic field values and critical currents. Thus, design parameter changes appear to be less dramatic than the first impression obtained from a look at the increase in refrigeration power. There are some particular safety considerations which are also discussed in the subsequent overall comparisons.

Composite-coolant interactions at the initiation of quenching and during post-quench conditions may be quite complicated, as recognized by ANL. An increase in pool pressure from top to bottom of about 10 kPa leads at 3 K to a change in saturation temperature of about 0.2 K. This amount does not appear to be excessive, even when actual stratification possibilities are taken into account. The size of vapor bubbles is slightly enlarged, with respect to 4 K bubbles, when other parameters are kept constant and saturated liquid is used. In contrast, the change in specific volume from saturated liquid to saturated vapor is distinctly increased. This poses the question of the maximum tolerable quality during gravitationally induced motion of vapor in the pool and windings.

Obviously, not all of these factors are entirely resolved. A remedy is the use of subcooled He^4 , as for instance employed in the Westinghouse prototype a.c. synchronous generator windings.

5.4. General Atomic EPR

In the General Atomic (GA) experimental power reactor design [5-3], the stabilization of the toroidal magnet appears to be guided by the preferential use of well-tested components, such as NbTi-Cu, and cryo-static stability. In the latter area, a constant heat flux cannot be the only design number when a constant cross-section is desired. The magnetic flux density is a function of position, and along with it the magnetoresistance of the copper. Therefore, in the normal state, different dissipation rates are encountered for a constant cross-section. The GA design is characterized by a graded cross-section, based upon various heat flux densities. According to the arguments presented, 0.4 W/cm^2 is the maximum value of nucleate cooling. Accordingly, values significantly above this figure have been avoided. As pointed out in conjunction with the ANL EPR, there is no clear-cut physical post-quench situation which mandates the choice taken.

One point of interest for adiabatic stability is the size of the filament. The diameter of 200 microns chosen appears to be rather large and less conservative than the other design choices of GA.

No special details of the gravity-fed cooling system are presented. Therefore, the dangers of stratification, in conjunction with enhanced leaks in the liquid He IV lines, may shift the apparently favorable design choice to less favorable conditions. Other comments pertaining to this topic are given in the "comparison" section. We turn first to the

UWMAK design, as one of the most comprehensive early approaches to controlled thermonuclear reactor concept evolution.

5.5. UWMAK-III

A comparison of the various systems of the UWMAK series shows a clear trend toward higher power density, implying a reduction in total size. This is not necessarily true for the entire magnet system, but UWMAK-III [5-4] is characterized by a reduction in mass, due to the use of aluminum alloy or aluminum for the toroidal field systems in both composite and structure; Alloy 2219 has been proposed.

UWMAK-III, as well as the EPR's, appears to incorporate stabilizers selected on the basis of cryostatic stability criteria of the steady state. The term "total cryostatic stability" has been used to characterize the particular conservatism in design which guards against important accident potentials, compared to UWMAK-I.

A shift in emphasis appears to have taken place by the incorporation of the detailed design calculations of Boom and co-workers. To some extent these design equations are representative of more sophisticated methods than the early design approach. Separate dewars are provided, instead of a large common dewar, for added definition and protection. Special mechanical improvements have been used against lateral loading when a magnet has to be taken out due to quench or damage. Circuitry incorporates power supply switching from normal to -10,000 volts.

It is not clear what happens when not only one but two dewars are affected by a disturbing event with damage and/or quenches of two coils. The large change in electric potential appears to be another unknown from the safety point of view. In conjunction with a lowered He^4 density,

the related electric fields may be considered to be high-- close to the tolerable limits. In case of a severe event causing damage to the stabilizer-type II integrity, previous concerns such as overheating and arcing seem to apply to the operational anomaly encountered.

For the initial UWMAK-I system, one single large dewar vessel poses the question of emergency cooling capability, e.g. by back-up systems. UWMAK-III has partitioned the total toroidal field system energy considerably. Thus, one dewar failure affects only a minor portion of the total TF energy, about 4% of the total, with 3% going to adjacent coils by mutual coupling and 1% removed.

The energy storage inductor design appears to be guided by the same conservative safety philosophy used for the TF coil assembly of UWMAK-III.

After these individual discussions, the system comparisons in the following section considers additional points in the areas of coolant states and various coolant-composite interactions.

5.6. System Comparisons

In line with the historical development of steam generators, some interesting parallel trends in the area of large superconducting magnets may be discovered. Pool boiling modes appear to be the natural choice, and forced flow is the next step, implemented by only a few magnet groups in the last years. Aside from the ORNL EPR, the reliance on pool boiling modes for the He⁴ coolant is the dominant feature of the other systems under consideration. Table 5-2 summarizes the highlights of the different designs.

As most systems incorporate liquid capable of nucleate boiling, the condition of the pool may include a large heat transfer coefficient at

Table 5-2. EPR Stability and Safety Constraints for Solid-Coolant Interactions in TF Coils.

Design	Mode of Cooling	Design Rate of (W/cm ²)	Specific Safety Problems	Extreme	Other Advantages or Disadvantages
GA	Pool	0.293-0.427	Cryo-Bath: Stratification in conjunction with large diameter filaments and quality deficiencies	LMC*	Limited extrapolation on the basis of bubble chamber magnets.
ANL	Pool	0.35	Cryo-Bath: Stratification (subcooler malfunction) at T < 4.2 K	LMC*	
ORNL	Forced flow	0.065 (for 0.24 m/s)	Supercritical low heat transfer coefficient; entropy removal in long coolant duct	LMC* LOF**	High field; new concepts with lack of big magnet experience

*LMC = Loss of magnet coolant with local degradation.

**LOF = Loss of magnet coolant flow with local degradation.

the upper range of this regime, or during subcooling. This point appears to be a strong argument, on the basis of experience with bubble chamber magnets [5-5] using NbTi-Cu. However, one would like to avoid any possible hazards related to the coolant. This induces a look at stratification effects in nearly all of the designs.

When gravity is used for the delivery of liquid, a supply of He⁴ near liquid-vapor saturation for a height of 10 m causes a pressure rise, from top to bottom, of about 10 kPa. If 4.2 K is the temperature on the top (near the boiling point), the boiling point at the bottom is about 0.1 K higher. Measurements in laboratory dewars have shown larger values due to some stratification. Furthermore, the peak heat flux density of a cryo-static pool will be affected by the stratification conditions. Therefore, further deterioration in quality in the feed transfer line may bring the system closer to a quench when NbTi is chosen. These figures have to be viewed in conjunction with the external field which each individual conductor section will "see."

A remedy is the use of subcooled liquid He⁴, which in turn requires a pressurization heat exchanger (subcooler) [5-6]. Thus, a safer cooling system (from the point of view of the stabilizer) adds complexity. Moderate use of forced flow by circulation pumps adds moving devices, with the possibility of failures and repair or replacement. After these additional steps have been taken, the original simplicity of the pool cooling system may have been lost. Accordingly, the advantage of the conventional liquid coolant system with respect to the ORNL system may have been considerably diminished. In general, a rather wide range of thermodynamic parameters has to be considered when the cooling system is modified further for optimum recovery after a quench. For these

Table 5-3. Results of Magnet Safety Studies: Safety Recommendations⁽¹⁾
for Solid-Cryocoolant⁽²⁾ Interaction⁽³⁾.

Pressure Choice	$P = 0.1 \text{ MPa} + \epsilon$ ϵ small for 2- phase coolant state $\epsilon \geq 0.13 \text{ MPa}$ for supercritical coolant state
Temperature Choice	$T \leq 4.2 \text{ K}$ for subcooler (GA, ANL designs) $T \leq T_{cp}$ (transposed critical temperature)
Flow Mode	Circulation pump (GA, ANL) Short length sectioning (ORNL)

Note: (1) Quality control mandatory.

(2) Emergency cooling system.

(3) Data: Reference [5-7].

steps, a few guidelines have been summarized in Table 5-3 for the specific case of operation with pressurized He^4 , including near-atmospheric pressures. These recommendations have been based on studies with pressurized cryo-fluid described elsewhere [5-7].

The choice of pressure should be 0.1 MPa or higher in large, high-energy magnets for three reasons: First, the danger of the leakage of foreign matter, e.g. through cracks, into the He^4 fluid space is diminished, and He^4 will flow out, instead of having an accumulation of "ice" in the magnet system; second, the post-quench scenario will be characterized by wall-adjacent fluid of relatively "high" density, compared to values at low pressure; third, the dielectric behavior will be improved when the density is kept high. Point two is related to film boiling excursions which inject a large thermal impedance between liquid and solid. This may introduce a slow recovery from severe quenches. If the pressure is raised further, to the vicinity of and beyond the thermodynamic critical pressure, the density change becomes insignificant. Thermal switching, through, may be noticeable unless large forced-flow velocities are used. The latter mode of operation may be necessary because of the rather low thermal conductivity of He I.

The dielectric behavior is influenced in a distinct manner by the thermodynamic state of the fluid. At present, the appearance of vapor is quite detrimental to breakdown safety. Therefore, organic coatings, such as formvar, have been quite common for laboratory solenoids. This point of the most suitable coatings and their required thicknesses has received relatively little attention in the EPR design work.

In this context also, the use of the low-temperature-phase He II has been contemplated for the storage inductors possibly to be used in

conjunction with UWMAK-III. Again, for this choice of thermodynamic state, the use of 0.1 MPa or higher pressures is recommended. Most of the arguments in favor of pressurized (non-saturated) He⁴ are also valid for He II. Thermal switching at the lambda-curve cannot be avoided when the system undergoes a severe quench. As the apparent thermal conductivity is quite high in He II, the attraction appears to be the avoidance of components with moving parts (i.e. circulation pumps). On the other hand, the post-quench situation in He I is characterized by a rather low thermal conductivity, which demands pumping for the steady availability of high recovery heat fluxes. Transient conditions, however, may permit large cooling rates for short times.

As many details of the EPR designs are unknown, several open questions remain. We mention but briefly the choice of a sensor system for open circuits. Voltage taps are potential arc and breakdown opportunities. The optimum surfaces for the composites have not been determined with regard to various conflicting requirements. In the latter area, both the rigid system (e.g. UWMAK-III) and the flexible design (e.g., jelly roll windings) have been chosen. The ultimate answers concerning these and other details still have to be found. However, continuing progress in magnet construction seems to show that more reliable and optimized magnets will be available in the future.

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