LMFBR Conceptual Design Study: An Overview of Environmental and Safety Concerns

**Prepared for the Environmental and Safety Engineering
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LMFBR CONCEPTUAL DESIGN STUDY:
AN OVERVIEW OF ENVIRONMENTAL
AND SAFETY CONCERNS

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SUMMARY

The U.S. Department of Energy (DOE)^(a) initiated the Liquid Metal Fast Breeder (LMFBR) Conceptual Design Study (CDS) with the objective of maintaining a viable breeder option. The project is scheduled to be completed in FY-1981 but decisions regarding plant construction will be delayed until at least 1985. This report provides a review of the potential environmental and safety engineering concerns for the CDS and recommends specific action for the Environmental and Safety Engineering Division of DOE. Based on this work, the following recommendations are made:

PRIORITY I

- A program should be established to monitor the performance of environmental control systems at the Fast Flux Test Facility. Currently this is the only opportunity in the U.S. to review and evaluate the components that must be scaled up for use in future 1000-MWe commercial facilities.
- Tritium control technology developed for Light Water Reactors (LWRs) should be reviewed to determine if it can be applied to the LMFBR design proposed by the CDS. Although total emissions from LMFBRs may be less than for LWRs, the guideline of controlling emissions to as low as possible still seems applicable. Also, recent experimental studies indicate that fast fission tritium yields may be an order of magnitude greater than previously reported.
- Hydrogen monitoring and control methods should be reviewed to determine if these methods can be used during the conditions postulated for a hypothetical core-disruptive accident. The hydrogen is produced when sodium comes into contact with concrete and it poses a potential explosion hazard.
- The licensability of the environmental control technology specified in the CDS should be determined. In the past, U.S. Nuclear Regulatory Commission (NRC) staff and DOE have often differed on the basic safety and implementation approach.

(a)Reactor Research and Technology Division, Office of Nuclear Energy Programs.

- Cold trap sodium purification technology should be reviewed to determine if the control efficiency of tritium and other radionuclides can be improved by design and/or process changes.

PRIORITY II

- The technology available for sensing and controlling sodium leaks should be reviewed, including methods for suppressing sodium smoke and preventing its release into personnel areas.
- The desirability of treating sodium wastes offsite should be evaluated. This has been mentioned as a possibility, presumably for safety reasons.
- The need to place a limit on the maximum activity that can be stored in any part of the Radioactive Argon Processing System at one time must be assessed. This would limit the amount of radiation that could be released in the event of a leak or equipment failure.
- Special considerations, if any, need to be determined for the decommissioning of a plant. Activated corrosion products plate-out and diffuse into system components; this may make decontamination and subsequent decommissioning more difficult.
- The shielding and maintenance practices associated with the primary sodium loop should be reviewed. Millions of pounds of sodium containing ^{22}Na , ^{24}Na and activated corrosion products are present in this system.

The environmental and safety aspects for the Fast Flux Test Facility (FFTF), Clinch River Breeder Reactor (CRBR) and Prototype Large Breeder Reactor (PLBR) project were reviewed. An attempt was made to identify potential problem areas that are unique to LMFBRs and not shared by LWRs and fossil plants. Fifteen areas of potential concern for the CDS were noted. The following findings provide the basis for the recommendations listed above.

FISSION AND ACTIVATION PRODUCTS

Many of the fission products are soluble in sodium and they are subsequently removed in a cold trap, mainly by precipitation processes. Volatile products are emitted to the argon cover gas over the liquid sodium.

A Radioactive Argon Processing System, consisting of multiple cryogenic charcoal beds removes essentially all of the activity. Argon-39 is not removed and it is recycled as a part of the cover gas. Krypton is separated by fractional distillation and put in bottles for storage.

Any leaks through valves and seals allows radioactive gas to enter the nitrogen cell atmosphere. The Cell Atmosphere Processing System uses hold-up tanks to allow decay of noble gases. Tritium is removed and converted to a solid form of tritiated calcium hydroxide.

Tritium and argon-39 are the largest radionuclide release from an LMFBR power plant. Radionuclide emissions are at least an order of magnitude less than for a comparable size LWR. However, recent experimental studies indicate that fast fission tritium yields are an order of greater than previously reported. So, additional control of tritium on LMFBRs will probably receive further consideration.

CORE-DISRUPTIVE ACCIDENTS

Any event that would cause the internal integrity of the reactor to be altered is called a core-disruptive accident (CDA). An energetic CDA is one in which criticality occurs in the core configuration after the accident. The NRC requires 24-hour containment integrity as a design criterion. For FFTF, the NRC staff noted that hydrogen resulting from sodium and core-debris interaction with concrete is a principal challenge to containment. It was therefore recommended that hydrogen monitoring and control methods be investigated for CDA conditions.

SODIUM SPILLS

Small leaks can develop into big leaks. The early detection of small leaks is important in both the primary and secondary coolant systems. If the leak is detected when it is small, the pressure can be reduced or the sodium can be drained. More research and development efforts are required to devise improved methods for sensing and controlling sodium leaks.

Large sodium spills represent a principal challenge to containment, especially with the hydrogen production associated with sodium-concrete reactions. Hydrogen recombiners are used in LWRs and, according to NRC staff, they need to be considered for LMFBRs also.

SODIUM PURIFICATION

The sole method used universally to purify sodium is cold trapping. The principal purpose of the cold trap is to control dissolved oxygen by removing it as disodium oxide. However, the cold trap is also the principal control device for many radionuclides, including tritium. While radionuclides do not seem to be a serious problem for LMFBR operations, few systematic studies have been conducted to understand the control mechanisms for various radionuclides. This makes it difficult to devise more efficient and effective cold traps.

RADIATION EXPOSURE

In addition to the reactor core, the primary sodium loop is the major source of radiation. Sodium-22 and sodium-24 are formed by activation of the sodium coolant as it passes through the reactor. Special shielding is therefore required for all areas where personnel must go during normal operations.

NRC reports that the estimated average collective dose to all onsite personnel at a 1000-MWe LWR is 250-450 man-rem/plant year. Estimates for CRBR are 280 man-rem/plant year.

The design, operational and maintenance procedures are such that the radiation protection for plant personnel in a commercial LMFBR will be consistent with the requirements of 10 CFR 20.

SODIUM/WATER REACTIONS

Sodium reacts with water to form hydrogen, a potential explosion hazard. The secondary sodium loop interfaces with the steam generation system. Leak detection and early sodium-water/steam reaction warning systems are used. In the event of a leak, procedures are used to shut the plant down or dump the

sodium and/or water into tanks. The secondary sodium heat exchanger is designed to prevent failure from pressure caused by reactions. This prevents radioactive sodium from being released. Thus, adequate equipment and procedures are available to prevent damage from potential sodium-water/steam reactions.

SODIUM/CONCRETE INTERACTIONS

When molten sodium comes into contact with concrete, water evolves and subsequently potentially explosive hydrogen is generated. This problem has been addressed and steel liners are used in containment vessels. Even so, NRC still recommends the use of monitoring and control equipment for hydrogen.

HEAT TRANSPORT SYSTEM

The integrity of the heat transport system is essential to reactor safety. The sodium inventory for a 1000-MWe plant is about 4,500,000 lb and coolant flow rates are 70,000 to 80,000 gpm. These figures are staggering and by themselves command a prudent approach.

Faults that might occur seem to have been considered and covered by design-safety considerations. Reliable systems are provided to sense heat removal faults and to provide for reactor shutdown and continued heat removal for fission-product decay. The specifics of decay heat removal systems have been the subject of critical review by NRC for CRBR and FFTF.

PLUTONIUM TOXICITY

Substantial quantities of plutonium oxide are used in LMFBR fuel elements. This material has been shown to be extremely toxic to laboratory animals but little data are available concerning its effects on humans.

Exposure is required for toxicity to be a problem. The core disruptive accident is one hypothetical case with plausibility. After extensive analyses, the NRC staff has concluded that this accident can be contained and the potential adverse exposure eliminated.

Illegal confiscation presents another avenue for exposure. Extensive safeguard programs have been incorporated to minimize this risk.

LICENSING

In the past, NRC staff and DOE have differed in basic safety and implementation approaches to CRBR and FFTF reactors. This must be recalled in any new LMFBF project. The licensing of commercial-size LMFBFs is difficult to predict, particularly until the technology has been demonstrated on a large scale.

SODIUM WASTES

Sodium wastes are converted into less reactive forms. The main sources of wastes are cold traps and contaminated hardware. Sodium waste from LMFBFs may be processed offsite.

ACCIDENTS WITH RADIOACTIVE WASTES

Fire and failure of the Radioactive Argon Processing System are two events of concern. If a fire should occur, the fire suppressing system is quickly engaged. The Cell Atmosphere Processing System would prevent significant release of radioactive materials.

On FFTF, the NRC staff concluded that failure of the Radioactive Argon Processing System was the most limiting event. A recommendation was made that limitations be placed on the maximum activity that can be stored in any part of the system at one time.

DECOMMISSIONING

Some activated corrosion products diffuse into steel components and may pose decommissioning problems. FFTF staff concludes that decommissioning a number of small power plants plus refueling and maintenance experience at FFTF provide an adequate basis for safe decommissioning.

FUEL HANDLING ACCIDENTS

Fuel handling accidents considered, although unlikely, include loss of cooling of fuel transfer machines, complete loss of cooling in interim fuel

assembly examination cells, and loss of cooling capability in interim decay storage tanks.

On FFTF, a postulated accident in the examination cell was evaluated. Because of the constant action of the Cell Atmosphere Processing System, virtually no release of radioactivity would take place. It was concluded that actions can be taken to minimize the potential for such accidents but if they do occur, any resultant releases will be relatively inconsequential.

CHEMICAL WASTES

Nonradioactive chemical wastes are produced mainly during water treatment for the steam generator and heat rejection systems. For this reason these wastes are not substantially different in type or quantity when compared to LWR or fossil plants. Therefore, chemical wastes are not a major concern for LMFBRs.

CONTENTS

SUMMARY	iii
1.0 INTRODUCTION	1.1
2.0 CONCLUSIONS AND RECOMMENDATIONS	2.1
2.1 MONITORING AT FFTF	2.1
2.2 TRITIUM CONTROL	2.1
2.3 HYDROGEN MONITORING AND CONTROL	2.1
2.4 LICENSING	2.2
2.5 SODIUM PURIFICATION	2.2
2.6 SODIUM LEAKS	2.2
2.7 SODIUM WASTES	2.3
2.8 ACTIVITY LIMITS	2.3
2.9 DECOMMISSIONING	2.3
2.10 SHIELDING	2.3
3.0 LIQUID METAL FAST BREEDER REACTOR PROJECTS	3.1
3.1 EARLY BREEDER ACTIVITIES	3.1
3.2 ESTABLISHMENT OF A PRIORITY BREEDER PROGRAM	3.2
3.3 FAST FLUX TEST FACILITY	3.3
3.4 CLINCH RIVER BREEDER REACTOR	3.4
3.5 PROTOTYPE LARGE BREEDER REACTOR	3.6
3.6 CONCEPTUAL DESIGN STUDY	3.7
3.7 DESIGN CRITERIA FOR CONCEPTUAL DESIGN STUDY	3.8
3.7.1 Plant Performance	3.9
3.7.2 Construction Schedule	3.9

3.7.3	Technology Basis	.	.	.	.	.	.	.	3.9
3.7.4	Proliferation Resistance	.	.	.	.	.	.	.	3.9
3.7.5	Site Location/Seismic Design Criteria	.	.	.	.	.	.	.	3.9
3.7.6	Design Safety Criteria	.	.	.	.	.	.	.	3.10
3.7.7	Codes and Standards	.	.	.	.	.	.	.	3.10
3.7.8	Inspection and Maintenance	.	.	.	.	.	.	.	3.10
3.8	CDS PHASE I STUDY RESULTS	.	.	.	.	.	.	.	3.11
3.8.1	Plant Size	.	.	.	.	.	.	.	3.12
3.8.2	Plant Type	.	.	.	.	.	.	.	3.12
3.8.3	Operating Temperatures	.	.	.	.	.	.	.	3.12
3.8.4	Reactor Core and Fuel	.	.	.	.	.	.	.	3.12
3.8.5	Fuel Handling and Storage	.	.	.	.	.	.	.	3.12
3.8.6	Containment/Confinement	.	.	.	.	.	.	.	3.13
3.8.7	Steam Cycle	.	.	.	.	.	.	.	3.13
3.8.8	Shutdown Heat Removal System	.	.	.	.	.	.	.	3.13
4.0	ENVIRONMENTAL AND SAFETY ENGINEERING CONCERNS	.	.	.	.	.	.	.	4.1
4.1	FISSION AND ACTIVATION PRODUCTS	.	.	.	.	.	.	.	4.2
4.1.1	Radioactive Argon Processing System	.	.	.	.	.	.	.	4.3
4.1.2	Cell Atmosphere Processing System	.	.	.	.	.	.	.	4.3
4.1.3	Comments	.	.	.	.	.	.	.	4.5
4.2	CORE-DISRUPTIVE ACCIDENTS	.	.	.	.	.	.	.	4.8
4.2.1	Comments	.	.	.	.	.	.	.	4.9
4.3	SODIUM SPILLS	.	.	.	.	.	.	.	4.14
4.3.1	Small Sodium Fires	.	.	.	.	.	.	.	4.14

4.3.2	Large Sodium Fires	.	.	.	.	.	.	.	4.16
4.3.3	Comments	.	.	.	.	.	.	.	4.18
4.4	SODIUM PURIFICATION	.	.	.	.	.	.	.	4.20
4.4.1	Comments	.	.	.	.	.	.	.	4.20
4.5	RADIATION EXPOSURE	.	.	.	.	.	.	.	4.22
4.5.1	Comments	.	.	.	.	.	.	.	4.23
4.6	SODIUM/WATER REACTIONS	.	.	.	.	.	.	.	4.26
4.6.1	Comments	.	.	.	.	.	.	.	4.26
4.7	SODIUM/CONCRETE REACTIONS	.	.	.	.	.	.	.	4.28
4.7.1	Comments	.	.	.	.	.	.	.	4.28
4.8	HEAT TRANSPORT SYSTEMS	.	.	.	.	.	.	.	4.30
4.8.1	Comments	.	.	.	.	.	.	.	4.32
4.9	PLUTONIUM TOXICITY	.	.	.	.	.	.	.	4.33
4.9.1	Comments	.	.	.	.	.	.	.	4.34
4.10	LICENSING	.	.	.	.	.	.	.	4.37
4.10.1	Comments	.	.	.	.	.	.	.	4.39
4.11	SODIUM WASTES	.	.	.	.	.	.	.	4.41
4.11.1	Comments	.	.	.	.	.	.	.	4.41
4.12	ACCIDENTS WITH RADIOACTIVE WASTES	.	.	.	.	.	.	.	4.42
4.12.1	Comments	.	.	.	.	.	.	.	4.43
4.13	DECOMMISSIONING	.	.	.	.	.	.	.	4.45
4.13.1	Comments	.	.	.	.	.	.	.	4.45
4.14	FUEL HANDLING ACCIDENTS	.	.	.	.	.	.	.	4.46
4.14.1	Comments	.	.	.	.	.	.	.	4.47

4.15 CHEMICAL WASTES	4.49
4.15.1 Comments	4.51
5.0 BIBLIOGRAPHY	5.1
APPENDIX A: DESIGN CRITERIA FOR PROTOTYPE LARGE BREEDER REACTOR . .	A.1

FIGURES

1.1 Schematic of LMFBR Facility	1.2
4.1 Radioactive Argon Processing System	4.4
4.2 Cell Atmosphere Purification System	4.6

TABLES

4.1 Estimated Radionuclide Releases for a 1,000-MWe LMFBR Power Plant and Reference LWR	4.5
4.2 FFTF Meltdown Accident Doses	4.35

1.0 INTRODUCTION

Light Water Reactors (LWRs) utilize only one to two percent of the energy available in the fuel. Furthermore, they must use uranium enriched to approximately three percent in ^{235}U . In naturally occurring uranium only 0.7% is the thermally fissionable isotope ^{235}U , the rest being ^{238}U . It is an expensive and complicated process to enrich ^{235}U , and the ^{238}U is eventually wasted. This inefficient use of uranium in LWRs detracts from the long-term availability of low cost uranium.

Because the breeder reactor converts nonfissionable ^{238}U to fissionable ^{239}Pu , the otherwise wasted ^{238}U is transformed into a useful fuel. More ^{239}Pu is produced than consumed; hence, the term "breeder." This fact of breeding enables the reactor to replenish its own fuel needs and eventually provide enough additional ^{239}Pu to operate another reactor. The breeder reactor, therefore, has the potential to extend available uranium fuel resources from a few decades to centuries.

Liquid sodium coolant was chosen for the U.S. breeder program because of predicted performance, industrial support, a broad base of technological experience and proven basic feasibility. It has excellent heat transfer properties, a tendency to react with, or dissolve, many fission products and retain them within the sodium, and its neutronic properties are such that it does not interfere with the breeding of ^{239}Pu . There are, however, certain drawbacks, including chemical reactivity with air and water and activation under irradiation.

A simplified schematic of the basic Liquid Metal Fast Breeder Reactor (LMFBR) components is shown in Figure 1.1. It consists of the reactor core, control rod drives, reactor vessel and associated pumps, piping, instrumentation, and equipment. Plutonium dioxide and uranium dioxide are fabricated into mixed oxides for the core fuel rods. Uranium dioxide is used in the blanket rods.

The heat transport system consists of sodium pumps, piping, intermediate heat exchanger, steam generator, turbine generator, and condenser (not shown).

FIGURE 1.1. Schematic of LMFBR Facility

Heat generated in the reactor core is transferred by forced circulation of the liquid sodium primary coolant to an intermediate heat exchanger and then through an essentially nonradioactive secondary sodium coolant system to steam generators in which superheated steam is produced. This steam drives a turbine-generator to generate electrical power. Waste heat is rejected either to a nearby river, lake, or ocean by once-through cooling or to the atmosphere in wet or dry cooling towers.

All of the sodium is covered with a blanket of argon gas. Gaseous radionuclides are removed by the Radioactive Argon Processing System. The purified argon is recycled.

The development of the LMFBR provides an additional option among the power generating technologies. To this objective, the U.S. Government has performed a number of studies and projects. A current project is the Conceptual Design Study (CDS) for a 1000-MWe plant. The purpose of the CDS is to incorporate experience from the Fast Flux Test Facility (FFTF), Clinch River Breeder Reactor (CRBR), and the Prototype Large Breeder Reactor (PLBR) study and serve as a focus for the next logical step in the U.S. LMFBR Program.

The purpose of this report is to:

1. briefly review U.S. LMFBR design and operational experience,
2. identify environmental and safety engineering concerns that should be addressed in the CDS study, and
3. recommend specific actions to be taken by the Environmental Safety Engineering Division in DOE with respect to its overview functions in the CDS.

The scope of this work was limited to the FFTF, CRBR and PLBR projects. A literature review, phone interviews and site visits have been used to gather information on the FFTF project. PLBR and CRBR reports provided the basis for reviewing these projects. This work was prepared using a literature base through September 1980.

2.0 CONCLUSIONS AND RECOMMENDATIONS

In FY-1981 the second phase of the Conceptual Design Study (CDS) will be completed. It is not clear how CDS will proceed after that. It appears that adequate funding will be provided to continue the project, but it will be at least 1985 before a specific decision is made about the construction of a 1000-MWe plant. In preparation for reviewing the CDS phase II report and future work, certain actions are recommended for the Environmental and Safety Engineering Division of DOE. They are described in the following subsections. Based on this work the first five recommendations are considered Priority I and the second five, Priority II.

2.1 MONITORING AT FFTF

A program should be implemented to monitor the performance of environmental control technology at FFTF. This is currently the only opportunity in the U.S. to review and evaluate the performance of large-scale equipment that might be later specified for commercial LMFBRs. This monitoring project would involve inspecting control equipment and evaluating performance. Such a monitoring program is essential to the Environmental and Safety Engineering Division.

2.2 TRITIUM CONTROL

Tritium control technology developed for LWRs should be reviewed to determine if it can be applied to the LMFBR design proposed by CDS. Although total emissions from LMFBRs are less than for LWRs, the guideline of controlling emissions to as low as possible still seems applicable. Recent experimental studies indicate that tritium yields are an order of magnitude greater than previously reported. Most of the tritium is collected in the sodium purification cold traps in the primary and secondary loops. However, if more stringent regulations are applied, additional control systems must be considered.

2.3 HYDROGEN MONITORING AND CONTROL

Hydrogen monitoring and control methods should be reviewed to determine if these methods can be used during the conditions postulated for a hypothetical

core-disruptive accident. The hydrogen is produced when sodium comes into contact with concrete and it poses a potential explosion hazard. The current approach is two-fold. First, steel liners are used to protect the concrete in the event of a spill. Second, analytical studies by FFTF and CLBR staff indicate that natural mechanisms will tend to prevent hydrogen buildup in the event of a spill. NRC staff, however, recommend the use of hydrogen control systems.

2.4 LICENSING

The licensability of the environmental control technology specified in the CDS should be determined. In the past, NRC staff and DOE have often differed on the basic safety and implementation approach. FFTF has received operational approval from NRC but there were several points unresolved on CLBR when that review was terminated by Presidential action in 1977. One area of contention is the use of sacrificial materials in the reactor cavity that do not produce large volumes of hydrogen and other gases in the event of a CDA or large sodium spill. NRC favors such an approach. Neither FFTF nor CRBR designs contain this feature.

2.5 SODIUM PURIFICATION

Cold trap sodium purification technology should be reviewed to determine if the control efficiency of tritium and other radionuclides can be improved by design and/or process changes. Cold traps are used in the primary and secondary sodium loops to control oxygen levels and thus reduce corrosion throughout the system. The oxygen is removed as disodium oxide. Many radionuclides are also removed in the cold traps but few studies have been conducted to determine removal mechanisms or changes that might increase control efficiency.

2.6 SODIUM LEAKS

The technology available for sensing and controlling sodium leaks should be reviewed, including methods for suppressing sodium smoke and preventing its release into personnel areas. Sodium is very reactive with air or water

and since millions of pounds of sodium are used, an ongoing concern exists about leaks. Detecting and controlling small leaks is one key to eliminating large spills.

2.7 SODIUM WASTES

The desirability of treating sodium wastes offsite should be evaluated. This has been mentioned as a possibility, presumably for safety reasons. This waste is a concern for two reasons. First, the sodium is highly reactive and hence caution is necessary, especially for onsite treatment. Second, the waste contains radionuclides because much of it comes from the sodium cold traps.

2.8 ACTIVITY LIMITS

The need to place a limit on the maximum activity that can be stored in any part of the Radioactive Argon Processing System at one time must be assessed. This would limit the amount of radiation that could be released in the event of a leak or equipment failure. This is a practical safety point that has been previously recommended by NRC for FFTF.

2.9 DECOMMISSIONING

Special considerations, if any, need to be determined for the decommissioning of a plant. Activated corrosion products plate-out and diffuse into system components; this may make decontamination and subsequent decommissioning more difficult. FFTF staff say that current technology and experience are available to decommission a facility such as FFTF.

2.10 SHIELDING

The shielding and maintenance practices associated with the primary sodium loop should be reviewed. Millions of pounds of sodium containing ^{22}Na , ^{24}Na and activated corrosion products are present in this system. While no new technology seems to be required to meet standards, it is a concern unique to LMFBRs and as such the Environmental and Safety Engineering Division staff needs to be in a position to review the system.

3.0 LIQUID METAL FAST BREEDER REACTOR PROJECTS

This section gives a brief history of U.S. LMFBR activities and then focuses on four recent projects: 1) Fast Flux Test Facility (FFTF), 2) Clinch River Breeder Reactor (CRBR), 3) Prototype Large Breeder Reactor (PLBR), and 4) Conceptual Design Study (CDS). Particular attention is given to the design criteria and plant specifications for the CDS. The environmental and safety engineering concerns for LMFBR projects are presented in the next section.

3.1 EARLY BREEDER ACTIVITIES

The U.S. interest in breeder reactors dates back to the Manhattan Project days in the early 1940s when the possibility of breeding was first recognized by pioneers in the nuclear field. To obtain the advantage of breeder reactors, the U.S. Government has been working for over 25 years on the sodium-cooled breeder reactor. The fast breeder research and development program has been continuous since 1945, although it has experienced variations in its priority for development. Much of the essential effort on the breeder has been conducted in the government national laboratories. One of the earliest steps in this program was the construction of the experimental Clementine fast neutron flux reactor at Los Alamos which, from March 1946 to December 1953, was used to explore the possibility of operating with a fast neutron flux, plutonium fuel and a liquid metal coolant--in this case mercury.

The Experimental Breeder Reactor I (EBR-I), pioneering in nature, operable in 1951, was the first reactor to prove the feasibility of breeding. Further, it was the first reactor to establish the engineering feasibility and technology of liquid-metal coolants and of small-scale liquid-metal components and instrumentation and control. Experience gained from the design, construction and operation of EBR-I contributed important information to LMFBR engineering and technology. Importantly, it made a noteworthy contribution to analyses of reactor stability and demonstrated that fast breeder reactors are inherently stable.

Further fast breeder reactor developments led to the construction in the U.S. in the mid-1950s of two fast reactors, the 62.5-MWt EBR-II and the

200-MWt Fermi Reactor. In the mid-1960s, construction was begun on the 20-MWt Southwest Experimental Fast Oxide Reactor (SEFOR) facility directed at providing a facility to conduct research on the safety of LMFBRs. Fast breeder reactor efforts were also begun in the United Kingdom, USSR and France.

3.2 ESTABLISHMENT OF A PRIORITY BREEDER PROGRAM

In 1962 the U.S. Atomic Energy Commission (AEC) reported to the President that the use of breeders could solve the problem of providing an adequate and economic energy supply for the foreseeable future. It was concluded that nuclear energy can and should make an important, and eventually a vital, contribution toward meeting our long-term energy requirements and that economic breeders were essential to any long-range, major use of nuclear energy.

In evaluating the future course to be taken by the U.S. advance reactor development program, the AEC, in early 1965, initiated a series of technical reviews. These reviews of the reactor program indicated that additional important engineering information was required, and that additional facilities and other resources were necessary to obtain that information. There was clear evidence of the need to strengthen the engineering capabilities of the laboratories and industry, and to assemble necessary and adequate resources if safe, reliable and economical breeder power plants suitable for operation in the utility environment were to be developed. These early reviews further indicated a requirement for in-depth review of each of the technical elements of the breeder program. Concurrently, it was necessary to initiate detailed plans for each of these elements.

During the early and mid-1960s remarkable advances were taking place in the development of light water reactor power plants, and nuclear power moved toward widespread acceptance as a new source of electrical energy. It was recognized that the plutonium produced in light water reactors could be most efficiently used in fast breeder reactors, and that the breeder would measurably reduce uranium ore requirements. The breeder development program was thus invested with a sense of urgency which had been lacking.

In early 1967, the AEC issued another report to the President and reaffirmed the promise of the breeder for meeting our long-term energy needs and established the LMFBR program as the AEC's highest priority civilian reactor development effort.

3.3 FAST FLUX TEST FACILITY

The AEC decided in 1965, based on a national consensus, that construction of an FFTF must be undertaken if the national objectives of the liquid metal-cooled fast reactor development program were to be achieved. The FFTF was initiated by the AEC in 1966. The sodium loops were filled in 1979, and the facility is scheduled to begin operation in 1980.

This nuclear complex consists of a fast flux test reactor and associated heat removal systems, coolant servicing systems, fuel handling systems, control systems, waste disposal and facilities, post-irradiation examination facilities, maintenance facilities and administrative offices. These facilities provide an advanced high-level fast neutron flux reactor facility for carrying out a comprehensive fuels and materials testing program to develop and demonstrate economical fuels and satisfactory materials for the LMFBR program. The FFTF incorporates closed loops in a fast flux environment, representative of LMFBR conditions, complete with coolant instrumentation and control (flow, temperature and impurities) and fuels and materials instrumentation. Also, the FFTF provides other in-core space, with an environment representative of LMFBR conditions, which is required to test statistically larger quantities of candidate fuels and materials specimens.

The FFTF, which is typical of many of the components and systems in LMFBR demonstration plants, has become a center for the application of disciplined LMFBR engineering to the design, construction, testing, operation and maintenance of first-of-a-kind nuclear plant projects. The FFTF provides a focus for management, design and development activities for much of the LMFBR program.

The major objectives of the FFTF are:

1. To provide a strong, disciplined engineering base for the LMFBR program, principally in the following areas:
 - Fast flux irradiation of LMFBR fuels and materials,
 - Design, construction, operation and maintenance of LMFBRs, including:
 - Statistically significant experimental data for the LMFBR program,
 - Verification of LMFBR design,
 - Verification of LMFBR analytical procedures,
 - Identification of needs for, and preparation of criteria, codes, standards and guides,
 - Physics,
 - Fuels and materials,
 - Safety,
 - Sodium technology,
 - Sodium components and systems,
 - Instrumentation and control,
 - Fuel cycle and
 - Plant design and overall planning and
 - Training ground for personnel from industrial organizations, utilities and national laboratories.
2. Provide fast flux testing.
3. Contribute to the development of a viable self-sustaining competitive U.S. LMFBR industry.

3.4 CLINCH RIVER BREEDER REACTOR

The CRBR, as part of the AEC's "Cooperative Power Reactor Demonstration Program," was announced in January, 1972. Commonwealth Edison Company of Chicago and Tennessee Valley Authority proposed to work with the AEC to design,

develop, construct and operate CRBR as the first LMFBR demonstration plant. To implement this proposal, two non-profit organizations, Breeder Reactor Corporation (BRC) and Project Management Corporation (PMC), were established to coordinate utility participation, with PMC providing centralized management of the CRBR through the completion of the five-year demonstration program.

In November of 1972, Westinghouse Electric Corporation was selected by AEC/PMC as the lead supplier of the Nuclear Steam Supply System. The General Electric Company and Atomics International (Division of Rockwell International Corporation), who also submitted proposals for the Nuclear Steam Supply System, were authorized to be retained by Westinghouse Electric Corporation as major subcontractors to design the Nuclear Steam Supply System. Burns and Roe, Incorporated, was retained by PMC in December of 1972 as the project architect-engineer.

The objectives of the CRBR are as follows:

1. To confirm and demonstrate the potential value and environmental desirability of the LMFBR concept as a practical and economic future option for generating electrical power;
2. To confirm the value of this concept for conserving important non-renewable national resources;
3. To develop, for the benefit of government, industry and the public, important technological and economic data;
4. To provide a broad base of experience and information important for commercial and industrial application of the LMFBR concept; and
5. To verify certain key characteristics and capabilities of LMFBR plants for operation on utility systems such as licensability and safety, operability, reliability, availability, maintainability, flexibility and prospect for economy.

The capability of the LMFBR as a safe commercial power producing facility must be demonstrated. Specifically, the CRBR is intended to help industry and government:

1. Demonstrate that the necessary technology is indeed available to scale up and successfully construct and operate commercial-sized LMFBRs;
2. Provide a technical basis for extending the technology to future commercial plants where improvements in fuel life, plant capacity and thermal efficiency will be made for economic reasons;
3. Develop operating data on the environmental impact of the LMFBR before large numbers of commercialized LMFBRs are constructed;
4. Provide a demonstration of the nuclear parameters necessary for commercial development;
5. Demonstrate the minimal impact from disposal of radioactive waste materials;
6. Demonstrate the equipment on a large scale; and
7. Demonstrate the breeder concept in an industrial environment.

3.5 PROTOTYPE LARGE BREEDER REACTOR

In November 1975, the U.S. Energy Research and Development Administration (ERDA) and the Electric Power Research Institute, Inc. (EPRI) initiated a three-phase LMFBR Plant Design Project for a 1000-MWe facility.

- Phase I - Establish plant characteristics and design basis,
- Phase I- - Develop preliminary overall conceptual design-- components, systems and structures,
- Phase III - Complete integrated plant conceptual design.

The Phase I effort was completed in June 1976 and was reported in two documents: "Pool-Loop-Hybrid Liquid Metal Fast Breeder Reactor Plant Comparison," and "Phase I Summary Report - Liquid Metal Fast Breeder Reactor Design Report." Both of these documents were issued as joint Atomics International and Burns & Roe project reports.

At the end of Phase I, ERDA-EPRI defined an interim Phase II period to allow further characterization of the pool and loop concepts. At the end of this interim period, ERDA-EPRI issued project guidelines to all contractors to begin Phase II work on plants employing a loop-type primary heat transport system with hot leg pumps. This Phase II effort was reported in three documents: "PLBR Core Selection for Low CDA Energetics," "Hockey Stick Steam Generator for Prototype Large Breeder Reactor," and "Liquid Metal Fast Breeder Reactor Conceptual Plant Design 1000 MWe."

Phase III of this project was not completed. EPRI did continue to work on certain portions of it after ERDA stopped supporting the project.

3.6 CONCEPTUAL DESIGN STUDY

The LMFBR CDS was initiated by the U.S. Department of Energy (DOE) on October 1, 1978, with the objective of developing an LMFBR plant conceptual design which will incorporate the experience from FFTF, CRBR, and PLBR; serve as a focus for the LMFBR base program; and provide a logical next step in the U.S. program.

The current study is organized in two overlapping phases--Phase I, running from October 1, 1978 to March 31, 1980, and Phase II, running from December 1, 1979 to March 31, 1981. During Phase I of the study, tradeoff studies were conducted to select the principal plant performance parameters and design requirements in accordance with the criteria, objectives and guidelines specified by DOE.

The objectives of Phase II of the CDS study are to: 1) complete the conceptual design of the development plant, with documentation to the system design description level; 2) conduct the plant level analyses necessary to document the performance, economic, and design safety aspects of the developmental plant and potential deployed plants; 3) develop the Conceptual Design Report; and 4) conduct design and analysis of critical aspects of an alternate plant configuration to maintain technical cognizance of this design option.

The CDS program is managed by the Reactor Research and Technology Division (RRT), Office of Nuclear Energy Programs of the DOE. Support is provided by the following national and engineering laboratories as a part of their overall LMFBF activities.

- Argonne National Laboratory
- Los Alamos Scientific Laboratory
- Oak Ridge National Laboratory
- Idaho National Engineering Laboratory
- Hanford Engineering Development Laboratory
- Energy Technology Engineering Center

Numerous organizations are actively participating in the CDS program. These include the technical integrator, five reactor manufacturers, and three architect-engineering organizations listed below.

- Technical Integrator
 - Boeing Engineering and Construction Co.
- Reactor Manufacturers
 - Atomics International Division
 - Babcock and Wilcox Company
 - Combustion Engineering Incorporated
 - General Electric Company
 - Westinghouse Electric Corporation
- Architect-Engineers
 - Bechtel National Corporation
 - Burns and Roe Incorporated
 - Stone and Webster Engineering Incorporated

3.7 DESIGN CRITERIA FOR CONCEPTUAL DESIGN STUDY

The design criteria for the CDS follow largely from the PLBR work. The PLBR design criteria are included in the Appendix. The main additional

constraints added for the CDS are: 1) resistance of the final cycle to diversion of weapons-grade materials, 2) near-term availability of technology, and 3) a construction schedule. The CDS work is being conducted with the following general criteria.

3.7.1 Plant Performance

The plant shall operate safely, reliably and economically and shall demonstrate significant design, operational, performance, and maintenance features of a large LMFBR power plant. Life cycle costs shall be minimized while adhering to the elements of these criteria, objectives and guidelines.

3.7.2 Construction Schedule

The plant shall be brought to full power operation within approximately 10 years of initiation of Title I.

3.7.3 Technology Basis

System designs shall be based primarily on application of current or near-term technology development (available in 5 years or less) and on experience in design, construction or operation. Consideration shall be given to use of existing designs for systems and components, such as were developed for FFTF, CRBR, or base program applications, where selection of such designs provide significant benefits. Selective consideration shall be given to advances in technology directed toward more economical or reliable designs.

3.7.4 Proliferation Resistance

Resistance of fuel cycle and plant design concepts to diversion of weapons-usable materials shall be a principal design criterion. Non-proliferation Alternative Systems Assessment Program (NASAP) and International Nuclear Fuel Cycle Evaluation (INFCE) results shall be utilized for guidance in this area.

3.7.5 Site Location/Seismic Design Criteria

The CDS Development Plant shall be designed for installation at an established government nuclear reservation, such as the Savannah River Laboratory (SRL), Idaho National Engineering Laboratory (INEL), or Hanford

Engineering Development Laboratory (HEDL). However, the plant shall also be suitable for location in the majority of potential U.S. sites and shall meet the requirements of the CDS Seismic Design Criteria.

3.7.6 Design Safety Criteria

The plant will undergo a safety review by the U.S. Nuclear Regulatory Commission (NRC), similar to that conducted for FFTF. This approach will be confirmed by Congress when they approve proceeding with final design and site selection.

The design safety criteria developed in Phase I shall be the primary basis for system design. (These criteria were developed using the general design criteria of 10 CFR, applicable NRC Regulatory Guides and design safety criteria developed in the PLBR studies as reference starting points.) Simplifying modifications or alternatives to these criteria shall be considered where a good technical basis, significant design advantage and good prospects exist for their timely incorporation into the conceptual design with appropriate engineering, research and development or test verification.

The safety criteria developed shall be consistent with the Line-of-Assurance (LOA) philosophy currently being developed under the LMFBF base program.

3.7.7 Codes and Standards

Application of codes and standards shall foster the assurance of a safe, reliable, economically feasible plant and the minimization of technical risk. The selection of specific standards to apply to the individual plant systems shall be accomplished in Phase II. Standards considered necessary but not yet available shall be identified and scopes defined. Voluntary consensus standards shall be employed where applicable and RDT standards used to supplement those where appropriate or as reference standards as needed.

3.7.8 Inspection and Maintenance

Inspection and maintenance capabilities and relative effectiveness of such efforts shall be considered as a principal factor in systems design.

The overall objective shall be to include the appropriate provisions for inspection, maintenance and removal/replacement into the design of plant components and systems.

3.8 CDS PHASE I STUDY RESULTS

Phase I was concerned primarily with the engineering analyses, trade-off studies and evaluations of the principal plant and fuel systems design variables and alternatives, and selection of those designs that collectively represent the most effective plant conceptual design basis. Where appropriate, additional engineering of the selected designs was performed to provide the design definition required for the succeeding work phase. Associated tasks to establish the overall plant design requirements and criteria, preliminary R&D needs, and systems requirements for major components under development were also conducted. The Phase I work was divided into the following 13 tasks.

1. Proliferation Resistant Fuel Cycle
2. Determine Plant Size
3. Primary System Arrangement and Operating Parameters
4. Steam Cycle Selection
5. Fuel Handling
6. Reactor Containment vs. Confinement
7. Systems Design Improvements and Simplifications
8. Safety and Licensing Criteria
9. Systems Requirements for Major Components
10. Preliminary R&D Requirements
11. Evaluation of CRBR Design Experience
12. Evaluation of FFTF Design Construction and Startup Experience
13. Principal Plant Design Requirements and Criteria.

Usually several contractors worked simultaneously on each task. The Technical Integrator, Boeing Engineering and Construction, performed an independent evaluation of the principal Phase I task submittals and provided recommendations to DOE for the design basis and plant characteristics to be adopted for Phase II. The principal items adopted for Phase II are as follows.

3.8.1 Plant Size

The CDS Development Plant shall be sized for 1,000 MWe (gross) power output.

3.8.2 Plant Type

The CDS Development Plant shall be a loop type with four primary and four intermediate heat transport loops.

3.8.3 Operating Temperatures

The reactor mixed mean outlet temperature shall be 950°F.

3.8.4 Reactor Core and Fuel

The CDS Developmental Plant core design shall be based on the following major requirements:

- Plutonium-uranium oxide fuel material and uranium oxide blanket material.
- Heterogeneous core configuration
- The design envelope for the reference oxide core must allow for physical interchange with a carbide core and must provide some considerations in key areas to maintain the possibility of future carbide reload cores.

3.8.5 Fuel Handling and Storage

The CDS Development Plant fuel handling design shall be based on the following major requirements:

- Head-on refueling with the use of three rotating plugs shall be provided.
- A straight-pull In-Vessel Transfer Machine (IVTM) shall be used to transport fuel within the reactor.

- A fixed fuel transfer cell shall be used to transfer fuel between the reactor vessel and the Ex-Vessel Storage Tank (EVST).
- Fuel storage shall be outside of the reactor containment building.
- The Fuel Handling System shall provide complete safeguards, surveillance, and accountability of all fuel assemblies at all times and locations at the reactor site.

3.8.6 Containment/Confinement

The CDS Developmental Plant containment design shall provide both containment and confinement.

3.8.7 Steam Cycle

The CDS Developmental Plant shall use a combined superheated steam cycle with the following characteristics:

- Turbine throttle steam conditions shall be 850°F and 2,200 psig
- Evaporator circulation ratio shall be variable, but no greater than 2.0.

3.8.8 Shutdown Heat Removal System

The CDS Developmental Plant SHRS design shall be based on the following major requirements.

- At least two independent, diverse, safety-grade systems shall be provided. One of the systems shall be made "inherent" to the maximum extent possible.
- Each system shall satisfy the following temperature limits with one of its loops out of service.
 - Maximum fuel cladding temperature (TBD)^(a)
 - Maximum core exit bulk sodium temperature (TBD)^(a)
- The SHRS shall meet a minimum reliability goal of 1.9×10^{-6} failures/year.

(a)To be determined.

GENERAL REFERENCES

Clinch River Breeder Reactor--Preliminary Safety Analysis Report. Chapter 12, Project Management Corporation, NRC Docket No. 50-537, April 1975.

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Safety Evaluation of the Fast Flux Test Facility, Project No. 448. U.S. Atomic Energy Commission, October 31, 1972.

4.0 ENVIRONMENTAL AND SAFETY ENGINEERING CONCERNS

In this section only environmental and safety engineering aspects that are unique to Liquid Metal Fast Breeder Reactors (LMFBRs) are considered. For example, there are many environmental concerns related to construction but these concerns are similar to those for Light Water Reactors (LWRs) or fossil plants and therefore are not discussed in this review.

During the review, fifteen major areas for potential concerns were identified. For each of these areas a brief description outlines the area, tells why it could be a serious concern and indicates basically how it is being handled at the present time. The comment sections offer a more detailed view of the concern and indicate the nature of any controversy or conclusions. Selected references are listed for each area of concern. A more detailed bibliography is presented in a following section.

The material in this section is the basis for the summary and recommendations presented previously.

4.1 FISSION AND ACTIVATION PRODUCTS

The sodium, in addition to being an excellent coolant, has the ability to retain fission products released from the small number of fuel failures or defects that might occur during reactor operation.⁽¹⁾ Corrosion products from the fuel cladding and structural components also dissolve in the sodium and become activated in the reactor.

The gaseous fission radioisotopes, particularly xenon and krypton, which are not held by the sodium, escape to the argon cover gas and are removed by gas-purification and recovery systems. Some of the metals and halogens normally plate out on metal surfaces or are removed by the sodium cold trap purification system. The principal fission-product impurities remaining in the sodium are the longer lived isotopes of cesium and of other alkali metals.

The main activation material found in the primary sodium coolant system during normal operation is ^{24}Na , which has a half-life of 15 hours. Because of its short half-life, it decays within a few days and thus poses only a minor maintenance problem. An additional activation product is ^{22}Na , which, though it has a longer half-life of 2.6 years, is produced in much smaller quantities (three orders of magnitude smaller) and is less radioactive than ^{24}Na . The primary sodium system is designed to operate at a lower pressure than is the secondary system, and therefore contamination of the secondary system by leakage from the radioactive primary system is highly improbable.

The tritium formed in the fuel and control elements diffuses through the cladding into the sodium coolant. However, most of the tritium is precipitated in the primary sodium-purification system cold traps. Of the small quantity (less than 10%) of tritium that diffuses through the intermediate-heat-exchanger tube walls to the secondary sodium system, most precipitates in the secondary sodium-purification cold traps. Finally, less than 1% diffuses through the steam-generator tube walls to the steam system and is released as tritiated water in steam-generator blowdown streams. One calculation shows that a total annual release rate of tritium (gaseous and liquid) from a 1,000-MWe LMFBR is approximately 120 Ci/plant-year.⁽²⁾ In comparison, the tritium release from the 1,000-MWe reference LWR is 850 Ci/plant-year.

4.1.1 Radioactive Argon Processing System

This system removes virtually all fission products from the argon cover gas. Figure 4.1 shows a schematic diagram of the system. The argon is collected, filtered, and then compressed into a surge tank. It is cooled in a heat exchanger with liquid nitrogen and then passes through a series of cryogenic charcoal beds. These beds collect and delay the radionuclides so that they can decay. The hold-up time of the system is such that the radioactivity from all the gases except ^{85}Kr (half-life of 10.8 years) decayed to insignificant levels. After further cooling the ^{85}Kr is removed in a fractional distillation column. The ^{85}Kr is stored in bottles and the purified argon is recycled to the cover gas system. The quantity of ^{85}Kr removed is between 2,000 and 6,000 Ci/yr. It can be stored in one 50-liter gas cylinder and shipped to a waste repository designed for the long-term storage of gaseous wastes.⁽²⁾

Argon-39 is not removed by this system. It will be retained and recirculated. The only way it escapes is by leaks in seals and valves. The half-life of ^{39}Ar is 260 years. Table 4.1 shows the estimated annual quantities released by leakage to the environment compared to those released from an LWR of comparable size.

4.1.2 Cell Atmosphere Processing System

All confinement cells are blanketed with nitrogen. Therefore leaks from the primary sodium piping system and reactor housing become a part of this atmosphere.

The major radioactive species present in the inert gas-filled enclosures under normal conditions is tritium that diffuses through the primary coolant system walls. However, radioactive argon, krypton, and xenon may enter the cells through leaks in the reactor to cover gas purification system and from the fuel handling cells.

A schematic of the Cell Atmosphere Processing System is shown in Figure 4.2. Effluents from the various cells are drawn into a suction tank, through a filter and then compressed into a receiver surge tank. The gases are

FIGURE 4.1. Radioactive Argon Processing System

TABLE 4.1. Estimated Radionuclide Releases for a 1,000-MWe
LMFBR Power Plant and Reference LWR

Nuclide	Gaseous Releases (Ci/plant-yr)		Liquid Releases (Ci/plant-yr)	
	LMFBR(a)	LWR(b)	LMFBR(a)	LWR(b)
Tritium	56	580	56	270
³⁹ Ar	75	--	--	--
^{85m} Kr	1	11	--	--
⁸⁵ Kr	1	380	--	--
⁸⁷ Kr	1	2	--	--
⁸⁸ Kr	1	14	--	--
¹³³ Xe	1	7,200	--	--
Others	1	180	--	--

(a)Based on Table 4.2-2, p. 4.2-36, of Reference 2.

(b)1,000-MWe LWR, PSEID, Vol. 1.

then passed through a tritium removal system. Tritium can be removed by conversion to tritiated water by catalytic oxidation or by passing the gases over hot copper oxide. The water is then absorbed in a suitable solid such as calcium oxide, which is then handled by the solid waste system. Gases other than tritium (e.g., noble gases that leaked into the cell atmosphere) are delayed and filtered prior to release to the atmosphere. This delaying action is achieved with cryogenic charcoal beds.

4.1.3 Comments

Some tritium release will occur. The results shown in Table 4.1 lead to a decontamination factor (DF) for tritium of about 186 and thus a total release of about 120 Ci/year of tritium. Use of lower cold trap "efficiencies" of 25% instead of 73% would lead to a reduced DF (about 35) and an increased release of tritium by about a factor of 5 (about 630 Ci/year). The failure of all the

FIGURE 4.2. Cell Atmosphere Purification System

primary cold traps (or all the secondary cold traps) would lead to an increase in the tritium release of about one order of magnitude, a rate leading to release of about 1200 Ci/year if uncorrected.⁽³⁾ Recent experimental studies indicate that fast fission tritium yields are an order of magnitude greater than previously reported.⁽⁴⁾ The selection of tritium control devices is essential to keeping these emissions as low as possible. Even if an LMFBR tritium release is less than for an LWR it will still be the subject of further consideration.

The processes and components specified for the control system are conventional and proven. Some alternatives to the reactor cover gas purification system are possible. For example a series of delay tanks could be substituted for the charcoal delay beds.

References

1. Preliminary Safety and Environmental Information Document (PSEID), Liquid-Metal Fast-Breeder Reactor NASAP, Vol. VI, U.S. Department of Energy, January 1980.
2. LMFBR Program Environmental Statement. WASH 1535, Vol. 2, U.S. Atomic Energy Commission, December 1974.
3. Kabele, T. J., Parametric Study of the Tritium Distribution in the FFTF. HEDL-TME 74-06, March 1974.
4. Till, J. E., et al., Tritium - An Analysis of Key Environmental and Dosimetric Questions. ORNL/TM-6990, May 1980.

4.2 CORE-DISRUPTIVE ACCIDENTS

Any event that would cause the internal integrity of the reactor to be altered is called a core-disruptive accident (CDA). Depending on the type of accident and the final configuration of the core, the event may be energetic or non-energetic. An energetic CDA would be one in which criticality is obtained inadvertently as a result of meltdown or disruption of the fuel. Since the probability of such an event is so low, it is often referred to as a hypothetical CDA.

In the past the U.S. Nuclear Regulatory Commission (NRC) staff took the position that an LMFBR containment system should be able to withstand not only design-basis events such as sodium fires, but also the consequences of low-probability accidents.^(1,2,3) Specifically, the NRC staff took the position that the containment system should be designed to provide protection comparable to the current LWR containment systems. This resulted in the 24-hour-containment integrity requirement for the Clinch River Breeder Reactor (CRBR). Since the termination of the CRBR review in April 1977, the NRC staff completed the Fast Flux Test Facility (FFTF) review and also completed a comparative study of the radiological consequences of core-meltdown events between land-based and offshore-sited floating nuclear plants. On the basis of this study,⁽⁴⁾ the NRC staff recommended the issuance of a manufacturing license for barge-mounted plants subject to the condition that "the applicant shall replace the concrete pad beneath the reactor vessel with a pad constructed of magnesium oxide (MgO) or other equivalent refractory material that will provide increased resistance to melt-through by the reactor core in the event of a highly unlikely core-melt accident and which will not react with core-debris to form a large volume of gases. . ."(5)

For the case of FFTF, the NRC staff analysis indicated that overpressurization and the generation of hydrogen resulting from sodium and core-debris interaction with concrete are the principal challenges to containment. The quantity of hydrogen generated that could create a potentially explosive or highly energetic flammable mixture in the FFTF containment building atmosphere or portions of the building, preceded the point of threatening containment integrity by

overpressurization. Although the NRC staff in the FFTF Safety Evaluation Report,⁽⁶⁾ considered various means to alleviate the build-up of pressure and hydrogen in the containment building following postulated core-meltdown events, some of the recommended steps to deal with the problem were probably not appropriate since the construction was almost complete. For example, even though refractory materials (e.g., similar to the MgO recommended for the floating nuclear plant design), which are highly resistant to molten core debris and do not generate hydrogen, could have been used in the reactor cavity and in the containment subcavity of the FFTF, their use would have been difficult, expensive, and maybe detrimental from an overall safety viewpoint, since the cavity and subcavity are already built and sealed.

4.2.1 Comments

For future large fast-reactor designs, the approach should be to integrate the necessary features and designs in the containment system design from the start; thus, the containment will be able to withstand and mitigate not only the consequences of design-basis events but also the consequences of lower-probability, higher-consequence accidents. Major emphasis should be placed on the prevention of accidents that could lead to core melting and disruption and the subsequent loss of containment integrity. The probability of such events must be low enough to justify their exclusion from the design-basis accident spectrum.⁽⁷⁾

In an LMFBF, the low-probability accidents that represent the principal challenge to containment are associated with core melt and disruption with the potential for concurrent energy release. The energy release is a result of either core vaporization (direct core disassembly and/or recriticality), or sodium vaporization from the transfer of heat from the molten core to the sodium vaporization from the transfer of heat from the molten core to the sodium coolant. Energetics could lead to early (order of minutes) containment failure if the containment system is not designed to accommodate the generated loads; on a longer time scale, failure of the containment would occur from the evolution could involve chemical reaction products and/or sodium vapor resulting involve chemical reaction products and/or sodium vapor resulting from the inadequate postaccident decay heat removal of a molten and/or disrupted core.

This could lead to hydrogen explosions or overpressurization and/or thermal and structural degradations; either one or a combination of these can cause containment failure. Without a particular design description, it is not possible to evaluate either the potential evolution of an accident scenario and its consequences or whether it will or can be mitigated and/or contained. Containments should be designed to mitigate or to reduce significantly the consequences of core melt and disruptive accidents. From the viewpoint of the two major accident sequences (i.e., early accident energetics and longer time meltdown consequences) that can threaten containment integrity, the following should be considered.⁽⁷⁾

Accident Energetics

In the past, some LMFBR designers have relied on the primary heat transport system to accommodate the potential energetics; this was especially true for the CRBR. At the time of suspension of the CRBR licensing review, the NRC staff and applicant had not resolved the question of whether the design was adequate to accommodate the value of the energetics described in NUREG-0122.⁽⁸⁾ Other designers (e.g., the United Kingdom in the case of the commercial fast reactor [CFR] design) have considered prestressed concrete vessels with inherent capability to accommodate large energetics. The choice of a particular vessel/containment system would depend, among other things, on the requirements derived from a specific design. Some of the key consideration that influenced the selection of the level of energetics for the CRBR were:

1. The potential for large work-energy release during the "initiating phase" (direct disassembly) due to the autocatalytic, positive-sodium-void effect without the presence of the mitigating effect of timely and substantial fuel dispersion.
2. The potential for large work energy release during the "transition phase" (recriticality).
3. The many uncertainties and unknowns associated with CDA phenomena including the potential for sodium as a working fluid; fuel pin failure dynamics; freezing, plugging, and remelting of molten fuel/steel mixes; and molten pool boiling dynamics.

Areas and parameters that will influence accident scenarios and consequences for the design(s) and fuel cycle(s) are:

1. The effect of a heterogeneous core (compared to a homogeneous core such as the CRBR) on accident progressions.
2. The effect of core size.
3. The effect of fuel type such as carbides and metals vs. mixed oxides. In the area of fuel-coolant interaction the effect may be major for both carbide and metal fuels because the potential for sodium becoming a working fluid is considerably enhanced.
4. The effect of various bondings for metal and carbide fuels (either helium or sodium).
5. The effect of fuel cycle types such as Pu/Th with Th blankets vs. Pu/U with uranium blankets.
6. The effect of a pot design vs. a CRBR-type loop design.
7. The effect of design specifics such as upper fission gas plena vs. lower plena, perforated subassembly ducts, and temperature profiles across subassemblies.

Core Meltdown

As was previously mentioned, a benign (i.e., nonenergetic) core meltdown can result in hydrogen explosions, overpressurization due to sodium vapor and noncondensable gas generation, and thermal/structural degradation. All of these effects can lead separately or contribute jointly to containment failure. The FFTF containment failure, for example, was predicted to occur either from hydrogen explosions in the time interval of 10 to 20 hours, or from overpressurization in the interval of 30 to 60 hours.

Evaluations performed by the NRC staff for the CRBR and FFTF, as well as the floating nuclear plant, indicate that containment integrity can be extended substantially or even indefinitely with the addition of refractory sacrificial materials and/or cooling systems in the lower reactor cavity area. In other areas outside the reactor cavity, steel liners constructed as

engineered safety features can be used to protect the concrete from sodium attack. For both cases, the objective is to reduce or eliminate the potential for the buildup of hydrogen and other noncondensable gases, as well as sodium vapor, that can threaten the containment integrity. Areas of work that should be pursued within the framework of future large LMFB design(s) are:

1. Examine refractory sacrificial materials that are highly resistant to core melt debris and do not interact to form a large volume of gases;
2. Examine cooling systems, both active and passive, to prevent sodium from evaporating following a core meltdown and to remove decay heat from the outer extremities of the refractory material, such that containment of molten core debris can be assured;
3. Investigate methods to monitor and control the hydrogen concentration in the containment building following postulated core meltdown events; and
4. Examine means to further reduce radiological releases from containment following postulated core-meltdown events, such as the addition of sand and gravel filters.

In summary, the NRC licensing staff believes that positive and clearly identifiable actions should be taken in large fast-reactor designs to mitigate significantly the consequences of core melt and disruptive accidents.

References

1. U.S. Nuclear Regulatory Commission, Final Environmental Statement Perkins Nuclear Power Station Nos. 1, 2, & 3. NUREG-75/088, Section 3.5, October 1975.
2. Clinch River Breeder Reactor-Preliminary Safety Analysis Report. Chapter 12, Project Management Corporation, NRC Docket No. 50-537, April 1975.
3. U.S. Atomic Energy Commission, Environmental Statement: Liquid Metal Fast Breeder Reactor Demonstration Plant. WASH 1509, April 1972.
4. Liquid Pathway, Generic Study Accidental Radioactive Release to Bodies of Water from Floating and Land-Base Nuclear Power Plants. NUREG-0440, U.S. Nuclear Regulatory Commission, December 1978.

5. Final Environmental Statement. Part III, "Floating Nuclear Power Plants," NUREG-0502, U.S. Nuclear Regulatory Commission, December 1978.
6. Fast Flux Test Facility Safety Evaluation Report. U.S. Nuclear Regulatory Commission, NUREG-0358, August 1978.
7. PSEID, Liquid-Metal Fast-Breeder Reactor. DOE/NE-0003/6, NASAP, Vol. VI, U.S. Department of Energy, Washington, D.C., January 1980.
8. J. F. Meyer, L. Lois, J. L. Carter, and T. P. Speis, An Analysis and Evaluation of the Clinch River Breeder Reactor Core Disruptive Accident Energetic. NUREG-0122, U.S. Nuclear Regulatory Commission, March 1977.

4.3 SODIUM SPILLS

At the operating temperatures of LMFBs, sodium will ignite and burn readily if sprayed into the air, even in reduced-oxygen atmospheres. It will also burn as a pool. The heat released from a sodium fire can damage concrete, and water released from the heated concrete reacts exothermally with sodium. The net effect of these reactions is to increase containment-cell temperature and pressures with structural degradation of the concrete and the production of potentially explosive hydrogen. However, because of the substantial experience that has been gained in handling sodium, the NRC staff has stated that it is possible to design features in the containment system to alleviate the sodium hazards.

The general safety design criteria for the CRBR containment issued by the NRC staff state that the reactor containment structure be so designed that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate, the calculated pressure and temperature conditions resulting from normal operation, anticipated operational occurrences, and any of the postulated accidents. In an LMFB, the accidents that represent the principal challenges to containment are sodium fires coupled with potential sodium-concrete reactions resulting from failure and subsequent release of sodium from the primary heat-transport-system equipment. The general containment-system design concept that appears to meet these needs is a containment-confinement system in which the steel containment building is surrounded by a thick concrete confinement shell, with the annulus between the two maintained at a reduced pressure. An annulus filter system could be added to reduce radioactivity release from the annulus to the environment while maintaining a reduced pressure in the annulus. Examination of the CRBR containment/confinement system design and the range of conditions to which it might be subjected led the NRC staff to conclude that it is technically feasible to implement design features to meet their stated criteria.

4.3.1 Small Sodium Fires⁽¹⁾

Two types of sodium circuits are used in LMFBs: 1) primary coolant circuits that circulate sodium through the reactor core and accumulate

significant amounts of radioactivity in the process, and 2) intermediate coolant circuits that are entirely outside the core and contain little or no radioactive materials.

The principal source of activity in the primary sodium is from neutron activation of the sodium itself. Sodium-24 (15-hr half-life) and smaller concentrations of Sodium-22 (2.6-year half-life) are formed. Depending on prior plant operations, significant levels of corrosion products may also be present.

Because of their relatively high activity, the primary coolant circuits are located within cells that contain an inert gas atmosphere. Consequently any leaks in the primary system will be into a confined region that, in turn, is housed in a low-leakage containment. The sodium in the secondary coolant circuit is not normally radioactive except for tritium that has diffused through the intermediate heat exchanger tubing. No potential radiation hazard of any consequence would result from leaks of secondary sodium. However, because of the potential for extensive facility damage in the event of a large sodium leak and because of danger to personnel from caustic fumes, equipment is provided to detect any leaks that might occur in either system. Because of the caustic nature of sodium combustion products, specific systems must be provided to assure that fires involving nonradioactive sodium are controlled and would not, of themselves, create significant risk to personnel and the public.

Experience to date has shown that one or more leaks, generally small in extent, may occur during a plant lifetime. In general, the failures can be classified as nuisance, minor, or major leaks. The nuisance leaks have been very small with no attendant fire or damage to adjacent equipment. Minor leaks have been moderate in magnitude and accompanied by small but controllable fires and light-to-moderate damage to adjacent equipment. Major leaks are spills of several hundred pounds which result in large intense fires and heavy equipment damage. Of the 80 leaks reported, 69 were nuisance types, 9 were minor, and 2 were major. Of the two major leaks, one caused extensive damage and the other light-to-moderate damage.

The smallest leaks observed were approximately 1 lb in magnitude and accounted for 79% of the entire leak population. The major portion (82%) of these leaks were caused by valve failures. The largest leak was approximately 600 lb. It occurred in a pipe section and was caused by human error. The second largest leak was about 400 lb and was the result of tube failure in a gas-fired sodium heater.

The construction of most sodium systems gives rise to the possibility that small leaks may develop which may not be immediately detected. Small leaks behind pipe insulation, within valve bonnets, and from systems within inert enclosures are some examples of this type of leak. If undetected, such leaks have the potential for progressing and causing significant damage. Leaks in secondary coolant system can produce hot sodium/air reaction products (oxide, peroxide, hydroxide, etc.) that in turn may cause rapid deterioration of austenitic steels and other metals or alloys used for sodium systems. Thus, in the presence of air, sodium from an initially small leak will create a very serious corrosion condition on the exterior surface of a metallic system. Similarly, flanges and flange bolts may also be seriously corroded by sodium from a small leak. Small leaks can develop into large leaks, the rate of growth of the leak depending on the original leak size, details of the container (such as thickness), system pressure, temperature, and the availability of oxygen and probably moisture. The early detection of small leaks is, therefore, very important in minimizing the consequences of the leak in both primary and secondary coolant systems. If a leak can be detected while it is still small, system pressure can be reduced and the sodium can be drained from the vicinity of the leak or it can be frozen; therefore, damage to the facility can be held to a minimum. This area receives special attention in plant design. Further research and development efforts are directed at devising new or improved methods for sensing and controlling sodium leaks.

4.3.2 Large Sodium Fires⁽¹⁾

Massive sodium leaks or spills (many thousands of pounds) are not expected to occur. However, the consequences of such an event are commonly evaluated as a test or measure of the capability of containment systems to accommodate a sodium fire.

The sequence of events typically postulated in defining the consequences of sodium fire are:

1. The reactor has been shut down for several days to permit decay of ^{24}Na activity, before opening a primary cell(s) for maintenance operations.
2. The cell is opened, and its inert atmosphere is replaced with air.
3. A failure, such as operator error, is then assumed which causes the loop sodium to drain onto the cell floor where it begins to burn.

The cells are typically lined with steel to prevent possible sodium-concrete reactions. Calculations based on such a sequence of events indicate peak pressures of a few psig and temperatures of 250°F. These values are well within typical containment building specifications.

Spills may be postulated to take place in closed cells; in this instance the low oxygen content (1 to 2%) will prevent rapid or complete combustion of the sodium. If the spill does not involve a large spray component, the resultant pressure-temperature transient does not impose a significant load on the cell (10 psig, gas temperature of a few hundred degrees). If, however, the sodium is ejected under pump pressure and causes splashing or fine dispersal of much of the sodium, somewhat higher pressures and gas temperatures could result. The exact conditions would probably be much less than the theoretical value (~30 psig and 1050°F); the specific value would be design specific and depend on a number of values such as cell geometry, break location, duration of spray volume coverage, and gas-liquid heat transfer. In addition to the option of providing cell designs of suitably high pressure and temperature capability, there are a number of design options that might involve, for example, deflectors or curtains to partially isolate and contain the discharged sodium. Other options include limited spill areas, sloped floors with sumps or gratings, and pressure relief to other volumes. Work is under way to develop an improved experimental and analytical basis for assessing the consequences of potential sodium fires and design features to mitigate such events.

Assessments of the response of such containment systems (including consideration of the pressure transient from the fire, leakage that may occur prior to containment isolation, action of filtration systems, and long-term leakage following the accident) indicate radiation doses (due to radioactive sodium and contaminants that become airborne as a result of the fire) at and beyond the site boundary which are well within applicable guideline values. A 2-hr, 2000-ft whole-body dose of 12 millirems and a 30-day, 2-mile whole-body dose of 4.5 millirems were estimated for a 1000-MWe LMFBR.

4.3.3 Comments

The accidents that represent the principal challenges to containment are sodium fires coupled with potential sodium-concrete reactions that result from failure of pipes and vessels containing sodium and the subsequent release of the sodium. Following sodium release, combustion with oxygen (even for those areas which are inerted) will result in increasing pressures and temperatures. The specific initiating events, as well as consequences, will be very system dependent. Based on the review of the CRBR and the FFTF, the sodium releases were based on a spectrum of postulated component and piping failures of different size, locations, and other properties sufficient to provide assurance that the entire spectrum of postulated sodium fire accidents is covered. Some of the specific challenges to the containment presented by sodium release accidents that should be considered in a containment design are as follows:⁽²⁾

- Mechanical. The deterioration of concrete by sodium can weaken structures, cause cracking, and enlarge leak paths; therefore, means should be used to prevent or reduce the likelihood of direct contact between sodium and concrete. For the FFTF and CRBR, cell liners were used to accomplish this.
- Thermal. The chemical heat of sodium reactions with oxygen or concrete can build up pressures within inerted cells or the containment building which must be included as part of the containment design basis.

- Explosive. The generation of hydrogen from reactions between sodium and water (or concrete) can lead to explosive mixtures in the air atmospheres of the reactor containment building; therefore, water should be kept to a minimum in buildings containing large amounts of sodium. Hydrogen recombiners are provided in LWRs to control hydrogen. For LMFBRs (the FFTF and the CRBR), the applicants have claimed in the past that the presence of sodium oxide has a catalytic effect in promoting recombination of hydrogen and oxygen and in keeping the hydrogen concentration below the explosive limit. Based on the available information, the NRC staff has previously been unwilling to accept the view that hydrogen can be depended upon to burn benignly under the natural processes associated with these accidents.
- Nonradiological toxicity. If released from containment or the steam generator building, large quantities of nonradioactive sodium could be used to suppress or extinguish sodium fires; isolation can prevent the release of the hazardous smoke.
- Filters. The dense smoke from sodium fires can rapidly plug ventilation filters. Scrubbers or prefilters are generally required to eliminate this problem.

In recognition of the above, the following general design criteria was issued for the CRBR, "the reactor containment structure, including access opening and penetrations, and if necessary, in conjunction with additional post-accident heat removal systems including ex-vessel systems, shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate, and with sufficient margin, the calculated pressure and temperature conditions resulting from normal operation, anticipated operational occurrences and any of the postulated accidents."

References

1. LMFBR Program Environmental Statement. U.S. Atomic Energy Commission, WASH-1535, Vol. II, December 1974.
2. PSEID Liquid-Metal Fast-Breeder Reactor. NASAP, Vol. VI, DOE/NE/0003/6, U.S. Department of Energy, January 1980.

4.4 SODIUM PURIFICATION

Experience gained from operation of LMFBRS shows that high levels of activity will accumulate on components in the primary circuit as a result of deposition of activated corrosion products, fission products and fuel particles from the liquid-sodium coolant. It is desirable to minimize this contamination by reducing the levels of these active materials in the coolant. Few methods of sodium purification have been developed; the method used universally is cold trapping.

The principal role of the cold trap in the primary circuit is to control the concentration of dissolved oxygen impurity by precipitation in the form of disodium monoxide, Na_2O . Hence acceptable levels of corrosion of steels and related release of corrosion products is maintained. However these traps also remove various fission and activation products. They are removed by the processes of precipitation, absorption, chemical reaction, or diffusion into the solid surface after deposition. All of these mechanisms can occur depending on the species and the environmental conditions in cold traps.

Fission products of volatile elements such as Cs, Rb, I, Te, Ag and Sb are easily dissolved in sodium. Radioactive corrosion products are major sources for system contamination. They can be formed by: 1) transport of non-radioactive products--mainly from the heat exchanger--to the core and activation, and 2) corrosion of activated cladding and other core structures materials. The most important radioactive corrosion product nuclides are ^{51}Cr , ^{54}Mn , ^{58}Co , ^{59}Fe , ^{60}Co and ^{182}Ta . (1)

4.4.1 Comments

In the operating LMFBRS, radionuclides have not been a serious problem for operation. On the other hand, the potential role of cold trapping as a means of removing radioactive products from the coolant has not been explored in depth. Few systematic studies have been made on cold trapping performance. Therefore, it is difficult to generalize about its performance. For example, ^{137}Cs is one of the most important isotopes to be considered yet the conditions to optimize its removal have not been identified. Another example, iodine, reacts to form sodium iodine but precipitation at cold trap

temperatures cannot be used for ^{131}I removal because concentration levels above the solubility level are not likely to be attained. Iodine appears to be amenable to cold trapping but by adsorption rather than precipitation.

Since some species are not effectively removed in cold traps by precipitation some researchers have proposed using radionuclide traps.^(2,3) The device uses a metallic element or alloy that acts as a collector for the undesirable radioactive component. For example, pure nickel has been found to be superior for removing ^{54}Mn and it is quite effective for removing ^{58}Co and ^{60}Co . Such a device has been tested on EBR-II, and it will be used in a FFTF test assembly.

References

1. Feuerstein, H. A. J. Hooper and F. A. Johnson, "Mechanism of Release of Radioactive Products into Liquid-Metal Coolant, Their Transport Within the Circuits and Removal from LMFBRs." Atomic Energy Review, Vol. 173, 1979.
2. McGuire, J. C. and W. F. Brehm, "A Radionuclide Trap for Liquid-Metal Cooled Reactors." Transaction of the American Nuclear Society, Vol. 30, p. 189, 1978.
3. McGuire, J. C., and W. F. Brehm, Progress in Radionuclide Trap Development. Report HEDL-TME 77-85, Hanford Engineering Development Laboratory, Richland, Washington, 1977.

4.5 RADIATION EXPOSURE

The reactor is the effective source of all radiation. The reactor cavity walls and walls of the surrounding cells serve to reduce the dose rate in the personnel access areas. The sodium coolant is also an extremely high radiation source during operation. Sodium-24 and sodium-22 are formed by activation in the reactor. Shielding is therefore required for the primary sodium loop. After the decay of the sodium coolant following shutdown, corrosion products and fission products are the primary source of personnel exposure.

The radiation protection design features help maintain occupational radiation exposures as low as is reasonably achievable.^(1,2) To minimize radiation doses associated with maintenance work, all major pieces of process equipment are located in individual rooms with individual access through labyrinthed entranceways from low radiation areas. Fluid transfer facilities are designed for easy flushing and internal cleaning. Double barriers (e.g., pipe inside a pipe) and shielding, are used for transfer lines passing through normal work areas. Radiation exposure and radioactive contamination is controlled during the transfer of process materials or maintenance of equipment. Rough surfaces, square corners, cracks or crevices, and absorbent materials are avoided in the design of hoods, cells, gloveboxes, and process equipment. Rough surfaces, square corners, cracks or crevices, and absorbent materials are avoided in the design of hoods, cells, gloveboxes, and process equipment to prevent crud buildup. To facilitate decontamination and control of atmospheric purity, selected cells containing radioactive equipment are lined with gas tight stainless steel liners and equipment with rinse systems. The hot cell complex is well shielded and houses remotely operated equipment for the performance of nondestructive examination of core components and maintenance of reactor plant equipment. Other design features intended to complement the "as low as is reasonably achievable" program include: 1) use of emergency lighting designed to provide adequate illumination after loss of preferred and emergency power system supplies; 2) routing of all piping carrying radioactive material to minimize personnel exposures; and 3) clear identification of radiation hazards using warning lights and/or authorized signs.

The radiation shielding is designed to provide protection against radiation for operating personnel both inside and outside the plant, and for the general public. Shielding is also used where it is necessary to protect materials and equipment for excessive radiation damage. The primary heat transport system components are located in individual cells to maximize the maintenance and operation flexibility while minimizing doses to personnel. The cells in the containment building are arranged so that the reactor and high activity cells are at the center and the lower radiation and source-free cells are located around the perimeter of the building. This assures low radiation levels outside of the containment structure.

4.5.1 Comments

On the basis of information compiled by NRC on past experience from operating nuclear power plants, it is estimated that the average collective dose to all onsite personnel at a 1,000-MWe LWR plant will be approximately 250 man-rem/plant-yr⁽³⁾ to 450 man-rem/plant yr.⁽⁴⁾ Although no directly relevant operating experience is available for a large LMFBR plant, an evaluation of the yearly exposure using the design parameters for the CRBR indicates an exposure using the design parameters for the CRBR indicates an exposure of approximately 280 man-rem/plant-yr.⁽³⁾ On this basis, the exposure level will be comparable to that of a current LWR plant. The plant system designs and operational and maintenance procedures are such that the radiation protection afforded to plant personnel in a commercial LMFBR plant will be consistent with the requirements of 10 CFR 20.

Doses to plant personnel are influenced by many variables, including the following:

1. The ability of fuel elements to retain fission products
2. The extent of deposition of activated corrosion products throughout the primary and auxiliary coolant systems
3. The plant layout
4. Operational and maintenance procedures

5. In-service inspection procedures
6. Radiation protection programs

However, a major portion of the radiation exposure of plant personnel is received during maintenance, radwaste handling, in-service inspection, refueling, and nonroutine operations.

In addition to the fission products, the erosion and corrosion products that become mobile and are activated constitute perhaps the principal source of radiation with respect to the exposures of plant personnel. Specific radionuclides that have been identified in crud in LWR plants are ^{58}Co , ^{60}Co , ^{54}Mn , ^{65}Zn , and ^{95}Zr .⁽⁵⁾ Similar nuclides are expected to be present in LMFBR coolant and loop components.

The LMFBR fuel is assumed to be spiked by pre-irradiation or by the addition of a small quantity of cobalt-60. The resultant radiation level would be high enough to prohibit handling the fuel without substantial shielding. The spikant would not affect plant operation or fuel characteristics during and after fuel irradiation.⁽⁶⁾

The effects of the radioactive spike (or pre-irradiated) fresh fuel will be to increase the occupational dose to plant personnel during fresh-fuel handling and refueling operations. Since these fuel-handling operations are performed by automated remote-control systems, the actual incremental dose will be relatively small and will depend on the shielding designs for the fuel-handling system and the refueling procedures at a given plant. In the absence of actual data, it is expected that the dose for the operating personnel during fuel-handling operation for the spiked fuel will be, at worst, twice that of the non-spiked-fuel case. However, the fraction of the dose received by plant personnel during fuel-handling operations is only 4% (or 11 man-rem/yr) of the total dose received.⁽³⁾ Therefore, the occupational exposure in a commercial LMFBR plant that uses spiked fuel may be about 22 man-rem/yr during fuel-handling operation. Thus, the total occupational exposure will be approximately 300 man-rem/plant-yr.

References

1. FFTF Safety Evaluation Report. NUREG-0358, Project 448, U.S. Nuclear Regulatory Commission, Washington, D.C., August 1978.
2. FFTF Final Safety Analysis Report. U.S. Energy Research & Development Administration, HEDL Contract No. AT(45-1)-2170, December 1975.
3. Clinch River Breeder Reactor - Preliminary Safety Analysis Report. Chapter 12, Project Management Corporation, NRC Document 50-537, April 1975.
4. Final Environmental Statement Perkins Nuclear Power Station No. 1, 2, & 3. NUREG-75/088, Section 3.5, U.S. Nuclear Regulatory Commission, October 1975.
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4.6 SODIUM/WATER REACTIONS

Sodium reacts with water to produce hydrogen, a potential hazard for explosions. Engineering design features are employed to minimize the possibility of leakage in steam generators that could result in reactions between sodium and water.⁽¹⁾ The water and steam in the steam generation system interface only with nonradioactive secondary sodium coolant. Leak detection and early sodium-water/steam reaction warning systems are designed to detect a leak early enough so that damage to the steam system and other secondary system components would be minimized by prompt plant shutdown and water or sodium dumping. Steam generator models have been tested in both sodium and water environments to assess leakage mechanisms and the extent of damage. Nevertheless, some leaks may occur, and provisions are made to relieve the pressures resulting from sodium-water/steam reactions by venting through rupture disks. The reaction product relief system is designed to separate and retain the sodium and most of the particulate reaction products and to vent the hydrogen to the atmosphere.

The protection system detects leaks through hydrogen or oxygen detection or by an overpressure in the intermediate sodium system. When a large reaction occurs, the system vents the sodium system of the affected loop to a sodium-water reaction products tank and shuts off the affected sodium loop. The plant is designed so that any pressure loadings from sodium-water reactions will not fail the intermediate heat exchange. In this way, no radioactive sodium is released and heat removal ability is retained.

4.6.1 Comments

Spontaneous hydrogen explosions do not take place unless the reactants are present in certain specific proportions and other conditions prevail. Following a sodium-water/steam reaction, should one occur, there would be little oxygen present, and several other reaction products would be present in addition to hydrogen. Thus, adequate equipment and procedures are available to prevent damage from a potential sodium-water/steam reaction.

References

1. LMFBR Program Environmental Statement. U.S. Atomic Energy Commission, WASH-1535, Volume II, December 1974.

4.7 SODIUM/CONCRETE REACTIONS

When molten sodium is spilled on concrete, water vapor is emitted. In subsequent interactions, hydrogen is generated. The hydrogen poses a serious problem if concentrations build up to explosive levels. The most serious sequence for this to happen is the core meltdown. In the event of a core melt-through, the quantity of hydrogen generated could create potentially explosive or highly energetic flammable mixtures in the containment building atmosphere or portions of the building, which could constitute a significant early threat to the integrity of the containment structure.⁽¹⁾

This potential problem has been recognized and it is the principal reason that a steel liner is used in containment vessels. In addition many studies have been conducted to determine the sodium concrete penetration rate. Calculations have been made to estimate the temperature and pressure used in a containment vessel in the event of a meltdown. These calculations typically assume the natural recombination of hydrogen in the presence of air and sodium.

4.7.1 Comments

The NRC reviewed this situation for FFTF⁽¹⁾ and determined that the generation of hydrogen was a principal challenge to containment which has not been fully resolved. They concluded that the facility should provide the necessary means to monitor and control hydrogen concentration in the containment under those conditions which the hydrogen concentration could threaten containment integrity. The following control means were listed for consideration and review:

- Sacrificial Materials - High-temperature sacrificial materials which do not generate hydrogen and which protect the liner and concrete from sodium and core debris could be added: 1) in the reactor cavity to protect the steel liner and 2) in the containment subcavity to protect the bare concrete floor and side walls. Because the reactor cavity and subcavity have been sealed, installation of additional materials in these regions would be difficult and expensive; however, it is not impossible prior to reactor startup.

- Spontaneous Recombination - Based on very small-scale tests reported in Reference 2, it is concluded that hydrogen will recombine spontaneously with oxygen provided the criteria of Reference 2 are satisfied. However, the recombination of hydrogen and oxygen in the presence of sodium aerosols has not been demonstrated for the large containment volume and the possible stratified conditions which may exist there.
- Catalytic Recombination - Recombining hydrogen catalytically at low rates of energy release is not considered to be feasible in the presence of an atmosphere that may be heavily loaded with sodium reaction products.
- Inert Gas - Dilution of the containment building atmosphere with an inert gas during a core melt accident is feasible but somewhat impracticable because of the large containment volume.
- Artificial Combustion - Inducing combustion of hydrogen artificially while it is at low enough concentration to avoid a serious detonation may alleviate this problem.
- Containment Subcavity - Eliminating the containment subcavity void by filling it with either concrete or, as mentioned earlier, a sacrificial material would mitigate and delay the significant containment response transient which occurs after the cavity floor is penetrated and the sodium and core debris come in direct contact with the large bare concrete surfaces of the subcavity. As has already been mentioned, installation of additional material in the subcavity would be difficult and expensive, although it is feasible.

References

1. FFTF Safety Evaluation Report. NUREG-0358, U.S. Nuclear Regulatory Commission, Project 448, Washington, D.C., August 1978.
2. Experimental Study of Hydrogen Formation and Recombination Under Postulated LMFBR Accident Conditions. HEDL-TC-730, Hanford Engineering Development Laboratory, Richland, Washington, December 1976.

4.8 HEAT TRANSPORT SYSTEMS

The integrity of the heat transport system is essential to reactor safety.⁽¹⁾ This system has the function of removing the heat generated by the reactor core and converting it to the rotational mechanical energy required by the generator to produce electrical power. It consists of multiple parallel circuits. Each circuit is comprised of three subsystems connected in series: 1) a radioactive primary system, 2) a secondary non-radioactive sodium system and 3) the steam system. The primary coolant system consists of several redundant circulating loops that conduct sodium from the core exit plenum of the reactor vessel and circulate it through intermediate heat exchangers wherein the heat is transferred to the sodium of the secondary coolant system. The primary sodium then returns to the reactor vessel. In the secondary system, secondary sodium is heated in the intermediate heat exchangers and is circulated to the steam generation system. Three parallel primary loops and three secondary loops, one serving each primary loop, are typically used, although six primary loops have been included in some concepts.

Potential heat transport system faults include: 1) loss of offsite power and consequent loss of full-power pumping capability, 2) mechanical difficulty with one or more of the main circuit pumps, 3) loss of main heat sinks through faults in the steam generators, the turbine generator, or the main condenser, and 4) rupture of a main line coolant pipe. Various design features can be provided to assure that an LMFBR can tolerate any of these faults without serious damage or significant release of radioactive material to the environment. Pony motors, powered by standby diesel generators, typically are used on the main coolant pumps for decay heat removal in the event of loss of offsite power. Further, elevated circuits typically are employed to ensure flow in the total absence of pumping power or for any other fault that might incapacitate the main pumps. Loss of the heat sink is made unlikely through the use of multiple loops, steam dump capability, and, in some designs, auxiliary systems that have some measure of independence from the main heat transport systems. Consideration of rupture of the main coolant line has led to several standard LMFBR design features.

Because of the low pressure, subcooled character of LMFBRs, coolant blowdown characteristic of a postulated LWR pipe rupture cannot occur. Rather, the potential problem is one of a draindown or pump-out condition. Elevation and/or containment of the system by guard tanks can, therefore, be employed, in combination with tripping the pumps, to limit the loss of coolant inventory. By restriction the loss of sodium inventory such that submergence of the reactor vessel main coolant nozzles is maintained, the other unaffected primary loops and associated primary sodium system components remain operable as redundant emergency and long-term decay heat removal systems. The flow provided by the coastdown of the pumps, following the reactor and pump trips, cools the core in the period immediately following a system rupture.

Because the reactor vessel outlet nozzles must remain covered after an accident, the elevation or containment of the remainder of the system is considered in relationship to those nozzles. All major components and directly attached piping are contained in guard vessels that provide double containment to an elevation above the reactor vessel outlet nozzles so that the pumps, when operating at pony motor speed, cannot pump sodium down to a level below the top of the reactor vessel outlet nozzles. The primary sodium piping, except for the connections to the components within the guard vessels, is installed above this minimum elevation without additional containment. The guard vessels are sized to prevent the sodium level in the vessel from dropping below the top of the outlet nozzle elevation.

Analyses of large pipe ruptures indicate that an LMFBR can respond to a large range of ruptures without impairing fuel capability or leading to significant core damage. Rupture downstream of a primary pump raises the possibility of pump-out of the sodium inventory in the reactor vessel over the reactor outlet nozzles and thereby a loss of suction in the unfaulted loops. This situation is prevented by prompt tripping of the pumps. Rupture at or near the reactor inlet nozzles would appear to lead to the greatest core-bypass flow and consequent core flow reduction. Additional protection against core damage can be provided by appropriate resistance to backflow

resistance to backflow at this location, if necessary, or through development of improved capability for rapid detection of leaks and reactor shutdown.

4.8.1 Comments

The sodium inventory for a 1000-MWe plant is about 4,500,000 lbs and coolant flow rates are 70,000-80,000 gpm.⁽²⁾ These figures are staggering and by themselves command a prudent approach. However, the potential faults that might occur seem to have been considered and covered by LMFBR design-safety considerations. Reliable systems are provided to sense heat removal faults and to provide for reactor shutdown and continued removal of fission-product decay heat.

References

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2. LMFBR Conceptual Plant Design: 1000 MWe. Rockwell International/Burns & Roe, Report TID-27701-2, Vol. II, May 1977.

4.9 PLUTONIUM TOXICITY

LMFBR fuel is composed of plutonium-uranium oxides. The fuel is about 20% plutonium dioxide by weight. A 1000-MWe Reactor would contain over 4000 kg of plutonium.

The situation is unusual in that large quantities of toxic material are present but under circumstances that containment can be rigorously insured.⁽¹⁾ The half-life is very long so risks persist for over a hundred thousand years. Also, although the toxicity is well demonstrated in experimental animals, there is no direct knowledge of effects in man.

A major issue regarding the health hazards of plutonium was raised during the review of the Proposed Final Environmental Statement (PFES) for the LMFBR Program.⁽²⁾ It centered about the so-called "hot particle" hypothesis which in essence postulates that the procedure of assessing plutonium health effect based on average organ doses is in error and that the health effects might be several orders of magnitude greater since the exposure is concentrated in limited areas of the lung in the vicinity of the "hot particles" with a much greater probability of cancer incidence than predicted using an average lung dose. In the opinion of the ERDA staff the evidence is overwhelmingly against the "hot particles" hypothesis and material attesting to this was presented at the Public Hearing on the PFES held on May 27-28, 1975.

"The outstanding issue is whether the hot particle hypothesis should be assumed as an additional degree of conservatism in projecting health effects from inhaled plutonium. In the judgment of the Board, (Internal Review Board), this dispute turns upon peculiarly recondite matters of health physics and cannot be resolved within the confines of an environmental impact statement. It must await the verdict of the scientific community. The conclusions of the PFES appear to be based upon the considerable weight of current informed opinion and are therefore as adequate for decision-making as the state of the art will allow."

An additional deterrent to human exposure is the extensive safeguards program. This program is aimed at eliminating illegal diversion and use of fuel materials for making explosives or creating radiological incidents.

4.9.1 Comments

A core-disruptive accident presents the greatest possibility for plutonium exposure to people. NRC has used a series of computer codes to assess the radiological effects of meltdown accidents for FFTF.⁽³⁾

Different approaches have been used to estimate the release of fuel from a pool of sodium. One method considers it as a molten material, subject to sparging by the boiling sodium vapors. Another approach is to regard the fuel as a fractured solid material but with only a small percent of its mass in the form of particles less than a few microns in diameter. NRC staff assumed a particle size distribution which places less than 10 percent of the plutonium in the dispersible particle size range. A part of these particles form sodium plutonium compounds and are removed by the pool. In essence, NRC staff assumed in its calculations that 0.1% of the core plutonium was released to the aerosol in the containment building at the end of the sodium boilup phase. This material is subject to fallout with the sodium oxide aerosol.

The radioactive dose results for an individual at the exclusion radius or at the low population zone are given in Table 4.2. Although the calculation is believed to cover an extremely unlikely combination of circumstances associated with an accident at FFTF, the consequences are not as severe as might have been expected. This is because the fallout within containment is a dominating effect during the time when the suspended radioactivity is at its highest. Eighty percent of the large doses from plutonium and solids results from the first half-hour after the major release from the pool at 24 hours. This suggests that temporary interdiction measures could be very effective in protecting downwind populations. Table 4.2 also shows the low-level doses that would be produced if no venting were required, or if venting could be delayed until 16 hours after the major release of fission products from the pool (40 hours from the start of the accident).

TABLE 4.2. FFTF Meltdown Accident Doses (rem)

Radionuclide	No Venting		Vent After 24 Hours	Vent After 40 Hours
	2 hr Excl. Rad	30 d LPZ(a)	4000 cfm 30 d LPZ(a)	10,000 cfm 30 d LPZ(a)
I - thyroid	10	3	180	3
Solids - bone	negligible	.03	27	.11
Solids - lung	negligible	.04	44	.17
Pu - bone	negligible	.03	45.0	.16
Pu - lung	negligible	.001	1.8	.006
Noble gases - WB ^(b)	.06	.04	2.6	1.7
- skin	.13	.12	12	8.8

Assumptions:

100 percent of noble gases released to RCB at time zero.

25 percent of iodines and 1 percent of solid fission products released to RCB at time zero.

Leak rate is 0.1 percent per day, until vent time.

At 24 hours, 1 percent of the remaining fission products and 0.1 percent of plutonium released to Reactor Containment Building (RCB).

At vent time, leak rate is increased to 400 percent per day (4000 cfm) or larger.

All solids (including halogens) undergo fallout within the RCB, along with Na oxides.
No fallout outside containment

Calculations show that 80 percent of the doses associated with the 24-hour venting result from the material vented in the first half-hour of venting.

(a)LPZ = outer boundary of Low Population Zone.

(b)WB = Whole Body.

The release fractions used here are comparable to those used in the analysis of light water reactors for similar low probability accidents, and the fission product inventories are lower for FFTF than for power reactors by a factor of 8. The mass of plutonium in FFTF is comparable to that in a light water moderated power reactor at the end of a cycle.

NRC staff concludes that the consequences of an FFTF core melt accident can be held to as a low level of probability as that of other reactors, provided that there is adequate management of releases from the containment, and preparations for these accidents are kept in readiness.

References

1. LMFBR Proposal Final Environmental Statement. U.S. Atomic Energy Commission, WASH-1535, Vol. II, Washington, D.C., December 1974.

2. LMFBR Final Environmental Statement. U.S. Energy Research & Development Administration, ERDA-1535, Vol. I, Washington, D.C., December 1975.
3. FFTF Safety Evaluation Report. U.S. Nuclear Regulatory Commission, NUREG-0358, Project No. 448, Washington, D.C., August 1978.

4.10 LICENSING

The 200-MWt Enrico Fermi Atomic Power Plant was a licensed reactor.⁽¹⁾
The 400-MWt government-owned FFTF was evaluated by the NRC staff and the Advisory Committee for Reactor Safeguards and found acceptable for construction.⁽³⁾
On the basis of this evaluation, it has been concluded that the FFTF can be operated safely as a test reactor with reasonable assurance that the health and safety of the public will not be endangered. A license application to construct a 975-MWt demonstration reactor, the Clinch River Breeder Reactor, had been under review by the NRC until the public hearings were indefinitely suspended.

The licensing evaluation of the CRBR conducted so far⁽⁴⁻⁶⁾ indicates that the state of technology and experience would result in a safe design, but the evaluation is incomplete and the particular design proposed has not yet been found satisfactory. The major concerns that have not been resolved include instrumentation to detect core abnormalities that could lead to accidents, inspection of the reactor system to provide continued confidence in the system integrity, reliability of the decay-heat-removal system, and containment design to withstand the consequences of low-probability accidents. All of these topics are being addressed in the United States and other countries.

Alternative fuel cycles have an effect on the plant design, safety, licensing. Of significant importance for licensing is the demonstrated technology; this is presently concentrated on the mixed oxides or uranium and plutonium, including most of the safety experiments and analysis models, but could be applicable to other oxide systems as well.⁽⁷⁾

Regarding containment system design, in the past the NRC staff took the position that it should be able to withstand not only design-basis events such as sodium fires, but also the consequences of low-probability or Class 9 accidents.⁽⁸⁻¹⁰⁾ Specifically, for the case of CRBR, the staff took the position that the containment system should be protected from the effects of low-probability accidents (commonly referred to as core-disruptive accidents

in LMFBRs or CDAs) such that, the inherent protection of LWR containment systems compared to core-melt events. This resulted in the 24-hour-containment integrity requirement for the CRBR which can be found in References 8-10. Since the termination of the CRBR review in April 1977, the staff completed the FFTF review and also completed a comparative study of the radiological consequences of core-meltdown event between land-based and offshore-sited floating nuclear plants. On the basis of this study,⁽¹¹⁾ the staff recommended the issuance of a manufacturing license for barge-mounted plants subject to the condition that "the applicant shall replace the concrete pad beneath the reactor vessel with a pad constructed of magnesium oxide (MgO) or other equivalent refractory material that will provide increased resistance to melt-through by the reactor core in the event of a highly unlikely core-melt accident and which will not react with core-debris to form a large volume of gases. . ."⁽¹²⁾

For the case of FFTF, the NRC staff analysis indicated that overpressurization and the generation of hydrogen resulting from sodium and core-debris interaction with concrete are the principal challenges to containment. The quantity of hydrogen generated that could create a potentially explosive or highly energetic flammable mixture in the FFTF containment building atmosphere, or portions of the building, preceded the point of threatening containment integrity by overpressurization. Although the staff in the FFTF Safety Evaluation Report (SER)⁽³⁾ considered various means to alleviate the buildup of pressure and hydrogen in the containment building following postulated core-meltdown events, some of the recommended steps to deal with the problem were probably not appropriate in view of the facility being essentially constructed. For example, refractory materials (e.g., similar to the MgO recommended to the floating nuclear plant design) which are highly resistant to molten core debris and do not generate hydrogen could have been used in the reactor cavity and in the containment subcavity of the FFTF. However, their use would have been difficult, expensive, and maybe detrimental from an overall safety viewpoint, since the cavity and subcavity were already built and sealed.

According to NRC, future large fast-reactor designs should integrate the necessary features and designs in the containment system design from the start; thus, the containment will be able to withstand and mitigate not only the consequences of design-basis events but also the consequences of lower-probability, higher-consequence accidents. Accordingly, three broad classes of accidents should be considered in designing large fast-reactor containment: 1) those postulated accidents considered in the design basis of plants (i.e., 10 CFR 50); 2) hazards not exceed by those from any accidents considered credible (i.e., 10 CFR 100 of Site Suitability Source Term); and 3) low-probability or Class 9 accidents.

The PLBR staff reviewed the NRC regulatory guides. They concluded that an LMFBF would have to comply with 105 of the 124 guides they reviewed. Those guides not applicable, applied specifically to PWRs or BWRs. Similarly it is anticipated that some guides will be promulgated that apply only to LMFBFs.

4.10.1 Comments

In the past, the NRC staff and the DOE differed in basic safety approach and implementation for both the CRBR and the FFTF reactors. These differences have been documented in great detail for the CRBR and the FFTF in correspondence between the staff and projects, and in the CRBR site-suitability report and final environmental statement (FES) and in the FFTF safety evaluation report (SER). It is imperative that the DOE recognize and understand these differences and factor them into their overall planning, in particular in their formulation of, and commitment to, a much-needed safety R&D program. Anything short of this could have serious implications for licensing. The licensing of commercial-size LMFBFs is difficult to predict, particularly until the technology has been demonstrated on a large scale.

References

1. Power Reactor Development Company, Enrico Fermi Atomic Power Plant, Technical Information and Hazards Summary Report. Part B, Revised License Application, AEC Docket No. F-16.
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3. U.S. Nuclear Regulatory Commission, Safety Evaluation Report Related to Operation of Fast Flux Test Facility. Project No. 448, NUREG-0538, August 1978.
4. Site Suitability Report by the Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission in the Matter of the Clinch River Breeder Reactor Plant. Energy Research and Development Administration, Tennessee Valley Authority, Project Management Corporation, Docket No. 50-537, March 4, 1977.
5. U.S. Nuclear Regulatory Commission, Final Environmental Statement Related to Construction and Operation of Clinch River Breeder Reactor Plant. NUREG-0139, Project Management Corporation, Tennessee Valley Authority, and Energy Research and Development Administration, Docket No. 50-537, February 1977.
6. J. F. Meyer, L. Lois, J. L. Carter, and T. P. Speis, An Analysis and Evaluation of the Clinch River Breeder Reactor Core Disruptive Accident Energetics. NUREG-0122, U.S. Nuclear Regulatory Commission, March 1977.
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8. U.S. Nuclear Regulatory Commission, Final Environmental Statement, Perkins Nuclear Power Station Nos. 1, 2, and 3. NUREG-75/088, Section 3.5, October 1975.
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12. Final Environmental Statement. Part III. "Floating Nuclear Power Plants," NUREG-0502, U.S. Nuclear Regulatory Commission, December 1978.

4.11 SODIUM WASTES

The purpose of the sodium waste system is to convert small amounts of metallic sodium waste, both radioactive and nonradioactive, into a less reactive form. The principal sources of sodium waste at the reactor are spent cold traps. Other possible sources are sodium-contaminated hardware such as equipment and spent fuel assemblies.⁽¹⁾

Because almost all of the tritium generated in the fission process is expected to precipitate out in the primary and secondary cold traps, the conversion of residual metallic sodium in spent cold traps requires special handling to minimize tritium releases to the environment. The procedure used will probably involve the following steps:

- draining as much sodium as possible into the primary or secondary sodium system before disconnecting the cold trap;
- deactivation of the contents with appropriate solvents with collection of all off gases and liquids;
- conversion of tritium hydride to tritiated water and routing of all liquids to the solid waste system for conversion to tritiated calcium hydroxide, $\text{Ca}(\text{OH})_2$;
- storing the chemically deactivated trap until removal for burial, salvage, and further cleanup.

After metallic sodium wastes have been chemically deactivated, they will be routed either to the liquid waste system or the solid waste system for further treatment, discharge, or shipment.

4.11.1 Comment

Sodium waste from LMFBR power plants may be processed offsite. A few special facilities may be built for this purpose. Wherever it is done, care must be taken to minimize tritium releases to the environment.

References

1. LMFBR Program Environmental Statement. U.S. Atomic Energy Commission, WASH-1535, Volume II, December 1974.

4.12 ACCIDENTS WITH RADIOACTIVE WASTES

The bulk of the solid radioactive wastes are miscellaneous low activity items. The principal sources of solid waste activity are from the sodium purification system and the concentrates resulting from the processing of liquid wastes. These wastes will be packaged and shipped offsite to a licensed disposal facility. One type of accident involving solid radioactive material could be a fire in the solid waste storage area. Combustible materials are not normally permitted in the area; however, if a fire should occur, the fire suppression system would act quickly, and the heating and ventilating system filters would prevent a significant release of radioactive materials.⁽¹⁾

The liquid radioactive system contains primarily low-level wastes. Liquid waste cells are equipped with sumps to enable any releases to be routed to another tank.

The only mechanism for release that can be identified is the release to the heating and ventilation system of small quantities of tritiated water vapor that have evaporated from the liquid pool during the time the spill fluid is on the floor. Neither the liquid nor the solid radioactive waste systems in the LMFBR contain a sufficient source of activity to cause significant impact on public safety if an accident did occur.

Two types of serious but very unlikely accidents are postulated for the gaseous radioactive waste system: 1) release of the primary system cover gas and 2) release of the contents of a gas storage or surge tank. The first type of accident could occur as a result of multiple failures of reactor vessel head seals, failure of a pipe connecting the reactor vessel, primary heat exchanger cover gas to the purifier, or a rupture of the purifier unit. A failure of the head seals would not result in a release directly to the environment, but rather to the containment. This failure could involve about 1000 Ci of noble gases (mostly short-lived isotopes of krypton and xenon). The cover gas is at low pressure (a few inches of water); therefore, the release would probably be at a slow rate. Prompt isolation of the containment would follow with no significant offsite exposures.

An accident of the second type (release of contents of cover gas storage or surge tanks) could result in a release to the low leakage cell housing the tank. Assuming, nominally, that 0.1% or less of the fuel is failed, and that the surge tank is at maximum tank pressure, approximately a half-million curies of krypton and xenon would be involved. Except for cell leakage, these gases can be diverted through the cell atmosphere processing system which will effectively prevent all but the ^{85}Kr and trace quantities of ^{39}Ar from being released (10 Ci); this capability will limit releases to concentrations below the levels specified in 10 CFR 20.

These two events are extremely unlikely failures; anticipated events such as leaking seals or temporary loss of cooling to some components would have lesser consequences.

The above discussion assumes the use of sealed fuel. Vented fuel concepts would routinely discharge radioactive gases or volatile materials into the cover gas and/or coolant, probably in excess of the levels conservatively assumed in the design of gas purification systems for 1% failure in sealed fuel. However, the system for effective cleanup and waste treatment would be well within existing design capability and would probably be similar in either case. Cleanup system inventories could be larger with vented fuel, but such changes would not materially alter the preceding discussion. In both systems, with proper consideration of possible faults in the design, the potential releases will not constitute a hazard to the public.

4.12.1 Comments

The radioactive waste treatment systems and procedures are suitable to comply with 10 CFR 20 in the event of an accident. On the FFTF, NRC determined that failure of the Radioactive Argon Processing System is the most limiting event.⁽²⁾

The system components are located in a series of gas-tight cells, and the cells housing the associated cold box and surge tank are steel lined. Both the cold box and the surge tank are isolated from each other and from the remainder of the system so that an event, plus a single failure, may

release only the inventory existing in the cold box and the surge tank. Therefore, the total failure of the surge tank and the cold box is an extremely unlikely event.

NRC calculations indicate that doses for failure of the surge tank and subsequent release of activity in the cold box over a two-day period would be a small fraction of 10 CFR 100 guideline doses.

NRC recommends that appropriate technical specifications be placed on the maximum activity that can be stored in any one tank at any time such that single failure of an active component will not result in radiological consequences that exceed small fractions of the 10 CFR 100 guideline dose.

References

1. LMFBR Program Environmental Statement. U.S. Atomic Energy Commission, WASH-1535, Vol. II, Washington, D.C., December 1974.
2. FFTF Safety Evaluation Report. U.S. Nuclear Regulatory Commission, NUREG-0538, Project No. 448, Washington, D.C., August 1978.

4.13 DECOMMISSIONING

Fission and activation products are carried throughout all parts of the primary sodium loop. They are deposited on internal surfaces of pipes, valves, and other components.

Some of the radionuclides diffuse into the steel after deposition.^(1,2) This is an important consideration to be taken into account when decontaminating the system.

4.13.1 Comments

FFTF staff believes decommissioning of the FFTF facility in 20 years will not be a problem. At the time the FFTF is decommissioned, the DOE will have authority over what procedures will be used for dismantling the plant. Based on experience available from the decommissioning of a number of small nuclear power plants, the FFTF staff believes that any safety-related technological problems incurred during decommissioning of the FFTF will not differ significantly from those that may occur during reactor refueling maintenance. The FFTF is designed for safe decontamination and dismantling, and the FFTF staff believes that it can be decommissioned with complete safety.⁽³⁾

References

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2. Colburn, R. P., "Characterization of Corrosion Product Deposits in Sodium Systems." Specialists' Meeting on Sodium REMoval and Decontamination (Proc. Meeting Richland, 1978), Rep. IWGFR/16, IAEA, Vienna 157, 1978.
3. FFTF Safety Evaluation Report. U.S. Nuclear Regulatory Commission, NUREG-0358, Project No. 448, Washington, D.C., August 1978.

4.14 FUEL HANDLING ACCIDENTS

The opacity of sodium and its chemical reactivity with air and water introduce some complexity to the LMFBR refueling process. Basically double containment barriers between fuel being handled and the outer containment atmosphere are employed.⁽¹⁾ A number of possible failures have been analyzed. These include complete loss of cooling of fuel transfer machines, complete loss of cooling of fuel transfer machines, complete loss of cooling in interim fuel assembly examination cells, and loss of cooling capability in interim decay storage tanks. These events would be unlikely.

An accident was postulated within the examination cell of the FFTF which involved complete loss of gas cooling. This event is considered to be extremely unlikely because there are redundant coolers, both of which are provided with a backup power supply. Fuel transfer activities are performed in sodium-filled pots that greatly increase heat removal capabilities (and in turn provide time for remedial action to restore effective cooling). For the maximum decay heat assembly (7kWt), pin rupture would occur approximately 25 min after loss of gas cooling, with clad melting occurring about 45 min after the event. The elements present in elements in the cell would be approximately:

- 6.6×10^4 Ci of noble gases ^{85}Kr , ^{133}Xe
- 8.6×10^4 Ci of halogens ^{131}I , ^{132}I
- 4.7×10^4 Ci of volatile solids
- 4.6×10^6 Ci of non-volatile fission products

As the cell atmosphere heats up, the internal pressure (originally negative with respect to surrounding air spaces) would rise and become positive; the examination cell design leak rate of 0.01 cfm would provide a leakage of approximately 0.002 Ci/sec into the containment work area. Even for this situation, confinement isolation would afford ample mitigation and control of the release. The cell atmosphere is constantly circulated to the Cell Atmosphere Processing System; virtually no release of radioactivity would take place. While the inventories and specific process flow activities might vary significantly in a commercial LMFBR, as compared with the various

analyses performed for the FFTF, it is concluded that design provisions related to fuel storage or transfer can and will be initiated to minimize the potential for serious accidents and, further, if such accidents should occur, to render any resultant releases relatively inconsequential.

4.14.1 Comments

The fuel handling equipment is designed with multiple cooling systems and multiple barriers to prevent the release of radioactivity outside the machine.⁽²⁾ In the unlikely event of loss of both of these redundant cooling mechanisms, or the even more unlikely possibility of mechanical damage, causing damage to the cladding or partial meltdown, partial release of the gaseous inventory might be expected. With the fuel handling machine located inside the containment building, the release occurs to the containment building atmosphere. Monitors on the containment building exhaust detect the abnormal rise in activity level and isolate the containment building. Any leakage that occurs prior to actuation of containment or due to leakage through penetrations after isolation is routed through the containment building ventilation filter system for removal of particulate matter. If a fuel handling incident occurs outside of containment in the reactor service building, release would occur to the reactor service building atmosphere. This atmosphere is also routed through the filter system to recover particulate matter. Thus, release to the environment due to fuel handling incidents is minimized, first, by multiple means of cooling the fuel element during transfer, second, by multiple barriers surrounding the fuel element within the fuel handling equipment design, third, by the containment isolation system if the accident occurs inside containment, and fourth, by the filtration system located outside containment.

The irradiated fuel that is removed from the in-vessel storage positions is normally placed in the interim decay storage vessel (IDS). This vessel is located in an inerted vault below the operating floor of the containment building. The IDS is equipped with redundant cooling systems to prevent any overheating in the event of a single failure. For FFTF based on 112 fuel subassemblies (not expected in practice), and decay heat of 240 KWt and

assuming loss of both cooling systems, it takes two days to reach sodium boiling. Fuel melting would not occur until all sodium boiled away. Corrective action can be taken before this happens. A battery operated cooling system could be connected to prevent clad failure. Even if all three cooling systems fail, no adverse environmental effect is expected since fission products released would be processed by the Cell Atmosphere Processing System. In addition, the gaseous fission products activity present in the IDS fuel has decayed to very low levels during the 100 days of in-vessel storage.

References

1. PSEID Liquid-Metal Fast Breeder Reactor Program. U.S. Atomic Energy Commission, WASH-1535, Vol. II, December 1974.
2. Fast Flux Test Facility Environmental Statement. U.S. Atomic Energy Commission, WASH-1510, May 1972.

4.15 CHEMICAL WASTES

The major source of chemical wastes is discharges from the water treatment systems.⁽¹⁾ The waste chemicals from these systems are discharged to the environment.

Condenser cooling water is treated to control biological slimes that can seriously interfere with the transfer of heat and adequate flow of water. The most common chemical used for this purpose is chlorine, added either as gaseous Cl_2 or as a hypochlorite solution. Other biocides are used for special conditions, particularly when salt water is used for cooling as the fouling deposits are often more extensive and contain a wider range of organisms than in the case when fresh water is used.

All closed-cycle evaporative systems after a period of operation will concentrate dissolved salts; the salts are the residue from water evaporation. The most common salts found in natural waters and hence in closed-cycle cooling systems are the bicarbonates, sulfates, chlorides and nitrates of calcium, magnesium, sodium, and potassium. These salts have various degrees of solubility within the water with calcium carbonate the least soluble. Whenever the solubility of the salt is exceeded, it sometimes precipitates jointly with other salts to form sludges or solid surface deposits called scale. The solubility of calcium carbonate decreases with increasing temperature so that scaling tends to occur first and to the greatest extent in the warm condenser where surface scale could seriously impair performance.

There are various ways to control scale and sludge formation in a closed-cycle system. One typically used is to blow down system water at a relatively low rate and replace it with fresh makeup water having a low salt concentration. This maintains lower salt concentrations within the system. Another method is to replace some of the carbonate (and the bicarbonate in equilibrium with it) with sulfate by adding sulfuric acid. The resultant carbon dioxide is released to the atmosphere. Sometimes sequestering agents are also used to tie up metallic ions which might otherwise form insoluble carbonates.

Other chemicals may be used in closed-cycle systems to provide treatment for corrosion, biological fouling, and silt deposition.

In summary, there are many chemicals which, by virtue of system water treatment, might appear in the blowdown discharge to the environment. For the most part, however, the blowdown merely contains higher concentrations of the salts found in the makeup water to the system and possibly residual chlorine. If sulfuric acid is used, there will be higher concentrations of sulfates and lower bicarbonate concentrations.

Very high purity water is required in the steam system to minimize corrosion and scale formation. To maintain purity, a portion of the process water is blown down and replaced with demineralized water. The demineralized water is produced from raw water through a series of water treatments which include clarification, filtration, and ion exchange. Clarification is the process of removing turbidity and sediment. An array of chemicals may be used in clarification such as alum, sodium aluminate, ferric sulfate, and others to coagulate finely divided and colloidal matter suspended in the water. The wastes from these operations, called sludges, are most often dewatered in settling basins. Other methods can be used to remove the water from the sludges, such as centrifugation. Filtration is used after clarification, and the residue materials are treated in the same manner as for clarification.

Ion exchange (demineralization) is typically the last step in the water purification process and it results in a relatively large chemical waste stream. In ion exchange, dissolved compounds in the form of ions are removed from the water to be purified by first attracting them to electrically charged resinous surfaces where they become tightly held and replaced with other ions. Typically the ion exchange resins are hydrogen cation and hydroxide anion types where the hydrogen ions are used to replace the cations in solutions such as magnesium and calcium, and the hydroxide ions are used to replace anions such as bicarbonate and sulfate. After a period of operation the ion exchange resins become ineffectual and must be regenerated. Sulfuric acid is generally used to regenerate the cation resin and sodium hydroxide and anion resin. In the process of regeneration, the original ionic materials that were trapped by the resins are released and replaced with hydrogen and hydroxide ions; the resulting regenerative waste stream is then composed of a solution of the original

ionic materials less the bicarbonates, which were converted to water and carbon dioxide, plus quantities of sodium sulfate, the reaction product of sulfuric acid and sodium hydroxide. These wastes are then normally discharged to the environment after diluting them with either the once-through condenser cooling water flow or the blowdown effluent from a closed-cycle cooling system.

Steam system blowdown is another possible source of aqueous waste from the system. Since the water is chemically treated for scale and corrosion control, it is expected that some of the chemicals used for this purpose might also be in the blowdown. Some of these chemicals might be sodium phosphates for scale control, sequestering agents, sodium hydroxide or carbonate for pH control, or hydrazine for oxygen control. The blowdown may also contain some corrosion products and other dissolved solids.

Other sources of aqueous chemical wastes discharged to the environment include spent chemicals for major component cleaning operations, sampling and laboratory wastes, sanitary wastes, and laundry wastes.

Cleaning of all the heat transfers surfaces must be done periodically to remove the possible scale deposits on all major plant components. Strong acids are usually used for this purpose. The waste stream from the scale cleaning operations results in a mixture of acids, copper, iron, phosphates, and other materials. Good practice requires that these wastes be collected in storage basins where they can be treated by neutralization and sedimentation after which they can be discharged to the environment with suitable dilution.

Plant sanitary wastes can be handled either by packaged treatment systems or by discharging them to a municipal treatment facility if available.

Nonradioactive solid wastes include demineralizer resins, filters, replaced equipment components, shop and laboratory wastes, garbage, water intake debris, and other trash. The disposal of these wastes is most likely performed by a disposal agency licensed and regulated by local authorities.

4.15.1 Comments

The nonradioactive chemical wastes for an LMFBFR are basically no different in type or quantity than for a LWR or fossil plant. The reason

for this is that the majority of these wastes are produced by the steam generator system and the heat rejection system. Therefore, there are not really any additional noteworthy concerns for LMFBRs.

Reference

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5.0 BIBLIOGRAPHY

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APPENDIX A

DESIGN CRITERIA FOR PROTOTYPE

LARGE BREEDER REACTOR

APPENDIX A
DESIGN CRITERIA FOR PROTOTYPE
LARGE BREEDER REACTOR

The current Conceptual Design Study is using the PLBR Design Criteria as a starting point. They apply as follows:

- Overall Requirements: Criteria 1-7
- Protection by Multiple Fission Product Barriers: Criteria 8-17
- Protection and Reactivity Control System: Criteria 18-25
- Cooling System: Criteria 26-40
- Reactor Containment: Criteria 41-51
- Fuel and Radioactivity Control: Criteria 52-56

CRITERION 1 - QUALITY STANDARDS AND RECORDS

Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of structures, systems, and components important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.

CRITERION 2 - DESIGN BASES FOR PROTECTION AGAINST NATURAL PHENOMENA

Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect:

1. Appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated,
2. Appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena, and
3. The importance of the safety functions to be performed.

CRITERION 3 - FIRE PROTECTION

Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the

probability and effect of fires and explosions. Noncombustible and heat-resistant materials shall be used wherever practicable throughout the unit, particularly in locations such as the containment and control room. Fire detection and firefighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Firefighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of structures, systems, and components.

CRITERION 4 - PROTECTION AGAINST SODIUM REACTIONS

Systems, components, and structures containing sodium shall be designed to limit the consequences of sodium chemical reactions resulting from a sodium spill. Special features such as inerted vaults shall be provided as appropriate for the reactor coolant system. Means to detect sodium chemical reactions and fire control systems shall be provided to limit and control the extent of such reactions to assure that the functions of components important to safety are maintained. Means shall be provided to limit the release of sodium reaction products to the environment as necessary to protect plant personnel and to avoid undue risk to the public health and safety. Materials which might come in contact with sodium shall be chosen to minimize the adverse effects of possible chemical reactions. In areas where sodium chemical reactions are possible, structures, components, and systems important to safety, including electrical wiring and components, shall be designed and located so that the potential for damage by sodium chemical reactions is minimized. Means shall be provided as appropriate to minimize possible contacts between sodium and water. The effects of possible interactions between sodium and concrete shall be considered in the design.

The sodium-steam generator system shall be designed to detect sodium-water reactions and limit the effects of the energy and reaction products released by such reactions to prevent loss of safety functions of the heat transport system.

CRITERION 5 - ENVIRONMENTAL^(a) AND MISSILE DESIGN BASES

Structures, systems, and components important to safety shall be designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, anticipated operational occurrences and postulated accidents. These structures, systems, and components shall be appropriately protected against dynamic effects - such as the effects of missiles, pipe whipping, and discharging fluids - that may result from equipment failures and from events and conditions outside the nuclear power unit.

CRITERION 7 - SODIUM HEATING SYSTEMS

Heating systems shall be provided as necessary for systems and components important to safety which contain, or may be required to contain, sodium. The heating systems and their controls shall be appropriately designed with suitable redundancy to assure that the temperature distribution and rate of change of temperature in sodium systems and components are maintained with design limits, assuming a single failure. The heating system shall be designed such that failure of any of its components will not impair the safety function of associated systems and components, assuming a single additional failure.

CRITERION 8 - REACTOR DESIGN

The reactor and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation and anticipated operational occurrences.

CRITERION 9 - REACTOR INHERENT PROTECTION

The reactor core and associated coolant systems shall be designed so that in the power operating range, the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.

(a) Natural phenomena are covered by Criterion 2.

CRITERION 10 - SUPPRESSION OF REACTOR POWER OSCILLATIONS

The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

CRITERION 11 - INSTRUMENTATION AND CONTROL

Instrumentation shall be provided to monitor variables and systems--over their anticipated ranges for normal operation--for anticipated operational occurrences, and for postulated accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

CRITERION 12 - REACTOR COOLANT BOUNDARY

The reactor coolant boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

CRITERION 13 - REACTOR COOLANT SYSTEM DESIGN

The reactor coolant system and associated control, protection, auxiliary, and sodium heating systems, shall be designed with sufficient margin to assure that the design conditions of the reactor coolant boundary are not exceeded during any condition of normal operation and anticipated operational occurrences.

CRITERION 14 - CONTAINMENT DESIGN

Reactor containment and associated systems shall be provided to establish an essentially leaktight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions

important to safety are not exceeded for as long as postulated accident conditions require.

CRITERION 15 - ELECTRIC POWER SYSTEMS

Onsite and offsite electric power systems shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that:

1) specified acceptable fuel design limits and design conditions of the reactor coolant boundary are not exceeded as a result of anticipated operational occurrences; and 2) the core is cooled, and containment integrity and other vital functions are maintained in the event of postulated accidents.

The onsite electric power supplies, including the batteries and the onsite electric distribution system, shall have sufficient independence, redundancy, and testability to perform their safety functions, assuming a single failure.

Electric power from the transmission network to the onsite electric distribution system shall be supplied by two physically independent circuits (not necessarily on separate rights of way) designed and located to minimize, as much as practicable, the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. A switchyard common to both circuits is acceptable. Each of these circuits shall be designed to be available in sufficient time, following a loss of all onsite alternating current power supplies and the other offsite electric power circuit, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant boundary are not exceeded. One of these circuits shall be designed to be available within a few seconds, following any postulated accident which does not involve loss of capability of the switchyard to provide power, to assure that core cooling, containment integrity and other vital safety functions are maintained.

Provisions shall be included to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident

with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies.

CRITERION 16 - INSPECTION AND TESTING OF ELECTRIC POWER SYSTEMS

Electric power systems important to safety shall be designed to permit appropriate periodic inspection and testing of important areas and features - such as wiring, insulation, connections, and switchboards - to assess the continuity of the systems and the condition of their components. The systems shall be designed with a capability to test periodically: 1) the operability and functional performance of the components of the systems, such as onsite power sources, relays, switches, and buses; and 2) the operability of the systems as a whole and, under conditions as close to design as practicable, the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system, and the transfer of power among the nuclear power unit, the offsite power system, and the onsite power system.

CRITERION 17 - CONTROL ROOM

A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal operations and to maintain it in a safe condition under accident conditions (including those conditions addressed in NRC Criterion 4 - Protection Against Sodium Reactions). Adequate radiation protection shall be provided to permit access and occupancy of the control room under postulated accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. Protection shall also be provided for the control room against release of toxic gases that may be stored in significant quantities on or in the vicinity of the site.

Equipment at appropriate locations outside the control room shall be provided with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a

safe condition during hot shutdown, and with a design capability for subsequent control of the reactor at any coolant temperature lower than the hot shutdown condition.

CRITERION 18 - PROTECTION SYSTEM FUNCTIONS

The protection system shall be designed: 1) to initiate automatically the operation of appropriate systems, including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences; and 2) to sense accident conditions and to initiate the operation of systems and components important to safety.

CRITERION 19 - PROTECTION SYSTEM RELIABILITY AND TESTABILITY

The protection system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that: 1) no single failure results in loss of the protection function, and 2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.

CRITERION 20 - PROTECTION SYSTEM INDEPENDENCE

The protection system shall be designed to assure that the effects of natural phenomena and of normal operating, maintenance, testing and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function.

CRITERION 21 - PROTECTION SYSTEM FAILURE MODES

The protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, sodium, sodium reaction products, and radiation) are experienced.

CRITERION 22- SEPARATION OF PROTECTION AND CONTROL SYSTEMS

The protection system shall be separated from control systems to the extent that failure of any single control system component or channel which is common to the control and protection systems, leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Interconnection of the protection and control system shall be limited to assure that safety is not significantly impaired.

CRITERION 23 - PROTECTION SYSTEM REQUIREMENTS FOR REACTIVITY CONTROL MALFUNCTIONS

The protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal of control rods.

CRITERION 24 - REACTIVITY CONTROL SYSTEM REDUNDANCEY AND CAPABILITY

Two independent reactivity control systems, preferably of different design principles, shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity to assure that under normal and anticipated operational occurrences, and with appropriate margin for malfunctions such as a stuck rod(s), specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure that acceptable fuel

design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical for any coolant temperature lower than the hot shutdown condition.

CRITERION 25 - COMBINED REACTIVITY CONTROL SYSTEMS CAPABILITY

The reactivity control systems shall be designed to have a combined capability of reliably controlling reactivity changes to assure that under postulated accident conditions, and with appropriate margin for stuck rod(s), the capability to cool the core is maintained.

CRITERION 26 - HEAT TRANSPORT SYSTEM DESIGN

The heat transport system shall be designed to reliably remove heat from the reactor and transport the heat to the turbine generator or ultimate heat sinks under all plant conditions, including normal operation, anticipated operational occurrences, and postulated accidents. Consideration shall be given to the provision of independence and diversity to provide adequate protection against common mode failures. The system safety functions shall be to:

- provide sufficient cooling to prevent exceeding specified acceptable fuel design limits during normal operation and following anticipated operational occurrences, and
- provide sufficient cooling to prevent exceeding specified acceptable fuel damage limits and to maintain integrity of the reactor coolant boundary following postulated accidents.

Following the loss of a flow path, the heat transport system shall include at least two independent flow paths, each capable of performing the safety functions following shutdown.^(a)

The system shall include suitable electrical interconnections, leak detection, isolation and containment capability to assure that, for onsite electric power operation (assuming offsite power is not available) and for

(a) This requirement is not intended to preclude two-loop operation provided the system safety functions can be appropriately met.

offsite electric power operation (assuming onsite power is not available), the safety function can be accomplished, assuming a single failure.

CRITERION 27 - ASSURANCE OF ADEQUATE REACTOR COOLANT INVENTORY

The reactor coolant boundary and associated components, control and protection systems shall be designed to limit loss of reactor coolant so that an inventory adequate to perform the safety functions of the heat transport system is maintained under normal operation, anticipated operational occurrences and postulated accident conditions.

CRITERION 28 - QUALITY OF REACTOR COOLANT BOUNDARY

Components which are part of the reactor coolant boundary shall be designed, fabricated, erected, and tested to the highest quality standards practicable. Means shall be provided for detecting and, to the extent practicable, identifying the location of the source of reactor coolant leakage.

CRITERION 29 - FRACTURE PREVENTION OF REACTOR COOLANT BOUNDARY

The reactor coolant boundary shall be designed with sufficient margin to assure that when stressed under normal operating, maintenance, testing and postulated accident conditions: 1) the boundary behaves in a non-brittle manner, and 2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures, service degradation of material properties, creep, and other conditions of the boundary material under normal operating, maintenance, testing and postulated accident conditions, and the uncertainties in determining: 1) material properties; 2) the effects of coolant chemistry and irradiation on material properties; 3) residual, steady-state and transient stresses; and 4) size of flaws.

CRITERION 30 - INSPECTION OF REACTOR COOLANT BOUNDARY

Component which are part of the reactor boundary shall be designed to permit: 1) periodic inspection of areas and features important to safety to

assess their structural and leaktight integrity, and 2) an appropriate material surveillance program.

CRITERION 31 - REACTOR AND INTERMEDIATE COOLANT AND COVER GAS PURITY CONTROL

Systems shall be provided to monitor and maintain reactor and intermediate coolant and cover gas purity within specified design limits. These limits shall be based on consideration of the potential for 1) chemical attack, 2) fouling and plugging of passages, 3) radioisotope concentrations, and 4) detection of sodium-water reactions.

CRITERION 32 - INTERMEDIATE COOLANT SYSTEM

The intermediate coolant system shall be designed to transport heat reliably from the reactor coolant system to the reactor residual heat extraction systems as required for the reactor coolant system to meet its safety function under all plant conditions including normal operation, anticipated operational occurrences, and postulated accident conditions. The intermediate coolant system shall contain coolant that is not chemically reactive with the reactor coolant. A pressure differential shall be maintained across a passive boundary between the reactor coolant system and the intermediate coolant system such that any leakage would flow from the intermediate coolant system to the reactor coolant system, unless other provisions can be shown to be acceptable on some defined basis.

CRITERION 33 - FRACTURE PREVENTION OF INTERMEDIATE COOLANT BOUNDARY

The intermediate coolant boundary shall be designed with sufficient margin to assure that, when stressed under normal operation, anticipated operational occurrences and postulated accident conditions, 1) the boundary behaves in a non-brittle manner, and 2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures, service degradation of material properties, creep, and other conditions of the boundary material under normal operation, anticipated operational occurrences,

and postulated accident conditions and the uncertainties in determining:
1) material properties, 2) the effects of coolant chemistry, 3) residual, steady-state and transient stresses, and 4) size of flaws.

CRITERION 34 - INSPECTION AND SURVEILLANCE OF INTERMEDIATE COOLANT BOUNDARY

Components which are part of the intermediate coolant boundary shall be designed to permit: 1) periodic inspection and testing of areas and features important to safety, to assess their structural and leaktight integrity; and 2) an appropriate material surveillance program for the intermediate coolant boundary. Means shall be provided for detecting intermediate coolant leakage.

CRITERION 35 - REACTOR RESIDUAL HEAT EXTRACTION SYSTEMS

The reactor residual heat extraction systems shall be provided to transfer reactor residual heat from the reactor coolant system to ultimate heat sinks under all plant shutdown conditions following normal operation, anticipated operational occurrences and postulated accident conditions. A passive boundary shall normally separate reactor coolant from the working fluids of the heat extraction systems. Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that, for onsite electrical power system operations (assuming offsite power is not available) and for offsite electrical power system operation (assuming onsite power is not available), the system safety function can be accomplished, assuming a single failure.

CRITERION 36 - INSPECTION OF REACTOR RESIDUAL HEAT EXTRACTION SYSTEMS

The reactor residual heat extraction systems shall be designed to permit appropriate periodic inspection of important components, such as heat exchangers and piping, to assure integrity and capability of the system.

CRITERION 37 - TESTING OF REACTOR RESIDUAL HEAT EXTRACTION SYSTEMS

The reactor residual heat extraction systems shall be designed to permit appropriate periodic pressure and functional testing to assure: 1) the

structural and leaktight integrity of their components; 2) the operability and the performance of the active components of the systems; and 3) the operability of each complete system and, under conditions as close to design as practicable, the performance of the full operational sequence that brings the systems into operation for reactor shutdown and following postulated accidents, including operation of applicable portions of the protection system and the transfer between normal and emergency power sources.

CRITERION 38 - ADDITIONAL COOLING SYSTEMS

In addition to the heat rejection capability provided by the reactor residual heat extraction systems, systems to transfer heat - from structures, systems, and components important to safety - to an ultimate heat sink shall be provided, as necessary. The system safety function shall be to transfer the combined heat load of these structures, systems, and components as required for safety under normal operating and accident conditions.

Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that, for onsite electric power system operation (assuming offsite power is not available) and for offsite electrical power system operation (assuming onsite power is not available), the system safety function can be accomplished, assuming a single failure.

CRITERION 39 - INSPECTION OF ADDITIONAL COOLING SYSTEMS

The additional cooling systems shall be designed to permit appropriate periodic inspection of important components, such as heat exchangers and piping, to assure the integrity and capability of the systems.

CRITERION 40 - TESTING OF ADDITIONAL COOLING SYSTEMS

The additional cooling systems shall be designed to permit appropriate periodic pressure and functional testing to assure: 1) the structural and leak-tight integrity of their components; 2) the operability and the performance of the active components of the systems; and 3) the operability of the complete

systems and, under conditions as close to design as practicable, the performance of the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system and the transfer between normal and emergency power sources.

CRITERION 41 - CONTAINMENT DESIGN BASES

The reactor containment structure shall be designed so that both the structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from normal operation, anticipated operational occurrences, and any of the postulated accidents. This design shall include access openings and penetrations, and, if necessary, be done in conjunction with additional post-accident heat removal systems, including ex-vessel systems. This margin shall reflect consideration of 1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as decay heat in released fission products, potential spray or aerosol formation, and potential exothermic chemical reactions; 2) the limited experience and experimental data available for defining accident phenomena and containment responses; and 3) the conservatism of the calculational model and input parameters.

CRITERION 42 - FRACTURE PREVENTION OF REACTOR CONTAINMENT BOUNDARY

The reactor containment boundary shall be designed with sufficient margin to assure, under normal operating, maintenance, testing and postulated accident conditions, that: 1) its metallic materials behave in a non-brittle manner, and 2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the containment boundary material during normal operation, maintenance, testing and postulated accident conditions, and the uncertainties in determining: 1) material properties, 2) residual, steady-state and transient stresses, and 3) size of flaws.

CRITERION 43 - CAPABILITY FOR CONTAINMENT LEAKAGE RATE TESTING

The reactor containment and other equipment which may be subjected to containment test conditions shall be designed so that periodic integrated leakage rate testing can be conducted at containment design pressure.

CRITERION 44 - PROVISIONS FOR CONTAINMENT TESTING AND INSPECTION

The reactor containment shall be designed to permit: 1) appropriate periodic inspection of all important areas, such as penetrations; 2) an appropriate surveillance program; and 3) periodic testing at containment design pressure of the leaktightness of penetrations which have resilient seals and expansion bellows.

CRITERION 45 - PIPING SYSTEMS PENETRATING CONTAINMENT

Piping systems penetrating reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus, and to determine if valve leakage is within acceptable limits.

CRITERION 46 - REACTOR COOLANT BOUNDARY PENETRATING CONTAINMENT

Each line that is part of the reactor coolant boundary and that penetrates reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

- One locked closed isolation valve inside and one locked closed isolation valve outside containment, or
- One automatic isolation valve inside and one locked closed isolation valve outside containment, or

- One locked closed isolation valve inside and one automatic isolation valve outside containment (a simple check valve may not be used as the automatic isolation valve outside containment), or
- One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to containment as practicable and, upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety. Other appropriate requirements to minimize the probability or consequences of an accidental rupture of these lines or of lines connected to them shall be provided as necessary to assure adequate safety. Determination of the appropriateness of these requirements, such as higher quality in design, fabrication, and testing, additional provisions for inservice inspection, protection against more severe natural phenomena, and additional isolation valves and containment shall include consideration of the population density, use characteristics, and physical characteristics of the site environs.

CRITERION 47 - PRIMARY CONTAINMENT ISOLATION

Each line that connects directly to the containment atmosphere and penetrates reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

- One locked closed isolation valve inside and one locked closed isolation valve outside containment, or
- One automatic isolation valve inside and one locked closed isolation valve outside containment, or
- One locked closed isolation valve inside and one automatic isolation valve outside containment (a simple check valve may not be used as the automatic isolation valve outside containment), or

- One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to the containment as practicable and, upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

CRITERION 48 - CLOSED SYSTEMS PENETRATING CONTAINMENT

Each line of a closed system that penetrates reactor containment and is neither part of nor directly connected to the reactor coolant boundary, nor directly to the containment atmosphere, shall have at least one containment isolation valve, unless it is demonstrated that containment isolation provisions for a specific class of lines are acceptable on some other defined basis. The isolation valve, if required, shall be either automatic, or locked closed, or capable of remote manual operation. This valve shall be outside containment and located as close to the containment as practicable. A simple check valve may not be used as the automatic isolation valve.

CRITERION 49 - CONTAINMENT ATMOSPHERE CLEANUP

Systems to control fission products, hydrogen, oxygen and other substances which may be released into the reactor containment, shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quantity of fission products released to the environment following postulated accidents. These systems shall also control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained. The necessity of such systems should consider the effects of sodium leakage and its potential reaction with oxygen and its potential for hydrogen when in contact with concrete.

Each system shall have suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment

capabilities to assure that, for onsite electric power system operation (assuming offsite power is not available), its safety function can be accomplished, assuming a single failure.

CRITERION 50 - INSPECTION OF CONTAINMENT ATMOSPHERE CLEANUP SYSTEMS

The containment atmosphere cleanup systems shall be designed to permit appropriate periodic inspection of important components, such as filter frames, ducts, and piping to assure the integrity and capability of the systems.

CRITERION 51 - TESTING OF CONTAINMENT ATMOSPHERE CLEANUP SYSTEMS

The containment atmosphere cleanup systems shall be designed to permit appropriate periodic functional testing to assure: 1) the structural and leaktight integrity of its components; 2) the operability and performance of the active components of the systems such as fans, filters, dampers, pumps, and valves; and 3) the operability of the systems as a whole and, under conditions as close to design as practicable, the performance of the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of associated systems.

CRITERION 52 - CONTROL OF RELEASES OF RADIOACTIVE MATERIALS TO THE ENVIRONMENT

The nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, and anticipated operational occurrences. Sufficient holdup capacity shall be provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment.

CRITERION 53 - FUEL STORAGE AND HANDLING AND RADIOACTIVE WASTE CONTROL

The fuel storage and handling, radioactive waste control, and other systems which may contain radioactivity, shall be designed to assure adequate safety under

normal and postulated accident conditions. These systems shall be designed:

- 1) with a capability to permit appropriate periodic inspection and testing of components important to safety;
- 2) with suitable shielding for radiation protection;
- 3) with appropriate containment, confinement, and filtering systems;
- 4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal;
- and 5) to prevent significant reduction in fuel storage coolant inventory under accident conditions.

CRITERION 54 - PREVENTION OF CRITICALITY IN FUEL STORAGE AND HANDLING

Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

CRITERION 55 - MONITORING FUEL AND WASTE STORAGE

Appropriate systems shall be provided in fuel storage and radioactive waste systems and associated handling areas to:

- 1) detect conditions that may result in loss of residual heat removal capability and excessive radiation levels,
- and 2) initiate appropriate safety actions.

CRITERION 56 - MONITORING RADIOACTIVITY RELEASES

Means shall be provided for monitoring the reactor containment atmosphere, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, anticipated operational occurrences, and postulated accidents.

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