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U. S. Nuclear
Regulatory Commission

Safety Evaluation Report

related to operation of

Office of Nuclear
Reactor Regulation

Fast Flux Test Facility

Project No. 448

U. S. Department of Energy

May 1979

MASTER

Supplement No. 1

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NUREG-0358

(Suppl. 1) May 1979

SUPPLEMENT NO. 1

TO THE

SAFETY EVALUATION REPORT

BY THE

OFFICE OF NUCLEAR REACTOR REGULATION

U.S. NUCLEAR REGULATORY COMMISSION

FOR

U.S. DEPARTMENT OF ENERGY

FAST FLUX TEST FACILITY

PROJECT NO. 448

21

TABLE OF CONTENTS

	<u>PAGE</u>
1.0 INTRODUCTION AND GENERAL DISCUSSION	
1.1 Introduction.....	1-1
1.2 General Plant Description.....	1-2
1.7 Status of Review Items.....	1-3
1.8 Requirements for Future Technical Information.....	1-7
1.9 Summary of Active Issues.....	1-9
3.0 DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS	
3.2 Classification of Structures, Components, and Systems.....	3-1
3.2.1 Seismic Classification.....	3-1
3.2.2 System Quality Group Classification.....	3-1
3.5 Missile Protection Criteria.....	3-1
3.5.1 Missiles Generated Outside of Containment.....	3-1
3.8 Design of Seismic Category I Structures.....	3-1
3.8.2 Containment Interior Structures.....	3-1
3.10 Seismic Qualification of Category I Instrumentation and Control Equipment.....	3-1
3.11 Environmental Qualification of Safety-Related Instrumentation and Control Equipment.....	3-2
4.0 REACTOR	
4.4 Thermal-Hydraulic Design.....	4-1
4.4.2 Thermal-Hydraulic Limits.....	4-1
4.4.9 Loose Parts Monitoring System.....	4-1
4.4.10 Evaluation and Findings.....	4-1

TABLE OF CONTENTS (Continued)

5.0	REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS	
5.2	Heat Transport System.....	5-1
5.2.7	Heat Transport System Materials Evaluation.....	5-1
5.4	Surveillance and Inservice Inspection.....	5-1
5.4.4	Recommendations.....	5-1
5.4.4.1	Leak Detection.....	5-1
5.4.4.2	Material Surveillance.....	5-2
5.4.4.3	Preservice Inspection.....	5-2
5.5	Decay Heat Removal System.....	5-2
6.0	ENGINEERED SAFETY FEATURES	
6.4	Control Room Habitability.....	6-1
7.0	INSTRUMENTATION AND CONTROLS	
7.1	Summary Description.....	7-1
7.2	Reactor Shutdown Systems.....	7-1
7.4	Systems Required for Safe Shutdown.....	7-1
7.5	Safety-Related Display Instrumentation.....	7-1
7.7	Control Systems Not Required for Safety.....	7-2
8.0	ELECTRIC POWER	
8.3	Onsite Power Systems.....	8-1
8.3.4	Environmental Qualification of Class IE Equipment.....	8-1
8.3.6	Electrical Penetrations.....	8-1
9.0	AUXILIARY SYSTEMS	
9.3	Process Auxiliaries.....	9-1
9.3.1	Compressed Air System.....	9-1

TABLE OF CONTENTS (Continued)

9.4 Heating and Ventilating Systems.....	9-1
9.4.1 Control Room Area Ventilation System.....	9-1
9.5 Service Systems.....	9-1
9.5.1 Fire Protection System.....	9-1
15.0 ACCIDENT ANALYSIS	
15.3 Analysis and Evaluation of Core Meltdown and Disassembly Accidents.....	15-1
15.3.6 Containment Response to a Core Meltdown Event.....	15-1
15.3.7 Radiological Consequences of Core Disruptive Accidents.....	15-2
16.0 TECHNICAL SPECIFICATIONS	
18.0 REVIEW BY THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS	
19.0 SUMMARY OF ADVICE AND RECOMMENDATIONS	
19.3 Advice and Recommendations.....	19-1
19.4 Status of Miscellaneous Issues.....	19-3
19.5 Summary of Active Issues.....	19-5
20.0 SUMMARY - CONCLUSIONS	

TABLE OF CONTENTS (Cont'd)

APPENDIX A	References
B	ACRS Report
C	NRC Letter to ACRS dated October 26, 1978, transmitting Supplemental Information
D	NRC Memo to ACRS, dated February 26, 1979, transmitting exchange of letters between NRC and DOE regarding DOE plans for addressing NRC and ACRS Advice on Start-up and Operation of FFTF
E	Aerospace Corporation Evaluation of Sodium-Concrete Interactions
F	Sandia Comments on L. D. Muhlstein's Summary of November 15-16, 1978 Meeting on Sodium/Concrete (Memo to H. Holz from T. H. Walker, April 6, 1979)
G	Sodium Concrete Interaction Test No. 13 (Memo to M. Silberberg from T. J. Walker, dated December 29, 1978)
H	Sodium/Magnetite Concrete Reaction Test No. 14 (Memo to M. Silberberg from T. J. Walker, dated April 11, 1979)
I	NRC Staff Recommendations Regarding Conventional FFTF Fire Protection Program (Non-Sodium)
J	Chronology

1.0 INTRODUCTION AND GENERAL DISCUSSION

1.1 Introduction

The Nuclear Regulatory Commission (hereinafter referred to as the Commission) staff issued its Safety Evaluation Report, NUREG-0358, related to the startup and initial operation of the Fast Flux Test Facility (hereinafter referred to as the FFTF) in August 1978. The staff, its consultants, and representatives of the FFTF Project, their subcontractor, Westinghouse Hanford Company, and the Advisory Committee on Reactor Safeguards (hereinafter referred to as the ACRS) met on September 7 and November 2, 1978 to review matters pertaining to the startup and operation of the FFTF. Previous to this, the ACRS subcommittee on FFTF considered the startup and initial operation of the FFTF in its July 12, and August 10, 1978 meetings. On November 8, 1978, the Commission received from the ACRS its report on the FFTF (this report is included as Appendix B to this supplement). The ACRS report provided advice and recommendations which are consistent with the advice and recommendations provided to the Department of Energy (DOE) in the Commission's Safety Evaluation Report. (The ACRS report is further discussed in Section 18.0 of this supplement.)

This supplement provides (1) the staff's evaluation of additional information received since issuance of the Safety Evaluation Report regarding previously identified uncompleted review items, (2) a discussion of comments made by the ACRS in its report of November 8, 1978, and (3) the staff's evaluation of additional or revised information related to new and/or old issues that have arisen since the issuance of the Safety Evaluation Report.

Sections of this supplement carry the same numbers as those of the Safety Evaluation Report (NUREG-0358) which they address or modify, and except where specifically noted, do not replace sections of the Safety Evaluation Report. Also where page numbers are included in the discussion in this supplement they refer to page numbers in the Safety Evaluation Report.

The construction of the FFTF is, at present, essentially complete and the facility is entering its preoperational and startup phase, the sodium has been loaded, and system testing is underway. Fuel loading is scheduled for July 1979. One objective of this supplement is to formally dispose of a number of open issues identified in the Safety Evaluation Report. Supplemental information and/or commitments have been obtained from the FFTF Project to clarify and/or resolve some of these issues. In some of these areas the commitments involve the implementation of facility modifications prior to unrestricted full power operation. The issues,

and their disposition, resolved in this supplement have been discussed with members of the ACRS subcommittee on FFTF as well as with the full Committee and are a matter of record in transcripts of those meetings. The issues not resolved are again restated in this supplement, e.g., the issue of containment margins for the core melt events (which involve design changes and possibly additional testing) and the issue regarding the adequacy of natural circulation which will have to be experimentally verified if plant modifications for forced convective cooling is to be avoided.

A third issue, closed loops, is also of special interest but has been deferred because its design is incomplete and may not be implemented for several years. The staff's review basis has been that the core positions to be occupied by the closed loops will be replaced with driver fuel. When the closed loops are finally installed the response of test elements within the loop might have an important effect on the safety evaluation of the FFTF due to the close neutronic coupling of the driver core and closed-loop test region.

The staff has completed its evaluation of the non-sodium fire hazards analysis submitted by the Project and has made three recommendations which are included in Appendix I.

The staff's review for this supplement includes consideration of FFTF, FSAR Supplement No. 30 and Amendment No. 36 reference material pertaining to the environmental qualifications of safety-related equipment, and those agreements and commitments of record which have been made known to the staff.

1.2 General Plant Description

The FFTF is designed for the purpose of testing fuels and materials in a fast neutron flux environment of 7×10^{15} n/cm²/sec. The reactor system design includes provision for 2 in-core closed loops, and 4 open test positions within the core matrix. The plant power is 400 Mwt including pump power. The driver fuel (73 subassemblies) is composed of mixed plutonium and uranium oxides. The reactor power is regulated through the use of 3 primary and 6 secondary control rods of B₄C. Either group of control rods can initiate reactor shutdown.

The containment structure consists of a steel shell, insulated above ground and embedded in concrete below ground. Adjacent concrete structures house the control room, auxiliary services, diesel generators and the radioactive waste system.

The reactor coolant is liquid sodium circulated in three parallel pipe loops common to the reactor vessel. Variable speed pumps adjust flow to match power and maintain an essentially constant core ΔT with power variations. Heat energy is transferred from the primary sodium coolant system to a secondary sodium system.

This system is also a piped loop system that rejects its heat to the atmosphere via the dump heat exchanger. All sodium heat transfer loops removing energy from the reactor are seismically designed Category I for the safe shutdown earthquake, but only the east secondary sodium main heat transfer system loop is tornado hardened against missiles.

In the event of failure of the offsite power systems, diesel generators are provided onsite. The diesel generators are intended only to augment or back up the offsite power sources and, although called emergency diesel generators, are not qualified as such.

In the event of loss of all alternating current (AC) power, the safety power system (battery/direct current inverter) supplies initial instrument and control function power for 15 minutes, for containment isolation and a controlled shutdown, and then provides essential plant monitoring functions for a period of one week. Natural circulation of sodium, monitored by instruments powered by the direct current Class 1E electric system, would be relied upon for ultimate decay heat removal in the event of loss of all offsite and onsite alternating current electrical power.

The control room is part of the auxiliary building substructure and is tornado missile protected by its location. Radioactive liquid wastes are generally stored and then transported offsite at a convenient time and low-level gaseous wastes are held up for decay and subsequently released to the atmosphere. The design of the FFTF is based upon a 20-year life expectancy.

1.7 Status of Review Items

Several construction changes were identified in Supplements Nos. 1 and 2 to the construction phase Safety Evaluation Report. For the initial operating phase of the FFTF our review and our technical assessment has been directed primarily to providing the Project with advice derived from our experience with varied types of commercial and test reactors in their operational phase. Significant FFTF modifications completed, planned or contemplated, and identified in our Safety Evaluation Report, NUREG-0358, have been resolved and are discussed in the following paragraphs of this supplement:

(1) Piping Integrity

(a) Inservice Inspection

The staff's summary position on piping integrity is documented in Section 19.3C of the Safety Evaluation Report. In a letter to the ACRS, dated October 26, 1978 and in discussions with the ACRS, at the November 2, 1978 full committee meeting, the staff indicated that it "will accept

the Project's construction quality assurance surveillance in lieu of a base line preservice inspection provided that traceability exists for inspection reports and radiographs for the reactor and reactor heat transport system." This in no way alters the staff's position regarding periodic monitoring of the six hot crossover welds in the primary sodium system as soon as practicable, following development of the high temperature ultrasonic testing methods. This position is consistent with the ACRS report. Again, we reaffirm that the staff does not consider monitoring the secondary sodium piping system as equivalent to periodic examination of the high stressed primary sodium system welds, as suggested by the FFTF Project in a letter dated November 24, 1978. Therefore, the staff requests that it be kept fully informed of UT development plans, schedules and progress of the development of this method of examination, and when it is expected that it will be utilized on the reactor coolant system.

(b) Leak Detection

The staff had recommended that at least one sodium leak detector system, preferably a sodium ionization aerosol type, be seismically qualified and furnished with Class 1E power. This is consistent with Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection System." It appeared to the staff that the sodium ionization detector offered the best chance of meeting leak detection sensitivity and response time requirements for weeping sodium leaks under insulated piping. The Project has been willing to provide the ionization type leak detector but not safety-grade quality nor Class 1E power. This issue is now considered resolved. The staff accepts the Project's position of reliance upon control power for leak detection on the basis that branch line failures can be tolerated without compromising decay heat removal from the core through adjacent loops. This situation could be hypothesized following a safe shutdown earthquake when control power might not be available. In the case of the FFTF only, which is an as-built facility, the cost of upgrading these systems to safety-grade requirements would not be commensurate with the overall improvement of safety.

The staff has also modified its position regarding aerosol leak detection sensitivity and response time based upon the Project's experimental verification of aerosol diffusion through piping insulation jackets which showed that aerosol leak detectors will respond to weeping leaks inside the insulation jacket in reasonable time intervals. Similarly, the staff accepts the Project's commitment to demonstrate the function of the leak detector system through the use of ionizing gases and ionizing type leak detectors in simulated tests.

(2) Electrical Penetrations Through Containment

Containment vessel penetrations which had not met the required normal, seismic and accident environmental qualifications have been modified. The protection of electrical power penetration through containment had still been under review by the staff at the time of the September 7, 1978 ACRS meeting. The staff has reevaluated its position in this matter, referred to in Regulatory Guide 1.63, "Electric Penetration Assemblies in Containment Structures for Light-Water-Cooled Nuclear Power Plants." This Regulatory Guide first came into existence in 1973, was revised in May 1977, and again in July 1978. The last version of this Regulatory Guide stipulates that it is applicable to construction permits docketed after August 31, 1978. Since the FFTF was docketed in October 1970, current guidance is waived. The FFTF electrical penetrations as presently installed provide adequate safety and are acceptable because the penetration module conductors are oversized such that current limiting protection is provided at the containment penetration through the first breaker or backup breakers before excessive wire temperatures are reached. (Reference FSAR Q. 040.29 response.)

(3) Post Accident Radiation Monitors

The Project has agreed to provide safety-grade, post-accident radiation monitors outside the containment. This matter is considered resolved.

(4) Closed Loops

By mutual agreement with the Project, we have discontinued our review of the interaction of the core with an experiment in a closed loop and of the capability of the closed-loop plant protection system to protect the reactor. At the close of the November 2, 1978 ACRS meeting, the staff requested that the Project inform the Commission of their future plans regarding the safety review of the closed loops for the FFTF. It is our understanding that they may not be installed for several years.

(5) Loose Parts Monitoring

The staff has recommended installation of a loose parts monitoring system in the FFTF when it is developed. The Project has indicated its intention to do so as soon as a system is developed that is practical to install. An experimental program is underway at Argonne National Laboratory. No further action is anticipated by the staff on this item. The staff requests that it be informed when this modification is made or if a decision is made not to implement the recommendation.

(6) Tornado Watch

Due to uncertainties existing with tornado missile protection, the staff has recommended shutdown of the facility when tornado watch conditions have been established for the immediate vicinity of the FFTF. The Project has agreed to this. The commitment alleviates a number of staff concerns regarding the use of non-Class 1E power for functions normally considered safety-related. The staff accepts this commitment.

(7) Control Room Habitability

The staff has recommended the installation of radiation, sodium aerosol and chlorine detectors, powered by Class 1E power and missile-protected, located ahead of the air intake to the control room, greater separation between air intakes, and inclusion of charcoal absorbers in the control room heating and ventilation system. The Project has agreed to provide the above detectors at air intakes with missile protection and Class 1E power except for the chlorine detector, which will be on control power. Improving the physical separation of air intakes is in process. Charcoal absorbers will not be provided, since they would be ineffective with sodium iodide. Normal FFTF control room operation is in the pressurized mode with optional control room isolation and/or internal air recirculation. The staff is in accord with this arrangement and considers this issue resolved.

(8) Fuel Channel Exit Temperature

The staff recommended the use of a fuel channel exit temperature of 1670°F for safety analysis rather than the proposed 1705°F. The staff's concern is related to the fact that a significant fraction of the pressure drop across the fuel subassembly is along the long fission gas chamber in the upper half of the fuel pin. The fuel channel exit temperature is thus not a good indicator of the fuel and clad temperatures in the power production region. We believe it to be more conservative to use the lower value until better data are available. The Project has agreed to this recommendation. The staff considers this matter resolved.

(9) Project Initiated Plant Modifications

Plant modifications initiated by the Project are identified in Table 1.3-1 of the Final Safety Analysis Report. Twenty-seven changes were listed by the Project contractors, along with the reasons for the changes. These changes were reviewed during the normal course of the safety review. We believe the reasons given in two instances deserve some further comments. Considering their importance, and for completeness they are restated:

- (a) Hardening of the diesel generators against tornados was eliminated. We view this decision to have been a departure from the intent of 10 CFR Part 50 and an acceptance of an increase in risk, not necessarily unsafe but certainly not improving the safety of the FFTF. The acceptance of this design decision is coupled to the demonstration and verification of natural circulation, which at this time remains an open safety issue until satisfactory acceptance tests have been completed. The Project has been made aware of our concerns. The staff conditionally accepts the Project's position pending the outcome of the natural circulation tests.
- (b) Tornado missile protection was provided for one secondary loop only. We believe it would have been prudent to have added the same protection to a second loop, although this was not done. We do not view one missile-hardened loop as meeting the intent of the design criteria in 10 CFR Part 50. We recommend that operation of the facility not be continued with any perturbation of the flow in the one hardened loop, such as erratic behavior of the primary check valve. Two-loop operation is outside the scope of the staff's review, but we have considered the safety consequences of accidental loss of one or two loops during three-loop operation. This is also related to natural circulation. Considering the foregoing stipulations, the planned three loop operation is acceptable for the FFTF.

1.8 Requirements for Future Technical Information

(1) Natural Circulation

In the Safety Evaluation Report, the staff conditionally accepted the premise that there is reasonable assurance that natural circulation will be demonstrated as a practical method of removing decay heat for the initial oxide cores and operating temperatures of the FFTF. Although the principle of natural circulation is not in dispute by the staff, the uncertainty regarding natural circulation relates to transient thermal behavior and also involves uncertainties in low flow behavior in the core during the transition to stable natural circulation, hot channel factor values, the fuel cladding temperature, and its time history in the core during this whole process. We also have some concerns regarding reestablishment of natural circulation, if once lost, in a timely manner, since multiple interconnected loops are involved and the thermal inertia may cause sluggish delayed response in overall system heat transport. For these reasons, which are safety-related, we have requested that the Project furnish us with the test specifications including those revisions to be implemented during the startup phase of testing. The updated natural circulation test specifications were received February 9, 1979 and are now under review. Further amplification of the

staff's advice and recommendations on this subject are given in Section 19.3A.

(2) Containment Adequacy for Core Melt Events

This issue is discussed in Sections 15.3.6, 15.3.7, and 19.3B. Supplemental information is also provided in Appendices C, D, E, F, G, and H. The staff finds the program proposed by the Project (Ref. 7 and Appendix D) in response to a request for plans and schedules addressing Commission and ACRS advice pertaining to startup and operation of the FFTF to be inadequate.

Specifically, the schedule and funding of the program for the HEDL Containment Air Cleanup Systems (Ref. 7, Encl. II) is, in our view, not compatible with the FFTF operating schedules and staff and ACRS advice.

The staff requests that the Project provide timely and definitive implementation plans and schedules to respond to our advice and recommendations provided in Section 19.3B of this supplement.

(3) Leak Detection

We have requested a demonstration of the performance of large-scale gaseous (or aerosol) sodium leak detection systems during the preliminary startup phase of the FFTF. More specifically, we have requested a demonstration of a large leak detection system because of the scale-up of the system size from laboratory size systems utilizing ionization detectors and differential pressure detectors. The Project agreed to in-plant testing in Supplement 16. We request that we be informed of the test results. Supplemental information is furnished in Appendix C, Attachment IV.

(4) Inservice Inspection

This subject was covered in Section 1.7, Item (1)(a) of this supplement and additional information is furnished in Appendix C, Attachment VI. We request that we be advised in writing of the Project's disposition of this matter with respect to future examination of the selected high stress welds in the hot crossover piping of the reactor coolant system.

(5) Cell Liner Test Plan

In our Safety Evaluation Report, we stressed the importance of cell liners to provide the function of preventing a sodium/concrete interaction in the event of a sodium spill. We have requested that the Project provide a test plan to periodically demonstrate the preservation of the functional capability of these liners during the life of the facility. The Project has provided a test plan for cell liners by letter dated January 31, 1979. We find the proposed test plan acceptable.

(6) Loose Parts Monitoring

We have recommended that the Project install a loose parts monitoring system. The Project is in agreement with this recommendation but believes the decision should be made when the development work is completed. We request to be kept informed regarding the ultimate disposition of this item.

1.9 Summary of Active Issues

- (1) Natural Circulation Verification - To be confirmed by reactor operation testing. (See Section 1.8(1))
- (2) Containment Adequacy for Core Melt Events - This issue remains active and is discussed in Sections 1.8(2), 15.3.6, 15.3.7, 19.3B and supplemental information is furnished in Appendices C, D, E, F, G, and H.
- (3) Closed Loops - The closed loop safety evaluation is not part of this review. (See Section 1.7 (4))
- (4) Conventional Fire Protection - Staff recommendations regarding non-sodium fire hazards are included in Appendix I. These recommendations were not part of the ACRS review.

3.0 DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

3.2 Classification of Structures, Components and Systems

3.2.1 Seismic Classification

Page 3-3, seismic classification of seal oil pumps. These pumps need not be seismically qualified based on the tests conducted which demonstrated the capability of the main sodium pumps to operate through flow coastdown. This issue is resolved. (See Section 19.4-3.)

Page 3-4, seismically qualified leak detectors. The staff has accepted nonseismically qualified leak detectors for the FFTF. This issue is resolved. (See Section 19.3C.)

3.2.2 System Quality Group Classification

Page 3-5. The staff has made the recommendation to upgrade the emergency control air to tornado harden the reserve air system to the air dump heat exchangers from Quality Group D to Quality Group C. The Project has agreed to do this and this issue is considered resolved. (See Section 19.4-4.)

3.5 Missile Protection Criteria

3.5.1 Missiles Generated Outside of Containment

Page 3-8. The staff has made the recommendation to shut down the FFTF if a severe tornado watch weather condition exists. The Project has accepted this recommendation. This issue is resolved. (See Section 19.3H).

3.8 Design of Seismic Category I Structures

3.8.2 Containment Interior Structures

Page 3-16. Bechtel through the Project Office has supplied supplementary analysis to show that the interior IHX cells could withstand short term pressures greater than 10 pounds per square inch gauge design pressure; the staff finds this supplemental analysis acceptable.

3.10 Seismic Qualification of Category I Instrumentation and Control Equipment

Page 3-24. Typical seismic qualification test reports were received on May 2, 1978 and upon review were found to be acceptable to the staff.

3.11 Environmental Qualification of Safety-Related Instrumentation and Control Equipment

Page 3-25, Item 7. Some plant protective system instrumentation may be subjected to external post-accident pressure greater than atmospheric - Table 3.10-7 in the FSAR lists this equipment. The staff has been provided information which substantiates that the protection system functions will not be impaired by high cell pressures from sodium spills. We find the Project's evaluation acceptable.

Page 3-25, Item 8. Environmental qualification of undervoltage relays. These relays are located in the control building-Room 104, outside of containment. This area is not subjected to abnormal in containment accident environmental conditions. In addition, a transformer failure in the relay results in a safe condition. We find this answer acceptable.

Page 3-26, Item 9. This question relates to the environmental qualification of post-accident monitoring. The Project has committed to environmentally qualifying the ex-containment radiation monitors and supplying them with Class 1E power. The staff accepts this commitment.

Page 3-26, Item 10. The staff questioned the monitoring of environmental conditions in which safety-related equipment is located - the Project has responded that physical separation of PPS equipment does not subject the complete system to a local heating and ventilation system malfunction or failure. A plant-wide failure of the heating and ventilation system would shut down the facility. In addition, roving operating personnel pass through accessible areas at least once a day and would correct a local malfunction if it occurred. The staff accepts these provisions for the FFTF.

Page 3-26, Item 11. The staff requested sample copies of environmental test reports. Test reports were supplied by the Project for pressure sensors, flux sensor, flux instrumentation, CIS radiation monitors, and secondary PPS comparators and logic. The staff finds the sample test reports acceptable.

4.0 REACTOR

4.4 Thermal-Hydraulic Design

4.4.2 Thermal-Hydraulic Limits

Page 4-21. The staff recommended a sodium fuel channel outlet temperature limit of 1670°F for an upper limit on sodium fuel channel exit temperatures, rather than 1705°F which the Project was using under low flow conditions. The Project agreed to the staff recommendation and the issue is resolved.

4.4.9 Loose Parts Monitoring System

Page 4-31. The staff recommended that a loose parts monitoring system be installed in the FFTF. The Project has agreed in principle once the loose parts monitoring system is developed. The staff requests that it be kept informed of the implementation of this recommendation but the Project's commitment is acceptable.

4.4.10 Evaluation and Findings

Page 4-32, under "Natural Convection Testing" - The staff requested verification of IANUS, FLODISC, VARR-II. See Sections 1.8(1) and 19.3A for further discussion of this subject. The Project's commitment to provide test information is acceptable.

5.0 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5.2 Heat Transport System

5.2.7 Heat Transport System Materials Evaluation

Page 5-11, Item 1. The staff recommended a baseline inspection be performed on selected highly stressed welds in the primary piping system. In our letter to the ACRS (October 26, 1978) the staff agreed to accept the Project's construction quality assurance program in lieu of a baseline preservice inspection (page VI-2, item 2). However, the staff still recommends that a volumetric examination of the six hot crossover piping welds be performed at an appropriate time early in plant life. This position was stated again by the staff at the November 2, 1978 ACRS meeting. (See Section 1.7(1)(a) and Appendix C.)

Page 5-11, Item 2. The staff had recommended venting of the annulus around the piping to the cell based upon information available. The Project, at an earlier subcommittee meeting, and on September 26, 1978 had stated that newer data existed which indicated very little difference in aerosol release between vented and nonvented piping. Documentation from the Project was received on November 16, 1978. This item was also discussed in Section 5.2.7, page 5-11 of NUREG-0358. The Project has committed to perform system tests to demonstrate the sensitivity and functional performance of the aerosol leak detection system. Based upon the preceding agreements and commitments, this item is considered resolved. (Ref. 10)

Page 5-11, Item 3. The Project has committed to a comprehensive materials surveillance and testing program which includes examination of removable reactor components (closed loops) and test coupons. They have instituted an out-of-pile test program for evaluating the fracture toughness of stainless steels. On the basis of the Project's planned program and commitments, this issue is considered resolved. (Ref. 8)

5.4 Surveillance and Inservice Inspection

5.4.4 Recommendations

5.4.4.1 Leak Detection

Page 5-16. The subject was previously discussed in Section 5.2.7, page 5-11 item 2.

5.4.4.2 Material Surveillance

Page 5-16. This subject was previously discussed in Section 5.2.7, page 5-11, item 3.

5.4.4.3 Preservice Inspection

Page 5-16. This subject was previously discussed in Section 5.2.7, page 5-11, item 1.

5.5 Decay Heat Removal System

Page 5-18. The staff recommended a test program to demonstrate that the main heat transfer pumps can run at pony motor speeds without forced lubrication - this is not currently a requirement unless the natural convection tests prove to be unsatisfactory. As such, the staff accepts the current pump design for flow coast down.

Page 5-19. The staff recommended bringing the plant to shutdown status under tornado watch conditions. The Project agreed to this recommendation.

6.0 ENGINEERED SAFETY FEATURES

6.4 Control Room Habitability

Page 6-12. The staff has recommended, for the control room, that the distance between air intakes be increased, that the air intakes be tornado-hardened, and that chlorine and sodium aerosol detectors be provided at each air intake. The Project has agreed to do this. An activated charcoal bed will not be provided because it would be ineffective against sodium iodide. The chlorine detector will be placed on control power since the source of chlorine is offsite. The sodium aerosol detector will be on IE power. The staff is in agreement with the Project's position on this item.

7.0 INSTRUMENTATION AND CONTROLS

7.1 Summary Description

Page 7-2, Item 5. The staff requested a comparison between RDT standards and IEEE Standard 323-1974. The Project's response was that RDT standards did not exist for environmental qualifications nor was IEEE Standard 323-1974 used by FFTF. The most severe environmental condition that is postulated to occur is given in Section 3.10 of the Final Safety Analysis Report. All protective system functions will have been performed before the time any of these severe environmental conditions are reached. This is acceptable for FFTF assuming natural circulation proves viable.

7.2 Reactor Shutdown System

Page 7-7, Item 6. The staff requested information regarding the limits of design for radiation for PPS instrumentation. The Project has no expectations of instrumentation performance beyond those conditions specified in the FSAR. This is acceptable.

Page 7-8, Item 12. The staff recommended that, due to all welded construction, PPS sensors be verified by intercomparison of responses. The Project believes there is no need to do this, but the capability exists on the basis that PPS sensors will be periodically checked by inter channel comparison using the plant computer in any event. The staff finds this acceptable.

7.4 Systems Required for Safe Shutdown

Page 7-12, Item 1. The staff evaluation of the closed loops has been deferred and is not included as part of this review.

7.5 Safety-Related Display Instrumentation

Page 7-12, Item 1. The Project has committed to provide a method of determining shutdown status. The method to be used has not been selected. The staff accepts the Project's commitment to provide a method of determining shutdown status. The staff requests that it be informed of the Project's decision and accepts their commitment.

7.7 Control Systems Not Required for Safety

Page 7-18, Item 3. The staff's concern deals with closed-loop and core interaction. This review is being deferred to the future. It is a safety-related system.

Page 7-18, Item 4. Item 4 was found acceptable in the Safety Evaluation Report but was not deleted from the last paragraph. It is acceptable as stated.

8.0 ELECTRIC POWER

8.3 Onsite Power Systems

8.3.4 Environmental Qualification of Class 1E Equipment

Page 8-5. We find the information submitted in Amendment 27 to the FSAR regarding environmental qualification of mechanical and electrical components for review to be acceptable.

8.3.6 Electrical Penetrations

Page 8-5, electrical penetrations through containment. Regulatory Guide 1.63 was first issued in October 1973. The FFTF was docketed in October 1970 and was exempt from current Regulatory Guide guidance. This subject, was discussed more fully in Section 1.7, Item 2. In any event an unsafe condition for the FFTF does not result using the early guidance provided at the time of its design (1970).

9.0 AUXILIARY SYSTEMS

9.3 Process Auxiliaries

9.3.1 Compressed Air System

Page 9-8. The staff has recommended to the Project that the standby air cylinders and associated piping for the hardened DHX be upgraded from Quality Group D to Quality Group C. It is our understanding that the Project will modify the control air system to comply with this recommendation. This is acceptable.

9.4 Heating and Ventilating Systems

9.4.1 Control Room Area Ventilation System

Page 9-12. The staff has recommended that the control room air intakes and exhausts be missile-protected. The Project has agreed to this recommendation. The subject is also discussed in Sections 1.7(7) and 6.4. This issue is considered resolved based on the Project's commitments.

9.5 Service Systems

9.5.1 Fire Protection System

Pages 9-12, 9-14 and 9-15. A fire protection site review was conducted on December 13 and 14, 1978. The staff's advice regarding conventional fire protection and emergency lighting is contained in Appendix I. These recommendations were not part of the ACRS review.

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15.0 ACCIDENT ANALYSIS

15.3 Analysis and Evaluation of Core Meltdown and Disassembly Accidents

15.3.6 Containment Response to a Core Meltdown Event

The issue of containment margins for core melt events was one of the principal outstanding items remaining at the time of publication of the staff's Safety Evaluation Report (August 1978). In Sections 1.9, 15.3.6, 15.3.7, 19.3 and Appendices C, D, and F of the Safety Evaluation Report the staff provided detailed evaluations, advice and recommendations regarding the adequacy of containment margins for postulated core meltdown events. Since publication of the Safety Evaluation Report the Project has provided additional information (Ref. 7) in the areas of sodium-concrete interactions and hydrogen disposition. The staff has evaluated this additional information and concludes there is insufficient basis for the staff to change its previous conclusions, advice and recommendations in this area.

We have also asked our consultants at the Materials Sciences Laboratory of the Aerospace Corporation to evaluate the additional information provided by the Project, as well as the entire area of sodium-concrete interactions, and their review is provided in Appendix E. As is evident, our consultants disagree with many of the Project's conclusions and recommend that the staff continue to use its earlier estimates contained in the Safety Evaluation Report of sodium-concrete interaction rates and assumptions regarding the disposition of hydrogen.

In addition to the above, the NRC Division of Reactor Safety Research is supporting a confirmatory research program at Sandia Laboratories on sodium-concrete interactions. We have asked the researchers at Sandia Laboratories to assess the recent Project information in the area of sodium-concrete interactions and hydrogen disposition, and their comments are provided in Appendix F. As indicated in Appendix F, Sandia also disagrees with many of the Project's conclusions in this area, especially in light of a recent test (No. 13) conducted at Sandia in which substantial penetration of sodium into the FFTF type of firebrick and concrete took place. Appendix G provides a brief description of the Sandia Test No. 13. With respect to sodium-concrete interactions, this test was probably the most prototypic of FFTF conditions performed so far, and it indicated that the rate of penetration of sodium into concrete is consistent with the estimates used by the staff in its FFTF containment margins evaluations. As mentioned in Reference 7, the Project's recent evaluation did not include consideration of the Sandia Test No. 13 because the data were not available in time. (Ref. 18)

The staff's advice and recommendations of the FFTF containment margins are reiterated in Section 19.3B.

15.3.7 Radiological Consequences of Core Disruptive Accidents

Subsequent to issuance of the Safety Evaluation Report, the staff performed a supplemental radiological assessment of postulated core meltdown events. This supplemental assessment which was transmitted to the ACRS on October 26, 1978, and also presented to the ACRS on November 2, 1978, is included as Attachment I to Appendix C. Additional discussion on this item, including the staff's advice and recommendations, is provided in Section 19.3B.

16.0 TECHNICAL SPECIFICATIONS

Technical specifications are subjected to many changes, and since the Commission does not have any direct authority over the operation of the FFTF, the Commission will provide general advice in response to the request by the Department of Energy regarding technical specifications. We understand that those technical specifications that are in the Final Safety Analysis Report are currently being revised. Our advice, given in our Safety Evaluation Report, is all that we are able to provide under these circumstances with one major exception. This exception concerns operation with failed fuel. It has been the staff's understanding that the Project would not initially operate the reactor with failed fuel. During the September 7, 1978 ACRS meeting (transcript pages 131 through 142) considerable discussion occurred concerning this point. The staff has performed its radiological evaluation on the basis of the release of fission gas from one percent failed fuel into the cover gas and with operation of the RAPS cleanup system. The staff does not view the one percent failed fuel as an initial operating technical specification limit without knowledge of the nature and cause of the fuel failure. We believe that the Project should review existing DOE documents and past statements made by the Project and proceed accordingly.

- (1) FFTF Environmental Statement, WASH 1510 dated May 1972, quote:

"An estimate of the amount of gaseous leakage from FFTF has been made by combining the data on inventories (Table IV.A.7.2) and the data on leakage (Table IV.A.7.3). If the reactor is operated with 0.1% of the fuel pins defective and 35% of the gas is actually released from the defected pins, the leakage is approximately 0.003 Ci/day. If the two closed loops are operated with 0.2% defective pins and 35% of this gas is released, an additional leakage of approximately 2×10^{-4} Ci/day would be expected. This would be a typical expected operating condition. The radioactive gas waste system is designed to accommodate 1% fuel failure in the reactor core and complete failure in all closed loops - a large margin above expected operating conditions." (emphasis added)

- (2) FFTF FSAR, Vol. 5, Page 12, 1-19, quote:

"The inventories of noble gases in the various portions of the reactor and the gaseous waste system used for estimating system leakage are given in Table 12.1-6. The values are based on operation with up to 0.1% of the fuel in the FFTF core defective with 35% of the fission

gases escaping. A similar inventory is presented for a typical closed loop with 0.2% of fuel defective and 35% gas released."

(3) FFTF FSAR Vol. 7, Appendix C, Page C.11-15, Paragraph C.11.2.5, quote:

"Operator Actions in Response to Failed Fuel Monitoring (FFM) System Alarms"

"This section outlines the operator actions in response to alarms from the FFM System for the abnormal conditions that will exist when fuel failure occurs. A more detailed set of instructions can be found in SDD-94 and SDD-31.

Operator actions with respect to the reactor are based on the assumption that the FFTF will not be operated for extended periods with known fuel failures even though the plant is designed to accommodate failures up to 1% of the fuel pins. This constitutes a conservative approach which is principally intended for initial operations. Once sufficient operating experience is accumulated, these procedures may be modified to permit continued operation with failures where only fission gas is released. However, during this period, if fuel particle release is detected, the assembly will be replaced. Further on in the plant lifetime, after operating experience with fuel leaking only fission gas has been acquired, removal of an assembly after detection of fuel particle release may be delayed.

It is recognized that fuel pins defected before power operation cannot be detected until generation and release of radioactive fission gas occurs. However, failures with only a few days irradiation are readily detectable, as proven in tests of the FFM in EBR-II in detecting failure of assembly X 232 after 5 operating days.

Operator actions are governed by the type of alarm(s) annunciated and the severity of the indicated failure. The following discussion summarizes operator actions in decreasing order of severity of the failure according to the types of alarm received in the reactor control room.

The detection of delayed neutrons is indicative of a fuel cladding failure which has exposed fuel to the sodium coolant. Such a failure is expected to depressurize the fuel pin and release the fission gas inventory. Therefore, delayed neutron alarms will normally accompany or follow on alarm from the fission gas monitor.

A high count rate on one or more of the delayed neutron monitor (DNM) channels coincident with a RCGM high alarm constitutes a confirmation of fuel failure. After verifying that cover gas sampling (Gas Tag Sampling Trap) has been initiated, the operator will initiate a controlled reactor shutdown. If,

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during execution of the cover gas sampling procedures, reactor or failed fuel instrumentation indicates a worsening condition within the reactor, a reactor scram will be initiated and sampling carried out following the scram. After cover gas sampling is completed, the plant will be cooled down to refueling conditions. The fuel element containing the failure will be removed following identification of the assembly by the Failure Location System.

In parallel to the foregoing procedure, three 30 to 40g sodium grab samples will be taken and analyzed for fission products. Finally, if installed, the delayed neutron monitoring (DNM) system procedures will be initiated. With the information from these sources, the identity of the assembly containing the failure will be determined.

If only one channel of the DNM exceeds the high alarm setting and there is no high radiation cover gas monitoring (RCGM) countrate, then there is likely a malfunction in that channel. Any pin failure detected by the DNM should also be identified by the RCGM since pin failure will result in a release of the fission gas inventory. In order to verify that the DNM channel is malfunctioning, the procedures discussed above (cover gas and sodium samples) will be implemented to check for fuel failures.

The detection of fission gas isotopes without a corresponding detection of delayed neutrons is indicative of a fuel failure in which no fuel has been exposed to the sodium. During the cover gas sampling procedure, the operator will monitor reactor and failed fuel instrumentation to see if conditions within reactor are worsening. If any of these conditions cause another alarm or exceeds an operational limit, the above procedures to scram the reactor will be implemented. If they continue to give normal readings, the procedures to characterize the fuel failure will be implemented as discussed in Section C.11.2.3. After verifying that three cover gas samples have been obtained with the gas tag sampling trap, the reactor will be shut down to hot standby. The plant will then be cooled down to refueling temperatures. Since it requires about five hours to obtain the three cover gas samples, the reactor will remain at power for a long period after the fission gas leak is first detected. As stated at the beginning of this section, these procedures represent a conservative approach for the initial operation of FFTF. They may be modified to permit operation with failed fuel as reactor operating experience is accumulated."

The staff is basically in accord with the procedure for operation described in Appendix C of the FSAR by the Project as quoted above. Consequently, the technical specifications in the FSAR do not have a permissible numerical value for failed pins under which full power operation is permitted.

We do not advocate a change at this time permitting initial FFTF operation with one percent failed fuel since this would allow $1\% \times 217 \times 74 = 160$ pins $\sim 73\%$ of the pins in one subassembly to be failed. We believe proceeding to that point at this time is premature and one should proceed more cautiously and we recommend that the procedures outlined by the Project for initial operation not be changed (.1% failed fuel limit).

- (4) Items recommended for the Project to consider in their technical specifications:
- (a) Requirement to periodically monitor the integrity of the cell liners to show continuing functional capability.
 - (b) Not operate the facility with flow perturbation in the tornado hardened loop, such as erratic check valve behavior. Prior to startup cycle increase the flow to flush check valves by ramping the pumps.
 - (c) Reassess plant protective system setpoints after acceptance test program.
 - (d) Shut down reactor when tornado (watch) conditions exist.
 - (e) Limit the fuel channel exit temperature to 1670°F rather than 1705°F.
 - (f) Maintain a fuel surveillance program for the initial fuel cycles.
 - (g) Periodic natural convection verification.
 - (h) Periodic examination of hot crossover piping welds.
 - (i) One-tenth percent failed fuel limit.

18.0 REVIEW BY THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

In the matter of considering the proposed operation of the FFTF the ACRS, in its review of the FFTF, held subcommittee meetings on July 12 and August 10, 1978, and full committee meetings on September 7 and November 2, 1978. A copy of the full committee's report, dated November 8, 1978, is included as Appendix B to this supplement. The Committee indicated that:

- (1) "In view of the fact that this reactor is unique, appropriate detailed procedures and standards of the kind used in the review of light water reactors were not available for the review of FFTF." The staff believes that the guidance provided by 10 CRF Part 50 Appendix A, modified where necessary, has provided an adequate basis for the safety evaluation of FFTF.
- (2) "Because the FFTF is a DOE facility the scope of the NRC review was defined by the DOE request that NRC provide advice regarding the adequacy of the FFTF design and technical specifications to ensure safe operation." NRC has provided its advice and recommendations in its Safety Evaluation Report, NUREG-0358, dated August 1978, and this supplement, dated May 1979 in response to DOE's request.
- (3) "The NRC review did not include construction audits, assessments of the "as built" configuration, or evaluation of acceptance test results that would verify that the plant was constructed in accordance with the design criteria and documentation. Provisions for safeguards and security were also excluded from review." The evaluation of the closed loops was also deferred. When the evaluation of the closed loops is requested by DOE, the ACRS will be informed.
- (4) "The NRC Staff agrees with the Project that failure of the reactor inlet piping need not be considered as a cause of a core disruptive accident (CDA) provided that certain conditions are met. These include preservice and inservice inspection of piping, leak detection instrumentation, and an appropriate materials surveillance program. The Project has committed to developing an ultrasonic testing (UT) device for high temperature use, to inservice inspection of selected, high stress welds of the secondary sodium loops when the UT device is available and to installation of aerosol leak detection systems in the heat transport cells. A materials surveillance program will be conducted using test subassemblies, as well as in-reactor and hot leg components that are removed. In addition, the NRC recommends that a cold preservice inspection be conducted on the hot crossover piping welds and

that inservice inspections be implemented as soon as practicable. The Committee supports these recommendations." The staff had recommended a pre-service inspection of primary system welds considering the status of construction (six hot crossover piping welds). In its letter to ACRS of October 26, 1978, the staff modified this position agreeing to the acceptance of construction inspection reports and radiographs in lieu of preservice UT examination. The staff, however, did recommend that the six primary system hot crossover piping welds be monitored as soon as practical when the high temperature UT method was developed. The staff did not agree that monitoring the secondary system piping only, without any primary sodium system inservice inspection monitoring, was sufficient. The staff believes its position is consistent with the intent of the advice provided by the ACRS. The purpose of performing a baseline inspection on the primary system six crossover pipe welds early in plant life is to establish a reference data base for possible later use. The opportunity for a cold preservice inspection has passed and it has become impractical due to preheating and filling the primary loops with sodium in the FFTF.

- (5) "The Project has performed studies of various postulated core disruptive accidents (CDA). The NRC Staff has concluded that the calculated prompt energetics from CDAs are within the capability of the containment system. The ACRS concurs with the Staff conclusion." As stated in our Safety Evaluation Report and in our presentation to the ACRS, both the Project and staff had concluded that the CDA energetics would not likely exceed the capability of the reactor coolant system.
- (6) "The FFTF does not have a Class 1E power supply to provide decay heat removal. Instead, the Project will depend upon natural convection cooling in the event of loss of offsite power and failure of the onsite diesel generators. The Project's calculations indicate that natural circulation will provide decay heat removal. It is proposed that the natural circulation decay heat removal be measured during the startup testing. The ACRS concurs that the adequacy of the decay heat removal by natural circulation should be experimentally verified." The ultimate heat sink for decay heat removal in the FFTF is by natural convection to the air cooled heat exchangers. The staff has recommended that natural convection and the thermal hydraulic codes being used by HEDL be verified experimentally using nuclear heat during startup testing. The Project has agreed to do this and provide the staff with a final test plan, preprediction results, summary analytical results, and supporting data pertaining to the test model verification and include a 35% power test point in addition to the other power levels proposed by the Project. The Project has also agreed to a periodic verification of natural convection and clarification of acceptance criteria.

- (7) "The NRC Staff and the Project have not yet agreed on the adequacy of containment for dealing with the consequences of some low probability accidents which lead to the potential for generation and release to containment of significant quantities of sodium aerosols, hydrogen and other volatile gases. In its report of July 15, 1975, the ACRS recommended that consideration be given to the possible usefulness of sand-and-gravel filters for the removal of airborne particulates. During the current review, the NRC staff has recommended that measures be taken by the Project to permit the measurement and control of the hydrogen concentration in the containment, to further reduce the chance of a damaging explosion. The NRC Staff has also recommended that means be included for controlled venting. The ACRS supports these Staff positions and recommends development of additional mitigation measures, such as a sand and gravel filter, for possible addition to the controlled venting system to provide still further assurance of limited radioactivity releases in the event that one of these low probability accidents should occur." The staff and the Project have not agreed on the adequacy of the existing design in relation to its containment margin for the low probability of a melt-through accident. As stated in the staff's December 15, 1978 letter to the Project (Items 3 and 4) Appendix C, we are concerned about the core debris - containment materials interaction rate, their experimental verification and the apparent divergency in experimental results. We are also concerned about the control of the hydrogen concentration in the containment before the pressure rise in the containment due to reaction products and heating. We recommend that the necessary parameters be monitored from the control room such as temperature, pressure, hydrogen concentration, percent oxygen and radiation field. Since hydrogen evolution is computed to reach critical concentrations before the design pressure of the containment is reached, different mitigating features may be required to protect the containment. The staff has advocated controlled venting to levelize containment pressure and radiological release. To further reduce the radiological releases the staff agrees with the ACRS advice of using scrubbers and/or filters in conjunction with the recommended controlled venting (see section 19.3B of this supplement).
- (8) "As pointed out in a previous report, the ACRS has recognized that the FFTF is a special test facility located on a favorable site. Both positive and negative aspects of this situation have been considered throughout the review. The ACRS believes that if due regard is given to the matters mentioned above, and in previous reports, it is acceptable for startup and operation of the FFTF to proceed." The staff advice and recommendations regarding the startup and operation of FFTF are given in Section 19.3B.

19.0 SUMMARY OF ADVICE AND RECOMMENDATIONS

19.3 Advice and Recommendations

A. Natural Circulation Decay Heat Rejection

The advice provided in the Safety Evaluation Report was supported by ACRS. Natural convection should be measured during startup testing using nuclear heat. This subject is also discussed in Sections 1.8.1, 4.4.10, and 18 (6) of this supplement. The staff's position was further delineated in its December 15, 1978 letter to the Project (Appendix D) quoted as follows:

"(2) Plans for Startup Testing

The NRC Staff and the ACRS are in accord that natural convection through the heat transport systems will have to be demonstrated using nuclear heat. The staff will review and evaluate the adequacy of information obtained from the natural circulation test program, including both natural circulation and computer model verification and DOE's analysis of these data in support of the position that natural circulation is a viable option for decay heat removal.

The staff, therefore, requests that it be kept informed of plans and schedules for startup testing, including natural circulation modes of operation, and any decisions or changes to the proposed FFTF Natural Circulation Test Plans."

This issue remains active for future resolution following startup.

B. Containment Margins for Core Melt Events

Our Safety Evaluation Report addressed the problem of the adequacy of containment margins for core melt events. The staff found that the challenge to containment from this low probability event is primarily controlled by hydrogen concentration initially and then by the buildup of containment pressure. It is the staff's judgment that autocatalytic recombination of hydrogen at low rates of energy release cannot be assured for the duration of this type of accident. In the Safety Evaluation Report the staff provided its advice and recommendations to alleviate the above concerns, but the Project has not yet provided a satisfactory response to the staff in these areas.

The ACRS in its July 15, 1975 letter, and again in its November 8, 1978 letter, recommended that consideration be given to the use of sand and gravel filters to assist in limiting radioactivity releases in the event of a low probability accident. We again recommend that the Project accelerate its R&D and design efforts in the area of controlling and monitoring hydrogen concentration within containment and the implementation of controlled venting with scrubbers and/or filters to control containment pressure and radiological releases from containment. Instrumentation readout in the control room should be provided as necessary.

The staff's advice is that this equipment be in place in the FFTF prior to unrestricted full power operation. The objective is to further reduce the radioactivity releases in the highly unlikely event of a core melt-through accident. The staff finds the program proposed in the Project's letter of January 26, 1979, in response to the staff's requests for plans and schedules addressing NRC and ACRS advice pertaining to startup and operation of the FFTF to be inadequate. Specifically, the schedule and funding for the program as specified in Enclosure II of the Project's letter for the HEDL Containment Air Cleanup Systems is, in our view, not compatible with FFTF operating schedules and the staff's and ACRS' advice. The staff will not endorse continued operation of the FFTF beyond startup and natural convection testing without adequate measures being in place to augment existing containment margins and control radiological releases from a low probability core melt-through accident.

C. Piping Integrity

The staff's recommendation in the Safety Evaluation Report was modified for the FFTF during the ACRS review. The staff's position was further delineated in its December 15, 1978 letter to the Project as Item 1 (Appendix D), quoted as follows:

"(1) Piping Integrity

The staff's summary position on piping integrity is documented in Section 19.3C of the Safety Evaluation Report. In a letter to the ACRS, dated October 26, 1978 and in discussions with the ACRS, at the November 2, 1978 full committee meeting, the staff indicated that it 'will accept the project's construction quality assurance surveillance in lieu of a base line preservice inspection provided that traceability exists for inspection reports and radiographs for the reactor and reactor heat transport system.' This in no way alters the staff's position regarding periodic monitoring of the six hot crossover welds in the primary sodium system as soon as practical, following development of the high temperature ultrasonic testing methods. This position is

consistent with the ACRS report. Again, we reaffirm that the staff does not consider monitoring the secondary sodium piping system as equivalent to periodic examination of the high stressed primary sodium system welds, as suggested by the FFTF project office in a letter to NRC, dated November 24, 1978. Therefore, the staff requests that it be kept fully informed of UT development plans, schedules and progress of development of this method of examination, and when it is expected that it will be utilized on the reactor coolant system."

D. Cell Liner Test Plan

As stated in the Safety Evaluation Report, the Project committed to developing a Cell Liner Test Plan at a meeting at NRC July 25/26, 1978. It is acceptable to the staff to proceed on the basis of the Project's commitment. A Cell Liner Test Plan was received by letter dated January 31, 1979. The Project commitment to proceed with this test plan is acceptable.

E. Loose Parts Monitoring

The staff accepts the Project commitment regarding this developmental item to be evaluated when it is developed and if practical to install in the FFTF.

19.4 Status of Miscellaneous Issues

(1) Closed Loops

The safety evaluation of the closed loops has been deferred and will be addressed when requested by DOE (estimated in late 1979).

(2) Conventional Fires

A Site Fire Protection Meeting was held on December 13 and 14, 1978. NRC's advice with respect to conventional fire protection is included in Appendix I.

(3) Seismic Qualification of Seal Oil Pumps

The staff's concern relates to flow coast down and seal preservation of the primary sodium pumps. The Project responded to this concern in Q001.49 Supplement 29. The response was found to be acceptable.

(4) DHX Reserve Air System

The Project has agreed to upgrade the DHX reserve air system from Quality Group D to Quality Group C. This item is resolved.

- (5) Environmental Qualification of Safety-Related Instruments
We find the information submitted in Amendment 27 acceptable.
- (6) H&V System Malfunction
This question related to the potential degradation of safety-related equipment. This subject was addressed in the response to Q30.60 Supplement 30. The staff finds the response acceptable.
- (7) PPS Instruments Exposed to High Levels of Radiation
This question relates to postulated abnormal events and was responded to in Q30.67 Supplement 30. The staff finds the response acceptable.
- (8) Recommend Comparative Time Response of All Non-Removable Safety-Related Sensors
The Project responded to this question in Q030.68 Supplement 30. An equivalent method for comparison is provided through comparison of signals from redundant channels and correlation with interrelated plant parameters. This method is acceptable.
- (9) Recommended Identification of Circuits in Containment Isolation System Using MIL HDBK 213 A&B.
The MIL HDBK was used primarily for methods and procedures and quantified values were not derived from this source. The response to Q30.70 Supplement 30 was considered acceptable.
- (10) Requested Documentation for Shutdown Status to be Displayed
The Project, in response to Q030.74, committed to providing a Class IE (PAM) System to verify shutdown status. We find the commitment to provide a safety grade system to be sufficient.
- (11) We proposed bellows seal pressure transducers for measuring containment pressure rather than an orifice type device.
The response provided by the Project to Q033.11 Supplement 22 was considered acceptable as a worst leak through the orifice would affect the leak rate at pressure by < 1%. This is considered acceptable.
- (12) We recommended that 18 valves be given type C Tests for leakage.
The response provided by the Project to Q022.02 was considered acceptable since they agreed to comply with the intent of the recommendation.
- (13) The staff requested supplemental environmental qualifications for IE electrical equipment and terminal block connectors.
Amendment 27 to Section 3.10 includes the environmental design of mechanical and electrical equipment by areas and systems. Safety-related equipment is

tabulated by area. Limiting environmental conditions within a given area can be found from the table. We consider this submittal acceptable.

- (14) We recommended independent and redundant valve position indication on the primary and secondary heat transport system isolation valves.
The Project response in Q040.9 and Q040.28 stated that normal position indication is open and that the power to valve operators is normally deenergized. A hot short would not shut the valves. The primary valves are not required to operate under accident conditions and redundancy in position indication is not required. Redundant circuitry for the secondary valves is not required since they are accessible and need not be immediately operated. The staff accepts this argument.
- (15) We recommended that the electrical power penetrations through containment be provided with circuit breakers rather than fuses.
The staff concluded that Regulatory Guide 1.63 Rev. 0 was complied with at the time FFTF was constructed and provides adequate protection for FFTF by means of oversize wire through the perforation plus breakers and fuses.
- (16) The staff previously accepted the use of control power for the air lock hatch indicator. This item is considered resolved.

19.5 Summary of Active Issues

- (1) Item A of Paragraph 19.3 (Natural Circulation)
- (2) Item B of Paragraph 19.3 (Containment Margins for Core Melt)
- (3) Item C of Paragraph 19.3 (Piping Integrity)
- (4) Item 1 of Paragraph 19.4 (Closed Loops)
- (5) Item 2 of Paragraph 19.4 (Recommendations regarding the conventional fire protection system are included in Appendix I).

20.0 SUMMARY-CONCLUSIONS

As a result of the staff's review of supplemental information submitted by the Project, including Amendments 31 through Supplement 30, the ACRS review and report and the staff's independent evaluation, described in the Safety Evaluation Report, NUREG-0358, August 1978, and this supplement, the staff concludes that if sufficient emphasis and due consideration are given to the implementation of the advice and recommendations stated in Sections 19.3 of the Safety Evaluation Report and this supplement there is reasonable assurance that the FFTF can be operated without undue risk to the public.

Certain stipulations regarding the operation of the FFTF have been delineated in the Safety Evaluation Report and this supplement, which are based on the design and operating characteristics set forth in the FFTF FSAR submitted by the Project. These stipulations include initial conditions for reactor operating temperatures, absence of closed loops, only three loop operation, postulating that natural convection is a viable method of decay heat removal, that the facility will not be operated when tornado (watch) conditions exist or erratic behavior of the check valve in the tornado hardened loop is detected, and that the constructed facility is as described in the FSAR and reference documents, that NRC did not have any inspection enforcement responsibility; further, that DOE is fully responsible for plant security and finally, that the Project furnish NRC with timely and definitive plans to respond to Section 19.3, items A, B and C of this supplement.

APPENDIX A

REFERENCES

Note: The documents, listed below, used to prepare this Safety Evaluation Report Supplement are on file for public inspection at the NRC Public Document Room, 1717 H Street, N.W., Washington, D. C. or are included as appendices to this supplement.

1. Transcript - Advisory Committee on Reactor Safeguards Subcommittee meeting on FFTF, Washington, D.C., July 12, 1978.
2. Transcript - Advisory Committee on Reactor Safeguards Subcommittee meeting on FFTF, Washington, D.C., August 10, 1978.
3. Transcript - Advisory Committee on Reactor Safeguards - 221 General Meeting, Washington, D.C., September 7, 1978.
4. Transcript - Advisory Committee on Reactor Safeguards - 223 General Meeting, Washington, D.C., November 2, 1978.
5. Advisory Committee on Reactor Safeguards Report on Fast Flux Test Facility, July 15, 1975.
6. Advisory Committee on Reactor Safeguards Report on Fast Flux Test Facility, November 8, 1978.
7. Plans for addressing NRC and ACRS advice regarding startup and operation of FFTF, letter to W. Gammill, AD for Standardization and Advanced Reactors, Division of Project Management, NRC from A. J. Rizzo, Deputy Director for Engineering FFTF PO, DOE, January 26, 1979. (Part of Appendix D)
8. Ultrasonic Inspection of FFTF Piping, letter to W. Gammill AD for Standardization and Advanced Reactors, Division of Project Management from A. Rizzo, Deputy Director for Engineering, FFTF PO, DOE, November 24, 1978.
9. Supplemental information provided ACRS, letter to R. F. Fraley, Executive Director, ACRS from T. P. Speis, Chief, Advanced Reactors Branch, DPM, NRC, October 26, 1978.

Attachment I - Radiological Consequences Following an FFTF Meltdown with vented Containment
Attachment II - Loss-of-Heat-Sink Scenarios
Attachment III - Control of Tritium
Attachment IV - Smoke Detectors

Attachment V - Fire Protection Program
Attachment VI - Piping Integrity and Leak Detection
Attachment VII - Neutron Fluences on Structures and Components.

10. Sodium Aerosol Release from Insulated FFTF Piping, letter to W. Gamill, AD for Standardization and Advanced Reactors, DPM, NRC from A. J. Rizzo, Deputy Director for Engineering, FFTF PO, DOE, November 16, 1978.
11. Analysis of FFTF Response to Postulated Earthquake with Primary Piping Leak and Cooling by Natural Circulation, letter to W. Gamill, AD for Standardization and Advanced Reactors, DPM, NRC from A. J. Rizzo, Deputy Director for Engineering, FFTF PO, DOE, November 22, 1978.
12. FFTF Cell Liner Test Plan, letter to W. Gamill, AD for Standardization and Advanced Reactors, DPM, NRC from A. J. Rizzo, Deputy Director for Engineering, FFTF PO, DOE, January 31, 1979.
13. HEDL-TC-698, "Flow and Stress Analysis of the FFTF Reactor Cavity Liner Vent System Under Ex-vessel PAHR Conditions," September 1976.
14. HEDL-TI-75279, "Large-scale FFTF Liner Sodium Spill Test Preliminary Report," October 1977.
15. HEDL-TI-75280, "High Temperature Liner Feature Test FT-3."
16. HEDL-TC-1059, Evaluation of the CLIRA Outlet Temperature PPS Function for Design Loss-of-Flow Events.
17. Sandia Comments on L. D. Muhlstein's Summary of November 15-16, 1978 Meeting on Sodium/Concrete (Memo to H. Holz from T. H. Walker, April 6, 1979) (Appendix F)
18. Sodium Concrete Interaction Test No. 13 (Memo to M. Silberberg from T. J. Walker, dated December 29, 1978) (Appendix G)
19. Sodium/Magnetite Concrete Reaction Test No. 14 (Memo to M. Silberberg from T. J. Walker, dated April 11, 1979) (Appendix H)
20. NRC Staff Recommendations Regarding Conventional FFTF Fire Protection Program (Non-Sodium) (Appendix I)

APPENDIX B
UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

November 8, 1978

Honorable Joseph M. Hendrie
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: REPORT ON THE FAST FLUX TEST FACILITY

Dear Dr. Hendrie:

During its 223rd meeting, November 2-4, 1978, the Advisory Committee on Reactor Safeguards completed its review of the proposed operation of the Department of Energy's (DOE) Fast Flux Test Facility (FFTF). The ACRS reported previously on the construction phase of this project on July 13, 1971, January 13, 1972, May 18, 1973, and July 15, 1975. The project was also considered during the 221st meeting, September 7-9, 1978 and at Subcommittee meetings held in Washington, DC on July 12, 1978 and August 10, 1978. During its review, the Committee had the benefit of discussions with representatives of the Reactor Research Technology Division of the DOE (Project), their contractors and consultants, and the NRC Staff. The Committee also had the benefit of the documents listed below.

The FFTF is a 400 Mwt sodium cooled fast reactor located at DOE's Hanford Reservation in Benton County, Washington. The site is about 4 1/2 miles from the Columbia River, the nearest boundary of the reservation, and about 10 miles north of Richland, Washington.

In view of the fact that this reactor is unique, appropriate detailed procedures and standards of the kind used in the review of light water reactors were not available for the review of FFTF. Because the FFTF is a DOE facility the scope of the NRC review was defined by the DOE request that NRC provide advice regarding the adequacy of the FFTF design and technical specifications to ensure safe operation. The NRC review did not include construction audits, assessments of the "as built" configuration, or evaluation of acceptance test results that would verify that the plant was constructed in accordance with the design criteria and documentation. Provisions for safeguards and security were also excluded from review.

The NRC Staff agrees with the Project that failure of the reactor inlet piping need not be considered as a cause of a core disruptive accident (CDA) provided that certain conditions are met. These include preservice and inservice inspection of piping, leak detection instrumentation, and an appropriate materials surveillance program. The Project has committed to developing an ultrasonic testing (UT) device for high temperature use, to inservice inspection of selected, high stress welds on the secondary sodium loops when the UT device is available, and to installation of sodium aerosol leak detection systems in the heat transport cells. A materials surveillance program will be conducted using test subassemblies, as well as in-reactor and hot leg components that are removed. In addition, the NRC recommends that a cold preservice inspection be conducted on the hot crossover piping welds and that inservice inspections be implemented as soon as practicable. The Committee supports these recommendations.

The Project has performed studies of various postulated core disruptive accidents (CDA). The NRC Staff has concluded that the calculated prompt energetics from CDAs are within the capability of the containment system. The ACRS concurs with the Staff conclusion.

The FFTF does not have a Class 1E power supply to provide decay heat removal. Instead, the Project will depend upon natural convection cooling in the event of loss of offsite power and failure of the onsite diesel generators. The Project's calculations indicate that natural circulation will provide decay heat removal. It is proposed that the natural circulation decay heat removal be measured during the startup testing. The ACRS concurs that the adequacy of the decay heat removal by natural circulation should be experimentally verified.

The NRC Staff and the Project have not yet agreed on the adequacy of containment for dealing with the consequences of some low probability accidents which lead to the potential for generation and release to containment of significant quantities of sodium aerosols, hydrogen and other volatile gases. In its report of July 15, 1975, the ACRS recommended that consideration be given to the possible usefulness of sand-and-gravel filters for the removal of airborne particulates. During the current review, the NRC Staff has recommended that measures be taken by the Project to permit the measurement and control of the hydrogen concentration in the containment, to further reduce the chance of a damaging explosion. The NRC Staff has also recommended that means be included for controlled venting. The ACRS supports these Staff positions and recommends development of additional mitigation measures, such

November 8, 1978

as a sand and gravel filter, for possible addition to the controlled venting system to provide still further assurance of limited radioactivity releases in the event that one of these low probability accidents should occur.

As pointed out in a previous report, the ACRS has recognized that the FFTF is a special test facility located on a favorable site. Both positive and negative aspects of this situation have been considered throughout the review. The ACRS believes that if due regard is given to the matters mentioned above, and in previous reports, it is acceptable for startup and operation of the FFTF to proceed.

Sincerely,

Stephen Lawroski
Chairman

References:

1. U.S. Energy Research and Development Administration (now DOE), "Final Safety Analysis Report, Fast Flux Test Facility," Vols. 1-10, dated December 1975 with Amendments 1-27 and Supplements 1-28.
2. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report, USDOE, Fast Flux Test Facility," USNRC Report, NUREG-0358, dated August 1, 1978.
3. Letter from R. L. Ferguson, ERDA, to R. P. Denise, NRC, Subject: NRC Review and Advice on the FFTF, dated November 13, 1975.
4. Letter from R. L. Ferguson, ERDA, to R. P. Denise, NRC, Subject: Scope of NRC Review of FFTF FSAR, dated August 20, 1976.
5. Letter from R. L. Ferguson, ERDA, to R. P. Denise, NRC, Subject: FFTF Safeguards and Security, dated July 14, 1977.

APPENDIX C

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

OCT 26 1978

Mr. R. F. Fraley, Executive Director
Advisory Committee on Reactor
Safeguards
United States Nuclear Regulatory
Commission
Washington, D. C. 20555

Dear Mr. Fraley:

At the 221st general meeting of the Advisory Committee on Reactor Safeguards held in Washington, D. C., on September 7, 1978, to consider the safety review of FFTF, a number of questions were raised for which the Committee asked the staff to address at the next general meeting on FFTF, now scheduled for November 2, 1978. Even though the staff will address these questions at the upcoming meeting, in order to facilitate the Committee's review of FFTF we are providing the Committee prior to the November 2 meeting with written material dealing with the particular subjects/concerns that were raised.

The supplemental information attached to this letter deals with the following subjects:

1. Attachment I - Radiological Consequences Following an FFTF Meltdown Accident with Vented Containment -

The reasons and the basis that the doses from a vented containment do not deviate substantially from 10 CFR 100 guidelines.
2. Attachment II - Loss-Of-Heat-Sink Scenarios -
The capability of primary system sodium to cool the FFTF core (either disrupted or intact) following a nuclear shutdown if a loss-of-heat-sink is postulated.
3. Attachment III - Control of Tritium
4. Attachment IV - Smoke Detectors
5. Attachment V - Fire Protection Program
6. Attachment VI - Piping Integrity and Leak Detection -
Additional information on the staff's position.

R. Fraley

- 2 -

7. Attachment VII - FFTF Neutron Fluence/Dose on Structures and/or Components

If we can be of further assistance we will be pleased to respond.

Sincerely,

Themis P. Speis, Chief
Advanced Reactors Branch
Division of Project Management

cc: A. Bates, ACRS Staff

ATTACHMENT I
RADIOLOGICAL CONSEQUENCES FOLLOWING AN FFTF VESSEL MELTTHROUGH ACCIDENT
WITH VENTED CONTAINMENT

The staff's analysis and evaluation of the radiological consequences of a core disruptive accident with vessel meltthrough in FFTF are documented in Section 15.3.7 of the FFTF SER(1) and have also been previously presented to the ACRS.

The FFTF project has also analyzed the radiological consequences from FFTF meltdown accidents and has recently documented two typical scenarios in HEDL-TC-1175(2). The ACRS has asked for more information concerning the variants of such accidents and the releases of radioactivity to the environment associated with them. In particular, the ACRS asked for substantiation of the fact that the doses from a vented containment could be limited to the order of 10 CFR 100 guidelines. This discussion has been prepared to draw attention to the key assumptions concerning aerosol behavior that enter into such an analysis, and to provide further substantiation for the staff recommendations for dealing with these situations. Although doses an order of magnitude greater than 10 CFR 100 guidelines can be calculated, the combination of circumstances and the conservatism in the calculations required to generate these doses will be discussed.

Radioactive releases associated with accidents involving large sodium fires are strongly influenced by behavior of the aerosol of sodium oxide smoke. Unless the radioactive species is inherently gaseous at the temperature of the atmosphere above the fire, the radioactive material will condense to fine particles which coagglomerate fairly uniformly with the sodium products. (Coagglomeration has generally been assumed in the past but is now being verified experimentally by work at ORNL.) It has been possible to set up codes which have been quite successful in calculating the agglomeration and fallout of the (predominantly) sodium oxide aerosols.

A code called HAA-3 was developed at Atomics International which described the collisions between particles based on simple geometric considerations, gravitational buoyancy, and Brownian kinetics, and which gave a reasonable description of aerosol behavior when certain empirical constants to describe collision efficiency and Stokes settling corrections were used. A major simplifying assumption in the code is that the size distribution of the particles follows a log-normal pattern. Distributions of this type have been quite commonly observed.

This code has been modified by the staff at Battelle Columbus Laboratories under contract to NRC's Office of Research, to permit some of these empirical factors to be varied within the computation in accordance with recent experimental observations on individual particles at various stages of growth. The modified code, HAARM-3, also incorporates the capability to calculate the composition of aerosols from certain mixed sources, the temperature effects of surfaces, and other conveniences.

Experiments for the verification of these codes have been carried out in vessels up to a height of 20-m, with aerosol concentrations up to 35 g/cm^3 of sodium oxide. Good agreement with experiments of this magnitude has provided the basis for extrapolation to reactor sizes, about 40 m high and about 100 g/cm^3 for a large sodium fire. Figure 1 shows a comparison of measured with calculated values.⁽³⁾

The vessel meltthrough accident does not include any mechanism for vaporizing a large fraction of the core, as might be the case with an initially more energetic core disruptive accident (CDA). Nevertheless, the sodium pool, boiling and reacting turbulently with concrete, is expected to disperse into the atmosphere some of the plutonium that may be present in fine particulate form. About 0.1% of the core plutonium may conservatively be estimated to be levitated into the containment building atmosphere. (See page 15-62 of Reference 1)

In addition to the plutonium, the containment building atmosphere may contain all the radioactive gases, the halogens and the more volatile solid fission products as aerosols, with a small fraction of non-volatiles. Since the consequences depend on the reduction of this material through aerosol processes, a typical scenario incorporating the aerosol processes described.

The gross behavior of the sodium oxide aerosol is illustrated in the attached Figure 2. With reference to this figure, the boiloff of sodium in the reactor cavity begins three hours after the initiation of the accident and continues to provide an aerosol source at the rate of about 130 gms per second until about 40 hours. This rate is consistent with the staff's estimated penetration rate of sodium into concrete of one inch per hour. During this time the concentration of aerosol in suspension in the RCB levels out to a steady state value amounting to about one half of one percent of the total 280,000 kg pool because the rate of fallout of suspended material becomes equal to the rate of introduction of new suspended material. If the introduction of new suspended material were stopped during this period, the suspended concentration would rapidly decay as shown by the dashed line at twenty hours.

If the floor of the cavity is penetrated, the exposure of additional concrete surface in the subcavity room will increase the rate of sodium reaction by about a factor of ten, to 1300 gm per second. The concentration of suspended aerosol soon thereafter levels out to a new steady state value, which however is only three times the former value, i.e., 1.5% of the total pool inventory. The 0.1% leak rate from containment during this period prior to containment failure makes little difference in the suspended concentration.

If, in order to relieve pressure, it became necessary to increase the leak rate sufficiently to vent the gases generated in the reactions, (an average of about 4000 cfm of gases are produced), the suspended concentration would be reduced slightly as shown at sixty hours. The reason the reduction is no greater is that the source and fallout terms still predominate in the aerosol equation over this seemingly large leak rate.

At the 1300 gm per second rate, total pool boiloff would occur in sixty hours. If 4000 cfm venting were continued throughout such a sixty hour period, only 12% of the total sodium would be released to the environment, because of the predominance of the fallout effect.

During some scenarios, a hydrogen deflagration is calculated to occur. This sudden event may require the prompt venting of about half the containment building contents. It can be seen from Figure 2, however, that because no more than 1.5% of the sodium pool can be suspended in the RCB, even such a large prompt puff release would not vent more than about 0.7% of the sodium.

The figures cited above for the venting of sodium would apply equally well to any radioactive isotopes released from the pool at the same rate as the sodium and coagglomerated with the sodium aerosol. The small amount of plutonium released from the pool is thought to be released uniformly with the sodium and would coagglomerate with it. The aerosol behavior is therefore an important factor in the reduction of the amount of plutonium released.

There are some important isotopes whose volatility differs from that of sodium to the extent that they are released from the boiling pool almost entirely at either the beginning or the end of the boiloff phase. Iodine, in the form of sodium iodide, is such a substance. It has been estimated (Fig. 3)⁽⁴⁾ that about 40% of the iodine in the pool would be released to the atmosphere during the final 10% of the sodium boiloff, between 84 and 90 hours on Figure 2. Thus, during this period, the fraction of the iodine inventory suspended with the aerosol would be four times greater than the fraction of the sodium inventory. This makes little difference in the case of long-term venting at 4000 cfm, since periods of highly concentrated release would be balanced by the periods of less concentrated release. In the case of a puff release accompanying hydrogen deflagration, the amount of iodine released would be quite dependent on the timing, being most serious if the puff were synchronized with the period of greatest sodium iodide boil-off. Since iodine is in many cases the critical isotope, we present here a hand calculation of the ^{131}I dose at the low population zone boundary, under the assumption that a puff release coincided with the period of maximum sodium iodide boiloff.

The dose, D, can be represented as

$$D = FxQx(X/Q) \times (BR)$$

where F is the conversion factor from curies inhaled to Rem, thyroid. For ^{131}I this is 1.49×10^6 . (Radioactive decay prior to inhalation has been neglected.)

The source, Q, is the product of the total ^{131}I shutdown inventory, the fraction remaining in the pool after the initial burst, the fraction of the sodium pool in suspension, the ratio of iodine to sodium being released from the pool, and the fraction of the containment volume vented in the puff. Thus,

$$Q = 1.07 \times 10^7 \times .75 \times .015 \times 4 \times .5 = 2.4 \times 10^5 \text{Ci}$$

The factor (X/Q) represents the ratio of downwind concentration to initial source, and is dependent on meteorological conditions. It has become customary in licensing calculations to use a value of (X/Q) that is not exceeded more than 5% of the time for the initial eight hours after the start of the accident, on the premise that the accident timing is random with respect to meteorology and could occur, without controls, at a time when meteorology was reasonably poor. For FFTF, this initial (X/Q) value is $2.7 \times 10^{-5} \text{ sec/m}^3$. Since these adverse meteorological conditions are unlikely to continue indefinitely, successive stages of the accident use (X/Q) values closer to the average conditions, and by the time of the puff releases, about thirty hours after the start of the accident (X/Q) may be reduced by a factor of three, to 8.9×10^{-6} . This procedure was used by the applicant in his treatment of puff releases in HEDL-TC-1175.

The staff reasons however that since the puff release occurs at a random time, uncontrolled by the operator, it should be treated as an initial release, using the higher (X/Q) value. All the above (X/Q) values are conservative because of neglect of any buoyancy effects associated with the elevated temperature of the release.

The breathing rate, BR, is also by custom taken to be time dependent, with the highest value, $3.47 \times 10^{-4} \text{ m}^3/\text{sec}$, corresponding to the active part of the day, used during the first eight hours of the accident. At approximately thirty hours after the start of the accident, normal procedure would be to use the daily average breathing rate, two-thirds of the above figure. The applicant has used the lower breathing rate, but the staff prefers the initial rate, consistent with the arguments for (X/Q).

The ^{131}I dose then becomes:

$$D = 1.49 \times 10^6 \times 2.4 \times 10^5 \times 2.7 \times 10^{-5} \times 3.47 \times 10^{-4} = 3300 \text{ Rem}$$

Thus, a dose an order of magnitude greater than the 10 CFR 100 guidelines can be calculated, based on the juxtaposition of a puff release at the time when radioactive concentration is greatest, using conservative (X/Q) and breathing rates. (We do not mean to imply that the 10 CFR 100 recommendations are limits that must be met in this case, but merely that they are guidelines to orient the significance of these results.) Conservative aspects of the calculation presented here are the neglect of the buoyant effect of the elevated temperature of the release and the neglect of fallout outside the containment. If these two factors were taken into account the ^{131}I dose would then become 660 Rem.

The generation of hydrogen associated with this accident requires some additional consideration. Hydrogen is generated when heat drives water out of concrete and the water reacts with a pool of sodium. In considering the possible hazards of hydrogen, the staff starts from the conventional position as expressed, for example, in the Reactor Handbook,⁽⁵⁾ that hydrogen is a common industrial explosion hazard. If an explosion serious enough to cause massive failure of the containment structure occurred, the radioactive material that had fallen out of the atmosphere of the containment building would be subject to redispersal and ejection to the environment. The consequences of the accident could certainly be multiplied several fold, perhaps an order of magnitude. The staff therefore wishes to avoid the possibility of such an explosion.

The applicant has presented data to indicate that under certain conditions the hydrogen would be introduced into the containment building atmosphere in such a way that it will ignite and burn without pressure or shockwave buildup (deflagration) especially in the presence of unreacted sodium. The staff and its consultants have expressed concern that perhaps not all the realistic conditions for hydrogen generation have been included in the applicant's analysis. Recent experiments at Sandia have attempted to reproduce realistically the conditions of sodium reactions with concrete, and these have occasionally been accompanied by hydrogen explosions. Thus, the staff is unwilling to accept the view that hydrogen can be depended upon to burn benignly under the natural processes associated with these accidents. The staff does recognize however that a range of compositions exists in which hydrogen will burn, but which is too dilute for an explosion to occur. This has led the staff to suggest that a controlled method of ignition be provided. This would render moot the question of whether the natural tendency of hydrogen to ignite was reliable

enough under the circumstances. To accomplish such a result effectively, it would probably be necessary to monitor the hydrogen and radioactive content of the containment building atmosphere, and to schedule the deflagration and release when these conditions were most favorable, considering also the external meteorology and population distribution.

There are several other features that could be added to the plant to reduce the consequences of the accident. Although they have not been investigated in detail by the staff, it is estimated that their cost would not be prohibitive, although probably larger than that of the procedures recommended. They include filtration through sand beds, for a possible reduction in particulate release between ten and one hundred. This would require some development because of the considerable amounts of sodium oxide present. Another possibility is an elevated stack for release of products. Rough calculations indicate that a 100 m stack would bring about a reduction of doses at the low population zone by a factor of twenty, for both gases and particulates. The staff has not made a detailed review of these possibilities, however.

In summary, the NRC recommendations presented in the SER are based on the importance of reducing the public hazard from such an accident to as low a figure as is reasonably achievable. This reduction is facilitated by monitoring the hazardous conditions of the containment building and by providing a minimum amount of control over the timing of deflagrations and releases.

FIGURE 1. PREDICTED AND MEASURED AEROSOL CONCENTRATION FOR CSTF RUN AB-1

Source		130 gm/sec	1300	1300	0
Steady State Values	Leakage	.02	.05	160 (0.2 % per hr.)	
	Fallout/ Plateout	130	1300	1140	
	Amount Released by a puff	0.2 %	0.75 %	0.7 %

REFERENCES

1. Safety Evaluation Report for FFTF, NUREG-0358 (August 1, 1978).
2. Jensen, R. L., et al, "Further Studies of Containment Margins in FFTF for postulated Failure of In-Vessel Post Accident Heat Removal," HEDL-TC-1175 (July 1978).
3. Gieseke, J. A., et al, "Aerosol Measurements and Modeling for Fast Reactor Safety: Annual Report for FY 1977," BMI-NUREG-1989 (December 23, 1977).
4. Castleman, A. W., Jr., "Fission-Product Behavior in Sodium," Nuclear Safety, 11, 5, p. 379 (September-October 1970).
5. Reactor Handbook, 2nd Edition, Vol. I, p. 1016, 1017.

ATTACHMENT II

LOSS-OF-HEAT-SINK SCENARIOS

In response to several questions raised at the last ACRS meeting on FFTF and, in particular, in response to Dr. Stratton's letter to the Committee of July 19, 1978, the staff has prepared the following response. The basic question was: What is the capability of primary system sodium to cool the FFTF core (either disrupted or intact) following a nuclear shutdown if a loss-of-heat-sink is postulated?

Five scenarios are presented here that we believe scope the possible accident progressions, two with an initially intact core and three with the initial assumption of core disruption. Table I brings together the key aspects of these five scenarios.

Briefly, the first two scenarios (I-A and I-B) assume an intact core initially at full power with a protected shutdown (rod scram) but with a loss-of-heat-sink. Starting with these initial conditions the scenarios were formulated to provide upper and lower bound estimates of how long after shutdown it takes to void the core and thus start core meltdown.

SCENARIO I-A

The first scenario assumes a breach in the Primary Heat Transport System (PHTS); this breach can be in the piping, branch piping, the vessel itself, or the overpressure protection system designed to relieve at 15 psig. After initiation of sodium boiling in the reactor vessel, the sodium vapor is vented to the containment building. The key assumption for scenario I-A is that the natural circulation flow, which is up through the core and down through the reflectors, is completely effective in cooling the core so that cladding dryout does not start until the core is uncovered (sodium level reaches the active core). For this scenario, the time to void the core is in the range of 175-220 hours (7-9 days).

SCENARIO I-B

The second scenario assumes: no breach in the PHTS, an adiabatic PHTS, and that the pony-motors maintain an isothermal system until $T = T_{\text{sat}}$. The added assumption here is that the natural circulation flow, assumed in Scenario I-A, is not present and thus cladding dryout commences well before Scenario I-A. For scenario I-B, the time to void the core is in the range of 11-15 hours.

Before considering the three disrupted core cases, we believe it is important to make a few comments concerning the above analyses and assessments.

- In 1976, the staff and, in particular, its consultants at BNL started analyzing the loss-of-heat-sink with scram problem as part of the CRBR review. The emphasis of this study was addressing the question of recriticality following sodium voiding (and subsequent cladding dryout), not the question of core/reflector natural circulation coolability. Thus, the three reports on this subject BNL-NUREG-23137, 23432, and 50910, concentrate on the CRBR core and emphasize the meltdown

progression and recriticality potential. Nevertheless, these studies are still valid in considering the meltdown progression and recriticality potential for FFTF for scenarios 1-A and 1-B once the core has been voided of sodium. The report "Loss-of-Heat-Sink in the Secondary Heat Transport System of the FFTF," BNL-NUREG-23269, dated September 1977, forms part of the basis for the results in Scenario 1-B. The other results quoted are from unpublished analyses performed in response to specific ACRS questions.

- For scenario 1-A, both the vessel head and walls were considered as condensation surfaces, allowing the sodium to drain back into the vessel pool. This cooling mechanism is very important and extends the core uncovering time by a factor of 8.
- The long core uncovering time (7-9 days) may be irrelevant in that it will take only about 3 days to pressurize the FFTF containment to 10 psig (due to sodium vapor entering the containment building and reacting with oxygen), thus, forcing containment venting to the outside atmosphere. Although ^{24}Na and ^{22}Na are only small contributors to site boundary dose, there would be a high level of activity in the containment building, probably precluding human intervention (e.g., to pump additional sodium into the vessel).
- It has been suggested that the containment building would provide sufficient condensation surfaces to eliminate containment pressurization. The key point here is that although there are significant heat losses to the containment, there are major energy additions to the FFTF containment from the significant exothermic reactions of Na-O_2 and $\text{Na}_2\text{O-H}_2\text{O}$. These exothermic chemical reactions provide the source of containment pressurization not the mass addition from the sodium vapor. It is particularly important to note that only 5% of the heat energy input to and generated in the containment is needed to pressurize the containment to 10 psig.
- The question was raised whether one could have core melt with the PHTS intact. It is the staff's position that whole core melt will result from a Loss-of-Flow without scram with the PHTS assumed intact. This class of accidents is discussed below. In addition, for the scenario 1-A scenario discussed above (with scram) a failure of the PHTS is not a necessary condition for fuel melting (that is assuming that the rupture disk functioning is not classified as a PHTS failure).

We review now the three disrupted core scenarios in order to clarify some points regarding the initial conditions needed for a core-melt accident, and in particular, we will address the question of whether there must be a breach of the PHTS in order to initiate a meltdown. A core meltdown event does not have to be initiated by failure of any of the three primary piping loops. The two postulated initiating events for a Core Disruptive Accident (CDA) followed by core meltdown which the staff has concentrated on are:

- (1) loss of flow (i.e., loss of pumping power) accompanied with failure to

scram, and (2) transient overpower with failure to scram. For both of these scenarios, the primary piping loops have been assessed to remain intact from the dynamic loads resulting from CDA energetics. So the question then becomes, following a meltdown event with the primary loops intact, will the core debris be contained in-vessel in a coolable configuration by convection cooling through the loops or will the fuel debris melt-through into the lower reactor cavity region. For the latter scenario, we are interested in the extent of FFTF containment margins to cope with this event. Here we briefly summarize the key points of three enveloping meltdown scenarios in Table I; namely, "In-Vessel Post Accident Heat Removal (PAHR) with Heat Sink" (Scenario II-A), "In-Vessel PAHR with Loss of Heat Sink" (Scenario II-B), and "Ex-Vessel PAHR" (Scenario II-C). Note that the scenarios are progressively more severe. All these three scenarios are discussed in the FFTF SER* as indicated in the table references. A brief summary of the three scenarios follow:

SCENARIO II-A - IN-VESSEL PAHR WITH HEAT SINK

Following a core meltdown caused by a Loss-of-Flow (LOF) without scram, the core fragments with 60-90% ejected upward onto the upper thermal baffles and 10-40% melts downward into the lower core support dome. Sodium convection through the DHXs provides an adequate heat sink and maintains the debris in a coolable configuration. For this scenario the Primary Heat Transport System (PHTS) remains intact, and the accident consequences (in terms of containment pressurization and hydrogen buildup) are negligible.

SCENARIO II-B - IN-VESSEL PAHR WITH LOSS-OF-HEAT-SINK

This scenario starts out as in scenario II-A above, but with the added assumption of a loss-of-heat-sink (LOHS). This LOHS is postulated to be caused by CDA - related structural damage of the primary system. Venting of sodium vapor from the reactor vessel to the containment building will take place once the pressure differential builds up to about 15 psig. It takes about 19 days for all the reactor vessel sodium to boil off. Well before this, however, the containment must be vented (2-3 days from accident initiation) to preclude overpressurization caused primarily by sodium vapor entering the containment building and reacting with oxygen.

SCENARIO II-C - EX-VESSEL PAHR

This scenario also starts out the same. However, unlike II-A and II-B, the whole core inventory is assumed to melt downward. Melt-through of the reactor and guard vessels occurs about 3 hours from the initiation of the accident. Before vessel melt-through, PHTS failure is not assumed. The CDA energetics are considered negligible. As indicated in Table I the containment loading is the most severe of the 3 scenarios.

*NUREG-0358, dated 8/78

We now proceed with a more detailed discussion of FFTF PAHR and associated containment response in order to clarify certain aspects of our assessment.

SCENARIOS II-A, -B, -C : FURTHER COMMENTS

With respect to an assessment the FFTF in-vessel core retention capability was provided in the staff's Supplement 2 to the Safety Evaluation Report issued on March 7, 1975. The staff concluded that the steel dome of the lower core support structure, Figure 1, is capable of containing between 20 to 40% of the core material as a molten pool for a substantial period of time, depending on the decay heat level and temperature of the inlet sodium. Tables 1 and 2 present results for the estimated lower dome creep lifetime for various decay heat levels and sodium inlet temperatures, respectively. It is evident from these results that the effects of decay heat level and sodium inlet temperature on the dome lifetime can be very significant. For example, increasing the temperature of the inlet sodium from 400°F to 800°F can reduce the dome lifetime by many orders of magnitude, depending upon the particular conditions of the mass and decay heat of the molten material in the dome. It is, therefore, essential that natural convection through the primary and secondary loops, as well as proper operation of the air dump heat exchangers, be assured such that the 400°F inlet sodium temperature is controlled and not permitted to rise.

This assessment of the FFTF in-vessel core retention capability, however, has been predicated upon a reasonable but somewhat limited understanding of certain critical physical phenomena, which affect the in-vessel post-accident heat removal capability. Three primary areas of uncertainty are discussed as follows:

1. Given that the staff is now predicting a relatively mild energetic CDA evolution, the disposition of core debris is uncertain in that more fuel material might be relocated, in the downward direction.
2. The coolability of fuel in the lower core support dome is very sensitive to the local heat flux distribution from the core debris to the steel dome surface and the downfacing heat transfer from the steel dome surface to the underlying sodium. Essentially no directly applicable experimental heat transfer data exists for this configuration, and the staff fuel coolability predictions, as well as those of the FFTF Project, are considered to be only rough estimates.
3. The structural capability of the lower core support steel dome is very difficult to assess because of the complex accident conditions which exist. Predicting the dome structural capability involves estimating the creep lifetime under non-uniform conditions of heat transfer, geometric, and material property distributions.

Based on the above uncertainties, the staff concluded in Supplement 2 that "the residual risk which arises from uncertainties in in-vessel post-accident heat removal makes it prudent to provide for the consequences of this low likelihood event in the best possible manner," and specifically to "consider means to alleviate high doses associated with vessel melt-through." Since

the staff's evaluation of the adequacy of in-vessel post-accident heat removal in Supplement 2, the FFTF Project has not provided any substantive new data or information which would make the staff change its previous position as reflected above.

With respect to the FFTF ex-vessel core melt scenarios, questions have been raised by the Committee as to why the staff is not given full credit for the melted core material to fragment and attain a coolable configuration. We have performed analyses, which are reported in SER Section 15.3.6., assuming the core debris to be either in a coolable particulate debris bed form (as assumed by the FFTF Project) or in a molten pool form. We felt it necessary to analyze the molten pool configuration because of the assumption made by the FFTF project, that following melt-through of the reactor and guard vessels the molten fuel/steel mixture will immediately fragment, self-level, and attain a coolable debris-bed configuration, is not fully substantiated by experimental data. The project's assumptions on coolable debris bed formation are based on very limited experimental data obtained with small quantities of molten materials ($<2\text{kg}$). There exists experimental information which indicates that the tendency to fragment decreases as the quantity of molten material increases. Therefore, in the absence of experimental data of a sufficiently large scale to demonstrate the validity of the project's assumptions, we considered it prudent to also analyze the containment response and penetration into the reactor cavity floor assuming that the core debris remains in a molten pool.

TABLE-1

INITIATOR	KEY ASSUMPTIONS/PARAMETERS	CORE HISTORY	CONTAINMENT HISTORY	ISSUES
<u>I-A LOSS-OF-HEAT SINK WITH SCRAM: EFFECTIVE CORE/REFLECTOR CIRCULATION</u>				
Postulated failure (breach) of PHTS system such that vessel is isolated from the heat transport system (with scram)	<ol style="list-style-type: none"> 1. A CACECO calculation, in which the decay heat is uniformly allocated to Na and in-vessel structure, was performed. 2. Na Vapor leakage is directly to RCB. 3. Condensation of Na on Reactor Vessel walls and Heat (and return to Na Pool.) 	<p>Time for Na level to reach top of core is approx. 175 (1) - 220(2) hrs.</p> <p>Time to uncover active core height is an additional 36 minutes.(3)</p> <p>After clad dryout, fuel will melt in an additional eighty minutes.(4)</p>	<p>10psi (containment overpressure) is expected to occur in approx. 80 hrs. Note that containment limit is reached 100 hrs before core is uncovered.</p>	<ol style="list-style-type: none"> 1. Clad may dryout before Na level drops to top of core (due to inadequate circulation of Na and therefore ineffective heat removal from some pins)
<u>I-B LOSS-OF-HEAT SINK WITH SCRAM: NO CIRCULATION</u>				
<p>Simultaneous loss of secondary heat transport trains - Furthermore assume that pony motors stay on until $T_{up} \sim T_{LP} \sim T_{sat}$.</p> <p>$up = \text{Upper Plenum}$ $LP = \text{Lower Plenum}$</p>	<p>If pony motors fail when temp. reaches T_{sat}, then core flow stops and it is assumed that the core cannot be adequately cooled by core/reflector natural circulation. If pony motors remain on after T_{sat} is reached, then if dryout does not occur before the core is uncovered, the scenario follows case I-A.</p>	<p>Voided core space at about 11-15hrs(5)</p> <p>meltdown proceeds in manner similar to that for CRBR as described in Ref. 4.</p>	<p>No breach of vessel until possibly after melt-through (after recriticality event). Any (small) breaches of vessel due to temp. or pressure-induced failures would not significantly impact on time scale of this scenario.</p>	<ol style="list-style-type: none"> 1. If meltdown same as case II, below.

INITIATOR	KEY ASSUMPTIONS/PARAMETERS	CORE HISTORY	CONTAINMENT HISTORY	ISSUES
<u>II-A IN-VESSEL PAHR WITH HEAT SINK 6, 7</u>				
1. Loss of Flow with failure to Scram	1. 60 to 90% of Core is ejected upward and coolable on upper thermal baffles. 2. 10 to 40% of Core Melts Downward and is Coolable in lower core support dome. 3. Inlet Sodium Temperature is controlled and maintained at 400°F and decay heat level is in the range of 1 to 2%.	1. Core debris is maintained in a coolable configuration indefinitely within reactor vessel for sodium inlet temp. of 400°F and decay heat level of 1%. (See Tables 1 and 2 in attachment 3 for results of core support dome creep lifetime for other sodium temperatures and decay heat levels).	Negligible consequences	1. Non-energetic accidents evolution and effects on downward disposition of core debris. 2. Coolability of core debris in lower dome; that is, a. Downfacing heat transfer b. Heat flux dist. 3. Lower core support dome structural capability.
<u>II-B IN-VESSEL PAHR WITH LOSS-OF-HEAT SINK 8, 9</u>				
1. Same as Scenario II-A., plus	1. Same as Items 1 & 2 under scenario II-A above,	1. Start boiling sodium in reactor vessel at 25 hrs.	1. Vent containment in a time interval of 50 to 80 hrs to avoid overpressurization	1. Same as II A.; However, even more uncertain because of loss-of-heat dissipation through primary system.
2. Failure of Primary System (NOTE: As in II-A., Initiator No. 1 results in a CDA followed by core meltdown)	2. CDA related structural damage of primary system ¹⁰ results in loss of heat sink. 3. Amount of sodium ejected from reactor vessel to reactor cavity. This parameter was varied in analysis.	2. Evaporate all of reactor vessel sodium at 450 hrs. (NOTE: Start with the core in a melted configuration within the reactor vessel)		2. CDA energetics sufficient to cause failure of primary system, ¹⁰

INITIATOR	KEY ASSUMPTIONS/PARAMETERS	CORE HISTORY	CONTAINMENT HISTORY	ISSUES
II-C EX-VESSEL PAHR ^{8, 9} (FAILURE OF IN-VESSEL CORE RETENTION)				
1. Same as Scenario II-A., plus	1. Core debris configuration	1. Start boiling sodium in reactor cavity at 50hrs. ¹²	1. Vent containment to preclude overpressurization in time interval of 30 to 60 hrs.	1. Core debris configuration; that is, coolable debris bed vs. molten pool.
2. Core debris melts through reactor and guard vessels and comes to rest on reactor cavity floor. ¹¹	2. Failure of reactor cavity liner. 3. Sodium-concrete interactions 4. Concrete water release 5. Effectiveness of liner venting system 6. Disposition of hydrogen	2. Evaporate all of primary sodium at 90 hrs. ¹²	2. Decrease containment hydrogen concentration to preclude hydrogen explosion in time intervals of: a. 15 to 45 hrs for a coolable core debris bed configuration, and b. 10 to 20 hrs for a molten pool core debris configuration.	2. Sodium-Concrete penetration rate and extent. 3. Disposition of hydrogen.

FOOTNOTES/REFERENCES FOR TABLE 1

1. BNL Analysis (unpublished) as reported at 8/10/78 ACRS Subcommittee Meeting, P. 67.
2. HEDL Analysis (as reported at ACRS meeting, 9/7/78, page 353).
3. BNL Analysis (unpublished) as reported at 8/10/78 ACRS Subcommittee Meeting, P. 68.
4. Value extrapolated to FFTF from BNL Calc. in BNL/CR-0427 (BNL-NUREG-50910) (in press).
5. BNL Analysis for FFTF in BNL-NUREG-23269.
6. Supplement 2 to the staff's Safety Evaluation Report for FFTF, issued March 7, 1975.
7. Safety Evaluation Report for FFTF, NUREG-0358, dated August 1978 (See Section 15.3.5)
8. See Section 15.3.6 of Reference 7.
9. "Containment Response to Postulated Core Meltdown Accidents in the FFTF," BNL-NUREG-24141-R, dated August 1978.
10. Because the staff is predicting relatively low CDA energetics in FFTF, we consider failure of the primary system from CDA related structural damage to be extremely unlikely.
11. Core debris melts through reactor and guard vessels in 3 hours; failure of PHTS is not assumed.
12. Times correspond to Case 12, Appendix F, Safety Evaluation Report for FFTF, NUREG-0358, August 1978.

TABLE 1
CALCULATED DOME LIFETIME

ANL AVERAGE CONDITIONS				NRC LOCAL CONDITIONS			
ANL Case	Steel Surface Temp. F*	Dome Thickness (in)	Calculated Lifetime † (hours)	Decay Heat	Steel Surface Temp. F*	Dome Thickness (in)	Calculated Lifetime † (hours)
a ₂₀	623	3.0	10 ¹⁵	0.78%	721	2.6	10 ¹³
d ₂₀	821	1.5	10 ⁷	2.0%	1006	0.8	10 ³
a ₄₀	765	2.1	10 ¹¹	1.05%	925	1.2	10 ⁵
d ₄₀	1067	1.0	6000	2.0%	1360	0.5	0

*Based on Sodium Inlet Temperature of 400°F

†Based on 50,000 lbs in the dome

TABLE 2
DOME LIFETIMES* - HOURS

ANL Case	Decay Heat	SODIUM TEMPERATURE		
		400 F (ANL)	600 F	800F
a ₂₀	0.78%	10 ¹⁵	10 ¹²	10 ⁹
d ₂₀	2.0%	10 ⁷	10 ⁵	10 ³
a ₄₀	1.05%	10 ¹¹	10 ⁸	10 ⁵
d ₄₀	2.0%	6000	60	3

*Based on 50,000 lbs in dome

REACTOR VESSEL SYSTEM

PC70-17

FIGURE 1

ATTACHMENT III
CONTROL OF TRITIUM

SUMMARY

An interest was expressed by the committee regarding the control of tritium in FFTF. The normal release of tritium is .088 Ci/day (page 11-5 SER and Table 12.1.8 of the FSAR). The staff agrees with this estimate for normal operation. Small amounts of tritium diffuse through the primary piping system walls mixing with the nitrogen atmosphere of the cells. The nitrogen cell atmosphere is processed to remove tritium. Ultimately trace amounts of tritium find its way to the atmosphere. Some tritium diffuses through the IHX tubes and then through the piping walls directly to the atmosphere. However, most of the tritium forms a hydride with sodium in the primary loops and is deposited in the primary cold traps along with NaO₂ and other radioactive mass transfer products. The staff would expect the cold traps to ultimately be disposed of and treated as solid waste. Some additional information pertaining to tritium Behavior and Control is reproduced from HEDL TME 73-39, as well as Table 12.1-8 reproduced from the FFTF FSAR and are both provided here (pgs. III-2 to III-4) for the Committee's information.

CONTROL OF TRITIUM

Reference Source HEDL TME 73-39
Sodium Engineering and Technology Report
Oct., Nov., Dec., 1977

C. TRITIUM BEHAVIOR AND CONTROL IN BREEDER REACTORS (FF126)

J. C. McGuire and E. O. Badgett

1. Objectives

The development program for Tritium Behavior and Control in Breeder Reactors was shared jointly by Argonne National Laboratory (ANL) and the Hanford Engineering Development Laboratory (HEDL). This program is now complete. At HEDL the Tritium Development Program objectives included determining permeation rates for hydrogen species through reactor piping and steam generator materials, testing tritium monitors and tritium analytical methods, and testing methods of tritium control for liquid metal cooled reactors by engineering analysis and experimentation.

2. Summary of Accomplishments

- Tritium permeation tests were conducted in Source Term Control Loop #3 (STCL-3). The Hydrogen Diffusion Apparatus was used for both hydrogen and deuterium work. Results on 304 SS, 316 SS and Croloy (2-1/2 Cr, 1Mo, Fe, and ferritic alloy) show that the permeation rate of hydrogen species through oxidized reactor piping and steam generator materials is very low. The low permeation rate coupled with high cold trap efficiency in picking up tritium and hydrogen almost eliminates the problem of tritium release from the reactor piping and containment. Typical permeation rates are shown in Table 2 for tritium concentrations of 1-40nCiT₂/gmNa.

TABLE 2

TRITIUM PERMEATION RATE THROUGH REACTOR MATERIALS

<u>Material</u>	Permeation Rate	
	$\frac{\text{mm}^3\text{-mm}}{\text{sec-m}^2\text{Pa}^{1/2}}$	$\frac{\text{cc-mm}}{\text{hr-cm}^2\text{-atm}^{1/2}}$
Clean 316 SS	24.7×10^{-4}	2.9×10^{-4}
Oxidized 316 SS	15.3×10^{-4}	1.8×10^{-4}
Oxidized 304 SS	18.7×10^{-4}	2.2×10^{-4}

- . In hydrogen and deuterium permeation studies at 13-1300 Pa (0.1-10 Torr), permeation rates through Croloy (2-1/4 Cr, 1Mo, Fe alloy) were strongly affected by the oxide layer formed at 300-550°C. Rates for oxidized Croloy were 30 times lower than those for "clean" Croloy. The 304 stainless steel showed only a factor of five difference between oxidized and unoxidized surfaces even when oxidized for 1000 hours at 550°C in air. The oxide layer on Croloy is unstable in vacuum and is rapidly reduced by hydrogen. This is not the case with 304 SS which gave almost identical permeation rates to vacuum and to air.
- . During the course of this research the most useful tool for determining tritium concentration in sodium was the ANL Tritium Meter.⁽³⁾ This simple device has a wide temperature range and is easily calibrated using a standard spiked sweep gas for the proportional counter and controlled injections of tritium into loop sodium. Tritium permeation through the test materials was detected by three methods, a commercial "Triton" meter, the gas proportional counter and liquid scintillation counting. For routine work the proportional counter was most convenient.
- . The cold trap was found to have a substantial effect on tritium concentration control in the loop sodium. With 10% flow fraction in the cold trap, and system turnover every 2.5 minutes, tritium content in the loop dropped by a factor of 10 within two hours and 20 within 16 hours. Cold trap temperature is important to tritium retention in the trap. When the cold trap operated at 155°C (312°F), loop tritium content was four times higher than when the cold trap was at 115.5°C (240°F). The above data were used by ANL-E in completing an analytical model of tritium control in FFTF and CRBR.⁽⁴⁾

3. Conclusion

This program was successful in demonstrating that tritium release from a liquid metal reactor can be adequately controlled. The results imply an even greater usefulness when considering the tritium problems to be addressed in the magnetic and laser fusion reactor programs.

FSAR
(REFERENCE SOURCE)

TABLE 12.1-8
ESTIMATED TRITIUM RELEASE^a

CASE	COLD TRAPPING RATE (gal/min)		TRITIUM CONCENTRA- TION (Ci/cm ³)		TRITIUM RELEASE RATE FROM DHX ^b (Ci/d)	TRITIUM RELEASE RATE ^c TO INERTED CELLS (Ci/d)	
	PRIMARY	SECONDARY	PRIMARY	SECONDARY		PRIMARY	SECONDARY
1	120	45	8.2×10^{-8}	5.3×10^{-9}	0.044	0.054	0.003
2	60	45	1.5×10^{-7}	1.0×10^{-8}	0.088	0.105	0.005
3	0.0	45	3.3×10^{-6}	2.1×10^{-7}	1.61	2.16	0.107
4	120	0.0	8.3×10^{-8}	5.0×10^{-8}	0.416	0.055	0.025
5	60	0.0	1.6×10^{-7}	9.8×10^{-8}	0.82	0.108	0.049
6	0.0	0.0	Release rate increases until 100% release occurs —				—
7	120.0	30.0	2.3×10^{-7}	4.8×10^{-8}	0.404	0.15	0.023

^aCases 1-6 — Hydrogen removal efficiency = 73%

Case 7 — Hydrogen removal efficiency = 25%

^bA release rate exceeding 1300 Ci/d is necessary to produce MPC for controlled areas at the DHX exit point. To produce MPC for uncontrolled areas requires a release rate greater than 50 Ci/d.

^cTritium released to inerted cells will be oxidized and collected in the CAPS, removing it as a source of environmental contamination.

Reference: — Kebele, T.J., "Estimates of the Tritium Distribution in the FFTF," HEDL-TME 74-6, 1975.⁽²⁶⁾

ATTACHMENT IV
SMOKE DETECTORS

The Committee expressed interest regarding the possible use of smoke detectors for containment isolation purposes. The main reason that radiation detection was selected for containment isolation was that it was the most reliable and sensitive (time response) parameter to use for containment isolation (Ref. SER, page 6-8). If smoke detection (NaO₂ Vapor) had been used, we may have situations where the NaO₂ would not get past the shielding around the reactor head, whereas the Xe, Kr and Argon 41 would get out into the building atmosphere under a seismic event. Radiation detection appears to provide the best overall parameter for containment isolation. In FFTF, several detector locations are used, one in the head access area and several in the heating and ventilating ducts.

ATTACHMENT V
FFTF FIRE PROTECTION PROGRAM

With respect to fire protection, the staff has not completed all of its review of fire protection for FFTF. The staff plans to address the more conventional fire hazards: electrical, cable spreading, penetrations, communications, and detection in a supplement to the SER after a site visit to be scheduled for this purpose. Sodium fires have been addressed in the SER pages 6-3 through 6-5 and are also mentioned in Section 9.8 of the Plant Fire Protection system in the FSAR.

A site visit was scheduled for October 11 and 12 to review the balance of the FFTF Fire Protection program but had to be delayed due to funding problems. It is now scheduled for November 7 and 8, 1978.

The principal mode of sodium fire protection in FFTF is by oxygen depletion, i.e., by maintaining nitrogen atmosphere in the primary system cells and also flooding the secondary system cells with nitrogen. Supplementary information regarding sodium fires is indicated on pages V-2 to V-7 of this attachment which outline the elements of the fire protection program and also provide FSAR References pertinent to the fire protection review. Portable CaCO_3 extinguishers and protective clothing are also available if needed.

FFTF FIRE PROTECTION PROGRAM

The Standard Review Plan (Section 9.5.1) provides the fundamental guiding concepts for fire protection for nuclear facilities,

- I. Defense-in-Depth to protect safety related functions
- II. Management responsibility for the fire protection program and its administration
- III. Establishment of specific fire areas (boundary conditions for fire walls and doors)
- IV. Formalized Fire Hazard Analysis for fire areas - heat content and fire protection in a given area
- V. Fire Detection and Suppression - System Design Basis for fire areas
- VI. Electrical cable, penetrations and trays and use of water on electrical cable fires
- VII. Control of Combustible materials throughout the facility.
- VIII. Ventilation control and isolation at fire area boundaries
- IX. Lighting and Communication in fire areas
- X. Portable equipment to aid fire fighting in the facility

FIRE PROTECTION REVIEW

- A. The SER NUREG-0358 addresses only sodium fires in and outside of the FFTF containment and their safety related significance to the initiation of potential radiological accidents and their evaluation in relation to plant safety.
- B. An outline of the FFTF Fire Protection System is given in Section 9.8 of the FSAR and Supplemental information is given in the General Plant Fire Hazards evaluation which is still being reviewed by the staff.
- C. The site assessment of the FFTF fire protection plan implementation was scheduled for October 19, 1978 and is now scheduled for November 7, 1978.
- D. The staff's evaluation of the FFTF conventional portion of the Fire Protection Plan will be addressed in a supplement to the FFTF SER.
- E. The FFTF FSAR References to Fire Protection are given in the last attached table on page V-5.
- F. FFTF Sodium spills and Fire Evaluation Report by the applicant is given in HEDL - TME 75-122 dated April 1976.
- G. Primary Sodium System Leaks
 - 1. The work in HEDL TME 75-122 and independent evaluations done by the staff's consultants indicates that spray leaks from the hot leg piping (pump discharge) from 50 lbs/sec upwards to 3000 lbs/sec only have relatively small effects on peak cell pressure, from 15 to 20 psig. The time of peak pressure rise is sodium rate dependent. A 50 lb/sec spray leak would be more characteristic of a split or branch pipe failure. The FFTF inert IHX cells can accommodate these peak transient pressures.
 - 2. Other cells containing primary sodium are also inerted, steel lined and tend to contain cooler sodium at lower pressures. The calculated effect from leaks in these sodium systems generally tend to form sodium pools on the steel lined floor which produces lower cell pressures and in the event of sprays also produce lower pressures.
 - 3. Leaks in primary sodium systems are in inerted steel lined cells and become self-extinguishing due to oxygen depletion.

FIRE PROTECTION REVIEW - ATTACHMENT V CONTINUED

H. Secondary Sodium System Leaks

1. Secondary sodium system fires are analogous to industrial chemical fires. There is inconsequential radioactivity associated with the burning of secondary sodium.
2. Secondary sodium piping is routed from the IHX through pipe chases to the secondary sodium pump tank cell. This cell has a steel lined floor, self-closing H&V dampers and provisions for flooding the cell with nitrogen in the event of fire.
3. Pipe tunnels from the pump cell to the DHX Building have steel lined drip pans below the piping, and provisions for nitrogen flooding of the tunnels.

I. DHX Sodium Leaks

1. The DHX Building has some limited capability for inert gas flooding and oxygen depletion control. There are no steel lined drip pans under piping in this area. The physical separation of the DHX units reduces the possibility of more than one unit or loop being involved at any given time in a fire.
2. Each DHX unit is fully isolatable and may be flooded with inert gas. Fans and dampers are interlocked for shutdown in the event of fire.

J. Instrumentation

A plant fire protection instrumentation system is provided throughout the facility to detect and annunciate the location of fires. Inert gas flooding is the primary means for controlling sodium fires and is manually activated.

K. Portable Equipment

Protective clothing, coveralls, gloves, helmets, portable breathing apparatus and hand-held and cart-type extinguishers are provided at appropriate locations in the facility.

FSAR REFERENCES TO THE FIRE PROTECTION SYSTEM

ATTACHMENT V

Section

1.4.2.6.8	Safety & Fire Protection
3.1.1.3	Design Criteria - Fire Protection
T 3.2	Seismic Category I Systems
5B.4.2.4	Reactor Coolant System
6.1.4.2.1	Sodium Spills
6.1.4.2.2	
6.1.4.2.3	Mobiltherm Fires
6.1.4.2.4	Insulation Fires
6.4	Habitability Systems
7.3.9	H&V Instrumentation
7.3.9.2.2	Control Area Air Conditioning
7.3.9.2.3	Control Area Life Support System
7.3.10	Fire Protection Instrumentation
8.2.3.8	Cable Tests
8.2.4.8.1	Electrical Fire Protection
9.3	Nitrogen System
9.4	H&V (includes control area)
9.6	Sanitary Water Supply
9.8	Fire Protection
11.10.2.6.5	IEM Cell Fire Protection
13.3.6.3.1	Emergency Measures, Corrective Action, Fire Control
15.2.4	Sodium Fire Analysis
15.2.5	Mobiltherm Fire Assessment
15.2.6	Electrical Fire Assessment

SODIUM AND COMBUSTIBLES IN CONTAINMENT

MATERIAL	MAXIMUM QUANTITY	LOCATION	TYPE OF FIRE PROTECTION
Primary Sodium	10,694.0 ft ³	Reactor Vessel, Containment	N ₂ inerting
	5,712.3 ft ³	HTS, Primary Loops, Containment	N ₂ inerting
	3,369.6 ft ³	Primary Processing & Storage System, Containment	N ₂ inerting
	19.0 ft ³	Primary Sampling, Containment	N ₂ inerting
Secondary Sodium	2,697.6 ft ³	HTS, Secondary Loops, Containment	N ₂ inerting
	1,509.8 ft ³	HTS, Secondary Loops, DHX-E	N ₂ flooding and space isolation
	1,509.8 ft ³	HTS, Secondary Loops, DHX-S	N ₂ flooding and space isolation
	1,509.8 ft ³	HTS, Secondary Loops, DHX-W	N ₂ flooding and space isolation
	240.9 ft ³	CLS, Secondary Loops, DHX-E	N ₂ flooding and space isolation
	240.9 ft ³	CLS, Secondary Loops, DHX-W	N ₂ flooding and space isolation
	47.7 ft ³	Secondary Processing System HTS-E	N ₂ flooding and space isolation
	488.8 ft ³	Secondary Processing System, HTS-W	N ₂ flooding and space isolation
	0.3 ft ³	Secondary Sampling, HTS-E	N ₂ flooding and space isolation
	0.7 ft ³	Secondary Sampling, HTS-W	N ₂ flooding and space isolation
Nak	79.2 ft ³	NaK IDS and Cold Trap Cooling System Containment	N ₂ inerting, Type "D" portable extinguisher
Mobiltherm (MTL)	830 gal.	IDS piping	(foam)

Fig. 1. Cell Pressure History for a Sodium Spill of 50 lb/sec in the Primary HTS Cell.

Fig. 2. Cell Pressure History for a Sodium Spill of 100 lb/sec in the Primary HTS Cell.

ATTACHMENT VI
PIPING INTEGRITY AND LEAK DETECTION

Considerable interest on the part of the Committee was expressed in the area of piping integrity and leak detection. Supplemental information is provided expanding on the piping integrity and leak detection issue.

The staff has argued that pipe failures in FFTF are not likely to be a CDA initiator (ref. SER page 5-10 Section 5.2.7 and our letter in Appendix B page B-12). However, to assure piping integrity, it is implicit that one has to be confident that no serious degradation processes are occurring during the life of the facility. This confidence is normally provided by periodic in-service inspection and meaningful continuous aerosol leak detection in LWRs (Reg. Guide 1.45). Historically, most sodium piping failures have been at branch connections, welds, or due to design or construction errors. We believe that periodically monitoring two high stressed welds in each loop is reasonable and minimal and is most likely to pick up any internal cracking if any unanticipated high temperature material degrading phenomena was in process. When the project's ultrasonic testing capability is developed, we believe it should be used as originally proposed on the selected welds.

It should be reiterated that piping failures leading to large leaks are considered by the staff to be very unlikely. Therefore, cold leg piping failures have been excluded as a CDA initiator by the staff for FFTF.

With respect to leak detection the staff has extracted from Reg. Guide 1.45 appropriate sections concerning leak detection which we believe are applicable to FFTF. We also point out in tests conducted at Atomics International and quote from a contractors meeting at DOE on August 16, 1976 (see page VI-7 and page VI-10) that the sodium aerosol had to get out of the insulation and reach the detector to be effective as a leak indicator. We do not believe that the statement given at the top of page 5-11 of our SER or in the recommendations on page 19-6 are inconsistent with information available to the staff. If other tests exist as implied during the projects presentation, we will evaluate them.

Additional information regarding leak detection is also provided in Attachment VI. Some questions were raised by the Committee regarding the design margin in the reactor coolant system for the SSE. There is no specific margin allocated for the SSE. In fact, it is treated as a single faulted event and at the time FFTF was designed one was allowed to use 90% of the plastic instability limit which has since been reduced to 70% in the ASME code. It is our opinion that the piping design satisfies the code requirements at the time it was built (see page VI-11 to VI-13).

PIPING INTEGRITY AND LEAK DETECTION ATTACHMENT VI

Piping Integrity

1. The staff has accepted the projects cold leg leak before break argument - cold leg break not a CDA Initiator.
2. The staff will accept the projects construction quality assurance surveillance in lieu of a base line preservice inspection providing that traceability exists for inspection reports and radiographs for the reactor and reactor heat transport system.
3. The staff continues to recommend that the hot cross over piping welds be volumetrically examined at an appropriate time interval to be established at a later date. The reasons for selection of these welds is given by the project in response to staff's Q1 120.6 5A and Q2 120.12, Supplements 3 and 23. Effectively this would defer the first inspection for a number of years after startup.
4. The staff recommends that the project retain the removable insulation sections in the hot cross over piping and continue with the development of its preferred UT method.
5. The staff accepts the projects statement regarding its proposed method of ultrasonic inspection as being insufficiently developed but encourages its use in the primary system on the six welds discussed in the future when it is developed.
6. The staff believes that the leak detection system and periodic inservice inspection programs are complementary in the sense that they serve the same safety function of assuring a minimum level of piping integrity that will make it very unlikely that serious accidents will occur. The staff believes that an integral part of this overall safety assurance is to know that at least one sensitive leak detector system in each loop will function following an SSE when control power is no longer available in FFTF. The staff has agreed to accept the FFTF Design based on control power considering the status of construction.
7. The staff believes that the most likely source of piping leakage will result from human error, lack of knowledge in design, unanticipated transients or their magnitude, mistakes in fabrication (typical construction errors), or plant operational errors. Further, most leaks will occur in smaller piping and will not likely be fatigue related.

8. In AI DOE 13225 Sodium Piping Survey to July 1977, sodium leaks were reported as follows:

DFR	1967	4" nozzle, 200 litres/day, 2 tons of sodium leaked 1-1/4" crack, 15 yrs operation
Phenix	1973	Mixing Tee, small fire
Raposdie	1966	2 tons, pipe burst while preheating
SCTI	1965	10 lbs, plumb bob welded to pipe
SRE	1969	10 lbs, heater arc, burn through
BR5	1966	Crack in weld
EBR II	1968	640 lbs, maintenance
Hallam	1964	Secondary expansion tank

Leak Detection

EXCERPT FROM REGULATORY GUIDE 1.45 REACTOR COOLANT PRESSURE BOUNDARY LEAKAGE DETECTION SYSTEMS

Seismic Qualification

Since nuclear power plants may be operating at the time an earthquake occurs and may continue to operate after earthquakes, it is prudent to require the leakage detection systems to function under the same conditions. If a seismic event comparable to a safe shutdown earthquake (SSE) occurs, it would be important for the operator to assess the condition within the containment quickly. The proper functioning of at least one leak detection system would assist in evaluating the seriousness of the condition within the containment in the event leakage has developed in the RCPB. The airborne particulate radioactivity monitoring equipment has the desirable sensitivity to indicate RCPB leakage, and it should be included for all plants. Components for the airborne particulate radioactivity equipment should be qualified to function through the SSE.

C. Regulatory Position

The source of reactor coolant leakage should be identifiable to the extent practical. Reactor coolant pressure boundary leakage detection and collection systems should be selected and designed to include the following:

6. The leakage detection systems should be capable of performing their functions following seismic events that do not require plant shutdown. The airborne particulate radioactivity monitoring system should remain functional when subjected to the SSE.
7. Indicators and alarms for each leakage detection system should be provided in the main control room. Procedures for converting various indications to a common leakage equivalent should be available to the operator. The calibration of the indicators should account for needed independent variables.
8. The leakage detection systems should be equipped with provisions to readily permit testing for operability and calibration during plant operation.
9. The technical specifications should include the limiting conditions for identified and unidentified leakage and address the availability of various types of instruments to assure adequate coverage at all times.

Leak Detection

- 1) Cell parameters, N₂, 1-2% O₂; 1000 VPPM moisture
- 2) Leak Rate Range 100 G/HR corrosion rate 30 MILS/MO @ 1050°F in inert gas 1000-6000 VPPM moisture
- 3) SID performance monitoring hot leg pipe annuli for 100 G/HR response: <1 HR monitoring cells response 10 hrs
- 4) DP cell performance monitoring hot leg pipe annuli for 100 G/HR: <10 hrs, monitoring cells. No response; need 30 GPM for 5 min. response at 700°F
- 5) It is not reasonable to monitor the cell ventilation for small leaks expecting the sodium aerosol to penetrate the inner steel jacket and outer aluminum insulation jacket and buildup the aerosol concentration to 10⁻¹¹ G/CC in the cell necessary for the SID to function
- 6) From the shape of the curves if the aerosol can get out into the 30,000 ft³ cell the SID will pick it up in less than one hour if it cannot it could be hours. A direct path to the annulus would be better.
- 7) The statements made at the October meeting on page 364 of the transcript regarding the small difference insulation played in leak detection does not appear to be supported except at low sodium temperatures <600°F when the evaporation and diffusion rates are about the same
- 8) The staff believes, based on the data furnished, that gross cell monitoring for aerosols has not been demonstrated to be adequate for detecting weeping type hot sodium leaks of 100 GR/HR through metallic jacketed insulation

AI TESTS
SMALL CELL MONITORING - NATURAL CONVECTION TEST RESULTS

Test Number	DP Cell		SID		Cable	Contact	H ₂
	Floor Area	Ceiling Area	Floor Area	Ceiling Area			
11	1.0 h	0.7 h	<0.1 h ¹	<0.1 h	0.1h	NR	NR
12	12.1 h	12.4 h	1.2 h ²	NR	NR	NR	NR } vented
16	NR	NR	115.6	NR	NU	NU	12.7 h
17	NR	NR	NR	NU	NU	NU	NR } not vented
11R	<0.5 h	<0.8 h	<1 min	<1 min	NR	NR	NR
13X	23.6 h	25.4 h	15.2 h	15.2 h	15.7	NR	0.58 h
-							

NU - Not Used

~~NR~~ - No Response

¹980°F

²640°F

Figure A-L Na Aerosol Concentration vs Time and Vault Size for a 100 gm/hr Leak

$$C = \frac{R}{P} (1 - e^{-Pt/V})$$

C = aerosol concentration (gm/cc)

R = aerosol leak rate through insulation gm/sec

V = Volume of that cell cm³

P = gas flow in sampling lines cc/sec

VI-7

TABLE VI
SMALL CELL MONITORING - NATURAL CONVECTION TEST PARAMETERS

Test Number	Moisture (vppm)	Oxygen (%)	Temperature (°F)	Sodium Leak Rate (gm/h)	Time (h)	Total Na Leaked (gm)	Remarks
11	1,000	1	980	2	100	200	24 in. pipe
12	1,000	1	640	2	100	200	24 in. pipe
16	1,000	1	500	2	100	200	3 in pipe
17	15,000	21	650	2	100	200	3 in. pipe

} vented

} not vented

ERDA/DOE - CONTRACTORS MEETING - ERDA-76/15

August 16, 1976
LEAK DETECTORS

We will talk about the sodium-to-gas leak detectors. We have liquid sodium detectors, hydrogen detectors, and smoke detectors. I will run through this fast. If there are questions, we have a fellow here looking into details on these and he can show you sketches. All of these detectors are subject to certain conditions for detection. These are temperature and leak rate. To give you a brief synopsis of what the test results look like and what the state-of-the-art is, we have listed these. We have listed both of the sodium ionization detectors in one column. Most of the results are specifically applicable to the Westinghouse sodium ionization detector.

This unit and this unit are critical to placement because you have to get sodium in contact with them, or sodium hydroxide in contact with them. They are electrical contact devices and you have to close the circuit. If the sodium doesn't touch them, they don't react. For the aerosol types, it's not important where you place them except that the aerosol has to get to them. If the aerosol doesn't get to them, they don't key contact. It's important from that aspect. If there is a pipe that is leaking and the insulation is tight enough that the aerosol never penetrates the insulation, and if you have the detectors out of the insulation, they will never detect. We have actually seen that case.

You might indicate where that was.

It was in a test cell.

Small piping with no annular gap. It's stainless steel sheeting material wrapped around it.

We designed that as leak tight as we could possibly make it. Another parameter is susceptibility to false responses. On most of these, we ended up with a question mark. With this old spark plug type, I think in every application I have seen of it, there has been far more false indications than sodium leaks that it has picked up. That has been the history of that device.

What sort of false indications do you get?

It indicates you have a leak. It may be a cabling problem or a short. We have seen rust between contacts.

Are those detectors made to the RDT standard?

There are only a few to the RDT standard. One had rust on it and it was made to the standard. Most of the rest were not made to the standard. I don't think it's germane whether they are made to the standard or not.

ATTACHMENT VII
FFTF NEUTRON FLUENCE/DOSE
ON STRUCTURES AND/OR COMPONENTS

The committee expressed an interest regarding structural radiation fluences. Attachment VII identifies structural elements of interest, their design life and the fluences to which they are exposed. Axial and radial neutron flux plots are also provided; in general, the project has attempted to keep the cumulative strain due to radiation damage below 1%.

Figure 2.2.2.1-3. Axial Neutron Flux Distribution in FTR

Fig. 2.2.2.1-4 Radial Neutron Flux Profile: Core to Cavity Wall

Figure 2.2.2.1.5 Axial Neutron Flux Throughout Reactor Vessel ($n/cm^2 \text{ sec}$)

ATTACHMENT VII
FFTF NEUTRON FLUENCE/DOSE-ON STRUCTURES
AND COMPONENTS

THE ATTACHED FIGURES SHOW THE AXIAL AND RADIAL NEUTRON FLUX PLOTS FOR FFTF.

FLUENCES AND DESIGN CRITERIA ARE GIVEN IN THE SER AS FOLLOWS:

FUEL SER - PAGE 4-16

.7% STRAIN INTEGRITY LIMIT CRITERIA GENERALLY APPEARS AS A REASONABLE LIMIT BASED ON D.F.R. DATA SUPPORTS A FLUENCE LIMIT OF 3×10^{22} N/CM²

INITIAL CONDITIONS OF FFTF REDUCED TEMP AND BURNUP SHOULD BE WELL BELOW THIS LIMIT. IN ADDITION THE FUEL CAN BE REMOVED AND EXAMINED AT FFTF.

CONTROL ROD DRIVE SER PAGE 4-33

THE DRIVE LINE EXTENSION IS SUBJECTED TO A FLUENCE OF 5×10^{20} N/CM²

CONTROL ROD ABSORBER ASSY SER PAGE 4-34

1.5×10^{23} N/CM² ACCEPTABLE DUCTILITY REMAINING AT <1% STRAIN

CORE SUPPORT STRUCTURE SER PAGE 4-35:

CALCULATED FLUENCE 4.7×10^{22} SECTION SER 4.5.2.1

RADIAL SHIELD SER SECTION 4.5.2.2

1×10^{23} N/CM²

INNER ROW 10 YEAR LIFE (REMOVABLE)

OUTER ROW 20 YEAR LIFE

.3% OF STRAIN CALCULATED

INSTRUMENT TREE SER PAGE 4-36

FLUENCE LIMIT SPECIFIED TO END OF LIFE DUCTILITY
>10% ELONGATION (REMOVABLE)

INVESSEL HANDLING MACHINE SER PAGE 4-37

FLUENCE 1.4×10^{20} N/CM²

DESIGN LIFE 10 YEARS

REFLECTORS SER PAGE 4-38

FLUENCE 2×10^{23} NVT

THE ACTUAL DESIGN LIFE WILL BE DETERMINED BY MONITORING
THE TWO ROWS OF REFLECTORS WHICH ARE REMOVABLE

APPENDIX D
UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

FEB 26 1979

MEMORANDUM FOR: T. C. McCreless, Chief, Project Review Branch No. 2

FROM: Themis P. Speis, Chief, Advanced Reactors Branch, DPM

SUBJECT: DOE PLANS FOR ADDRESSING NRC AND ACRS ADVICE REGARDING
START-UP AND OPERATION OF FFTF

In August 1978, the Nuclear Regulatory Commission (NRC) issued its Safety Evaluation Report (SER), NUREG-0358, related to the operation of the Fast Flux Test Facility. On November 8, 1978, the Commission received from the ACRS its report on the Fast Flux Test Facility.

On December 15, 1978 the Regulatory Staff requested the Department of Energy (DOE) (Enclosure 1) to provide their plans and schedules and in general the information and/or documentation showing how DOE is or will be implementing the Commission's and the ACRS advice and guidance regarding the startup and operation of FFTF. DOE responded to this staff request in a letter dated January 26, 1979, which is attached to this memorandum for your information (Enclosure 2). In addition, DOE has provided their interpretation of some recent sodium/concrete interaction tests performed at SANDIA under NRC sponsorship (this does not include test No. 13 which was performed December 11, 1978 in which substantial sodium-concrete interaction occurred). The staff is presently evaluating this information and will document its assessment in the FFTF SER Supplement which will also address all the other items that either remained outstanding during the FFTF review or were resolved during the later stages of the FFTF SER preparation but due to time constraints were not documented in the SER. Please note that even though DOE has provided some information on their "air cleaning program" including funding levels through 1982, it is not clear from their response if and how they plan to address/implement the ACRS advice regarding "--- development of additional mitigation measures, such as a sand and gravel filter, for possible addition to the controlled venting system to provide still further assurance of limited radioactivity releases"

If we may be of further assistance to you please contact me or Mr. H. B. Holz the LPM for FFTF.

A handwritten signature in dark ink, appearing to read "Themis P. Speis", is written above the typed name.

Themis P. Speis, Chief
Advanced Reactors Branch
Division of Project Management

Enclosures:

1. NRC Ltr. dtd. December 15, 1978
2. FFTF Project Letter dtd. January 26, 1979

cc: A. Bates, ACRS

Enclosure 1

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

DEC 15 1978

Project 448

Dr. R. L. Ferguson
Program Director
Nuclear Energy Programs
Department of Energy
Washington, D. C. 20555 (MS-B107)

Dear Dr. Ferguson:

SUBJECT: NUCLEAR REGULATORY COMMISSION ADVICE REGARDING STARTUP AND
OPERATION OF FFTF

In August 1978, the Nuclear Regulatory Commission (NRC) issued its Safety Evaluation Report (SER), NUREG-0358, related to the Operation of the Fast Flux Test Facility. As has been documented in the SER and discussed with the Advisory Committee on Reactor Safeguards (ACRS), the staff's review was directed toward evaluating the adequacy of the design to ensure safe operation of the facility. The Department of Energy (DOE) has the responsibility for deciding how to implement the NRC and ACRS advice, and has the final responsibility for the design and safe operation of the FFTF facility. On November 8, 1978, the Commission received from the ACRS its report on the Fast Flux Test Facility. The ACRS report provided advice and recommendations which are consistent with the advice and recommendations provided to DOE in NRC's Safety Evaluation Report.

Our normal procedure at this point of facility completion is to close out open issues identified in the SER relating to plant operation by issuing an SER Supplement based on our evaluation of additional information received from the applicant and or developed by the NRC staff. This supplement also includes a discussion of the advice and/or comments by the ACRS in its report on the FFTF.

Based on the ACRS report and the Safety Evaluation Report the following comments are provided to assist you in the development of plans, schedules, and preparation of future documentation regarding DOE's implementation of the Commission's and the ACRS advice and guidance:

(1) Piping Integrity

The staff's summary position on piping integrity is documented in Section 19.3C of the Safety Evaluation Report. In a letter to the ACRS, dated October 26, 1978 and in discussions with the ACRS, at the November 2, 1978 full committee meeting, the staff indicated that it "will accept the

project's construction quality assurance surveillance in lieu of a base line preservice inspection provided that traceability exists for inspection reports and radiographs for the reactor and reactor heat transport system." This in no way alters the staff's position regarding periodic monitoring of the six hot crossover welds in the primary sodium system as soon as practicable, following development of the high temperature ultrasonic testing methods. This position is consistent with the ACRS report. Again, we reaffirm that the staff does not consider monitoring the secondary sodium piping system as equivalent to periodic examination of the high stressed primary sodium system welds, as suggested by the FFTF project office in a letter to NRC, dated November 24, 1978. Therefore, the staff requests that it be kept fully informed of UT development plans, schedules and progress of development of this method of examination, and when it is expected that it will be utilized on the reactor coolant system.

(2) Plans for Startup Testing

The NRC staff and the ACRS are in accord that natural convection through the heat transport systems will have to be demonstrated using nuclear heat. The staff will review and evaluate the adequacy of information obtained from the natural circulation test program, including both natural circulation and computer model verification and DOE's analysis of these data in support of the position that natural circulation is a viable option for decay heat removal.

The staff, therefore, requests that it be kept informed of plans and schedules for startup testing, including natural circulation modes of operation, and any changes to the proposed "Interim Summary of FFTF Natural Circulation Test Plans," dated September 1977.

(3) Core Debris-Containment Materials Interaction Experiments

Because of the limited amount of applicable experimental data regarding core debris (sodium, fuel, steel) interactions with FFTF containment materials (steel liners, concrete), a need exists to obtain more information in this area to improve the confidence for predicting the behavior of such material interactions in the event of a core meltdown accident. This need has been further reinforced by the apparent divergence in the results from some sodium-concrete reaction experiments performed by the Hanford Engineering Development Laboratory (HEDL), under DOE sponsorship, and by Sandia Laboratories, under NRC sponsorship. The staff, therefore, requests that it be furnished with DOE's plans and schedules of future work in this area, as well as the experimental data and results when they become available, and the potential impact of this work on the implementation of means to control the containment hydrogen concentration and a controlled venting system (e.g., better quantification of the venting rates) for FFTF (see item (4) below).

(4) Containment Margins for Core Melt Events

The NRC staff in its SER recommended that means be provided to monitor and control the hydrogen concentration in the containment following a low probability core melt accident with reactor vessel melt-through, and that means be provided to vent the containment in a controlled manner to prevent containment failure. The ACRS concurred with this staff recommendation and in addition recommended the "... development of additional mitigation measures, such as a sand and gravel filter, for possible addition to the controlled venting system to provide still further assurance of limited radioactivity releases...." The staff is therefore requesting that DOE provide the necessary information (i.e., designs, plans and schedules, etc.) regarding the implementation of the staff's and the ACRS' recommendations regarding the above.

If we may be of further assistance to you, please contact Mr. H. B. Holz, LPM for FFTF.

Sincerely,

William P. Gammill
Assistant Director for Standardization
and Advanced Reactors
Division of Project Management

Enclosure 2

Department of Energy
Fast Flux Test Facility Project Office
P. O. Box 550
Richland, Washington 99352

JAN 26 1979

Nuclear Regulatory Commission
Washington, D. C.

ATTN: Mr. William Gammill, Assistant Director
for Standardization and Advanced Reactors
Division of Project Management

Gentlemen:

PLANS FOR ADDRESSING NRC AND ACRS ADVICE REGARDING START-UP AND
OPERATION OF FFTF

Reference: Letter, W. P. Gammill to R. L. Ferguson, "Nuclear Regulatory
Commission Advice Regarding Start-up and Operation of FFTF,"
dated December 15, 1978.

In August 1978, the Nuclear Regulatory Commission issued the Safety
Evaluation Report (SER), NUREG 0358, related to the operation of the
Fast Flux Test Facility (FFTF). Subsequently on November 8, 1978,
the Advisory Committee on Reactor Safeguards (ACRS) issued its report
on the FFTF. The reference letter requested amplification of project
plans for disposition of the NRC and ACRS advice. This letter provides
information regarding FFTFPO plans to address the issues identified in
the reference letter.

As noted in the reference letter, the principal issues identified in the
NRC and ACRS reports relate to containment margins and design provisions
to reduce the consequences of a postulated melt-through of the FFTF core
into the reactor cavity under hypothetical accident conditions. During
the NRC and ACRS review, the FFTF Project indicated that an HCDA followed
by failure of the reactor vessel to contain the core debris is of such an
exceedingly low probability that it is considered to be a Class 9 accident
in terms of the NRC guidelines. Design features are not required to accom-
modate such an accident. Moreover, the consequences of such an event
calculated for the existing containment design, utilizing conservative
models and assumptions, are considered acceptable for this hypothetical
accident. On the basis of these safety assessments, the FFTF Project Office
concluded that modification of the FFTF containment is unnecessary and
unwarranted.

JAN 26 1979

In their Safety Evaluation Report, the NRC staff recommended that the FFTF Project provide "....the necessary means to monitor and control the hydrogen concentration in the containment under those conditions in which the hydrogen concentration could threaten containment integrity..." and provide "....the means to vent the containment building in a controlled manner to prevent containment failure by over pressurization." We believe that information presented to ACRS by NRC and Sandia regarding the sodium/concrete testing and the H₂ deflagration observed during Sandia's sodium/concrete testing was incomplete and may have led to incorrect conclusions. Since the last ACRS meeting, we have met at Sandia with NRC, HEDL, and Sandia to reassess the differences between Sandia and HEDL tests and to review the incidents in which hydrogen deflagration occurred in the Sandia test chamber. A meeting report containing details of these evaluations is provided as Enclosure I to this letter. In the Sandia testing where a hydrogen deflagration was observed, hydrogen apparently accumulated in a nitrogen inerted chamber which housed an inner test enclosure. Approximately six hours after completion of the sodium/concrete reaction the inerted chamber had cooled down, the nitrogen purge supply to the chamber was terminated, air was introduced into the chamber, and hydrogen deflagration ensued. This result is not unexpected based on hydrogen flammability limits, and indeed was predictable. The experience at Sandia does not alter the validity of the conclusion that hydrogen will be ignited by the presence of sodium and burn in the containment air atmosphere for conditions corresponding to a reactor melt-through accident. The enclosed meeting report also documents H₂ auto-ignition data from the 1 yd² HEDL LFT-5 test. These data further support the conclusion that hydrogen will ignite and burn for conditions existing in the melt-through accident analyses. This information was included in a presentation to ACRS, but has not previously been documented and provided to NRC.

The Sandia sodium/concrete interaction test data reviewed in Enclosure I did not indicate substantial discrepancies between HEDL and Sandia tests. We are aware that Sandia has since performed an additional test, and we plan to evaluate its results as well. Also we would propose serious consideration of joint test planning by HEDL and Sandia to assure development of a mutual understanding of the technology and request that we be kept fully informed of plans, schedules, and results of the NRC testing at Sandia.

Our technical assessment for FFTF remains that the existing safety margins are adequate; however, we intend to re-examine the technical information, including the most recent test data, relevant to the accident probabilities and safety impacts of containment venting and hydrogen control design features. To aid in these assessments, we have authorized efforts by HEDL

to establish criteria and prepare designs of systems for containment venting and for hydrogen detection and control. These studies will be available to support disposition of NRC/ACRS advice regarding containment design modification.

In addition to the features suggested by NRC, the ACRS report recommended "...development of additional mitigation measures, such as a sand and gravel filter, for possible addition to the controlled venting system..." A substantial DOE sponsored development effort in this area is already funded and under way. The program objective is principally to develop and proof test containment air cleaning technology for application to future breeder reactor designs. In response to the ACRS advice, we have reviewed the scope of the current containment air cleaning program to ensure the ACRS concerns are being addressed. The funding level for this program in FY 1979 is \$1,139K with a projected total program cost from FY 1974 through FY 1982 of \$7,770K. The program work scope and major milestones are summarized in Enclosure II.

The reference letter also requests that NRC be kept informed of plans, schedules and progress in the areas of ultrasonic testing (UT) technology development and natural circulation start-up testing. Plans and schedules for development of UT technology utilizing the FFTF secondary piping system were provided in our letter of November 24, 1978. Results from this program will be supplied when they become available. In addition to the "Interim Summary of FFTF Natural Circulation Test Plans" dated September 1977, NRC has been supplied with and placed on routine distribution for changes to the natural circulation Test Specifications (TS-51-3A018, TS-51-4A009, and TS-51-5A008). Additional test predictions and test results will be provided to NRC when they become available.

We thank NRC and ACRS for your review and advice on these matters and we will continue to keep you informed regarding the disposition of your advice.

Very truly yours,

A. J. Rizzo, Deputy Director
for Engineering
FFTF Project Office

FTFGDB:E086

Enclosures: (I) Report on Sandia Meeting, Nov. 15-16, 1978
(II) HEDL Containment Air Cleaning Development Program

NRC
Page 4

JAN 1 1977

cc w/enclosures:

H. B. Holz, NRC

Helen Gearin, NRC

T. P. Speis, NRC

H. Hollister, E-201

T. A. Mangelsdorf, HEDL

D. E. Simpson, HEDL

From: Safety Systems Development

Hanford Engineering Development Laboratory

Position: 2-2312 W/221T HE

Date: December 18, 1978

Subject: HEDL CONTAINMENT AIR CLEANING DEVELOPMENT PROGRAM

To: J. M. Atwood
D. E. Simpson

cc: H. Alter DOE/RRT-HQ
WF Sheely C-44
DD Stepnewski C-75

The following description of the HEDL Containment Air Cleaning Development Program was given to G. D. Bouchey, DOE/FFTFPO, Friday, December 15, 1978, to be included as an attachment to a letter to NRC. The purpose of the description was to inform NRC of the present program.

The evaluation of air cleaning system (ACS) concepts for FFTF (Item 3.1.3) is within the applicable work scope of 189a FF287/NTB Subtask D, Containment Air Cleaning Systems, with no additional funding required for FY-1979. However, any further work in relation to an ACS concept for FFTF which might be scheduled for FY-1980, such as conceptual design and/or cost estimate, is not within the present scope of Subtask D. Additional funding or a major re-direction of the present Containment Air Cleaning program may be required. I would hope that further discussions would be held, with the opportunity of obtaining additional funding as first choice, before a decision was made to redirect the present program.

I feel the program as presently outlined is responsive to the NRC questions regarding an air cleaning system for FFTF and will develop the required technology.

L. D. Muhlestein
Manager

gmg

Attachment: HEDL Containment Air Cleaning Development Program

December 18, 1978

HEDL CONTAINMENT AIR CLEANING DEVELOPMENT PROGRAM

Sponsored by U.S. Department of Energy
Division of Reactor Development and Demonstration
Office of Reactor Safety

Prepared by L. D. Muhlestein

1.0 PROGRAM OBJECTIVE: Develop and proof test containment air cleaning systems (ACS) for controlling effluents as an added safety margin for postulated breeder reactor accidents.

2.0 PROGRAM TASKS:

- Evaluation of Containment Air Cleaning System Concepts
- Aerosol Behavior Tests
- Air Cleaning System Development Tests
- Large-Scale ACS Demonstration Tests.

3.0 PROGRAM DESCRIPTION:

3.1 Evaluation of Containment Air Cleaning System (ACS) Concepts

3.1.1 Task Objectives: Provide evaluation, technical guidance, assistance, and correlation for development of breeder reactor air cleaning systems.

3.1.2 Progress:

- Evaluation completed of various ACS concepts for postulated head release accident and various containment options. Report issued (HEDL-TME 76-41).
- Evaluation completed of various ACS concepts for postulated containment vent/purge modes following a postulated reactor melt-through accident. Report issued (TC-836).
- Coordinate DOE/NRC aerosol behavior test programs, and serve on various U.S. and international aerosol working committees.

3.1.3 Continued Work Scope:

- Complete analytical evaluation of applicable ACS concepts for FFTF postulated accident conditions.

3.2 Aerosol Behavior Tests

3.2.1 Task Objectives: Provide baseline data for evaluation of air cleaning tests, and provide large-scale data for evaluation of aerosol behavior and containment response computer models.

3.2.2 Progress:

- Six aerosol behavior tests defined and reviewed by DOE/NRC.
- Two tests performed with sodium pool fire aerosol source. Reports completed (HEDL-SA-1381, HEDL-SA-1466, HEDL-SA-1686).
- Two tests performed with sodium spray fire aerosol source. Preliminary report completed (HEDL-SA-1686).

3.2.3 Continued Work Scope:

- Complete remaining aerosol behavior tests and report results.
- Compare aerosol behavior tests data with computer codes.

3.3 Air Cleaning System Development Tests

3.3.1 Task Objectives: Perform development tests to evaluate candidate air cleaning concepts and support large-scale ACS test program.

3.3.2 Progress:

- Facility modification complete and test loop operational.
- Aerosol mass loading characteristics measured for HEPA filters. Report completed (HEDL-SA-1465).
- Sand and gravel bed test equipment fabricated.

3.3.3 Continued Work Scope:

- Complete sand and gravel bed filter development tests.
- Complete development tests with novel air cleaning concepts.

3.4 Large-scale ACS Performance Tests

3.4.1 Task Objectives: Demonstrate performance of air cleaning systems under postulated breeder reactor accident conditions.

3.4.2 Progress:

- First ACS concept selected to be proof tested. Concept is an aqueous scrubber system consisting of a spray quench chamber, a venturi scrubber, and a high efficiency fibrous scrubber.

- Design of aqueous scrubber system completed; components procured and system installed. Tests to be initiated January 1979.
- Completed evaluation of nucleation and capture of condensible airborne contaminants in an aqueous scrubbing system. Report issued (HEDL-TME 78-82).

3.4.3 Continued Work Scope:

- Complete proof test series for aqueous scrubber system to include testing of modifications to concept design as required depending on test results.
- Design, fabricate and install second ACS concept; complete proof testing.
- Complete proof testing for developed novel air cleaning concepts.

4.0 MAJOR PROGRAMMATIC MILESTONES

• Program initiated	December 1974
• ACS concepts evaluation completed for head release accident	December 1975
• ACS concepts evaluation completed for melt-through accident	February 1977
• Containment Systems Test Facility completed	May 1977
• First pool fire aerosol behavior test completed	September 1977
• First spray fire aerosol behavior test completed	January 1978
• HEPA filter loading tests completed	July 1978
• Aqueous scrubber system concept selected and installed at CSTF	November 1978
• Complete sand and gravel bed development tests	August 1979
• Complete analytical evaluation of ACS concepts applicable to FFTF	August 1979
• Complete air cleaning development tests for aqueous scrubber concept	September 1979
• Complete small-scale development tests for alternate concepts	September 1980
• Complete aerosol behavior tests	June 1981
• Complete large-scale performance tests	March 1982

APPENDIX E
AEROSPACE CORPORATION
(NRC Contractor)

REVIEW OF SODIUM-CONCRETE INTERACTIONS

Summary

After reviewing sodium-basalt concrete interactions, we have reached the following conclusions:

1. A sodium-basalt concrete penetration rate of one inch per hour is recommended. In the absence of firm evidence indicating when sodium penetration will end, it is felt that a continuous penetration rate is appropriate.
2. Hydrogen recombination seems to us to be unlikely in significant amounts.
3. The firebricks employed in these tests are not a barrier to sodium. Indeed, this particular firebrick-liner combination may be detrimental if it comes into contact with liquid sodium. The pressure generated as reaction products form may cause the concrete to crack. Firebricks that are inert to sodium should be employed in place of the ones selected. Siliceous firebrick should be avoided.
4. It is difficult to say whether basalt concrete is better or worse than limestone concrete in a sodium environment. Basalt concrete seems to be somewhat more resistant to penetration by sodium but it also seems more prone to cracking than limestone concrete.
5. The questions of the effects of scale, geometry and mode of restraint remain unresolved.
6. Sodium pool temperature appears to affect the chemical reactions that occur and the reaction products that form. A lower temperature sodium environment (600°C) seems to have a more severe effect on basalt concrete than the boiling sodium environment.
7. Sodium pool depth may affect the time required for the occurrence of a reaction and the extent of reaction.
8. The chemistry of the sodium-basalt concrete and sodium-limestone concrete interactions are speculative. Hypothesis have been proposed which seem reasonable but they have not as yet been confirmed.

9. The physical aspects (cracking, spallation) of the sodium-basalt and sodium-limestone concrete interactions are poorly understood.
10. Further research is needed to shed light on both the chemical and physical aspects of sodium-concrete interactions. Given the differences between the experimental conditions, it is difficult to coherently correlate the results. Attention should be paid to devising and selecting experiments to close this gap and gain better insight. More detailed coordination in selecting experiments and experimental parameters for study would be desirable.

Introduction

As of this time, 44 sodium-concrete interactions tests have been performed at Hanford Engineering Development Laboratory (HEDL) and Sandia Laboratories. These tests have examined concretes similar to those used at the Fast Flux Test Facility (FFTF) and proposed for use at the Clinch River Breeder Reactor Plant (CRBRP). The experimental results have varied considerably even under what appear to be similar initial conditions. This variation is not completely understood and is the principal reason for concern about predicting long term effects of a large sodium spill.

Review of Sodium-Concrete Experiments

The small scale HEDL experiments have investigated the penetration of liquid sodium into approximately equal numbers of limestone, basalt and magnetite concrete specimens. Generally in these tests, 50 pounds of liquid sodium were poured on one square foot area specimens. Typically, the HEDL pours were made at higher temperatures than those at Sandia. In a majority of the HEDL tests, sodium was poured at 871°C. HEDL has conducted one large scale, 1000 pound sodium pour on a faulted liner in which firebricks separated the sodium from the basalt concrete.

Although Sandia initially concentrated on studies of the interactions between sodium and limestone concrete, recent experiments have examined basalt and magnetite concrete interactions with sodium. Fewer tests have been performed at Sandia than at HEDL although the experiments have been on a significantly larger scale. Typical experiments have involved pouring approximately 180 kg of liquid sodium on specimens

with areas of about one square meter. The pouring temperature of the sodium in the experiments has varied from 450°C to 760°C. Recent experiments have included steel liners and FFTF firebricks covering the concrete.

Results from the HEDL experiments are summarized in Table I and results from Sandia's tests are shown in Table II. Blanks can be found in some columns for certain tests because data has not been reported. In the two tables, concretes are grouped together by type (limestone, basalt, magnetite). The specimen thickness, area and orientation are listed. Sodium temperatures, masses and depths are also shown. Information is given on the test duration, whether or not an exothermic reaction was observed, hydrogen evolution, extent of penetration and whether or not noises were observed suggesting a vigorous interaction.

Historically, there appeared to be a substantial difference in the results obtained by Sandia and HEDL. In the first three large scale Sandia tests, highly energetic reactions were observed which consumed all the sodium in a short period of time and penetrated as far as 15 cm into the limestone concrete within 3 hrs. In the first seven HEDL small scale tests (1 ft²) on magnetite and limestone concrete, penetration of less than half this distance occurred over 24 hours with an excess of sodium. All these tests were made on horizontal surfaces.

The differences in results initially were attributed to scale effects since the surface areas in the Sandia tests ranged from 3 to 15 times those of HEDL. Significantly, the sodium pool depths in the initial Sandia sodium pours were approximately half those used by HEDL. Also, the initial HEDL sodium temperatures were 300°C higher than those used by Sandia. It was thought at the time that the higher temperatures would produce more energetic reactions.

Following these initial tests, the experimental results changed. The fourth Sandia limestone concrete test employed a 0.5 m² surface area and twice the previous sodium depth. A penetration depth of only 0.5 cm was observed. This was followed by 6 more tests which produced little penetration, but some experimental difficulties such as hydrogen deflagrations or explosions. HEDL conducted two limestone and two magnetite concrete tests using vertical test surfaces and observed penetrations which came close to 15 cm. HEDL observed that the penetration into a vertical surface exceeded that into a horizontal surface by a factor of two. On the other hand, Sandia observed greater penetration downward than radially.

TABLE I: H E D L SODIUM-CONCRETE INTERACTION TESTS

(All tests are listed by type of concrete)

TEST NUMBER	CONCRETE TYPE/ THICKNESS	SURFACE AREA ORIENTATION	TEMPERATURE AVE. /MAX.	SODIUM MASS/ DEPTH	EXOTHERM. REACTION YES/NO	TIME OF REACTION	HYDROGEN EVOLVED	PENETRATION MAX. /AVE.	NOISES (POPS, BUMPS, ETC)
SC-4	Limestone 30 cm	0.092 m ² Horizontal	677°C 802°C	22.7 Kg 25.4 cm	Yes	8 Hrs	0.18 Kg	8.4/6.1 cm	Yes
SC-5	Limestone 30 cm	0.092 m ² Horizontal	871°C 871°C	22.7 Kg 25.4 cm	No	2 Hrs	0.10 Kg	4.3/2.5 cm	No
SC-6	Limestone 30 cm	0.092 m ² Horizontal	871°C 871°C	24.4 Kg 27.3 cm	No	8 Hrs	0.26 Kg	4.8/3.8 cm	No
SC-8	Limestone 30 cm	0.092 m ² Horizontal	871°C 871°C	23.6 Kg 26.4 cm	Yes	24 Hrs	0.28 Kg	5.3/4.4 cm 15 cm dam.	No
SC-10	Limestone 30 cm	0.092 m ² Vertical	871°C 1093°C	18.1 Kg 20.3 cm	Yes	8 Hrs	0.25 Kg	13/8.9 cm	Yes
SC-12	Limestone 30 cm	0.092 m ² Vertical	871°C 954°C	19.1 Kg 21.4 cm	Yes	24 Hrs	0.23 Kg	14/9.2 cm	Yes
SC-13	Limestone 30 cm 3.15 Kg NaOH	0.092 m ² Horizontal	871°C 932°C	22.7 Kg 25.4 cm NaOH 24 Hrs	Yes	26 Hrs	Yes	7.6/4.4 cm	Yes
SC-14	Limestone 30 cm 3.15 Kg NaOH	0.092 m ² Horizontal	871°C	22.7 Kg 25.4 cm NaOH 0 Hrs		24 Hrs	Yes	7.6/5.7 cm	?
SC-18	Limestone 30 cm	0.092 m ² Horizontal	871°C	3.15 Kg NaOH + Con- crete chunks		8 Hrs	No	5.0/2.5 cm	?

TABLE I: H E D L SODIUM-CONCRETE INTERACTION TESTS (CONTINUED)

TEST NUMBER	THICKNESS	SURFACE AREA ORIENTATION	TEMPERATURE AVE. /MAX.	SODIUM MASS/ DEPTH	EXOTHERMIC REACTION YES/NO	TIME OF REACTION	HYDROGEN EVOLVED	PENETRATION MAX. /AVE.	NOISES (POPS, BUMPS, ETC)
SC-19	Limestone 30 cm 3.15 Kg NaOH	0.092 m ² Horizontal	677°C	22.7 Kg 25.4 cm		8 Hrs	Yes	6.4/3.8 cm	?
SC-15	Basalt 30 cm	0.092 m ² Horizontal	871°C	22.7 Kg 25.4 cm		2 Hrs	Yes	4.4/3.2 cm	No
SC-16	Basalt 30 cm	0.092 m ² Horizontal	871°C	22.7 Kg 25.4 cm		8 Hrs	Yes	7.6/7.0 cm	No
SC-17	Basalt 30 cm	0.092 m ² Horizontal	871°C	45.4 Kg ? Na Leak		24 Hrs	Yes	7.0/5.7 cm	?
SC-20	Basalt 30 cm 3.15 Kg NaOH	0.092 m ² Horizontal	500°C	22.7 Kg 25.4 cm NaOH@0 Hrs		8 Hrs	Yes	0.6 cm	?
SC-21	Basalt 30 cm 3.15 Kg NaOH	0.092 m ² Horizontal	649°C	22.7 Kg 25.4 cm NaOH@0 Hrs		8 Hrs	Yes	2.5 cm	?
SC-22	Basalt 30 cm	0.092 m ² Horizontal	871°C 829°C	22.7 Kg 25.4 cm		24 Hrs	Yes	7.6 cm	No
SC-23	Basalt 60 cm	0.092 m ² Horizontal	871°C 860°C	45.4 Kg 25.4 cm Na Leak		16 Hrs		4.4 cm	
SC-24	Basalt 30 cm	0.092 m ² Horizontal	593°C No Heating	22.7 Kg		? Cooled rapidly		1.3 cm	
SC-25	Basalt 60 cm	0.092 m ² Horizontal	871°C 871°C	45.4 Kg Na Leaks		100 Hrs		Significant side wall attack 16.5 cm	

TABLE I: H E D L SODIUM-CONCRETE INTERACTION TESTS (CONTINUED)

TEST NUMBER	CONCRETE TYPE/ THICKNESS	SURFACE AREA ORIENTATION	TEMPERATURE AVE/MAX.	SODIUM MASS/ DEPTH	EXOTHERMIC REACTION YES/NO	TIME OF REACTION	HYDROGEN EVOLVED	PENETRATION MAX./AVE.	NOISES (POPS, BUMPS ETC)
SC-1	Magnetite 30 cm	0.092 m ² Horizontal	677°C 677°C	22.7 Kg 25.4 cm	No	8 Hrs	0.18 Kg	3.8/1.9 cm	No
SC-2	Magnetite 30 cm	0.092 m ² Horizontal	871°C 871°C	23.6 Kg 26.4 cm	No	2 Hrs	0.08 Kg	4.1/1.6 cm	No
SC-3	Magnetite 30 cm	0.092 m ² Horizontal	871°C 960°C	24.0 Kg 26.9 cm	Yes	8 Hrs	0.28 Kg	4.8/3.4 cm	Yes
SC-7	Magnetite 30 cm	0.092 m ² Horizontal	871°C 871°C	24.5 Kg 27.4 cm	No	24 Hrs	0.29 Kg	6.9/4.5 cm	No
SC-9	Magnetite 30 cm	0.092 m ² Vertical	760°C 996°C	19.5 Kg 21.8 cm	Yes	8 Hrs	0.31 Kg	14.7/7.7 and 15 cm hole	Yes
SC-11	Magnetite 30 cm	0.092 m ² Vertical	871°C 927°C	18.1 Kg 20.3 cm	Yes	24 Hrs	0.36 Kg	12.4/8.6 cm	Yes
LFT-1	Basalt under 1.6cm steel liner 0.3cm gap	0.002 m ² hole Horizontal (5cm hole in plate)	871°C	22.7 Kg		8 Hrs	?	3.8 cm	?
LFT-2	Morex Firebrick Insulating Brick Mortar Basalt	0.002 m ² hole Horizontal 5 cm hole in plate	871°C	22.7 Kg		8 Hrs	?	6.4 cm into 7.6 cm Firebrick	
LFT-3	Limestone Steel Liner with 4" gap	0.002 m ² hole Horizontal 3/4" MgO aggregate	871°C	22.7 Kg		8 Hrs		5.7/1.9 cm	
LFT-5	1.6 cm steel 0.6 cm gap 7.6 cm firebrick 6.4 cm insulating brick 0.6 cm mortar Basalt	0.018 m ² Hole (0.85 m ²) 5 gauge	704°C 860°C	454 Kg 56 cm	100 Hrs Na Consumed @70-80 Hrs		H ₂ @113 l/min max.	14 cm max @ctr	

TABLE II: SANDIA SODIUM — CONCRETE INTERACTION TESTS

(All tests are listed by type of concrete)

TEST NO.	CONCRETE TYPE/ THICKNESS	SURFACE AREA/ ORIENTATION	SODIUM TEMPERATURE POUR/MAX.	SODIUM MASS/ DEPTH	EXOTHERMIC REACTION YES/NO	TIME OF REACTION	HYDROGEN PENETRATION EVOLVED MAX. /AVE.	NOISES (POPS, BUMPS, ETC)
1	CRBRP Limestone 30.5 cm	0.29 m ² ? Horizontal	550°C 800°C	21 Kg 8.6 cm	Yes	22 min.	Yes 8.3 cm	Explo.
2	CRBRP Limestone 38.1 cm	1.17 m ² Horizontal	550°C 800°C	108 Kg 11.2 cm	Yes	45 min	Yes 9.1/7.6 cm	Yes
3	CRBRP Limestone 38.1 cm	1.47 m ² ? Horizontal	550°C 740°C	186 Kg 15 cm	Yes	3 Hrs	Yes 15 cm	Explo. . Large spalled chunk
4	CRBRP Limestone 38.1 cm	0.55 m ² Horizontal	450°C	188 Kg 30 cm	No	4 Hrs (8 min)	Yes 0.5 cm	Yes(once)
5	CRBRP Limestone	1.17 m ² Horizontal	540°C 460°C	186 Kg	No	2 Hrs	? slight (0.5 cm)	No
6	No. 5 reused + 36 Kg NaOH		700°C	186 Kg	No		? slight	No
8	CRBRP Limestone	1.17 m ² Horizontal	550°C	127 Kg 15 cm	Moderate		?	
9	CRBRP Limestone	Horizontal	600°C	182 Kg 35 cm	Yes		Yes-H ₂ expl. T + 2 min,	
10	CRBRP Limestone	Horizontal	625°C	182 Kg			Yes-H ₂ expl. T + 30 sec.	

TABLE II: SANDIA SODIUM - CONCRETE INTERACTION TESTS (CONTINUED)

TEST NO.	CONCRETE TYPE/ THICKNESS	SURFACE AREA/ ORIENTATION	SODIUM TEMPERATURE POUR/MAX.	SODIUM MASS/ DEPTH	EXOTHERMIC REACTION YES/NO	TIME OF REACTION	H ₂ Evolved	PENETRATION MAX./AVE	NOISES (POPS, BUMPS, ETC)
7	Basalt	Horizontal	485°C	45 Kg	No		Yes	Slight	No
11	Basalt 38.1 cm	1.6 m ² Horizontal	760°C 600°C	128 Kg 15 cm	Yes	4.5 Hrs Na Limited	Yes	18/5 cm	Yes
14	Magnetite + Bricks + Liner	1.17 m ² Horizontal	639°C 870°C	128 Kg 13 cm	Yes	1.8 Hrs Na limited	Yes	extensive ≥ 15 cm	Yes
12	1 cm steel liner + insu- lating firebrick + Basalt concrete	0.65 m ² Horizontal	600°C 865°C	69.9+45.5 30+20 cm	Yes	10.5 Hrs (exp) Na Limited	Yes	? severe cracks leaks	Yes
13	steel liner firebrick-14 cm Basalt-38.1 cm	0.55 m ² Horizontal	600°C 925°C	182 Kg 25 cm	Yes	10 Hrs? 12 Hrs Na limited	Yes	14 cm brick +38.1/25 cm (total penetration) 52.1 cm total	Yes

During this period, sodium-limestone concrete interactions were of considerable interest as limestone concrete had been proposed for use in the CRBRP. At this point, Sandia proposed a mechanism to explain the different results obtained at Sandia and HEDL. It was suggested that NaOH rather than sodium was primarily responsible for the observed attack on concrete. The NaOH formed when sodium reacted with water driven from the concrete by heat. It was suggested that the concrete would experience little attack during the period before the sodium became saturated with NaOH. When additional NaOH was formed, the liquid NaOH formed a separate layer which in turn attacked the concrete. It was argued that the shallow pools used in the first three Sandia limestone concrete tests quickly saturated with NaOH allowing the energetic reaction to begin promptly. The deeper sodium pools used by HEDL, and also used in Sandia test No. 4, required a longer time to reach saturation with NaOH. In addition, it was suggested that reaction products would provide some shielding for the concrete from further attack. This proposed mechanism led both Sandia and HEDL to perform tests in which NaOH was added to the liquid sodium before it came into contact with limestone concrete in order to achieve immediate saturation. Sandia's test with NaOH (test No. 6) showed less reaction than before and HEDL's tests (tests SC-13, SC-14 and SC-19) showed about the same penetration. These results did nothing to confirm the NaOH attack hypothesis.

Basalt Concrete

More recently, emphasis has shifted towards study of the sodium-concrete interactions expected in the type of basalt concrete that is being used in FFTF. HEDL has conducted nine small scale basalt concrete tests at temperatures from 500°C to 871°C. With one exception, the total penetration in these tests varied from 3.2 to 7.6 cm but could not be correlated to either test duration or temperature. All of the successful sodium tests were performed at 871°C. The total penetration data for the 2, 8 and 24 hour tests are conflicting, with the 24 hour data showing less penetration in one case than the 8 hour tests. However, the 100 hour test showed a 16.5 cm penetration suggesting continuing attack for a long period. Furthermore, the sodium leaked out before the conclusion of the experiment so that the test was probably sodium limited. HEDL also conducted two lower temperature tests with an initial charge of NaOH added

to the sodium. The penetrations observed in these tests were less than the penetrations in tests with sodium alone.

When Sandia test No. 11 is considered, the basalt concrete data look much worse. In that test, penetrations of as much as 18 cm were observed in a 4.5 hr test at 600°C. The average depth of penetration over the entire bottom of the crucible was 5 cm. Sandia also has conducted two tests with basalt concrete protected by firebrick and a steel liner. Substantial penetration and severe cracking occurred in both of the tests. All of the Sandia basalt tests were limited by exhaustion of the sodium supply so that it is not possible to say how far penetration might have proceeded.

Post test examination of the reaction products has shown differences that suggest that two different reaction mechanisms are at work. In the quiescent HEDL high temperature horizontal tests, a dense, hard layer, apparently Na_2O formed between the unreacted sodium and the concrete below. This layer is believed to provide some degree of protection to the concrete from further attack. On the other hand, the reaction zone in the lower temperature Sandia tests is a porous, cinder-like sodium silicate.

Up to this time, there have been nine tests at HEDL with sodium and bare basalt concrete plus three others with liners. Sandia has completed one bare basalt concrete test and two other tests where liners and firebricks provided protection from sodium. It is useful to try to compare the effect of different parameters on these tests.

Sodium Temperature

The temperature of the sodium pool in most of the relatively quiescent HEDL tests was much higher than that of the relatively energetic Sandia test when the exothermic reaction started. The HEDL tests were maintained at 871°C whereas the Sandia basalt test was conducted at 760°C. At both of these temperatures, water is driven from the concrete. However, it has been suggested by Sandia that, at the lower temperature, water reacts with sodium to form NaOH and NaH. At higher temperatures, Na_2O alone is formed. NaOH is a liquid at these temperatures and attacks concrete. In addition, the reaction products are soluble in NaOH so that there is little opportunity for a protective layer to form. On the other hand, at higher temperatures, the reaction generating Na_2O is apparently favored. Although Na_2O can attack concrete, its potential is limited as it is a solid at these temperatures. This seems to be one instance

where the chemical environment is more severe at lower temperatures than at higher temperatures. It should be noted that the temperature of the sodium is elevated to 800°C or higher by the exothermic reaction once it starts. There is also some evidence from thermocouple data that the areas in which the reactions occur can reach temperatures in excess of 1000°C.

There is one flaw in the above explanation of the difference between the HEDL and Sandia tests that should be noted. In HEDL tests, SC-20 and SC-21, 3.15 kg of NaOH were added to sodium at temperatures of 500°C and 649°C, respectively. This quantity should have saturated the sodium. Under the hypothesis advanced above, extensive penetration should have occurred. However, the observed penetrations were only 0.6 and 2.5 cm, respectively. There are other factors that may affect the extent of penetration that should also be considered; one of these is sodium pool depth.

Sodium Pool Depth

If the concrete is principally attacked by NaOH, rather than by sodium, a longer time will be required for attack to occur in a deeper pool. A longer time is required in order to saturate the sodium with NaOH when the pool is deep. The NaOH has only limited contact with the concrete until the pool is saturated. At that time, a separate NaOH layer forms in contact with the concrete allowing the attack to proceed expeditiously. In the Sandia shallow pool test, this saturation would have occurred relatively quickly leading to rapid penetration. In tests with deeper sodium pools, in this temperature range, a longer time should be required for saturation to occur.

It should be noted that if this hypothesis is used to explain the lack of a reaction in HEDL tests SC-20 and SC-21, which were the only HEDL tests in the same temperature range as Sandia's tests, there is a problem. In these tests, NaOH was added to the sodium in a quantity that should have assured initial saturation. Consequently, the results in tests SC-20 and SC-21 remain unexplained.

Failed Liner Tests

These experiments have been conducted to simulate the effects of a failed steel liner. The steel liner is part of a system consisting of the liner and firebricks and is intended to protect the underlying basalt concrete from liquid sodium. In each of the

tests, sodium reacted with the firebricks and, in some of these tests, the sodium was consumed before the bricks were totally penetrated. In some of the later tests at Sandia, using more sodium, the reaction continued into the concrete, apparently until all the sodium was consumed. In tests at HEDL, less concrete penetration was observed and excess sodium remained.

The reaction products expand considerably, exerting extreme pressures on the containing surfaces. The pressure can bend upward the "failed" steel plate and force reaction products through the hole. In addition, the expansive forces produced under these conditions can cause serious cracks in the concrete allowing sodium to penetrate to the bottom of the concrete crucible. In the last tests at Sandia, the pressure caused the failure of all of the rebars as well as severe cracking in the basalt and magnetite concrete structures.

Further, the reaction products appear to be porous and provide little or no barrier to the flow of sodium to the reaction zone. In Sandia test No. 12, additional sodium was added to the top of the reaction mass and the exothermic reaction was restarted shortly thereafter. This demonstrates clearly that the reaction products are porous to sodium and offer no protection to the concrete below.

In reviewing the sodium concrete data, it appears that a major constituent in the fire-resistant brick and insulating brick is involved in the chemical reactions with liquid sodium. The first choice candidate is SiO_2 , which is also present in the basalt concrete. Sodium reacts with silica and forms a host of complex chemical compounds involving the rest of the components of concretes. Some of these compounds have low melting points while the behavior of other compounds is not easily predicted due to their complex phase diagrams. In any event, it would appear best to avoid, to the extent possible, the inclusion of siliceous materials in the reaction zone. It would be better to replace the fire-resistant and insulating bricks with a material such as MgO , which is inert in liquid sodium.

Cracking

HEDL and Sandia have observed very different amounts of cracking. Cracking is important because it reduces strength, provides a pathway for sodium to reach and attack fresh concrete and because it provides a potential escape route for gases, such as hydrogen, generated by sodium reactions.

In HEDL's tests with basalt concrete, except for SC-25, there has been little or no evidence of cracking. On the other hand, Sandia observed substantial 1/8 inch radial cracking in the bare basalt test (test No. 11) and very severe cracking in tests 12 and 13 where firebricks and liners were also present. In these tests, sodium preferentially penetrated into cracks as far as 20 cm.

HEDL has asserted that the cracking observed by Sandia is due to thermal expansion in the Sandia crucible geometry. In test No. 11, the HEDL interpretation has some merit. However, since only hairline cracks were observed in similar crucibles of limestone concrete (admittedly a different material), it seems that other factors besides thermal expansion may be involved. In tests No. 12 and No. 13, the pressure exerted by the voluminous reaction products trapped between the steel liner and the concrete almost certainly was a major contributor to the severe cracking that was observed.

In the actual reactor system, firebricks and a steel liner are present and, consequently, it seems quite possible that reaction products may form that will generate pressures that could cause cracking. Furthermore, cracking is quite possible at the point where the reactor room floor and sidewalls meet. Consequently, the problems that cracking may cause should be given further consideration.

Penetration Rates

Under the conditions of the 2, 8 and 24 hour HEDL tests, the extent of sodium penetration into concrete ranged from 3.2 to 7.6 cm. In HEDL test SC-25, however, a penetration of 16.5 cm was observed over 100 hr at 871°C. At some point during the test, sodium leaked away so that it is not possible to determine if the penetration would have proceeded further.

In Sandia test No. 11, a penetration of 18 cm was observed after only 4.5 hours at 600°C. There is no way of knowing how far the sodium would have penetrated in this test as the sodium supply was exhausted. In Sandia test No. 13, sodium penetrated all the way through 14 cm of firebrick and 25 cm of concrete in ten hours at 600°C. Although the reaction was sodium limited, some sodium penetrated entirely through the 38-cm thick basalt specimen along a crack and was found underneath the crucible. From information supplied by thermocouple temperature excursions, Sandia was able to

determine that the reaction front advanced into the concrete in test No. 13 at a rate given by $D = 1.6 t^{1/12}$ where D is the penetration distance and t is time in minutes.

About a year ago, we recommended the use of a continuous penetration rate of one inch per hour for sodium attacking basalt concrete. Although we believed that there were factors that might eventually cause the penetration to end, such as the buildup of reaction products, there was no firm evidence suggesting when this might occur. Consequently, in the absence of such evidence, a continuous penetration rate was recommended. After reviewing the new data that has been obtained in the intervening time, we feel that the earlier estimate is still reasonable.

Hydrogen Generation

The explosions that occurred at Sandia demonstrate the problems that can occur if hydrogen can escape through cracks or other paths and collect elsewhere. When a proper air mixing ratio and ignition source appear at the collection point, a deflagration or explosion can result.

Substantial hydrogen generation can be expected from reactions between sodium and water from the concrete. The hydrogen data obtained in HEDL test LFT-5 is not conclusive because of the limited sodium penetration beyond the firebricks into the concrete. Furthermore, we do not believe that hydrogen recombination from an autocatalytic reaction is probable to any significant extent.

Comparison of Basalt and Limestone Concretes

The penetration rates for sodium attacking limestone and basalt concrete under the conditions of the experiments at HEDL are similar. Some experiments at Sandia suggest that sodium will penetrate limestone concrete more rapidly than basalt concrete under the conditions of the Sandia tests. However, the basalt test specimen (test No. 11) at Sandia displayed greater cracking than similar limestone specimens. Although arguments have been made that this cracking was caused by the effects of geometry and thermal expansion, these arguments are speculative since both the limestone and basalt specimens at Sandia had the same geometry. Experiments with molten UO_2 on both concretes suggested that limestone concrete was more resistant to attack by molten core debris, probably due to its higher melting point and because of extensive evolution of CO_2 which can carry away some of the heat.

Based on the data available today, it is difficult to say that one concrete is better than the other. Limestone concrete seems to be more crack resistant than basalt concrete but it can also be argued that it may be more easily penetrated by sodium. Limestone concrete is more resistant to molten UO_2 . Additional experimentation is needed as the mechanisms of the sodium concrete interactions are not well understood.

Response to L. D. Muhlestein's Trip Report of December 14, 1978

We are unable to agree with a number of the positions stated in the conclusions of the HEDL trip report. Our response to the individual conclusions follows.

1. The Sandia tests show evidence of substantially greater penetration rates than the HEDL tests. Furthermore, since the Sandia tests are sodium limited, the actual total penetration may be substantially greater. We are recommending a continuous penetration rate of one inch per hour for basalt concrete.

The different reaction products observed also suggest that two different chemical processes may be occurring under the different conditions of the tests performed at the two laboratories. It appears that more severe attack may occur at lower sodium temperatures.

Although the HEDL specimens displayed little cracking, the Sandia specimens were severely cracked. At least some of this cracking is apparently due to the accumulation of reaction products which in turn generated substantial pressures.

2. The Sandia and HEDL faulted liner tests gave dissimilar results. After 12 hours, all of the firebricks were consumed and the reaction apparently ended because the sodium supply was exhausted.

In the HEDL test, after 100 hours, metallic sodium remained and only half of the firebricks reacted with sodium.

3. The cracking in the Sandia tests is relevant because the cracking may have been generated by pressure produced by the accumulation of reaction products between the liner and concrete. Furthermore, if cracking occurs, fresh surfaces are presented to the sodium for attack so that sodium penetration will be enhanced.

4. The hydrogen deflagration observed in the Sandia tests is very relevant to FFTF containment margins. If hydrogen can escape through cracks and accumulate, it can later, under the right conditions, cause an explosion.
5. LFT-5 did not penetrate very far beyond the firebricks into the concrete, unlike some of the Sandia tests. The reaction between the water driven from the concrete and sodium will generate substantial quantities of hydrogen.

Hydrogen recombination in significant amounts seems unlikely.

APPENDIX F
UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

APR 6 1979

MEMORANDUM FOR: H. B. Holz, NRR
A. R. Marchese, NRR

FROM: T. J. Walker, RSR

SUBJECT: SANDIA COMMENTS ON L. D. MUHLSTEIN'S SUMMARY OF THE
NOVEMBER 15-16, 1978 MEETING ON SODIUM/CONCRETE

The subject comments are enclosed and forwarded for your information. The comments assume familiarity with the subject but additional detail will be provided if needed.

One detail that I have added is the relationship of the data in Figure 1 of the enclosure to the source of the measurement. The first four points on the curve are thermocouple transients from couples located between the firebrick layers, below the second row of firebrick; and within the concrete. The "penetration" is thus the thermocouple position at the time the Na/concrete reaction reached the immediate vicinity of the thermocouple. The first point denotes this rapid temperature rise at the thermocouple between the two layers of firebrick and the second is a similar rise at the thermocouple just below the second layer of brick. The third point is 1 cm into the concrete, while the fourth point is 6 cm into the concrete. The last point is derived from the actual penetration measurement of the sectioned crucible (the test was sodium limited).

T. J. Walker
T. J. Walker
Experimental Fast Reactor
Safety Research Branch

Enclosures:
As stated

March 27, 1979

Dr. T. J. Walker
Experimental Fast Reactor Safety
Research Branch
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

- Ref. 1) Letter, T. J. Walker, NRC, to R. L. Coats, Sandia,
dated 2/12/79
- 2) Trip report, HEDL/Sandia Working Group Meeting on
Sodium-Concrete and Faulted Liner Tests; November 15-
16, 1978, L. D. Muhlestein, HEDL to D. E. Simpson,
HEDL, dated 12/14/78.

Dear Dr. Walker:

In your referenced letter to R. Coats, you asked two
questions:

1. What is Sandia's interpretation of the Sandia and
HEDL sodium/concrete results?
2. What is Sandia's best current evaluation of the
hydrogen explosion mechanism and your best estimate
of basalt attack rate as a function of time?

In answering these questions, we were specifically asked to
comment on the referenced trip report by L. Muhlestein.

HEDL provided a summary of the trip report in which
opinions were stated and conclusions reached. Every Sandia
test to date on basalt concrete has been sodium limited. HEDL
refers to this fact numerous times--then ignores it in drawing
conclusions, i.e., in discussing test 12, the following state-
ment is made: "The actual data, which is most important,
demonstrates that even in the right circular cylindrical geo-
metry, the combination of firebrick, insulating brick and
refractory mortar was quite successful in inhibiting the sodium
migration to the concrete." Since test 12 was sodium limited,
we cannot conclude that more sodium would--or would not-- have
caused further reaction. However, Sandia test 13 has since
shown that the only inhibition offered by the above combination

of firebricks is that it takes a certain quantity of sodium to react with the bricks--beyond that--the bricks are not a barrier to the attack of concrete by sodium.

HEDL SUMMARY ITEM 1

HEDL states that, to date, Sandia and HEDL test data agree. The actual three areas of dissimilarity are:

1. Penetration depths roughly agree--rates do not. HEDL tests typically run many times longer than Sandia tests and show less penetration.
2. Sodium pool temperatures--HEDL typically tests above 800°C, Sandia tests at 550-600°C. The important point here is that different reaction products are formed. In the HEDL tests, sodium oxide, a solid, is produced forming a hard impervious barrier. In the Sandia tests, sodium hydroxide, a liquid, is formed as an intermediate, with the final product being a porous, cinder-like sodium silicate.
3. Cracking of concrete test article--Sandia's large scale basalt crucibles have shown severe radial cracking in both a bare crucible and a crucible with firebrick and steel insert. Subsequently, a test has been run in which the crucible was presented with nearly a one-dimensional sodium attack. In this test, severe horizontal cracking occurred. HEDL's one large scale test showed little, if any, cracking.

HEDL SUMMARY ITEM 2

It is stated that Sandia's test 12 appears very comparable to HEDL's LFT-3 and LFT-5. All three tests involved firebricks and a flawed steel liner. The Sandia test terminated at 12 hours. No metallic sodium remained. Essentially all of the firebricks had been consumed by the reaction with sodium. The HEDL tests ran for 100 hours. Metallic sodium remained after the tests, only about one-half of the firebricks had been consumed by the reaction with sodium.

HEDL SUMMARY ITEM 3

The cracks in Sandia's large scale basalt concrete crucibles are postulated by HEDL to be due to crucible geometry and not germane to FFTF safety analyses. Sandia is not involved in FFTF Safety Analyses and therefore has no comment. The cracks are further postulated not to lead to extensive additional sodium-concrete interaction. In large scale tests on limestone concrete, Sandia has seen minimal cavity sidewall attack and only hairline radial cracks. In the one bare, large scale basalt concrete test, Sandia found extensive, non-uniform sidewall attack with severe sidewall cracking. The sodium had preferentially attacked into the cracks as far as 20 cm. In view of the fact that the test was sodium limited, Sandia cannot concur that cracks do not lead to extensive additional sodium-concrete interaction.

HEDL SUMMARY ITEM 4

The hydrogen gas explosion (or deflagration) that occurred in Sandia's large scale basalt concrete test 12 is the subject of conjecture. Sandia's position on this explosion is only that, with cracking, a mechanism has been shown to exist whereby hydrogen gas being evolved by the sodium-concrete interaction region can escape from that region and collect elsewhere--where the gas can await proper mixing ratios and an ignition source.

HEDL SUMMARY ITEM 5

HEDL states that hydrogen gas data from LFT-5 confirms the bounds used in safety analysis. LFT-5 did not penetrate much beyond the firebricks into the concrete. Most of the water which reacts with the sodium to produce hydrogen is in the concrete. How the WATRE code's agreement with experimental data is justified is not clear.

Sandia's current best estimate of sodium penetration into siliceous firebrick and basalt concrete is derived from the experimental data of test 13 which presented a one-dimensional attack and showed that the attack progressed uniformly over a large area unaided by cracking. The rate derived from this data is shown in Figure 1.

Dr. T. J. Walker

-4-

, I hope these remarks have answered your questions. If you desire further discussion, please call me on 475-6162.

Sincerely,

R. U. Acton

RUA:1537:am

Copies to:

1530 W. E. Caldes
1537 N. R. Keltner
1537 L. A. Kent
4420 J. V. Walker
4422 R. L. Coats
5534 J. E. Smaardyk
5846 R. A. Sallach

Penetration of the Sodium Reaction Front into the Firebrick and Concrete of Test 13

Figure 1

APPENDIX G
UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

DEC 29 1978

MEMORANDUM FOR: M. Silberberg, Chief
Experimental Fast Reactor
Safety Research Branch

FROM: T. J. Walker
Experimental Fast Reactor
Safety Research Branch

SUBJECT: SODIUM/CONCRETE INTERACTION TEST # 13

The relatively extreme wall cracking for the basalt crucible in test # 12 resulted in an attempt to run test # 13 in a one dimensional mode. A cross section of the test crucible is shown in Figure 1. The mild steel liner extended to the inner bottom surface of the crucible. The firebrick was mortared on top of the concrete and an inner "liner" bottom was tack welded to the liner on top of the brick. Products of the reaction of sodium and concrete or firebrick could thus break the tack welds and force the liner bottom upward within the cylindrical liner similar to a piston in a cylinder. Thermocouple locations are shown in the figure and were located to provide information on the progress of the sodium/concrete reaction. The liner and MgO isolated the walls of the crucible to effectively one dimensionalize the sodium/concrete reaction in the vertical direction.

The test (#13) was run on Dec. 11, 1978 with the following principal results:

- There was no sidewall attack on the crucible.
- All the sodium was consumed although the maximum system capability of ~400 lb. of Na was dumped.
- The crucible bottom is about 15 in. thick and about 10 of the 15 in. were penetrated. In fact some reaction products were found under the crucible although these may have exited through a crack.
- The bottom of the crucible is effectively cracked off of the upper cylindrical part and separated from the top part by about 4 inches. There are about 15 vertical rebars in the crucible walls and apparently all have been broken.
- The bottom of the liner was forced upward, breaking the tack welds, and tilted about 6 inches. About 2 in. of reaction products were found on top of the liner. The liner had a "standard" defect, 1/4 x 6 in. Chunks up to a cubic foot in volume were broken out of the crucible near the bottom although the upper cylindrical portion is intact. Some fine cracking is evident in the cylindrical part but no major cracks are evident above the firebrick level.

An analysis and report are in preparation, however, a few obvious results are evident from the figures and will be noted here:

Figure 2: Penetration to the 1 cm T/C, CR01, required about 2.5 hr. The reaction with the concrete was exothermic and the temperature exceeded the initial pouring temperature. (The sodium in the crucible is temperature controlled to 600°C.) However, the relatively flat tops (with the exception of short excursions) of the temperature - time traces suggest that the reaction is self terminating at about 750°C. Note the gradual conduction and temperature rise at CR02 which is 6 cm below the surface of the concrete.

A reaction at the 6 cm location initiates at about 4.6 hr. and again appears self limiting at 700 to 750°C.

Figures 3 & 4: The thermocouples in the firebrick indicate clearly the times at which the sodium reaction with the firebrick has penetrated to the T/C locations, these times are consistent with those indicated by the concrete T/C locations.

Figure 5: The pool temperature traces show significant variations at about the time the firebrick thermocouples were showing fast excursions. This is probably the time that the bottom of the liner was forced upward by the products of the reaction and the relative location of the thermocouple is no longer valid. The temperature reading of the thermocouples are thus also questionable after these large displacements.

Figure 6: The MgO thermocouple trace appears to rise quite uniformly and reasonably. The side wall temperature may indicate some reaction at the 6 to 7 hour point or it may be the result of the extreme distortion within the liner and crucible wall. An attempt to evaluate the most probable cause and effect will be made after disassembly.

An analysis and report will be prepared by Sandia and the above comments must be considered as preliminary. However, it appears that the conclusions will include:

1. The test design (Test # 13) was successful in providing one dimensional penetration rate data.

2. An analysis of the restraint of the base of this crucible and that of a typical cell should provide further confidence in the validity of the test.
3. The apparent 700 to 750°C temperature limit to the reaction should be studied in the separate effects tests.

Test # 13 concludes the large scale basalt tests introduced into the generic study of sodium/concrete reactions. These tests were performed expeditiously within CY'78 by Sandia in support of the NRR review of FFTF. The Sandia staff associated with this program are to be commended for an excellent performance on a task requiring ingenuity, quality and timeliness.

T. J. Walker

T. J. Walker
Experimental Fast Reactor
Safety Research Branch

AKSR Staff

cc: H. Holz, NRR
A. Marchese, NRR
T. Speis, NRR
W. Gammill, NRR
R. Savio (5), ACRS
J. Walker, Sandia
R. Coats (3), Sandia
R. Acton, Sandia
W. Von Rieseman, Sandia
N. Keltner, Sandia
H. Alter, DOE
L. Muhlestein, HEDL

Figure 1. Location of Instrumentation (NOT TO SCALE)

TEST 13

SODIUM POOL DEPTH 576.8 CM

SODIUM DUMPED AT 700°C

TEST DATE 12/11/78

* 12 THERMOCOUPLES IN CONCRETE (1, 7 & 10 at 1 cm; 2 at 6 cm; 3 & 9 at 19 cm)

4 IN Na POOL (PL1 at 0 cm from BOTTOM; PL2 at 5.1 cm; PL3 at 58 cm & PL4 at 61 cm)

1 IN MgO

6 IN FIREBRICK (TOP 1 & TOP 2 ON SURFACE OF DENSE FIREBRICK NEXT TO LINER BOTTOM

MID 1 & MID 2 IN HORIZONTAL JOINT BETWEEN DENSE & INSULATING FIREBRICK

BOT 1 & BOT 2 IN HORIZONTAL JOINT BETWEEN INSULATING FIREBRICK & CONCRETE

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

APR 11 1979

MEMORANDUM FOR: M. Silberberg, Chief
Experimental Fast Reactor
Safety Research Branch

FROM: T. J. Walker
Experimental Fast Reactor
Safety Research Branch

SUBJECT: SODIUM/MAGNETITE CONCRETE REACTION TEST
MARCH 28, 1979

The subject test is #14 in the series of large scale sodium/concrete pours into crucibles. The magnetite concrete crucible cavity was 4 ft. in diameter and 11 in. deep.

Sodium (128 kg.) at 913°K was poured into the crucible making a pool depth of 13 cm. The pour cooled to 733°K but increased to 823°K in 6.5 minutes. This rate of heating requires more power than the heater capacity. Therefore an exothermic reaction started within this 6.5 minute period. The pool temperature continued to increase until it leveled off at 1,100°K at 16.5 minutes. Some noise was picked up by the microphone but no loud reports. The noise was described as a hiss or rush of gas. During a five minute period that the heating rate was linear with the heaters turned off, the power to the sodium was calculated at 137 kw in addition to all losses to the concrete and top hat.

The test cell was vented by leaving open a man sized port in the side. A fan circulated air to prevent hydrogen build-up in the cell and thus prevent explosions similar to those in previous tests. Five minutes into the test visual observations were made of smoke and flame. The flame looked orange in color through the smoke. Fifteen to twenty minutes into the test, the smoke diminished and side wall cracks were apparent that looked .5 in. wide. However, later it was apparent that the cracks were very thin ($< 1/16$ in.) and that water was running from the cracks and dripping to the floor.

Thirty to thirty five minutes into the test the sodium fire smoke disappeared and thin hydrogen flames were apparent at the cracks and around the base of the top hat seal. At forty-two minutes a loud report was heard with flames and sparks. Then the reaction rate diminished rapidly and the test cooled.

APR 11 1979

M. Silberberg

-2-

Early Observations After Disassembly

The crucible had very little side wall cracking. The cracks in the cooled condition are "hairline" widths. No metallic sodium remained. The surface on the top and in the center of the pool area was a clinker like material. The outer half of the radius of the top surface was covered with a slag like material that is quite hard. The side walls had no appreciable attack. Penetration measurements will be reported after the crucible is sectioned.

T. J. Walker
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Experimental Fast Reactor
Safety Research Branch

cc: ARSR Staff
H. Holz, NRR
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W. Gammill, NRR
R. Savio (5), ACRS
J. Walker, Sandia
R. Coats (3), Sandia
R. Acton, Sandia
S. Key, Sandia
W. Von Rieseman, Sandia
N. Keltner, Sandia

APPENDIX H

UNITED STATES

DEPARTMENT OF ENERGY

WASHINGTON, D.C. 20545

ON

APR 11 1979

MEMORANDUM FOR: M. Silberberg, Chief
Experimental Fast Reactor
Safety Research Branch

FROM: T. J. Walker
Experimental Fast Reactor
Safety Research Branch

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S. Key, Sandia
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N. Keltner, Sandia

APPENDIX I

NRC STAFF RECOMMENDATIONS REGARDING CONVENTIONAL FFTF FIRE PROTECTION PROGRAM (NON-SODIUM)

Based on our review and the assumption of safe plant shutdown by natural convection of sodium and air, we conclude that after implementation of the following staff recommendations, the FFTF fire protection program for plant areas not containing sodium will assure that no single fire in these areas will prevent safe plant shutdown and will meet the intent of General Design Criterion 3 and is, therefore, acceptable.

1. It is our recommendation that a valve be installed in the fire pump discharge header between the two discharge lines to the underground fire main and that any other single failure incidents which could shutdown supply from both fire pumps be corrected by adding pipe and/or valves to prevent such an occurrence.
2. It is our recommendation that the applicant provide fixed, self-contained lighting consisting of fluorescent or sealed-beam units with individual 8 hour minimum battery power supplies for areas that must be manned for safe shutdown and for access and egress routes to and from these areas. Access and egress routes to and from all other areas should be provided with similar lighting units with 2 hour battery power supplies.
3. The fixed aqueous film forming system (AFFF) installed to protect the mobiltherm system inside containment indicates that the system utilizes closed head sprinklers. NFPA 16 permits only open head systems for form-water sprinkler or spray systems. We recommend that the AFFF

system be revised to use open head sprinklers to comply with NFPA 16. To preclude a single spurious detector alarm from causing actuation of the foam system, a cross-zoned detector system using any combination of ionization, photoelectric, or heat actuated detectors with coincident circuitry would be consistent with NFPA 16 guidelines.

GAGE-BABCOCK & ASSOCIATES, INC.

CHICAGO OFFICE • 135 ADDISON AVENUE, ELMHURST, IL 60126 • (312) 530-1494

PAUL D. SMITH, P.E., *President*
BERT M. COHN, P.E., *Senior Vice President*

April 10, 1979
File 7917-2

U.S. Nuclear Regulatory Commission
Division of Systems Safety
Washington, D.C. 20555

Attn: Mr. V. Benaroya

Gentlemen:

Fast Flux Test Facility
Fire Protection Program Review

The Fire Protection Program review for the Fast Flux Test Facility, as stated in the Memorandum for W. P. Gammill, Assistant Director for Standardization and Advanced Reactors, DPM, from S. H. Hanover, Assistant Director for Plant Systems, DDS, dated April 5, 1979, has been reviewed and adequately reflects our concerns and recommendations. We concur in the NRC findings and the conclusions that upon implementation of the recommendations listed in the memorandum the fire protection program will be acceptable. Our review was based on the guidelines set forth in Appendix A to Branch Technical Position APCS 9.5-1 to the extent that Appendix A guidelines can be applied to FFTF, and on the conditions and assumptions set forth in the above referenced memorandum.

The review process of the plant's fire protection program was a joint effort of NRC and GBA staff personnel, and frequent contact was maintained during the review process. However, the GBA concurrence is based on an independent evaluation of documents submitted by Hanford Engineering Development Laboratory, of conditions noted during a site visit in December 1978, and of information supplied in response to requests generated during the evaluation.

Very truly yours,

Robert D. Barnes, P.E.
Sr. Fire Protection Engineer

APPENDIX J

CHRONOLOGY OF RADIOLOGICAL REVIEW
FAST FLUX TEST FACILITY

August 3, 1978	Safety Evaluation Report, NUREG-0358, issued
August 10, 1978	ACRS Subcommittee meeting
September 7, 1978	ACRS meeting
October 9, 1978	Submittal of Supplement 30 to FSAR, consisting of responses to questions concerning instrumentation and control
October 26, 1978	Letter to ACRS (T. Speis to R. Fraley) transmitting material relating to their concerns raised at their September 7, 1978 meeting
October 30, 1978	Letter from DOE (A. Rizzo to W. Gammill) transmitting information on seismic qualification of sodium leak detectors
November 2, 1978	ACRS meeting
November 8, 1978	ACRS report issued
November 16, 1978	Letter to NRC (A. Rizzo to W. Gammill) concerning sodium aerosol release from insulated piping.
November 24, 1978	Letter to NRC (A. Rizzo to W. Gammill) concerning ultrasonic inspection of FFTF piping.
November 26, 1978	Letter to NRC (A. Rizzo to W. Gammill) forwarding an analysis of FFTF response to postulated earthquake with primary piping leak and cooling by natural circulation.
December 13-14, 1978	Meeting at the site with DOE staff to discuss fire protection
December 15, 1978	Letter to DOE (W. Gammill to R. Ferguson) containing advice regarding the startup and operation of the FFTF

January 26, 1979

Letter to NRC (A. Rizzo to W. Gammill) advising of plans for addressing NRC and ACRS advice regarding startup and operation of the FFTF.

January 31, 1979

Letter to NRC (A. Rizzo to W. Gammill) concerning the cell liner test plan.

February 9, 1979

Submittal by DOE of updated natural circulation test specifications.