FINAL SAFETY ANALYSIS REPORT

FUEL STORAGE FACILITY

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INTRODUCTION AND GENERAL DESCRIPTION OF FUEL STORAGE FACILITY

INTRODUCTION AND GENERAL DESCRIPTION OF FUEL STORAGE FACILITY

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1.0 INTRODUCTION AND GENERAL DESCRIPTION OF THE FUEL STORAGE FACILITY

1.1 INTRODUCTION

1.1.1 Purpose of the Facility

The Fuel Storage Facility (FSF) is an integral part of the Fast Flux Test Facility. Its purpose is to provide long-term storage (20-year design life) for spent fuel core elements used to provide the fast flux environment in FFTF, and for test fuel pins, components and subassemblies that have been irradiated in the fast flux environment. The facility is shown in Figure 1.1-1.

Such a long term storage facility is necessary because there is only very limited storage capacity within the reactor vessel and in the Interim Decay Storage (IDS) Facility, and because all spent fuel used must be stored at this time since there is no fuel reprocessing available.

1.1.2 The Final Safety Analysis Report

This Final Safety Analysis Report (FSAR) and its supporting documentation provides a complete description and safety evaluation of the site, the plant design, operations, and potential accidents.

This FSAR generally follows the requirements of the LMFBR edition of Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, issued February 1974. However, since the FSF is not a reactor, and is unique in that it does not fit any of the categories for which standard SAR formats are available, the paragraphs (in the LMFBR edition) that do not apply to the FSF have been omitted. This provides for a more concise document.

FIGURE 1.1-1. Fuel Storage Facility.

1.2 SUPPORTING DOCUMENTATION

Lists of references are contained within the individual chapters of this report.

1.2.1 Requirements, Specifications, and Limits

The FSF is designed, constructed, and operated to the requirements, specifications, and limits applicable from the ERDA Manual, Code of Federal Regulations (10 CFR), NRC Regulatory Guides, and appropriate standards and codes such as those of ASME.

1.2.2 System Design Description (SDD)

The Fuel Storage Facility System Design Description (SDD)⁽¹⁾ contains all of the design bases and detailed descriptive information on the FSF building and components. Material in the SDD is included in the FSAR by both reference and quotation.

1.2.3 Final Safety Analysis Report for the Fast Flux Test Facility

Since the FSF is an integral part of the FFTF and is operated by the FFTF staff, many parts of the FSF FSAR are common to the FFTF FSAR.⁽²⁾ Significant material in the FFTF FSAR has been included by reference and by quotation.

1.2.4 Supplementary Documentation

Key documents supporting the conclusions in this FSAR are referenced to provide details of studies and analyses that form the basis of the conclusions.

1.3 GENERAL FSF DESCRIPTION

The FSF, as constructed and as it will be operated and maintained, provides a highly assured capability for reliable and predictable operation and an inherent capacity to prevent the occurrence of accidents.

The FSF is constructed so that, in the event of off-normal conditions, the design margins and engineered safety features will arrest the event and limit its consequences to well-defined and acceptable levels.

1.3.1 Summary of Site Characteristics

The FSF is on the DOE-owned Hanford Reservation in Benton County, Washington. The site is 4-1/2 miles (7.2 km) inside the closed reservation boundary, and 6 miles (8 km) from the city limits of Richland, Washington. The Richland city limits extend well beyond the present city development, and the facility is about 9 miles (14 km) from concentrated residential areas.

The climate is semi-arid, with mild winters and hot summers. The region, known as the Columbia Basin, experiences winds of a local character much of the time with occasional dust and sandstorms.

The site is well above the level of any potential flood. Tornado occurrence in the Columbia Basin is extremely rare; for the entire State of Washington the frequency of tornadoes is less than one per year. The site lies in a zone where the potential for earthquake damage is classified as minor to major. Refer to Chapter 2 for details of the site characteristics.

1.3.2 Summary Description of the Facility

The FSF is adjacent to the FFTF at the 400 area on the Hanford reservation. The FSF is in a building of standard industrial above-grade construction and reinforced concrete below-grade construction.

A large sodium-filled vessel (below grade) is provided to store up to 466 fuel assemblies and canisters containing fuel pins. The stored material is supported in the vessel by a rotatable turntable storage rack.

Decay heat is removed from the stored materials by natural circulation of the vessel sodium. This sodium transfers heat to two separate NaK heat transfer loops. The NaK loops dump the heat to the atmosphere through a natural draft heat exchanger (NDHX) in each loop.

All vaults with sodium or NaK containing piping and components are provided with catch basins with perforated fire-suppression covers.

Vault atmosphere and concrete temperatures are maintained well below design limits by a vault cooling system. An electric blower forces vault air over a cooling coil containing a chilled fluid.

Radioactive contamination carried into the storage vessel by the fuel assemblies or generated by accidental damage to the stored fuel assemblies is confined. The cover gas from the storage vessel is contained in a holdup tank and is vented to the atmosphere via the NDHX stack only after it is verified that such release is within safe and acceptable limits.

The FSF has been designed to be compatible with FFTF fuel-handling equipment for the insertion and removal of fuel assemblies. The FSF uses the Bottom Loading Transfer Cask (BLTC) and the FFTF floor valve for these operations.

All major components of the FSF are designed for a 20-year life.

1.4 IDENTIFICATION OF AGENTS AND CONTRACTORS

This section identifies those principal government organizations and contractors which bear the responsibility for design, construction, and operation of the FSF.

Department of Energy (DOE) -- DOE, as the owner of the Fuel Storage Facility, has the responsibility for approval of standards and acceptance of the work performed under this project. The management responsibilities for overall consideration, contract administration, and approvals are delegated to the Fast Flux Test Facility Project Office (FFTF-PO), located at Richland, Washington.

Rockwell International, Atomics International Division -- Atomics International (AI) has the responsibility for the design of the sodium and supporting systems for the facility, the preparation of the Systems Design Description, and the Preliminary Safety Analysis Report. In addition, AI is responsible for the procurement of sodium components and other long-lead items to be supplied to the installation contractor during the construction phase. AI will also provide Title III engineering services for the FSF sodium and process systems installation.

Norman Engineering Company (NECO) -- NECO has the responsibility for the design of the building site, structures, and utilities. NECO will also provide Title III engineering services for the FSF building construction.

J. A. Jones Construction Company (JAJ) -- JAJ, as the construction manager, will be responsible for the management of the building construction and installation contracts. JAJ or its agents will be responsible for the procurement of conventional items used in the construction.

Westinghouse Hanford Company (WHC) -- WHC has the responsibility for acceptance testing and operation of the facility. Additionally, HEDL, operated by Westinghouse Hanford Company, has prepared this Final Safety Analysis Report.

Vitro Engineering -- Vitro Engineering will provide for all quality assurance functions for the building and process systems during FSF construction and provide final "as-built" drawings.

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In addition to major project participants listed above, the following consultants and service organization are listed:

Consultants -- Woodward and Clyde -- Performance of Seismic Analysis Studies

Service Organizations -- Rockwell Hanford Operations -- Services include general warehousing, security, transportation, radioactive waste disposal, utilities, fire protection, and grounds maintenance.

1.5 RESEARCH AND DEVELOPMENT PROGRAMS

The FSF program does not involve any research or development activities carried out specifically for the FSF. The FSF design effort has, however, drawn extensively from state-of-the-art R&D programs such as those conducted for FFTF and CRBRP. In particular, several of the liquid metal system components will be selected from among FFTF or CRBRP seismic-qualified components to avoid an extensive testing effort within the FSF program.

1.6 REFERENCES

1. Fuel Storage Facility System Design Description, N106SDD210001, Rev. A, Atomics International, Canoga Park, CA, April 1978.
2. Fast Flux Test Facility Final Safety Analysis Report, HEDL-TI-75001, Hanford Engineering Development Laboratory, Richland, WA, December 1975.

CHAPTER 2
SITE CHARACTERISTICS

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2.0 SITE CHARACTERISTICS

2.1 GEOGRAPHY AND DEMOGRAPHY

The Fuel Storage Facility (FSF) site is consistent with the siting requirements of 10 CFR 100 (Reference 1). The description which follows demonstrates how the site location and characteristics satisfy the exclusion requirements (isolation and controlled access) and the physical criteria in terms of geography and demography, meteorology, hydrology, geology, and seismology.

2.1.1 Site Location

The FSF is located north of the city of Richland, in Benton County, in the south-central portion of the state of Washington. The site specific geographic location is $46^{\circ} 26$ min, 8 s north latitude and $119^{\circ} 21$ min, 31 s west longitude. This is within the confines of the DOE-owned Hanford Reservation described by Figure 2.1-1. The FSF is about 4.5 miles (7.1 km) from the nearest point on the reservation boundary on the west bank of the Columbia River.

The reservation areas to the northeast and north of the Columbia River are leased to and under the control of the U.S. Bureau of Sport Fisheries and Wildlife and the Washington State Game and Fisheries Departments. Sections of this area have been designated as a U.S. game refuge while other sections are open to the public for controlled fishing and hunting. In addition, a 77,000-acre (311-km^2) section in the southwest portion of the reservation has been set aside as an arid lands ecology reserve. An illustration of the reservation boundaries, major geographical features, and DOE facilities in relation to the surrounding area is provided by Figure 2.1-2.

Public access to the reservation south and west of the Columbia River is restricted except for a narrow corridor through which Washington State Highway 240 passes. Roads leading directly into the reservation proper are posted or barricaded. The entire 599-mi^2 (1448-km^2) reservation could be

FIGURE 2.1-1. Location of Hanford Reservation.

FIGURE 2.1-2. Topography Within 50 Miles of FSF

considered a controlled area, within the intent of ERDAM 0524 (Reference 2), since the DOE controls access and all activities therein. Plant security forces monitor access to specified areas and regularly patrol within the reservation boundaries.

2.1.2 Site Description

The FSF is located within the 400 Area of the Hanford Reservation, and is adjacent to the FFTF which it serves. The physical location of the FSF site on the reservation is shown on Figure 2.1-3, including features within the immediate vicinity and within a 10-mile (16-km) radius of the FSF. An area plan of the FSF is shown on Figure 2.1-4. The fence around the FFTF complex provides a minimum 400-ft (122-m) perimeter around the FSF. Access within the perimeter is controlled. The reservation boundaries nearest to the FSF site are 1) the Columbia River 4.5 miles (7.2 km) to the east, and 2) the Richland City limits about 6 miles (10 km) to the south.

As determined by 10 CFR 100, the exclusion radius, low population zone, and population center distance all fall within the reservation. In determining potential exposure to the public, the nearest point of approach to the reservation is the boundary 4.5 miles (7.2 km) to the east. Washington Public Power Supply (WPPSS) Plant No. 2, approximately 3 miles to the north-east, is the nearest facility to the 400 Area on the Hanford Reservation. However, in anticipation of the future development of Hanford as a nuclear center, a 1.5-mile (2.4-km) control radius is used in evaluating radiation exposures to personnel at other Hanford facilities.

2.1.2.1 Exclusion Area Control

At this time there are no activities unrelated to plant operations being conducted within the 1.5-mile (2.4-km) established control radius which require evacuation plans for personnel based on emergency situations at the 400 Area. The control area is traversed by one highway, RT-4, approximately 1 mile (1.6 km) from the FSF site. This highway is owned by the DOE and is

utilized for access and road shipments to and from the Hanford 100, 100A, and 200 Areas and the WPPSS construction site. Traffic control in the event of an emergency will not present a major problem or hazards to the general public.

2.1.2.2 Boundaries for Establishing Effluent Release Limits

The FSF is located on the reservation, which is a restricted area. Within the reservation, the FFTF is fenced around the perimeter at a minimum distance of 500 ft (150 m) from the containment vessel (Figures 2.1-3 and 2.1-4). The FSF is within this fence at a minimum distance of 400 ft (120 m). Access to the reservation is controlled and access to the FSF site is further controlled within the reservation boundaries.

The FSF is designed based on the philosophy of as low as reasonably achievable (ALARA) radioactivity release. Accidental release of radioactivity is evaluated at both the 1.5-mile (2.4-km) control distance and at the closest reservation boundary, 4.5 miles (7.2 km).

2.1.3 Population and Population Distribution

Population in the area surrounding the FSF site and the reservation is sparse. The local population is generally well informed concerning nuclear power applications and, for the most part, welcomes expansion of the area's nuclear industry. The proposed nuclear park to be developed on the reservation will include the construction of other power reactors.

The impact of known development has been considered in the population projection provided in the following sections.

At this time, two additional facilities related to the FFTF operation are being proposed for construction at the 400 Area. These include the following:

- 1) Fuel and Materials Examination Facility (FMEF) -- to provide the capability under controlled environmental conditions for completely remote and safe handling and examination of irradiated fast reactor fuels and materials and to provide an integrated pilot-scale plutonium fuels facility with supporting laboratories and offices to develop improved and advanced Liquid Metal Fast Breeder Reactor (LMFBR) fuels manufacturing processes, equipment, and handling systems.
- 2) Maintenance and Storage Facility (MASF) -- to provide maintenance and operations support equipment and capabilities for FFTF component repairs.

The projected schedule for completion of these facilities is 1980 through 1983, and planned operations would increase the working population at the site by approximately 350. It is expected that the operation of these facilities will not affect the established controlled access or the control radius of 1.5 miles (2.4 km).

It is difficult to predict the number of power units and facilities that might be constructed over the next 20-year period; however, present site requirements and other criteria will probably limit the working population within 5 miles (8 km) of the 400 Area to approximately 4,000 for the next decade.

2.1.3.1 Population Within Ten Miles and Between Ten and Fifty Miles

Within 10 miles (16 km) of the FFTF site, about 14,550 residents of the Tri-Cities area are employed at nearby DOE operations, attend schools and work in private industry. A population of about 189,000 (compared to 185,000 in 1979) now resides within a 50-mile (80 km) radius of the FSF site. The Tri-Cities urban population increase accounts for approximately 3,700 of the 4,000 difference. Approximately 525,000 people live within a 100-mile (160 km) radius of the facility.

The 1970 geographic distribution of population is shown in Figure 2.1-5. The same data in terms of cumulative resident population outward from FSF is shown in Figure 2.1-6.

2.1.3.2 Low Population Zone

This section is not applicable to FSF.

2.1.3.3 Transient Population

Daily and seasonal variations in populations within the boundary of the reservation are insignificant. Variations in the working population distribution during regular day and night shift operations are provided in Table 2.1-1.

2.1.3.4 Population Center

Resident population to the north, west, and east outside the reservation consists primarily of farms and farming communities. The Tri-Cities incorporated areas (Richland, Kennewick, and Pasco), located to the south and southeast, represent the major population concentrations in the area. The nearest residential area in Richland begins about 8 miles (13 km) south-southeast of the FSF site. The center of Richland is approximately 12 miles (19 km) from the site.

2.1.3.5 Public Facilities and Institutions

Facilities and institutions located within 10 miles (16 km) of the FSF site are identified and located with respect to the FSF site on Figure 2.1-3. Populations of each are provided in Table 2.1-1.

2.1.4 Uses of Adjacent Lands and Waters

The area around the reservation is primarily agricultural. The farms in this area are irrigated with water supplied by the Bureau of Reclamation,

FIGURE 2.1-5. Population Distribution About FSF (1970 Census).

FIGURE 2.1-6. Cumulative Population Distribution About FSF (1970 Census).

TABLE 2.1-1
WORKING AND SCHOOL POPULATIONS

ACTIVITY/AREA (JULY 1974)	DAY	NIGHT
Hanford (ERDA) and Contractors		
100	816	169
200	1618	138
300	2438	101
400 (FFTF Site)	295	46
700 ^a	(1219)	(54)
1100	328	9
3000	1122	22
Private Industry Within 10 Mi. (March 1975)		
Battelle Northwest Laboratory (BNW)	1766	15
Bovee and Crail Construction	644	34
Exxon Nuclear Industries, Inc.	477	35
Holosonics Inc.	60	0
J. A. Jones	841	0
NORTEC Corp.	25	1
Olympic Eng.	51	0
Pride Pak Foods	130	140
Richland Airport	17	1
SIGMA Research	12	0
United Nuclear Industries	104	1
United States Testing Co., Inc.	66	1
Western Sintering	21	4
WPPSS - Engineering Office	395	0
- Engineering Office (December 1975)	585	0
- Site (During peak construction 1976) ^b	1500	0
Schools Within 10 Miles (March 1975)		
Edwin Markham	195	
Hanford (K-12)	1772	
Sacajawea	430	
Joint Center for Graduate Study		
Continuing Education	125	350
Academic	30	150

^a Outside the Reservation and more than 10 miles south of FFTF

^b If WPPSS WNP-1 construction is on anticipated schedule

Columbia River Basin Project. Approximately 5,000 acres (20 km^2) of farmlands to the west of Pasco are irrigated by water pumped from the Columbia River between Richland and Pasco. In addition, dry-land grain production is prevalent throughout the area. Planned expansion of irrigation projects on the northwest and south will allow the cultivation of these arid lands. Acreages and yields for principal crops are shown in Table 2.1-2.

The nearest dairy farms are located about 8 miles (13 km) to the southwest of the FSF site and approximately 4 miles (7 km) east of the Columbia River. These are identified on Figure 2.1-3.

Aside from their use for irrigation, the rivers of the area are used for hydroelectric power, shipping, and recreation. The Columbia River is closed to boating near the old Hanford townsite and commercial barge traffic has no designation further than the Port of Benton slip (see Figure 2.1-3) north of Richland. Information regarding surface and groundwater uses are covered by Section 2.4 of this chapter.

The extent of commercial and sport fishing within the Hanford area, including species and yields taken and relative abundance, is defined in Table 2.1-3.

2.1.5 References for Section 2.1

1. Code of Federal Regulations, 10 CFR 100, "Reactor Site Criteria," USGPO, Washington, DC.
2. ERDA Manual, "Standards for Radiation Protection, Chap. 0524, US Department of Energy, Washington, DC, March 1977.
3. 1970 Census Population, US Department of Commerce/Bureau of the Census, Washington, DC.
4. J. R. Walker, State of Washington, Population Trends, 1972, Information Systems Division, Office of Program Planning and Fiscal Management, Olympia, WA, 1972.
5. State of Washington, Population Trends, 1973, Information Systems Division, Office of Program Planning and Fiscal Management, Olympia, WA, 1973.

TABLE 2.1-2
PRINCIPAL CROPS WITHIN 80 km OF FSF

CROP	ACREAGE	YIELD (PER ACRE)
Wheat	800,000	41 Bu
Alfalfa	250,000	5 Ton
Barley	110,000	50 Bu
Potatoes	75,000	200 Cwt
Sugar Beets	80,000	25 Ton
Corn	40,000	110 Bu
Mint	35,000	80 Lb
Grapes	20,000	9 Ton
Apples	15,000	9.5 Ton
Asparagus	12,000	1.5 Ton

TABLE 2.1-3
FISHERY IN HANFORD AREA^a

SPECIES	ABUNDANCE	CATCH	EFFORT
Steelhead Trout	30,000	4,300	20,000 Angler Days
Whitefish, Small Mouth Bass, Bullhead Catfish, Crappie	Abundant		
Large Mouth Bass, Sunfish, Perch, Shad	Moderate	280,000	140,000 Angler Days
Rainbow Trout, Channel Catfish, Walleye Pike	Few		
Salmon	180,000	300 1,500 ^b	Not Estimated
Sturgeon	Abundant	300	Not Estimated

^a Annual averages based on 1970, 1971, 1972 Washington State Fisheries data.

^b Yakima Indian Fishery on Yakima River.

6. Population and Household Trends in Washington, Oregon, and Northern Idaho, 1970-1985, Business Research Division, Pacific Northwest Bell Telephone Company, January 1972.
7. Washington Population, Employment, and Housing Units Projected to 1990, Bonneville Power Administration, U.S. Department of the Interior, February 1973.

2.2 NEARBY INDUSTRIAL, TRANSPORTATION, MILITARY FACILITIES

Potential accidents as a result of extraneous activities in the vicinity of the FSF have been studied to determine what effects, if any, could jeopardize the safety of the facility. This section identifies those activities, such as industrial, transportation, and military, in the vicinity of the plant and addresses their potential effects on the plant's continued operation and safety.

2.2.1 Location

Figure 2.1-3 shows nuclear and industrial facilities, laboratories, airports, waterways, railways, highways, and public facilities located within a 5- and 10-mile (8- and 16-km) radius of the FSF site. Of those, only the WPPSS reactor site is within the vicinity of the FSF. Its location is approximately 2-3/4 miles (4.4 km) northwest. The only military installation is the Yakima Firing Range located 45 miles (72 km) to the west-northwest of the FSF site.

2.2.2 Description

The major facilities (industrial and laboratory) shown on Figure 2.1-3 within the boundary of the Hanford reservation (excluding the WPPSS reactors) are generally assigned to DOE-funded work. Potential hazards from most of these facilities are radiological and are assessed below in Section 2.2.3. The three WPPSS reactors, however, will store industrial chemicals* for

*Sodium hypochlorite, sodium hydroxide, sulfuric acid, hydrazine, ammonia, and chlorine

water treatment and chemistry control. Of these, the only chemical that presents a potential hazard to FSF is chlorine. Chlorine is intended for use in all three reactors and probably will be stored in 10 DOT 1-ton containers at each site. Usage rate is projected to be 4.2 tons (3.8 tonnes) per week for the three-reactor complex. An evaluation of the potential hazards of chlorine to FSF is given in Section 2.2.3. The major industrial facilities located outside of the principal Hanford Reservation (south of the 300 Area) within 10 miles (16 km) of FFTF are shown on Figure 2.1-3 and are listed below by function:

<u>Name of Firm</u>	<u>Activity</u>
Pacific Northwest Laboratory	Research, Consulting
NORTEC Corporation	Nondestructive Testing Equipment
EXXON Nuclear	Fuel Fabrication
Washington Public Power Supply System (WPPSS)	Power Co. Offices, Engineering
U.S. Testing	Industrial and Agricultural Testing Service
Western Sintering	Metal Fabricators
Bovee and Crail Const. Company	Industrial Contractor
J. A. Jones Company*	Construction and Industrial Services
Rockwell Transportation* and Central Stores	General Services for the Hanford Reservation (DOE-funded)
Pride Pak	Food Processing
SIGMA Research	Contract Research and Development
Olympic Engineering	General Engineering Consultants

Most of the industries listed above are engaged in activities which do not present hazards to FSF. Radioactive materials are processed at the EXXON facility, and a potential accident assessment is given in Section 2.2.3.

*On DOE property

2.2.2.1 Waterways

Barges of 3,000-ton (2.7-tonne) capacity can be accommodated on the Columbia River. The manufacturers of large equipment can use barge transportation for delivery. Components must be unloaded at the Port of Benton barge slip, which is located approximately 7-1/2 miles (12 km) from the FSF site. From this point, components will be loaded on DOE-owned or contracted transportation vehicles for delivery to the site. Hazardous materials (i.e., chemicals, explosives, munitions, or liquid and gaseous fuels) have not been transported by waterways within the Hanford Reservation and approval criteria must be met before such shipments can be made.

2.2.2.2 Airports

The Richland airport, under the direction of the Port of Benton, is located approximately 8-1/2 miles to the south-southeast of the FSF facility. Figure 2.2-1 shows a master plan of its existing facilities and proposed potential expansion program which at this time has not received favorable response from the Richland city government.

The dominant role of the Richland airport is classified as General Utility. Activities include recreation flying, pilot training, charter flights and air taxi, some business flying, soaring and sail plane operations, and parachute jumping and skydiving. The present level of aircraft operations is 69,300 (takeoffs and landings) per year. A total of 65 aircraft are based at the airport. Storage tanks buried at the airport include 16,000 gallons (6×10^4 liters) for 80-octane fuel and 18,000 gallons (6.8×10^4 liters) for 100-octane fuel.

Typical day movements expected at the Richland airport in 1979 are 170 takeoffs and landings. The type aircraft normal to airport operations is less than 12,500 pounds. Based on its present and expected level of operation,

PORT OF BENTON

2952 GEORGE WASHINGTON WAY
P O BOX 300
RICHLAND, WASHINGTON 99352
TELEPHONE (509) 943 9115

FIGURE 2.2-1. Richland Airport.

the Richland airport is considered not to have potential consequences serious enough to affect the safety of the FSF plant.

The Tri-Cities Airport at Pasco, Washington, is the main airport in the area for commercial air carriers. The Tri-Cities Airport is located 17 miles to the southeast of the FSF facility. Daily movements at the Tri-Cities Airport average 306 per day.

2.2.2.3 Highways

The DOE-owned road system serves the areas of the reservation with approximately 270 miles (435 km) of paved two-lane and four-lane primary roads. State and national highway standards prevail for designing and maintaining DOE roads. Figure 2.1-3 shows the primary arteries, both public and DOE-owned, which traverse and/or bypass the reservation.

Road shipments to and from the Hanford 100, 100N, and 200 Areas, the FFTF, and WPPSS construction site often utilize Route 4 (DOE-owned) which passes about one (1) mile (1.6 km) from the FSF. The type of materials normally shipped via this route are reagents and chemicals from the Reservation General Supplies. Commercial shipments of these types of materials are delivered to the Purchasing and Stores Department close to Richland (13 on Figure 2.1-3). Smaller quantities (pounds and gallons) are then transported to the various sites by Rockwell (under contract to DOE) on a normal use or as-needed basis. Gasoline and diesel fuel 2 are delivered in 4,000-gallon (1.5×10^4 liter) and 6,000-gallon (12.2×10^4 liter) tank trucks.

2.2.2.4 Railways

The DOE-owned railroad system has a capability of moving approximately 12,000 cars per year over 150 miles (240 km) of reservation track. The system includes five main lines and two classification yards and is connected to the Burlington Northern and Union Pacific Lines in the south and to the Chicago, Milwaukee, St. Paul, and Pacific Line in the north. Delivery is

usually made to the reservation central stores area in the south with final delivery to the sites by DOE-owned railroad equipment.

Table 2.2-1 identifies the hazardous material types and amounts and the frequency of shipments transported currently by rail within 2-1/2 miles (4 km) of the FSF site. Increases in the frequency of shipments of chlorine can be expected when the WPPSS reactors come on-line. As anticipated, however, the shipments will be in one-ton containers as are presently being transported. The potential hazards of these chlorine shipments are addressed in Section 2.2.3.

2.2.2.5 Projections of Industrial Growth

Within the scope of proposed development for the reservation, there are no known unrelated facilities or industries (with the exception of WPPSS) planned in the vicinity of FSF site.

2.2.3 Evaluation of Potential Accidents

In general, rail and truck shipments or stored industrial materials do not pose a significant hazard to the FSF. The shipments are either infrequent, in small quantities, or insufficiently hazardous to present a potential problem. In addition, if it should be necessary to evacuate the facility, leaving it unattended, this is an acceptable personnel protection method because 1) alarm signals for parameters associated with safe operations of FSF are given in the continuously manned FFTF control room, and 2) the change in the principal safety parameters is very slow, in most cases allowing days to take appropriate action as discussed in subsequent chapters. The FSF is normally unmanned during operation.

Richland Airport - Probabilities of aircraft crashes near air fields are given below (see Table 2.2-2). Based on a value of 0.14×10^{-8} probability per square mile per aircraft movement at 8.5 miles and a roof target

TABLE 2.2-1

RAIL SHIPMENTS OF HAZARDOUS MATERIALS VIA DOE-OWNED
FACILITIES ON THE HANFORD RESERVATION: JAN. 1973-JUNE 1975

MATERIAL	TOTAL CAR LOADS (30 MOS.)	TOTAL TONNAGE (30 MOS.)	AVG. PER CAR LOAD (TONS)	MAX. SHIPMENT (TONS)	AVG. FREQ. (Car Loads/Yr)
Chlorine	8	120	15	15 (In One Car Load)	3.2
Caustic Soda	101	6381	63.2	316 (In Five Car Loads)	26
Carbon Tetrachloride	3	69	23	23 (In One Car Load)	1.2
Aluminum Sulfate	4	224	56	62 (In One Car Load)	1.6
Sulfuric Acid	19	1444	76	150 (In Two Car Loads)	7.6
Sodium Nitrate	2	71	35.5	51 (In One Car Load)	0.8
Scrap Radioactive Waste (Low Level)	43	2164	50.3	545 (In Nine Car Loads)	24

TABLE 2.2-2
PROBABILITY OF AN AIRCRAFT CRASH
(WASH-1400)

<u>Distance from End of Runway (mile)</u>	<u>Probability ($\times 10^{-8}$) of a Fatal Crash Per Square Mile Per Aircraft Movement</u>
0 - 1	16.7
1 - 2	4.0
2 - 3	0.96
3 - 4	0.68
4 - 5	0.27
5 - 6	0.0*
6 - 7	0.0
7 - 8	0.0
8 - 9	0.14
9 - 10	0.12

*No crashes occurred at these distances within a 60° flight path.

area of 10^{-3} square miles, the probability of an air crash into the roof of the FSF is estimated to be 1×10^{-8} . Therefore, this accident is deemed to be noncredible and is not analyzed further.

Pasco Airport - The Tri-Cities Airport at Pasco, Washington, is the main airport in the area for commercial air carriers. Operations at the Tri-Cities Airport are considered to have no impact on the safety of the FSF. However, a small fraction of the aircraft traffic entering and leaving the Tri-Cities Airport is controlled by the FAA into air corridors adjacent to the 400 Area. The probability that such traffic would result in an accident whose impact would potentially exceed the 10CFR100 guidelines is $1.9 \times 10^{-8} \text{ yr}^{-1}$ which is well within the limits of Regulatory Guide 1.70 and Section 3.5.1.6 of NUREG-75/087 ($1 \times 10^{-7} \text{ yr}^{-1}$).

Potential hazards to FSF from radiological accidents were assessed for the N reactor in the 100 N Area, the waste processing facilities in the 200 areas, the WPPSS reactors, and the EXXON fuel fabrication facility. The N reactor and the waste processing facility do not pose a hazard due to the distances involved. The design basis accidents (DBAs) for the WPPSS reactor were reviewed as a potential hazard at the FFTF site (Reference 2). The worst case found, a loss of coolant accident (LOCA) occurring at either WNP-1 or 4, hypothetically could yield significant thyroid doses in the vicinity of FSF. The consequences of this event are mitigated, however, by the fact that personnel could evacuate to the FFTF control room. As discussed in Paragraph 15.10.6.1, the FSF can remain unmanned for long periods of time. Postulated radiological accidents occurring at EXXON, as addressed in Reference 3, were also assessed as to potential hazard to FSF, and no problem was found to exist.

2.2.4 References for Section 2.2

1. Stevens, Thompson, and Runyan, Richland Airport Master Plan, the Port of Benton; Richland, WA, November 18, 1974.
2. Fast Flux Test Facility Final Safety Analysis Report, HEDL-TI-75001, Hanford Engineering Development Laboratory, Richland, WA, December 1975.
3. Final Environmental Statement Related to Operation of Uranium Oxide Fuel Plant, EXXON Nuclear Co., Docket 70-1257, U.S. Atomic Energy Commission, Directorate of Licensing, March 1974.

2.3 METEOROLOGY

The following section presents atmospheric data for the Hanford region. Accident analyses for the FSF site have used the meteorological assumptions in Regulatory Guides 1.3⁽¹⁾ and 1.4⁽²⁾ for diffusion estimates. Data presented in this section shows the general agreement and conservatism of these referenced assumptions for use in the accident analysis considered in the FSF design.

2.3.1 Regional Climatology

The climate of the Hanford reservation is typical of that found in inter-mountain areas at this northerly latitude on the western side of the continent. The Columbia River Plains region in which the reservation lies has the lowest elevation of any part of Central Washington. The reservation is protected by mountains: to the west, Cascades; north, Okanogan Highlands; east, Bitterroot and Salmon Hills; and south, Horse Heaven Hills and the Blue Mountains. The climatic characteristics of the reservation include a small amount of precipitation that falls in the colder half of the year, light snowfall, continental temperatures, winds of a local character much of the time, extremely low humidity in summer, high humidity in winter, and frontal passages primarily during the winter. More specific data on the region provided in the following sections are from Reference 3.

2.3.1.1 General Climate

The annual mean temperature (1912-1970) at Hanford is 53.2°F (11.5°C). The summertime average is 73.7°F (22.9°C); however, temperatures exceeding 100°F (37°C) may be expected approximately 14 days a year. Wintertime temperature average is 32.4°F (0°C), but temperatures below 0°F (-18°C) may be expected approximately 4 days per year. The maximum temperature ever observed at Hanford was 115°F (46°C) in July 1939, and the absolute minimum observed temperature was -27°F (-33°C) in December 1919.

The primary observation site, from which the mean values have been derived, is approximately 16 miles (26 km) northwest of the FSF site. The absolute minimum temperature cited above was recorded at a cooperative observer site approximately 10 miles (16 km) east-northeast of the present observing facility.

A statement regarding the interplay between synoptic scale processes and terrain characteristics of the area may best be made by a verbatim quote from Phillips:⁽⁴⁾

"The climate is influenced by topography, distance and direction from the ocean, the prevailing westerly winds, and the position and intensity of the high and low pressure centers over the Pacific. Elevation (terrain) varies from 400 ft (120 m) near the confluence of the Snake and the Columbia, 1,000 to 3,000 ft (305 to 914 m) over the agricultural areas between the Snake and the Blue Mountains, and 6,000 ft (1800 m) or more on the higher slopes of the mountains.

The Rocky Mountains shield the Inland Basin from cold arctic air masses moving southward. The Cascade Range forms a barrier to the easterly movement of moist and comparatively mild air in winter and cool air in summer. Most weather systems crossing the Inland Basin move with the prevailing westerly winds. In summer, air from over the continent results in low relative humidity and high temperature, while in winter it produces clear, cold, dry weather. Extremes in both winter and summer generally occur when the Inland Basin is under the influence of air from over the continent. Maritime air from the Pacific has a moderating influence throughout the year."

Precipitation statistics have been recorded at Hanford since 1946. These data show that the mean annual total precipitation is 6.25 in. (0.16 m). Three months, November through January, contribute 42% of this total. On only 29 days between 1956 and 1970 have more than 0.5 in. (0.013 m) of observed precipitation been recorded. The highest daily precipitation observed (Hanford Townsite and Hanford Meteorological Facility) since 1912 is 1.91 in. (0.044 m), which was in October 1957. The highest annual total precipitation occurred in 1950 when 11.45 in. (0.29 m) was recorded. The heaviest annual snowfall occurred in the 1955-1956 winter, when 41.6 in. (1.06 m) accumulated in Richland, and 38.5 in. (0.98 m) accumulated at the Hanford site. The greatest snowfall in the immediate area was recorded in November 1978, when a fall of 8.3 in. (0.21 m) was recorded in Richland.

The maximum snow depth recorded on the ground was 12.1 in. (0.31 m) in December 1964.

Hail is rare at Hanford. There is no record of more than two occurrences in any one year. When hail storms do occur, they are not of unusual intensity. The greatest hail stone size ever observed at Hanford was about 0.4 in. (0.01 m) in diameter.

Ice storms (freezing rain) occur on the average of 7 times each year. Most of these are of slight intensity. The two worst occurrences were February 11-12, 1954, and November 23-24, 1970. These storms resulted in traffic disruption but caused no known damage to transmission lines. Thunderstorms may occur in any month at Hanford; however, the thunderstorm season is essentially limited to the months of April to September. The area averages 11 occurrences per year. Precipitation rates associated with thunderstorms have never exceeded 1.65 in./hr (0.04 m/hr). A fall of 0.55 in. (0.14 m) was observed in a 20-minute period on June 12, 1969. Winds associated with thunderstorms average 30 to 40 mph (50 to 65 km/hr).

The entire state of Washington averages less than one tornado per year. Those that do occur are of less severity than those affecting the Great Plains and Gulf State areas. Only 14 tornadoes have been observed since 1916 within a 100-mile (160-km) radius of the FSF. None of these tornadoes resulted in major property damage or loss of life. A survey of tornadic events in the Pacific Northwest developed by Fujita⁽⁵⁾ identifies that within the immediate Hanford area only five small tornadoes have been recorded between 1950 and 1970 - none since 1970. Because of the relative infrequency of tornadoes in the Pacific Northwest (and especially in the Hanford area) and the lack of precise arithmetic tornado intensity measurements, standard statistical criteria cannot be applied to tornado prediction. Fujita's survey⁽⁵⁾ has however established two significant trends: (1) if Hanford area tornadoes do occur, they will be more infrequent and of lower intensity than those occurring elsewhere in the Pacific Northwest, and (2) the Hanford area is not within any of the charted Pacific Northwest "tornado alleys."

2.3.1.2 Severe Weather

Severe storm criteria are discussed in Reference 6. These criteria are as follows:

- Very warm moist air lower than 5,000 ft (1500 m)
- A wedge of very dry air, near 10,000 ft (3000 m), that could be superimposed over the lower warm moist layer
- Low level wind maxima (30 kts (15 m/s) or greater) in moist air
- Strong winds near 20,000 ft (10,000 m) with strong horizontal shear and branching jet stream
- Strong vertical shear between low level and mid-level wind flow
- Cooling of mid-level air mass
- Extreme instability

Futher definition⁽⁶⁾ of the criteria states:

"... severe storms, of the type that occur in 'Tornado Alley,' require that very specific atmospheric conditions be present simultaneously in time and space. The 'fuel' for this type of storm is very warm, moist buoyant air in the lower 5,000 ft (1500 m) of the atmosphere. This type of air mass has been termed 'continental tropical.' Additional requirements are: the availability of a very dry continental air mass upwind, a low level jet in the moist air, a strong band of winds near the area at or near 20,000 ft (6000 m) strong vertical shear, significant cooling aloft, and explosive instability. The first and last of these items are unknown in the Pacific Northwest. The other conditions do exist in the Pacific Northwest, but seldom coexist. We see no reasonable route over which an unmodified tropical air mass (warm, moist, buoyant) could be transported into the Eastern Washington-Oregon area. The mountainous features that are present to the east, south, and west of this region would deplete such an air mass of its moisture by orographic lift prior

to its arrival. If the moisture is not available then the fuel is also not present. Further, the unavailability of the tropical air, in the lower atmosphere, also negates the possibility of an explosive instability."

and

"In response to the added interest in design criteria for nuclear facilities and the interface with the wind and pressure forces of a potential tornado, the calculations and conclusions in the FFTF FSAR have been reviewed. The tornadic probabilities as set forth in the FFTF FSAR are very conservative as these calculations did not take into account the diminished tornado path area typical of the Pacific Northwest storms. Further study of the physical characteristics to be expected in a Pacific Northwest storm shows that it is reasonable to expect that the severe midwest-type storm would not occur in the Pacific Northwest. The physical features of the local storms are characteristic of cold core systems, short path lengths and small vortex radii. Storms of this category have been calculated to have maximum tangential wind speeds in the 110 mph (180 km/hr) range."

In conclusion, the formation of a severe tornado of the midwest type is highly unlikely within the climatological and orographic characteristics of the Pacific Northwest. It has been determined that the possibility of occurrence of any tornado is low, on the order of 10^{-6} per year.⁽⁶⁾

Even though the tornado probability is low, and the force of any tornado, compared to the midwest type, is relatively low, a margin of safety against tornadoes is provided. While a fully verified tornado has never been recorded on the Hanford reservation (but funnel clouds have been observed), the possibility of a tornado touchdown cannot be ruled out. As previously cited,⁽⁶⁾

"Storms of this category have been calculated to have maximum tangential wind speeds in the 110 mph (180 km/hr) range."

Winds of this magnitude could not cause sufficient damage to the FSF to result in a release of radioactivity. However, to insure that structures and equipment will maintain the plant in a safe shutdown condition, the criteria shown below apply.

2.3.1.2.1 Criteria for Tornado Design

Due to the low probabilities of tornadoes in the vicinity of the FSF and their expected lesser intensity in comparison to those recorded in the Pacific Northwest "tornado alleys," the below conservative design criteria are applied. Refer to technical justification.⁽⁶⁾

- 1) Wind Speed: 175 mph (280 km/hr) horizontal wind, combined tangential and translational components, applied over the full height of the structures.
- 2) Pressure Transient: 0.75 lb/in.^2 (5200 Pa) reduction in atmospheric pressure in 3 s, followed by return to original pressure at the same rate.
- 3) Missiles:
 - a) 2 in. x 12 in. (.05 m x .30 m) plank 12 ft (3.7 m) long, weighing 54 lb (245 kg), traveling end-on at 100 mph (160 km/hr), at any height.
 - b) 4 ft x 8 ft (1.2 m x 2.4 m) plywood sheet, 3/4 in. (0.2 m) thick, traveling end-on at 150 mph (240 km/hr), at any height.
 - c) 26 in. and 20 ft (.66 m x 6.1 m) sheet of No. 20 corrugated steel siding, traveling end-on at 150 mph (240 km/hr), at any height.

Other strong winds do occur in the Hanford area. These linear wind phenomena usually occur in winter and spring. The highest wind speed ever recorded at

the 50-ft (15-m) level of the primary observing facility was a gust to 80 mph (130 km/hr) that occurred in January 1972; during this period the average wind speeds were recorded at 51 mph (82 km/hr). Since April 1957 the Hanford Design Standards for all facilities has been an 82 mph (131 km/hr) wind for heights from 0-50 ft (0-15 m).

2.3.1.3 Pollution Potential

Holzworth has shown that this area lies on the northern edge of a high pollution potential area.⁽⁷⁾ His analysis shows that for the period of August 1963 to April 1970, a total number of forecast days of high pollution potential for this area would be about 30.

2.3.2 Local Meteorology

2.3.2.1 Local and Extreme Values of Meteorological Parameters

Hanford meteorological summaries, of many parameters, are available in Climatology of the Hanford Area⁽³⁾ by W. A. Stone, et al. Some of these are presented in Tables 2.3-1 through 2.3-10 and Figure 2.3-1. Table 2.3-11 presents the annual occurrence of atmospheric stability classifications from 50 to 250 ft (15 to 75 m), converted to the usual format °C/100 m.

2.3.2.2 Potential Influence of the Plant and Its Facilities on Local Meteorology

It is not expected that operation of the FSF will have any influence on local meteorology. The only associated facilities that have any potential for modification of the immediate environs are the heat exchangers. The thermal influence of these facilities will be dissipated within 1.5 miles (2.4 km), if average winds prevail. The local topography is presented by the map of Figure 2.1-2. It is not expected that the terrain would have an effect on

HEDL 7404-89.3

FIGURE 2.3-1. Average Monthly Precipitation Amounts (1912-1970).

TABLE 2.3-1

MONTHLY EXTREMES OF DAILY MAXIMUM AND MINIMUM TEMPERATURES AND DAILY TEMPERATURE RANGES - HANFORD TOWNSITE (NEAR) AND HMS* BASED ON PERIOD OF RECORD: 1912-1970

MONTH	DAILY MAXIMUM TEMPERATURE				DAILY MINIMUM TEMPERATURE				DAILY TEMPERATURE RANGE ^a			
	RECORD HIGHEST		RECORD LOWEST		RECORD HIGHEST		RECORD LOWEST		GREATEST		LEAST	
	°F	DATE	°F	DATE	°F	DATE	°F	DATE	°F	DATE	°F	DATE
JAN	66	20, 1968	-2	31, 1950	51	1, 1924	-23	17, 1916 ^b	42	25, 1962	1	13, 1952 ^b
FEB	71	12, 1924	-3	1, 1950	55	26, 1932	-23	3, 1950 ^b	42	1, 1947	2	9, 1954 ^b
MAR	83	25, 1960	24	3, 1960	54	9, 1942	6	5, 1955	41	30, 1945 ^b	2	6, 1957
APR	95	22, 1934 ^b	41	7, 1945	60	21, 1956	12	1, 1935	47	16, 1954 ^b	8	7, 1953 ^b
MAY	103	26, 1936 ^b	49	8, 1918	70	19, 1956	28	1, 1954	48	11, 1946	6	24, 1962
JUNE	110	25, 1912	56	18, 1938 ^b	81	30, 1924	33	9, 1933	45	25, 1970	8	28, 1946 ^b
JULY	115	27, 1939	59	2, 1966	82	30, 1925	41	5, 1935	44	9, 1952	7	2, 1966
AUG	113	4, 1961	63	25, 1920	81	4, 1961	40	18, 1918	43	31, 1955 ^b	10	20, 1959 ^b
SEPT	102	2, 1950	52	23, 1934 ^b	72	7, 1955	25	23, 1926	44	6, 1962	10	27, 1955 ^b
OCT	90	6, 1933	31	30, 1935	60	27, 1937 ^b	6	31, 1935	43	11, 1946 ^b	3	20, 1947
NOV	73	5, 1934	14	13, 1955 ^b	52	24, 1927 ^b	-1	14, 1955	36	24, 1970	1	26, 1952
DEC	68	22, 1933	-3	13, 1919 ^b	53	20, 1933	-27	12, 1919	37	3, 1956	1	27, 1947 ^b
ANNUAL	115	JULY 27, 1939	-3	FEB 1, 1950 ^b	82	JULY 30, 1925	-27	DEC 12, 1919	48	MAY 11, 1946	1	NUMEROUS DATES

^a Statistics tabulated for the daily temperature range are based only on the period of record 1945-1970.

^b Denotes also on earlier dates.

*Hanford Meteorological Station

TABLE 2.3-2

COMPOSITE RECORD OF MONTHLY AND ANNUAL MEAN TEMPERATURES (F)
HANFORD TOWNSITE (NEAR) AND HMS: 1912-1970

YEAR	JAN	FEB	MAR	APR	MAY	JUNE	JULY	AUG	SEPT	OCT	NOV	DEC	ANNUAL
1912	—	—	—	—	62.6	71.1	74.8	70.5	62.3	49.1	42.6	35.2	—
1913	25.8	28.2	43.2	54.1	61.4	70.6	76.0	73.7	62.2	49.5	43.0	31.2	51.6
1914	37.4	35.5	46.8	53.7	62.9	65.7	75.2	73.8	60.2	53.4	41.0	22.6	52.4
1915	25.2	38.2	48.2	55.4	60.2	67.2	72.8	77.5	62.4	54.0	37.8	32.4	52.6
1916	16.1 ^a	32.4 ^a	46.9	53.0	56.8	—	—	73.6	63.8	50.8	35.5	25.4	—
1917	28.4	34.6	43.2	50.5	60.0	66.0	77.4	75.7	66.2	54.3	42.9	41.0	53.4
1918	34.8	36.0	45.3	53.8	58.8	73.4	76.9	72.1	68.5	56.2	40.6	35.6	54.3
1919	31.1	37.1	45.5	54.5	60.8	66.9	76.6	74.4	64.0	49.4	38.8	18.5	51.5
1920	26.8 ^a	35.2	46.4	51.5	58.9	67.6	78.6	74.0	63.2	51.8	40.6	37.2	52.6
1921	35.3	38.1	47.6	51.5	63.6	72.5	75.3	75.4	59.8 ^b	54.8	40.4	31.4	53.8
1922	19.1	30.7	44.4	52.2	61.3	75.4	78.5	75.3	66.4	54.7	38.4	23.0	51.6
1923	37.0	28.8	46.0	54.8	62.2	67.8	76.6	77.5	68.4	53.7	42.8	33.2	54.1
1924	25.0	43.7	46.9	54.3	68.3	71.2	75.2	73.5	65.5	54.6	39.4	23.8	53.4
1925	34.8	44.4	47.3	56.4	63.8	71.0	79.2	72.8	63.4	50.6	37.6	38.0	54.9
1926	33.9	43.7	49.8	58.9	61.8	72.6	78.7	72.6	58.8	53.6	42.8	27.8	54.6
1927	28.3	40.8	44.6	52.4	58.6	70.0	75.6	72.9	61.5	52.3	44.0	26.2	52.3
1928	28.4	36.2	47.8	51.4	65.6	68.9	77.6	72.0	63.9	52.5	40.8	32.2	53.1
1929	20.0	21.4	46.8	51.8	61.8	67.4	74.6 ^a	74.4	62.2	54.2	32.9	34.6	50.2
1930	12.2	39.7	48.6	57.8	61.5	67.8	77.6	75.6	65.2	48.8	38.6	31.3	52.1
1931	35.3	38.2	46.2	54.4	67.0	69.0	78.4	75.4	65.1	52.1	37.4	27.8	53.9
1932	26.4	34.0	46.4	56.0	61.2	72.3	73.0	74.0	65.3	53.4	43.8	30.4	53.0
1933	35.8	26.6 ^a	42.6 ^a	51.6	56.6	68.4	74.4	73.2	60.4	54.1 ^a	41.6	42.0	52.3
1934	39.6	41.8	49.6	59.6	64.4	70.7	74.2	73.6	62.0	55.6	46.0	37.5	56.2
1935	33.1	37.0	43.0	49.5	60.5	67.2	73.4	70.7	66.9	50.7 ^a	35.3	32.4	51.6
1936	35.4 ^a	21.5	43.8	57.3	66.2	70.3	75.2	73.8	63.0	54.2	31.8	35.6	52.4
1937	13.0	30.5	46.7	51.4	61.5	68.4	77.3	70.7	65.4	56.2	44.9	37.8	52.0
1938	36.7	34.0	45.0	53.0	61.1	71.2	79.0	72.0	70.0	54.0	37.4	33.8	53.9
1939	36.6	34.4	45.9	54.6	63.4	67.2	77.3	75.5	66.7	53.4	39.9	39.0	54.5
1940	32.0	39.6	49.0	54.4	64.6	72.8	76.6	74.0	69.5	57.3	35.6	33.8	54.9
1941	33.4	38.1	47.6	55.4	60.8 ^a	68.5 ^a	80.0	73.4	61.0	52.2	43.8	36.7	54.2
1942	25.4	37.8	43.2	54.5	60.2	66.8	76.0	76.8	66.8	54.4	38.2	32.0	52.7
1943	22.7	36.2	42.9	—	—	—	—	—	—	—	—	32.8	—
1944	29.4	36.2	41.5	52.9	61.8 ^a	—	—	—	—	—	—	—	—
1945	33.9	38.6	42.1	50.3	61.7	67.5	78.0	77.5	64.6	56.4	40.6	32.7	53.6
1946	34.4	39.6	45.5	53.7	64.2	66.9	76.1	76.6	63.5	49.5	35.8	34.8	53.4
1947	27.3	39.9	49.6	56.2	68.8	67.8	75.4	71.8	65.5	53.3	41.2	33.1	54.2
1948	32.0	31.8	42.1	49.3	58.4	72.5	72.8	71.9	64.4	50.9	40.8	26.8	51.1
1949	14.0	31.9	45.1	55.5	67.0	69.4	75.0	74.9	68.1	50.3	45.3	35.1	52.6
1950	12.1	30.7	42.3	49.9	59.0	66.5	75.4	76.4	67.5	51.1	40.7	36.2	50.6

TABLE 2.3-2 (Cont'd)

YEAR	JAN	FEB	MAR	APR	MAY	JUNE	JULY	AUG	SEPT	OCT	NOV	DEC	ANNUAL
1951	33.0	36.9	40.1	54.1	61.1	69.4	76.7	74.2	66.8	51.5	39.5	27.4	52.6
1952	25.2	36.7	44.1	55.2	62.7	67.1	77.0	74.0	69.0	<u>59.0</u>	34.0	34.8	53.2
1953	<u>42.5</u>	41.2	46.2	51.0	58.0	<u>63.0</u>	75.8	74.0	67.8	<u>55.4</u>	43.4	37.6	54.7
1954	28.9	39.3	41.5	51.4	62.9	<u>65.5</u>	73.9	71.4	66.1	51.4	<u>46.0</u>	34.0	52.6
1955	30.0	35.3	<u>39.4</u>	<u>47.5</u>	57.0	70.2	73.0	75.5	66.4	53.3	<u>31.3</u>	29.4	50.7
1956	31.8	25.6	43.8	56.2	65.3	65.7	78.9	75.3	67.3	52.1	36.6	34.6	52.8
1957	16.5	34.1	44.0	55.2	65.9	70.8	74.3	72.9	69.0	56.7	40.4	38.5	52.7
1958	37.1	<u>44.5</u>	43.5	51.3	68.1	73.9	81.2	79.4	65.6	54.4	40.6	35.2	<u>56.2</u>
1959	32.0	<u>35.5</u>	45.1	54.2	57.5	68.6	77.7	71.8	62.6	53.4	36.4	33.1	<u>52.3</u>
1960	23.3	37.4	45.1	52.6	58.5	70.1	<u>81.8</u>	71.4	67.7	54.5	41.2	29.0	52.7
1961	35.0	43.7	46.1	52.3	60.0	74.0	79.4	80.2	63.8	51.6	35.3	33.7	54.6
1962	29.8	36.6	42.6	55.6	56.9	68.3	76.0	71.9	67.1	52.6	43.2	36.8	53.1
1963	25.4	38.3	46.4	49.8	61.7	69.4	<u>72.4</u>	75.7	71.1	56.0	42.8	30.2	53.3
1964	35.6	38.1	43.8	50.2	59.7	67.7	<u>74.5</u>	69.8	63.0	53.3	38.2	25.4	51.6
1965	32.3	40.5	42.9	54.8	60.5	69.3	76.5	<u>74.7</u>	62.4	57.1	43.1	33.0	53.9
1966	34.0	39.9	<u>45.4</u>	54.6	63.2	66.9	73.3	75.6	68.8	53.4	43.7	38.2	54.8
1967	39.8	43.7	44.3	47.6	60.5	72.5	78.6	81.5	<u>71.7</u>	55.1	41.5	33.1	55.8
1968	35.7	41.8	49.0	51.3	62.4	69.8	79.7	<u>71.5</u>	<u>66.8</u>	50.3	41.7	30.6	54.2
1969	19.8	31.7	45.8	52.2	64.6	75.1	76.0	72.8	67.4	51.0	40.2	34.6	52.6
1970	30.7	40.6	45.0	49.0	61.5	73.6	78.6	76.3	61.8	50.9	39.7	30.8	53.2
AVG	29.4	36.2	45.2	53.2	61.8	69.4	76.4	74.2	65.2	53.1	40.0	32.6	53.1
^b NORMAL	29.4	37.4	44.2	52.5	61.7	69.2	76.6	74.6	66.2	53.0	40.2	33.1	53.2

^aAverage adjusted for one or more days of missing data

^bNormals are averages for the period 1941-1970

— The highest and lowest averages in each column are underscored

TABLE 2.3-3

FREQUENCY DISTRIBUTION IN PERCENT OF DRY BULB TEMPERATURE VS RELATIVE HUMIDITY

MARCH, APRIL, AND MAY 1957-1970

JUNE, JULY, AND AUGUST 1957-1970

RELATIVE HUMIDITY								RELATIVE HUMIDITY							
DRY BULB	<30	30/44	45/59	60/74	75/89	>89	TOTAL	DRY BULB	<30	30/44	45/59	60/74	75/89	>89	TOTAL
0/4	0	0	0	0	0	0	0	30/34	0	0	0	0	0	0	0
5/9	0	0	0	0	0	0	0	35/39	0	0	0	a	0	0	a
10/14	0	0	0	0	0	0	a	40/44	0	0	0	0.07	0	0	0.07
15/19	0.03	0.02	0.06	0.03	a	0.02	0.16	45/49	0	a	0.14	0.15	0.05	0.03	0.37
20/24	0.03	0.07	0.10	0.08	0.06	0.06	0.40	50/54	0.01	0.22	1.06	0.62	0.34	0.09	2.34
25/29	0.06	0.10	0.14	0.35	0.45	0.24	1.33	55/59	0.09	1.41	3.16	1.16	0.48	0.16	6.46
30/34	0.06	0.08	0.69	1.24	1.14	0.42	3.63	60/64	0.38	4.30	5.38	1.42	0.58	0.16	12.22
35/39	0.13	0.39	2.31	2.19	1.58	0.45	7.05	65/69	1.71	7.40	4.54	0.73	0.17	0.02	14.57
40/44	0.18	1.42	4.78	3.44	1.97	0.83	12.62	70/74	4.35	9.00	2.16	0.24	0.03	a	15.78
45/49	0.33	3.61	6.05	3.40	1.60	0.73	15.72	75/79	6.88	6.55	0.57	0.02	0	0	14.02
50/54	1.36	5.62	6.08	2.47	0.81	0.42	16.76	80/84	8.83	3.27	0.08	0	0	0	12.18
55/59	2.32	6.18	3.62	1.12	0.53	0.12	13.89	85/89	7.92	0.85	a	0	0	0	8.77
60/64	3.24	5.03	2.12	0.48	0.15	0.02	11.04	90/94	6.81	0.10	0	0	0	0	6.91
65/69	3.01	3.28	0.79	0.13	0.02	0	7.23	95/99	4.24	0	0	0	0	0	4.24
70/74	2.74	1.70	0.27	0.02	0	0	4.73	100/104	1.76	0	0	0	0	0	1.76
75/79	1.97	0.63	0.04	a	0	0	2.64	105/109	0.29	0	0	0	0	0	0.29
80/84	1.44	0.16	0.01	0	0	0	1.61	>109	0.02	0	0	0	0	0	0.02
85/89	0.70	0.05	0	0	0	0	0.76	TOTAL	43.29	33.10	17.09	4.41	1.65	0.46	100.00
90/94	0.37	0	0	0	0	0	0.37								
95/99	0.06	0	0	0	0	0	0.06								
100/104	a	0	0	0	0	0	a								
105/109	0	0	0	0	0	0	0								
>109	0	0	0	0	0	0	0								
TOTAL	18.03	28.34	27.06	14.95	8.31	3.31	100.00								

aDenotes less than 0.01 percent of 30912 total observations

aDenotes less than 0.01 percent of 30912 total observations

aDenotes less than 0.01 percent of 30912 total observations

TABLE 2.3-3 (Cont'd)

SEPTEMBER, OCTOBER, AND NOVEMBER 1957-1970

DECEMBER, JANUARY, AND FEBRUARY 1957-1970

RELATIVE HUMIDITY								RELATIVE HUMIDITY							
DRY BULB	<30	30/44	45/59	60/74	75/89	>89	TOTAL	DRY BULB	<30	30/44	45/59	60/74	75/89	>89	TOTAL
-10/-6	0	0	0	0	0	0	0	< -20	0	0	a	0.01	0	0	0.01
-5/-1	0	0	0	0	0	0	0	-20/-16	0	0	0	0.03	0.01	0	0.04
0/4	0	0	0	0	0	0	0	-15/-11	0	a	0.03	0.04	0.01	0	0.08
5/9	0	0	0.01	a	0	0	0.01	-10/-6	0	0.07	0.03	0.10	0.01	0	0.21
10/14	0	0.02	0.02	0.03	0.01	0	0.08	-5/-1	0	0.09	0.05	0.18	0.04	0.01	0.37
15/19	0	0.06	0.02	0.06	0.07	0.05	0.26	0/4	0	0.08	0.10	0.30	0.12	0.02	0.68
20/24	0.02	0.15	0.07	0.24	0.32	0.20	1.00	5/9	0.04	0.06	0.24	0.30	0.43	0.07	1.14
25/29	0	0.06	0.23	0.53	0.82	0.60	2.24	10/14	0.07	0.15	0.30	0.37	0.61	0.52	2.02
30/34	0.02	0.12	0.36	1.24	2.07	1.73	5.54	15/19	0.11	0.19	0.27	0.47	1.12	1.39	3.55
35/39	a	0.26	0.97	2.34	2.73	1.79	8.09	20/24	0.08	0.20	0.54	0.95	2.35	3.04	7.16
40/44	0.05	0.46	2.37	3.64	3.20	2.31	12.03	25/29	0.06	0.37	0.66	1.51	4.85	6.47	13.92
45/49	0.11	1.03	3.61	3.62	2.89	1.59	12.85	30/34	0.14	0.31	0.94	3.38	7.37	10.51	22.65
50/54	0.23	2.44	4.76	3.42	1.78	0.44	13.07	35/39	0.06	0.38	2.49	4.94	5.43	5.54	18.84
55/59	0.51	3.41	4.39	2.24	0.66	0.22	11.43	40/44	0.08	0.82	3.57	4.33	2.65	2.73	14.18
60/64	1.18	3.84	4.17	1.38	0.24	0.06	10.87	45/49	0.16	1.41	3.00	2.70	1.07	0.37	8.71
65/69	1.41	3.69	2.39	0.30	0.03	0	7.82	50/54	0.16	1.13	1.68	1.15	0.21	0.05	4.38
70/74	1.98	3.32	0.82	0.01	0	0	6.13	55/59	0.10	0.41	0.79	0.28	0.02	0	1.60
75/79	2.00	1.79	0.11	0	0	0	3.90	60/64	0.03	0.18	0.19	0.03	0	0	0.43
80/84	1.95	0.62	0	0	0	0	2.57	65/69	0.01	0.01	0.01	0	0	0	0.03
85/89	1.23	0.09	0	0	0	0	1.32	70/74	0	0	0	0	0	0	0
90/94	0.65	0.01	0	0	0	0	0.66								
95/99	0.13	0	0	0	0	0	0.13								
100/104	0	0	0	0	0	0	0								
105/109	0	0	0	0	0	0	0								
TOTAL	11.47	21.37	24.30	19.05	14.82	8.99	100.00	TOTAL	1.10	5.86	14.95	21.07	26.30	30.72	100.00

^aDenotes less than 0.01 percent of 30312 total observations^aDenotes less than 0.01 percent of 30576 total observations

TABLE 2.3-4

MONTHLY AVERAGES OF PSYCHROMETRIC DATA BASED ON PERIOD OF RECORD: 1950-1970

		AVERAGES												
		JAN	FEB	MAR	APR	MAY	JUNE	JULY	AUG	SEPT	OCT	NOV	DEC	YEAR
DRY BULB		30.3	37.5	44.0	52.5	61.8	69.9	77.5	75.3	67.0	53.2	40.1	33.4	53.5
WET BULB		27.9	33.6	37.3	42.8	49.1	54.5	57.9	57.3	52.6	45.4	36.4	31.2	43.8
REL. HUM.		76.0	69.7	55.0	46.4	41.8	39.4	31.5	34.9	39.9	57.7	72.6	80.8	53.8
DEWPOINT		23.2	27.4	27.3	30.4	36.0	41.2	42.3	42.8	39.5	36.9	31.1	27.5	33.8
		MONTHLY AVERAGE EXTREMES												
DRY BULB	HIGHEST	43.0	44.0	48.7	56.2	68.7	75.5	82.8	82.5	72.0	59.1	45.8	38.8	56.3
	YEAR	1953	1958	1968	1956	1958	1969	1960	1967	1967	1952	1954	1953	1958
	LOWEST	12.9	25.8	39.6	48.3	57.2	64.2	73.2	70.6	61.6	50.3	32.3	26.5	51.0
	YEAR	1950	1956	1955	1955	1962	1953	1963	1964	1970	1968	1955	1964	1955 ^a
WET BULB	HIGHEST	39.3	40.7	40.8	45.1	54.6	58.6	61.2	61.1	56.5	47.7	42.3	35.8	46.5
	YEAR	1953	1958	1968	1962	1958	1958	1958	1961	1963	1962	1954	1966	1958
	LOWEST	12.4	23.4	32.9	39.3	45.4	51.4	55.6	54.9	48.3	42.4	29.6	25.0	41.8
	YEAR	1950	1956	1955	1955	1959	1954	1954	1964	1970	1970	1955	1964	1955
REL. HUM.	HIGHEST	89.0	87.0	66.0	64.0	52.0 ^b	54.0	40.0	44.0	55.0	74.0	80.0	90.0	58.0
	YEAR	1960	1963	1950	1963	1962 ^a	1950	1955	1968	1959	1962	1956	1950	1950 ^a
	LOWEST	60.0	54.0	44.0	37.0	31.0	34.0	22.0	24.0	34.0	42.0	64.0	69.0	49.0
	YEAR	1963	1967	1965	1966	1966	1960	1959	1967	1952	1952	1963 ^a	1968	1967
DEWPOINT	HIGHEST	34.4	36.7	34.0	37.1	43.8	47.5	46.6	46.9	45.4	43.5	38.3	34.3	37.7
	YEAR	1953	1958	1961	1963	1957	1958	1958	1961	1963	1962	1954	1950	1958
	LOWEST	6.5	17.3	20.8	26.2	30.4	37.5	35.4	38.4	33.8	32.1	24.0	21.0	31.5
	YEAR	1950	1956	1965 ^a	1955	1964	1954	1959	1955	1970	1970	1959	1961	1955

^a Also in earlier years^b Although not included in these tables, an average of 63% was recorded in 1948.

TABLE 2.3-5

COMPOSITE RECORD OF MONTHLY AND ANNUAL PRECIPITATION (INCHES)
HANFORD TOWNSITE (NEAR) AND HMS: 1912-1970

YEAR	JAN	FEB	MAR	APR	MAY	JUNE	JULY	AUG	SEPT	OCT	NOV	DEC	TOTAL
1912	—	—	—	—	0.93	0.22	0.09	0.93	0.01	0.55	1.19	0.46	(4.41)
1913	1.39	0.16	0.05	0.16	0.61	1.11	0	T	0.03	1.42	0.75	0.76	5.44
1914	1.12	1.15	T	0.54	0.60	0.29	0.73	0	0.32	1.75	0.58	0.74	7.82
1915	0.65	0.61	0.58	0.39	0.90	0.76	0.21	T	T	0.20	1.56	1.10	6.96
1916	0.69	1.54	0.55	0.77	0.01	—	—	0	0.02	0	1.20	0.87	(5.65)
1917	1.00	0.28	0.31	0.72	0.57	0.09	T	0	0.07	0	0.81	0.50	4.35
1918	0.63	0.50	0.22	0	0.34	0.68	0.28	0.38	0.15	0.54	0.69	0.49	4.90
1919	1.04	1.09	0.38	0.32	0.64	0	T	T	0.74	0.21	0.88	1.00	6.30
1920	1.46	T	T	0.78	0.05	0.69	T	<u>1.18</u>	0.75	0.51	0.99	1.22	7.63
1921	0.36	1.25	T	0.60	0.33	0.12	T	0.14	0.06	0.29	1.61	0.69	5.45
1922	0.37	0.45	0.26	0.14	T	0.65	0	0.26	0.26	0.72	0.28	1.03	4.42
1923	0.37	0.40	0.05	0.85	0.27	1.11	0.33	T	0.43	1.03	0.06	0.56	5.46
1924	1.07	0.78	0.10	0.01	1	0.14	0.37	0.17	0.83	0.77	1.36	1.16	6.71
1925	0.66	0.70	0.06	0.80	0.87	0.34	0	T	0.49	0.07	0.96	0.66	5.61
1926	0.93	0.57	0.08	0.04	0.26	0.80	0	0.10	0.30	0.40	<u>3.05</u>	0.78	7.31
1927	1.29	0.77	0.45	0.03	0.09	T	T	0.29	1.14	0.62	1.26	0.40	6.34
1928	1.34	0.09	0.63	0.71	0.12	0.12	0.08	0.15	0.05	0.01	0.49	1.33	5.58
1929	1.37	0.07	0.02	0.05	0.25	0.89	T	0.07	0.15	0.07	T	0.82	4.78
1930	0.58	0.78	0.14	0.24	0.43	0.09	T	0.06	0.40	0.04	0.32	0.30	3.38
1931	1.35	0.17	0.99	0.11	0	0.51	0.19	0	0.16	0.47	1.15	<u>2.53</u>	7.63
1932	1.52	0.15	0.63	0.96	0.29	0.06	0.03	0.10	T	0.77	0.65	0.24	6.40
1933	0.86	0.50	0.43	0	1.06	0.17	0.10	0.13	0.44	0.59	0.48	1.20	5.96
1934	0.41	0.10	0.57	0.29	1.56	0.12	0	0.45	0.71	0.74	1.05	0.50	6.50
1935	0.41	0.31	0.05	0.08	0.13	0.37	0.46	0	0.05	0.39	0.30	0.87	3.42
1936	1.54	0.65	0	0.35	0.33	0.50	0.04	0.01	0.28	0.06	T	0.98	4.74
1937	0.80	0.63	0.76	0.73	0.01	1.09	0.24	0.06	0.16	0.39	1.84	1.08	7.79
1938	0.35	0.77	<u>1.15</u>	0.18	0.11	1.78	0.06	0.14	0.02	0.79	0.67	0.71	6.73
1939	0.59	0.80	0.52	T	0.07	0.07	0	0.28	0.11	0.25	T	1.69	4.38
1940	0.84	<u>3.08</u>	0.47	0.26	0.28	0.11	0.34	0	0.88	2.03	1.45	1.52	11.26
1941	1.82	0.65	0.16	0.31	0.83	1.16	0.08	0.59	0.57	0.41	0.46	0.40	7.44
1942	0.46	0.47	0	0.20	1.02	1.28	0.02	0	T	0.54	1.34	1.19	6.52
1943	0.40	0.37	0.33	—	—	—	—	—	—	—	—	0.23	(1.33)
1944	0.39	0.56	0.03	0.77	—	—	—	—	—	—	—	—	(1.75)
1945	—	—	—	—	—	—	—	—	—	—	—	—	—
1946	—	—	—	—	—	—	0.15	0.35	0.52	0.65	0.66	<u>0.11</u>	(2.44)
1947	0.32	0.27	0.42	0.70	0.02	1.07	0.71	0.68	<u>1.34</u>	2.20	0.81	0.75	9.29
1948	1.36	0.69	0.07	0.95	<u>1.71</u>	1.47	0.40	0.39	0.16	0.45	0.95	1.11	9.71
1949	<u>0.13</u>	0.68	1.12	0.02	0.16	0.01	0.01	0.03	0.23	0.10	1.47	0.16	4.12
1950	1.80	1.06	0.87	0.47	0.27	<u>2.92</u>	0.07	T	0.01	<u>2.46</u>	0.55	0.97	<u>11.45</u>

TABLE 2.3-5 (Cont'd)

YEAR	JAN	FEB	MAR	APR	MAY	JUNE	JULY	AUG	SEPT	OCT	NOV	DEC	TOTAL
1951	0.84	0.51	0.46	0.53	0.43	1.39	0.37	0.15	0.10	0.71	0.82	0.70	7.00
1952	0.65	0.50	0.06	0.13	0.58	1.07	T	0.08	0.08	0.04	0.20	0.77	4.16
1953	2.16	0.25	0.17	0.77	0.28	0.55	T	0.96	0.13	0.20	0.96	0.49	6.92
1954	1.48	0.28	0.59	0.07	0.41	0.10	0.22	0.42	0.51	0.42	0.86	0.35	5.71
1955	0.56	0.22	0.17	0.40	0.59	0.28	0.57	0	0.77	0.40	1.54	2.03	7.53
1956	1.71	0.56	0.10	T	0.22	0.86	T	0.38	0.01	1.03	0.15	0.58	5.60
1957	0.58	0.73	1.86	0.38	0.82	0.47	0.05	0.02	0.34	2.72	0.39	0.53	8.29
1958	1.74	1.48	0.46	0.64	0.74	0.81	0.02	T	0.05	0.19	0.77	1.84	8.74
1959	2.05	1.17	0.40	0.20	0.50	0.23	T	0.03	1.26	0.56	0.41	0.26	7.07
1960	0.51	0.58	0.67	0.53	0.71	0.14	T	0.26	0.23	0.23	0.92	0.65	5.42
1961	0.33	2.10	1.02	0.48	0.80	0.42	0.15	0.09	T	0.07	0.49	0.88	6.83
1962	0.13	0.90	0.14	0.34	1.35	0.12	T	0.50	0.38	0.92	0.62	0.60	6.03
1963	0.95	0.69	0.53	1.17	0.43	0.28	0.31	0.01	0.02	0.04	0.74	1.14	6.31
1964	0.37	0.01	0.03	0.11	0.04	0.90	0.04	0.24	0.09	0.28	0.94	2.34	5.39
1965	0.93	0.14	0.03	0.09	0.15	0.49	0.11	0.03	0.11	0.01	1.17	0.39	3.15
1966	0.68	0.03	0.39	0.03	0.05	0.43	0.81	T	0.27	0.39	2.25	0.60	5.93
1967	0.32	T	0.14	0.90	0.56	0.57	T	T	0.05	0.13	0.16	0.43	3.26
1968	0.88	0.58	0.02	0.01	0.06	0.19	0.04	0.51	0.25	0.93	1.23	1.25	5.95
1969	1.24	0.54	0.10	1.22	0.51	0.75	T	T	0.48	0.10	0.13	1.29	6.36
1970	2.47	0.75	0.24	0.45	0.54	0.25	0.01	T	0.03	0.24	0.71	0.61	6.33
AVG	0.93	0.62	0.36	0.40	0.45	0.57	0.14	0.19	0.30	0.58	0.85	0.86	6.25
^a NORMAL	0.97	0.58	0.38	0.44	0.53	0.70	0.15	0.21	0.30	0.61	0.80	0.81	6.48

The highest, and if measurable, the lowest totals in each column are underscored

T Denotes a trace, an amount too small to measure

() Denotes an incomplete annual total

^a Normals are averages for the period 1941-1970

TABLE 2.3-6

AVERAGE NUMBER OF DAYS WITH PRECIPITATION
OF SPECIFIED AMOUNT: 1956-1970

MONTH	TRACE OR MORE	0.01 INCH OR MORE	0.10 INCH OR MORE	0.25 INCH OR MORE	0.50 INCH OR MORE	1.00 INCH OR MORE
JAN	17	9	3	1	b	0
FEB	12	7	2	1	b	0
MAR	11	6	2	b	0	0
APR	11	5	2	b	b	0
MAY	10	5	2	1	b	0
JUNE	10	5	2	1	b	b
JULY	4	2	1	b	0	0
AUG	6	2	1	b	b	0
SEPT	7	3	1	b	b	0
OCT	11	6	2	1	b	b
NOV	15	8	3	1	b	0
DEC	18	10	3	1	b	0
ANNUAL	132	68	24	7	1 ^a	b

^aAlthough the number of days with 0.50 inch or more averages less than 0.5 days for any one month, 29 such days have been recorded during 25 years of record for an average annual number of 1

^bUsed to denote an average of less than ½ day

TABLE 2.3-7

MONTHLY AND ANNUAL AVERAGES AND EXTREMES IN TOTAL
TIME WITH PRECIPITATION OBSERVED: 1946-1970

MONTHS	AVERAGES		GREATEST		YEAR	LEAST		YEAR
	NO. OF HOURS	% OF TIME	NO. OF HOURS	% OF TIME		NO. OF HOURS	% OF TIME	
JAN	96.4	13.0	212.0	28.5	1969	29.2	3.9	1949
FEB	56.7	8.4	106.8	15.9	1958	4.7	0.7	1967
MAR	37.1	5.0	135.2	18.2	1957	7.3	1.0	1968
APR	31.5	4.4	69.2	9.6	1953	1.8	0.2	1956
MAY	34.9	4.7	89.9	12.1	1948	4.8	0.6	1947
JUNE	33.4	4.6	80.8	11.2	1950	3.2	0.4	1949
JULY	9.6	1.3	38.2	5.1	1966	1.0	0.1	1958
AUG	13.5	1.8	61.7	8.3	1968	0.0	0.0	1955
SEPT	17.0	2.4	52.2	7.2	1959	1.3	0.2	1950
OCT	38.8	5.2	119.9	16.1	1947	3.6	0.5	1952
NOV	68.1	9.6	132.2	18.4	1955	21.4	3.0	1957
DEC	93.8	12.6	179.6	24.1	1963	31.5	4.2	1946
ANNUAL AVG	44.2	6.1	738.0 ^a	8.4	1950	355.4 ^a	4.1	1949

^aAnnual extremes

TABLE 2.3-8

TOTAL DURATION^a AND MAXIMUM PERSISTENCE OF FOG
TABULATED IN HOURS: 1945-1970

ALL FOG (VIZ 0-6 MILES)	AVG TOTAL DURATION	MAX TOTAL DURATION	YEAR	MIN TOTAL DURATION	YEAR	AVG DURATION PER DAY OF OCCURRENCE	MAX. PERSISTENCE ^e	YEAR
JAN	68.3	193.4	1965	0	1949	7.2	58.1	1955
FEB	36.4	206.2	1963	0	1967	6.2	58.0	1963
MAR	4.4	20.6	1951	0	1968+	3.1	12.2	1949
APR	0.3	2.8	1950	0	1970+	1.4	2.8	1950
MAY	0.3	2.7	1958	0	1970+	1.2	2.7	1958
JUNE	b	0.5	1948	0	1970+	0.5	0.5	1948
JULY	b	0.7	1966	0	1970+	0.7	0.7	1966
AUG	b	1.0	1959	0	1970+	1.0	0.7	1959
SEPT	0.3	5.5	1957	0	1970+	2.0	2.6	1957
OCT	7.6	63.6	1962	0	1970+	3.9	39.0	1962
NOV	55.4	148.0	1952	1.0	1960	6.8	65.4	1963
DEC	105.4	193.8	1947	6.5	1968	8.7	72.3	1947
Y	278.4	462.5 ^c	1964-65	147.7 ^d	1948-49	7.0	72.3	1947-48
DENSE FOG (VIZ ¼ MI OR LESS)								
JAN	20.4	52.4	1955	0	1949	3.4	15.0	1953
FEB	12.7	86.7	1963	0	1967+	3.8	16.7	1963
MAR	1.8	7.8	1949	0	1968+	2.2	5.0	1961
APR	0.1	1.8	1955	0	1970+	1.8	0	-----
MAY	0.1	1.6	1958	0	1970+	1.6	1.6	1958
JUNE	0	0	-----	0	-----	0	0	-----
JULY	0	0	-----	0	-----	0	0	-----

TABLE 2.3-8 (Cont'd)

DENSE FOG (VIZ ¼ MI OR LESS)	AVG TOTAL DURATION	MAX TOTAL DURATION	YEAR	MIN TOTAL DURATION	YEAR	AVG DURATION PER DAY OF OCCURRENCE	MAX. PERSISTENCE ^e	YEAR
AUG	b	1.0	1959	0	1970+	1.0	0.7	1959
SEPT	0.1	3.2	1957	0	1970+	3.2	1.4	1957
OCT	3.1	35.2	1962	0	1970+	3.1	15.8	1962
NOV	21.1	71.4	1952	0	1960	4.1	20.6	1963
DEC	42.0	119.8	1947	1.3	1968	5.4	47.0	1957
Y	101.4	221.5 ^c	1962-63	47.3 ^d	1948-49	4.2	47.0	1957-58

^a Total duration denotes total number of hours and tenths in which fog is observed.

^b Denotes less than 0.05 hour

^c Denotes the greatest number of hours in a season

^d Denotes the least number of hours in a season

^e Maximum persistence of dense fog is based on the period 1953-70

TABLE 2.3-9

ANNUAL PERCENT OCCURRENCE OF WIND SPEED VS ELEVATION:
1950-1970

SPEED (MPH)	ELEVATION (FT)		
	50	200	400
0 to 1	9	7	7
1 to 3	16	13	11
3 to 7	33	25	23
7 to 12	24	23	21
12 to 18	12	19	19
over 19	6	14	19

TABLE 2.3-10

ANNUAL PERCENT OCCURRENCE OF WIND DIRECTION VS
ELEVATION: 1950-1970

SECTOR	ELEVATION (FT)		
	50	200	400
N	3.6	4.0	3.9
NNE	3.3	3.6	3.3
NE	2.0	2.2	2.1
ENE	2.4	2.5	2.5
E	2.6	2.7	2.4
ESE	3.7	3.9	3.8
SE	2.8	2.8	2.6
SSE	3.2	2.7	2.6
S	4.1	3.6	3.5
SSW	7.2	6.4	6.5
SW	8.5	7.3	7.1
WSW	9.8	8.3	7.9
W	16.1	16.3	16.8
WNW	16.7	19.7	19.9
NW	4.9	6.3	6.9
NNW	4.5	4.9	5.1
VAR	2.4	1.3	1.1
CALM	2.2	1.5	2.2

TABLE 2.3-11
ANNUAL PERCENT OCCURRENCE OF STABILITY CLASSES: 1950-1970

PERCENT FREQUENCY OF OCCURRENCE WIND SPEED (MPS) VS WIND DIRECTION STABILITY CLASS A						PERCENT FREQUENCY OF OCCURRENCE WIND SPEED (MPS) VS WIND DIRECTION STABILITY CLASS B					
SECTOR	1-3	4-7	8-11	12-18	GT19	SECTOR	1-3	4-7	8-11	12-18	GT19
N	.18	.11	—	—	—	N	.07	.03	—	—	—
NNE	.20	.10	—	—	—	NNE	.10	.05	—	—	—
NE	.10	.01	—	—	—	NE	.07	—	—	—	—
ENE	.11	—	—	—	—	ENE	.08	—	—	—	—
E	.08	—	—	—	—	E	.05	—	—	—	—
ESE	.08	—	—	—	—	ESE	.05	.02	—	—	—
SE	.06	.02	—	—	—	SE	.02	—	—	—	—
SSE	.14	.02	—	—	—	SSE	.07	.01	—	—	—
S	.18	.09	—	.01	—	S	.07	.05	—	—	—
SSW	.18	.29	.12	.04	—	SSW	.10	.18	.04	.01	—
SW	.09	.30	.04	.01	—	SW	.07	.15	.02	.01	—
WSW	.07	.10	.02	—	—	WSW	.06	.06	.02	—	—
W	.06	.20	.04	—	—	W	.06	.15	.02	—	—
WNW	.17	.55	.13	.03	—	WNW	.14	.28	.15	.01	—
NW	.09	.03	—	—	—	NW	.08	.02	—	—	—
NNW	.29	.07	—	—	—	NNW	.17	.04	—	—	—

Number of Cases of Calm 167.

Number of Cases of VRBL 652.

A — Indicates an Occurrence Fraction of
Less Than 0.01 Percent

Number of Cases of Calm 148.

Number of Cases of VRBL 221.

A — Indicates an Occurrence Fraction of
Less Than 0.01 Percent

TABLE 2.3-11 (Cont'd)

PERCENT FREQUENCY OF OCCURRENCE WIND SPEED (MPS) VS WIND DIRECTION STABILITY CLASS C						PERCENT FREQUENCY OF OCCURRENCE WIND SPEED (MPS) VS WIND DIRECTION STABILITY CLASS D					
SECTOR	1-3	4-7	8-11	12-18	GT19	SECTOR	1-3	4-7	8-11	12-18	GT19
N	.05	.03	—	—	—	N	1.17	.40	—	—	—
NNE	.09	.02	—	—	—	NNE	1.27	.26	—	—	—
NE	.03	—	—	—	—	NE	.76	.06	—	—	—
EWE	.07	—	—	—	—	ENE	.82	.02	—	—	—
E	.04	—	—	—	—	E	1.01	.04	—	—	—
ESE	.05	—	—	—	—	ESE	1.05	.09	—	—	—
SE	.02	—	—	—	—	SE	.63	.10	—	—	—
SSE	.06	.01	—	—	—	SSE	.82	.26	.03	—	—
S	.07	.03	—	—	—	S	.70	.53	.10	.07	—
SSW	.15	.09	.02	.01	—	SSW	.88	1.26	.37	.14	—
SW	.05	.09	.02	.01	—	SW	1.00	1.52	.20	.05	—
WSW	.05	.03	—	—	—	WSW	1.10	.94	.06	.02	—
W	.07	.11	.01	—	—	W	1.93	2.57	.21	.02	—
WNW	.12	.17	.03	—	—	WNW	3.30	3.19	.51	.04	—
NW	.11	.01	—	—	—	NW	1.71	.36	.01	—	—
NNW	.10	.02	—	—	—	NNW	1.86	.40	—	—	—

Number of Cases of Calm 121.

Number of Cases of VRBL 114.

A — Indicates an Occurrence Fraction of
Less Than 0.01 Percent

Number of Cases of Calm 4510.

Number of Cases of VRBL 852.

A — Indicates an Occurrence Fraction of
Less Than 0.01 Percent

TABLE 2.3-11 (Cont'd)

PERCENT FREQUENCY OF OCCURRENCE WIND SPEED (MPS) VS WIND DIRECTION STABILITY CLASS E						PERCENT FREQUENCY OF OCCURRENCE WIND SPEED (MPS) VS WIND DIRECTION STABILITY CLASS F					
SECTOR	1-3	4-7	8-11	12-18	GT19	SECTOR	1-3	4-7	8-11	12-18	GT19
N	.58	.11	—	—	—	N	.18	.01	—	—	—
NNE	.68	.07	—	—	—	NNE	.18	—	—	—	—
NE	.49	.03	—	—	—	NE	.12	—	—	—	—
EWE	.65	.04	—	—	—	ENE	.21	.02	—	—	—
E	.87	.03	—	—	—	E	.31	.01	—	—	—
ESE	1.22	.12	.01	—	—	ESE	.65	.10	—	—	—
SE	.75	.20	.01	—	—	SE	.62	.17	—	—	—
SSE	.78	.23	.04	.02	—	SSE	.58	.05	.01	—	—
S	.67	.50	.11	.10	—	S	.61	.07	.01	—	—
SSW	1.21	1.39	.25	.10	—	SSW	1.10	.15	—	—	—
SW	1.64	1.23	.07	—	—	SW	1.98	.55	—	—	—
WSW	3.15	1.66	.62	.01	—	WSW	2.44	.43	—	—	—
W	4.87	4.22	.17	—	—	W	3.00	1.16	—	—	—
WNW	3.70	3.21	.17	—	—	WNW	2.82	1.10	—	—	—
NW	1.27	.20	—	—	—	NW	.74	.09	—	—	—
NNW	1.03	.16	—	—	—	NNW	.41	.02	—	—	—

Number of Cases of Calm 5293

Number of Cases of VRBL 388

A — Indicates an Occurrence Fraction of
Less Than 0.01 Percent

Number of Cases of Calm 2790

Number of Cases of VRBL 248

A — Indicates an Occurrence Fraction of
Less Than 0.01 Percent

the diffusion within 5 miles (8 km) of the FSF site. However, the terrain would have an effect on the wind field due to channelling around the terrain. This effect is not apparent in the speed component of the wind field.

2.3.3 Onsite Meteorological Measurements - Programs

The Hanford meteorological data have been gathered since 1946. The procedural methodology and equipment utilized are defined by "Climatology of the Hanford Area."⁽³⁾

The 408-ft (124 m) Hanford meteorology tower, 16 miles (26 km) west-northwest of the FSF, is the primary source of meteorological information (temperature, temperature difference, humidity, wind direction, and velocity) for the Hanford complex. It also furnishes standard surface observations. An anemometer, installed in 1971, is 13 ft (4 m) above a small knoll that is 1000 ft (305 m) east-northeast of the FFTF containment vessel; it provides wind velocity and wind direction data for the site. In addition, there is a 300-ft (91 m) meteorological tower, about 3 miles (5 km) north of the FSF, for correlation purposes.

There are plans being formulated to increase the Hanford meteorological effort to provide a plume trajectory analysis capability. It is planned to include the FFTF surface wind speed and direction data as input into this system.

2.3.4 Short-Term (Accident) Diffusion Estimates

The distances that will be used for the atmospheric dilution factor calculations are 1.5 and 4.5 miles (2.4 and 9.2 km).

Atmospheric dilution factor calculations, for the cited distances, have been made in accordance with procedures suggested in Reference 2. The results are tabulated in Table 2.3-12.

TABLE 2.3-12

ATMOSPHERIC DIFFUSION FACTORS^a

<u>Distance</u>	<u>Time</u>	<u>X/Q(sec/m³)</u>
1.5 mi (2.4 km)	0-8 hrs	1.5×10^{-4}
4.5 mi (7.2 km)	0-8 hrs	3.7×10^{-5}
1.5 mi (2.4 km)	8-24 hrs	4.0×10^{-5}
4.5 mi (7.2 km)	8-24 hrs	8.0×10^{-6}
1.5 mi (2.4 km)	1-4 days	1.4×10^{-5}
4.5 mi (7.2 km)	1-4 days	2.7×10^{-6}
1.5 mi (2.4 km)	4-30 days	3.0×10^{-6}
4.5 mi (7.2 km)	4-30 days	5.8×10^{-7}

^aCalculated in accordance with Reference 2.

2.3.5 References for Section 2.3

1. AEC Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors," Rev. 2, US Atomic Energy Commission, Washington, DC, June 1974.
2. AEC Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors," Rev. 2, US Atomic Energy Commission, Washington, DC, June 1974.
3. W. A. Stone, D. E. Jenne and J. M. Thorp, Climatology of the Hanford Area, BNWL-1605, Pacific Northwest Laboratories, Richland, WA, June 1972.
4. E. L. Phillips, Washington Climate for the Counties Asotone, Benton, Columbia, Franklin, Garfield and Walla Walla, EM-3127, Washington State University, Pullman, WA, January 1970.
5. T. T. Fujita, Estimate of Maximum Wind Speeds of Tornadoes in Three Northwestern States, University of Chicago, IL, December 1970.
6. D. E. Simpson, Review of Tornado Considerations for FFTF, HEDL-TME 71-35, Hanford Engineering Development Laboratory, Richland, WA, January 1971.
7. G. C. Holzworth, Mixing Heights, Wind Speeds, and Potential for Urban Air Pollution Throughout the Contiguous United States, U.S. Environmental Protection Agency, Washington, DC, January 1972.

2.4 HYDROLOGIC ENGINEERING

2.4.1 Hydrologic Description

The principal streams in the area are the Columbia River and one of its tributaries, the Yakima River, with drainage areas of 169,000 and 5,600 mi² (4.4×10^5 and 1.4×10^4 km²), respectively. The Columbia River is extensively developed for hydroelectric power and irrigation supply upstream of the Hanford Reservation area and for hydroelectric power and navigation in a downstream direction. With its tributaries, the Columbia represents one-third of the potential hydropower of the United States. Its annual discharge of 180,000,000 acre-feet (1 a-ft = 43,560 ft³) (2.2×10^{11} m³) makes it the largest of all rivers flowing into the Pacific Ocean from North America. The Yakima is a minor stream with a mean flow of 3,240 ft³/s (92 m³/s)⁽¹⁾ and which has undergone extensive development for irrigation supply. The Columbia has a mean flow (in the Hanford area) of 114,000 ft³/s (3200 m³/s). The average peak flow for the Columbia River was 346,000 ft³/s from 1960 to 1973, Reference 11.

2.4.1.1 Site and Facilities

The FSF is located on a desert hillock about 4-1/2 mi (7.2 km) from and 210 ft (64 m) above the Columbia River. The FSF utilizes natural draft heat exchanger for rejection of decay heat. Water requirements are modest, limited to potable and sanitation needs of the operating personnel. The dry waste heat resulting from operations will be discharged to the atmosphere where it will quickly dissipate.

2.4.1.2 Hydrosphere

There are a number of multipurpose dams on the Columbia River and its major tributaries upstream of the site. The associated reservoirs provide an active storage of 49,690,000 a-ft (6.1×10^{10} m³). The largest reservoirs

are: Mica (12,000,000 a-ft) ($1.5 \times 10^{10} \text{ m}^3$); Arrow (7,100,000 a-ft) ($8.8 \times 10^9 \text{ m}^3$) in Canada; Libby (4,965,000 a-ft) ($6.2 \times 10^9 \text{ m}^3$); Hungry Horse (2,982,000 a-ft) ($3.7 \times 10^9 \text{ m}^3$); and Grand Coulee (5,232,000 a-ft) ($6.4 \times 10^9 \text{ m}^3$).

2.4.2 Floods

The flood control storage in upstream reservoirs for regulation purposes is 35,900,000 a-ft ($4.4 \times 10^{10} \text{ m}^3$) for the standard project flood (SPF) and 45,400,000 a-ft ($5.6 \times 10^{10} \text{ m}^3$) for the probable maximum flood (PMF). The FSF site elevation of 550 ft (170 m) above mean sea level is 210 ft (65 m) above the low water level (flow rate - 36,000 ft^3/s) (1020 m^3/s) of 340 ft (104 m) at River Mile 348 (due east of the site on the Columbia River).

The site elevation is 186 ft (57 m) above the Regulated SPF level of 364 ft (111 m). Research reveals that SPF has not been exceeded in recorded history. A safety margin greater than 160 ft (49 m) under conditions of a regulated PMF, indicates that flood protection is not required at the site.

2.4.2.1 Flood History

Columbia River floods are generally divided into two categories:

- 1) Spring floods caused by the melting snowpack usually in the upper reaches of the Columbia Basin east of the Cascades.
- 2) Winter floods caused by intense rain occasionally augmented by snow in the basins west of the Cascades. (These floods occur downstream of Hanford.)

The maximum flood of record at Hanford is that of June 1894, which resulted in a peak discharge of 740,000 ft^3/s ($9.1 \times 10^8 \text{ m}^3/\text{s}$) as estimated from the high water mark at Wenatchee, Washington.⁽²⁾ Because the flood of 1894 occurred before the establishment of upper Columbia River gaging stations,

its flow in the Hanford area was determined by relating a recorded high water mark to rating (stage-discharge) curves at Wenatchee, River Mile 465, and then routing the flood downstream by U.S. Corps of Engineers routing techniques. The largest recent flood, occurring in June 1948, had an observed peak discharge of $690,000 \text{ ft}^3/\text{s}$ ($8.5 \times 10^8 \text{ m}^3/\text{s}$) at Hanford. These floods were spring floods resulting from the melt of a large snowpack combined with the spring rain. For design of dams on the Columbia River, the U.S. Corps of Engineers has defined two hypothetical floods: the standard project flood (SPF) and the probable maximum flood (PMF).⁽³⁾ The SPF hydrograph represents the critical concentrations of runoff from the most severe combinations of precipitation and snowmelt that are considered reasonably characteristic of the drainage involved. The hypothetical characteristics (peak discharge, volume, and hydrograph shape) of the PMF are those that are considered to be the most severe reasonably possible at a particular location, based on a relatively comprehensive hydrometeorological analysis,^(3,4) critical runoff producing precipitation, snowmelt, and hydrological factors favorable for maximum flood runoff. Both the SPF and PMF for the Columbia River were considered to be caused primarily by snowmelt over an extended period of two to three months, but with significant runoff from storm rainfall. They are similar to the floods of 1894 and 1948, except that the severity of the meteorological conditions was increased.

At River Mile 348 the SPF and PMF flows and water levels are as follows:⁽⁵⁾

- 1) Standard Project Flood (SPF): Natural - $740,000 \text{ ft}^3/\text{s}$ ($9.1 \times 10^8 \text{ m}^3/\text{s}$) flow; water level at 369 ft (msl) (113 m).
Regulated - $570,000 \text{ ft}^3/\text{s}$ ($7.0 \times 10^8 \text{ m}^3/\text{s}$) flow; water level at 364 ft (msl) (111 m).
- 2) Probable Maximum Flood (PMF): Natural - $1,600,000 \text{ ft}^3/\text{s}$ ($2.0 \times 10^9 \text{ m}^3/\text{s}$) flow; water level (msl) at 389 ft (119 m).
Regulated - $1,440,000 \text{ ft}^3/\text{s}$ ($1.8 \times 10^9 \text{ m}^3/\text{s}$) flow; water level (msl) at 386 ft (118 m).

2.4.2.2 Flood Design Consideration

No special flood protection facilities are provided for the FSF site. The facility is safe from any imaginable flood. Predicated water levels at River Mile 348 are representative of potential conditions at the site level (which is about 550 ft (168 m); thus a safety margin of 160 ft (49 m) would prevail under PMF conditions.

2.4.3 Potential Dam Failure

The failure of upstream dams is considered only as a hypothetical event; the probability of such events resulting from such natural causes as earthquakes has not been established.

Since 1951, several studies of flooding associated with artificial flood considerations for the Columbia River Dams have been made by the U.S. Corps of Engineers^(6,7,8) and New York University.⁽⁹⁾ These studies indicate that the limiting discharge is at the Chief Joseph Dam resulting from a DBE failure of the Grand Coulee Dam. The maximum flood expected from such a dam failure would not be any greater than the Probable Maximum Flood (PMF) of 1,440,000 ft³/s (1.8×10^9 m³/s) hypothesized for the most severe combination of meteorological and hydrological conditions that are reasonably possible in that region. The studies indicate at least a 160-ft (49-m) margin between the water level and site evaluation.

2.4.4 Groundwater⁽¹⁰⁾

The DOE, through its contractors, has drilled more than 2,200 wells in the Hanford Reservation area and presently monitors approximately 255 wells for contamination.⁽¹²⁾ Figure 2.4-1 displays well locations in the Hanford Reservation as of about January 1969. In addition, DOE monitors approximately 400 additional wells in the surrounding area. In general, the water table is highest in the southwest, near Rattlesnake Mountain, and

FIGURE 2.4-1. Water Table Map.

decreases in elevation toward the 200 Areas in the center of the reservation; from the 200 Areas the general decrease in elevation is to the northeast and southeast.

Groundwater below the FSF site flows from west to east toward the Columbia River. Test borings at the FFTF site have determined depth to groundwater to be 170 to 175 ft (52 to 54 m) below grade. The bottom of the vault floor, which is the lowest projection of the FSF, is about 130 ft (40 m) above the present groundwater level.

A typical Hanford groundwater contour map is shown on Figure 2.4-1. A predicted Hanford groundwater contour map, based on construction of the proposed Ben Franklin Dam at approximately River Mile 348, is illustrated by Figure 2.4-2; the groundwater level at the site would raise about 20 ft (6 m) if the dam were constructed.

2.4.5 References for Section 2.4

1. Water Resources, Appendix V, Volume I, Pacific Northwest River Basins Commission, Vancouver, WA, April 1970.
2. V. W. Wood, A Summary of Columbia River Hydrographic Information Pertinent to Hanford Works - 1894 to 1954, HW-30347, Hanford Engineering Development Laboratory, Richland, WA, February 1954.
3. Memorandum Report Columbia River Basin Lower Columbia River Standard Project Flood and Probable Maximum Flood, U.S. Army Corps of Engineers North Pacific Division, Portland, OR, September 1969.
4. Snow Hydrology, U.S. Army Corps of Engineers North Pacific Division, Portland, OR, 1956.
5. Letter, from U.S. Army Corps of Engineers North Pacific Division to V. C. St. Clair, Atomic Energy Commission, Richland (WA) Office, dtd 2 November 1970.
6. Artificial Flood Possibilities on the Columbia River, U.S. Army Corps of Engineers Washington District, Washington, DC, November 1951.
7. Artificial Flood Considerations for Columbia River Dams, U.S. Army Corps of Engineers Seattle District, Seattle, WA, August 1963.

8. Artificial Flood Considerations for Columbia River Below Chief Joseph Dam to Richland, Washington, U.S. Army Corps of Engineers North Pacific Division, Portland, OR, January 1968.
9. E. Isaacson, Calculation of Severe Floods in a Developed River, New York University, New York, NY, January 1965.
10. R. C. Newcomb, J. R. Strand and F. J. Frank, Geology and Groundwater Characteristics of the Hanford Reservation of the U.S. Atomic Energy Commission, U.S. Geological Survey Professional Paper 717, Washington, DC, 1972.
11. J. N. Paglieri, 300 Area Site Description, HEDL-TME 75-13, Hanford Engineering Development Laboratory, Richland, WA, February 1975.
12. Hanford Wells, BNWL-2296, Pacific Northwest Laboratory, Richland, WA, June 1977.

2.5 GEOLOGY AND SEISMOLOGY

2.5.1 Basic Geologic and Seismic Information

The geology of the Hanford region is generally well documented and is specifically described in Blume.⁽¹⁾ This section summarizes the seismic and geologic characteristics of the site.

2.5.1.1 Regional Geology

The FSF is located on a bench believed to be a remnant of giant sand and gravel deposits of an ancient Columbia River in the late Pleistocene period. The bench is in the Hanford valley which is itself the northwest portion of the Pasco Basin, an important physiographic and structural feature of the Columbia Plateau. The Hanford valley is bounded by the Rattlesnake Hills to the southwest, Yakima and Umtanum Ridges to the west, Saddle Mountains on the north, and White Bluffs, which also form the east bank of the Columbia River, on the east. Figure 2.1-2 is a topographic map of this area.

The Pasco Basin can be traced to a tremendous outpouring of lava during the Miocene period. About 10 million years ago the volcanism ceased, the Pasco Basin began to subside, and folding of the basalts was initiated. The slowly rising Horse Heaven Hills anticline and the subsidence of the Pasco Basin during the Pliocene period caused damming of the Columbia River and deposition of the Ringold formation. During the Pleistocene period, high floods occurred as the glaciers retreated. These floods scored vast regions of northeastern Washington, creating glaciofluvial deposits across the Hanford Reservation, finally breaching the Horse Heaven Hills anticline at Wallula Gap. The basin surface is now covered with irregular, poorly developed, or subdued dunes of wind-blown sand 5 to 15 ft (2 to 5 m) in depth. The dunes are stabilized by sparse growths of sagebrush and cheatgrass.

A tectonic map is included as Figure 2.5-1; from this it may be seen that the closest potentially active fault is the Rattlesnake-Wallula Fault about 9 miles (14 km) southwest of the site. A subsurface exploration confirmed that there were no hidden faults of significance to the FSF.⁽²⁾

2.5.1.2 Site Geology

The FSF is located upon the glaciofluvial deposits at an elevation of 550 ft (msl). The bottom of the vault door, the lowest FSF feature, is at a depth of about 30 ft (9 m) (520 ft (160 m) msl). The glaciofluvial deposits at the site are underlain by sediments of the Ringold formation from about 400 ft (120 m) elevation to basalt at about 50 ft msl. The thickness of the basalt exceeds 10,000 ft (3050 m); however, the total thickness is unknown.

Excavations for FFTF structures revealed "sand dikes"⁽³⁾ which are nearly vertical seams of elastic sedimentary material. These dikes are common to the area and are attributable to surface cracking and filling during the ice ages.⁽⁴⁾ The excavations provided no evidence of soil displacements which might be associated with seismic activity.

2.5.2 Vibratory Ground Motion

Seismic investigations and evaluations for the FFTF and other Hanford area activities have been summarized and reported by J. A. Blume and Associates, Engineers of San Francisco, California.^(1, 5, 6)

The Pasco Basin has historically been an area of low seismic activity with recorded shocks of below Richter Magnitude 4.0.

The two largest historical earthquakes in the region were the Umatilla, Oregon, quake in 1893 with estimated Modified Mercalli (MM) intensity VII located 33 miles (53 km) southeast of the site, and the Milton-Freewater quake in 1936 of MM VII which was located 57 miles (91 km) southeast of the site. Calculations indicate that the maximum ground acceleration at the site caused by the Umatilla, Oregon earthquake was 0.03 g.⁽¹⁾ Similarly, the Milton-Freewater quake was calculated to have produced an acceleration at the site of 0.02 g. or less.

In a conservative interpretation of data, the activity of the Rattlesnake-Wallula fault has not been ruled out. Therefore, this fault is considered to be a potential generator of earthquakes. The fault is shown on the tectonic map of Figure 2.5-1. Field investigations by J. A. Blume geologists of the site,⁽²⁾ which include a line of hills near Horn Rapids and the area in the vicinity of Badger Coulee,⁽⁵⁾ and a detailed literature review⁽¹⁾ indicated that there are no significant faults closer to the site. Examination of historic records and calculation of peak ground motion possibly felt at the site from earthquakes which have occurred within 200 miles (320 km) of the site showed that an earthquake on the Rattlesnake-Wallula fault would cause maximum ground motion at the site. The design basis earthquake (DBE) was therefore located on this fault.

Comprehensive site geological and seismological studies have established a design basis earthquake (DBE) of Richter Magnitude 6.8, located about

9.5 miles (15 km) from the site and at a focal depth of 6 miles (10 km). Geophysical field measurements were made to establish representative geotectonic properties at the site. These site properties, together with the DBE magnitude and distance from the site, were used to calculate the maximum ground acceleration that would be induced at the site if the DBE should occur. Several methods of calculating the maximum ground motion were utilized. The calculated results were then considered in conjunction with the seismic history of the area. It was concluded that a reasonably conservative prediction of maximum horizontal ground motion that would be induced at the site by a DBE occurring on the Rattlesnake Hills-Wallula fault is 0.25 g.

The response spectra for the DBE is considered to be for free-field ground motion. The shape of the DBE response spectra was developed after careful consideration of the geotectonic properties of the site, the proximity and focal depth of the DBE, and other related factors.⁽⁶⁾ The vertical ground motion response spectra are considered to be equal to two-thirds of the horizontal response spectra.

2.5.3 Surface Faulting

A reconnaissance geologic survey of the area within 5 to 10 miles (8 to 16 km) of the FSF site sought evidence of faulting near the site. With the exception of one minor fault, no known surface rupture or displacement by faulting was observed.⁽⁵⁾

The minor fault was observed in the Yakima River gap, striking N 20°E, and would pass within 3 miles (5 km) of the site at its nearest point if it continued northward beyond the slopes of Rattlesnake Mountain. No evidence was found to indicate that it does continue northward. Therefore, it does not constitute a surface rupture hazard to the plant.

The reconnaissance survey was followed by detailed geologic examination of the Horn Rapids Lineament. No evidence of faulting which might pass beneath or close to the FFTF was found.⁽⁷⁾

2.5.4 Stability of Subsurface Materials

A soil investigation⁽⁷⁾ concluded that the structures can be supported on the dense glaciofluvial sands or on fills constructed from these sands. The geologic investigations⁽²⁾ near the site indicated that the basalt surface is topographically subdued and showed no evidence of faulting. The dense silt and clay of the Ringold Formation forms a bedrock-like stratum at about 400 ft (120 m) elevation. The dense glaciofluvial deposits above this level are the subject of the Soil Investigation Report.⁽⁷⁾ Significantly, all tests made in situ in the FFTF reactor containment building excavation exceeded 80% relative density. As previously stated, the groundwater is within the Ringold Formation at 387 ft (msl) (118 m) and could be 407 ft (msl) (124 m) if the Ben Franklin Dam were constructed. A liquefaction analysis is presented in the Soil Investigation Report.⁽⁷⁾ This analysis concludes that a factor of 1.6 exists against liquefaction.

The next allowable bearing pressure determined for 1 in. total settlement for shallow foundations is 14.2 kips/ft² (6.8×10^5 Pa) for spread footings 5 ft (1.5 m) wide, and 10.8 kips/ft² (5.2×10^5 Pa) for mats 20 ft (6.1 m) wide or larger. For deep foundations, the corresponding values are 28.4 and 21.6 kips/ft² (1.4×10^6 and 1.0×10^6 Pa).

2.5.5 Slope Stability

The Soil Investigation Report⁽⁷⁾ determined the angle of internal friction to be 39.5° at 80% relative density. Permanent excavations have slopes of 2.5 to 1 (21.8°) and temporary excavations 1.5 to 1 (33.7°); thus, these slopes are stable. The closest natural slopes of significance, which are higher than the FSF, are 5 miles (8 km) distant and on the far side of

natural channels - the Columbia River and Cold Creek. Thus, the site will not be affected by landslides.

2.5.6 References for Section 2.5

1. Seismic and Geologic Siting Evaluation, Fast Flux Test Facility, JABE-WADCO-01, Hanford Engineering Development Laboratory, Richland, WA, August 1970.
2. Subsurface Geologic Investigations for the FFTF Project in Pasco Basin, JABE-WADCO-07, Hanford Engineering Development Laboratory, Richland, WA, October 1971.
3. A Discussion of Sand "Dikes" in the FFTF Evacuation (Bechtel Corporation), HEDL-TME 71-101, Hanford Engineering Development Laboratory, Richland, WA, June 1971.
4. R. L. Lupton, "Clastic Dikes of the Columbia Basin Region, Washington and Idaho," Bulletin of the Geological Society of America, 55, pp. 1431-1462, December 1944.
5. Supplementary Geologic Investigations for Seismic Evaluation of the FFTF Site Near Richland, Washington, JABE-WADCO-04, Hanford Engineering Development Laboratory, Richland, WA, February 1971.
6. A Summary Report of Seismic Evaluation and Development of Ground Acceleration and Response Spectra for FFTF Site, JABE-WADCO-03, Hanford Engineering Development Laboratory, Richland, WA.
7. Fast Flux Test Facility Final Safety Analysis Report, HEDL-TI-75001, Chap. 2, Hanford Engineering Development Laboratory, Richland, WA, December 1975.

CHAPTER 3
DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

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3.0 DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

3.1 FUEL STORAGE FACILITY (FSF) DESIGN

3.1.1 Quality Standards

The functions and requirements for the FSF are given in Section 1 of the FSF System Design Description.⁽¹⁾ The FSF has been designed to comply with the intent of 10 CFR 50.⁽²⁾ Each of the criteria has been examined, and a set of criteria has been established for the FSF. Sections 3.1.1 through 3.1.13, below, present these criteria along with a discussion of how the FSF design meets them. Other criteria which appear in 10 CFR 50, but not here, were examined and found to be applicable only to reactor design and not to fuel storage facilities. The Facility is shown in Figures 3.1-1, 3.1-2 and 3.1-3.

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

Discussion -- As a part of the FSF design effort, the systems and components important to safety were identified. This was done by researching the accident studies of nuclear facilities of a similar sort or which perform similar functions, and by an accident analysis of the FSF itself (refer to Chapter 15). Quality standards for the safety-related components of the facility were established based on the importance of the safety functions to

FIGURE 3.1-1. Fuel Storage Facility.

FIGURE 3.1-2. Ground Floor Plan.

GROUND FLOOR PLAN

FIGURE 3.1-3. Basement Plan.

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be performed. A quality assurance program (Chapter 17) assures that components, structures, and systems will satisfactorily perform their safety functions.

3.1.2 Design Bases for Protection Against Natural Phenomena

Criterion -- Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, and floods without loss of capability to perform their safety related functions. The design bases for these structures, systems, and components shall reflect: (1) appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated; (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena, and (3) the importance of the safety functions to be performed.

Discussion -- Design basis natural phenomena for the FSF were established based on studies conducted at the Hanford 400 Area for the FFTF. The results of these studies are summarized in Chapter 2 and include:

- 1) Earthquakes (Section 2.5, also References 1, 2, 7, and 8 of Section 2)
- 2) Tornadoes (Sections 2.3.1.2 and 2.3.1.2.1)
- 3) Hurricanes (Section 2.3.1.2.1)
- 4) Floods (Sections 2.4.2 and 2.4.3)

Safety-related components of the FSF are designed to either withstand these natural phenomenon design bases or are protected from them to prevent losing their ability to provide the safety function.

3.1.3 Fire Protection

Criterion -- Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Non-combustible and heat-resistant materials shall be used wherever practical throughout the facility. Fire detection and extinguishing systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Fire-fighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.

Discussion -- FSF components and systems have been designed to conform to the improved risk requirements of 5480.1, Chapter VII, Fire Protection. They have been constructed of noncombustible materials to provide fire protection for safety-related equipment and to assure that no fire can preclude decay heat removal or result in inadvertent criticality. Below-grade vaults for sodium containing components and physical separation of the two redundant heat-removal loops are examples of measures taken to protect safety-related systems. Leak detectors and product-of-combustion sensors are provided to warn of liquid metal spills or fires. A Halon system in the control room, a sprinkler system, and portable fire extinguishers throughout the rest of the facility are provided to deal with local fires. Water sprinkler systems are specifically excluded from operation in the high bay area because of large quantities of sodium in the storage vessel. Exterior walls are four-hour fire-rated, and below-grade concrete partitions preclude the spread of fire within the FSF. Fire and smoke dampers (activated by the smoke detection system) are provided in the stairwells and west entry room.

3.1.4 Environmental and Missile Design Bases

Criterion -- Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. These structures, systems and components

shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the facility.

Discussion -- Safety-related equipment in the FSF is designed to continue performing its safety function under extremes of operation including normal operation, upset conditions, and accident situations in which the proper operation of the equipment is necessary for plant safety. Missile protection is provided for safety-related equipment by locating such components in the below-grade areas or in the tornado-hardened natural draft heat exchange (NDHX) structures.

The heat removal capability of the NDHX is enhanced by backup fans and gasoline-driven generator that provide cooling air to the heat exchanger tube. The NDHX stacks are not designed to withstand either the design basis earthquake (DBE) or tornado missiles or loads.

3.1.5 Sharing of Structures, Systems and Components

Criterion -- Structures, systems, and components important to safety shall not be shared among units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions.

Discussion -- The FSF shares several safety-related systems with the FFTF. These are the Bottom Loading Transfer Cask (BLTC), BLTC power supply, electrical (offsite) power, fire protection services, water supply, and security arrangements. An analysis of possible accident situations in the FSF and/or FFTF indicates that neither facility will have a significant safety impact on the operation of the other facility. These conclusions were reached based on the Seismic I rating of the BLTC rail clamps, and BLTC power supply independent from FSF, the ability of the FSF to operate safely without electrical power or water supply, and the FFTF Emergency Plans which assure that sharing fire protection services and security services would not adversely affect protection at either facility.

3.1.6 Instrumentation and Control

Criterion -- Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions. Instrumentation shall be as appropriate to assure adequate safety and integrity of the heat removal system and coolant pressure boundary. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

Discussion -- The FSF is provided with instrumentation to verify that safety-related systems are fulfilling their functions. Temperature, pressure, and sodium and NaK level instrumentation provide information on the performance of the heat removal system. The radiation, leak, and fire detection instrumentation systems provide information relating to pressure boundary integrity, liquid metal spills, and other hazards.

3.1.7 Storage Vessel Pressure Boundary

Criterion -- The storage vessel pressure boundary shall be designed, fabricated, tested, installed, and operated so as to have an extremely low probability of leakage.

Discussion -- All components of the storage vessel pressure boundary conform to the quality standards of the ASME Boiler and Pressure Vessel Code,⁽³⁾ Section III, Class 2. In addition, a Section III, Class 3, guard vessel is provided to assure that no primary vessel leak can uncover the NaK coolers or the fuel region of the stored fuel. An estimate of the probability of coolant leakage from the storage vessel/guard vessel combination is less than one occurrence per million years (Chapter 15).

3.1.8 Decay Heat Removal System Design

Criterion -- The decay heat removal system and associated auxiliary, instrumentation, and control systems shall be designed with enough margin to

assure that the design conditions of the storage vessel pressure boundary are not exceeded during any condition of normal operation including anticipated and unlikely events. See Chapter 15 for definition of these terms.

Discussion -- The individual natural convection heat removal loops meet this requirement at the rated heat load. The availability of two physically independent loops enables the heat removal system to meet this criterion even under single failure conditions.

3.1.9 Control

Criterion -- Control capability shall be provided to assure safe operation of the facility and to maintain it in a safe condition even under accident conditions. Adequate radiation protection shall be provided to permit control of the facility under accident conditions without personnel receiving radiation exposure in excess of 5 rem whole body or equivalent to any part of the body for the duration of the accident.

Discussion -- The only control function required for operation of the FSF is adjustment of the NDHX dampers that control the rate at which cooling air passes over the heat exchanger tubes. These are normally adjusted remotely from the control room but can be operated manually in the NDHX rooms. The NDHX rooms are earthquake, tornado, and missile hardened, and access to them is from the exterior of the FSF building. None of the postulated FSF accidents cause doses in excess of 700 mrem in the FSF or 10 mrem off site. (Refer to Chapter 15). The FSF is also provided with interlocks that prevent damage to fuel assemblies during fuel handling operations: with the BLTC grapple in the down position the turntable will not rotate nor will the floor valve close, and the BLTC grapple will not lower unless the floor valve is open.

3.1.10 Decay Heat Removal System Redundancy

Criterion -- The decay heat removal system shall provide suitable redundancy in components and features to assure that for a loss of electric power incident the system can function assuming a single failure.

Discussion -- Each of the two natural circulation heat removal loops is designed to remove the design heat load without electric power. These two loops are redundant and completely independent of each other.

3.1.11 Control of Releases of Radioactive Material to the Environment

Criterion -- The facility design shall include means to suitably control the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal facility operation including anticipated and unlikely occurrences.

Discussion -- The FSF is provided with a gaseous radwaste system consisting of a holdup tank and suitable radiation monitoring equipment. Activity of gas to be released is measured in the holdup tank to assure that the levels are suitable for release prior to allowing release. There is no significant source of radioactive liquids that might be discharged. Amounts of solid radwaste will be small over the facility life and they will be incorporated into the FFTF solid waste system, which is designed for near zero release.

3.1.12 Prevention of Criticality

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

Discussion -- The storage vessel turntable is designed to ensure a subcritical configuration for the stored fuel assemblies. The design requirement states that the k_{eff} shall be 0.95 or less for all credible conditions of loading,

operation, or accident. The FSF design meets these requirements by a combination of (1) administrative control to assure that only the proper type of fuel assemblies and canisters are loaded, (2) positive physical spacing to assure that the designed center-to-center distances between fuel assemblies (9 in. minimum) are maintained, (3) an annular B_4C ring placed adjacent to the fueled region of the stored assemblies which serves as a nuclear poison and is effective in reducing reactivity, and (4) the sealing of the access port to ring 5 until reactivity levels on an "as-loaded" basis within the other nine rings is established. Calculation of the k_{eff} for the FSF using the most reactive loading configurations and adding 95% probable uncertainty gives a k_{eff} less than 0.95. Since the FSF is designed to be inherently subcritical under all credible conditions, the 10 CFR 50⁽²⁾ paragraphs relating to reactivity or criticality control "processes" do not apply. See Chapter 15 for discussion.

3.1.13 Monitoring Radioactivity Releases

Criterion -- Means shall be provided for monitoring occupied areas, effluent discharge paths, and the facility environs for radioactivity that may be released for normal operations including anticipated operational occurrences, and from postulated accidents.

Discussion -- Area radiation monitors are in the process areas of the FSF and in the gaseous effluent line. The environment radiation monitoring system associated with the FFTF reactor is utilized to monitor the FSF environs.

3.2 CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

3.2.1 Seismic Classification

FSF structures, systems, and components important to safety are designed to retain their safety related function in event of a design basis earthquake (DBE). They are classed as Seismic Category I and include but are not limited to those structures, systems, and components that are necessary to:

- 1) Physically retain the stored fuel assemblies in a noncritical array.
- 2) Maintain the cladding of the fuel pins below the 1500°F (816°C) assumed threshold temperature of failure.
- 3) Prevent uncontrolled release of any potentially radioactive cover gas.
- 4) Maintain the liquid metal temperatures within the faulted limits for the storage vessel and heat removal system components (Table 4.2-1).

A list of seismic categories is shown in the FSF System Design Description (Reference 1). These components, systems, and structures were selected for the Category I classification, based on the intent of Regulatory Guide 1.29,⁽⁴⁾ to provide reasonable assurance that the FSF would not present an unacceptable hazard in the event of a DBE. Other FSF structures and nonsafety related equipment are designated Category III and designed to the Uniform Building Code⁽⁵⁾ Zone 2 standards for industrial facilities.

3.2.2 System Quality Group Classification

The FSF systems, components, and structures are classified as Safety Class 2, Safety Class 3, or nonsafety class (SC-2, SC-3, or NSC) depending upon the importance of the safety function provided. These assignments are made according to Reference 6 and 7. Shown below are the relationships of safety classes to code classes. Refer to Reference 1 for additional component information.

RELATIONSHIP OF SAFETY CLASSES TO CODE CLASSES

<u>Safety Class</u>	<u>ASME B&PV Section III Code Class</u>
SC-2	2
SC-3	3

3.3 WIND AND TORNADO LOADING

All of the structures in the FSF are designed to withstand the design basis winds. The portions of the facility that are necessary for safe storage of the fuel are also designed to withstand the design basis tornado (including tornado projected missiles). These portions include the storage vessel, heat removal loops and the cover gas system. They are also designed as Seismic Category I items. The design basis winds and tornados for the site are discussed in Chapter 2.

3.3.1 Wind Loadings

3.3.1.1 Design Wind Velocity

The design wind velocity for the FSF is 87 mph (39 m/s) for heights 0-30 ft (0-9 m) above ground with a 1.1 gust factor. The recurrence interval for this design wind is 100 years.

3.3.1.2 Gust Factor

Gust factor and increase of wind velocity with height above grade are in accordance with Reference 8. The gust factor is taken as 1.1 in accordance with Recommendation 8 of Reference 8, because the geometry of the FSF complex is such that wind blown against it in any direction will exceed a horizontal dimension of 125 ft (38.1 m) transverse to the wind.

Increase of design wind velocity with height is in accordance with EQ(z) of Reference 8 as follows:

$$V_z = V_{30} \left(\frac{z}{30} \right)^x, \quad \begin{array}{l} x = 0.25 \text{ (Fig. 3)} \\ z = 50 \end{array}$$

<u>Elevation</u>	<u>Wind Velocity</u>
550 ft 0 in. to 580 ft 0 in. (168 m to 177 m)	87 mph (39 m/s)
580 ft 0 in. to 600 ft 0 in. (177 m to 183 m)	99 mph (44 m/s)

3.3.1.3 Determination of Applied Forces

Calculation of pressures (equivalent static loads) on structures resulting from design wind velocities is as follows (Reference 8):

$$q = 0.002558 V^2$$

$$p = q \times G_F \times C_D$$

in which q is the dynamic pressure in pounds per square foot, V is the design wind velocity in miles per hour, G_F is the gust factor considered as 1.1, C_D is the drag coefficient and p is the average pressure on the projected area of the building in pounds per square foot. Again, from Reference 8, the values selected for C_D are as follows:

- 1) Windward walls; $C_D = 0.9$
- 2) Leeward Walls; $C_D = 0.5$ (suction)
- 3) Flat roofs; $C_D = 0.6$ (suction)
- 4) Building; $C_D = 1.4$

The equivalent static loads, p , calculated for the exposed parts of various buildings at different elevations are summarized below:

- 1) Building below Elevation 580 ft 0 in. (177 m), $p = 30 \text{ lb/ft}^2$
(1436 Pa)
- 2) Building above Elevation 580 ft 0 in. (177 m), $p = 39 \text{ lb/ft}^2$
(1867 Pa)

- 3) Roofs below Elevation 580 ft 0 in. (177 m), $p = 13 \text{ lb/ft}^2$ (622 Pa)
- 4) Roofs above Elevation 580 ft 0 in. (177 m), $p = 17 \text{ lb/ft}^2$ (814 Pa)

3.3.2 Tornado Loadings

The structures designed for tornado wind loads include the NDHX rooms and the inherently hardened below-grade portions of the FSF. These provide protection for Category I components including the storage vessel, heat removal loops, holdup tank and associated piping and equipment.

3.3.2.1 Design Parameters

3.3.2.1.1 Design Tornado Wind Velocity

The design tornado as specified in Chapter 2, Section 2.3.1.2.1 is 175 mph (78 m/s), made up of 150 mph (67 m/s) rotational velocity and 25 mph (11 m/s) translational velocity. This combined velocity is applied uniformly over the full exposed height of the structures.

3.3.2.1.2 Design Transient Pressure Differential

A design pressure transient is 0.75 lb/in.^2 (5200 pa) reduction in atmospheric pressure in 3 seconds, followed by a return to the original pressure at the same rate as specified in Chapter 2, Section 2.3.1.2.1.

3.3.2.2 Determination of Forces on Structures

3.3.2.2.1 Wind Pressure Calculation, Load Combination and Stress Limits

Calculation of Wind Pressures -- Pressures (equivalent static loads) on structures resulting from tornado wind velocity are calculated using the following equations from Reference 8.

$$q = 0.002558 V^2$$

$$p = q \times C_D$$

where q and p are as defined previously. V is the design tornado wind velocity in miles per hour, i.e., $V = 175$. C_D is the drag coefficient as given in Section 3.3.1.3.

Applicable Load Combination and Stress Levels -- Tornado loadings were factored into FSF design in the following combinations:

- 1) Category I Steel Structures (Closure Head)
- 2) Category I Reinforced Concrete Structures

$$U = 1.10 (D+L) + T_o + W' + L_p$$

$$\text{or } U = 0.90 D + T_o + W' + L_p$$

where:

U = required strength of concrete members

D = dead loads

L = live loads

T_o = thermal effects (operating conditions)

W' = tornado loads

L_p = operating loads on structures

3.3.2.3 Ability of Hardened Structures to Perform Despite Failure of Structures Not Designed for Tornado Loads

The design of all permanent structures, systems, and components not designed for tornado loadings were analytically checked to minimize the chance that missiles will be generated that have more severe effects than those listed in Section 3.5.3. This provided assurance that structures, systems, and

components required for continued safe storage after a tornado will perform their intended functions. (See Reference 9.)

Nonhardened structures whose collapse could result in loss of required structures, equipment, or systems, were analytically determined not to collapse when subjected to extreme environmental loads.

The building shell (roof deck, exterior wall, siding) can fail without collapsing the high bay crane supporting structure when subjected to tornado wind pressure and missile impact. The heat exchanger rooms are tornado hardened.

1) The following structural elements are tornado hardened:

- a) high bay columns
- b) roof girders
- c) crane runway beams
- d) heat exchanger room roof
- e) heat exchanger room exterior walls

2) The following structural elements do not require tornado hardening:

- a) roof beams and purlins, except as in 3 b) below
- b) roof deck
- c) wall siding
- d) girts
- e) columns not in high bay

3) The following structural elements are not tornado hardened but are designed for tornado wind pressure:

- a) high bay roof bracings, struts, and chords
- b) high bay wall bracings and struts

Any one of the bracings may fail due to missile impact without endangering the overall structural stability of the building, since the remaining structure is adequate for the redistributed loads.

The following design features have been incorporated in the FSF to assure that the capability for continued safe storage is not lost.

- 1) Backup fans and power supplies are provided and situated in a hardened area to assure the capability for continued operation of the heat removal system in the event that the NDHX stacks are damaged by a tornado.
- 2) The closure head of the storage vessel assembly is designed to withstand impact from tornado generated missiles and falling debris from the above grade non-Category I structures.
- 3) The high bay crane is designated Category I and designed to withstand the design basis tornado in order to prevent collapse and possible impact with the storage vessel closure head.
- 4) The heat removal system is designed to be capable of maintaining storage vessel temperatures within safe limits in the event of loss of the vault cooling system (Refer to Chapter 15).

3.4 WATER LEVEL DESIGN

A design flood is not applicable for FSF, since the deepest foundation is approximately 125 ft (38 m) above the maximum probable flood.

<u>Plant Elevation/Event</u>	<u>Elevation Ft (Meters) Above MSL</u>
Nominal plant elevation	550 (168 m)
Deepest FSF foundation	514 (157 m)
Maximum probable flood (Section 2.4)	386 (119 m)

Accordingly, no special account is taken of flooding in the FSF structural design.

3.5 MISSILE PROTECTION

3.5.1 Missile Barriers and Loading

Missile barriers are designed to protect FSF components necessary for continued safe storage from hazardous effects of internally or externally generated missiles. The missile barriers provided in the FSF are as follows:

<u>Structure</u>	<u>Missile Barrier</u>	<u>Remarks</u>
NDHX Rooms	Hardened concrete construction of the walls and missile rated steel louvered air inlets and NDHX stack missile grilles consisting of a pair of square grids built of 1/4" x 6" steel plate set 6 in. on centers. (See Figure 3.5-1)	All components inside NDHX rooms are protected
Below grade vaults	Concrete floor at elevation 550 ft (168 m) and the striker plate over vessel closure head from the upper hardened boundary.	All areas below elevation 550 ft except stair openings are protected

The missile barriers are designed to withstand the design basis tornado load within the elastic range of the material and the impingement load effects of the design missiles postulated in Section 3.5.3.

3.5.2 Missile Selection

3.5.2.1 External Missiles

Falling objects generated from commercial, industrial, or military activities near the plant site are not included as a design basis. All commercial and

FIGURE 3.5-1. NDHX Stack W/Tornado Grill.

industrial activities on the Hanford Reservation are administratively controlled by DOE. The nearest military installation is the Yakima Firing Range, 45 miles (72.42 km) west-northwest of the plant site. This is not a guided missile facility, and no missile hazard is postulated for the FSF. A detailed description of local air traffic is provided in Chapter 2.

One FSF plant component was identified and analyzed as a potential external missile hazard to the FSF. It is the dump heat exchanger fan rotor. The fan is a double width, double inlet centrifugal fan with a maximum speed of 1200 rpm. Design analysis has shown that the maximum stress at any point in the fan rotor assembly is 17,000 psi (117 MPa).

This provides a 76% safety margin to the design maximum stress of 30,000 lb/in.² (1.4×10^6 Pa) and a 194% safety margin to the material yield stress of 50,000 lb/in.² (2.4×10^6 Pa). The fan is not expected to provide an identifiable external missile hazard to the FSF.

The external missiles considered are debris associated with the 175 mph (71 m/s) design tornado. Heavy, non-aerodynamic objects such as cars or pieces of pipe cannot be lifted by the 150 mph (67 m/s) rotational wind. A heavy plank such as 4 in. x 12 in. (0.1 m x 0.3 m) probably would not become airborne and in any case could not attain high speed. Lightweight, large surface area objects, such as plywood sheets or sheets of steel siding, however, could be lifted and projected at relatively high speed. Intermediate objects such as 2 in. (0.05 m) planks might become airborne but at a lesser projected velocity than the 175 mph (78 m/s) design velocity.

3.5.2.2 Internal Missiles

There are no large rotating components in the plant. High-pressure gas bottles are built to ICC requirements and racked to prevent damage. Therefore, no internal missiles resulting from such systems are identified as a design basis.

3.5.3 Selected External Missiles

A summary of the design basis missiles that result from tornado considerations follows:

<u>Missile</u>	<u>Dimensions</u>	<u>Weight (lb&kg)</u>	<u>Velocity (mph&m/s)</u>	<u>Status</u>	<u>Missile Energy $E=1/2MV^2$</u>
Wood Plank	2 in. x 12 in. x 12 ft (0.05 m x 0.3 m x 3.6 m)	63 (29 kg)	100 (45 m/s)	Traveling end-on at any height	21.2×10^3 ft-lb (28.7×10^3 J)
Plywood Sheet	3/4 in. x 4 in. x 8 ft (0.02 m x 0.1 m x 2.4 m)	80 (36 kg)	150 (67 m/s)	Traveling end-on at any height	60.1×10^3 ft-lb (81.5×10^3 J)
Corru- gated 20-Gauge Steel Siding	26 in. x 20 ft (0.66 m x 6.1 m)	71 (32 kg)	150 (67 m/s)	Traveling end-on at any height.	53.3×10^3 ft-lb (72.2×10^3 J)

3.5.4 Barrier Design Procedures

The missile barriers are designed to withstand the tornado wind loads and the effects of missile impact according to the following procedures:

- 1) Tornado Wind Loads - The tornado wind force on the barrier structure is considered as a static load and calculated on the basis of a 175 mph (78 m/s) horizontal wind, consisting of 150 mph (67 m/s) rotational velocity and 25 mph (11 m/s) translational velocity components, applied over the full height of the barrier structure.

The applicable portion of wind design methods described in Reference 8 is used. Load combinations along with stress criteria for the tornado wind load on the barrier structure are given in Reference 8.

- 2) Missile Impact - All missile barriers were evaluated to ensure adequate protection against the design missiles listed above.

3.5.5 Missile Barrier Features

The missile barriers in the FSF are designed to withstand impacts of the selected missiles identified in Section 3.5.3.1.

Refer to Reference 1 for a description of the features of missile barriers such as the striker plate, closure valve, etc.

3.6 PROTECTION AGAINST DYNAMIC EFFECTS ASSOCIATED WITH THE POSTULATED RUPTURE OF PIPING

The piping systems in the FSF are generally low-pressure systems and therefore are not subject to severe dynamic effects in the event of rupture, with the exception of the argon supply bottles and instrument (compressor supplied) air system. The following pressures are the maximum unrelieved values for the systems tabulated.

Argon System	275 lb/in ² gage (1.90 x 10 ⁶ Pa)
Instrument Air System	150 lb/in ² gage (1.03 x 10 ⁶ Pa)
Storage Vessel and Purification Loop	6 lb/in ² gage and head (4 x 10 ⁴ Pa)
NaK Heat Removal Loops	15 lb/in ² gage and head (1 x 10 ⁵ Pa)
Below Grade Vaults	1 lb/in ² gage (7 x 10 ³ Pa)
Vault Cooling System Water/Glycol Loop	30 lb/in ² gage (2.1 x 10 ⁵ Pa)

3.7 SEISMIC DESIGN

3.7.1 Seismic Input

The input spectrum upon which the seismic analysis of FSF components and structures is based was determined by extensive analysis of the site geologic conditions, soil interactions, and effects imposed by nearby structures, notably the FFTF reactor and Reactor Service Building. The resultant foundation input spectrum derived for the FSF is in Figure 3.7-1. Further details of the seismic analysis are in References 10 and 11.

3.7.2 Seismic System Analysis

A time-history analysis was performed on a mathematical model of the FSF structure. From this analysis, response spectra for several building elevations were calculated for use in the analysis of FSF components. The seismic-induced stress components along with the mechanical and thermal stress components were determined so that the maximum primary and primary plus secondary stress intensities could be evaluated for comparison with the allowable stress in the components. In addition, nonseismic equipment in contiguous areas was evaluated for debris hazard to Seismic Category I equipment and the results were reported in Reference 9.

3.7.3 Seismic Subsystem Analysis

The major components of the Fuel Storage Facility were separately analyzed to verify compliance with applicable ASME code requirements.

- 1) The storage vessel assembly, including primary vessel, turntable, bearing, support ring, and closure head are designed as Seismic Category I components. The stress analysis, including details of the seismic model, is in Reference 11.

FIGURE 3.7-1. Computed Response - Vertical Excitation.

FIGURE 3.7-1A. Computed Response - Horizontal Excitation.

- 2) The NDHX and associated piping is similarly designed to withstand the DBE, and the details of that seismic analysis are in Reference 12.
- 3) Those portions of the pipings systems in the FSF that are required to withstand the DBE without loss of function were analyzed to determine their response to the seismic event, and the results are reported in Reference 13.
- 4) Seismic design of the Category I structures is addressed in Section 3.8.
- 5) The cold trap in the sodium purification system is designed to withstand the DBE. The seismic analysis is presented in Reference 14.

Table 3.7-1 presents a list of Seismic Category I components along with function type, qualification methods, and environment.

3.7.4 Seismic Instrumentation Program

Continued safe storage of spent fuel at FSF is assured by the design of passive components and engineered safety features. There are no specific safety-related control functions that must be initiated by the instrumentation and control systems in the event of a DBE. The onsite ground motion input spectrum will be monitored by the FFTF seismic monitoring instrumentation. No further seismic monitoring instrumentation is necessary for safe operation of the FSF.

3.7.5 Seismic Design Control

The quality of the FSF seismic design features is assured by the quality assurance program. Each step of the design process is attended by a review procedure to establish that the design meets all of the applicable criteria. The design requirements set by such procedures are followed through the complete procurement and installation program including spec writing, bid packages, procurement and fabrication inspection, quality assurance

TABLE 3.7-1
SEISMIC CATEGORY EQUIPMENT
(ENVIRONMENT, FUNCTION, QUALIFICATION)

Equipment	Post-DBE				Design	
	Temp.	Interior/Exterior Environment	Qual. Method (*)	Function (**)	Temp.	Interior/Exterior Environment
Primary Vessel	<788°F	Na - Argon ⁽¹⁾ /Air ⁽²⁾	A	C/S	800°F ⁽²⁾	Na - Argon ⁽¹⁾ /Air
Turntable and Storage Tubes	<788°F	Na - Argon ⁽¹⁾	A	S	800°F ⁽²⁾	Na - Argon ⁽¹⁾
Turntable Brg.	<230°F	Argon ⁽¹⁾	A	S	250°F	Argon ⁽¹⁾
Vessel Support Ring	<155°F	Argon ⁽¹⁾ /Air ⁽³⁾ - Air	A	C/S	250°F	Argon ⁽¹⁾ /Air
Vessel Closure Head	<210°F	Argon ⁽¹⁾ /Air ⁽⁶⁾	A	C/S	250°F	Ar ⁽¹⁾ /Air-Drpd Fl Vlv ⁽⁶⁾
Turntable Dr. Pinion Assembly	<210°F	Argon ⁽¹⁾ /Air	A	C	250°F	Argon ⁽¹⁾ /Air
Striker Plate	<130°F	Air ⁽⁶⁾	A	S	150°F	Air-Drpd Fl Vlv ⁽⁶⁾
Floor Valve Adapter	<210°F	Argon ⁽¹⁾ /Air	A	C	250°F	Argon ⁽¹⁾ /Air
Guard Tank	<650°F	Air ⁽³⁾	A	S	700°F	Air ⁽³⁾
NaK Cooler (In-Vessel)	<788°F	NaK/Na - Argon ⁽¹⁾	A	C	800°F ⁽²⁾	NaK/Na - Argon ⁽¹⁾
NDHX ⁽⁷⁾	<788°F	NaK/Air	A	C	800°F ⁽²⁾	NaK/Air
NDHX Support	<103°F	Air	A	S	200°F	Air
NaK Expansion Tank	<150°F	NaK/Air	A	C	650°F	NaK/Air
NaK Valves (Process & First Service Connection)	<788°F	NaK/Air	T	C	1220°F	NaK/Air
NaK Lines (In-Vessel Vault)	<788°F	NaK/Air ⁽³⁾	A	C	800°F ⁽²⁾	NaK/Air
NaK Lines (Ex-Vessel Vault)	<788°F	NaK/Air	A	C	800°F ⁽²⁾	NaK/Air
Na Lines (to Block Valves)	<788°F	Na/Air ⁽³⁾	A	C	800°F	Na/Air ⁽³⁾
Na Block Valves	<788°F	Na/Air ⁽³⁾	A	A ⁽⁴⁾	800°F	Na/Air ⁽³⁾
Na Cover Argon Valves/Rup. Disc	240°F	Argon ⁽¹⁾ /Air	A	C/A	400°F	Argon ⁽¹⁾ /Air
Na Cover Argon Relief Valves	103°F	Argon ⁽¹⁾ /Air	A	A	150°F	Argon ⁽¹⁾
Na Cold Trap	600°F	Na/Air ⁽³⁾	A	C	825°F	Na/Air ⁽³⁾
NDHX Backup Fans	103°F	Air	A	S	150°F	Air
NDHX Backup Motor Generators	103°F	Air	T	C	150°F	Air

TABLE 3.7-1 (Cont'd)

Equipment	Post-DBE				Design	
	Temp.	Interior/Exterior Environment	Qual. Method (*)	Function (**)	Temp.	Interior/Exterior Environment
NaK Expansion Tank Valves	103°F	Argon ⁽¹⁾ /Air	A	C	150°F	Argon ⁽¹⁾ Air
NDHX Temperature Readout	103°F	Air	T	A	≥125°F	Air
High Bay Crane and Support	103°F	Air ⁽⁶⁾	A	S	≥125°F	Air ⁽⁶⁾
Plug Handling Fixture	103°F	Air ⁽⁶⁾	A	C	125°F	Air ⁽⁶⁾
BLTC Transporter	103°F	Air ⁽⁶⁾	A	S	125°F	Air ⁽⁶⁾
Subgrade Vault with Catch Pans	465°F	Air ⁽³⁾	A	S	650°F ⁽²⁾	Air ⁽³⁾
Subgrade Vault (noninert)	103°F	Air	A	S	150°F	Air
Seismic Switches	103°F	Air	T	A ⁽⁵⁾	≥125°F	Air

(*) A - Analysis; T - Test

(**) A - Active; C - Containment; S - Structure

NOTES:

(1) w/sodium or NaK vapor

(2) Faulted condition

(3) w/sodium fire comb. products

(4) Valve to close only

(5) Cause open circuit

(6) w/tornado debris

(7) Includes NDHX damper

documentation, shipping, installation, and checkout. The personnel involved in the QA program report along separate lines to top management, thereby assuring resolution of any problems in a visible and equitable fashion. Further details of design control procedures are in the quality assurance functional plan and the quality program index (refer to Section 17).

3.8 DESIGN OF CATEGORY I STRUCTURES

The FSF building structures that are Category I are so designated because of their function as structural supports for safety-related components, or missile barriers to protect safety-related components, or both. No portion of the FSF building serves a containment function.

3.8.1 Internal Structure

All internal structures that contribute to the support of safety-related Category I FSF components are also designated Category I and are designed accordingly. These structures include, but are not limited to:

- 1) Storage vessel vault foundations and walls
- 2) Support ring and guard tank embedment
- 3) Below-grade foundations and walls that support the holdup tank and associated piping
- 4) NDHX foundations and structures used for HRS piping supports
- 5) Facility crane support member foundations
- 6) Fuel handling equipment (BLTC and Transporter) rails and rail foundations

The seismic design details of Category I structures are in the FSF building design report (Reference 15).

3.8.2 Other Category I Structures

All portions of the FSF structure required for missile protection for safety-related FSF components and separation of redundant HRS components are

also designated Category I and designed accordingly. This includes the walls and roof of the NDHX rooms, grade level floors and hatches that protect below-grade components, and stairwell walls and doors adjoining below-grade areas.

The details of the design of FSF building missile barriers are in Reference 15.

3.8.3 Post-Accident Environments

The post-accident environment of the Seismic Category I equipment has been determined and is presented in Table 3.7-1. All of the Seismic Category I equipment is designed to withstand both the normal operating environment and the design basis accident environment (Table 4.2-1).

3.9 CATEGORY I MECHANICAL SYSTEMS AND COMPONENTS

3.9.1 Dynamic System Analysis and Testing

Except for the seismic switches coupled to the sodium isolation valves (see paragraph 6.1.1), there are no active mechanical FSF components that must perform a safety-related mechanical function during any seismic event.

3.9.2 ASME Code Section III, Class 2 and 3 Components

The design of ASME Code, Section III, Class 2 and 3 Category I Components is carried out to assure that the components meet the code requirements when subjected to the following design conditions:

- 1) Thermal design conditions including design (650⁰F) and faulted (800⁰F) temperatures (i.e., stress within limits for static pressure and dead weight loads).
- 2) Stress limits (design or faulted) involve static pressure plus dead weight plus DBE at design temperatures.

3.9.2.1 Analytical and Empirical Methods for Design of Valves

The design primarily uses only valves and other small components that have already been seismically qualified under the FFTF, SPTF, CRBRP, or other similar programs. All valves used will have documentation of their successful qualification in the system.

3.9.2.2 Pressure Relief Device Design and Installation Criteria

The design, installation, and periodic test of the pressure relief devices for Category I components in the FSF conform to the ASME code requirements of the vessel that they protect. The storage vessel is equipped with redundant parallel pressure relief valves with a pressure relief setting of 15 lb/in.² gage (103 kPa). Rupture discs in series with the relief valves protect them from sodium vapor. A small vent line behind each rupture disc allows the escape of any gas that may leak through a flaw in the rupture disc. Additionally, valves are supplied to allow isolation and testing of each pressure relief path. These valves are normally in the locked open position and are interlocked to prevent more than one being closed at a time. The holdup tank is also supplied with a pressure relief valve, but rupture discs are not necessary due to the absence of sodium vapor. NaK expansion tanks have one pressure relief valve each with a protective rupture disc and provision for periodic inspection with pressurized gas. The calculated reaction forces from relief action of any of these valves are negligibly small when compared with the deadweight and seismic forces that these components are designed to withstand.

3.9.2.3 Stress Level for Category I Components

Piping system stress analyses have been performed for all Category I piping systems. The details of the piping stresses are in the piping stress analysis (Reference 13).

3.9.2.4 Field Run Piping Systems

The only field run piping in the FSF is the below-grade pneumatic distribution lines from the distribution header to the valve operators. None of these lines is a Category I line. Failure of any pneumatic line will not change valve status, except for the sodium purification loop isolation valves that close on loss of actuating pressure, which is the safe condition.

3.9.3 Components Not Covered by ASME Code

Most of the Category I systems contain components that come under the ASME Code. However, some Category I components are not designed to the ASME Code. These are:

1) NDHX Backup Fans

The NDHX fans must retain their functional capability in the event of an earthquake and are protected from tornados. These components were qualified by testing.

2) Motor/Generator for NDHX Backup Fans

The motor/generators must maintain their functional integrity in the event of an earthquake and are protected from tornados. The motor/generator sets have been seismically qualified by test before installation.

3.10 SEISMIC DESIGN OF INSTRUMENTATION AND ELECTRICAL SUPPLY

3.10.1 Seismic Design Criteria

The components of the instrumentation, control and electrical systems are designed to withstand various levels of seismic disturbance based upon the importance of their safety function. Accordingly, the following criteria have been set.

<u>Component or System</u>	<u>Seismic Basis</u>	<u>Category</u>
NDHX Inlet & Outlet Temp. Indicator (local)	DBE	I
NDHX Backup Fan	DBE	III/I
NDHX Backup Motor Generator	DBE	III/I
EI&C Backup Power (Battery)	UBC	III
Balance of System	UBC	III

The seismic Category I components are selected from components approved for service under the DBE conditions or qualified by test. The motor/generator set and backup fan are not required to operate after the DBE but will remain intact and not cause failure of Category I equipment in the vicinity.

3.10.2 Seismic Analysis Testing Procedures and Restraint

The majority of seismic Category I EI&C components will be mounted directly on the NDHX. The NDHX will be qualified by analysis to withstand the DBE under the requirements of the ASME Code. The generator mounting in the NDHX vault will also be qualified along with all of the individual components. The temperature indication instruments have been selected from among components already qualified through the FFTF.

3.11 ENVIRONMENTAL DESIGN OF MECHANICAL AND ELECTRICAL EQUIPMENT

3.11.1 Equipment Identification

Category I mechanical and electrical equipment is located as follows:

<u>Equipment</u>	<u>Location</u>
NDHX Backup Fan	NDHX Room
Backup Generator	NDHX Room
NDHX Thermocouple Readout	NDHX Room
NDHX Damper System	NDHX Room

The worst-case environment for these components is the site environmental extremes, and the component must be designed to function acceptably under these conditions.

3.11.2 Qualification Tests and Analyses

Mechanical and electrical components designated Seismic Category I will be either analyzed or tested under conditions at least as severe as the expected environmental extremes or selected from components that have been qualified for environments that are at least as severe.

3.11.3 Loss of Ventilation

Loss of the building HVAC system does not affect the performance of any safety-related Seismic Category I components.

3.11.4 Special Considerations

Electrical instrumentation and conduit that serve safety-related functions are enclosed according to the requirements of the National Electrical Code Class I, Division (XX) (Reference 16). The NDHX fans are not necessarily required to operate in the event of or in the aftermath of a sodium leak or fire. This is because a leak or fire implies that the heat removal loop is inoperable and a backup fan is superfluous.

3.12 REFERENCES

1. Fuel Storage Facility System Design Description, N106SDD210001, Rev. 1, Atomics International, Canoga Park, CA, April 1978.
2. Code of Federal Regulations, 10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants," USGPO, Washington, DC.
3. ASME Boiler and Pressure Vessel Code, Section III, The American Society of Mechanical Engineers, New York, NY.
4. Regulatory Guide 1.29, "Seismic Design Classification," Rev. 3, US Nuclear Regulatory Commission, Washington, DC, September 1978.

3.12 REFERENCES (Cont'd)

5. Uniform Building Code (UBC), Zone 2, International Conference of Building Officials, Whittier, CA, 1979.
6. ANSI B18.2.1, Square and Hex Bolts and Screws, The American Society of Mechanical Engineers, New York, NY, November 1972.
7. Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive Waste-Containing Components of Nuclear Power Plants," Rev. 3, US Nuclear Regulatory Commission, Washington, DC, February 1976.
8. "Wind Forces on Structures," ASCE Paper 3269, Transactions of the American Society of Civil Engineers, Vol. 126, Part II, 1961.
9. Earthquake Debris Damage Assessment, N106T1230009, Atomics International, Canoga Park, CA, June 1978.
10. Dynamic Analysis of the Fuel Storage Facility, N106SR410001, Atomics International, Canoga Park, CA.
11. Final Stress Analysis of FSF Vessel Subsystem, N106SR220002, Atomics International, Canoga Park, CA.
12. Stress Analysis of Preliminary Design of Natural Draft Heat Exchanger, N106SR310006, Atomics International, Canoga Park, CA.
13. Stress Analysis of FSF Piping System, N106SR440001, Atomics International, Canoga Park, CA.
14. FSF Cold Trap Stress Analysis, N106SR220004, Atomics International, Canoga Park, CA, November 1977.
15. FSF Title I Engineering Final Report, FSF 8954, Vol. I and II, Norman Engineering Co., Los Angeles, CA, September 1977.
16. National Electric Code, NFPA No. 70-1978, National Fire Protection Association, Boston, MA, 1978.

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4.0 FUEL STORAGE VESSEL

4.1 SUMMARY DESCRIPTION

The fuel storage vessel system is located in the tornado and DBE hardened reinforced concrete subgrade vault. Only the upper surface of the closure head is exposed outside of this vault. The vessel system is designed to maintain its structural integrity and functional capability when it is subjected to DBE-generated forces at the mounting point, with the BLTC mated to the vessel closure head in the fuel transfer mode. The vessel provides for the storage in a sodium environment of up to 466 FFTF spent fuel assemblies and canisters of fuel. The canisters, subsequently termed cans, are used to contain fuel pins that have been disassembled from fuel elements either because they underwent examination or because they had cladding failures. The storage vessel system (Figure 4.1-1) consists of the primary vessel, turntable, support ring, seals, closure head assembly, turntable positioning system, floor valve adapter, and storage tubes. The storage vessel assembly is compatible with the FFTF BLTC and floor valve.

The primary vessel is about 21 ft (6.9 m) in diameter and 24 ft (7.3 m) long. It is supported by a flange at the top and contains about 238,000 pounds (107,955 kg) of sodium. Eutectic NaK coolant, circulated through the vessel via two closed loops, removes heat from the vessel by natural convection.

The turntable supports and provides the proper spacing of the 466 storage tubes that contain the bare fuel assemblies. It has a basket configuration and is supported at the top by a ball bearing race attached to the upper flange of the primary vessel.

A support ring, on which the primary vessel is mounted, carries the load out to the wall of the concrete vault. The support ring is a part of the vessel system pressure boundary and also carries the weight of the closure head. Metal O-ring seals prevent leakage between the support ring and the primary vessel and the closure head.

FIGURE 4.1-1. Storage Vessel Subsystem.

The closure head assembly provides boundary for the primary vessel. It also acts as a thermal barrier and gamma-neutron shield and serves as a support structure for the BLTC during fuel handling operations. There are ten access ports in the closure head. Each port is aligned with one of the ten concentric rings of storage locations in the turntable. Other penetrations in the closure head provide for turntable position indication, instrumentation thimbles, and the turntable drive pinion shaft.

A turntable positioning system, consisting of a variable-speed dc drive motor and gear train, is mounted on the head of the vessel. The drive shaft penetrates the closure head, and the pinion gear engages the turntable bull gear, which is integral with the inner race of the bearing.

The floor valve adapter interfaces between the FSF floor valve and the fuel transfer ports in the closure head during the fuel handling operations. It also becomes part of and maintains the integrity of the vessel system containment boundary through the use of elastomeric seals at the closure head junction and by providing a mating surface for the floor valve seals.

The storage tubes, mounted in the turntable, contain and position the fuel assemblies and cans within the storage vessel. They also align the stored components when under the storage ports so that the BLTC grapple can grasp them for removal. The fuel assemblies in stored position extend above the molten coolant level and the level depends primarily upon temperature, but is also influenced by loading. The exposed upper portion of the assembly protrudes approximately 2' at initial loading and approximately 1'4" in fully loaded condition.

Material Summary -- The materials used in the sodium system are generally carbon steel. The NaK heat removal systems are generally stainless steel except for the carbon steel NaK cooler piping within the storage vessel. The materials are summarized in Table 4.1-1. Implications for fuel assembly material resulting from exposure above the sodium pool are discussed in 4.2.2.3.2.

TABLE 4.1-1
MATERIALS IN CONTACT WITH LIQUID METAL

<u>Component</u>	<u>Reference Paragraph</u>	<u>Material</u>
Storage Vessel	4.2.1.3.1	SA 516 (shell) SA 508 (flange)
Turntable	4.2.2.3.1	SA 516 (shell) SA 533 (flange)
Storage Tubes	4.2.2.3.2	ASTM A 334
Closure Head	4.2.1.3.2	SA 533
Guard Tank	6.2.2.6	SA 516
NaK Cooler (within Storage Vessel)	5.1.3	SA 106
NaK Piping	5.5.3	304 SS
NDHX Tubes	5.5.1	304 SS
NaK Expansion Tank	5.5.2	304 SS

4.2 MECHANICAL DESIGN

4.2.1 Pressure Boundary

4.2.1.1 Design Bases

The storage vessel pressure boundary (Figure 4.2-1) encloses the vessel, the closure head, the inner portion of the support ring, the floor valve adapter, and the positioning system components located in and below the closure head. The detail, design, construction, structural, design life, seismic, and other requirements of the individual components are specified in Reference 1, Section 2.2, as amended, and in the FSF storage vessel subsystem design specification (Reference 2).

The components were analyzed and designed in accordance with seismic design criteria.⁽³⁾ A summary of the minimum design and operating condition requirements of the pressure boundary components is in Table 4.2-1.

4.2.1.2 Design Description

The design layout drawings of the major pressure boundary components are summarized in Table 4.2-2. The detailed design description is in Reference 1.

FIGURE 4.2-1. Storage Vessel Pressure Boundary.

TABLE 4.2-1.

MINIMUM DESIGN AND OPERATING CONDITION REQUIREMENTS FOR THE PRESSURE BOUNDARY COMPONENTS

Component	Design Conditions Temp @ Loads	Normal Conditions (Cold Trapping) Temp @ Loads	Faulted Conditions Temp @ Loads
Vessel -			
o Shell	1. 650°F (343°C) @ 15 psig (100 kPa) plus static head of sodium plus dead weight of structure	600°F (316°C) @ 0.5 psig (3.4 kPa) plus nozzle loads as per Design Specification plus dead weight plus content weight	800°F (427°C) @ 6 psig (40 kPa) plus dead weight plus contents plus nozzle loads as per Design Specification
	2. ≤90°F (32.2°C) @ Full Vacuum		DBE @ 600°F and 0.5 psig
o Flange	1. 250°F (121°C) @ 15 psig (100 kPa) plus dead weight of structure plus content	200°F (93°C) @ 0.5 psig (3.4 kPa) plus same as shell	250°F (121°C) @ 6 psig (40 kPa) plus same as shell
	2. ≤90°F (32.2°C) @ Full Vacuum		DBE @ (1) normal operation, turntable loaded plus (2) BLTC attached to the top head, floor valve and floor valve adapter in place; (3) consider an imbalance from partial loading of the turn- table

TABLE 4.2-1 (Cont'd)

Component	<u>Design Conditions</u> Temp @ Loads	<u>Normal Conditions</u> (Cold Trapping) Temp @ Loads	<u>Faulted Conditions</u> Temp @ Loads
Closure Head -	250°F (121°C) @ 15 psig (100 kPa) plus dead weight of structure plus loads from fully loaded BLTC plus dead weight of striker plate ≤90°F (32.2°C) @ Full Vacuum	200°F (93°C) @ 0.5 psig (3.4 kPa) plus dead weight of head plus loaded BLTC plus striker plate plus turntable drive assembly	250°F (121°C) @ 6 psig (40 kPa) plus dead weight of head, loaded BLTC, striker plate, turntable drive assembly DBE @ 200°F (93°C), 0.5 psig (3.4 kPa) plus BLTC wheel load Impact Load @ 200°F (93°C), 0.5 psig (3.4 kPa) plus impact load from dropping floor valve from 2-ft (0.6-m) height onto the striker plate

TABLE 4.2-1 (Cont'd)

Component	Design Conditions	Normal Conditions	Faulted Conditions
	Temp @ Loads	(Cold Trapping) Temp @ Loads	Temp @ Loads
Support Ring -	250°F (121°C) @ 15 psig (100 kPa) plus dead weight of structure plus loads from head, striker plate, floor valve adapter, floor valve, loaded BLTC vessel content, loaded turntable, plus bearing assembly	150°F (66°C) @ 0.5 psig (3.4 kPa) plus loads as in design conditions	200°F (93°C) @ 6 psig (40 kPa) plus loads as in design conditions DBE @ 150°F (66°C), 0.5 psig (3.4 kPa) plus loads as for the vessel
Floor Valve Adapter -	250°F (121°C) @ 15 psig (100 kPa) plus dead weight of floor valve and BLTC (cask only)	200°F (93°C) @ 0.5 psig (3.4 kPa) plus same as design conditions	200°F (93°C) @ 6 psig (40 kPa) plus same as design conditions DBE @ 200°F (93°C), 0.5 psig (3.4 kPa) plus loads as for vessel Dropped Floor Valve: Floor valve dropped from 2-ft (0.6-m) height

TABLE 4.2-1 (Cont'd)

Component	<u>Design Conditions</u> Temp @ Loads	<u>Normal Conditions</u> (Cold Trapping) Temp @ Loads	<u>Faulted Conditions</u> Temp @ Loads
Turntable Position System -			
● Drive Motor and Gear Box	150°F (66°C) @ static and dynamic loads of drive system		250°F (121°C) @ TBD
● Component above and within the closure head	200°F (93°C) @ N.A.		
● Components below the closure head	250°F (121°C) @ N.A.		250°F (121°C) @ TBD
Seals -			
● Elastomere	250°F (121°C) @ negligible	N.A.	N.A.
● Metallic	800°F (427°C) @ variable depending on material and design	N.A.	N.A.

TABLE 4.2-2
PRESSURE BOUNDARY DESIGN LAYOUT DRAWINGS

<u>Drawing Number</u>	<u>Component</u>
N106000218	Primary Vessel Assembly
N106000249	Support Ring Weld Assembly
N106000178	Closure Head Assembly
N106000198	Floor Valve Adapter Assembly
N106000148	Pinion Drive Assembly

4.2.1.3 Design Evaluation

The requirements and the design conditions for the pressure boundary components are in this section. The calculated safety margins for these components under the design and faulted conditions are summarized in Table 4.2-3. The allowable stress for the faulted condition was taken as 70 percent of the ultimate strength at the faulted temperature. The Final Stress Analysis Report⁽⁴⁾ references ASME Code⁽⁵⁾ Case 1481-1, Elevated Temperature Design; and Code Case 1607-1, Stress Criteria for Section III, Division 1, Class 2 and 3 Vessels. The detailed analysis in which these margins were established is in Reference 4.

4.2.1.3.1 Storage Vessel

Material Data

1. Shell-(Cylindrical Section) ASME SA-516, Grade 65 plate:
t = 1.00 in. (0.025 m).
2. Head-(Std. ASME 2:1 Ellipsoidal) ASME SA-516, Grade 65 plate:
t = 1.00 in. (0.025 m).

3. Flange-ASME SA-533 plate or SA-508 forging; the vessel flange is machined on all surfaces to tolerances noted on Drawing N106R000084.

Design Analysis Basis -- The vessel design analysis follows the criteria set forth in the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NC, for Class 2 components. In particular, the alternative design rules for vessels, Paragraph NC-3200, are followed. Paragraph NC-3200 allows the use of analysis with the higher design stress intensity values of Table I-1.0.

Allowable Stresses

<u>Material</u>	S_m <u>(lb/in.²)</u>	<u>Temperature</u> <u>(°F)</u>
SA-516 plate (P 1), Grade 65	17,000 (120 MPa)	650°F (343°C)
SA-533 plate or SA-508 forging (P 3), Grade B CI.1	26,700 (180 MPa)	650°F (343°C)

Corrosion Allowance -- The design margins are based upon a minimum material thickness that takes into account a corrosion allowance of 50 mils maximum over the life of the vessel (20 years).

4.2.1.3.2 Closure Head

The stress analysis is contained in Reference 4; a summary as pertains to the closure head is included in Table 4.2-3.

4.2.1.3.3 Floor Valve Adapter

ASME SA-240 plate, Type 304 stainless steel.

TABLE 4.2-3

CALCULATED PRESSURE BOUNDARY COMPONENTS DESIGN MARGINS

(Sheet 1 of 2)

Component	Condition	Loading	Allowable Stress psi (MPa)	@ Temp. °F (°C)	Design* Margin
VESSEL:					
Shell - Cylindrical Section	Design	External Pressure	-29.9 (-0.2)	300 (149)	0.99
Shell - Cylindrical Section	Design	Compression	8000 (55)	650 (343)	2.43
Shell - Cylindrical Section	Design	Shell Thickness	.22 in. required	700 (371)	3.23
Shell - Elipsoidal Head	Design	Shell Thickness	0.33 in. required	650 (343)	1.88
Shell - Elipsoidal Head	Design	External Pressure	-63.5 (-0.4)	700 (371)	3.23
NaK Nozzle Body, Interior	Design	Press. + DW + TE	17400 (120)	650 (343)	2.87
NaK Nozzle Body, Interior	Faulted	Press. + DW + TE + DBE	41760 (290)	650 (343)	5.10
Shell - Local Nozzle Penetration	Design	Press. + DW + TE	17000 (120)	650 (343)	7.92
Shell - Local Nozzle Penetration	Design	Press. + DW + TE	25500 (180)	650 (343)	5.52
Shell - Local Nozzle Penetration	Design	Press. + DW + TE	51000 (350)	650 (343)	1.65
Shell - Local Nozzle Penetration	Faulted	Press. + DW + TE + DBE	45500 (310)	650 (343)	0.16
Flange	Design	Press. + DW	17000 (120)	650 (343)	6.71
Flange	Design	Press. + DW	25500 (180)	650 (343)	9.09
Flange	Faulted	Press. + DW + TE + DBE	45500 (310)	500 (260)	2.86
Flange - Bolt	Faulted	DBE	65800 (454)	800 (427)	0.01
NaK Nozzle Body, Exterior	Design	Press. + DW + TE	17400 (120)	650 (343)	.05
NaK Nozzle Body, Exterior	Faulted	Press. + DW + TE + DBE	49000 (340)	650 (343)	.41

$$* \text{Design Margin} = \frac{\text{Allowable Stress}}{\text{Applied Stress}} - 1$$

TABLE 4.2-3 (Sheet 2 of 2)

Component	Condition	Loading	Allowable Stress psi (MPa)	@ Temp. °F (°C)	Design Margin
CLOSURE HEAD:					
Closure Head	Design	Primary	27000 (180)	250 (121)	>9.28
Closure Head	Faulted	Primary	27000 (180)	250 (121)	>9.28
Closure Head	Design	Press. + DW	40000 (280)	250 (121)	>.11
Closure Head	Faulted	Press. + DW	40000 (280)	250 (121)	>.11
FLOOR VALVE ADAPTER:					
	Design	Press. + DW	23750 (160)	250 (121)	>5
	Faulted	Press. + DW + DBE	40600 (280)	250 (121)	>5
Sodium Level Probe	Design/ Faulted	Bending	16000 (110)	650 (343)	>.79
Sodium Level Probe	Design/ Faulted	Buckling	8900 (61)	650 (343)	>.006

Design Analysis Basis -- The floor valve adapter design analysis follows the criteria in ASME Boiler and Pressure Vessel Code, Section III, Class 2, per NC-3200.

Allowable Stress -- Material allowable stress = $S_m = 20,000 \text{ lb/in}^2$
(140 MPa) @ 250°F (121°C)

Loading Conditions -- During the fuel transfer mode when the transfer dolly is centered over a transfer port, the dead weight of the transfer cask 100,000 lb (45,360 kg) and the floor valve 17,000 lb (7711 kg) is imposed upon the floor valve adapter.

The floor valve adapter, being a pressure containment item during fuel transfer operations, is subjected to internal pressure. The only other loads encountered are during the postulated DBE which produces an additional acceleration loading to be added to the dead weight loading. The DBE loading is classified as a faulted condition.

4.2.1.3.4 Support Ring

Design Analysis Basis -- Design per ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NC-3200.

Design Condition 1

Temperature (T) = 250°F (121°C)

Pressure (P) = 15 lb/in.² gage (10⁵ Pa)

Dead Weight Loads, including BLTC (DW)

Load Combination

P + DW

Stress Limits

NC-3200

$$P_m \leq kS_m$$

and

$$P_L + P_b \leq 1.5 kS_m$$

where k = 1.0

Design Condition 2

Temperature (T) = 90°F (32°C)

Pressure (V) = Full vacuum

Dead Weight Loads, not including BLTC (DW)

Load Condition

V + DW

Stress Limits

NC-3200

$$P_m \leq kS_m$$

and

$$P_L + P_b \leq 1.5 kS_m$$

where k = 1.0

Operating Condition (Faulted) 1

$T = 200^{\circ}\text{F}$ (93°C)

Pressure (P) = 6 lb/in.^2 gage (40 kPa)

Dead Weight Loads (includes BLTC) (DW)

DBE Seismic Loads (DBE)

Load Combination

$P + DW + DBE$

Stress Limits

NCC- 3200

$P_L + P_b \leq 0.7 S_u$
(see Appendix F)

Operating Condition 2

$t = 150^{\circ}\text{F}^*$ (65°C)

Pressure (P) = 0.5 lb/in.^2 gage (3.5 kPa)

Dead Weight Loads (includes BLTC) (DW)

Loads caused by temperature effects (T)

Load Combination

$P + DW + T$

Stress Limits

$P_L + P_b + Q \leq 3 S_m$

*Use for evaluating allowables.

4.2.1.4 Testing and Inspection

Testing and inspection requirements are not applicable to the SC-3 System.

4.2.2 Turntable and Storage Tube

4.2.2.1 Design Bases

The detailed design, construction, structural, design life, seismic, and other requirements for these components are specified in Reference 1, as amended, and in Reference 2.

The design provides adequate safety margins during normal and faulted operating conditions. The components were analyzed and designed in accordance with seismic design criteria.⁽³⁾ A summary of the design, normal operating, and faulted conditions of these components is presented in Table 4.2-4.

4.2.2.2 Design Description

Refer to Reference 1, Section 2, for a detailed description of the turntable and storage tube components.

The design layout drawings of the turntable and storage tube assemblies are summarized in Table 4.2-5.

TABLE 4.2-4 (Sheet 1 of 2)

DESIGN AND OPERATING CONDITIONS TURNTABLE AND STORAGE TUBES

Component	Design Conditions		Normal Conditions		Faulted Conditions	
	Temp. °F (°C)	Loads	Temp. °F (°C)	Loads	Temp. °F (°C)	Loads
TURNTABLE:						
- Shell	650 (343)	Dead weight of fully loaded turntable plus: starting and running torque loads	600 (315)	Dead weight of the fully loaded turntable plus the loads resulting from the turntable rotation	HIGH TEMPERATURE:	
- Flange	250 (121)		200 (93)		800 (426) Plus dead weight of the loaded turntable 250 (121)	
					DROPPED FUEL ELEMENT:	
					The dead weight of a fuel element 381 lb (173 kg) dropped from the BLTC into a storage tube from the fully retracted BLTC grapple height of 23 feet (7.6 m).	
					SEISMIC EVENT:	
					Design Basis Earthquake (DBE), Temperatures: Shell-600°F (315°C), Flange 200°F (93°C). DBE assumed to occur a maximum of one time in the life of the FSF under the conditions:	
					(A) Normal operation - turntable completely loaded with fuel assemblies,	
					(B) BLTC attached to the top head, FFTF floor valve and FSF floor valve adapter in place, and	
					(C) Consideration shall be given to the imbalance resulting from the outer row of storage tube locations being half-filled with fuel assemblies.	

TABLE 4.2-4 (Sheet 2 of 2)

Component	Design Conditions		Normal Conditions		Faulted Conditions	
	Temp. °F (°C)	Loads	Temp. °F (°C)	Loads	Temp. °F (°C)	Loads
STORAGE TUBE	650 (343)	One Fuel Element 381 lb (173 kg)	600 (315)	One Fuel Element 381 lb (173 kg)	<u>HIGH TEMPERATURE</u>	
					800 (426)	One Fuel Element 381 lb (173 kg)
					<u>DROPPED FUEL ELEMENT</u>	
					600 (315)	One Fuel Element plus another dropped from fully retracted BLTC grapple from height of 23 feet (7.6 m)
					<u>SEISMIC EVENT</u>	
					600 (315)	Design Basis Earthquake with one fuel element (Turntable completely loaded)

TABLE 4.2-5
NONPRESSURE BOUNDARY DESIGN LAYOUT DRAWINGS

<u>Drawing Number</u>	<u>Component</u>
N106000211	Turntable Assembly
N106000223	Storage Tube Assembly

4.2.2.3 Design Evaluation

The requirements and the design conditions for the turntable and the storage tube are in this section. The calculated design margins for these components under the design and faulted conditions are all positive. The detailed analysis in which these margins were established is presented in Reference 4.

4.2.2.3.1 Turntable

Design Data

1. Design per ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NF, Class 2, "Component Supports."
2. Design pressure: (None); Design temperature: 650⁰F (343⁰C)

Material Data

1. Shell, grid plates, support plates: ASME SA-516, Grade 65
2. Flange: ASME SA-533, Grade B, Class 1 plate

Design Analysis Basis

The turntable design analysis follows the criteria set forth in Reference 1 Subsection NF. In particular, Paragraph NF-3300, the design of Class 2 component supports, is used. The design procedure used for the analysis of the turntable structure was design-by-analysis, using the maximum stress theory. Table NF-3132.1(b)-1 indicates the various code paragraphs used for the different types of supports.

Allowable Stresses

The allowable stress for Class 2 plate and shell type construction are from Reference 4.

SA-516 plate (Grade 65)	$S = 16,200 \text{ lb/in.}^2 @ 650^\circ\text{F} (343^\circ\text{C})$ $(1.1 \times 10^3 \text{ Pa})$
SA-533 plate or SA-508 forging (Type B C1.1)	$S = 20,000 \text{ lb/in.}^2 @ 650^\circ\text{F} (343^\circ\text{C})$ $(1.4 \times 10^8 \text{ Pa})$

The allowable stresses for Class 2 linear type component support structure are from Reference 4 (yield strength).

SA-516 plate (Grade 65)	$S_y = 25,400 \text{ lb/in.}^2 @ 650^\circ\text{C} (343^\circ\text{C})$ $(1.8 \times 10^8 \text{ Pa})$
SA-533 plate or SA-508 forging (Grade B C1.1)	$S_y = 41,400 \text{ lb.in.}^2 @ 650^\circ\text{F} (343^\circ\text{C})$ $(2.8 \times 10^8 \text{ Pa})$

Corrosion Allowance

No corrosion allowance is accounted for in the subsequent calculations of the design analysis of the FSF turntable structure.

Design Condition Stress Limits

$$\sigma_1 < S \quad \text{and} \quad \sigma_1 + \sigma_2 < 1.5 S$$

Faulted Operating Condition

Load combination: Dead load plus pressure plus DBE

Note: The only essential difference between the faulted condition and design condition is the inclusion of loads that result from the DBE loading.

4.2.2.3.2 Storage Tube

Material

ASTM A-334 seamless carbon steel tubing. Note: The equivalent ASME material specification is SA-192.

The storage tubes are not coded items; however, for this analysis (Reference 4), the allowable stress for SA-192 that appears in the ASME Boiler and Pressure Vessel Code, Section VIII, Division 1, was used.

$$S_{\text{allowable}} = 11,700 \text{ lb/in}^2 (8.1 \times 10^7 \text{ Pa}) @ 650^\circ\text{F} (343^\circ\text{C})$$

Design Temperature

650°F (343°C) (Note: This agrees with the design temperature of the turntable.)

Loading Conditions

The storage tubes, under normal conditions, have as their only function to maintain the stored fuel assemblies in a upright position. All void space on the tubes if filled with liquid sodium of the vessel pool. The only structural function then is to carry their own dead weight. Dead weight stresses are negligible: $100 \text{ lb/in}^2 (6.89 \times 10^5 \text{ Pa})$.

Stored Components - Fuel Assembly Corrosion

The corrosion rate of fuel cladding due to sodium impurity compounds is projected to be <2 microns per year at 600°F. Mechanical degradation due to bulk corrosion is therefore acceptable on the basis that only 1.6 mil (40 microns) are affected in 20 years. Elevated levels of oxygen or hydroxide on the sodium pool surface could encourage local pitting of cladding in the fuel assemblies. Local pitting may lead to cladding breach, release of fission product gases, and subsequent sodium flooding of an irradiated fuel pin. Cladding breach with fission product release is evaluated in Chapter 12 and found to be acceptable. Sodium flooding of an irradiated fuel pin or sodium oxide/hydride accumulation on the upper clad are also acceptable and are a consideration only in cases of potential future assembly cleaning, examination and disposal.

4.2.3 Reactivity Control

Not applicable since the facility is designed to remain subcritical under all conditions.

4.3 CRITICALITY CONTROL

4.3.1 Description

The primary function of the FSF is to store spent fuel assemblies from the FFTF. There are 466 storage positions in a single tier. These positions are arranged in 10 concentric rings, with 9" between adjacent radii (see Figure 4.3-1). During initial operations only 9 rows will be used. The circumferential spacing is a minimum of 9" center-to-center. At each storage location is a 7.0" OD, 0.25" thick seamless carbon steel (ASTM-A344) storage tube. The fueled portion or canisters* will be placed inside these storage tubes. This entire configuration will be immersed in molten sodium. In addition, the turntable has an internal annulus made of natural B₄C which has a 39.5" (1.00 m) OD, 48.7" (1.2 m) length, and 2" (0.05 m) thickness. This will assure a $k_{eff} < 0.95$. See chapter 15 for a discussion of the safety analysis.

*Steel canisters are used to package fuel pins disassembled from assemblies for examination in the IEM Cells of FFTF. They contain fewer pins per canister than the assemblies do. See Reference 6 for discussion.

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FIGURE 4.3-1. FSF Storage Array.

4.3.2 Summary of Results (from Reference 7)

The final design of the FSF storage array has 466 storage positions. The design includes an annulus of B_4C just inside the innermost ring of positions. The expected (but not required) arrangement for the fully loaded FSF is mixed inner and outer driver assemblies from Cores 1 and 2 (154 assemblies) loaded toward the center of the array, and mixed inner and outer driver assemblies from Cores 3-6 (312 assemblies) loaded toward the outside.

Driver fuel is expected (but not required) to have burned to an average discharge burnup of 60,000 MWd/Mtm. The k_{eff} for this expected arrangement was calculated to be 0.824 with all ten rings filled.

When the innermost (most reactive) driver assemblies in the above arrangement were replaced by one core loading (75 assemblies) of unburned Cores 1 and 2-type fuel, the k_{eff} was calculated to be 0.863 with all ten rings filled.

For the FSF loaded with all discharge fuel in the most reactive configuration, the k_{eff} was calculated to be 0.853. The most reactive configuration was found to be one in which the enrichment of the fuel increased toward the center of the array.

Although the anticipated loading contains discharge fuel, the storage of unburned fuel assemblies is not prevented. Therefore, the worst case loading of all unburned fuel loaded in the maximum criticality arrangement was calculated. The reactivity was found to be 0.932 ± 0.017 using a diffusion-based calculational method and 0.930 ± 0.019 using a Monte Carlo-based method. The B_4C annulus which was included in the system would be needed to meet the criterion of 0.950 for the criticality plus uncertainty. Analysis has shown that less than 0.0001% of the boron would be burned out during the 20-year design life. The expected arrangement with all unburned fuel was calculated to have a criticality of 0.913.

The driver fuel in each of Cores 3-6 is assumed to be the same and is designated Cores 3- and 4-type fuel. The reactivities for various configurations and compositions are shown in Table 4.3-1 and discussed in Reference 7.

Because of (1) the large uncertainty introduced by extrapolation from a much smaller experimental model, (2) the core close approach of the k_{eff} plus the 95% confidence level to 0.95, and (3) maintaining a high degree of versatility by restricting neither the burnup nor the loading sequence, the access port to ring 5 shall be sealed during initial fuel loading.* This provides 422 storage ports for immediate use. The k_{eff} for this condition is 0.881 ± 0.028 at the 95% confidence level (Chapter 15). After significant loading has occurred, ring 5 may be released for use based on the "as-loaded" condition of the loaded ports. (See Chapter 16, Technical Specifications).

4.4 THERMAL HYDRAULIC DESIGN

The temperature distributions in the storage vessel had been determined from a two-dimensional thermal-hydraulic model which considered heat transport by conduction plus natural convection; such a model is appropriate because of the evident cylindrical symmetry of the system, except for azimuthal temperature gradients in the cold sink tubes which were taken into account. In normal steady-state operation (205 kW power, both NaK cooling loops operative), natural convection proved powerful in eliminating temperature peaking throughout the sodium pool. The temperature difference between the hottest point in the pool and the coldest point in the downflow area between the turntable and vessel walls is approximately 1.5°F ($\sim 0.8^{\circ}\text{C}$).

* See Chapter 15, Safety Analysis, which discusses the results of various calculations, including check calculations for which the sum of the k_{eff} + 95% CL is greater than 0.95 (0.942 ± 0.025) with all ten rows used to store unirradiated fuel and with an unrestricted loading sequence.

TABLE 4.3-1
CALCULATED CRITICALITIES FOR VARIOUS LOADINGS*

(10 Rings Loaded)

<u>Fuel Loading Arrangement (Inside to Outside)</u>	k_{eff}
Mixed inner and outer Cores 1 and 2; mixed inner and outer Cores 3 and 4, all with average discharge burnup	0.824
As above, except Cores 1 and 2 and Cores 3 and 4 fuel interchanged	0.853
As first case, except innermost 75 assemblies are Cores 1 and 2 unburned fuel	0.863
As second case, except innermost 75 assemblies are Cores 1 and 2 unburned fuel	0.877
Inner Cores 1 and 2; inner Cores 3 and 4; outer Cores 1 and 2; outer Cores 3 and 4, all unburned fuel	0.892
Mixed inner and outer Cores 1 and 2; mixed inner and outer Cores 3 and 4, all unburned fuel	0.913
Outer Cores 1 and 2; inner Cores 1 and 2; outer Cores 3 and 4; inner Cores 3 and 4, all unburned fuel	0.923
Outer Cores 3 and 4; outer Cores 1 and 2; inner Cores 3 and 4, inner Cores 1 and 2, all unburned fuel	0.935

*These values are mutually consistent but do not reflect some minor changes included in the final design.

It was assumed that all the power is generated in a 3-ft (0.9 m) axial region in the fuel assemblies; the radial power profile was assumed flat but the axial power profile was given a 1.3 to 1 peak to average chopped cosine distribution. It was assumed that the sodium pool reached only the minimum leak level, axially, just above the top of the fuel and 7 ft (2.1 m) above the inlet plenum. Only this 7 ft (2.1 m) region was modeled. Thus, the calculation is conservative since no credit was taken for heat transfer paths above the minimum leak level or below the fuel assembly region. In addition: 1) conservatively large values were assumed for the total kinetic energy (inlet and out) and flow loop losses, 2) only conduction was allowed in the inner sodium annulus, and 3) the vessel wall was assumed adiabatic. Therefore, values for the convective mass transport, flow velocities, and Reynold's numbers are likely to be lower bounds, and the temperature gradients upper bounds. The expression used for frictional losses assumes turbulent flow, but even for the laminar flow cases encountered, the frictional contribution to pressure drop is outweighed by the assumed inlet and exit losses.

The flow logic assumed vertical flow within the sodium annuli, in the storage tubes, and between the rings of storage tubes. Cross flow between rings was neglected as a second order effect, but convective flows to the cold sink (assumed to be horizontal and at the minimum leak level) from various rings were allowed to mix. The influence on ΔP of the lower plenum and outlet flow to the cold sink were assumed accounted for by the conservative inlet and exit losses. The possibility of counter flow (i.e., locally offsetting upward and downward convection that effectively traps heat, causing temperature peaking) was considered but rejected as remote, since the high sodium thermal conductivity would not allow the prerequisite radial temperature gradients to exist at 205-kW power levels. Note: Individual fuel elements are limited to 1.4 kW by BLTC operations.

The problem of the cooling tubes' azimuthal temperature asymmetry was solved by modeling them as two constant temperature nodes at 432°F (222°C). The heat transfer coefficients from the down flow area to the cold sink NaK were

also taken from that calculation. If the initial average pool temperature is 462°F (239°C), with a 1.5°F (~0.8°C) axial fuel gradient across each fuel region, this model predicts cooling of the sodium pool to an average steady-state temperature of approximately 440°F (227°C) within 35 hours. In addition, the ΔT (hottest interior node to coldest down flow node) = 1.5°F (~0.8°C) and ΔT axial (each ring) = 2°F (~1.1°C) and ΔT (hottest interior node to cold sink) = 10°F (5.6°C). Thus, the effect of natural convection is evident.

4.5 REFERENCES

1. Fuels Storage Facility System Design Description, N106SDD210001, Rev. 1, Atomics International, Canoga Park, CA, April 1978.
2. Equipment Specification, Storage Vessel Subsystem - Fuel Storage Facility, N106NV220001, Atomics International, Canoga Park, CA, February 1977.
3. Seismic Design Criteria for the Fast Flux Test Facility, JABE-WADCO-02, Revision 2, Hanford Engineering Development Laboratory, Richland, WA, July 1975.
4. Final Stress Analysis Report of the Fuel Storage Facility (FSF) Vessel Subsystem, N106SR220002, Atomics International, Canoga Park, CA.
5. ASME Boiler and Pressure Vessel Code, Section III, Division I, Subsections NC and NF, The American Society of Mechanical Engineers, New York, NY.
6. Fast Flux Test Facility Final Safety Analysis Report, HEDL-TI-75001, Hanford Engineering Development Laboratory, Richland, WA, December 1975.
7. Criticality Calculations for the Final Design of the FSF Storage Vessel, N106TI210005, Atomics International, Canoga Park, CA, June 1977.

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5.0 HEAT REMOVAL SYSTEM NaK HEAT TRANSFER LOOPS

5.1 SUMMARY DESCRIPTION

The NaK heat transfer loops remove decay heat from the storage vessel through internal NaK coolers to the NDHX's, where the heat is rejected to the atmosphere. The system is shown in Figure 5.1-1.

The heat is removed by natural convection of the NaK in the heat transfer loops and by natural circulation of air through the NDHX's. Two heat transfer loops provide independent and redundant heat removal paths; each NDHX is sized for a maximum heat load of 205 kW. Heat rejection rates are varied by manually adjusting the air flow through the NDHX.

Since two physically separate heat transfer loops are provided, no single failure in a NaK heat transfer loop prevents the removal of decay heat from the vessel under normal conditions nor will any failure in one loop simultaneously affect the other. In addition, the operational pressure of the NaK loop is greater than that of the storage vessel system to reduce the contamination of the NaK by the sodium in the event of a leak in the in-vessel Na/NaK heat exchanger.

The heat transfer loop piping and components, except the NDHX stacks, are constructed to the requirements of ASME Section III, Class 3 (Reference 1), and also to withstand the FSF DBE. All liquid metal containment welds are examined to Class 2 requirements. They are housed within tornado-hardened structures. The loss of the stack above the hardened structure will not impair the capability of each heat transfer loop, with natural convection air flow through the NDHX, to maintain the vessel sodium temperature below 800°F under the maximum vessel heat load of 205 kW (see Reference 2). An electrically driven standby blower on each NDHX can be used to assist the natural convection air flow under emergency conditions, providing additional backup heat removal capability, but it is not required subsequent to DBE.

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FIGURE 5.1-1. Heat Removal System.

5.1.1 Flow Diagram

A schematic flow diagram of one of the heat removal loops is shown in Figure 5.1-1. Normally, each of the two NaK loops function independently to prevent a failure in one loop from affecting the other.

5.1.2 Piping and Instrument Diagram

Although two NaK coolers (in-vessel loops) are located inside the vessel, they are part of the NaK heat transfer loops. The pressure of the NaK at this sodium/NaK interface is maintained at least 5 lb/in.² gage (35 KPa) above the sodium pressure to prevent possibly contaminated sodium from entering the NaK loops in the event of a leak in the cooler.

5.1.3 Elevation Drawing

Figure 5.1-2 shows the elevations of the heat transfer loops. Decay heat is removed from the storage vessel at a low point in the system and rejected to the atmosphere at the high point. The density difference between the hot and cold legs of the loop creates a pressure gradient large enough to circulate NaK by natural convection.

All NaK piping entering the vessel passes through the vessel and guard vessel boundaries at points above the maximum sodium operating levels to minimize the chance of serious leaks.

The two NaK coolers, which are part of the NaK heat transfer loops are integral with the primary vessel. They are 3 in. (0.08 m) carbon steel pipes installed around the inside perimeter of the vessel between the vessel wall and the turntable above the active portion of the fuel, but below the sodium low-leak level.

FIGURE 5.1-2. Convective NaK Loop (Design Conditions for 1 Loop).

The components of each NaK loop, except stacks, are housed either in separate pipe trench vaults or in separate rooms. The loop components are fabricated to ASME Section III (Reference 1), and are seismic-hardened. The surrounding vaults and trenches are designed and constructed according to ACI Code and are seismic, tornado, and missile hardened.

5.2 INTEGRITY OF NaK HEAT TRANSFER LOOP

5.2.1 Design of NaK Heat Transfer Loop

The NaK heat transfer loops are designed to meet the following requirements:

- 1) Operation is within the imposed temperature limitations (Table 5.2-1) at heat loads from zero to 205 kW.
- 2) No single failure will prevent the removal of decay heat from the storage vessel when operating under any normal operating conditions.
- 3) Radioactive materials are prevented from reaching the environment except under controlled conditions.
- 4) Provision is made for maintenance of the loops without impairment of function.
- 5) Provisions are made to permit startup and shutdown.
- 6) The NaK loops are independent and redundant.
- 7) The system is designed to withstand the FFTF site Design Basis Earthquake (DBE) and FFTF site tornadic force inputs.
- 8) The fuel and vessel temperatures are maintained below 600°F (315°C) in all but faulted conditions (Table 5.2-2). The system design temperature is 650°F (343°C).
- 9) All NaK piping and components meet ASME Section III, Class 3. All NaK containment welds are examined per Class 2 requirements.

- 10) The system is designed for a service life of 20 years.
- 11) The system is protected from overpressurization by relief valves in the argon system.
- 12) The heat removal loops components will be subject to the following environment:
 - a) Air atmosphere, temperature 150°F (66°C) - all equipment in the storage vessel and cold trap vaults
 - b) Air, temperature range of -10° to 103°F (-23° to 39°C) - all equipment in the NaK pipeways and the NDHX vaults.
- 13) Components subject to wear or with limited lives are replaceable.
- 14) Liquid metal leak detection equipment is provided.
- 15) Provision for thermal expansion in the heat transfer loops is incorporated.
- 16) The basic design philosophy provides a facility needing as little operator attention as possible and, at the same time, minimizing the need for automatic controls.

TABLE 5.2-1

HEAT REMOVAL SYSTEM (HRS) TEMPERATURE LIMITATIONS

<u>Heat Load</u>	<u>Operational Mode</u>	<u>Vessel Temperature</u>	<u>Condition</u>
0 - 205 kW (at NDHX)	Two Loops	<500°F (260°C) >300°F (149°C)	Design Point
0 - 205 kW (at NDHX)	One Loop	<600°F (316°C) >300°F (149°C)	Design Point
0 - 144 kW Decay Heat	Two Loops	<500°F (260°C) >300°F (149°C)	Normal

TABLE 5.2-2

CONDITIONS AFFECTING HEAT REMOVAL SYSTEM

Condition	Category
Vault cooling inoperative	HRS status remains normal
Loss of NDHX stacks; backup blowers operational	Upset
Loss of electrical power	HRS status remains normal*
One loop down (other than for maintenance)	Upset
Both loops inoperative	Faulted
Sodium leak	Upset
Normal operation plus DBE	Faulted

5.2.2 Overpressure Protection

Pressure relief valves in the argon system protect the NaK loops from overpressure. The internal design pressure of the loops is 50 lb/in.² gage (345 kPa). Valves relieving the NaK expansion tank cover gas are vented to the atmosphere. Rupture discs are installed ahead of the relief valves to prevent liquid metal from depositing on the valve seats. The relief system also conforms to ASME Code Section III, Class 3, requirements. Pressure relief is set at 35 lb/in.² gage (241 kPa) compared to normal operating pressure of 5 to 10 lb/in.² gage (34 kPa to 68 kPa). However, initial cover gas pressure will not be increased to the operating point of 5 lb/in.² gage (34 kPa) before the NaK is heated to the cold trapping temperature of ~600°F (316°C), since the increase in NaK

* If this condition occurs at low decay heat (requiring use of heaters for temperature maintenance), this would be in upset category.

volume during operation may otherwise inadvertently rupture the discs. Other than temperature change in the storage vessel sodium, there is no event which causes simultaneous pressure change in both NaK loops. Because of the large heat capacity of the storage vessel and sodium, the temperature changes and, thus, the pressure changes will be very gradual. In no case would the relief pressure be reached even if the sodium reaches the 800°F (427°C) faulted condition of the storage vessel.

5.2.3 General Material Considerations

The material for the stainless steel Type 304 components will meet the requirements of the ASME Code Section III, Class 3.

5.2.3.1 Austenitic Stainless Steel -- Cleaning and Contamination Protection

Cleaning may be by mechanical means; by solvents; or by using water, steam, or detergent, cleaning agents, or any combination thereof. However, the following restrictions for austenitic steels shall be applied:

- 1) Halogenated cleaning agents and solvents, except Trichlorotrifluoroethylene (TCTFE) conforming to MIL-C-81302, shall not be used for cleaning or degreasing of austenitic steel.
- 2) Mechanical cleaning tools such as grinding, polishing, filing, deburring, and brushing tools shall be clean and shall not have been used on aluminum, copper, lead, or materials containing lead or lead components, or other low melting point materials. One set of tools not previously used on carbon or low alloy steel shall be maintained, clearly marked, and used only on austenitic steel.
- 3) Only resin-bonded or vitrified grinding wheels shall be used on austenitic steel. When rubber-bonded grinding wheels are used on stainless steel surfaces, all traces of residue shall be removed by subsequent cleaning.

- 4) Brushing of austenitic steel shall be performed only with austenitic steel brushes.
- 5) Austenitic stainless steel that has become sensitized during heat treatment, welding, or forming shall be solution-annealed or shall be protected against corrosion and moisture during all subsequent manufacturing, testing, and storage operations.

For further details, refer to Reference 3.

5.2.4 Leak Detection System

The aerosol, product-of-combustion (POC) detectors that are a part of the general fire protection system sense airborne liquid metal or products of combustion. They are installed in the pipe galleries and the NDHX cells where a NaK leak will react with the air. Leak detector type and position are given in Table 5.2-3.

All detectors have on-line test capability. Annunciation is provided in the FSF control room. All alarms activate the smoke or liquid metal leak alarm channels to the FFTF control room.

Each NaK expansion tank is provided with a thimble-mounted, inductive-level probe which, when below normal, indicates NaK leaks. This monitoring system is provided with backup power.

Although the likelihood of a single NaK leak appears to be credible (probability of occurring more than once in 10^6 years), the redundancy attained through the use of the two separate heat removal paths makes the permanent loss of the heat removal system not credible. (For further details of the leak probability analysis, refer to Reference 4.)

An optical-type smoke detector is mounted in the NDHX stack. Smoke detection will be annunciated in the FSF control room. The alarm activates the smoke or

liquid metal leak alarm channel to the FFTF control room. Also, a smoke detector is mounted in each NDHX vault/pipe trench.

5.2.5 Inservice Inspection

Inservice inspection is not required.

5.3 THERMAL AND HYDRAULIC SYSTEM DESIGN

Decay heat from the storage vessel is transferred to the NaK loops at the NaK coolers. The NaK flows by natural convection between the NaK cooler and the NDHX. Flowing through the NDHX, the hot NaK is cooled by ambient air, -10°F to 103°F (-23° to 39°C), flowing over 1 in. (0.025 m) diameter finned tubes. The air circulates through the NDHX by the natural draft created by the stack. If the stack is damaged, the heat can still be rejected through the shortened stack (Reference 2) and, in addition, a backup blower capable of delivering additional air through the NDHX is also provided.

Thermal expansion of the NaK is accommodated by a NaK expansion tank located at the high point of each loop.

In the operating mode, the system maintains storage vessel temperatures between 300°F (149°C) and 600°F (316°C) at heat loads varying from zero to 205 kW and an ambient air temperature range of -10°F to 103°F (-23° to 39°C). The calculated expected maximum heat load for the vessel is 151 kW with a full load of 466 fuel assemblies. Heat losses from the system were not considered in sizing the system components, but they were taken into account in evaluating system performance.

Performance data for the NaK loop is shown in Table 5.3-1 for various normal and design conditions. Sufficient heat transfer area is provided between the sodium and the NaK cooler such that the NaK outlet temperature approaches the sodium temperature within 1°F (0.6°C).

TABLE 5.2-3

FSF LIQUID METAL LEAK DETECTORS

Location	Type	Purpose	Testing Capability
1. Sodium storage vessel/ guard tank annulus	Conduction probe	Large sodium leak	Manual continuity check at instrument panel
2. Isolation valves (two) on sodium purification loop	Conduction probe (one/valve)	Bellows rupture	Manual continuity check at instrument panel
3. Bottom of cold trap cooling shroud	Conduction probe	Large sodium leak	Manual continuity check at instrument panel
4. Sodium storage vessel/ guard tank annulus	Hydrogen gas analyzer	Small pinhole leak	Built-in test capability with standardized gas sample
5. Sodium heater	Conduction probe	Large sodium leak	Manual continuity check at instrument panel
6. NDHX stack	Optical smoke detector	Small NDHX leaks	
7. Throughout building	POC monitors	All leaks	Fail safe
8. Sodium storage vessel/ guard tank annulus	Plugging filter	Small/intermediate leaks	Flowrate and ΔP indication

Detailed discussions of the system performance in both normal and off-normal operational modes are given in Reference 5.

5.4 STORAGE VESSEL AND APPURTENANCES

(Refer to Chapter 4.)

5.5 COMPONENTS AND SUBSYSTEMS

5.5.1 NDHX

Decay heat removed from the storage vessel is rejected to the atmosphere by natural circulation of air through the NDHX's. Although maximum expected heat load in the vessel is 151 kW, the design heat load of each NDHX is 205 kW. Variations in the heat loads are accommodated by manually varying the air flow rate through the NDHX. The NDHX's pressure boundaries are designed to Section III, Class 3, and the stacks and NDHX fans are designed to manufacturer's standards. Both NDHX's and fans are housed in tornado-hardened areas and are Seismic I components.

The NDHX is a cross-flow NaK-to-air heat exchanger consisting of four banks of 15 finned tubes 46 in. (1.2 m) long. The tubes are 1 in. (0.025 m), 0.083 in. (2.1 mm) wall, Type 304 stainless steel. A 48 ft (14.6 m) high stack of 24 in. (0.61 m) Schedule 20 pipe produces the necessary draft. A schematic of the NDHX is shown in Figure 5.5-1. The design pressure of the heat exchanger section is 50 lb/in.² gage (340 kPa) and of the NDHX structure is 0.5 lb/in.² gage (3.4 kPa).

The backup fan, which may be used in the event of a stack damage, has 2300 scfm air flow capacity. Table 5.5-1 shows the important design parameters.

NDHX's are provided with the following auxiliary equipment and features:

TABLE 5.3-1

HEAT REMOVAL SYSTEM PERFORMANCE

Event Description	Performance Parameter							
	Net Heat Load (kW)	Air Temperature °F (°C)		Air Flow Rate Per Loop scfm (l/s)*	NaK Temperature °F (°C)		NaK Flow Rate Per Loop gpm (l/s)	Sodium Temperature °F (°C)
		Inlet	Outlet		Cold Leg	Hot Leg		
Design Point - 2 Loops, 205 kW No Heat Losses	205	103 (39)	345 (174)	1300 (615)	300 (149)	453 (234)	25.3 (1.6)	453 (234)
Design Point - 1 Loop, 205 kW No Heat Losses	205	103 (39)	404 (206)	2100 (990)	341 (171)	582 (305)	33.1 (2.1)	583 (306)
Normal - 2 Loops, 151 kW With Heat Losses	114	103 (39)	335 (168)	750 (355)	300 (149)	405 (207)	20.4 (1.3)	405 (207)

* Standard conditions are: 1 atm and 60°F (15°C)

FIGURE 5.5-1. Natural Draft Heat Exchanger (NDHX).

- 1) Backup power source (gasoline-engine-driven generator set) to provide 120 V, 60 Hz power for the NDHX backup blower.
- 2) Smoke detectors in the NaK pipe galleries, NDHX vaults, and the NDHX stacks.
- 3) Equipment to control and indicate the NDHX damper position.
- 4) Temperature monitoring instrumentation for inlet and outlet air of the NDHX.
- 5) Temperature monitoring instrumentation for NaK lines to and from the NDHX (readout device to be seismic hardened).
- 6) Physical and electrical separation of each NDHX.
- 7) Manifold venting to the expansion tank cover gas supply/vent line.

5.5.2 Expansion Tank

The NaK expansion tanks accommodate thermal volume changes in the loops. Each tank has one penetration (1 in.) for the argon cover gas and a second (1 in.) to the NaK heat transfer loop. Pressurized argon cover gas maintains the internal NaK pressure at the NaK coolers at 5 lb/in.² (34 kPa) above the sodium pressure.

The tank volume is approximately 18 gallons (0.07 m³), 1 ft (0.3 m) diameter by 3 ft (0.9 m) high, and the design pressure is 50 lb/in.² gage (340 kPa). The design temperature is 650°F (343°C) temperature. Each tank is designed to ASME Code, Section III, Class 3. All NaK containment welds are examined per Class 2 requirements.

5.5.3 Piping System

The piping is arranged so that one NaK cooler is placed above the other at a storage vessel elevation below the minimum leak level. The vessel nozzles are arranged so that both loops flow in the same direction to maximize thermal performance (in terms of the hot NaK temperature approaching the bulk sodium pool temperature) while retaining the simple geometry of a 360° loop. The cooler characteristics are summarized in Table 5.5-2.

TABLE 5.5-1
NATURAL DRAFT HEAT EXCHANGER SUMMARY

1. Design Configuration	Cross Flow Extended Surface
2. Tube Dimensions	
O.D., in.	1.0 (0.025 m)
I.D., in.	0.834 (21 mm)
Length, in.	46 (1.2 m)
3. Number of Tube Banks	4
4. Number of Parallel Tubes per Bank	15
5. Tube Spacing	
Between Tubes, in.	2.25 (0.057 m)
Between Banks, in.	2.78 (0.07 m)
Pitch	Triangular
6. Fin Dimensions	
O.D., in.	2.0 (0.05 m)
Thickness, in.	0.035 (0.9 mm)
7. Fin Density, Fins per inch	7
8. Stack Dimensions	
O.D., in.	24.0 (0.6 m)
I.D., in.	23.25 (0.59 m)
Height, ft	48.51 (14.8 m) (above the tube bank)
9. Material	
Tubing	304 Stainless Steel
Fins	Copper (brazed)
10. Air Flow Control	Butterfly valve in outlet and adjustable inlet louvers
11. Code Class	Tube bundle designed to ASME, Section III, Class 3, welds examined to Class 2 requirements

TABLE 5.5-2
NaK COOLER CHARACTERISTICS SUMMARY

1. Pipe Dimensions (Schedule 40)	
O.D., in.	3.5 (0.09 m)
I.D., in.	3.068 (0.078 m)
2. NaK Cooler Diameter, in.	249 (6.32 m)
3. NaK Cooler Heat Transfer Length, ft (min)	
Loop I	63.1 (19.2 m)
Loop II	65.1 (19.8 m)
4. Heat Transfer Area (min) sq ft	
Loop I	57.8 (5.4 m ²)
Loop II	59.7 (5.5 m ²)
5. Material	Carbon Steel
6. Code Class	ASME Section III, Class 3, weld examined per Class 2 requirements

5.5.4 Valves

The NaK valves are bellows-type sealed gate valves designed for 125 lb/in gage (860 kPa) pressure and 650°F (343°C). The process and first service connection isolation valves are required to meet Section III, Class 3 of the ASME Code. The service connection backup isolation valves are required to meet ANSI B31.1. These valves were purchased for FFTF and are Section III, Class 1, valves. (The FFTF requirements are more stringent than the FSF's.)

5.5.5 Instrumentation

Each NaK heat transfer loop is provided with flow and temperature instrumentation. NaK flow through each loop is monitored by a permanent magnet flowmeter, installed in the return line with flow indication in the FSF control room.

Temperature measurements are by thermocouples, attached to the process piping. The inlet and outlet temperatures of each heat transport loop are monitored and recorded with out-of-limit annunciation. These temperature alarms activate the FSF process malfunction alarm channel to the FFTF control room. The temperature monitoring instruments are provided with backup power.

As a backup to the above temperature instrumentation, the temperature of NaK piping adjacent to the inlet and outlet nozzle of each NDHX is monitored by thermocouples with local indication at each NDHX. A selector switch is used to select which line (inlet or outlet) is to be displayed. The system is a self-powered type, i.e., the thermocouple generates the signal to directly operate the indicator, and requires no external power. All components of this systems are seismically hardened.

5.6 REFERENCES

1. ASME Boiler and Pressure Vessel Code, Section III, The American Society of Mechanical Engineers, New York, NY.
2. NDHX Emergency Heat Rejection Study, N106TI310009, Rev. A, Atomics International, Canoga Park, CA, July 1977.
3. Cleaning and Packaging, General Requirements, FSF, N106A0610001, Atomics International, Canoga Park, CA, April 1977.
4. Probabilities of NaK and Na Leaks, N106TI230004, Atomics International, Canoga Park, CA, March 1977.
5. FSF Heat Removal System, N106TI300005, Rev. A, Atomics International, Canoga Park, CA, July 1977.

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CHAPTER 6
ENGINEERED SAFETY FEATURES

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6.0 ENGINEERED SAFETY FEATURES

6.1 GENERAL

The FSF is equipped with engineered safety features, the dynamic action of which prevent or limit the consequences of incidents that would otherwise severely impact plant safety. These features include:

- 1) Redundant sodium purification loop isolation valves with fail-closed features and redundant seismic switches. This assures that earthquake-caused damage in the purification loop will not result in significant sodium loss from the fuel storage vessel.
- 2) Self-powered thermocouple readouts for monitoring NaK heat transfer fluid temperature. This feature assures that information to control the decay heat removal process will always be available.
- 3) Hardened NDHX damper controls, which assure that NDHX dampers will not prevent proper operation of the redundant heat removal loops.

6.1.1 Purification Isolation Valves

6.1.1.1 Design Bases

All portions of the sodium purification loop from the fuel storage vessel up to and including the second redundant isolation valves in the cold trap vault are designated Category I and are required to withstand the design basis earthquake (DBE) and tornado effects. The components in these portions of the purification loop are also designed to withstand the full range of environments to which they may be exposed under normal operating and accident conditions, including a sodium fire in the cold trap vault.

6.1.1.2 System Design

Redundant sodium isolation valves placed in the loop at the seismic boundary assure that leakage in nonseismic portions of the sodium purification loop will not result in drawing significant amounts of primary coolant from the fuel storage vessel. Each of these valves is qualified for operation after a DBE. Closure of only one valve in each line is required to isolate the fuel storage vessel from the Category III portion of the sodium purification loop. Each valve of the redundant sets is of the FAIL CLOSE type (closes upon loss of electrical control or pneumatic power source). The electrical control signal to the solenoid-operated pneumatic valve for the pneumatic actuator on each isolation valve is routed through a seismic switch-controlled relay. The qualified seismic switches are set to cut off relay holding coil power when seismic forces exceed Category III (Uniform Building Code Zone 2) limits. The seismic switches are powered by the facility power, without any backup. Thus, isolation valves closure can be initiated manually from the control room, automatically by the seismic switches, or by the loss of facility power. The overall storage vessel isolation system is FAIL SAFE.

6.1.1.3 Design Evaluation

The design of the isolation valves and associated piping conforms to the requirements of the ASME Boiler and Pressure Vessel Code, Section III, for Class 2 components. The valves are qualified to survive the FSF DBE in operating condition. There are two redundant valves in each leg of the purification loop. The valves are both fail-closed type valves. Each valve is independently supplied with a control signal through a seismic switch. There is no single failure which can prevent isolation of the vessel from the sodium purification loop.

6.1.1.4 Tests and Inspection

During operation of the FSF cold trapping, sodium heating and sample taking are expected to occur only on an intermittent basis with the purification loop shut down when not in use. Isolation of vessel sodium from the loop (i.e., closing of the isolation valves) will be verified as a part of the operational procedure. In the event the loop is being operated, the isolation capability of the valves can be verified by momentarily switching the valve selector to "close" and noting the valve position indicators and the flow meters. The functional performance, including the trip point of each seismic switch, will be verified by means of an auxiliary test kit provided.

6.1.1.5 Instrumentation Requirements

The seismic switches required to sense the occurrence of an earthquake in excess of the UBC Zone 2 earthquake are designed to withstand the FSF design basis earthquake. They are designed to meet the requirements for reactor plant protection system components. They are independent, fail-safe on loss of power, and no single seismic switch failure can prevent the safety function (isolation).

6.1.2 Self-Powered NDHX Thermocouple Readouts

6.1.2.1 Design Bases

The self-powered thermocouple readouts on the NDHX are designated Seismic Category I. They are capable of withstanding the environmental extremes of all of the design basis conditions in the NDHX room (see Table 3.7-1). The readouts are protected from the design basis missiles by their location within the NDHX room missile barriers.

6.1.2.2 System Design

The design of the NDHX self-powered thermocouple readouts is similar to the design used in the FFTF and qualified by test for use as engineered safety features. The design bases and environments under which the thermocouple readouts were qualified are more severe than the design basis conditions in the FSF. The design is therefore acceptable for incorporation in an engineered safety feature in the FSF.

6.1.2.3 Design Evaluation

An evaluation of the design of the thermocouple readouts was performed, including shake table testing, and the design was found to be acceptable. The readouts meet the single failure criterion in that an independent readout is provided for each of the two redundant heat removal loops. Either of these loops is capable of removing the full design basis heat load.

6.1.2.4 Test and Inspection

The self-powered NDHX thermocouple readouts are in continuous operation during normal plant operation. The readouts will be inspected and verified periodically by comparison with the normal HRS loop instrumentation and the storage vessel sodium temperature.

6.1.2.5 Instrumentation Requirements

The self-powered NDHX thermocouple readouts do not require any additional instrumentation for their operation during the design basis accident. During normal operation, the verification test is provided by normal facility temperature instrumentation.

6.1.2.6 Materials

The materials of construction of the self-powered thermocouple readouts are not expected to cause any problems with the operation of other systems during any of the accidents hypothesized for the FSF.

6.1.3 NDHX Damper Controls

The majority of the Heat Removal System (HRS) components are passive due to the nature of the convective flow heat transfer concept used in the FSF. The NDHX damper is, however, an active component which must be capable of being placed in the full open position after the DBE, which assumes that the stacks are no longer standing. For this reason, it is treated separately as an engineered safety component.

6.1.3.1 Design Bases

All portions of the heat removal system NDHXs, including the damper controls, must survive the DBE. In addition, all necessary HRS components must be shielded from the effects of tornado missiles. They are also required to withstand the full range of environments to which they may be exposed during normal operation and during the design basis events.

6.1.3.2 System Design

The NDHX dampers are normally a remote manually-controlled component. After a DBE, it may be necessary to move the dampers to the full open position if the NDHX stacks have been damaged. A means of decoupling the damper from the servo motor has been provided in the design so that direct positioning can be manually accomplished at the damper. Timely post-DBE access to the damper controls is provided as described in Section 15.2.6 and Appendix 15.8. The damper and NDHX themselves are Seismic Category I devices designed to withstand the DBE.

6.1.3.3 Design Evaluation

The design on the NDHX damper assures that even if hardware failures occur in the damper servo motors or if there is a loss of power, the dampers can still be brought to a fully open position, if necessary, after a DBE. Furthermore, each of the redundant HRS loops is provided with an independent NDHX and damper to assure that any single failure in an NDHX damper will not prevent removal of the entire decay heat load.

6.1.3.4 Testing and Inspection

The motion of the NDHX dampers is tested during normal operation as the dampers are adjusted to the varying heat loads in the FSF. An inspection port in the side of the NDHX allows visual confirmation that the damper is operating as desired. Functional confirmations of the operation of the NDHX can be obtained from the normal facility temperature instrumentation.

6.1.3.5 Instrumentation Requirements

In the event of a DBE, functional confirmation that the NDHX dampers are allowing heat removal through the NDHX can be obtained from the Seismic Category I self-powered thermocouple readouts attached to the NDHX inlet and outlet pipes.

6.2 ADDITIONAL DESIGN FEATURES

In addition to the engineered safety features, the Fuel Storage Facility is equipped with design features that either prevent or limit the consequences of incidents that would otherwise have some impact on plant operation or safety. These features include:

- 1) Redundant natural convection cooling loops which allow decay heat removal without the need for external power sources. The physical separation and independence of the two heat transfer loops prevents single failures from stopping heat removal capability.
- 2) A guard tank surrounds the storage vessel in the storage vessel vault, and the sodium purification loop lines are provided with a siphon breaker to prevent the storage sodium from falling below the level of the fuel assembly core region in the event of leaks in the storage vessel. This allows decay heat removal to continue even in the event of substantial sodium leakage from the primary vessel.
- 3) A cover gas holdup system provides control over the release of possible contaminated gas to the atmosphere. This system allows the facility to maintain releases within the concentration guides of ERDAM 0524 (Reference 1) even under a majority of the accident conditions.
- 4) Backup fans and power supply are provided for the natural draft heat exchangers to enhance continued heat removal in the event of stack damage.
- 5) Fire suppression catch pans reduce the hazards that may be associated with a liquid metal leak or spill.

6.2.1 Redundant Natural Convection Cooling

6.2.1.1 Design Bases

All portions of the NaK heat removal loops are designated Category I and are designed to withstand the design basis earthquake and tornado. The NDHX stacks, which are attached to the building exterior, are designed to withstand the 100-year environmental extremes. Each of the redundant loops is designed to be protected from the consequences of any event in the other loop. Each loop is designed to handle the entire 205-kWt maximum heat load as a design condition. This heat load would be present only in the unlikely event of an FFTF core dump. The radiation environment of the heat removal loop components is not severe enough to cause any materials problems.

6.2.1.2 System Design

The design of the heat removal system loops is described in detail in Chapter 5 of Reference 2, and in Reference 3.

The heat removal loops are physically separate systems and are isolated from each other by fireproof and earthquake and tornado-hardened barriers to prevent accidents or single component failures in one loop from affecting the other. Primary vessel sodium is cooled by natural convection without the use of any exterior power source, thus eliminating any significant adverse effect on the facility because of loss of electrical power.

6.2.1.3 Design Evaluation

The design of the natural convection NaK heat removal loops was evaluated with the aid of a computer model of the system. Heat transfer, flow rate, density, and heat loss calculations were balanced to evaluate system performance under a variety of conditions. The computer code used for this purpose is described in Appendix B of Reference 3.

The reliability of this type of natural convection system is high as evidenced by the perfect performance record of the EBR II shutdown coolers over a period of 12 years. The EBR II coolers are similar in concept and sizing to the FSF heat removal system.

Before operation of the FSF, the heat removal system will be tested during the acceptance testing program. At this time, the heat removal system capabilities will be verified as a part of facility qualification.

6.2.1.4 Tests and Inspection

During operation of the FSF, the heat removal loops will be in continuous operation. Flow and temperature measurements in the loops will be continuously available for examination. Thus, verification that the natural convection heat removal loops are operating properly is continuously present and formal readiness inspections are not warranted.

6.2.1.5 Instrumentation Requirements

The design of the natural convection heat removal loops allows operation with minimal operator intervention. The only instrumentation required for operation is inlet and outlet temperature indications of the NaK loops. This is provided by thermocouples on the pipes and temperature indicators in the control room. As a backup system, a self-powered seismic-hardened thermocouple readout is located in the NDHX room.

6.2.2 Guard Vessel and Siphon Breakers

The fuel storage vessel is provided with a guard vessel and siphon breakers to prevent the sodium level from dropping to levels that would impair heat removal system performance in the event of a leak.

6.2.2.1 Design Bases

The design of the guard tank around the storage vessel and the siphon breakers in the purification loop lines prevents the sodium level in the primary vessel from falling below the level of the active core region of the stored fuel assemblies in the event of a leak under the normal operating pressures in the storage vessel. The tank can withstand the DBE and tornado missile effects. The tank is capable of containing liquid sodium at temperatures of up to 650°F (343°C).

6.2.2.2 System Design

The guard vessel is constructed according to the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Class 3. The upper lip of the tank is positioned at an elevation sufficient to support a sodium column equivalent to the pressure difference between the storage vessel cover gas and the inert vault atmosphere. A siphon breaker in the purification loop prevents lowering of the sodium below the minimum leak level at elevation 536 ft, 7 in. (163.4 m). The tank is in a below-grade vault to provide tornado missile protection. Specific details of the design of these components can be found in Reference 2 and in the storage vessel assembly stress report, Reference 4.

6.2.2.3 Design Evaluation

Considerable experience over a number of years has shown the guard tank approach to be an acceptable method of limiting the effects of leakage in vessels. The selection of the ASME Code, Section III, Class 3, requirements as a design guide assures that the guard tank will perform as required.

The sizing of the tank was verified by calculating the equilibrium sodium levels for a variety of hypothesized leakage events. Details of these calculations are presented in Reference 5. The siphon breakers will consist of a simple hole drilled in the purification loop downcomer in the primary vessel. This hole will be located below the normal sodium operating level to prevent blockage.

6.2.2.4 Test and Inspection

The guard tank and siphon breakers will be tested and inspected according to the requirements of the ASME Code.

6.2.2.5 Instrumentation Requirements

The vessel/guard tank annulus is provided with a hydrogen sniffer, a plugging filter and conduction types of liquid metal leak detectors. These detectors are removable for periodic inspection and maintenance. In the event of a massive sodium leak into the guard tank annulus, the sodium level therein could be estimated from the indicated storage vessel sodium level and the indicated differential pressure between that of the storage vessel cover gas and of the storage vessel vault atmosphere. There are no instrumentation requirements associated with the siphon breaker.

6.2.2.6 Materials

The guard tank is constructed of carbon steel. This material is not expected to cause any problems with the operation of other systems during any of the accidents hypothesized for the FSF.

6.2.3 Argon Holdup System

The FSF argon holdup system provides a means for monitoring and controlling the release of gaseous radionuclides.

6.2.3.1 Design Bases

The argon holdup system is capable of controlling the release of cover gas from the storage vessel under all normal operating conditions.

Components of the holdup system are capable of withstanding a tornado without significant leakage of radionuclides. Release of radionuclides following a DBE will not exceed established criteria.

6.2.3.2 System Design

The holdup tank in the argon cover-gas system is operated at a vacuum of between -3 and -8 lb/in gage (-21 and -55 kPa). It is designed to be of sufficient volume to store all of the exhaust from one fuel transfer operation. At the end of the cycle, the contents of the holdup tank can be monitored before release to the atmosphere. The holdup tank is constructed to Section III, Class 3, of the ASME Code. Its below-grade location prevents adverse tornado or missile effects. Further details of the design of the argon holdup system are supplied in the FSF Argon System Description, Reference 6.

6.2.3.3 Design Evaluation

The specification of ASME Code for the holdup tank assures that its design will be adequate for its intended function. Sizing of the holdup tank assures sufficient capacity to handle the maximum expected effluent from a single fuel handling cycle. Sufficient time is required (approximately 8 hours) between cycles to allow monitoring of the holdup tank and pump down before the start of the next fuel handling cycle.

6.2.3.4 Testing and Inspection

Periodic, regular usage of the system constitutes an adequate test program during the life of the FSF. In addition, routine visual inspection of the components and operational records will verify that the system has not sustained any apparent degradation.

6.2.3.5 Instrumentation Requirements

The holdup tank has a radiation monitor to determine the activity of the cover-gas effluent before release. In addition, there is a second monitor in the effluent line that is used to record the integrated release for the operational record.

6.2.3.6 Materials

The materials of the holdup tank and associated piping are carbon steel. The radiolytic and pyrolytic decomposition products of these components are not expected to cause interference with any safety-related system even under the most severe accident conditions.

6.2.4 Backup Powered Operations

In the event of a design basis earthquake or tornado, the NDHX stacks above the NDHX cell roof may be damaged such that their effectiveness as natural draft chimneys is reduced. Although a single NDHX, with its stack truncated at the cell roof level, is capable of maintaining the storage vessel sodium temperature below the vessel faulted temperature (Reference 7), the built-in blower in the NDHX can be operated to increase the cooling capacity. A gasoline-engine-driven electrical generator is provided in each NDHX cell for powering the blower if the facility power is also lost.

6.2.4.1 Design Bases

The electrically powered blower and the motor/generator set are selected for their ability to maintain their structural integrity, including the engine fuel system, under the Seismic Category I conditions. This insures that there will be no impact on other associated Category I equipment in the vicinity. Being placed in the tornado-hardened NDHX vaults, these components are protected from any consequences of tornado events. The operation of the backup fan is not essential to the safety of the FSF, and thus these components are

not required to be functional after the seismic event. Likewise, the IEEE-IE emergency power requirements are not applicable.

6.2.4.2 System Design

The backup fans and generators were selected from commercially available units and were tested for the DBE. The location of the units in the NDHX rooms will provide tornado and missile protection. Mounts for the fan and generator will restrain the fan and generator during startup, operation, and earthquake conditions.

6.2.4.3 Design Evaluation

The backup fans and the generators that supply them with power are each sized to produce 2500 ft³/min (1.2 m³/s) airflow over the NDHX tube bundles. This will allow dissipation of 205 kW from each of the heat removal loops. The use of separate generators and fans for each of the NDHX's assures that heat removal system redundancy is preserved, and that a single component failure cannot prevent the heat removal system from performing its function.

6.2.4.4 Testing and Inspection

The backup generators and fans will be scheduled for periodic testing as a part of normal FSF routine maintenance. The test will include starting the generator and adding a load to the circuit by switching the fans on. The system will be allowed to run this way until steady-state is reached, and it is verified that the entire system is operating properly. The generators will then be turned off and the fuel supplies topped off.

6.2.4.5 Instrumentation Requirements

The heat removal loop instrumentation will be used to determine whether the backup generators and fans are operating properly by monitoring NaK coolant temperatures.

6.2.5 Vault Catch Pans

Catch pans are used in the storage vessel vault and the cold trap vault to limit the consequences of a fire if there should be a sodium leak.

6.2.5.1 Design Bases

The catch pans are designed and fabricated to contain the sodium from any spill and to limit the amount of surface that could react with the surrounding atmosphere.

6.2.5.2 System Design

The catch pans consist of steel basins sized for the maximum leak and placed beneath the sodium-containing components. The catch pans are provided with fire suppressing cover plates.

6.2.5.3 Design Evaluation

The catch pans are not safety-related systems because they serve a backup role to the Category I sodium components and piping. The limited air ingress and fire suppression catch pans in the below-grade vaults provide an additional margin to limit significant combustion (Reference 8).

6.3 REFERENCES

1. ERDA Manual, "Standards for Radiation Protection", Chapter 0524,
US Department of Energy, Washington, DC, March 1977.
2. Fuel Storage Facility System Design Description, N106SDD210001,
Rev. 1, Atomics International, Canoga Park, CA, April 1978.
3. FSF Heat Removal System Description, N106TI300005, Rev. B-1, Atomics
International, Canoga Park, CA, June 1978.
4. Final Stress Analysis of the FSF Vessel Subsystem, N106SR220002,
Rev. NC-1, Atomics International, Canoga Park, CA, September 1978.
5. Determination of Minimum Sodium Leak Level in the FSF Storage Vessel,
N106TI220005, Rev. C, Atomics International, Canoga Park, CA,
June 1978.
6. FSF Argon System Description, N106TI300007, Rev. B, Atomics Inter-
national, Canoga Park, CA, May 1978.
7. NDHX Emergency Heat Rejection Study, N106TI310009, Atomics International,
Canoga Park, CA, June 1978.
8. K. Weston, Alkali Metal Fire and Dose Analysis for FSF Preliminary
Safety Analysis Report, N106TI230006, Rev. A, Atomics International,
Canoga Park, CA, October 1977.

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THE
INSTRUMENTATION AND CONTROL
SYSTEMS
OF THE
FACILITY

CHAPTER 7 INSTRUMENTATION AND CONTROL

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7.0 INSTRUMENTATION AND CONTROL

7.1 INTRODUCTION

The Electrical, Instrumentation and Control (EI&C) System is designed for fail-safe manual operation. Annunciation and warning lights are provided for events that require operator intervention or corrective action to maintain system operation within normal range. Remote alarms signifying off-normal conditions are provided to the FFTF control room by three grouped alarm channels:

- 1) Process Malfunctions
- 2) High Radiation Levels
- 3) Smoke or Liquid Metal Leaks

7.1.1 Safety-Related Systems

The seismic switches and the self-powered NDHX inlet and outlet temperature indicators are the only components of the EI&C System required for the safe operation of the FSF. The balance of the EI&C system provides monitoring and diagnostic information of the following FSF parameters:

- 1) Radiation
- 2) Oxygen concentration in accessible below-grade vaults
- 3) Liquid metal leak detection
- 4) Storage vessel sodium low-level and sodium temperature
- 5) NDHX inlet and outlet temperature (in addition to the safety-related temperature indicator)

7.1.2 Identification of Safety and Diagnostic Functions

The engineered safeguard seismic switch operation is required to maintain the FSF safety under the design basis earthquake accident conditions. The function of the switches is to provide an OPEN in the electrical circuit which

controls the redundant isolation valves on sodium inlet and outlet pipes from the storage vessel. The loss of electrical continuity initiates the closure of the FAIL CLOSE valves. The electrical continuity initiates the closure of the FAIL CLOSE valves. The electrical circuitry associated with the switches and the valves are designed so that no single failure will affect the safety function. The switches are qualified by testing under environmental conditions equal to or more severe than the FSF DBE. These switches are located in the tornado-proof subgrade electrical equipment room.

The safety-related NDHX inlet and outlet temperature indicating systems provide data to assess the performance of the storage vessel heat removal system. The temperature indicator systems are of self-powered design, and have been seismically qualified by tests to meet FSF seismic requirements. The components are located in the tornado-proof NDHX vaults.

The balance of the instrumentation are not safety-related, but they do provide useful information on the operation of the FSF. The significant instrumentations are described below.

Radiation monitors are placed at various locations (Figure 7.1-1) to detect and annunciate above-normal radiation levels. Levels that exceed the high trip point will trip FSF building evacuating alarms. Radioactive effluent released from the argon hold-up tank is continuously monitored to control release rates within the limits imposed by ERDAM 0524.⁽¹⁾ The FSF storage vessel is provided with a neutron monitoring system for diagnostic purposes. Neutron detection for diagnostic purposes is accomplished by BF_3 type detectors providing high neutron sensitivity. These monitors are described in Chapter 11.

All routinely accessible areas that have a risk of an oxygen deficient atmosphere (<19.5%) have permanent oxygen monitors. These areas include the vault cooling equipment room, the argon hold-up tank vault, electrical equipment room, and subgrade corridors. All such nonroutinely entered areas are posted and require personal oxygen monitors.

FIGURE 7.1-1. Radiation Monitors.

The FSF is provided with liquid metal leak detectors in areas where sodium or NaK spills are possible. A variety of leak detectors, (conduction type, H₂ analyzer, optical smoke, and POC's) are located as shown in Table 5.2-3. All detectors have on-line test capability. Annunciation is provided in the FSF control room and all alarms activate the smoke or liquid metal leak alarm channels to the FFTF control room.

The storage vessel is equipped with a level-sensing probe and a low-level trip which will alarm if there is significant sodium leakage. Temperature sensing is also provided and alarmed to notify operations personnel that corrective action may be necessary. Both of these alarms are retransmitted to the FFTF control room as process malfunction alarms.

The NDHX inlet and outlet temperatures are additionally monitored by conventional temperature sensors to verify that the decay heat removal is proceeding as required for plant operation. High-temperature alarms are provided in the FSF control room and retransmitted to the FFTF control room on the process malfunction circuit.

Annunciation and warning lights are provided for events which require operator intervention or corrective action to maintain system operation within normal ranges. Remote alarms signifying off-normal conditions are provided to the FFTF control room by three grouped alarm channels. These are Process Malfunctions, High Radiation Levels, and Smoke or Liquid Metal Leaks. A separate fire alarm channel is also provided.

The FSF is provided with an intercom/PA system capable of placing the FFTF control room in contact with any of the FSF building areas.

To minimize the chance of a single credible event disabling important FSF EI&C System operation, electrical separation is maintained from the point of the equipment to either the Motor Control Center (MCC) or control room depending upon the nature of the system. The use of PVC and Teflon wire insulation is prohibited. All wiring conforms to the Insulated Power Cable Engineers Association (IPCEA) Vertical Flame Test, outlined in IPCEA S-61-402 (Reference 2).

7.2 SAFETY CONTROL SYSTEMS

There is no safety-related EI&C control system except as noted in 7.1.1. The control systems allow remote operation of equipment and provide protection against equipment damage in the event of malfunctions; however, the failure of any related equipment (heaters, motor controllers, etc.) does not affect the ability of the FSF from maintaining safe fuel storage.

7.3 NONSAFETY CONTROL SYSTEMS

The control systems on the FSF adjust process parameters and prevent undesirable action in the event of malfunctions. The design of the suction line to the purification loop and syphon breaker in the return line is such that sodium cannot drain below the stored fuel in the event of damage to the purification loop. Manual closing of the isolation valves provides backup to this system.

The storage vessel turntable is rotated by a motor control system. This system is operated by a control pendant. A dead-man feature prevents unattended operation. Interlocks prevent turntable rotation during fuel transfer operations.

The holding tank vacuum pump is equipped to automatically shut off when the tank pressure drops to -8 lb/in.^2 gage (-55 kPa).

Each NDHX has a damper control system. A backup blower ensures adequate heat removal in the event of NDHX stack damage.

7.4 REFERENCES

1. ERDA Manual, "Standards for Radiation Protection," Chap. 0524, US Department of Energy, Washington, DC, March 1977.
2. Thermoplastic-Insulated Wire and Cable for the Transmission and Distribution of Electrical Energy, IPCEA-NEMA Standards Publication, IPCEA Pub. No. S-61-402 (Third Edition), NEMA Pub. WC 5-1973, National Electrical Manufacturers Association, New York, NY, 1973.

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CHAPTER 8
ELECTRICAL POWER

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8.0 ELECTRICAL POWER

8.1 INTRODUCTION

The Fuels and Materials Examination Facility (FMEF) substation provides 13.8-kV feeder power to the FSF via a 13.8-kV/480-V, 3-phase, 60-Hz, 1000-kVA outdoor transformer. The 480-V power is provided to the FSF Building Motor Control Room, where the 480-V power distribution center is located. The distribution center contains the circuit breakers necessary to supply power to the motor control centers and heater control center.

8.2 OFF-SITE POWER

8.2.1 Design Basis

The electrical power for FSF is supplied via an underground 13.8-kV feeder from the FMEF substation (451B), which in turn is continuously supplied by Bonneville Power Administration (BPA) service to the Hanford 400 Area, Figure 8.2-1.

The arrangement of the electrical power system provides for an adequate maintenance and testing program with no significant reduction in operating efficiency. Equipment used is standard and interchangeable to the maximum extent practical.

8.2.2 Design Parameters

Power at 1000 kVA is continuously available through a 15-kV, 1200-ampere, 500-MVA line circuit breaker, which is located in the 451B substation and provides protection for the underground 13.8 kV feeder to FSF substation. In addition, a 15-kV air interrupter switch located at the FSF substation isolates the 1000 kVA transformer when required. Design parameters are as follows:

FIGURE 8.2-1. FMEF Substation 451B.

- 1) Voltage nominal: 13.8 kV
- 2) Voltage minimum: 13.11 kV
- 3) Voltage maximum: 14.52 kV
- 4) Frequency: 60 Hz $\pm$ 0.02 Hz
- 5) Capacity, transmission line: 1 MVA

Power is delivered to a 13.8-KV, 480-V, 3-phase, 60-Hz, 1000 kVa substation located north of the facility.

8.2.3 Service and Transient Conditions

The electrical system is constructed and equipped with protection in accordance with industry standards for environmental and operational transient conditions prevalent in the area in regard to wind, ice loading, temperature, lightning, flooding, seismic, and switching surges.

8.2.4 Documentary Criteria

- 1) NESC, Part 2, Safety Rules for the Installation and Maintenance of Electrical Supply and Communication Lines
- 2) HWS-10000, The Fuel Storage Building and Utilities, Project A-003
- 3) HWS-10006, Hanford Standard Design Criteria (used as a guide only)
- 4) State of Washington, Department of Labor and Industries, Rules and Regulations for Electrical Construction
- 5) NEC (National Electric Code) (used as a guide only)

8.2.5 Design Description

Electrical power for the Fuel Storage Facility (FSF) is supplied from the Fuels and Materials Examination Facility (FMEF) outdoor substation. As shown on Figure 8.2-1, Bonneville Power Administration (BPA) will supply 115-kV, 3-phase electrical power from the Benton Switching Station via an overhead

line. A future 115-kV overhead line from the White Bluffs Substation is proposed for completion in the fall of 1981; this will provide electrical power via two routes for critical loads connected to the FMEF substation. Bypass switches (BS-1 and BS-2) will provide an intertie between Benton Switching Station and White Bluffs substation. Oil circuit breakers (OCB) and disconnect switches (DS) connect 20 MVA transformers (TRR) to the incoming 115-kV power. Power is supplied to the FSF via load circuit breaker (5) from transformer TRR-2 through an underground electrical duct bank.

8.2.6 System and Feature Tests

The line will be thoroughly checked and tested before it is energized. Such items as insulation resistance, circuit switchgear operation, current transformer polarity, protective relay setting, and interlocking will all be tested for adequacy. Underground 15-kV cable will be tested to 55 kV dc, and such tests will be repeated every 4 to 5 years to detect deteriorating insulation.

8.2.7 Acceptance

The following construction testing will be performed on the electrical equipment to verify the functional and safety characteristics of components and subsystems of the electrical power system:

- 1) Megger test--power circuits and apparatus
- 2) Electrical equipment external connection and grounding
- 3) Electrical equipment internal completeness and cleanliness
- 4) Integrity of lightning arrestors
- 5) Circuit breaker visual inspection
- 6) Circuit protective system function
- 7) Control circuit functional test

8.2.8 Evaluation

The design capacities provide adequate reserve for the design basis load of about 500 kVA for FSF. The results of loss of power events have been analyzed (paragraph 15.2.6.4) and shown to have no radiological consequences.

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CHAPTER 9
AUXILIARY SYSTEMS

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9.0 AUXILIARY SYSTEMS

9.1 GENERAL

The auxiliary systems assist the safe steady-state operation of the storage vessel and heat removal systems. Their functions include provisions for sodium and NaK cover gas; cooling of the sodium components contained in below-grade vaults; heating and insulation of sodium and NaK containing components; fuel-handling interface with FFTF equipment; and instrumentation and control to support the operation of the storage vessel, heat removal system, and other storage vessel support systems. None of these systems are, however, essential for the safety of the FSF, and a loss of any of them will not cause public hazard conditions. An additional description of these systems is in Section 2 of Reference 1.

9.2 FUEL HANDLING SYSTEM

The Fuel Handling System (FHS) includes the Bottom Loading Transfer Cask (BLTC) transporter, the BLTC and its dolly, the floor valve, the floor valve adapter, and the plug handling fixture. The BLTC system and the floor valve are furnished by the FFTF system, and the floor valve adapter and plug handling fixture are part of the fuel storage vessel system.

The FHS is operated in conjunction with the BLTC and the FFTF floor valve to transport spent fuel assemblies and IEM canisters from the FFTF to the FSF. The FHS lowers and retrieves fuel assemblies and canisters into and out of the FSF storage vessel.

The BLTC transporter is a rail-guided carriage onto which the BLTC is driven and secured for travel between the FFTF Reactor Service Building Extension (RSBE) and the FSF. The rails on the transporter align with the two sets of rails at the FFTF RSBE and the three rails at the FSF, which form two sets of BLTC rails. The transporter can support the BLTC containing the FFTF

fuel assembly and retain it in the vertical orientation under forces resulting from an FSF DBE condition. A reeled power cable provides the electrical power and control for the BLTC transporter traction motor. A second reel powers the BLTC during transit. The transporter also serves as the vehicular bridge across the rail trench at the RSBE.

The FSF plug handling fixture mates with the FFTF floor valve and removes and stores the fuel transfer port plug before handling fuel assemblies. The plug handling fixture is configured as a vertical cylindrical can. A gate valve at the lower end maintains an interior argon atmosphere. The seals on the lower mating flange of the fixture are compatible with the upper seal surface of the FFTF floor valve. Purge and vacuum connections mate with those on the FFTF floor valve to inert the interior areas before the plug handling operations. The upper portion of the fixture has a grapple and a plug lifting and opening mechanism for handling the transfer port plugs. Lifting lugs on the outside of the fixture allow handling by the FSF crane. An electrical interlock prevents floor valve operation unless the plug handling fixture or BLTC is mated and sealed to the floor valve in normal operation. Provisions are included to maintain the argon atmosphere within the fixture at a slight positive pressure during its storage, using a bottled argon supply.

9.3 TRACE HEATING

9.3.1 Functions and Requirements

The trace heating system preheats the piping, valves, and sodium purification loop components before filling and operation, maintains elevated liquid metal temperatures during operation, and prevents sodium from freezing in the piping. The system maintains temperature at a level that prevents sodium deposition and buildup. The system also preheats the guard tank and fuel storage vessel and assists in heating sodium for cold trapping. The storage vessel trace heating system is also a backup for the flow through sodium heater in the cold trap vault.

9.3.2 System Description

In addition to the subsequent text, the system is described in Section 2.2.3.4 of Reference 1.

9.3.2.1 Trace Heaters for Na and Argon Pipes

The pipes of the sodium purification system are trace-heated to maintain the sodium temperature within the pipe at approximately the bulk temperature of the vessel sodium. Argon piping of the vessel cover gas is also trace-heated between the upstream check valve and the downstream sodium vapor trap. The trace heaters on the pipes are stand-off mounted within the 1 in. (0.25 m) annular cavern formed around the pipe by the metallic sheath and the thermal insulation, as shown on Figure 9.3-1. A schematic of the heat removal system is shown on Figure 9.3-2. Redundant heaters are provided on the inaccessible pipes within the vessel vault.

The heaters for the sodium pipes are designed for approximately 50 W/ft (164 W/m) output at the operating voltage and those of the argon pipe for approximately 20 W/ft (66 W/m). Heaters are arranged in zones for control purposes. The heater within each zone is controlled by an on-off controller to maintain the pipe temperature within the selected setpoints. Redundant thermocouples, one for control and the other for independent temperature indication, are attached to the pipe in the appropriate location with each zone. To prevent any possible overpressure conditions, the piping heaters on the plugging temperature indicator (PTI), in-line sodium heater (including the heater itself), and cold trap subloops of the sodium purification system are interlocked to the respective isolation valves. Thus, when a subloop is isolated, the associated heaters are deactivated.

FIGURE 9.3-1. Pipe Heater Standoff.

IEDL 7908-194.2

FIGURE 9.3-2. Heat Removal System.

Individual heaters are located to minimize the unheated length of pipe either at the component or at the junction of heaters. A maximum unheated length of the sodium and argon piping of approximately 18 in. (0.45 m) occurs at the vessel wall penetration seal.

The calculated length of unheated piping to cool stagnant sodium within it from 400°F (204.4°C) to 250°F (121.1°C) is 5.4 ft (1.54 m), and for stagnant argon is 2 ft (0.6 m). With sodium flowing at any expected rate, the equivalent unheated length would be considerably greater. The consequence of losing a heater on the argon piping is that this section can act as cold trap to precipitate sodium vapor. To mitigate this possibility, the argon pipes are deliberately 2-in. (5 cm) oversized to provide adequate time to activate the redundant heater or replace the heater. Once the repairs are made, any sodium deposition can be remelted.

9.3.2.1.1 Trace Heaters for Sodium Argon Valves

Each valve in the sodium and argon lines is heated with a separately controlled cartridge type heater attached to the valve body. Two thermocouples, one for control and the other for independent indication, are also attached to the valve body. Heaters for the pneumatically actuated valves are designed for 120 W output at the operating voltage, and the power for manually operated valves is 50 W.

9.3.2.1.2 Trace Heaters for the Vapor Trap

The sodium vapor trap in the argon cover gas line is trace-heated by two longitudinally placed strap-on heaters located approximately 180 degrees apart. Each of the heaters is designed for 50 W output at the operating conditions. The pair is operated in an on-off mode to maintain the vapor trap within the $240 \pm 10^\circ\text{F}$ ($115.5 \pm 5.5^\circ\text{C}$) operating range. Two thermocouples, one for control and the other for independent indication, are provided. The latter may be used for control, if necessary.

9.3.2.2 Guard Tank Standoff Heaters

The stand-off guard tank heaters (24 total) are placed in four zones in the guard tank side walls and are the only heaters to be controlled with variable voltage. The heaters are placed so that there is a ring of tubular heaters approximately every 3 ft (0.91 m) along the cylindrical section of the guard tank. The guard tank insulation covering the heaters is a permanent installation, and since the storage vessel vault is subjected to radiation from spent fuel elements in the storage vessel, tank heaters are nonreplaceable. Therefore, excess total heater capacity is provided as a backup in case of individual heater failure.

The guard tank stand-off heaters are used for dry heatup and maintaining sodium temperature under low decay heat conditions. If sodium should freeze in the storage vessel, there will be no difficulty remelting to a free surface due to heater arrangement. These heaters are designed for up to 10.2 kW output per ring or a total of 40.8 kW.

9.4 SODIUM PURIFICATION

9.4.1 Functions and Requirements

The sodium purification loop, shown schematically in Figure 9.3-2, protects the integrity of materials stored in the FSF by:

- 1) Limiting oxide buildup to a level which permits the sodium to drain from the fuel assemblies when they are removed (after up to five years of storage).
- 2) Preventing localized overheating of fuel assemblies due to plugging of flow channels with impurities.

In addition to the piping used for sodium circulation, there are provisions for sodium fill. Samples can be obtained from the sodium purification loop for analysis of impurities, radioactive contamination, or both.

No single failure in the sodium purification loop, under operating conditions, interferes with the removal of decay heat from the primary vessel.

Design provisions in the sodium purification loop prevent failure by plugging. The sodium plugging temperature is kept at least 25°F (13.9°C) below the NaK cold leg temperatures during operation, and in no case above 300°F (149°C). A plugging temperature monitor is provided in the purification loop.

The design temperature of the sodium purification system is 650°F (343°C), and the design pressure of the piping is 50 lb/in.² gage (340 kPa), plus the static sodium head. The sodium purification loop components and piping can operate in the below-grade vault air environment when filled with sodium at temperatures up to 650°F (343°C).

An in-line sodium heater in the loop heats the liquid sodium to 600°F (316°C) for the initial cold trapping and can maintain the sodium temperature at any desired level when there is little or no heat generation from stored fuel.

9.4.2 System Description and Performance

The purification loop, shown schematically in Figure 9.3-2, is operated at 600°F (316°C) before fuel elements are actually stored in the vessel to initially purify the sodium. During steady-state operation, the loop is operated at the vessel's sodium temperature of 300°-500°F (149°-260°C). It is expected that two cold traps will be required during initial purification. A third cold trap, with a capacity large enough to remove the oxides formed during at least half of the 20-year operating life of the facility, will then be installed.

The sodium purification loop consists of a cold trap, a cold trap fan, a plugging temperature indicator (PTI) with a blower, an electromagnetic (EM) linear induction pump, a sampling station, a flowthrough sodium heater, and associated valves and instrumentation. Purification is by precipitation of oxides in the cold trap, and sodium purity is monitored by the PTI. During normal cold trap operation, the plugging temperature is kept at least 25°F (13.9°C) below the NaK cold leg temperature and is not allowed to exceed 300°F (149°C). The flow rate and temperature of sodium through the plugging meter are recorded whenever the unit is in use.

The sampling station consists of block valves with pipe stubs that allow a pinch tube to be installed to obtain sodium samples for analysis. A flow-through sodium heater heats the vessel sodium to 600°F (316°C) before and during the initial cold trapping. The sample station is in a subgrade equipment room.

The EM pump in the sodium purification loop circulates up to 10 gpm ($6.3 \times 10^{-4} \text{ m}^3/\text{s}$) through the cold trap. The pump is capable of flow rate up to 30 gpm ($1.9 \times 10^{-3} \text{ m}^3/\text{s}$). The higher flow rate aids the initial sodium heatup using the in-line heater. The difference between the inlet and outlet temperatures to the cold trap is maintained at about 50°F (28°C). The 10 gpm ($6.3 \times 10^{-4} \text{ m}^3/\text{s}$) flow is used during initial cold trapping at 600°F (316°C). During steady-state fuel storage operation, with the vessel sodium temperature at 300°-500°F (149°-260°C), the flow rate is set between 1 and 5 gpm (6.3×10^{-5} and $3.2 \times 10^{-4} \text{ m}^3/\text{s}$).

The section of the loop between the fuel storage vessel that includes the second isolation valve from the storage vessel meets Section III, Class 2, of the ASME Code (Reference 2). This Section III piping and the valve are Seismic Category I. The remainder of the loop meets Section VIII of the Code (Reference 2) and is Seismic Category III. All parts of the sodium purification loop are in the below-grade fuel storage vessel and cold trap vaults.

A sodium fill/connector on the piping allows the fuel storage vessel and the sodium purification loop to be filled from tank cars or barrels. Sodium flow to the vessel and the purification loop is started by pressurizing the sodium supply.

The sodium in-line heater assembly is a vertical cylindrical tank with 36 replaceable tubular-type heaters located around the sodium pipe in an oven. The heater assembly is rated at 72 kW. The nominal sodium flow rate through the heater assembly is 30 gpm ($1.9 \times 10^{-3} \text{ m}^3/\text{s}$). The sodium in-line heater, in conjunction with the guard tank trace heating, can heat the storage vessel sodium from 300⁰-400⁰F (149⁰-204⁰C) to 600⁰F (316⁰C) without exceeding the vessel and piping design temperature of 650⁰F (343⁰C).

9.4.3 Instrumentation and Control

The sodium purification system has temperature, flow, pressure, and liquid metal leak detection instrumentation. Temperature measurements are made by thermocouples attached to the process components. Control valves are supplied with position sensors that provide remote valve position status (open/closed) to the FSF control room.

Liquid metal flow is measured by permanent magnet flowmeters.

An EM pump circulates sodium through the cold trap. The EI&C System provides interlocks that automatically shut off the EM pump in case of:

- high pump temperature
- leakage in either the cold trap or in-line heaters
- loss of flow

The cold trap blower system circulates vault air through the annular space around the trap. The EI&C System provides instrumentation to control the blower. A hand-indicating controller allows variable speed control of the blower. An interlock automatically shuts the blower off upon sensing a low sodium flow condition.

The cold trap itself has four thermocouples mounted in a single instrument thimble. Three of the four thermocouples monitor local temperature. The fourth is connected to a recorder in the FSF control room, with low-temperature annunciation. An additional thermocouple in the outlet duct monitors air outlet duct temperature. The cold trap is preheated by externally mounted resistance heaters.

The EI&C System for the plugging temperature indicator provides temperature, flow rate, leak detection, and control instrumentation. Flow rate measurements are by a permanent magnet flowmeter installed in the outlet line and are recorded in the FSF control room. A damper control system, consisting of a hand-indicating controller and damper position sensors, controls the flow of vault atmosphere around the plugging temperature indicator for plugging temperature measurements. Temperature measurements are obtained from thermocouples attached to the process components. Signals from the thermocouples are recorded in the FSF control room. Associated control valves with position sensors (open/closed) and hand-indicating controllers permit remote valve operation.

The sodium in-line heater assembly has six sets of six (36 total) 2-kW heaters. A controller operating from a thermocouple on the sodium in-line heater outlet line controls one to six sets of heaters (selected by individual manual circuit breakers) to maintain the selected outlet temperature. An out-of-limit controller operating from the same thermocouple shuts off power to the heaters on HIGH HEATER SETPOINT TEMPERATURE. To maintain electrical separation, a second thermocouple on

the heater outlet line monitors and records the temperature in the control room, turns off heater power and annunciates when the HIGH-HIGH HEATER outlet limit is reached. Both of these alarms must be manually reset before heater power can be applied to the in-line heater. These alarms activate the FSF process malfunction alarm channel to the FFTF control room.

9.5 INSTRUMENT AIR SYSTEM

9.5.1 Function

The air system, shown schematically in Figure 9.5-1, supplies pressurized air to actuate the pneumatically operated valves located in the various inerted vaults.

9.5.2 Design Requirements

The air system was designed to meet the following requirements:

- To provide enough capacity to actuate the pneumatically operated valves.
- To meet the ASME Power and Piping Code ANSI B31.1 for all process piping and valves.

FIGURE 9.5-1 AIR SYSTEM

9.5.3 Design Description

The FSF air system comprises piping and valves which distribute compressed air at 90/100 lb/in² (620/689 kPa) to pneumatically operated valves in FSF. The source of air is the instrument air system of FFTF, the inter-connection being made in the Reactor Service Building.

The piping system in FSF has a design pressure and temperature of 150 lb/in² gage (1.0 M Pa) and 150°F (65.5°C) respectively, and is provided with a relief valve and alarms which indicate either interrupted or over pressurization. The system is designed to the requirements of B31.1.0.

FFTF instrument air is supplied at a dewpoint of -40°F (-40°C) or less and at a pressure of 115/125 lb/in² (793/862 K Pa). The FFTF system comprises three air compressors which are, also, connected to the emergency electric power system to assure a continuous air supply.

The air utilization design value for FSF is 10 std ft³/min (0.005 m³/sec) with the maximum anticipated usage being < 4 std ft³/min (0.002 m³/sec).

9.5.4 Safety Considerations

Pressure reliefs at each pressure-reduction stage protect the system from overpressure if a pressure regulator fails to open. The fuel storage vessel and cold trap vaults also have pressure reliefs to prevent the vault pressure from exceeding 1 lb/in² gage (6.9 kPa).

9.6 VAULT COOLING SYSTEM

9.6.1 Functions

The vault cooling system removes heat radiated and convected to the vaults' atmospheres from process equipment (primarily the heat removal sytem and the storage vessel system equipment). The system is shown schematically in Figure 9.6-1.

9.6.2 Design Requirements

The vault cooling system meets the following requirements:

- 1) Maintains structural bulk concrete temperatures in the vaults at or below 150°F (66°C) for expected FSF operational heat loads, and local hot spots at or below 200°F (93°C).
- 2) Maintains bulk structural concrete temperatures in the vaults at or below 350°F (176°C) during emergency periods that may result from failures in the heat removal system or a single failure in the vault cooling sytem, and maintains local hot spots at or below 650°F (343°C).
- 3) Rejects theheat to the atmosphere where the temperature is 103°F (39.4°C) or less.
- 4) Rejects the heat loads outlines in Table 9.6-1.
- 5) The components that are part of the air circulation loop are designed to operate in the following normal environment:

Radiation: Maximum 10 mr/hr

Atmosphere: Air

Pressure: Atmospheric with pressure relief protection against an overpressure above the design pressure of 2 lb/in² gage (13.8 kPa)

Temperature: 150°F (66°C) maximum
50°F (10°C) minimum

- 7) The components in the coolant circulation are designed to operate in the following normal environment:

Atmosphere: Water glycol mixture
Normally 40% glycol (by weight)

Pressure: Maximum 50 lb/in² gage (340 kPa)

Temperature: 100°F (37.8°C) maximum
-10°F (-23.3°C) minimum

TABLE 9.6-1

ESTIMATED HEAT LOADS (IN kW) ON VAULT COOLING SYSTEM

Heat Removal System Operational Mode	Vault Cooling System Heat Loads	
	Cold Trap (kW)	Storage Vessel (kW)
Normal Operation and Cold Trapping $T_{\text{sodium}} = 475^{\circ}\text{F} (246^{\circ}\text{C})$	14	27
Initial Preheat/Sodium Loading $T_{\text{sodium}} \leq 400^{\circ}\text{F} (204^{\circ}\text{C})$	Not applicable	22
Initial Cold Trapping $T_{\text{sodium}} = 600^{\circ}\text{F} (316^{\circ}\text{C})$ (maximum)*	20 to 40	38
Off-Normal Operation (between 500°F and 600°F) $T_{\text{sodium}} = 500^{\circ}\text{F} - 600^{\circ}\text{F}$	Not applicable	29 to 36**

*The bulk sodium temperature will gradually be reduced to $400^{\circ}\text{F} (204^{\circ}\text{C})$ as the sodium plugging temperature is decreased. The storage vessel vault heat loads would, thus, gradually decrease to 22 kW.

**Approximately linear increase in heat loss from vault over temperature range $500^{\circ}\text{F} - 600^{\circ}\text{F}$.

FIGURE 9.6-1. Vault Cooling System.

9.6.3 Design Description

The principal components of the vault cooling system (Figure 9.6-1) are an air loop, a coolant (water-glycol) loop, a tube-fin type cooler, a coolant chiller, and a coolant expansion tank.

In the air loop, a blower circulates the air from the vaults through the air to water cooler, then back to the vaults. The air ducting to the storage vessel vault directs the cold air from the vault cooling system into a swirl pattern around the storage vessel, creating a turbulent flow pattern in the entire annulus and preventing the formation of local vault hot spots. In the cold trap vault, a blower supplies a variable volume of vault air to cool the cold trap, discharging directly into the return duct. Air from the cold trap vault joins with the cold trap shroud duct through a tee. During periods when cold trapping is not required, the cold trap fan is turned off and isolated by an inlet damper. Meanwhile, cold air still flows through the cold trap vault. Moisture is condensed from the circulating air in the air cooler and is removed from the system through the air cooler drain line.

In the thermostatically controlled coolant loop, a coolant pump circulates the coolant from the air cooler through the chiller, where the process heat is rejected to the atmosphere, and back to the air cooler. Valves isolate each of the major loop components. An expansion tank accommodates changes in coolant volume caused by thermal expansion and contraction. The system is filled through the expansion tank. The expansion tank and all low points in the coolant system have drain lines. All controls and valves in the loop are manually operated, on-off or open-closed.

The instrumentation and control system measures the following process variables:

Temperatures

Remote temperature indicators and high temperature alarms are provided for each vault.

Air cooler inlet and outlet coolant temperatures are locally indicated.

Pressures

Air cooler outlet coolant pressure and coolant pump suction pressure are indicated locally. A high coolant pump discharge pressure automatically shuts the pump off.

Flow

A flow measuring orifice downstream from the coolant chiller provides for remote flow indication.

Level

A level alarm is on the expansion tank.

Oxygen

Oxygen levels are monitored.

All process alarms are indicated in the FSF control room. In addition, a process malfunction alarm indicates remotely in the FFTF. Remote hand switches are provided for the air blower and the coolant pump.

All coolant piping and equipment meets ASME Power and Piping Code, ANSI, B31.1

9.6.4 Performance Description

Table 9.6-1 summarizes the anticipated maximum heat loads for the range of normal operating conditions expected. These normal operating conditions refer primarily to the status of the storage vessel and heat removal systems. The most severe normally expected condition occurs during the

initial cold trapping period of the storage vessel when the vessel sodium is heated to 600°F (316°C); the resulting maximum total heat load to the vault cooling system is 87 kW.

A computer simulation of the storage vessel vault under these conditions determined the steady-state concrete temperature as a function of sodium temperature. It was concluded that the vault air temperature should be maintained at 100°F (37.8°C) during operation with sodium at 600°F (316°C).

The same cooling rate necessary to maintain the above conditions would maintain the vaults at approximately 91°F (32.7°C) with the storage vessel sodium at 475°F (246°C) and the normally expected cold trapping rates. Safety analysis is based on a minimum vault cooling system capacity of 90 kW.

9.6.5 Safety Considerations

The vault cooling system operates under the expected environmental conditions outlined in Paragraph 9.6.2. The design includes provision for isolating the cold trap vault from the fuel storage vessel vault for personnel access. The pressure relief system on the coolant circulation loop protects the equipment from exceeding the design pressure.

The active components are located and arranged to permit access and contact maintenance with no restrictions during plant operation. In the event of off-normal operating conditions, the vault cooling system will perform as follows:

- 1) In the event of off-normal operation of the heat removal system, it is expected that cold trapping could be temporarily discontinued. This would reduce the total vault cooling system heat loads so that the concrete temperature could be maintained at below 150°F (66°C).

- 2) In the event of a temporary loss of the vault cooling system, sodium cold trapping would be discontinued to minimize the heat load to the cold trap vaults. A computer simulation estimates that at steady-state with sodium at 475°F (246°C) (i.e., the removal system remains functioning), the peak storage vessel vault concrete temperature would reach 296°F (146.6°C) under equilibrium conditions. If the vault cooling system were to remain inoperative for a prolonged period of time, the peak storage vessel vault concrete temperature could be held to about 200°F (93°C) by increasing the heat removal rate to reduce the sodium temperature to 300°F (140°C).

As presented in the accident analysis section (Chapter 15), accidents such as loss of vault cooling will not result in sodium and concrete temperatures that exceed design limits. Accidents such as excessive vessel cooling and loss of vault cooling flow will not cause the release of fission products.

9.7 ARGON SYSTEM

9.7.1 Functions and Requirements

The argon system fulfills the following functions:

- 1) Provides inert cover gas above the sodium in the fuel storage vessel.
- 2) Provides inert cover gas above the NaK in the NaK expansion tanks at a pressure that ensures that the NaK pressure in the fuel storage vessel is always at least 5 lb/in.² (34 kPa) greater than the sodium pressure.
- 3) Supplies argon to the buffer seals of the fuel storage vessel.
- 4) Supplies argon to the BLTC, floor valve, and plug handling fixture.
- 5) Protects the fuel storage vessel and NaK expansion tanks from overpressurization.
- 6) Provides the capability to draw a vacuum in the sodium and NaK components and piping.

The following requirements are met in the design of the argon system:

- 1) The argon system provides for the initial evacuation and backfill of the system it supports.
- 2) A holdup tank is included in the system to temporarily store the argon released from the vessel and the BLTC. At this point a detector monitors the argon for radioactivity to ensure that the release rates are within acceptable limits when waste argon is released to the atmosphere.
- 3) The system protects the fuel storage vessel and the NaK expansion tank from pressurization above design pressures. The remainder of the argon system is protected from overpressurization in accordance with the codes mentioned in Item 4.
- 4) Those portions of the argon system that are part of the storage vessel system pressure boundary are designed to ASME B&PV Code, Section III, Class 2 or 3, as explained in Chapter 4.0. Other components and piping are designed according to the standards listed in Appendix A of Reference 1.
- 5) A source of emergency argon supply is provided in the tornado-hardened area.
- 6) Fuel storage vessel heat removal is operable at all times and will not be interrupted by an argon system or component failure.
- 7) The system has a design life of 20 years.
- 8) Those portions of argon system piping containing or possibly containing sodium vapors are trace-heated and insulated.
- 9) Argon purity meets the following limits:

Oxygen	10 ppm
H ₂	8 ppm
H ₂ O	8 ppm
All Carbonaceous Gases	25 ppm

9.7.2 System Description

The argon system is composed of the compressed argon supply, the auxiliary argon supply, the argon distribution network, and the effluent argon holdup and vacuum equipment. The system is shown schematically in Figure 9.7-1 and in more detail in Reference 1.

High purity argon at 2400 lb/in² gage (916 MPa) is supplied from compressed bottles (290 std ft³ [8.5 M³] per bottle) located outside of the building. The same connections (CGA fittings) used for the compressed gas bottles can be used to connect liquid argon cylinders to the system to provide argon for the initial inerting of the vessel and heat removal systems. The supply pressure is reduced by a series of pressure regulators to 30 lb/in² gage (204 kPa) at a distribution header. From this point, argon is distributed through additional pressure reducers to:

- NaK expansion tanks through a regulator and normally closed blocking valves to a pressure of 5 lb/in² gage (35 kPa) during initial fill
- BLTC/floor valve connection at 30 lb/in² gage (210 kPa)
- Vessel seals at 3 lb/in² gage (21 kPa)
- Storage vessel at 8 in. water gage (iwg) (2 kPa)

To reach 8 iwg (2 kPa) in the vessel, 30 lb/in² gage (204 kPa) argon is throttled by two regulators in series [30 lb/in² to 2 lb/in², and 2 lb/in² to 8 iwg (204 kPa to 14 kPa, and 14 kPa to 2 kPa)] to provide an extra level of overpressure protection. Argon can flow to the vessel through either one of two oversized argon nozzles. These nozzles are provided in the

① See Text

② Check Valve Poppet Removed

Figure 9.7-1. Argon System

nozzle should become blocked by sodium deposits. Argon from the storage vessel normally can be vented at 12 iwg (3 kPa) through a vapor trap (<1 ppm Na in outlet) to the spent argon holdup tank (normally at -3 lb/in.² gage to -8 lb/in.² gage (-21 to -55 kPa)).

Spent gas from the BLTC, floor valve, and plug handling fixture operations becomes a part of the vessel cover gas. The excess gas is removed by vacuum (-3 lb/in.² gage to -8 lb/in.² gage)(-21 kPa to -55 kPa) to the argon holdup tank. This argon, after being monitored for radioactivity, may be discharged to the atmosphere by a vacuum pump.

Other design features follow:

- 1) Emergency argon gas cylinders are located below grade in tornado-hardened areas.
- 2) Pressure relief valves/rupture disks are on each NaK expansion tank.
- 3) Connections are provided to aid initial evacuation of vessel systems and NaK heat transfer lines by portable vacuum pumps.
- 4) Infrequently used pressures for testing the vessel system and all buffer seals can be obtained by using bypasses around pressure reducing regulators.
- 5) The argon lines from the 2 lb/in.² gage to 8 iwg (14 kPa to 2 kPa) pressure reducer to the storage vessel are sloped toward the vessel. The argon line from the vessel to the vapor trap is sloped toward the vessel. Both of the above lines are trace heated at 250⁰F (121⁰C) to prevent plugging from condensed sodium vapors.

9.7.3 System Performance

9.7.3.1 Cover Gas Operations

The cover gas above the storage vessel sodium and the NaK in the expansion tanks is normally static once the systems have been inerted. As the average temperatures in the heat transfer loops rise, the NaK expands; this causes the argon pressure in the expansion tanks to increase gradually. (It may be necessary to occasionally vent the argon to the atmosphere to maintain the NaK cover gas at 5 lb/in² gage (35 kPa). Relief valves, protected from NaK vapors by rupture disks, relieve the NaK loops at a pressure of 35 lb/in² gage (233 kPa). The storage vessel cover gas is relieved at a pressure of 15 psig (100 kPa).

The cover gas above the storage vessel sodium is supplied at 8 iwg (2 kPa). The argon supply is reduced from the supply header pressure [50 lb/in² gage (340 kPa)] by two regulators in series to ensure against overpressure. In addition, the storage vessel is relieved by redundant relief valves and rupture disks [at 6 lb/in² gage (41 kPa)] to the NDHX stacks. A backpressure regulator normally limits the vessel pressure to 12 iwg (3 kPa); any vented argon is held in the argon holdup tank until released to the atmosphere. A bypass line around the backpressure regulator allows the system to be depressurized to 8 iwg (2 kPa) after accumulation of argon from fuel transfer operation. Sodium is removed from the gas before venting by a filter vapor trap; the sodium content in the vent gas will be less than 1 ppm.

9.7.3.2 Fuel Transfer Operations

The fuel transfer operations that directly affect the design of the argon system are plug removal/installation (plug handling fixture, PHF), floor valve purging, and fuel loading and unloading (BLTC). Table 9.7-1 summarizes the requirements and operations involving the BLTC and the floor valve.

TABLE 9.7-1
ARGON SYSTEM/BLTC & FLOOR VALVE INTERFACES

<u>Argon Flow/Description</u>	<u>Argon Source</u>	<u>Argon Sink</u>	<u>Flow</u>	<u>Pressure</u>
1. FSF Argon Supply to BLTC	FSF Argon Supply Header, 30 psig (2.1×10^5 Pa)	BLTC, Floor Valve*	10 scfm ₃ (0.28 m^3)	30 psig (2.1×10^5 Pa)
2. Floor Valve Purge	Floor Valve	Argon Holdup Tank	2 scf (0.05 m^3)	(-3) - (-8) psig (-2.1×10^4 to -5.5×10^4 Pa)
3. BLTC Vacuum Line Purge	BLTC	Argon Holdup Tank	1 scf (0.028 m^3)	(-3) - (-8) psig (-2.1×10^4 to -5.5×10^4 Pa)
4. Miscellaneous Seals	BLTC, Floor Station	Argon Holdup Tank	1 scf (0.028 m^3)	(-3) - (-8) psig (-2.1×10^4 to -5.5×10^4 Pa)
5. Trickle Purge to Storage Vessel During Fuel Transfer	BLTC	Storage Vessel	10 scf (0.28 m^3) (nom) 1-10 scf (0.028 - 0.28 m^3) (range)	1 psig
6. Argon Blast to Remove Sodium During Fuel Removal	FSF Argon Supply Header 30 psig (2.1×10^5 Pa)	Storage Vessel	8 scf (0.226 m^3) (Included in range of Flow 5)	30 psig (2.1×10^5 Pa)
Total flow directly to Holdup Tank			4 scf (0.11 m^3)	
Total flow directly to Storage Vessel			1-50 scf (0.028 - 1.4 m^3)	
Required flowrate to BLTC			10 scfm, 30 psig ($4.7 \times 10^3 \text{ m}^3/\text{sec}$, 2.1×10^5 Pa)	
Required vacuum at BLTC			-2 to -7 psig (-1.4×10^4 to -4.8×10^4 Pa)	

*Argon Flows 2, 3, 4, and 5 originate from Argon Flow 1.

The argon system supplies up to 10 std ft³/min (9.7×10^{-3} m³/sec) of argon to the BLTC at 30 lb/in.² gage (210 kPa) from a floor station in the FSF high bay area. The quick-disconnect-type connection is compatible with the BLTC. A similar connection provides a vacuum line for exhaust gases from the BLTC and floor valve. This is a direct connection to the argon holdup tank. Other argon flows from the BLTC pass directly to the storage vessel and are vented to the holdup tank at 12 iwg (3 kPa).

A summary of the expected fuel transfer operations is presented in Table 9.7-2.

9.7.3.3 Buffer Seals

There are 19 sets of buffered seals on the storage vessel and one on the floor valve adaptor that are pressurized by the argon system.

<u>Storage Vessel</u>	<u>Total Seals</u>
10 access ports, 1 seal set for each port	10
1 floor valve adapter seal set	1
1 vessel seal set, metal O-ring	1
1 closure head seal set, metal O-ring	1
1 turntable drive, 2 seal sets	2
1 spare turntable drive, 1 seal set	1
2 instrument penetrations, 1 seal set each	2
1 inspection port for periscope, 1 seal set	1
 <u>Floor Valve Adaptor</u>	
1 interface seal, requires flexible connection from floor station	1
	<hr/> 20

These seals are maintained at a pressure of 3 lb/in.² gage (21 kPa); however, they can be tested for leakage at higher pressures by a bypass on the pressure regulator. Because these seals are normally in a static condition,

TABLE 9.7-2
FSF FUEL TRANSFER OPERATIONS - SUMMARY

1.	Basis of Calculations	
a.	Plant Life	20 Years
b.	Refueling Cycle	130 Days
c.	Fuel/Cycle	35 Elements and Canisters
d.	Maximum Loading Rate	8 Hours/Transfer
e.	Maximum Unloading Rate	9 Transfers/3 Days and Unknown Time for Shipping, etc.
f.	Maximum Gas Usage/Transfer	54 scf (1.5 m ³)/Transfer
2.	Maximum Incoming Assemblies/Plant Life	1960 Fuel Assemblies and Canisters
3.	Maximum Outgoing Assemblies/Plant Life	1960 Fuel Assemblies and Canisters
4.	Total Assembly Transfers	3920 Transfers
5.	Estimated Gas Usage for Transfers, Average	10584 scf (300 m ³)/yr**

**This value has 2 assumptions that tend to cancel one another. Part of the assemblies will be in an unknown number of fuel canisters; it may take more than one canister per fuel assembly (disassembled). This increases the total number of transfers. Second, it is doubtful the maximum transfer gas would be used each time.

they will not normally be open to the argon supply. The pressure will be checked periodically to evaluate the condition of the seals.

9.7.3.4 Spent Argon Holdup and Release

Because the argon cover gas is potentially radioactive, a temporary holdup of excess argon is provided so that release to the atmosphere can be monitored and manually controlled. Argon from the storage vessel is released to the holdup tank at pressures greater than 12 iwg (3 kPa); otherwise, a bypass around the backpressure regulator balances the pressures in the tank and vessel until 8 iwg (2 kPa) is reached in the vessel.

The argon cover gas is monitored for radioactivity; then normally released to the NDHX stacks through a manually controlled vacuum pump. The argon can be released at the maximum rate of the pump ($4.7 \times 10^{-3} \text{ m}^3/\text{s}$) and the pump stops automatically when a vacuum of -8 lb/in.^2 gage (-55 kPa) is reached. See Chapter 11 for a discussion of release data. If the activity is unsuitable for release (see Technical Specifications), the gas can be removed from the tank through a valved connection for storage and/or processing.

9.7.3.5 Infrequent Operations

Before sodium and NaK fill, the vessel and heat removal systems must be inerted and preheated. The systems are inerted to approximately 99.998% argon by:

- 1) drawing a vacuum of approximately 0.5-1 in. Hg (124 to 249 Pa);
- 2) backfilling with argon to 1 atm ($1.0 \times 10^5 \text{ Pa}$); and
- 3) repeating steps 1 and 2.

The vacuum is drawn by a portable vacuum pump through connections provided expressly for this purpose. Argon backfilling is from pressurized liquid argon cylinders through the regular distribution network (Union Carbide PG-45 cylinders or equivalent). Each cylinder used is capable of 66 std ft³/min

(0.031 m³/s) at a supply pressure of 100 lb/in² gage to 125 lb/in² gage (690 kPa to 860 kPa).

The storage vessel is preheated by the guard tank trace heater exterior to the storage vessel. As the argon gas expands, it is necessary to vent some gas to the atmosphere. Likewise, as sodium and NaK are loaded into the systems, it is necessary to vent argon. After purification of the sodium and NaK systems the argon cover gas pressures will be held static.

The NaK loops or the storage vessel will be drained by pressurizing the storage vessel or the NaK expansion tanks.

9.7.4 Instrumentation and Control

The main argon process supply line has local pressure indicators both upstream and downstream of the main pressure regulator. Two secondary supply lines, that are used to pressurize the two NaK expansion tanks, have pressure transducers to allow remote pressure monitoring in the FSF control room with out-of-limit annunciation. These alarms activate the FSF process malfunction alarm channel to the FFTF control room.

The third subsidiary supply line supplies cover gas and buffer gas to the storage vessel and associated seals. Local pressure indication is provided on the upstream and downstream sides of all the storage vessel cover gas pressure regulators. The outlet line from the storage vessel to the holdup tank has local and remote pressure indication and out-of-limit (high-low) annunciation at the FSF control room. The out-of-limit alarm activates the FSF process malfunction alarm channel to the FFTF control room.

The backup argon supply of two argon gas bottles is monitored by a pressure switch that annunciates a low-pressure condition in the FSF control room. This alarm activates the process alarm channel to the FFTF control room.

All pressure relief valves that may be subject to sodium or NaK vapors are isolated from the process lines by rupture discs. Where blocking valves are used to isolate the relief valve from the process line during maintenance or testing, two relief systems are provided and are blocking-valve-interlocked so that the two systems cannot be isolated at the same time. The interlock is accomplished by placing each blocking valve's VALVE OPEN limit switch (switch closed only when the valve is open) in the electrical control circuit of the other blocking valve. This is shown schematically in Reference 1.

The argon holdup tank pressure is indicated both locally and at the FSF control room. In addition, the holdup tank vacuum pump is interlocked to automatically shut down when holdup tank pressure is reduced to -8 lb/in.^2 gage (-55 kPa).

9.8 PROTECTIVE SYSTEMS

9.8.1 Radiation Monitoring System

The FSF radiation monitoring system detects and measures radioactive release from the facility to the environment and detects and measures above-normal radiation levels within the facility. All radiation monitoring instrumentation has backup power (via battery pack with automatic activation and up to 8 hours operation without recharging) to ensure the operability of the system should normal site power be lost. The system is described in the following paragraphs and shown in Figure 9.8-1.

9.8.1.1 Storage Vessel Neutron Monitoring

The FSF storage vessel has a neutron monitoring system for diagnostic purposes. Neutron detection is by boron trifluoride type detectors, providing high neutron sensitivity. Two detectors, which are mounted in a single steel thimble located in the storage vessel vault wall, are used in the system. The instrument thimble is located so that an unobstructed view of the storage vessel is obtained by the detectors. The detectors are placed within the

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FIGURE 9.8-1. Radiation Monitoring System.

vault wall to provide maximum sensitivity of the detectors, due to the thermalizing properties of the vault wall and to physically protect the detectors. The detectors are positioned within the thimble so that the neutron-sensitive portions of the detectors face the active portions of the stored fuel assemblies. The system contains both status indicators (test, operational) and malfunction indicators (loss of low voltage, loss of high voltage).

9.8.1.2 Argon Holdup Tank Monitoring

The argon holdup tank, which provides storage of argon cover gas, has a gamma-sensitive monitoring system. The monitoring system ascertains the radioactivity level resulting from the fission gas and determines the applicable effluent discharge rate when the tank is vented to atmosphere. Gamma radiation level detection is by a G-M type detector. The detector is in proximity to the holdup tank, with an unobstructed viewing angle.

Signals from the detector are processed both locally and in the FSF control room. Local indication and annunciation of radiation levels are provided near the holdup tank vent vacuum pump switch to permit operating personnel to verify the activity level before venting the tank to atmosphere. The FSF control room is also provided with indication and annunciation of holdup tank activity. The high-radiation level annunciation is actuated if the contents of the holdup tank are such that, if released at the maximum discharge rate, the concentration guides as defined in ERDAM 0524 (Reference 3) would be exceeded. Periodic system test is by a permanently installed check source in the detector.

9.8.1.3 Effluent Discharge Monitoring

To ensure and document that the effluent released from the argon holdup tank does not exceed the concentration guides of ERDAM 0524, an in-line beta gamma scintillation-type detector is installed in the discharge line between the argon holdup tank and the NDHX. A flow transmitter is also installed in this line to monitor total effluent discharge. Continuous indication and recording

of effluent discharge parameters are available in the FSF control room which includes effluent discharge flow rate [std ft³/min (m³/s)] and effluent activity level. In addition, the total amount of radioactive discharge is computed and recorded based upon activity level and flow rate. A high radiation alarm is triggered if the discharge is such that the concentration guides, as defined in ERDAM 0524, would be exceeded if this discharge rate were to continue for one hour.

The effluent monitoring system is designed in accordance with ERDAM 0513 (Reference 4). The system is powered by 120-V, 60-Hz power with backup power. Periodic system test is by an integral check source in the detectors.

9.8.1.4 Cold Trap Monitoring

The cold trap, which is part of the sodium purification system, is provided with a gamma sensitive detector of the ion chamber type. The detector is mounted in proximity to the cold trap with an unobstructed viewing angle. This detector serves a purely diagnostic function and monitors the radioactivity level of impurities removed from the storage vessel sodium. Indication of cold trap activity level is available in the FSF control room. There are no alarms or annunciation associated with this detector. The detector operates on 120-V, 60-Hz power. Periodic test is by an integral check source in the detector.

9.8.1.5 Vault Atmosphere Monitoring

The cold trap and fuel storage vessel vault atmospheres are monitored for radioactivity levels by monitoring the activity level of the main vault cooling duct located in the vault cooling equipment room. A beta/gamma sensitive detector of the G-M type is installed in the cooling duct for this purpose. The vault atmosphere radiation indication and annunciation is provided both locally in the vault cooling equipment room and in the FSF control room. A high-radiation alarm is triggered should the radiation level exceed one order of magnitude above the maximum operating level. The high-radiation alarm activates the FSF high-radiation alarm channel to the FFTF control room.

The vault monitoring system operates on 120-V, 60-Hz power. Periodic test is by an integral check source in the detector.

9.8.1.6 FSF Ventilating System Monitoring

The FSF ventilating system has radiation monitoring instrumentation to ascertain the FSF internal atmospheric radioactivity levels. A beta/gamma sensitive detector of the G-M type is installed in the main air return path. Both local (H&V equipment room) and remote (FSF control) indication and annunciation are provided. A high-radiation alarm is triggered if the radiation level exceeds one order of magnitude above the maximum operating level. If the radiation level exceeds three orders of magnitude above the maximum operating level, an immediate high-radiation alarm is triggered. The FSF building evacuation alarm produces a mid-frequency sound wave, amplitude modulated at subsonic frequency of not less than 75 decibels at every location where operating personnel may be present. The high-radiation alarm activates the FSF high-radiation channel to the FFTF control room.

The ventilation system monitoring system operates on 120-V, 60-Hz power with backup power to both the monitoring system and the evacuation alarm. Periodic test is by an integral check source in the detector.

9.8.1.7 High-Bay Area Monitoring

The high bay area is provided with three beta/gamma sensitive detectors of the G-M type. Two detectors are near the operating floor level and are protected from physical damage by an open wire screen surrounding the detectors. The remaining detector is approximately 15 ft (4.6 m) above the floor in the high bay area. Indication and annunciation are both local in the high bay and in the FSF control room. A high-radiation alarm is triggered if the radiation level exceeds one order of magnitude above the maximum operating level. If the radiation level exceeds three orders of magnitude above the maximum operating level, an immediate high-radiation alarm (refer to Section 9.8.1.6)

is triggered. The high-radiation alarm activates the FSF high-radiation alarm channel to the FFTF control room.

The high-bay radiation monitoring system operates on 120-V, 60-Hz power with backup power.

9.8.2 Oxygen Monitoring System

All routinely accessible areas that have a risk of an oxygen deficient atmosphere (less than 19.5%) are provided with permanent oxygen depletion monitors. These areas include the vault cooling equipment room, the argon holdup tank vault, electrical equipment room, and subgrade corridors.

The monitor uses an amperometric oxygen sensor for oxygen detection. The sensor is inside the area to be monitored while the electronic and readout equipment is outside the area, to permit operating personnel to verify oxygen concentration before making entry in the area. Both local and remote (FSF control room) indication and annunciation are provided. The alarms associated with the oxygen monitors activate the FSF process malfunction alarm channel to the FFTF control room. The oxygen monitoring system has backup power. Portable detectors are provided for areas that are not routinely accessible.

The vault atmosphere is monitored for low (below 19.5%) oxygen concentration.

9.8.3 Liquid Metal Leak/Smoke Detection System

This system is described in Section 5.2.4.

9.9 OTHER AUXILIARY SYSTEMS

9.9.1 General

These other auxiliary systems include the following four systems:

- 1) Heating, ventilating, and air conditioning systems for the total facility except for the subgrade vault.
- 2) Automatic and portable fire protection systems.
- 3) Lighting system for the interior of the facility.
- 4) Communication system.

9.9.2 Description and Design Features

9.9.2.1 Heating, Ventilating, and Air Conditioning

Process Area Air Handling System -- The louver intakes for this subsystem are at the north end of the Air Conditioning Equipment Room. This room contains a 35% filter section, supply air fan, air washer with washer pump, a 50-kW electric duct heater, and control dampers. During the heating cycle, 98% of the air is recirculated.

Stairwell and Basement Passage Air Handling System -- The air intake for this subsystem is on the roof of Stair 2. The unit consists of a supply air fan, and a 3-kW electric duct heater. The Electrical Equipment Room has an exhaust fan.

Control Room Air Handling System -- The air intake unit for this subsystem is on the roof of the Control Room. The unit consists of a roof-mounted packaged air conditioner with a 13-kW electric heater in the discharge duct.

Toilet Room Air Handling System -- The air intake for this subsystem is on the roof. The unit consists of a filter section, a supply air fan and an 8-kW duct heater. Additional heating in the toilet room is provided with a wall-mounted 0.5-kW electric heater.

Vault Cooling Equipment Room -- This room is ventilated by a wall louver inlet equipped with an automatic damper and gravity roof ventilator. Heating, to prevent freezing, is by two 1-kW electric wall heaters.

Motor Control Room -- This room has an exhaust fan which starts when a preset temperature is exceeded.

Sanitary Waste System -- A 6-in. (0.15 m) diameter extra strength, vitrified clay pipe picks up sanitary sewage from the water closet, urinal, lavatory, janitor's sink, and floor drains in the facility. The sewer is at the south-east corner of the building. A sump and sump pump in the basement copes with water reaching the basement via the stairwells.

9.9.2.2 Fire Protection System

This system is used to protect against ordinary hazards. Automatic fire protection systems are installed in areas adjacent to the high bay (process) area. Special provisions, using portable fire extinguishers, provide the fire suppression system in the high bay area and other areas where the presence of liquid metal may present a unique hazard. Liquid metal fires are covered in detail in Section 15.2.4. Area coverage of the fire protection system is listed in Table 9.9-1, and shown on Figure 9.9-1.

The sprinkler system protecting the areas referenced in Table 9.9-1 is to be designed, installed, and tested in accordance with National Fire Protection Association (NFPA) Standard 13. The sprinkler system will be connected to underground fire mains in accordance with NFPA Standard 24. This system provides protection against the ordinary hazards found in these areas. Flow and/or reduced pressure will alarm locally in the FFTF control room and fire loop systems. Curbs prevent water from entering the high bay.

Halon System

The Halon 1301 system, protecting the high value electronics and electrical equipment used in normal control of the FSF, is to be designed, installed, and

TABLE 9.9-1
FIRE PROTECTION AREA COVERAGE

<u>Areas</u>	<u>Flammable Materials</u>	<u>Detectors</u>	<u>Protection</u>
Vestibule Toilet Janitor	Plastic Wood Paper and Cloth	Sprinkler Heads	Automatic Sprinkler
Control Room	Wood and Paper Electrical Insulation Plastic	Smoke Detectors	Automatic Halon 1301
NDHX	Fuel NaK Electrical Insulation	Smoke Detectors	Manual Portable Extinguisher
Motor Control Room	Electrical Insulation	Sprinkler Heads	Automatic Sprinkler
Mechanical Equipment	Electrical Insulation	Sprinkler Heads	Automatic Sprinkler
Vault Cooling Equipment Room	Electrical Insulation	Sprinkler Heads	Automatic Sprinkler
Electrical Equipment Room	Electrical Insulation	Smoke Detectors	Manual Portable Extinguisher
Argon Vault	Electrical Insulation	Smoke Detectors	Manual Portable Extinguisher
Stairwell Passages Entry	None	Sprinkler Heads	Automatic Sprinkler
High Bay Area	Protective Clothing Electrical Insulation Liquid Metal	Smoke Detectors	Manual Portable Extinguishers
Inerted Areas Trenches & Vaults	Electrical Insulation Liquid Metal	Smoke Detectors	Catch Basins with Perforated Fire Sup- pression Cover Plates

FIGURE 9.9-1. Fire Protection Plan.

tested in accordance with NFPA Standard 12-A. Automatic discharge of the Halon system is initiated by smoke detectors through a time-delay relay. This time-delay relay allows local action to control a fire before flooding the control room with gas. If a fire can be controlled quickly by local action, the automatic system action can be terminated. Manual trip capability of the Halon system is also provided. The smoke detectors, initiating the Halon system, will alarm locally, as well as the FFTF control room and fire loop system. When the Halon 1301 system is activated, the air circulating blower is turned off and the dampers are closed through automatic interlocks.

Manual Portable Fire Extinguishers -- Portable extinguishers for manual fire control will be installed in accordance with NFPA Standard 10. Five separate stations will be set up and each station will contain the following:

- 1 Ansul 28-lb hand extinguisher, Model NX-30-D (NA-X)
- 1 Ansul 15-lb hand extinguisher, Model CDj-15-17224 (CO₂)

A portable Halon 1301 system will be installed on the crane for protection of the crane's motors.

Catch Basin -- Catch basins are installed under liquid metal containing components and lines to contain any spills or leakage. These catch basins protect the floor and facilitate fire suppression by retaining and cooling any liquid metal spills. Basic fire suppression in the inerted vaults is by inerting to exclude oxygen. Basic fire suppression in the noninerted area is with Na-X contained in pails and portable extinguishers.

Building Exits -- Building exits are designed, constructed, and identified in accordance with NFPA Standard 101.

Alarms -- Fire signals by detectors, flow signals, or local manual stations will be announced locally and transmitted to the FFTF control room and to the fire loop system. Fire signals will actuate fire and smoke dampers located in the east and west stairwells and the west entryroom, Room 919.

Reliability -- The reliability of conventional fire protection systems conforms to applicable requirements of the National Fire Protection Code. The reliability of the inerting systems for fire protection in the sodium component containing vaults is commensurate with the inherent reliability built into the ANSI B31.1 standard applied to inerting system components and piping. The vault leak rates are tested at less than 1% per day at operating conditions, and the nitrogen supply capacity and delivery rate are provided with large design margins.

9.9.2.3 Lighting System

Facility Lighting -- General building lighting is furnished by fluorescent mercury vapor and incandescent lamps, with the power supplied from a panelboard for operation and maintenance. Duplex receptacles are supplied from a panelboard.

Minimum lighting levels are as follows:

Control Room - 100 FC (1.08×10^3 lux)
Equipment Rooms - 30 FC (323 lux)
Stairways - 30 FC (323 lux)
Work Areas - 50 FC (538 lux)
Emergency Egress - 5 FC (54 lux)

Emergency Lighting -- Emergency power is a central AC power pack for emergency lighting, exit signs, and other essential life safety systems and is automatically activated upon loss of normal power.

9.9.2.4 Communications System

General

The building is equipped with a PAX telephone and paging system. The telephone system is an integrated dial-operated telephone system. These

telephone instruments are reached by dialing a specified extension number. The public address and paging system are an integrated part of the telephone system. The building is paged by dialing an appropriate number for any telephone.

PAX Telephone System -- The telephone system allows communication between the work areas and the operations of the control room. The system is used for either a station-to-station call or a conference call. Telephones are in the control room, the high bay, and in the BLTC when mated to the facility.

Evacuation Alarm -- The 400 Area evacuation alarm system is interconnected throughout the FFTF site and is designed to alert the entire area whenever a significant hazard develops. This facility interfaces with the existing system. The evacuation alarm system is activated from the FFTF control room.

9.10 REFERENCES

1. Fuel Storage Facility System Design Description, N106SDD210001, Rev. 1, Atomics International, Canoga Park, CA, April 1978.
2. ASME Boiler and Pressure Vessel Code, The American Society of Mechanical Engineers, New York, NY.
3. ERDA Manual, "Standards for Radiation Protection," Chap. 0524, US Department of Energy, Washington, DC, March 1977.
4. ERDA Manual, "Effluent and Environmental Monitoring and Reporting," Chap. 0513, US Department of Energy, Washington, DC, March 1974.
5. Equipment Specification, Storage Vessel Subsystem - Fuel Storage Facility, N106NV220001, Atomics International, Canoga Park, CA, February 1977.

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CHARTER

1. STEAM AND POWER CONVERSION

2. If no steam produced by the system nor is there any power
produced. Therefore, this chapter is not applicable to the system.

CHAPTER 10
STEAM AND POWER CONVERSION SYSTEM

10.0 STEAM AND POWER CONVERSION

There is no steam produced by the system nor is there any useful power produced. Therefore, this chapter is not applicable to the FSF.

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CHAPTER 11

RADIOACTIVE WASTE MANAGEMENT

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11.0 RADIOACTIVE WASTE MANAGEMENT

11.1 GENERAL

This chapter describes the FSF radioactive waste management. Operation of FSF will result in various small quantities of radioactive gaseous waste, and a minimum of solid waste. The gaseous waste is principally associated with the storage vessel cover gas and purging of the fuel transfer machine. Solid waste is principally clothing, rags, etc., from any decontamination required; probably no more than one cold trap change in mid-life; and sodium samples taken as needed for appropriate analyses. Any liquid waste will be small in quantity; it will be picked up by rags and disposed of as solid waste. This chapter describes the systems used to safely process these wastes.

The FSF waste processing systems achieve near zero release for the following reasons:

- 1) No leaking fuel pins will enter the system. If a failure of cladding has occurred prior to entry, the leaking element will be placed in a sealed container prior to storage in FSF.
- 2) Once in storage the system is passive other than rotation of the turntable for new fuel addition, hence there is little potential for clad rupture subsequent to storage.
- 3) All radioactive solid wastes (if any) are transferred to the FFTF solid waste handling system where the solid wastes are packaged and sent to the Hanford 200 Area for disposition. The FFTF system, described in Chapter 12 of Reference 1, is designed for near zero release.
- 4) If any radioactive liquid is generated it will be of small volume, absorbed in rags and processed as solid waste. Therefore it will not be available for liquid release.

11.2 SOURCE TERMS

The source terms used in the design and evaluation of the radioactive waste systems consist of the types and quantities of activity that remain in spent fuel elements after a minimum 200-day decay time when it is removed from the FFTF interim decay storage (IDS) vessel for transport to FSF and those that may be carried over from the cover gas of the IDS to the FSF via transfer using the bottom loading transfer cask (BLTC).

11.2.1 Gaseous Waste

This chapter discusses normal operations and the maximum credible accident associated with the gaseous waste system. Source terms and the consequences of postulated accident situations that could lead to direct release, through other than the discharge system, are presented in Chapter 15. Three conditions are considered: (1) normal operations, (2) design basis operations, and (3) maximum credible accident conditions. Although accident conditions are presented in Chapter 15, the maximum credible accident involving the gaseous waste system is discussed in this chapter as well for continuity.

For normal operation, the only source of activity in the argon cover gas is carryover from the IDS, introduced during fuel addition through the BLTC. A reasonably conservative estimate of the equilibrium inventory in the FSF vessel is 1.2 curies of inert radioactive gas composed of 75% Xe-133, 2% Xe-131m, and 23% Kr-85.

For design basis operation, it has been assumed that 0.1% of the fuel pins rupture when in the FSF and their total inventory of inert fission gas reaches the argon cover gas; as discussed in Section 11.1, this occurrence is extremely unlikely. Due to the long decay time (>200 days) prior to storage in the FSF, the only isotope of concern is the 131 curies of Kr-85 added to the cover gas inventory with other noble gas isotopes, contributing less than 1% of the additional activity. Note that as stated in Section 11.1, no ruptured fuel pins are expected in the FSF, but for the design basis case 0.1% was assumed, which is 10% of that assumed for the IDS (Reference 1).

For the maximum credible accident condition it was assumed that, in loading, a fuel assembly was dropped into an already occupied position. In this event it was assumed that all pins in both assemblies were ruptured such that all fission products are released in the vessel, adding to the activity already present in the design basis case. The cover gas radioactive inventory would be 99.9% Kr-85 at a level of 909 curies.

11.2.2 Liquid Wastes

There is not expected to be any liquid waste processing. The only radioactive liquid that may be expected would occur if decontamination is required for items such as the floor valve. If small quantities of radioactive liquid are generated they will be absorbed in rags and disposed of as solid waste. No significant accidents, other than coolant losses, discussed in Chapter 15, were identified.

11.2.3 Solid Wastes

Radioactive solid wastes will consist of occasional pieces of hardware and paper, rags, etc., used in maintenance work. Since the FSF is designed to operate 20 years without major deterioration of the sodium, the expected replacement items that will be contaminated are the cold trap and the plugging temperature indicator, each of which is expected to be replaced once during the 20-year service life. Sodium samples will be taken for analysis on an as-needed basis but the frequency is expected to be low. A summary of expected solid wastes is given in Table 11.2-1. The solid waste generated will be processed by the FFTF system as discussed in Chapter 12 of Reference 1. No significant accidents were identified for solid waste handling.

TABLE 11.2-1
RADIOACTIVE SOLID WASTE ESTIMATE

Source	Approximate Unit		Annual Quantity		Radio-Activity Level		Remarks
	Size	Weight	Ft ³	lb	High	Low	
Sodium Cold Trap		2077 lb (942 kg) empty 2811 lb (1275 kg) full	Once in 20 Years		X		1 cold trap in the 20-year life plant
Sodium Vapor Trap			Once in 20 years			X	1 vapor trap in the 20-year life plant
Sodium Sample	1/2 in. x 6 in. (0.013 m x 0.15 m)	1 lb (0.45 kg)	.1	2 lb (0.9 kg)		X	Reacted sodium in polyethylene bottles with vermiculite
Maintenance and Decontamination Operations Miscellaneous Parts			10	50		X	Estimated annual quantity

11.3 LIQUID WASTE SYSTEMS

As discussed in Section 11.2.2, no liquid wastes are expected for processing. If any liquid wastes are generated that require processing, they will be processed by the FFTF system (Reference 1, Chapter 12).

11.4 RADIOACTIVE GASEOUS WASTE SYSTEM

11.4.1 Design Basis

The radioactive gaseous waste system is designed to collect and discharge the fuel storage vessel cover gas effluent during normal operation, design basis operation, and credible accident conditions. A diagram of the system is shown in Figure 11.4-1. The cover gas effluent is collected in a holdup vessel, monitored and then released to the atmosphere under controlled conditions as specified in Reference 2, noting that for expected operations there should be little or no activity in the cover gas. The dominant radioactive nuclide is xenon-133 for normal operations, and in the design basis case and maximum credible accident, krypton-85. For each, the maximum permissible concentration (MPC_a^{168}) is 3×10^{-7} $\mu\text{Ci/cc}$ offsite -- 4.5 miles (7.24 km) -- and the MPC_a^{40} is 1×10^{-5} $\mu\text{Ci/cc}$ onsite based on a continuous averaged, yearly release rate. The normal and design basis operational cases do not exceed the MPC_a^{40} at the stack discharge on a 24-hour-averaged release rate. It should be noted that although the site boundary is at a distance of 4.5 miles (7.24 km), calculations were also made at a distance of 1.5 miles (2.4 km), which represents a minimum distance to facilities other than those in the 400 Area on the site. At the 1.5 mile (2.4 km) distance, an MPC_a^{40} of 1.3×10^{-6} $\mu\text{Ci/cc}$ was used since there are no living quarters on the site although the uncontrolled dose limit of 500 mrem/yr was used.

FIGURE 11.4-1. Argon System.

11.4.2 System Description

The gaseous effluent from the fuel storage vessel cover gas is collected in a holdup tank, then control-vented to atmosphere. If fission product gases have escaped from elements in storage this effluent may contain radioactivity. The fuel storage vessel ullage volume, 2700 ft^3 (76.5 m^3), is controlled to 8 iwg to 12 iwg (2 to 3 kPa) by a backpressure regulator. The storage vessel effluent dumps into a holdup vessel of 80 ft^3 (2.26 m^3), which is at a negative pressure of -3 lb/in.^2 gage to -8 lb/in.^2 gage (-21 to -55 kPa). The cycle volume for the fuel storage vessel and the holdup vessel is approximately 28 to 30 std ft^3 (0.79 to 0.85 m^3). The effluent, when discharged, exits via the two NDHX stacks where initial dilution takes place. The only normal cover gas throughput is associated with fuel transfer operations where nominally 28 ft^3 (0.79 m^3) of argon purge gas is dumped from the BLTC during connection and disconnection. BLTC operations can occur no more than once every 6 to 8 hours, the minimum transit time. After each transfer operation, the holdup vessel is pumped down to -8 lb/in.^2 gage (-55 kPa) through a $10 \text{ ft}^3/\text{min}$ ($0.28 \text{ m}^3/\text{min}$) (maximum flow) evacuation pump. This corresponds to an average flow of $6.3 \text{ std ft}^3/\text{min}$ ($3.0 \times 10^{-3} \text{ m}^3/\text{s}$). The evacuation pump effluent is discharged through the NDHX exhaust stacks, each of which has a $2500 \text{ ft}^3/\text{min}$ ($1.2 \text{ m}^3/\text{s}$) air throughput, performing the first stage of dilution (located in Figure 11.4-2).

The limiting maximum credible accident operation is a fuel assembly dropped on another fuel assembly, and that their total fission product inventory is released. A description of this accident is found in Chapter 15, Paragraph 15.5.3.2. The resultant radioactive gases are totally contained within the BLTC/fuel storage vessel cover gas system. The radioactive non-inert gases are absorbed in the sodium coolant and do not contribute significantly to the gaseous effluent.

Table 11.4-1 summarizes the pertinent factors and radioactive concentrations for the FSF gaseous waste released under expected operations, design basis

FIGURE 11.4-2. Fuel Storage Facility (FSF).

TABLE 11.4-1
RADIOACTIVE CONCENTRATIONS

(Sheet 1 of 2; footnotes on
sheet 2 of 2)

Condition	Units	Expected Operation	Design Bases Operation	Maximum Credible Accident
1 Radioactive Inventory*	Curies (Ci)	1.20	131	905 ⁽¹⁾
2 Specific Activity in the Covergas	μCi/ml	0.016	1.7	12
3 Holdup Vessel Pump-out Rate (Average)	cc/min	1.8×10^5	1.8×10^5	1.8×10^5
4 Holdup Vessel Pump Down Time (Minimum)	Minutes	4.4	4.4	4.4
5 Air Flow Rate in NDHX	cc/min	1.4×10^8	1.4×10^8	1.4×10^8
6 Radioactive Release Per Pumpout	Ci	0.013	1.3	9.5 (initial) 1.4 (final)
7 Radioactive Release Rate (Maximum)	μCi/sec	0.047×10^3	5.1×10^3	36×10^3
8 Radioactive Concentration from NDHX Stack(s)	μCi/ml as stated	0.002×10^{-3} 75 (Fuel	2.2×10^{-3} 75 (Fuel	15.3×10^{-3}
9 Number of Releases Per Year (Expected Maximum)		Transfers)	Transfers)	143
10 Radioactive Releases Per Year (Maximum)	Ci/yr	0.98	107	778
11 Dispersion Factor X/Q Onsite ⁽²⁾ (10 m)	sec/m ³	1.5×10^{-3}	1.5×10^{-3}	1.5×10^{-3}
12 Dispersion Factor X/Q at 1.5 miles (2.4 km) ⁽³⁾	sec/m ³	1.7×10^{-4}	1.7×10^{-4}	1.7×10^{-4}
13 Dispersion Factor X/Q Offsite ⁽³⁾ at 4.5 miles (7.24 km)	sec/m ³	3.8×10^{-5}	3.8×10^{-5}	3.8×10^{-5}
14 Concentration Onsite During Discharge	μCi/ml	7.1×10^{-8}	7.7×10^{-6}	2.7×10^{-5} (avg) 5.4×10^{-5} (max)
15 Concentration at 1.5 miles (2.4 km) During Discharge	μCi/ml	8.0×10^{-9}	8.7×10^{-7}	3×10^{-6} (avg) 6.1×10^{-6} (max)
16 Concentration at Site Boundary During Discharge	μCi/ml	1.8×10^{-9}	1.9×10^{-7}	6.7×10^{-7} (avg) 1.3×10^{-6} (max)
17 Average Yearly Concentration Onsite	μCi/ml	4.5×10^{-11}	4.8×10^{-9}	3.2×10^{-8}
18 Average Yearly Concentration at 1.5 Miles (2.4 km)	μCi/ml	5.0×10^{-12}	5.5×10^{-10}	3.6×10^{-9}
19 Average Yearly Concentration at Site Boundary	μCi/ml	1.1×10^{-12}	1.2×10^{-10}	8.0×10^{-10}
20 Allowable Concentration Onsite ⁽⁴⁾	μCi/ml	1.0×10^{-5}	1×10^{-5}	1×10^{-5}
21 Allowable Concentration at 1.5 miles (2.4 km) (see text)	μCi/ml	1.3×10^{-6}	1.3×10^{-6}	1.3×10^{-6}
22 Allowable Concentration at Site Boundary ⁽⁴⁾	μCi/ml	3×10^{-7}	3×10^{-7}	3×10^{-7}
23 Release Ratio Onsite During Discharge (Release/Allowance)		.007	.77	2.7**
24 Release Ratio at 1.5 miles (2.4 km) During Discharge		.006	.70	2.3**
25 Release Ratio at Site Boundary During Discharge (Release/Allowance)		.006	.63	2.3**
26 Yearly Average Release Ratio Onsite (Release/Allowance)		4.5×10^{-6}	4.8×10^{-4}	3.2×10^{-3}
27 Yearly Average Release Ratio at 1.5 miles (2.4 km) (Release/Allowance)		3.8×10^{-6}	4.8×10^{-4}	3.2×10^{-3}
28 Yearly Average Release Ratio at Site Boundary (Release/Allowance)		3.7×10^{-6}	4.0×10^{-4}	2.7×10^{-3}

FSF TABLE 11.4-1
RADIOACTIVE CONCENTRATIONS
(Sheet 2 of 2)

- (1) Chapter 15, Section 15.2.2.3.2, "Fuel Storage Facility FSAR Accident Analysis Section."
- (2) Meteorology and Atomic Energy - 1968, TID 24190, USAEC (1969), US Department of Energy, Washington, DC.
- (3) FFTF Atmospheric Dispersion Factors, FFTF FSAR Table 2.3-12, Calculations Per Regulatory Guide 1.4, June 1973.
- (4) "Standards for Radiation Protection," ERDAM Appendix 0524, Annex A, April 8, 1975.

* For the expected operation case the inventory is assumed to be 22.5% Kr-85, 2.5% Xe-131m, 75% Xe-133. For the design basis operation and the maximum credible accident the inventories are assumed to be 100% Kr-85.

** These concentrations assume that the operator would release the total activity to the atmosphere, which is prevented by administrative controls based on measured activity limits. The activity if bled to the atmosphere would be well below limits of the concentration guide.

operations, and hypothetical accident scenarios (Chapter 15, Section 15.2.2.3.2). The only nuclide of any significance in the latter two cases is Krypton-85 while the other gaseous nuclides, iodine and xenon, are many orders of magnitude below permissible concentrations of Reference 2. The radioactive inventory associated with expected operation was derived by assuming the inert gaseous activity in the IDS was at 10% of its design value of 500 Ci, that this activity equilibrated in the relative volume of the BLTC, and the activity in the BLTC was introduced into the FSF cover gas. During fuel addition to the BLTC at the IDS, argon gas flow is from the BLTC, but gaseous equilibration was assumed in the calculation. The average radioactive gaseous equilibration was assumed in the calculation. The average radioactive gaseous mixture in the FSF at equilibrium was calculated to be 0.27 Ci Kr-85, 0.029 Ci Xe-131m, 0.001 Ci Xe-133m, and 0.90 Ci Xe-133. The design basis operational radioactive inventory was derived from Reference 3 considering a filled fuel storage vessel and 0.1% of the radioactive gases from the fuel is released to the cover gas volume. The radioactive inventory was assumed to be evenly distributed throughout the cover gas and holdup vessel.

The results shown in Table 11.4-1 indicate that for any operating condition the concentration guides of Reference 2 should not be exceeded even instantaneously and the yearly average concentration will be < 1% of these limits. If the maximum credible accident condition occurred and the operator released the maximum concentration of activity, the limits may be exceeded by a factor of approximately 2-1/2 for four minutes, but even in this case the yearly average concentration limits of Reference 2 could not be exceeded.

11.4.3 System Design

The gaseous radwaste system is shown schematically in Figure 11.4-1; it consists of the fuel storage vessel ullage volume, lines and backpressure regulator, floor connections to the BLTC, holdup vessel, vacuum pump, radiation monitor, and NDHX stacks. The principal equipment items are the fuel storage vessel ullage space, holdup vessel, and vacuum pump.

The storage vessel is described in Chapter 4. The ullage space of approximately 2700 ft^3 (76.5 m^3) represents the pivotal system volume since the BLTC basically vents into this space. The maximum credible accident is assumed to release all the non-sodium-absorbed radioactive gases from the two ruptured fuel elements into the vessel volume. The holdup vessel is designed to Section III, Class 3, of the ASME Code (Reference 4). Basically, it is an 80-ft^3 (2.3 m^3), $\pm 15 \text{ lb/in.}^2$ gage (0-100 kPa) vessel designed to operate at ambient temperatures. The holdup vessel is sized to accommodate the gas associated with one fuel transfer via the BLTC connect and disconnect purges. The vessel discharge is limited by a $10 \text{ ft}^3/\text{min}$ ($4.7 \times 10^{-3} \text{ m}^3/\text{s}$) vacuum pump.

The holdup vessel is pumped down by a vacuum pump, which operates in a differential pressure of up to 9 lb/in.^2 (62 kPa) with a nominal displacement of $10 \text{ ft}^3/\text{min}$ ($4.7 \times 10^{-3} \text{ m}^3/\text{s}$).

From the fuel storage vessel through the holdup tank, the argon system pressure boundary is designed to Seismic Category I. Since all radioactive gas is contained in this part of the system until intentionally released, failure of non-Category I components and equipment will not result in a radioactive release.

11.4.4 Operating Procedures

The operating procedures will contain administrative controls to assure that appropriate monitoring of radioactivity released is recorded and that limitations in emergency situations are observed. During non-emergency operations no administrative controls are required to assure that the airborne radioactive concentration limits of Reference 2 are not exceeded.

Should the maximum credible accident release occur, under the most conservative limits the discharge will be controlled to the limit conditions on site and at the site boundary.

11.4.5 Performance Tests

Besides normal periodic instrumentation calibration, the radioactivity release will be monitored as described in Chapter 9 of Reference 1.

11.4.6 Estimated Release

The only release from the gaseous waste system will be through the NDHX stacks. Table 11.4-1 presents the expected release from FSF both for normal and maximum credible accident situations.* Other releases arising from accident situations are described in Chapter 15.

11.4.7 Release Points

The radioactive gaseous waste system releases through the NDHX stacks shown in Figure 11.4-2. These stacks have the following characteristics:

Base Elevation	EL 550 ft
Orifice or exit elevation	EL 610 ft
Orifice or pipe inside diameter	23.25 in. (0.59 m)
Effluent velocity, nominal	3.8 ft/s (1.16 m/s)
Effluent velocity, maximum	14 ft/s (4.26 m/s)
Exit temperatures	350 ⁰ F (117 ⁰ C) (approx.)

There are no radwaste system effluents released through the H&V building vent systems. Accident situations are discussed in Chapter 15.

* These releases represent the worst case situations arising from a full storage vessel, which meet the requirements in Reference 3.

11.4.8 Dilution Factors

The dilution factors used for the FSF radwaste system discharges are presented and referenced in Table 11.4-1.

11.4.9 Estimated Doses

The specific controlled and uncontrolled area concentrations are presented in Table 11.4-1. During expected operations the only source of activity is carryover from the IDS which at equilibrium in the FSF is approximately 23% Kr-85, 2% Xe-131m, and 75% Xe-133. The only isotope of significance in the design case or the maximum credible accident case is Kr-85. The radioactive properties of these isotopes are given in Table 11.4-2. The recommended concentration guide (CG) in air for these radioactive gases, as listed in Reference 2, is based on the external dose rate that would be received by a person when he is completely submerged in a semi-infinite cloud (i.e., intercepted by the ground plane) of the isotope. Controlled areas concentrations are based on 40 hours exposure per 7-day period at an occupational dose limit, and uncontrolled area concentrations are based upon continuous exposure for 7 days (168 hours) at a non-occupational dose limit. The concentration guide used at the 1.5-mile (2.4-km) distance was based on 40 hours per 7-day period and non-occupational dose limits. The appropriate concentration guides are (see Table 11.4-1):

$$\text{Controlled areas (occupational), } \text{MPC}^{40} = 1 \times 10^{-5} \frac{\mu\text{Ci}}{\text{cc}} (40 \text{ hr/7 days}) \\ (5 \text{ rem per year})$$

$$\text{Uncontrolled areas (non residence), } \text{MPC}^{40*} = 1.3 \times 10^{-6} \frac{\mu\text{Ci}}{\text{cc}} (40 \text{ hr/7 days}) \\ (1.5 \text{ mile/2.4 km}) \quad (0.5 \text{ rem per year})$$

$$\text{Uncontrolled areas (general public), (4.5 mile/7.2 km Site Boundary)} \quad \text{MPC}^{168} = 3 \times 10^{-7} \frac{\mu\text{Ci}}{\text{cc}} (168 \text{ hr/7 days}) \\ (0.5 \text{ rem per year})$$

*Derived, see text.

TABLE 11.4-2
RADIATIVE PROPERTIES OF ISOTOPES

Isotope	Kr-85	Xe-131 m	Xe-133
Half Life	10.8 yr	12.0 day	5.3 day
Principle Decay Type	(99+%)	M	β
Energy of Particle (Max)	0.67 Mev	None	0.35 Mev
Gamma Abundance	0.4%	100%	100%
Gamma Energy	0.51 Mev	0.164 Mev	0.081 Mev
MPC _a ⁴⁰ , Controlled	1 x 10 ⁻⁵ μCi/cc	2 x 10 ⁻⁵ μCi/cc	1 x 10 ⁻⁵ μCi/cc
MPC _a ¹⁶⁸ , Uncontrolled	3 x 10 ⁻⁷ μCi/cc	4 x 10 ⁻⁷ μCi/cc	3 x 10 ⁻⁷ μCi/cc

TABLE 11.4-3
YEARLY DOSE TO HYPOTHETICAL POPULATIONS RESIDING AT SPECIFIED
LOCATIONS FOR THE INDICATED TIME PERIODS*

<u>Condition</u>	<u>Occupancy Time (hr/wk)</u>	<u>Limit (mrem)</u>	<u>Expected Operation</u>	<u>Yearly Dose (mrem)</u>	
				<u>Design Worst Case Operation</u>	<u>Maximum Credible Accident Condition</u>
On Site	40	5000	<<1	3	16
At 1.5 miles	40*	500*	<<1	<1	2
At Site Boundary	168	500	<<1	<1	<1

* Most limiting dose listed are to the skin, see Table 15.1-2

The yearly doses to the various hypothetical populations were calculated by multiplying the yearly average release ratio (Table 11.4-1) by the dose per year for the corresponding MPC^{ahr}. The yearly doses are given in Table 11.4-3. With the exceptions of the maximum credible accident condition on site, which is less than 0.5% of the annual limit, the annual doses are so low as to be nonmeasurable over natural background radiation for the area (~100 mrem/yr).

11.5 PROCESS AND EFFLUENT RADIOLOGICAL MONITORING SYSTEM

11.5.1 Design Objectives

The FSF has a radiation monitoring system to provide data concerning radioactive releases from the facility to the environment as well as to detect above-normal radiation levels within the facility with respect to equipment malfunction and operating personnel safety. All radiation monitoring instrumentation has backup power to ensure the operability of the system if normal site power fails. A high radiation alarm actuates an annunciator in the FSF control room and a single channel in the FFTF control room. A schematic of the monitoring system is given in Figure 11.5-1.

11.5.2 Description and Design Features

11.5.2.1 Storage Vessel Monitoring

The FSF storage vessel is provided with a neutron monitoring system for diagnostic purposes. Neutron detection is made using boron trifluoride type detectors providing high neutron sensitivity. Two detectors mounted in a steel thimble in the storage vessel vault wall are used in the system. The instrument thimble has been located such that an unobstructed view of the storage vessel is obtained by the detectors. The detectors are placed within the vault wall to provide maximum sensitivity of the detectors because of thermalizing properties of the vault wall and to physically protect the detectors. The detectors are positioned within the thimble so that the

FIGURE 11.5-1. Radiation Monitoring System.

neutron sensitive portions of the detectors face the active portions of the stored fuel assemblies. The system is not intended for use as a radiation alarm system component.

11.5.2.2 Argon Holdup Tank Monitoring

The argon holdup tank, which provides storage of argon cover gas, has a gamma-sensitive monitoring system. The monitoring system ascertains the radioactivity level resulting from the fission gas (Kr-85/Xe-133) and indicates the applicable effluent discharge rate when the tank is vented to atmosphere. Gamma radiation level detection is by a scintillation detector. The detector is in proximity to the holdup tank with an unobstructed viewing angle.

Signals from the detector are processed both locally and in the FSF control room. Local indication and annunciation of radiation levels are provided near the holdup tank vent vacuum pump switch to permit operating personnel to verify the activity level before venting the tank to atmosphere. The FSF control room also indicates and annunciates holdup tank activity. The high-radiation level annunciation is actuated if the contents of the holdup tank are such that, if released at the maximum discharge rate the uncontrolled area concentration guide, as defined in Reference 2 for the most limiting case of isotopic mixes, would be exceeded by a factor of 3 at the stack exit based on a weekly average. Periodic system test and calibration is by a permanently installed check source in the detector.

11.5.2.3 Effluent Discharge Monitoring

To ensure and document that the effluent released from the argon holdup tank is well below the concentration guides given by ERDAM 0524, an in-line beta and gamma scintillation-type detector, calibrated for Kr-85, is installed in the discharge line between the argon holdup tank and the NDHX. A flow transmitter is also installed in this line to monitor total effluent discharge. Continuous indication and recording of effluent discharge parameters,

such as effluent discharge flow rate (std ft³/min) and effluent activity level $\mu\text{Ci/cc}$), are available in the FSF control room. In addition, the total amount of radioactive discharge is computed and recorded based upon activity level and flow rate. A high-radiation alarm is triggered if the discharge is such that the maximum permissible concentrations, as defined in ERDAM 0524, would be exceeded if this discharge rate were to continue for one hour.

The effluent monitoring system is designed in accordance with ERDAM 0513 (Reference 5). Periodic system test and calibration is made using an integral check source in the detectors.

11.5.2.4 Cold Trap Monitoring

The cold trap, which is part of the sodium purification system, has an ion chamber-type gamma sensitive detector mounted in proximity to the cold trap with an unobstructed viewing angle. This detector monitors the radioactivity level of impurities removed from the storage vessel sodium. Indication of cold trap activity level is available in the FSF control room. There are no alarms or annunciation associated with this detector. Periodic test and calibration is by an integral check source in the detector.

11.5.2.5 Vault Atmosphere Monitoring

The cold trap and fuel storage vessel vault atmospheres are monitored for radioactivity levels by monitoring the activity level of the main vault cooling duct in the vault cooling equipment room. A GM type beta/gamma sensitive detector is installed in the cooling duct for this purpose. The vault atmosphere radioactivity level is indicated and annunciated both locally in the vault cooling equipment room and in the FSF control room. A high-radiation alarm is triggered should the radiation level exceed one order of magnitude above the operating level. The high-radiation alarm activates the FSF high-radiation alarm channel to the FFTF control room. A periodic functional test is made using an integral check source in the detector.

11.5.2.6 FSF Ventilation System Monitoring

The FSF ventilating system has radiation monitoring instrumentation to measure the FSF internal atmospheric radioactivity levels. A GM type beta-gamma sensitive detector is installed in the main air return path. Both local (H&V equipment room) and remote (FSF control room) indication and annunciation are provided. A high-radiation alarm is triggered if the radiation level increases one order of magnitude above the operating level. Should the radiation level exceed three orders of magnitude above the operating level, an immediate FSF building evacuation alarm is triggered (not to be confused with 400 Area general evacuation alarm). The FSF building evacuation alarm is designed using guidance of ANSI N2.3 (Reference 6) and produces a mid-frequency sound wave at every building location where operating personnel may be present. The amplitude is modulated at subsonic frequency of not less than 75 decibels nor more than 115 decibels and at least 10 decibels above the maximum background noise level. The high-radiation alarm activates the FSF high-radiation alarm channel to the FFTF control room.

The ventilation system monitoring system is powered by 120 V, 60 Hz power with backup power to both the monitoring system and the evacuation alarm. Periodic functional test is by an integral check source in the detector.

11.5.2.7 High-Bay Area Monitoring

The high-bay area is provided with three GM type beta-gamma sensitive detectors. Two detectors are near the operating floor level and are protected from physical damage by an open wire screen surrounding the detectors. The third detector is approximately 15 ft above the floor in the high-bay area. Indication and annunciation are both local in the high bay and in the FSF control room. A high radiation alarm is triggered should the radiation level exceed one order of magnitude above the operating level. If the radiation level exceeds three orders of magnitude above the operating level, an immediate evacuation alarm (see Sections 11.5.2.6 and 9.8.1.7) is triggered.

The high radiation alarm activates the FSF high-radiation alarm channel to the FFTF control room. Periodic functional testing will be performed using external calibration sources.

11.6 RADIOACTIVE SOLID WASTE SYSTEMS

FSF operation will not generate a significant quantity of radioactive solid waste, and what is generated will be processed through the FFTF solid waste disposal system.

11.6.1 Design Objectives

The solid waste volume generated by FSF will be only one or two items per year in the beginning and less near the end of the FSF design life cycle.

11.6.2 System Inputs

The potential FSF solid wastes will come from the sodium cold trap, argon cover gas, vapor trap, sodium sample station, and miscellaneous drop cloths, rags, etc. The maximum sodium radionuclide inventory is listed in Table 11.6-1.

11.6.3 Equipment Description

Solid wastes are packaged as required for transfer into the FFTF radioactive solid waste handling system (described in Chapter 12 of the FFTF FSAR). Packaging materials and personnel to perform these operations are supplied by FFTF.

11.6.4 Expected Volumes

The maximum estimated volumes and weight that FSF may be expected to generate are shown in Table 11.2-1.

TABLE 11.6-1

SUMMARY

FSF SODIUM - RADIOACTIVE SOURCES
(238,000 lb (107,956 kg) Sodium)

Isotope	Source			Total Potential Inventory $\mu\text{Ci/gm}$
	IDS* Carryover $\mu\text{Ci/gm}$	Max Credible Accident $\mu\text{Ci/gm}$	Random Failure 0.1% Failed Fuel Elements 466 Assemblies $\mu\text{Ci/gm}$	
^{134}Cs	1.9×10^{-3}	5.61	0.9	6.5
^{136}Cs	3.1×10^{-3}			
^{137}Cs	6.3×10^{-2}	16	5.1	21.2
^3H	1×10^{-4}			**
^{131}I	3.9×10^{-3}			**
^{22}Na	7.7×10^{-3}			**
^{24}Na	1.1×10^{-1} (Na activation)			0.1
^{241}Pu	1.5×10^{-3}			**

*After 20 years of operation

**Negligible compared with ^{137}Cs

11.7 OFFSITE RADIOLOGICAL MONITORING PROGRAM

The Hanford site covers about 560 square miles (1500 km²). Operations within this boundary include those of the DOE contractors and privately owned facilities. In support of contractor operations, DOE and its predecessors have maintained an environmental surveillance program at and near Hanford continuously since 1944.

The current Hanford environmental surveillance program is conducted for DOE by Pacific Northwest Laboratories operated by Battelle. Results of this effort are published annually in conformance with Reference 5. In the following sections, the program and expected results associated with external radiation doses are discussed, since external doses are the only expected dose-to-man effects from operation of the FSF. A more detailed discussion of the environmental surveillance program, including air particulate, soil, water, vegetation and animal monitoring, is given in Reference 1, Section 12.1.

11.7.1 Expected Background

Although the environmental monitoring program is designed to monitor those nuclides found in gaseous and liquid effluents of the Hanford site, only the airborne inert gases are of significance from operations of the FSF, as discussed earlier in this chapter. These nuclides contribute significantly only to the dose from external sources. The average external radiation dose (from all sources including natural background) is approximately 0.2 mrem/day at the site boundary. The operation of the FSF will not be expected to cause any measurable increase in this background.

A more detailed discussion of all background levels is given in Section 12.1.6.1 of Reference 1.

11.7.2 Critical Pathways

The evaluation of environmental surveillance and operational data considers dose to man via the pathways from all Hanford site releases of radioactivity, which is described in Section 12.1.6.2 of Reference 1. For operation of the FSF the only expected releases to the environment would be inert gases, whose only significant pathway to cause radiant energy depositon (radiation dose) to the body is from the decay of the radioisotopes in the atmosphere external to the body.

11.7.3 Sampling Media, Locations, and Frequency

External dose measurements are made using thermoluminescent dosimeters (TLDs). They are located at the site perimeter communities (13 currently) and at distant communities (6 currently) for comparison as well as at locations within the site and at site boundaries. The communities selected surround the site. The TLDs are continuous monitors, processed monthly. Expected releases of noble gases from FSF, combined with those from FFTF, are not expected to produce any measurabe increase in the radiation level.

11.7.4 Data Retention and Presentation

All environmental surveillance data is maintained on computer files to facilitate storage, retrieval, and processing. Graphic plots of the data are prepared to aid in detection of significant trends and anomalous data points. Monthly reports of significant observations are made to HEDL and DOE.

Environmental surveillance reports are published annually in compliance with Reference 5.

11.8 REFERENCES

1. Fast Flux Test Facility Final Safety Analysis Report, HEDL-TI-750001, Hanford Engineering Development Laboratory, Richland, WA, December 1975.
2. ERDA Manual, "Standards for Radiation Protection," Chap. 0524, Annex A, US Department of Energy, Washington, DC, March 1977.
3. Fuel Storage System System Design Description, N106SDD210001, Rev. 1, Atomics International, Canoga Park, CA, April 1978.
4. ASME Boiler and Pressure Vessel Code, Section III, The American Society of Mechanical Engineers, New York, NY.
5. ERDA Manual, "Effluent and Environmental Monitoring and Reporting," Chap. 0513, US Department of Energy, Washington, DC, March 1974.
6. ANSI N2.3, American National Standards Institute, The American Society of Mechanical Engineers, New York, NY.

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CHAPTER 12
RADIATION PROTECTION

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12.0 RADIATION PROTECTION

12.1 GENERAL

It is the policy in the FSF to assure as low as reasonably achievable (ALARA) occupational radiation exposures, as specified in ERDAM 0524 (Reference 1). The FSF is located adjacent to and within the FFTF site boundary. As such, the policy considerations, area monitoring, and health physics program applicable to the FFTF and described in the FFTF FSAR (Reference 2) also apply to the FSF. Specific applicable portions of the FFTF FSAR are: Section 12.2.1.1, Policy Considerations; Section 12.2.3.4, Area Radiation and Airborne Radioactive Monitoring Instrumentation; and Section 12.2.5, Health Physics Program. The radiation protection procedures unique to the FSF are described in this chapter.

12.2 SHIELDING

The shield design philosophy is threefold: (1) to have expected dose rates as low as reasonably achievable, (2) to have design basis operational dose rates not to exceed the desired occupancy criteria dose rates described in Paragraph 12.2.1, and (3) to have the maximum credible accident dose rates low enough to perform necessary functions. The expected dose rates are based on having no uncontained fuel elements in the FSF. This condition is expected because (1) any elements with deliberate or accidental cladding failures from FFTF would be recanned prior to insertion into the FSF, and (2) the environment within the FSF is passive, which would not be expected to cause any cladding failure. The design operational dose rates assume a 0.1% cladding failure of elements with their full inventory of fission products released into the storage vessel. The maximum credible accident case assumes the complete rupturing of all pins in two peakpower fuel assemblies and their activity adding to that of the activity induced from the design operational case. The potential for buildup of activity over the 20-year service life was considered where applicable.

12.2.1 Design Objectives

Radiation levels in the FSF shall be controlled and monitored according to the following requirements:

- 1) Personnel exposure limits shall be set per Reference 3. To achieve these limits, the design basis radiation levels shall not exceed the following levels.
 - 0.2 mrem/hr - uncontrolled area, routine occupancy (control room, restrooms, and high bay area when not fueling)
 - 2 mrem/hr - controlled area, nonroutine occupancy <2 hr/day (NDHX vault and Electrical Equipment Room)
 - 20 mrem/hr - controlled area, limited access, some routine tasks such as radiation surveys and maintenance performed for <1 hr/wk (below grade corridors and argon holdup tank vault)
 - 200 mrem/hr - controlled area, limited access, only non-routine tasks performed, <1/2 hr/month (below grade process equipment vaults)
- 2) Personnel monitoring devices shall be provided per Reference 3.
- 3) Radioactive effluent control, disposal, and monitoring shall be performed in accordance with Reference 3.
- 4) A k_{eff} of 0.95 or less shall be maintained under all credible configurations per Reference 4. Criticality calculations shall be based on the most reactive configuration of the following fuel assemblies:

Cores 1 and 2, inner zone	59 assemblies
Cores 1 and 2, outer zone	95 assemblies
Cores 3 and 4, inner zone	120 assemblies
Cores 3 and 4, outer zone	<u>192 assemblies</u>
	466 assemblies

- 5) Neutron detectors shall be provided for diagnostic purposes.
- 6) Personnel doses at the site boundary resulting from any postulated credible accident shall not exceed the limits of 10CFR100 (Reference 5).
- 7) Shielding calculations shall be based on the inventory of 60,000 MWt day burnup fuel having 200 days of decay.

12.2.2 Design Description

12.2.2.1 FSF Storage Vessel Shielding and Dose Rates

Figures 12.2-1 and 12.2-2 show the shielding arrangements for the FSF vessel head area in the high bay. The dose rates above a fully loaded facility are summarized below in Table 12.2-1.

TABLE 12.2-1

SUMMARY OF RESULTS
OF THE FULLY LOADED FACILITY
(Operational Dose Rate Allowance is 0.2 mrem/hr)

<u>Dose Point</u>	<u>Shield</u>	<u>Operational Dose Rate (mrem/hr)</u>	<u>Maximum Credible Accident Dose Rate (mrem/hr)</u>
Point A, Figure 12.2-1	12.0 in. (0.3 m) Carbon 12.0 in. (0.3 m) Closure Head 4.0 in. (0.1 m) Borated Polyethylene 3.0 in. (0.08 m) Striker Plate	0.12	0.39
Point B, Figure 12.2-1	12.0 in. (0.3 m) Carbon 12.0 in. (0.3 m) Closure Head 6.5 in. (0.16 m) Rail	0.16	0.40
Point C, Figure 12.2-2	9.0 in. (0.22 m) Support Ring 4.0 in. (0.1 m) Borated Polyethylene 4.0 in. (0.1 m) Steel Shield 1.0 in. (0.025 m) Cover Plate	0.04	0.08
Point D, Figure 12.2-2	9.0 in. (0.22 m) Support Ring 4.0 in. (0.1 m) Borated Polyethylene 6.5 in. (0.16 m) Rail	0.04	0.08

FIGURE 12.2-1. General Arrangement of the FSF Shielding.

FIGURE 12.2-2. Shielding Above the Support Ring.

Figure 12.2-3 shows the basement plan, and Figure 12.2-4 shows the ground floor plan of the FSF facility.

The predicted operational dose rates meet the allowable dose rates of 0.2 mrem/hr. Except for Point B which has a 20% margin and is a localized area, there is a very large margin, hence exposures will be low. For the normal and design basis operational conditions there is no significant difference.

In the maximum credible accident case, the element in the channel would be withdrawn, and routine operations could not continue until it was. The total fission product inventory of the two affected assemblies were calculated to be released into the sodium coolant and cover gas. Since the fission product gamma rays contribute only a small portion of the dose rate (<0.005 mrem/hr) (Reference 6), the operating dose rate would increase only <0.04 mrem/hr subsequent to the accident case.

The shield design and dose rates associated with fuel transfer are presented in Section 11.11 of Reference 2. During fuel transfer operations personnel are protected from excessive radiation exposure by adequate radiation shielding, controlled accessibility and limiting the time that personnel remain in the radiation fields. A maximum transient (~ 20 seconds) dose rate of 200 mrem/hr can occur at the closure valve/floor valve region during the actual vertical transfer. At all other times dose rates of less than 2 mrem/hr are predicted at the surface of the transfer cask (BLTC), the fuel storage vessel top and the connection between the two. The operating and emergency conditions for fuel transfer are discussed in Section 11.11 of Reference 2.

At the basement level, lateral to the vessel, 4 ft to 4-1/2 ft (1.2 m to 1.4 m) of concrete provides adequate shielding to attenuate the radiation from the stored assemblies so that all passageways, the argon holdup tank room, and the electrical equipment room have dose rates less than

FIGURE 12.2-3. Basement Plan.

HEDL 7909-194.7

FIGURE 12.2-4. Ground Floor Plan.

0.2 mrem/hr. The basement plan is shown in Figure 12.2-3, indicating the arrangements and the concrete shielding.

12.2.2.2 Auxiliary Equipment Shielding

The shield design for these components is presented in Reference 7.

Cold Trap -- The cold trap removes, with varying degrees of efficiency, the impurities in the FSF sodium. The activity for expected normal operations includes carryover from the IDS (transferred activity) accumulated over the 20-year operational life and sodium activation. The design basis operation adds the fission product activity deposited from 0.1% random pin failure from 466 average powered fuel assemblies. The accident case adds the inventory of fission products from two crushed assemblies. The source terms for the accident case are in Paragraph 12.2.3. To meet the 200-mrem/hr criterion in the cold trap room for design basis operations, a lateral steel shield nearly 6 in. (15 cm) thick would be needed adjacent to the cold trap. This shielding for the design basis operation has not been installed as it can be added if conditions warrant. Note that the unshielded dose rate for the design basis operation is 155 rem/hr (end of design life), assuming no cold trap replacement. For the expected operations the dose rate would be <10 mrem/hr at 1 ft (30.4 cm) and for the accident condition <730 rem/hr at 1 ft (30.4 cm) with no steel shield addition. If the dose rate from the cold trap builds to a dose rate level near 200 mrem/hr, there are two options in addition to shielding addition: (1) changing out the cold trap, and (2) redesignating and controlling the area as a no-access-permitted area. The change-out option is the most attractive option, from both radiological and operational standpoints.

In areas away from the cold trap, dose criteria are met. At the unrestricted high bay area (ground level), the distance from the cold trap and 3-ft (.9 m) concrete floor provide adequate attenuation so that the expected normal operating dose rate is <0.2 mrem/hr, the design basis

operating dose rate is <0.2 mrem/hr and the accident case is <10 mrem/hr. In the below grade corridor, the 2-ft (.6 m) concrete walls provide dose rates of <0.01 mrem/hr for the expected operational case, 3 mrem/hr for the design basis operational case, and 11 mrem/hr for the accident case. The criterion of less than 20 mrem/hr has been met in all cases.

Argon Holdup Tank -- The argon holdup tank does not present a significant exposure problem because it must have an activity such that the concentration of Kr-85 at the site boundary is $<3 \times 10^{-7}$ $\mu\text{Ci/cc}$ for release to the atmosphere. From Reference 7, the dose rate at the surface of the tank is approximately 1.5 mrem/hr corresponding to an activity in the tank of 0.86 Ci of Kr-85, which meets the criteria. This activity is approximately the activity if the cover gas were at the design activity.

Natural Draft Heat Exchanger Vault -- The NaK coolant is isolated from the sodium primary coolant, therefore there is no fission product activity in the secondary coolant. Hence, there would be no significant difference in dose rates between the three cases previously considered. The only source of activity is the sodium and potassium activation from spontaneous fission neutrons in the storage vessel. In Reference 7, calculations have shown that the activity induced in the NaK would give an unshielded dose rate at the surface of a large pool of this NaK of 0.16 mrem/hr. Therefore the dose rates in the vault are well below the criterion of 2.0 mrem/hr.

TABLE 12.2-2
SUMMARY OF MAXIMUM DOSE RATES

<u>Location</u>	<u>Criterion (mrem/hr)</u>	<u>Expected Operation (mrem/hr)</u>	<u>Design Worst Case Operation (mrem/hr)</u>	<u>Max. Credibl Accident Condition (mrem/hr)</u>
High Bay, etc.	0.2	0.16	0.16	<10
Cold Trap Vault	200	<10	~200*	~200*
Argon Holdup Tank	20	<0.2	<1.6	<12
Natural Draft Heat Exchanger Vault	2	0.2	0.2	0.2
Underground Corridors	20	<0.1	3	11
Electrical Equipment Rm.	2	<0.1	<1	2

* If no action were taken, the design basis case operations would give a dose rate at 1 ft (.3 m) of 155 rem/hr, and the accident case would give 730 rem/hr. Administrative control will be used to normally limit the dose rate to approximately 200 mrem/hr by changing out the cold trap or adding shadow shielding. If necessary the area could be redesignated a no-access permitted area; and the area could be completely secured. Change out would be the most attractive option normally from both radiological and operational standpoints.

12.2.3 Source Terms

The source terms for the FSF facility are shown in Appendixes A and B.

The design source term for the argon holdup tank is $0.38 \mu\text{Ci/cc}$ of Kr^{85} in a total volume of 80 ft^3 (2.26 m^3) (Reference 7). The induced activity in the NaK loop is $4.6 \times 10^{-5} \mu\text{Ci/cc}$ (Reference 7). The activity in the cold trap under accident conditions is as shown in Table 12.2-3.

TABLE 12.2-3
FSF COLD TRAP ACTIVITY FOR THE ACCIDENT CASE

<u>Isotope</u>	<u>Total Inventory (curies)</u>	<u>Fraction in the Cold Trap</u>	<u>Activity in the Cold Trap (curies)</u>
Pu 241	0.16	1.0	0.16
Ce 144	2.9	0.10	0.29
Cs 137	4000	0.80	3230
Cs 136	0.33	0.05	0.017
Cs 134	0.20	0.60	0.12
I 131	0.42	0.05	0.021
Sb 125	110	0.90	100
Ru 106	7.1	0.10	0.71
Sr 90	3.3	0.10	0.31
Na 24	12	0.0046	0.06
Na 22	0.83	0.0046	3.8×10^{-3}
H 3	0.01	0.98	0.01

12.2.4 Area Monitoring

The FSF is provided with features which permit monitoring of:

- Storage vessel neutron flux
- Argon holdup tank radionuclides
- Effluent discharge activity
- Cold trap contamination
- Vault atmosphere activity
- HVAC system activity
- High bay radiation levels

A description of the detector types, locations, and operation is contained in Reference 8.

12.2.5 Operating Procedures

The majority of plant operations will be conducted in areas where continuous occupancy is allowed. Fuel transfer operations, which will involve somewhat higher dose rates during the time that spent fuel is being transferred through the floor valve, will be monitored and provided with additional temporary shielding, if necessary. The procedure for monitoring and operating the BLTC is contained in the BLTC Operations and Maintenance Manual. The FSF fuel handling shielding analysis is documented in Reference 9. Predicted dose rates are consistent with DOE limits and ALARA concepts in Reference 1.

Maintenance at the FSF will be carried out with appropriate radiological monitoring and precautions per the operating procedures of the Hanford Engineering Development Laboratory (HEDL) which has been assigned as the operator of the FSF. The operating procedures used by HEDL have passed DOE review on numerous occasions for nuclear facilities of a similar type, and which involve similar radiological hazards.

12.3 VENTILATION SYSTEM

The FSF building ventilation system does not perform a radiation protection function. There are no provisions for building atmosphere isolation in the event of radiation release accident because the identified accidents (see Chapter 15) do not release sufficient radionuclides to require isolation or confinement in order to meet radiological safety criteria. The ventilation system is described in Reference 7 and it has a variable ventilation rate of 500 ft³/min to 16,000 ft³/min (0.24 m³/s to 7.5 m³/s). In the event of a fire in the control room, the air circulating fans will shut off, and the dampers to that area will be closed automatically to prevent interference with the operation of the Halon fire suppression system (See Section 9.9.2.2).

12.4 HEALTH PHYSICS PROGRAM

12.4.1 Organization and Objectives

The health physics program for the FSF will be administered by the FFTF Health and Safety Organization. This section describes the policy considerations and organization of HEDL Safety Department, of which FFTF Health and Safety is a part.

12.4.1.1 Objectives

The policy of HEDL regarding the exposure of employees, visitors, and the general public to ionizing radiations attributable to HEDL operations is to reduce the exposure to the minimum practicable level in conformance with applicable sections of Reference 1. This policy is responsive to the recommendations of recognized authoritative bodies in the field of radiation protection and complies with the applicable law and regulations including ERDAM 0524. DOE regulations for safety apply to all

construction and operation of DOE facilities. Radiation exposure is minimized by identifying potential radiation hazards, shielding, confining radiation sources; planning operations to minimize individual and total man-rem exposure, training in radiation protection, and measuring and limiting the time of external exposure or quantity of ingested radioactive material.

HEDL operating controls shown in Reference 3 further regulate monthly and annual radiation doses to employees to levels significantly less than occupational dose limits of ERDAM 0524. Guides for emergency radiation doses to save vital equipment and human life are presented in Reference 3.

The FFTF Project Manager is responsible for operations of the plant and related programs in conformance with the HEDL policies to assure that radiation exposure is maintained as low as practicable. Health physics functions essential for day-to-day operation are located at the plant site and results are reported to the FFTF Plant Manager. Radiation monitors on all shifts are part of the FFTF Health and Safety staff and are administratively independent of the FFTF Project Manager.

Effectiveness of the radiation protection program is verified by compilation, review, and reporting the results of individually worn dosimeters, urinalyses, whole body counts, and liquid and airborne effluent samples. Further verification of the effectiveness of the radiation safety programs comes from environmental soil, plant, foodstuff, animal, air, and water samples taken on the Hanford Site and off-site by another DOE RL contractor.

Environmental and Radiological Engineering perform periodic radiation protection audits of radioactive material handling and storage operations, and administrative procedures in accordance with Reference 3.

12.4.1.2 Organization

HEDL health physics functions, along with those for criticality safety, environmental control, industrial safety, and fire protection, are organized to be administratively independent of the operations staffs. The health physics program for the FSF is carried out as a part of the FFTF program. It is administered by the Manager of FFTF Safety who reports to the Manager of the Safety Department who in turn has the overall responsibility for safety in all HEDL operations.

12.4.1.2.1 Health Physics Staff

The full-time staff assigned to FFTF consists of the Manager of FFTF Health and Safety, an experienced health physics engineer, a radiation monitor supervisor, qualified radiation monitors, and a nuclear safety and training engineer. The specific qualifications required for each of these positions are given in Reference 2. In addition to the full-time staff of FFTF, the experienced and knowledgeable staff of HEDL is available.

12.4.1.2.2 Personnel Health Physics Orientation

The management and staff that operate the FSF are the operating staff of the FFTF. As such they are indoctrinated and qualified in radiological and nuclear safety, both in basic concepts and in individually assigned functions including those associated with the FSF, as discussed in Reference 2.

12.4.2 Facilities, Equipment, and Instrumentation

The FSF is integrally associated with the FFTF and will utilize its facilities, equipment, and personnel.

12.4.2.1 Facilities

Some of the significant facilities at FFTF that may be of significant importance to operation of the FSF (including accident cases) are:

- the continuously manned control room where high radiation levels measured in the FSF are annunciated, as shown in the Radiation Monitoring System Schematic, Figure 12.4-1
- a low background counting room for sample analysis
- a change room with protective apparel and frisking instrumentation located within it
- storage for portable radiation monitoring instrumentation

For any occasion when more sophisticated backup may be required, the facilities of the HEDL 300 Area and other DOE-RL contractors at the Hanford Site are available.

12.4.2.2 Equipment and Instrumentation

The radiation monitoring equipment installed in the FSF is described in Chapter 11. A schematic of this system is shown in Figure 12.4-1.

Instrumentation is available for air, liquid, smear, personnel contamination, and radiation level monitoring. These systems are capable of being used to sample and/or measure activity at remote locations. Special kits containing instruments and other equipment designed for emergency use are maintained at the FFTF Emergency Control Center and at other accessible locations throughout the Hanford Site. More specific details of instruments and equipment at FFTF are given in Reference 2.

FIGURE 12.4-1. Radiation Monitoring System.

12.4.2.2.1 Portable Radiation Instruments

Portable dose rate meters are used by radiation monitors to measure beta, gamma, and neutron doses over a wide range of dose rates.

Count rate meters are provided for field detection of radioactive contamination. Alpha radiation detectors and beta-gamma radiation detectors are available.

All radiac instruments are maintained and calibrated by the Calibrations Laboratory in the 300 Area. A schedule for calibration and routine maintenance for the instruments is based on the historical optimum operating performance for each type of instrument.

12.4.2.3 Air Sampling

The heating and ventilation air and the vault cooling air activity are continuously monitored in their ducts in the FSF, Figure 12.4-1.

In addition, air particulate sampling can be accomplished, as needed, using filter air sampling heads or high volume samplers (impactors).

12.4.2.4 Protective Clothing and Respirators

Protective clothing for contamination control and air purifying respirators are maintained in the FFTF Change Room. The canisters and the filter cartridges for the respirators are extensively tested to assure proper filtration and the assembled masks are leak tested. Self-contained breathing systems are available.

12.4.2.5 Procedures

12.4.2.5.1 Radiation Surveys

The methods, determination of frequencies and the procedures for conducting radiation surveys are included in Reference 3. They are performed with portable radiation instruments. Radiation surveys are performed to evaluate the radiation and contamination status of facilities and equipment, the conditions to which personnel are exposed, and the effectiveness of existing radiation control measures. The radiation survey program provides information through routine and special surveys. Routine surveys are concerned with the conditions existing in area, buildings, and rooms and on equipment. These surveys are conducted inside and outside of radiation zones according to a schedule that is based on the probability of contamination and undesirable dose rates. FFTF Radiation Monitoring has a list of routine contamination and radiation surveys to be made at prescribed intervals. Special surveys are performed to assure adequate radiation control during specific jobs or experiments.

12.4.2.5.2 Control Programs for ALARA Occupational Radiation Dose

Access controls to the various radiation hazard zones and area monitoring by fixed equipment are discussed in Section 12.2. These systems and their associated procedures (Reference 3) are reinforced and performance-assured by continuous surveillance, including supplemental routine and special surveys performed by FFTF Health and Safety. Such surveys are made with portable instruments described in Section 12.4.2.2.1. Safety personnel are trained to assist in planning for major and minor activities so that personnel radiation doses are ALARA without causing operational difficulties and delays that may ultimately result in greater hazards.

HEDL radiation dose control levels are listed in Section 2.1.2 of Reference 3. These upper-bound working levels help assure that

occupational dose limits are not exceeded, and they provide a basis for work planning. These dose control levels require more stringent exposure control, which is accomplished by (1) identification of radiation hazards, (2) added shielding and other protective devices where feasible, (3) training with mockups to save exposure time in radiation fields, (4) procedure and equipment tests before radiation zone entry, and (5) training and orientation of all employees who may be involved with ionizing radiation or radioactive materials.

Procedures do allow some latitude within the limits for external radiation exposure control; however, risk for internal deposition of radioactive material is minimized. The rigid policy is necessary due to the extremely small radionuclide quantities permissible in the body and the difficulty of quickly and accurately assessing internal doses. Sections 3.1 and 3.2 of Reference 3 provide control methods to prevent internal deposition of radionuclides.

Where the internal and/or external dose received by an individual is uncertain or an emergency dose above normal control guides has been received by an individual, the individual is not assigned to work involving possible radiation exposure until detailed dose assessment is completed and further individual controls have been established.

12.4.2.5.3 Control of Access and Residence Time

All entry and work in a radiation zone is regulated by an applicable valid Radiation Work Procedure (RWP) or an Emergency Procedure under emergency conditions. The RWP is prepared and signed by FFTF Health and Safety and other involved management. The RWP includes the period in effect, locations covered, job description and limitations, known radiation sources and hazards, required protective clothing and equipment, required radiation measurements, and special instructions and requirements.

Long-term RWPs are reviewed periodically. Surveillance is established to detect changing conditions that may alter the controls required. When the established radiation work time limit is less than 30 minutes, an independent time keeper is required. Protective clothing contamination checks may be required during the job and at egress from the zone.

12.4.2.5.4 Protection Against Airborne Radioactivity

Respiratory protective equipment requirements are specified by FFTF Radiation Monitoring prior to or during the course of radiation zone work. Respiratory protection is required whenever personnel work with uncontained and radioactive material quantities that exceed values specified in Reference 3. These values conform with the requirements of Reference 1. Known hazards and air samples representative of the air inhaled by the worker(s) are used to determine need for respiratory protection equipment. Continuous air monitoring is required whenever work is performed without benefit of containment equipment and with radioactive material quantities that exceed those specified in Reference 3. A job is terminated when exposures may exceed 4 (MPC)-hours. The number of (MPC)-hours is given by the product of the measured airborne radioactivity concentration (in $\mu\text{Ci/ml}$) and the exposure period (in hours) divided by the appropriate MPC.

Reference 3 also defines conditions of exposure to airborne radioactive material requiring evaluation of internal dose, and designates concentrations to which exposure is prohibited with or without masks.

Mask fitting and training are required for all individuals who may be required to wear masks on planned work assignments or under emergency conditions. Fitting and training are effected in conjunction with employee physical examinations, since both are conducted by the Hanford Environmental Health Foundation in Richland, where testing equipment is installed.

Each individual is also responsible for fitting his own mask to ensure a proper seal with his face before entering an area requiring the use of a mask.

Only masks approved by HEDL Environmental and Radiological Engineering may be used for control of exposure to airborne radioactive materials. Only masks which use replaceable filters or are air-supplied are used for protection against inhalation of airborne radioactivity. All masks are sterilized before they are worn. A mask may be reworn by the same individual if it is determined to be free of contamination.

Supplied air respiratory protection is required if it is anticipated that the air concentration of radioactivity may exceed 100 times the values in Reference 4.

12.4.2.5.5 Routine Radiation Safety Audits

Periodic audits of the effectiveness of controls, the facility services, operational health physics and radiation protection documentation are conducted by a radiation protection group not directly associated with the operation of the facility. Formal audits are conducted at least annually and informal audits are conducted at least quarterly.

12.4.2.5.6 Training Program

Extensive time and effort are devoted to assure that employees understand radiation and radiation protection as it applies to their work. The amount and type of training depends on the nature and location of their work. Specific details of the scope of frequency of training are contained in HEDL procedures manuals References 3 and 12.

Organization and coordination of HEDL radiation protection training programs is assigned to Environmental and Radiological Engineering. Operating management implements the training program for the employees' work locations.

The application of these HEDL training requirements to FFTF personnel, who operate the FSF, at all levels is included in training programs described in Reference 2. Participation is placed on record in the attending individual's file.

New operational health physics personnel receive special training by senior operational health physics personnel. Reference 11 is the guide for performance of health physics activities.

Shortly after employment by HEDL, radiation workers are provided a general radiation protection orientation covering basic policies, procedures, and identifying signs and symbols.

Before employees work in radiation zones they receive additional training by the operating unit and FFTF Health and Safety. Operating management has responsibility for implementing the training program and provides necessary training to assure that the employee can operate equipment and perform procedures safely. Health Physics personnel provide information on radiation protection practices and procedures applicable to the employee's work location or assignment, the radiations or contaminants he may encounter, his responsibilities for radiation protection, emergency procedures, and the monitoring devices and protective equipment provided for use in radiation measurement and control.

At least once every two years, training sessions are presented which review and update information described above. Presentations are by the employee's management and health physics personnel.

12.4.3 Personnel Dosimetry

Personnel dosimeters are required for entry to the 400 Area. The minimum requirement is a basic thermoluminescent dosimeter (TLD), sensitive to beta and gamma radiation. For radiation workers, a multipurpose TLD, sensitive to beta-gamma and neutron radiation is required. In special instances where hand exposure may be a problem, ring dosimeters are issued. Pocket ionization chambers are used to provide interim exposure information before processing the TLDs and to provide backup in the event of loss of the required dosimeter. The processing and calibration of the TLDs are routinely done by a DOE vendor. The Pacific Northwest Laboratory (PNL) conducts audits of the calibration programs. In addition, PNL operates the Hanford-wide, computer-programmed dosimetry records program. Internal dosimetry results are included in the records program.

12.4.3.1 External Dose Measurement

Individual external ionizing radiation doses are determined and recorded from TLDs.

- 1) Employees who do not work with radioactive materials or in radiation fields and who do not require radiation zone entry in the normal course of their work are assigned a Hanford Basic TLD, which is exchanged annually. The dosimetry is described in Reference 13.
- 2) Employees whose work involves routine entry into radiation zones (i.e., a radiation worker) are assigned a Hanford Multipurpose TLD (beta, gamma, and neutron dose measurement). For any person whose annual whole body dose to penetrating radiation is expected to be greater than the values given in Reference 3, the dosimeter is exchanged monthly. For all other radiation workers, the dosimeters are exchanged quarterly. The badge is described in Reference 14.

- 3) Employees whose extremity dose may be expected to exceed the values given in Reference 7 wear ring dosimeters (gamma dose measurement, only). Those dosimeters are changed monthly.

TLDs are processed and the results recorded and submitted to HEDL under contract with DOE-RL. The contract covers TLD quality control, processing, and audit provisions for TLDs used by all RL contractors at Hanford.

Provision is made in the contracts: (1) for prompt notification of any high dose measurement, (2) for rapid emergency service during off-hours and on holidays, and (3) maintaining and providing HEDL dosimetry records.

12.4.3.2 Internal Dose Measurement

The HEDL program for internal dose measurement uses the techniques of bioassay examination, using both in-vitro and in-vivo techniques. These techniques are used as warranted by the radio-nuclides and activity levels involved.

Employees who will work with radioactive material will have any previous internal deposition of radioactivity evaluated before assignment to radiation work. Employees who work with radioactive materials in dispersible form while employed by HEDL receive an evaluation of internal dose before termination. Employees who work with radioactive material in dispersible form receive regularly scheduled internal dose evaluations. FFTF Health and Safety determines types and frequencies of routine internal dose measurements for each employee according to the risk of internal deposition of radionuclides. They also request special internal dose evaluation when a measurable deposition is suspected.

In-Vivo Analysis -- In-vivo examinations in the DOE-RL whole body counter facility are normally scheduled with the routine employee medical examination. The facility has capability for chest, whole body, bone, thyroid, and wound measurement.

In-Vivo Analysis -- Each month, a DOE-RL contractor laboratory collects and performs specified radiochemical and autoradiographic analyses of excretory specimens from specified HEDL employees. Periodically, blind samples containing known quantities of radionuclides are submitted by another contractor for audit purposes.

The bioassay contractor laboratory's contract provides for prompt notification of any high results, and for rapid emergency service during off-hours and on holidays.

12.5 REFERENCES

1. ERDA Manual, "Standards for Radiation Protection," Chap. 0524, US Department of Energy, Washington, DC, March 1977.
2. Fast Flux Test Facility Final Safety Analysis Report, HEDL-TI-75001, Hanford Engineering Development Laboratory, Richland, WA; December 1975.
3. Radiation Protection Procedures, WHAN-M-8, Hanford Engineering Development Laboratory, Richland, WA
4. J. N. Paglieri and C. A. Rogers, Criteria for Criticality Safety, Limits, and Controls at HEDL, HEDL-TME 73-36, Hanford Engineering Development Laboratory, Richland, WA, April 1973.
5. Code of Federal Regulations, 10 CFR 100, "Reactor Site Criteria," USGPO, Washington, DC, January 1980.
6. Dose Rates Above the Top Shield of the FSF, N106TI210007, Rev. A, Atomics International, Canoga Park, CA, June 1977.
7. Miscellaneous FSF Shielding Analysis, N106TI210004, Rev. A, Atomics International, Canoga Park, CA, June 1977.
8. Fuel Storage Facility System Design Description, N106SDD210001, Rev. 1, Atomics International, Canoga Park, CA, April 1978.
9. Fuel Storage Facility Fuel Handling Shielding Analysis, N106TI210009, Atomics International, Canoga Park, CA.
10. Portable Radiation Survey Instrument Manual, BNWL-MA-62, Pacific Northwest Laboratory, Richland, WA, August 1969.
11. Operational Health Physics Procedures, HEDL-TME 72-20, Hanford Engineering Development Laboratory, Richland, WA, December 1975.
12. Criticality Safety Procedures, WHAN-M-6, Hanford Engineering Development Laboratory, Richland, WA, January 1979.
13. R. L. Kathren, L. F. Kocher, and G. W. R. Endres, Thermoluminescence Personnel Dosimetry at Hanford, BNWL-SA-2793, Pacific Northwest Laboratory, Richland, WA, April 1970.
14. L. F. Kocher, et al., The Hanford Thermoluminescent Multipurpose Dosimeter, BNWL-SA-3955, Pacific Northwest Laboratory, Richland, WA, May 1971.
15. Norman Engineering Final Report, FSF-8958, Norman Engineering Co., Los Angeles, CA, March 1978.

A P P E N D I X A .

NEUTRON SOURCE IN FUEL REGION

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GAMMA SOURCE IN FUEL REGION

<u>HEDL Group No.</u>	<u>E (Mev)</u>	<u>Photons/Sec/Ass'y</u>
13	2.30	2.1 + 12*
12	2.10	1.5 + 13
9	1.475	1.3 + 13
8	1.225	1.2 + 12
7	1.0	3.6 + 13
6	0.825	2.0 + 15
5	0.65	9.9 + 14
4	0.475	7.3 + 14
2	0.25	1.7 + 13
1	0.15	3.1 + 14
TOTAL		4.1 + 15

INTERPOLATED VALUES FOR 39-GROUP (18 GAMMA GROUPS)

SNAP SHIELD LIBRARY

<u>Gamma Group No.</u>	<u>E_{max} (Mev)</u>	<u>Photons/Sec/Ass'y</u>
1	10	0
2	8	0
3	7	0
4	6	0
5	5	0
6	4	0
7	3.5	0
8	3	0
9	2.5	1.7 + 13
10	2	0
11	1.6	2.5 + 13
12	1.2	3.6 + 13
13	0.9	3.0 + 15
14	0.6	7.3 + 14
15	0.4	1.7 + 13
16	0.21	3.1 + 14
17	0.12	0
18	0.07	0
TOTAL		4.1 + 15

*read 2.1×10^{12}

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CHAPTER 13
CONDUCT OF OPERATIONS

13.0 CONDUCT OF OPERATIONS

The Fuel Storage Facility (FSF) will be operated and maintained by the HEDL FFTF crews. The FSF is normally unmanned and will be included in one of the FFTF ex-containment watchtours. All FSF facility and process equipment alarms are displayed in the FSF control room and are group retransmitted to the FFTF control room. When an FSF alarm is received in the FFTF control room, an operator will be dispatched to the FSF control room to take corrective action as necessary.

Fuel handling operations in the FSF will be conducted by the FFTF fuel handling crew trained in the operation of the BLTC, BLTC Transporter Floor Valve, Plug Handling Fixture, and Storage Vessel Turntable.

Irradiated FFTF fuel will normally be transferred to and from the FSF during FFTF power operation, thus minimizing conflicts during FFTF refueling and maintenance periods. An accurate current inventory of the FSF vessel will be maintained by the Refueling Manager similar to the FFTF in-containment fuel inventory control.

The FSF is contiguous to the FFTF, and all emergency signals are transmitted between the two buildings to allow personnel present to take appropriate action.

For an in-depth discussion of the HEDL organization, personnel training, site emergency plans, procedures, and records for the FFTF complex, see Chapter 13 of the FFTF FSAR.

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CHAPTER 14
INITIAL TESTS AND OPERATIONS

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14.0 INITIAL TESTS AND OPERATIONS

This chapter describes the scope of tests and operations to be performed on the Fuel Storage Facility (FSF) systems and components from the point of turnover from the construction contractor to the final acceptance run with sodium ready to accept FFTF spent fuel. Also included is a brief discussion of construction testing with emphasis on tests required to meet the ASME Code (Reference 1). The extensive testing program for FSF ensures that the equipment and systems perform in accordance with design criteria and that initial liquid metal heatup is accomplished in a safe and efficient manner. During this initial test and operation period, systems and/or components are tested and evaluated according to written, approved test procedures. The procedures include the test purpose and requirements, prerequisites, initial plant conditions, safety precautions and limitations, detailed test methods, and identification of the data required. Test results are analyzed to verify that each system and/or component performs within acceptance limits. As the systems and/or components are tested, the applicable plant operating procedures are implemented to the extent practicable and changed as necessary until the final operating procedure implemented is verified to be a safe, comprehensive and workable procedure for use after acceptable design performance is verified.

The FSF testing and operation program is under the direct control of the Hanford Engineering Development Laboratory (HEDL), including preparation of test procedures and performance of tests. The sodium system design contractor, Atomics International (AI), planned and prepared test specifications which are used for developing written test procedures. Test results are evaluated by the responsible design organization, as required.

14.1 TEST PROGRAM

14.1.1 Construction Testing

Numerous tests must be completed prior to the start of acceptance testing. These are categorized as Construction Tests which are conducted by the construction contractor (J. A. Jones). This testing is not expected to require extended operation of equipment nor performance of hazardous operation, including the handling of sodium. Formal acceptance of test results is required as one of the prerequisites for turnover of the system from the construction contractor. Contractor tests include:

- system hydrostatic, pneumatic and leak tests, dependent on the type system (hydrostatic tests are not conducted on systems handling liquid metal)
- wiring continuity checks
- megger tests on wiring and selected electrical equipment
- initial coupling alignments and bumping of motors
- checking control system component functions to verify that wiring is per design

The pressure and leak testing required to establish the basic integrity of the various liquid metal systems and to provide for their certification under the ASME Code are conducted during Construction Testing.

Each system that handles inert gas is pneumatically tested per the applicable ASME code requirements. Inspection for leaks is performed using soap bubble and/or helium leak-detection techniques.

Following construction testing, systems are turned over to HEDL for calibration, run-in, and testing during the Acceptance Test Program.

14.1.2 Acceptance Testing

Acceptance tests for FSF are those events required to be performed by FFTF Operations, with FFTF Test Engineering (FTE) technical direction, subsequent to turnover to confirm design, construction, performance and operating procedures of the FFTF. This testing has been divided into three phases as described below:

Phase 1 -- Preoperational Tests -- These are events required on components or portions of systems to determine that the components are ready to support subsequent system tests. The test includes such items as checking the operation of individual mechanical and electrical components to the extent permissible under existing conditions, adjusting setpoints and limit switches, calibrating instrumentation, checking interlocks, and grooming and aligning control circuits.

Systems and equipment associated with fuel handling are tested to verify that they will perform as expected in a non-sodium environment and that the equipment and systems are ready for subsequent testing with sodium in the system. Included in this testing is an operational check of the fuel handling equipment in air.

Phase 2 -- System Startup Tests (Without Liquid Metal) -- These are events required on systems or portions of systems subsequent to their preoperational tests and prior to sodium and NaK fill. These are tests to demonstrate that the system is satisfactory at that stage and that the plant is ready to support subsequent phases of Plant Acceptance Testing.

Systems associated with sodium handling, storage and processing are tested at ambient temperature and up to fuel handling temperature in a non-sodium environment to prepare the system for Phase 3 sodium fill. Procedures for heatup of Liquid Metal Systems from ambient to fuel handling temperature are checked out, including such items as check out of heater circuits and control systems for preheat.

Phase 3 -- System Operational Tests (With Liquid Metal) -- These are tests required on systems or portions of systems that use sodium and NaK. These tests demonstrate that the overall plant performance is acceptable and ready for fuel storage.

Following sodium and NaK fill, a series of fluid, mechanical and electrical systems tests are conducted to verify the overall operational performance of the plant to the extent possible without decay heat source.

Sodium and NaK purification is performed during sodium circulation testing up to 600°F using cold trapping techniques. Fuel handling equipment is tested to demonstrate satisfactory performance in a sodium environment, including performing a fuel handling operation using a dummy assembly.

14.2 ORGANIZATION RESPONSIBILITIES

During the FFTF Acceptance Test Program, WHC's FFTF Operations group has overall responsibility for the safe and efficient operation of the FFTF plant and its supporting facilities. It is directly responsible for compliance with all safety requirements as defined by WHC management and DOE. All test operations are conducted in strict compliance with approved procedures by the FFTF operating crews under the control of the Shift Operations Manager. Specific responsibilities, qualifications, and training of FFTF Operations personnel are described in Chapter 13 of Reference 2.

14.2.1 Test Engineer

Test Engineers are assigned by WHC management to be responsible for the technical direction of a test or tests being performed. The Test Engineer may either be assigned to a specific test or to a shift. The Test Engineer is responsible for assuring that all test prerequisites are satisfied and that required plant conditions are established before starting a test. He is responsible for the following:

- briefing fellow test engineers and operators
- coordinating dry runs of test procedures when required
- providing concurrence for FFTF Operations to initiate the major steps of a detailed test procedure
- assuring that the test is conducted in accordance with the test procedure
- reviewing and assembling test data in a data report which identifies the data which is satisfactory and those which are out of specification
- initiation of test exception and responsibility for their close-out, if required

14.2.2 Personnel Qualifications

Persons assigned by HEDL Management to participate in FSF testing shall meet all the recognized standards of experience and training for personnel providing technical support to a nuclear plant. The basis for assuring that test personnel meet the qualifications is a combination of selective hiring policies, wherein academic background and nuclear experience are prime considerations, and periodic training on appropriate subjects. A test engineer assigned to provide technical direction of a given test shall have demonstrated competence in his field; shall have demonstrated a satisfactory knowledge of the functions and requirements of applicable plant systems and an awareness of plant and radiological safety; and shall understand the content of all applicable Test Instructions.

The qualifications of key test personnel (managerial and technical) will be identified as these positions are filled.

14.3 PROCEDURES

WHC is responsible for coordinating and conducting all FSF acceptance testing activities. Test requirements identified by the sodium system designer (AI) are called Test Specifications (see References 3 through 7). Test Procedures are prepared by HEDL and reviewed by the designer. As the test is conducted and deviations from the expected results are identified, a Test Exception is prepared. Upon completion of the specific test, a Data Report is prepared which is reviewed by the designer. Finally an Evaluation Report summarizes the data, and analyses of the system performance are prepared. For a more detailed discussion of this procedure see Chapter 14.1.2 of Reference 2.

14.4 TEST SEQUENCE

Construction testing is performed under the direction of J. A. Jones, the FSF Construction Manager. This includes flushing and cleaning of all fluid lines (except liquid metal), testing and code stamping to meet ASME Code requirements, and electrical continuity and ground checks. In addition, all building functions are verified operational prior to turnover to HEDL for plant acceptance testing. This includes facility fire protection system, heating and ventilating system balancing, water system, all lighting (for operation and required intensity), rollup doors, and building crane. The battery chargers, batteries and inverters are tested for correct operation and load carrying capability.

The first item to be performed upon facility turnover for acceptance as a part of Phase 1 is calibration of all the electrical equipment to assure that all measurements are correct and that operating limits are not exceeded.

These tests are performed in air at ambient temperature to provide maximum hands-on operation with minimum risk to personnel. The Phase 1 tests in addition to calibration include:

- storage vessel seal leakage
- storage vault and cold trap vault leakage
- storage vessel turntable drive, position and index
- storage vessel, sodium in-line and trace heater controls
- fuel handling demonstration
- radiation and oxygen monitor operation verification
- storage vessel and cold trap vault cooling system checkout.

Phase II of the acceptance test inerts the storage vessel and NaK piping system. These tests are prerequisite to adding liquid metal to the facility. The tests are:

- filling the argon headers with argon, and operating the system
- evacuation and backfilling the storage vessel, sodium piping and NaK piping
- balancing the H&V system
- operating the trace heating system

Phase III preheats and adds liquid metal to the system. These tests include:

- filling NaK system at ambient temperature
- preheating storage vessel and sodium piping
- fuel handling demonstration at temperature
- filling storage vessel with sodium and starting circulation

- heating storage vessel to 600°F and purifying sodium and NaK
- completing NDHX performance tests
- stabilizing storage vessel at 400°F

After satisfactory completion and evaluation of the proceeding tests, the facility is ready to receive fuel for storage.

14.5 REFERENCES

1. ASME Boiler and Pressure Vessel Code, The American Society of Mechanical Engineers, New York, NY, July 1980.
2. Fast Flux Test Facility Final Safety Analysis Report, HEDL-TI-75001, Hanford Engineering Development Laboratory, Richland, WA, December 1975.
3. Acceptance Requirements for FSF Process Systems, N106ACR210001, Atomics International, Canoga Park, CA.
4. Acceptance Requirements for FSF Fuel Handling System, N106ACR210002, Atomics International, Canoga Park, CA.
5. Acceptance Requirements for the FSF Trace and Sodium Heating System, N106ACR210003, Atomics International, Canoga Park, CA.
6. Acceptance Requirements for the FSF Protective System, N106ACR210004, Atomics International, Canoga Park, CA.
7. Acceptance Requirements for the FSF Storage Vessel System, N106ACR220001, Atomics International, Canoga Park, CA.

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15.0 ACCIDENT ANALYSIS

15.1 GENERAL

Other sections of this FSAR have addressed the safety features of the facility design in terms of engineered safeguards, safety codes, and standards, and the high level of inherent safety margin designed into the FSF. This section demonstrates that the consequences of events that fall within the range of credible accidents do not adversely affect public health or safety.

The key criterion which must be met under all credible accidents is that the radiological consequences do not exceed the guidelines imposed by 10CFR100 (Reference 1). This requirement states that whole body doses to members of the public outside of the controlled access site boundary shall not exceed 25 rem to the whole body for any credible accident. When organs other than whole body are exposed, the dose limit ratio of organs has been adopted to provide supplementary data for comparison of FSF accident consequences with recognized regulatory direction. These are 10CFR20 (Reference 2) and ERDAM 0524 (Reference 3) which provide guidance for a wider scope of radiation exposure incidents.

The controlled access boundary for purposes of evaluating these doses is the nearest approach of the Hanford Site boundary, 4.5 miles (7.2 km) distant. Other doses quoted in the text were calculated using a distance of 1.5 miles (2.4 km), which is assumed to be the closest distance to the 400 Area where the FSF is located that any other facility may be built.

To assure that the criteria are met, a wide range of accidents was postulated and analyzed. The accidents presented here represent a cross-section of the casualty events to which the FSF might be subjected. An examination of the accidents selected for this FSAR was made to determine the relative severity of their consequences. Those considered to have the maximum credible consequences are given particularly close scrutiny, and in some cases are used as a basis of comparison for other similar but less severe accidents.

Many of the postulated accidents can be shown to have consequences that are less severe than the consequences of some other event analyzed in this report. For such events, it is considered sufficient proof of compliance with the safety criteria if it is shown that:

- 1) The consequences of the event do not exceed the consequences of the more severe event.
- 2) The consequences of the more severe event comply with the safety criteria.

The undesirable consequences of the postulated accidents ultimately resolve into four basic consequence classes. These are release of radioactive material, release of chemically toxic material (such as fire aerosols), radiation exposure and/or release of radioactive material from a criticality, and/or radiation exposure associated with inadequate shielding. Table 15.1-1 presents the major conclusions of the accident analysis. Each accident type that is examined in this FSAR is listed along with the section in which it appears. The radiological consequences of each event are listed for exposure offsite, onsite,* and inside the FSF building.

The information in the table demonstrates that the FSF meets the requirements of 10CFR100 by large margins in every case. In addition, the offsite doses for all of the events fall within the acceptable normal operating limits of ERDAM 0524 and 10CFR20 if annual dose averaging techniques are used as allowed by these documents.

Onsite doses were (predictably) calculated to be greater than those encountered offsite, but even these were found to be small. The expected doses for each accident fall within the annual dose limit of 10CFR20 and ERDAM 0524. Doses within the FSF were found to be larger than the onsite

* Onsite doses are those that might occur to an unprotected individual outside and in the immediate vicinity of the FSF.

TABLE 15.1-1

ACCIDENT SUMMARY TABLE
(doses in mrem)

Accident	Section	Off Site Dose			On Site Dose			IN FSF			Remarks ()
		Whole Body	Skin	Lung	Whole Body	Skin	Lung	Whole Body	Skin	Lung	
Cover Gas Accidents	15.3										
A. Into FSF Bldg.		<0.1	<0.1	<0.1	0.1	1.5	1.3	0.6(1) 10.0(2)	42.0(1) 680.0(2)	40.0(1) 670.0(2)	(1) Assumes evacuation. (2) Assumes continuous manning.
B. Through FSF Stack		<0.1	<0.1	<0.1	<0.1	0.3	0.2	<0.1	0.3	0.2	
Criticality (3)	15.4	--	--	--	--	--	--	--	--	--	(3) Not credible
Fuel Handling Accidents	15.5	<0.1	<1.0	<1.0	0.2	16.0	14.0	0.2	16.0	14.0	
Loss of Heat Removal Capacity (4)	15.6	--	--	--	--	--	--	--	--	--	(4) No release of radioactivity or increased radiation fields by the FSF.
Liquid Metal Fires	15.7	<1.0	<1.0	<1.0	<8.0	<16.0	<4.0	400	800	200	(5) Assumes continuous manning in the plume for 1 hour.
Loss of Shielding Accidents	15.8	<0.1	<0.1	<0.1	<0.1	<0.1	<0.1	62.0	62.0	0	(6) Assumes 10% of time manned over vessel head for 1 year.
Natural Disasters (7)	15.9	--	--	--	--	--	--	--	--	--	(7) No release of radioactivity or increased radiation fields by the FSF.
Other Accidents (8)	15.10	--	--	--	--	--	--	--	--	--	(8) No release of radioactivity or increased radiation fields by the FSF.

doses for several accident scenarios. However, when credit is taken for the radiation alarm system and reasonable evacuation times, doses to personnel within the FSF fall within the acceptable annual limits for normal operation.

A description of the analysis performed for each event in the summary table can be found in the following portions of this section. Further details relating to the specific analyses are in the appendices to this section and the referenced documents.

15.2 ACCIDENT GROUPS

The accidents addressed in this portion of the FSAR are divided into several major categories for convenience. These groups are:

- cover gas release accidents
- fuel handling accidents
- loss of decay heat removal accidents
- sodium and NaK fires
- inadequate shielding accident
- natural disasters
- other accidents

Each of these are discussed in subsequent sections.

15.3 COVER GAS RELEASE ACCIDENTS

The cover gas release accidents encompass those events which cause a breach of the cover gas pressure boundary and allow an abnormal release of argon cover gas and (possibly) fission gas contamination. Accidents considered in this group include releases from:

- primary vessel, support ring, and closure head
- seals and associated buffer gas supply
- argon piping downstream of the cover gas inlet check valve

- holdup tank
- pressure regulation and relief system valves and components

Two distinct modes of release are possible within the group. First, there are the events which allow a leakage of cover gas into the pressure relief or effluent gas lines. These events vent any contamination to the exterior of the building normally through the stacks. FSF operating personnel are not exposed to doses greater than those onsite and exterior to the FSF building. Second, there are events which allow contaminated cover gas to leak into the occupied areas of the building or to the below-grade vaults. In those cases, the dose rates inside the FSF building may be substantially higher than those exterior to the building and evacuation may be necessary.

The requirement of 10CFR72 (proposed) or 10CFR100 (site boundary whole body dose for a given accident not to exceed 25 rem) is the basic acceptance upper limit applied in judging accidents of this group. Limits for other organs were taken to be the same ratio as those listed in ERDAM 0524.

15.3.1 Sources of Radioactivity and Radiation Involved in Cover Gas Release

Two cover gas inventories of radioactivity were considered in the analyses. The first was the design basis activity. The second was a reasonably conservative estimate of the expected activity.

15.3.1.1 Design Activity

For the design basis it was assumed that 0.1% of the full FSF inventory of fuel elements have their cladding ruptured while stored in the FSF and that these ruptures are such that their fission product inventory is released to the sodium coolant. The cover gas would contain 131 curies of krypton-85 with activities of other radioisotopes much less than four orders of magnitude lower, based on the inventories of fuel pins that have undergone 200 days of decay.

The value of 0.1% fuel clad failures was an assumed number. As discussed subsequently, there should be no fuel with cladding failures in the FSF, but for a design basis it was desirable to assume some failure. The value chosen is 10% of that used as a design basis for the FFTF in its FSAR (Reference 4), where the higher temperatures, high fluxes, and more extensive fuel handling are much more conducive to cladding failures.

There is not expected to be any release of fission products from the stored fuel elements because of the following reasons.

- 1) If an assembly were leaking prior to transport to the FSF, the pins would be removed from the assembly and sealed in special cans in the Interim Examination and Maintenance (IEM) prior to delivery to the FSF.
- 2) The environment in the FSF is passive. The only motion is periodic indexing of the turntable for fuel addition, and sodium coolant flow by natural convection.
- 3) The operating temperature is low, $<600^{\circ}\text{F}$, which is much lower than what the cladding experiences in the reactor and the Interim Decay Storage (IDS).

15.3.1.2 Expected Activity

The activity expected in the cover gas would have as its only source any activity that might be carried over from the IDS through the Bottom Loading Transfer Cask (BLTC). A source term was arrived at by assuming that the activity in the IDS cover gas was at 10% of its design value, that this activity equilibrated in the relative volume of the BLTC, and then that the BLTC activity was introduced into the FSF cover gas. The results of these assumptions are expected to be conservative because little if any flow from the IDS to the BLTC should occur because an argon purge from the BLTC to the IDS is made during fuel transfer. Based on these assumptions,

an equilibrium concentration of activity in the FSF was calculated, which was 0.27 Ci Kr-85, 0.029 Ci Xe-131 m, 0.001 Ci Xe-133 m, and 0.90 Ci Xe-133.

15.3.2 Mechanisms of Cover Gas Release

15.3.2.1 Removal of a Floor Valve Without a Transfer Port Plug in Place

This accident involves the removal of the floor valve from a position over an open fuel transfer port. This is basically a breach of the cover gas pressure boundary.

There are three ways in which the floor valve may be accidentally removed from its position over a transfer port. These are earthquake, removal by use of the building crane, and collision with the BLTC.

The floor valve is prevented from moving in an earthquake by a retaining lip on the floor valve adapter (FVA). The Design Basis Earthquake (DBE) vertical acceleration at the closure head of 0.7 g does not exceed the 1 g necessary for the floor valve to "jump" over the lip. The DBE horizontal accelerations of 3 g are accommodated in the design of the floor valve adapter lip and in the design of the rest of the FVA components, which transmit this horizontal loading to the closure head. The 30-mil clearance provided by the lip does not allow enough movement to damage the seals between the floor valve and the FVA. A floor valve console support pad beneath the floor valve transmits loads on the console directly to the striker plate, preventing any tilt of the floor valve. Because of these design features which enable the floor valve to withstand the DBE without breaching the cover gas seals and because earthquakes in excess of the DBE are not credible, a breach of the cover gas boundary due to earthquake motion of the floor valve is not credible.

The floor valve may be removed by use of the building crane. Two ways of doing this have been identified. Both require that the operator make

multiple errors in procedure in order to lift the floor valve while the transfer port plug is not in place.

In the first case, the bare floor valve might be lifted off of the FVA. In order to do this while the transfer port plug is out, the operator will have to ignore the following procedural steps:

- 1) Check to see if a transfer port plug is present in the plug handling fixture. (If it is, then place the plug in the transfer port before continuing.)
- 2) Check the pressure of the transfer port plug seal buffer space to verify that the plug is properly seated.
- 3) Purge the space between the floor valve and the transfer port plug into the holdup tank and bring the pressure in this space to ambient pressure. (It is not possible to bring this space to ambient pressure if the port plug is not in place without causing a process alarm on low pressure in the storage vessel. If such an alarm occurs, the operator must acknowledge the alarm and clear the cause before proceeding.)

In the second case, the floor valve might be lifted if the operator does not unbolt the plug handling fixture (PHF) from the floor valve before attempting to lift the PHF. To lift the floor valve with the PHF, the operator will have to ignore the following procedural steps:

- 1) Set the crane hoist load limiter to 10,000 pounds (4,536 kg). (This limit permits the crane to lift the PHF but will stop the crane and cause an alarm if the floor valve is attached.)
- 2) Unbolt the PHF from the floor valve and inspect the PHF to verify that there are no obstacles to lifting the PHF free from the floor valve.

The precautions written into the operating procedures for the FSF will reduce the likelihood of an accident in which the floor valve is removed without the transfer port plug in place to an unlikely event. This accident is treated as a credible occurrence.

Collision with the BLTC may cause the floor valve to be moved from its position over the transfer port. The positive floor valve positioning provided by the floor valve adapter lip, the low speed of the BLTC, and the existence of obstacle sensors on the BLTC dolly make this event unlikely, but credible.

15.3.2.2 Opening of a Bare Floor Valve Without a Transfer Port Plug in Place

This accident involves the opening of a floor valve without the plug handling fixture or BLTC in place and without a transfer port plug in place. This is basically a breach of the cover gas pressure boundary.

These are two ways in which the floor valve can be inadvertently opened. These are motor-driven and manual operation.

The floor valve is prevented from motorized operation when it is not mated to either the BLTC or the plug handling fixture. The valve motor circuit must be connected to the 480-V plug on the mating machine. In order to open the floor valve under motorized operation, the operator would have to purposely bypass this physical interlock against procedural warnings to the contrary.

The floor valve is equipped with a provision for manual operation. This consists of a hand crank which mates with a shaft protruding from the valve body. The crank is kept under lock and key with specific precautions regarding its use. For an accident of this type to occur, not

only must the normal precautions regarding operation of the floor valve be ignored, but also the added precautions surrounding the procurement and use of the hand crank must be ignored.

This event is considerably less likely than the floor valve lift accident, due to the presence of interlocks and administratively controlled, limited access to special tools.

15.3.2.3 Overpressure Relief of Storage Vessel Cover Gas

Cover gas pressurization in excess of 6 lb/in.² gage (40 kPa) in the storage vessel may cause a release of contaminated argon via the pressure relief system. This constitutes a breach of the cover gas pressure boundary.

Overpressurization of the storage vessel may be caused by exposure of the cover gas to a high pressure source of large capacity. The storage vessel cover gas may be connected to two distinct pressure sources. These are the FSF argon supply and the BLTC argon supply.

The FSF argon supply source is comprised of a six-bottle supply, manifolded for a maximum of three pressurized argon gas bottles at a time with a combined capacity of 900 std ft³ (27 m³), and a maximum pressure of 2400 lb/in.² gage (17 MPa). This represents a capability to pressurize the cover gas to ~8.2 lb/in.² gage (56 kPa). Exposure to this source would not necessarily cause release via the relief system, but release will be assumed for the analysis of this accident. There exist five major pathways through which the cover gas may be exposed to the FSF argon supply. These are:

- 1) Cover gas makeup line
- 2) Buffer seal space supply line
- 3) BLTC purge supply line

- 4) NaK loop No. 1 cover gas supply
- 5) NaK loop No. 2 cover gas supply

The cover gas is protected from direct exposure to the pressure source along each of these paths. This is accomplished by a combination of pressure regulators and/or block valves and/or pressure boundary seals in the case of the first three paths, and by a combination of a block valve and a passive ASME Section 3, Class 3, pressure boundary in the case of paths 4 and 5 (NaK coolers). An analysis of the failure probabilities of the components which protect the cover gas from exposure to this pressure source was made using failure data in References 5 and 6. The event frequency, where an event was defined to be an exposure of the FSF cover gas to the FSF argon supply at a delivery pressure greater than 6 lb/in.² gage (42 kPa), was calculated at $\sim 1 \times 10^{-4}$ events/year. The overpressure relief event is, therefore, considered to be credible, but extremely unlikely, and the safety consequences are evaluated.

The BLTC pressure source, which also represents a possible source of high pressure exposure, consists of a 550 std ft³ (15.6 m³) supply at 2500 lb/in.² (17 MPa). It does not comprise a cover gas overpressure relief hazard because its capacity is insufficient to cause pressurization greater than 5 lb/in.² (35 kPa), which is below the relief valve setting. Thus, it is not considered a credible initiator for this event.

15.3.2.4 Storage Vessel Overpressure with Relief System Failure

This accident involves the overpressure of the storage vessel similar to that treated in Section 15.3.2.3, with the added accident of pressure relief system failure.

As was discussed in Chapter 4, the overpressure of the storage vessel can occur in a variety of ways; however, such an event is considered unlikely but credible at 1×10^{-4} events/year. Such an event could be considered

a "demand" on the pressure relief system. This relief system consists of two parallel pressure relief valves. The probability of failure in the system is conservatively calculated from Reference 5. data to be 1×10^{-7} /demand. This indicates tht the overpressure accident with failure of pressure relief would happen less than 1×10^{-11} /year. The event is considered to be noncredible, and the consequences of this event are not analyzed.

15.3.2.5 Leak or Rupture of Argon Holdup Tank

This accident involves the failure of the holdup tank to contain the contaminated effluent from the storage vessel and fuel handling operations. This accident may be handled as a special case of breach of the cover gas pressure boundary.

Possible initiators for this accident were evaluated in four categories. These were mechanistically-induced failures, thermally-induced failures, overpressure-induced failures, and random failures.

The holdup tank is not specifically required to withstand the DBE-induced forces. It is protected against missiles and extreme weather (design basis tornado) by the concrete vault in which it is located. No other credible mechanical inputs of significance could be identified. Consequences of failure are consistent with those in 15.3.3.1.

The holdup tank is designed to withstand the extremes of the thermal environment in which it is expected to operate.

The holdup tank is protected from overpressure by one pressure relief valve. For the holdup tank to sustain a pressure in excess of the 15 lb/in² gage (100 kPa) for which it was designed, there would have to be at least one pressure regulator or valve failure in the argon supply system and failure of the holdup tank relief valve. Scoping calculations showed that the probability of such an occurrence was approximately 10^{-10} events/year. This failure mode was considered to be not

credible. Proper relief of the pressurization via the overpressure relief valve may, on the other hand, be a credible initiator of release. The consequences of such a failure mode are bracketed by the analysis done for overpressure relief of the entire storage vessel (15.2.1.3) and are not separately addressed in this report.

Random leakage or rupture of holdup tank welds was found to be a credible initiator for this event at 1.3×10^{-3} events/year. The accident was therefore considered to be within the bounds of credibility and was analyzed for its safety consequences. It is assessed as an unlikely event.

15.3.2.6 Failure of Storage Vessel Relief Rupture Disc (Low-Pressure Leak)

This accident involves a leakage failure of one or both of the rupture discs used in the storage vessel pressure relief system. This is a breach of the cover gas pressure boundary.

Although the storage vessel pressure relief rupture disc will be well-protected from mechanically and thermally induced failures (overpressure is treated elsewhere in 15.3.2.3), there is some small probability that a flaw in the rupture discs will cause some premature failure. Probability data on such an event was not found in the literature; however, the accident was assumed to be credible and was evaluated for safety consequences. This accident was evaluated against the criteria for an unlikely event. Failure during a DBE is consistent with 15.3.3.1.

15.3.2.7 Leak in Double-Buffered Pressure Boundary Seals

This accident involves the failure of both seals in a buffered pair. There are 18 separate buffered seal pairs in the FSF storage vessel in addition to the floor valve and BLTC seals, which occasionally form a part of the cover gas boundary where this is possible. This accident is basically a breach of the cover gas pressure boundary.

Double failure and leakage of buffered pressure boundary seals may occur in several ways. There may be a common mode failure through overpressure or thermal degradation, or there may be independent random failures.

The overpressure failure mode requires both a pressure regulator failure and a pressure relief valve failure in addition to an operator error in order to subject the seals to a pressure source above their design pressure. The failure rate of the paired pressure regulator/pressure relief valve system alone is 0.8×10^{-6} events/year.

Thermal degradation of buffered elastomer seals can occur due to some of the fuel handling accidents. Discussion of these is contained in Section 15.5.4.

Independent random failure of both seals in an elastomer seal pair is calculated to occur 6×10^{-5} times per year if credit is taken for routine surveillance and replacement as necessary within one month of failure. Failure data on buffered metal O-ring seals is scarce, but they are certainly more reliable than elastomers. The accident is, however, considered to be credible based on the failure rate for elastomer seals (frequency $> 1 \times 10^{-6}$ occurrences/year). This accident was assessed as an extremely unlikely event.

15.3.2.8 Leak in the Argon Cover Gas Lines/Valves/Components

This accident involves leaks in the components forming the cover gas pressure boundary other than seals. This is basically a breach of the cover gas boundary.

The components of the cover gas pressure boundary may fail by overpressure, leakage at welds, or component or pipe rupture during a DBE.

Overpressure failures have been addressed in Sections 15.3.2.3 and 15.3.2.4. Overpressure to the point of failure has been shown to have a

very low probability ($<1 \times 10^{-6}$ events/year) for the FSF systems analyzed. This occurs because the components are designed so that there is a low probability they will see a high-pressure source, and there is also a low probability that their relief systems will fail if they do see a high-pressure source.

Leakage at weld and component rupture was estimated at $\sim 3 \times 10^{-3}$ /year based on a weld and valve count and the additive failure probabilities of all these components. Such an accident is credible and was analyzed for safety consequences. This is an unlikely event.

15.3.3 Method of Analysis

For each of the two source terms, design activity and expected activity, in the cover gas (Section 15.3.1) the doses (1) in the FSF, (2) in the 400 area, (3) at a distance of 1.5 miles (2.4 km), and (4) at a distance of 4.5 miles were calculated using the COMRADEX digital computer program which was qualified for use for Reference 6. The distance of 4.5 miles (7.2 km) is that to the site boundary. Calculations were made at 1.5 miles (2.4 km) which was considered to be the closest distance that facilities may be located to those in the 400 area. A conservative meteorology was used for a 1-m/s wind speed under Pasquel condition F with no crosswind.

15.3.3.1 Cover Gas Release in the FSF Building

For those accidents that release cover gas into the FSF building, such as leaks through seals, the initial rate of release would be controlled by the cross-sectional area of the leak. Since this rate cannot be postulated exactly, the conservative assumption was used that this area is very large; hence, there would be an initial instantaneous release based on differential pressure across the interface.

Under normal operating conditions, the maximum pressure differential between the cover gas and the process area is 12 iwg (3 kPa). Pressure regulators in the storage vessel cover gas argon inlet line would attempt to make up any drop in pressure below 8 iwg (2 kPa). Thus there would be a continuous purge of argon supplied to the cover gas causing it to exhaust through the leak at the delivery capacity (10 std ft³/min, 4.7×10^{-3} m³/sec) of the argon supply pressure regulator. This purge would continue until the argon supply was exhausted (1000 std ft³, 28.3 m³) or until the purge was manually terminated. No credit is taken for operator intervention in this case.

With a large leak, the storage vessel would quickly depressurize from 8-12 iwg (2-3 kPa) to ambient and then slowly purge at 10 std ft³/min (4.7×10^{-3} m³/sec). The depressurization portion of the release would consist of argon gas at the initial contamination levels. The purged portion of the release would consist of argon with successively diluted concentrations of contaminant. In the absence of any well defined mixing in the cover gas, a realistic but still conservative model can be constructed by saying that the release consists of an instantaneous decompression cloud followed by a 10 std ft³/min (4.7×10^{-3} m³/sec) purge at the original argon cover gas average contamination terminating when the argon supply is exhausted. This would be equivalent to an instantaneous release of 3% of the source term followed by a continuous release of 0.55% of the source term per minute for a period of 100 minutes; these releases were used in the calculations.

The principal pathway for the release of activity out of the building would be by the building ventilation system. This ventilation is a 16,000 std ft³/min (11.8 m³/sec) system using a 98% recirculation for heating and no recirculation for non-heating ventilation. Hence, the air rejection rate is 500 std ft³/min (0.24 m³/sec) when heating and 16,000 std ft³/min (11.8 m³/sec) when not. Activity concentration calculations were made for both rates and are shown in Figure 15.3-1.

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FIGURE 15.3-1. Concentration of Activity in the FSF Building Subsequent to a Cover Gas Release into the FSF Building.

15.3.3.2 Cover Gas Release Through the FSF Stacks

The cover gas pressure relief system vents through the NDHX stacks. For an overpressure relief event, the depressurization phase would last from 15 lb/in.² gage (105 kPa) (the relief valve setpoint) down to atmospheric pressure. The continued depressurization after the system drops below 15 lb/in.² gage (105 kPa) is due to a small bypass vent line installed between the rupture disc and the relief valve. Once the rupture disc has been breached, the depressurization continues even though the relief valve resets. The depressurization phase would release 51% of the cover gas and its associated contamination as opposed to the 3% cited in 15.3.3.1. Calculationally, this was assumed to occur instantaneously. When the cover gas pressure drops below 8 iwg (2.0×10^3 Pa), the cover gas argon supply pressure regulators will initiate the purged portions of the release by supplying 10 std ft³/min (4.7×10^{-3} m³/sec) of argon in an attempt to repressurize the cover gas. This purge phase would release 0.26% of the cover gas volume (and source term) per minute until the remaining argon was exhausted. This exhaustion of the balance of the argon would happen almost immediately because nearly all of the gas would be required to cause pressurization. The total release would involve no more than 60% of the source term.

There is no release to the building interior due to this accident. The containment effluent is vented through the relief system to the NDHX stacks after the cover gas reaches 15 lb/in.² gage (105 kPa) and never enters the process area. Doses inside the FSF are, therefore, not calculated and are assumed to be similar to the onsite dose calculated for this accident.

15.3.4 Results of Analyses

The results of the analyses are tabulated in Table 15.3-1 assuming the design level of activity in the cover gas, and in Table 15.3-2 assuming the expected level. The only location at which non-trivial doses may be

TABLE 15.3-1
TOTAL ENVIRONMENTAL DOSES FROM A COVER GAS RELEASE ACCIDENT
WITH DESIGN ACTIVITY AS THE SOURCE

<u>Location</u>	<u>DOSES</u>		
	<u>Whole Body (mrem)</u>	<u>Skin (mrem)</u>	<u>Lung (mrem)</u>
A. Release into the FSF bldg.			
*On site at 10 m	0.1	1.5	1.3
At 1.5 miles (2.4 km)	0.1	0.1	0.1
At 4.5 miles (7.2 km) (site boundary)	<0.1	<0.1	<0.1
B. Release through the FSF stack			
*On site at 10 m	<0.1	0.3	0.2
At 1.5 miles (2.4 km)	<0.1	0.2	0.1
At 4.5 miles (7.2 km) (site boundary)	<0.1	<0.1	<0.1

*The uncertainties in the calculational model are greater at the 10m distance than at the larger distances.

TABLE 15.3-2
TOTAL ENVIRONMENTAL DOSES FROM A COVER GAS RELEASE ACCIDENT
WITH EXPECTED ACTIVITY AS THE SOURCE

<u>Location</u>	<u>DOSES</u>		
	<u>Whole Body (mrem)</u>	<u>Skin (mrem)</u>	<u>Lung (mrem)</u>
A. Release into the FSF bldg.			
*On site at 10 m	<0.1	<0.1	<0.1
At 1.5 miles (2.4 km)	<0.1	<0.1	<0.1
At 4.5 miles (7.2 km) (site boundary)	<0.1	<0.1	<0.1
B. Release through the FSF stack			
*On site at 10 m	<0.1	<0.1	<0.1
At 1.5 miles (2.4 km)	<0.1	<0.1	<0.1
At 4.5 miles (7.2 km) (site boundary)	<0.1	<0.1	<0.1

*The uncertainties in the calculational model are greater at the 10m distance than at the larger distances.

received is within the FSF building itself when there is a cover gas release into the building. The concentration of activity as a function of time in the building is given in Figure 15.3-1 for both design and expected activity levels in the cover gas and for both ventilation rates (98% recycle for heating and no recycle for non-heating) using the leak rates described in Section 15.3.3.1. It is difficult to project a dose for individuals because (1) the facility is unmanned most of the time, (2) the actual leak rate in the accident is uncertain, (3) because of (2) it is uncertain whether the high radiation alarm would sound, and (4) the ventilation rate is uncertain. Assuming the worst case being the maximum leak rates into the building described previously, the minimum ventilation rate of 98% recycled air and the design level of activity, the high radiation alarm would sound on all three high bay detectors and on the detector in the ventilation duct; assuming evacuation occurred in approximately one minute a whole body dose of 0.6 mrem, a skin dose of 42 mrem, and a lung dose of 40 mrem would be obtained. If the high ventilation were either on or turned on, as required by operating instructions in the event of an alarm of the ventilation detector, and a person remained in the building continuously throughout the event, the doses would be approximately 10 mrem whole body, 680 mrem skin, and 670 mrem lung. If the cover gas activity was at the expected level, the dose estimates above would be a factor of 110 lower.

On the basis of these results, no individual in the FSF, in the 400 area, or off-site, would exceed any quarterly operational limit much less any accident limit.

15.4 CRITICALITY

The design and operation of the FSF fuel storage system is such to preclude criticality using the guidance of DOE Manual Chapter 0530 (Reference 7) and applicable sections of Chapter 0540 (Reference 8).

15.4.1 Identification of Causes

The FSF storage vessel, turntable, and storage tubes are designed and fabricated to assure that subcritical multiplication is maintained under all credible conditions. The only methods by which criticality could be obtained in the FSF are by (1) changing the fuel composition, such as increasing the plutonium composition, (2) changing the neutron spectra or cross-sections, such as adding moderation, (3) changing the geometry, such as rearranging the fuel with closer spacing, or (4) changing the neutron distribution, such as removing poison.

The composition/reactivity of the elements to be stored in the FSF is administratively controlled, supported by physical limitations and diagnostic measurements. Storage of fuel can only be physically accomplished, by use of the bottom loading transfer cask (BLTC). The BLTC can physically accept only one fuel element or can at a time, and only from the IDS, the IEM, or the cask loading station. This limits the fuel that can be loaded into the FSF to FFTF fuel or test specimens.* Hence, the reactivity is limited by the administrative and physical controls of the FFTF as well as administrative controls for FSF. These controls as well as the additional reactivity that could be accommodated without the system going critical make criticality non-credible. Addition of moderator material to the fuel region is also not possible because of the following features:

- 1) Facility location makes general flooding noncredible.
- 2) No water is allowed in high bay area, including water lines.
- 3) Curbs are installed between the high bay and water-containing areas.
- 4) Storage vessel is designed with a high-reliability pressure boundary.
- 5) Floor level is above exterior grade to prevent local flood.
- 6) Waterless firefighting instructions are posted and equipment is available.

Geometry considerations were examined to determine the credibility of deformations large enough to cause criticality problems. Poison consideration, included inadvertent removal, migration, impacts or accelerations were studied. Worst credible closure head impact loads were selected based on design basis tornado missiles and maximum crane load drop weights. Analysis of these missiles and drop loads showed that the stored fuel geometry was unaffected. Worst credible storage tube impact loads were selected based on the most energetic fuel handling accident. (Fuel assembly and BLTC grapple drop from full up position into an empty storage tube.) Stress analysis indicates that the storage tube remains undeformed. Worst case accelerations for the entire assembly were selected based on those expected from the DBE. Ground motion in excess of these levels is considered to be noncredible in the Hanford 400 area. Stress analysis of the supporting structure, primary vessel, and turntable components according to the ASME Code (Reference 9) indicates that no change of fuel geometry will result from these earthquake accelerations. A mechanically induced critical accident is, therefore, considered to be not credible. Second, thermal failures leading to geometry degradation were examined. These are discussed in Section 15.6. Heat removal system failures were analyzed to determine whether temperatures in the range of significant creep could result from credible HRS accident sequences. No credible thermal accidents were found that could cause geometry changes other than normal thermal expansion. Thermally induced criticality accidents are, therefore, considered to be noncredible. Third, although the lattice pitch of 9-in. (0.23 m) in the FSF turntable is sufficient to produce a $K_{eff} < 0.95$ for spent FFTF fuel assemblies, a pessimistic loading of all fresh fuel assemblies in the most reactive configuration requires the presence of a nuclear poison to be maintained within the criticality limits. The FSF turntable design calls for an encapsulated B_4C annulus in the central sodium region. This absorber is enclosed in an ASME Code Section III, Class 2 boundary and is

*See Reference 4 for discussion of fuel handling and controls in that facility.

also insoluble in sodium at FSF operating temperatures.* Migration of the B_4C is therefore not credible. Inadvertent removal of the absorber is impossible without turntable disassembly because of the absorber envelope and turntable assembly sequence. Burnup of the B_4C was calculated to be less than 0.0001% in 20 years of FSF operation. A criticality accident due to B_4C misplacement is therefore considered to be not credible.

15.4.2 Analysis of Effects and Consequences

Analyses were performed using both Monte Carlo and diffusion calculations, Reference 10. Monte Carlo check calculations were also performed by HEDL. Initial calculations by Atomics International (AI) were made using loading in all ten rings of non-irradiated FFTF fuel (loading of spaces beyond that required for FFTF cores 1 through 4 was assumed to be all of core 4 outer driver assemblies, which are the most reactive assemblies identified). The HEDL check calculations assumed that the entire loading was made of core 3 and 4 outer assemblies. The results are in Table 15.4-1. It should be noted that the large difference between the HEDL uncertainty compared to the AI uncertainty is due to the larger uncertainty assigned by HEDL in normalizing calculations to experiment (ZPPR-2) because of significant geometry differences.

To provide an increase in the margin of safety, the fifth-from-inside ring's access port (43 storage positions) shall be sealed until a significant loading has occurred in the remaining nine rings, at which time calculations based on the "as loaded" assemblies can be used to provide the basis to modify the technical specifications (Chapter 16) to allow the use of ring 5. Since this loading will undoubtedly be irradiated fuel, the expected eigenvalue will be less than 0.90. The eigenvalues and uncertainties are shown in Table 15.4-2.

* The 2 inch annulus is filled with boron carbide granuals compacted to approximately 64% of theoretical density. The distribution is verified by radiography, the isotopic content by analyses.

TABLE 15.4-1
CALCULATED K_{eff} FOR FILLING OF TEN RINGS
WITH UNIRRADIATED FUEL

<u>Calculation</u>	<u>K_{eff}</u>	<u>Uncertainty at 95% CL</u>	<u>Sum</u>
AI Monte Carlo	0.930	0.019	0.949
AI Diffusion	0.932	0.017	0.949
HEDL Monte Carlo	0.942	0.025	0.967

TABLE 15.4-2
CALCULATED K_{eff} FOR FILLING NINE RINGS WITH
UNIRRADIATED FUEL

<u>Calculation</u>	<u>K_{eff}</u>	<u>Uncertainty at 95% CL</u>	<u>Sum</u>
HEDL Monte Carlo	0.881	0.028	0.909

Although the margins shown in Table 15.4-2 are adequate to meet the requirements of <0.95 for the calculated $K_{eff} + 95\% \text{ CL}$, there is significant additional margin that should be realized. Although no prohibition has been placed on loading unirradiated fuel into the FSF, it would not serve its design function to support FFTF (Chapter 1) if it were used for this purpose. Reference 10 reports an eigenvalue of 0.853 for mixed cores 1 through 4 with average discharge burnup. Therefore, the expected K_{eff} is well below 0.9.

The margin to criticality is large enough to cover any type of accident identified as discussed below:

- 1) Fuel Bowing - Fuel bowing was included in the HEDL Monte Carlo calculations (Table 15.4-1) modeling such that for a cluster of four boxes, the fuel was 2.29 cm (0.9 in.) closer to the common corner. Therefore bowing has been accounted for.
- 2) Inadvertent Use of Ring 5 - Although ring 5 shall be sealed until the Technical Specification is changed based on "as-loaded" calculations, criticality would not be obtained if it were used (Table 15.4-1).
- 3) Temperature Change - HEDL Monte Carlo calculations indicate a change in K_{eff} in changing the temperature from 70°F (21°C) to 400°F (204°C) of -0.013 ± 0.002 due to change in sodium density, and -0.020 ± 0.009 due to doppler broadening. Therefore if the system were allowed to cool to room temperature, the system would remain subcritical.
- 4) Over Loading - A change in plutonium content of 7% w/w in one-fourth of the assemblies would produce a $+0.108$ in K_{eff} according to HEDL Monte Carlo calculations. This would be approximately $+0.08$ in K_{eff} if the plutonium content of one-fourth of the assemblies were increased 5% of the maximum plutonium content (from approximately 30% w/w to 35% w/w), and the system, if it occurred, would remain subcritical. Reactivity and fuel region location in assemblies or canisters is verified prior to transfer to FSF (Chapter 16, Technical Specifications).
- 5) Loss of Absorber - In Reference 10, it was reported that the K_{eff} would be <1 if over half the B_4C absorber were not present.

The net result is that the double contingency criteria is met (Reference 7) and a criticality accident is incredible.

15.5 FUEL HANDLING ACCIDENTS

The fuel handling accidents encompass those events which involve operational errors in use of the fuel handling equipment in the FSF. Accidents* considered in this group include 1) events which may cause a fuel assembly to be dropped, and 2) fuel assembly stalling or sticking

Mechanical and thermal failure of in-situ fuel is rendered noncredible by design features of the facility (15.6 and 15.9.1). The only credible failure modes for fuel cladding are, therefore, connected with the handling of individual fuel assemblies as they are either loaded into or retrieved from the storage vessel.

15.5.1 Fuel Assembly Dropped During Transfer

This accident involves the inadvertent release and/or drop of a fuel assembly during an attempted transfer between the BLTC and the FSF fuel storage vessel. This event may result in the rupture of the fuel assembly cladding and release of fission gases into the cover gas.

15.5.1.1 Identification of Causes

Rupture or a severe break of the BLTC latch or grapple hardware fuel assembly shoulder latch hardware, or both BLTC grapple chains can cause the premature release of a fuel assembly during transfer. Improper grapple latch grasp of the assembly may also result in a fuel assembly drop.

The combined probabilities of these mechanical failure modes are assumed to make this accident a credible though unlikely event.

* Consideration of storing an element that has not had a 200-day decay period subsequent to inpile irradiation is not discussed since the shorter-lived decay products would be encapsulated and a second accident would be required to release them.

15.5.1.2 Consequence of the Accident

The dropped fuel assembly with or without all or part of the grapple attached to it will plunge into the storage vessel sodium, strike the bottom of the fuel assembly storage tube and come to rest as it reaches the 3-in. (0.08-m) thick steel support plate just below the bottom of the storage tube. (Stress analysis of this accident mode is discussed in Reference 11.) Although only the nose (intake region) of the fuel assembly is expected to deform in this accident, fuel pin damage and release of the entire fuel assembly fission gas inventory is assumed to take place for purposes of this analysis. The vessel sodium will provide adequate cooling of a fuel assembly in the dropped position. The heat transfer characteristics of a dropped fuel assembly without sodium circulation were determined; the thermal conductivity alone is sufficient that 1.7 kW of decay heat power will not cause the fuel clad temperature to exceed 800°F (426°C) during normal operation even when the entire assembly decay heat is lumped in the center pin and conservative geometry assumptions are made. The vessel gas pressure boundary will remain intact and no radioactive gas escape will result from this accident.

The release of any fission product gas would be a planned release and as such was presented in Section 11.4 in discussing the gaseous discharge system. Because the source in this accident is only half of the activity used in Section 11.4 maximum credible accident analysis, the doses would be less than those reported in Table 11.4-3.

15.5.2 Attempt to Load Fuel Assembly into an Occupied Position

This accident involves the BLTC lowering a fuel assembly onto the top of another fuel assembly already in the vessel position below the open port.

15.5.2.1 Identification of Causes

Operator error or incorrect storage inventory control may result in loading a fuel assembly into an occupied fuel storage position. These accident

modes result from human error rather than hardware failures. The probability of such a human error is unpredictable. Because of this uncertainty, this analysis makes the assumption that this accident is a credible event. The effects of loading a fuel assembly into an occupied position are evaluated here as an unlikely event.

15.5.2.2 Consequence of the Accident

The BLTC grapple maximum speed while lowering a fuel assembly is 10 ft/min (2.5 m/min). This impact speed will not cause a rupture of the cladding in either fuel assembly because this is the equivalent of dropping the fuel from about 0.005 in. (1.3×10^{-4} m). FFTF fuel assemblies are designed to survive a drop of many times this severity. The fission gases will remain within the fuel assembly and will not be released to the cover gas. The radiological consequences of this accident are therefore insignificant.

15.5.3 Fuel Assembly Dropped During Attempt to Load into Occupied Position

This accident involves the inadvertent drop of a fuel assembly from the BLTC onto the top of another fuel assembly already in position. This event could result in the rupture of fuel pin cladding in both assemblies.

15.5.3.1 Identification of Causes

Dropping a fuel assembly into an occupied position is actually a combination of the previous two accidents. The circumstances cited under Sections 15.5.1.1. and 15.5.2.1 (i.e., Identification of Causes) are both necessary in order to cause this "combined" accident. The causes for the dropped fuel assembly accident and the attempt to insert a fuel assembly into an occupied position accident are both considered to be credible although they are difficult to quantify. Dropping a fuel assembly into an occupied position is also possible and is treated here as an unlikely credible event.

15.5.3.2 Consequence of the Accident

The fuel assembly can free fall a maximum distance of 15 ft (4.6 m) and achieve a maximum velocity of 31 ft/sec (9.5 m/sec) before impact with the fuel assembly in the vessel occupied position. This assumes the fuel assembly was initially in the up position limit within the BLTC with the closure and floor valves open. Although it is unlikely that significant pin damage would occur in either assembly, it is assumed that all 217 pins in both assemblies rupture and release their gaseous fission products to the cover gas system where no uncontrolled release would occur. Since there would be no inadvertent release of radioactivity, the consequences of planned releases are discussed in Section 11.4. Maximum whole body doses (Table 11.4-3) of 16 mrem on site and <1 mrem at the site boundary were calculated.

15.5.4 Fuel Assembly Stuck During Transfer

This accident involves the sticking or jamming of a fuel assembly somewhere along the transfer path from the BLTC to a storage tube in the FSF.

15.5.4.1 Identification of Causes

Jamming of a grapple chain, jamming or functional failure of the grapple gear drive, failure of electric power sources, binding of a warped or bent fuel assembly in the floor valve opening or inadvertent closure of either the floor valve or BLTC gate valve during transfer may cause a fuel assembly to be stuck in a partially lowered position in the opening from the BLTC to the vessel.

These failure modes involve either hardware failure, control failure, or possible human error. Based on the fuel handling experience to date in systems performing similar functions, this accident is assumed to be a credible event.

15.5.4.2 Consequence of the Accident

When a fuel assembly sticks in a partially lowered position, diminished cooling, from lack of forced argon convection, will allow the fuel assembly temperature to rise. Thermal analysis done for a similar situation (immobilization of a fuel assembly in the IDS shield region) indicated that fuel pin failure would not occur due to the expected elevated temperatures of ($\sim 1290^{\circ}\text{F}$, $\sim 699^{\circ}\text{C}$) even with an element liberating 10 kW of heat. However, the presence of a hot fuel assembly in one of the FSF transfer ports may cause thermal degradation of adjacent elastomer seals. Thus, contamination argon cover gas may be released through a breach in the cover gas pressure boundary. The release of cover gas through failure of seals is discussed previously in Section 15.3

If the fuel element were loaded such that the most radioactive portion of the element was in the floor valve area, dose rates as high as 200 mrem/hr at the surface of the valve could be obtained. This dose rate would not cause any significant dose to personnel because 1) the BLTC operator is shielded from this source, and 2) by procedure, other personnel are in < 2 mrem/hr areas during the actual transfer of fuel.

15.6 LOSS OF HEAT REMOVAL CAPABILITY

The loss of heat removal accidents encompass those incidents which disrupt normal dissipation of decay heat from the stored fuel. These accidents occupy a position of importance in the safety study for two reasons. First, the integrity of the storage vessel system and its supporting concrete structure depend upon the maintenance of a proper thermal operating envelope. Second, in-place fuel pin cladding thermal failures must be prevented in order to keep the cover gas contamination levels within acceptable limits.

Decay heat is removed from the FSF fuel storage vessel by one of three principal paths: through each of two independent and redundant NaK heat

transfer loops or by the vault cooling system. Relatively minor amounts of heat are lost through the vessel head and through the concrete walls.

A partial loss of heat removal capability could be caused by any one of three general types of accidents:

- 1) A failure in the heat removal system (HRS). Failures in one or both NaK loops are considered.
- 2) A failure in the vault cooling system (VCS). These failures are considered alone and with a concurrent failure in one HRS loop.
- 3) A loss of coolant failure. This group considers NaK leaks from the loops and considers sodium leaks from the purification loop or from the vessel and/or guard tank.

The ultimate safety criterion for evaluating all of these events is that they do not lead to release of fission products that could exceed the dose limits of 10CFR72 (proposed) or 10CFR100. An accident will not cause the release of fission products from in-place fuel under the following conditions:

- 1) The sodium temperature stays below 1200°F (649°C). The vessel stress to rupture at this temperature is 1500 lb/in.² (11 MPa) for 10,000 hours and the 1% creep is 800 lb/in.² (5.5 MPa) for 10,000 hours, whereas the system dead weight stresses are approximately 400 lb/in.² (2.8 MPa); therefore, the storage vessel integrity as a container for the sodium coolant remains intact. Also, fuel pin clad temperatures will not exceed 1500°F (816°C) if the fuel pins remain in contact with the sodium at the 1200°F (649°C) bulk temperature. Damage to the pin cladding would not occur below the 1500°F (816°C) fuel pin emergency limit and thus radioactive release from in-place fuel would not occur. It is also acceptable if the sodium temperature does not exceed the 1200°F (649°C) limit within the time required to

make an emergency repair. This analysis conservatively assumes that repairs of liquid metals systems leakage will take as long as 15 days in accessible areas. Under these conditions, the maximum sodium temperature will not exceed 800°F (472°C).

- 2) The concrete temperature stays below 400°F (204°C). (See Reference 9, Section III, Division 2, Subsheet CB, Table CB-3430-1, Failure Temperature.) Below this temperature, the concrete will remain intact and continue to provide support for the vessel.
- 3) The sodium level in the vessel remains above elevation 535 ft-7-1/2 in. (163 m). This elevation corresponds to the top of the active portion of the fuel assemblies and to the upper NaK cooler. Therefore, as long as sodium is at this level or above, there will be cooling of the fuel cladding.

15.6.1 Loss of Heat Removal System

15.6.1.1 Identification of Causes

The HRS constitutes the primary path by which decay heat is rejected to the atmosphere. The loops are composed of essentially passive components: piping, manual valves, heat exchangers, etc. During operation, adjustments are made only to the NDHX louvers and dampers (both manually operated). Diagnostic instrumentation provides information on process status and is used by the operator to establish the required manual actions for process control only over a long period of time as heat loads decrease/increase or as the seasons change.

An analysis was performed to determine the failure modes which represent credible initiators of a loss of HRS. (See Appendix A). Random weld failure in heat removal loop pressure boundaries was determined to be the most credible failure mode. Other failure modes, such as inadvertent closing of the NDHX dampers or loss of the NDHX chimney, were found to be less severe

than events which affected the NaK pressure boundary because recovery times were much shorter. The thermal heatup transients associated with these less severe accidents are identical to the more severe events, except that they are terminated earlier and, therefore, do not reach as high a temperature. A discussion of the temporary loss of heat removal loops is presented in Sections 15.9.1, "Earthquake" and 15.9.2, "Tornado." Inadvertent closing of the NDHX dampers is less severe than the DBE and DBT events because of a quicker recovery time. Calculations made to scope the effect of a high-powered fuel assembly (1.7 kW) showed that the clad temperatures in the assembly would remain within acceptable limits (see 15.5.1.2). The calculation of the failure probabilities for a wide range of possible accidents indicated that the worst credible loss of HRS accident is the loss of both heat removal loops at 144 kW. Although 205 kW is the upper bound on any credible decay heat load; this could occur only after an FFTF core dump with space in the storage vessel for exactly 75 fuel assemblies dumped. A heat load of 205 kW would not be accepted into FSF with one heat removal loop already inoperable because the normal maximum acceptable decay power will be administratively controlled to a level where the vault cooling system alone would provide adequate heat removal capability if the remaining loop is lost (via Technical Specifications). This accident exhibits a frequency of 1.44×10^{-5} events/year or a mean time between failures (MTBF) of 69,444 years. This is an extremely unlikely accident. The consequences of this event were analyzed to verify compliance with the safety criteria for HRS accidents (15.6).

15.6.1.2 Consequence Analysis

A thermal analysis (Reference 12) was performed to evaluate the consequences of a loss of both heat removal loops at 144 kW. The analysis takes credit for suspension of fuel handling activity after the initial failure, and operation of the vault cooling system at maximum capacity by increasing the sodium flow through the cold trap and the nitrogen flow in the cold trap cooling shroud. The calculated sodium and concrete thermal transients are shown in Figures 15.6.1 and 15.6.2. The maximum sodium temperature of

FIGURE 15.6-1. Sodium Temperature History After Loss of HRS Accident.

FIGURE 15.6-2. Concrete Temperature Response After Maximum Credible HRS Accident.

910°F (487°C) is reached only after 43 days, and the maximum concrete temperature is reached after two months. Both of these maximum temperatures lie within the criteria set in 15.6. Based on this analysis, it is concluded that no additional in-place fuel pin failures will occur due to this accident. Moreover, the vessel "faulted" temperature limit will not be reached for at least 16 days under these accident conditions, which would be expected to be more than adequate to take remedial action.

15.6.2 Loss of Vault Cooling

This accident involves a failure of FSF components which prevents effective use of the vault cooling system to cool the concrete in the storage vessel vault.

15.6.2.1 Identification of Causes

The Vault Cooling System (VCS) provides cooling of the vault atmospheres. The VCS safety function is to maintain the storage vessel vault concrete at acceptable temperatures (bulk concrete below 150°F (66°C), local hot spots below 200°F (93°C) under normal operating conditions, and below 350°F (177°C) bulk, 650°F (343°C) local under emergency conditions (Reference 13). The vault cooling system normally removes from 20 kW to 25 kW from the storage vessel via the vault atmosphere and between 1 kW and 40 kW from the sodium cold trap. The VCS, therefore, represents a significant, secondary means of decay heat removal.

The VCS components are chosen from commercially available hardware. The components are not designed to withstand the DBE, and the chiller is not protected from the DBT. A failure in these components is capable of disrupting cooling to the below-grade vaults. Such failures are credible though unlikely and the consequences of a loss of vault cooling are, therefore, evaluated.

15.6.2.2 Consequence Analysis

The loss of vault cooling consequence analysis considers the failure of the vault cooling system concurrent with one heat removal loop failed at a decay heat load of 205 kW. This is considered to be the most severe credible scenario in which loss of vault cooling plays a significant part. The 205 kW heat load represents the maximum heat load after FSF has accepted 75 assemblies from an FFTF core dump. The heat removal loop is assumed to fail shortly thereafter. This represents a highly unlikely though still credible set of circumstances and is considered for purposes of conservatism. Such an event would be extremely unlikely.

After the heat removal loop and vault cooling system fail, the sodium and concrete temperatures will begin to rise toward a new equilibrium level. Figure 15.6-3 illustrates the temperature history of the hottest part of the vault concrete during this transient. The vessel sodium comes to equilibrium at 561°F (294°C) which is below the design temperature of 650°F (343°C). The vault concrete equilibrates at 323°F (162°C) (it does not reach 300°F (149°C) until after 15 days).

Figure 15.6-4 shows the equilibrium concrete temperature (no VCS) as a function of bulk sodium temperature while Figure 15.6-5 illustrates the sodium temperature as a function of decay power for one-loop operation of the HRS at maximum thermal capacity. Operation of the NaK cold leg at temperatures lower than 300°F (149°C) will produce lower sodium temperatures and correspondingly lower concrete temperatures.

15.6.2.3 Results of Analysis

The worst accident of this category (loss of the VCS and one heat removal loop at 205 kW decay heat load) causes maximum sodium temperatures of 561°F (294°C) and the maximum concrete temperatures of 323°F. Both are within the safety criteria set forth in Paragraph 15.6 (1200°F for sodium and 400°F for concrete).

FIGURE 15.6-3. Concrete Temperature (Surface) Response After Loss of One Loop and VCS.

FIGURE 15.6-4. Equilibrium Concrete Temperature.

FIGURE 15.6-5. Minimum Achievable Sodium Temperature vs Decay Heat in Storage Vessel.

are within the safety criteria set forth in 15.6 (1200°F for sodium and 400°F for concrete).

15.6.3 Liquid Metal Leaks

15.6.3.1 Identification of Causes

The probability of a leak in the NaK piping has been discussed in paragraph 15.6.1, Appendix A, and in References 5 and 6. Other than the thermal consequences already noted in paragraph 15.6.1, there will be no impact upon the vessel sodium. The consequences of a fire resulting from such a leak are discussed in Section 15.7. This paragraph, therefore, concentrates on the probabilities and effects of sodium leaks.

Three separate sodium leakage events are examined to determine the probability of sodium leaks which may affect decay heat removal from material stored in the FSF. These leakage events are:

- sodium leak in the purification loop
- leakage in the primary storage vessel below the sodium level
- leakage in the primary storage vessel (as above) with concurrent leakage in the guard tank below the equilibrium sodium level

First, the sodium purification loop consists of piping, valves, the PTI, the sodium heater, the cold trap, and the EM pump. These components are all in the cold trap vault. All the components are operated at low pressure (30 lb/in.² gage (210 kPa) maximum operating pressure contrasted with a 50 lb/in.² gage (340 kPa) design pressure), and normally the temperatures are 600°F (316°C) or less. Conceivably, a sodium leak could occur through a weld failure, a bellows failure in a valve, or a gross rupture of the piping. Based on containment quality weld failure probability data provided in Reference 5, leakage in the sodium purification loop was found to be a credible event.

Second, leakage of the primary vessel will occur only if the vessel design conditions are exceeded to the point of failure or if a random weld failure occurs. The worst credible heat removal system failures (see 15.6.1) were found to result in the sodium temperatures below those required for vessel failure. Thermally induced storage vessel rupture is, therefore, considered to be not credible. Vessel overpressure to the point of failure is also considered to be not credible based on the analysis done for Section 15.3 because the system is protected from overpressurization by redundant relief valves as well as a pressure regulator.

All the vessel welds are made to ASME Section III, Class 2 standards; each weld is x-rayed on the first and last pass and subject to a dye-penetrant test on every pass. The reliability of such welds on large vessels is known to be extremely high. An analysis performed in Reference 6 found the primary storage vessel welds to have a collective mean time between failure (MTBF) of 910 years. Such a leak, though extremely unlikely, is regarded as credible and its consequences are evaluated.

Third, leakage of both the storage vessel and the guard tank requires that, in addition to the unlikely leakage of the vessel (MTBF 910 years), some sort of failure must also occur in the guard tank. The guard tank is an unpressurized vessel and, therefore, overpressure is not a credible failure mode. The guard tank temperatures do not exceed the storage vessel temperatures for any of the credible thermal transient events and thus overtemperature failure is not credible. The tank is designed, built, and inspected to ASME Section III, Class 3 standards and weld reliability is correspondingly high. An analysis (Reference 6) was performed to determine the probability of guard tank weld failures. The MTBF calculated for the guard tank was 1,220 years. The combined probability of a primary vessel and guard tank failure was also calculated. The MTBF for this failure mode is 1.11×10^6 years.

The very low probability of weld leakage in both the storage vessel and the guard tank (less than 1×10^{-6} year⁻¹) provide reason not to consider this accident further.

15.6.3.2 Analysis of Effects and Consequences

The consequences of the two credible leakage accidents (purification system leak and storage vessel leak) were analyzed. The maximum leak rate from the sodium purification loop is no more than 50 gpm (3.2×10^{-3} m³/sec) flow rate through the sodium pump (the pump is nominally rated at 10 gpm, 165×10^{-3} m³/sec). In this accident, the pump could drop the vessel level to no lower than the elevation of the sodium intake nozzle at 536'5" (164 m). A siphoning accident could lower the sodium level to no lower than the siphon breaker elevation, also at 536'5" (164 m) elevation. In addition, there is an appreciable amount of time before the sodium level in the primary vessel can drop to the antisiphon level through siphon action (approximately 8 hours). During this time, there would be alarms transmitted to the FFTF control room due to low sodium level in the primary vessel and leak detection in the cold trap vault. Immediate action could be taken from the FFTF control room or the FSF control room to shut the purification loop isolation valves thus terminating the leak long before the sodium level gets down to the siphon breaker. The lowest equilibrium sodium level for this accident is above the active portion of the fuel and the NaK coolers; therefore, the heat removal capabilities of the FSF are not impaired. The cold trap vault floor is provided with a "catch pan" for spilled sodium. A large spill of sodium could cause some water loss from the concrete, but damage would be limited to a relatively shallow layer with no loss of structural integrity (Reference 14 and Paragraph 15.7.1.3.2).

The sodium volume in the storage vessel is sufficient to assure that any leak into the storage vessel/guard tank annulus will not allow the sodium to fall to a level lower than the 536'5" (164 m) elevation at a temperature of 300°F (149°C). This level is above the top of the active fuel and above the NaK coolers; therefore, the heat removal capabilities of the FSF are not impaired (Reference 15). The sodium trapped between the vessel and guard vessel will cause heat losses to the vault to increase slightly.

15.6.3.3 Results of Analysis

Because the sodium level in the vessel cannot drop to an elevation below 536'5" (164 m), the worst credible accident of this kind meets the safety criteria established in paragraph 15.1.

15.7 LIQUID METAL FIRES

The sodium and NaK fires accidents encompass all of the events in which a liquid metal fire may occur. This group of accidents has safety consequences for three reasons. First, radionuclide contamination in the sodium due to fuel pin failures, IDS sodium carryover, or neutron activation may be dispersed in a sodium fire. Second, liquid metal fire aerosols contain chemical irritants that may represent a health hazard when inhaled. Third, liquid metal fires may adversely affect the operation of plant components in their vicinity.

The presence of either sodium or NaK at FSF creates the possibility of an alkali metal fire. The FSF design makes the probability of such a fire remote and limits the consequences if one should occur. Some of the factors reducing the chance or extent of fire are:

- 1) The location of the alkali metals, Over 99% of the alkali metal at FSF is in a below-grade closed vaults.
- 2) Integrity of liquid metal systems. All liquid metal pressure boundaries are designed, fabricated, and tested to meet the requirements of Section III or Section VIII of Reference 9. Piping from the vessel to isolation valves in the cold trap vault and the cold trap itself are seismically hardened.
- 4) System pressures. The FSF liquid metal systems operate at very low pressure (50 lb/in² gage, 340 kPa) at all times. If a leak should occur, the low system pressures make it doubtful that a "spray" of any consequence could be formed.

- 5) Drip pans are provided beneath all liquid metal components to catch and contain leakage. The pans are sized to contain anticipated leaks. Thus, the liquid metals will be kept from contact with concrete unless an additional unlikely event (drip pan leakage) occurs. The low operating temperature of the liquid metals makes it unlikely that a significant concrete interaction would occur. The drip pans are covered with suppression plates which limit the reaction rate should a large leak occur (Reference 16).
- 6) Redundant Seismic Category I isolation valves with associated seismic switches prevent spilling of large amounts of sodium in a seismic event.

The accident analyses consider both alkali metal aerosols and possible offsite doses from any fission products present. Table 15.7-1 summarizes the important fission products postulated to be present in the storage vessel sodium. The sodium fire analyses considers the case of sodium contaminated by the design and expected activity levels.

The safety evaluation criterion for these accidents is that the site boundary dose to members of the unprotected public does not exceed the limits of 10CFR72 (proposed) or 10CFR100 (25 rem, off-site).

15.7.1 Sodium Spills and Fires

The sodium fires analysis reported here is intended as a reasonably conservative estimate of sodium spill/fires consequences.

15.7.1.1 Identification of Causes

The sodium inventory in the FSF is located in two below-grade vaults. The storage vessel vault has approximately 236,300 lb (107,184 kg) and the cold trap vault has approximately 770 lb (349 kg) of sodium (based on 56.316/ft³ at 400°F). The vessel, the sodium piping from the vessel to the storage vessel isolation valves (which are "fail-closed" valves requiring no operator action in the event there is damage to control circuitry or pneumatic lines to these valves), the NaK piping, and the cold trap are all seismic hardened. An estimate of the failure frequency for weld leaks in this system indicates that leakage may occur at the rate of 4×10^{-3}

events/year (Reference 6). If a rupture to a "nonhardened" pipe at the low point in the piping is assumed, a maximum of 26,000 lb (11,793 kg) of sodium could be siphoned in no less than 8 hours from the vessel to the cold trap vault (Reference 16). Of this amount only a small portion, 1100 lb (499 kg) could be released from a seismic event (assumed complete pipe rupture) due to closure of the seismic switches (assuming a very conservative one minute closure time). If leaks occur, they would be detected long before action of the siphon breaker could occur. Therefore, the analysis is based on a 1100-lb (499-kg) spill. The cold trap vault floors are provided with drip pans to prevent sodium/concrete reactions. If splashing causes some sodium to come in contact with concrete, the effects should be limited to local spalling of the concrete. A very large spill of sodium into the vault could be expected to heat some water out of the concrete; this would be a slow process and would be limited to a shallow surface layer. The increased moisture in the air could serve to lower the self-ignition temperature of the sodium. The system operates at very low pressures; the highest sodium operation pressure is less than 50 lb/in² gage (340 kPa). The normal operating temperature range of the sodium is 300°F to 500°F (149°C to 260°C). The expected sodium temperatures are 416°F (213°C) or less. At these low temperatures, self ignition is unlikely but it is assumed for conservatism. Because of this assumption and the fact that sodium leakage is credible, sodium fires are treated as credible accidents. Their consequences are analyzed and compared with the criteria for extremely unlikely events.

15.7.1.2 Consequence Analysis

Sodium from a leak will splatter somewhat, but most should drain below a fire suppression plate; this plate is assumed to reduce the burning rate to 3% of the burning rate without a fire suppression plate. The basic burning rate used is commonly associated with pool fires of sodium, at 950°F (510°C) or higher is air. Air to support continued combustion was assumed to enter the vault by natural convection through the vault cooling ducts (see Reference 16 for complete analysis). The parameters used to evaluate this fire were:

- Total sodium consumed - 1100 lb (499 kg)
- Burning area (gross) - 198 ft² (17.8 m²)
- Burning rate, 3% x 8 lb/ft²-hr = 0.24 lb/ft²-hr (11.49 Pa/sec)
- Aerosols released to atmosphere - 35%

TABLE 15.7-1
FISSION PRODUCTS IN STORAGE VESSEL SODIUM

Fission Product*	Expected Inventory No Leakers (μ curies/gm-Na)	Expected Inventory From 0.1% Leakers (μ curies/gm-Na)
CS-134	<0.1	0.9
CS-137	<0.1	5.0

* All other activities are more than three orders of magnitude lower than cesium.

- Dispersion factors - see Table 2.3-12
- Site boundary - 4.5 miles (7.2 km)
- Fission product contamination - Column 2 of Table 15.7-1

At the above conditions, even if the sodium fire were allowed to burn itself out, it would take approximately 22-1/2 days to consume all the spilled sodium. Assuming no dilution with excess air convected through the vault cooling ducts, a man standing in the duct outlet would receive a whole body dose at a rate of 780 mrem/hr or less if he were in a "beta suit".* As this "plume" would be highly visible, it is doubtful a person would remain in such a plume.

Offsite integrated doses were evaluated at the site boundary assuming a person was stationed in the plume for the duration of the event. The upper limit of radiological doses were calculated to be:

Maximum internal dose to any organ - <1 mrem (whole body)

Maximum external dose - <1 mrem (whole body)

The maximum sodium aerosol concentration at the site boundary would be <1 mg (NaOH)/m³.

15.7.1.3 Results of Analysis

15.7.1.3.1 Transient Pressure Rise

When hot sodium is spilled into an enclosed vault, heating of the vault atmosphere occurs and induces a rise in pressure. In the FSF, with its

*If activity were at the expected level, the dose rate would be less than 1 mrem/hr. The term "beta suit" is used to denote any type(s) of covering over all skin areas that has an attenuation factor approximately equal to greater than that of the skin it covers. Without the beta suit the skin or dose rate would be approximately 1600 mrem/hr.

limited amount of sodium spilled in a seismic event, this pressure rise is limited to an amount which will not cause vault overpressure.

15.7.1.3.2 Concrete Degradation

When hot sodium comes in contact with concrete*, certain chemical reactions must be considered. The heat of the sodium may cause the concrete to give up water vapor with a subsequent loss of structural integrity. Since the cold trap vault shares a wall with the storage vessel vault, it is necessary to assure that the supporting structures for the storage vessel will not be weakened. Data from sodium spill experiments were scaled in Reference 16 to the FSF case and it was concluded that no more than the first 2" of concrete could be affected. This damage is clearly insignificant in a wall that is 54" thick.

15.7.1.3.3 Hydrogen Generation

When concrete is heated, such as in a sodium spill, there is some evolution of water vapor which enters the vault atmosphere. This water vapor will react with the sodium and release hydrogen gas. Since the integrity of the vault cooling system is not assumed during an earthquake of design basis magnitude, oxygen will be present and the problem of explosive hydrogen concentrations assumes considerable importance in evaluating the sodium spill. Extrapolation from studies which have been performed to determine water loss and hydrogen generation shows that the maximum amount of H₂ that can be expected in the cold trap vault is less than 1-1/2% by volume which is well below the 4% by volume lower bound of the explosive concentration range (Reference 14).

*It is assumed here that the drip pans fail.

15.7.1.3.4 Radiation and Aerosol Release

Offsite doses from any FSF sodium fire will meet the safety criteria for such an event by a wide margin. Offsite doses from such an event are below the limits permitted for "anticipated" events. The sodium aerosols would not be excessive.*

The maximum onsite dose rate of 790 mrem/hr to the skin is significant; however, this dose rate applies to exposure directly in the plume at its source. Onsite access may be required during the considered event to engage in possible HRS recovery activity; access (in beta suits) of several hours without exceeding exposure limits (Reference 3) could be accomplished.**

15.7.2 NaK Spills and Fires

15.7.2.1 Identification of Causes

A NaK fire could occur if there should be a rupture or weld failure in one of the loops in the air environment of the NDHX cells or the pipeways. Based upon probability studies, the most likely place for a rupture or weld failure is in the piping composing the NDHX tube bundle. Such a fire is limited to a NaK supply of <475 lb (216 kg). In addition, a passive fire suppression system, composed of a sump and a grate that limits oxygen transport to a pool type fire, limits the burning time of such a NaK fire. A fire in the pipeways is less limited by the supply of NaK than by the supply of oxygen. The pipeways are below grade with only one opening through which combustion products and oxygen must pass in opposite directions. The NaK fires are evaluated as unlikely.

*The "safe occupational" threshold limit value for alkali metal aerosols is 2 mg/m^3 for continuous exposure during a normal workday. The aerosol would not exceed the 8-hr time-weighted average of 2 mg/m^3 as specified in 29CFR1910 (Reference 17).

**If activity were at the expected level, the dose rate would be less than 7 mrem/hr.

15.7.2.2 Consequence Analysis

In terms of the oxygen availability, the potential for a weld or pipe failure and the possibility of an ignition source, a NaK fire in the NDHX room is the worst NaK accident. It is conservatively assumed that 475 lb (216 kg) of NaK is spilled, that the burning area is limited to 5% of floor area ($\sim 14.5 \text{ ft}^2$, 1.3 m^2), and that the NaK burning rate is limited to $8 \text{ lb/ft}^2\text{-hr}$ ($650 \text{ g/m}^2\text{-sec}$). Radioactive contamination in the NaK is assumed to arise from activation in the NaK coolers where the coolant is exposed to the neutron flux from spontaneous fission of the stored fuel. This calculation (Reference 18) was performed to give a conservative estimate of the activation species in the NaK. The aerosol containing these activation products is assumed to be released at ground level.

The NaK fire, if allowed to burn continuously, could burn at the postulated rate for approximately four hours. Assuming that 35% of the NaK burned is released to the atmosphere as aerosols, the rate of release is 390 grams of aerosol per minute (75% potassium oxide aerosol). This results in a maximum site boundary (4.5 miles, 7.2 km) concentration of $0.9 \text{ mg-aerosol/m}^3$. The site boundary whole-body dose is $5.3 \times 10^{-4} \text{ mrem}$ due to the Na^{24} contamination of the aerosol.

15.7.2.3 Results of Analysis

The dose at the site boundary is $<0.1 \text{ mrem}$ and it complies with the 10CFR100 limit of 25 m by a substantial margin. The pessimistically calculated aerosol concentration arising from this event is below the 29CFR1910 limit (2 mg/m^3).*

*The "safe occupational" threshold limit value for alkali metal aerosols is 2 mg/m^3 for continuous exposure during a normal workday. The aerosol would not exceed the 8-hr time-weighted average of 2 mg/m^3 as specified in 29CFR1910.

15.7.3 BLTC Drip Pan Sodium Spill and Fire

This accident involves the spill of sodium contained within the BLTC onto the FSF building floor. Basically, this is a sodium fire in the FSF building and consequent release of sodium aerosols and radiation into the FSF building and then into the environment.

15.7.3.1 Identification of Causes

The design of the BLTC is such that there is no known failure mode that will result in a spill of a significant amount of sodium from the BLTC. If the BLTC were not mated to the floor valve and the BLTC bottom closure valve were open with the drip pan rotated away from below the fuel assembly, some drops of sodium from the film on the surface of the fuel assembly could potentially fall to the floor below. The oxygen contained in the FSF building atmosphere could potentially enter the opening and combine with the drip pan sodium and any sodium film or droplets from the fuel assembly. This accident would involve operator error or operational malfunction in opening the bottom closure valve when the BLTC is not mated to the floor valve.

The quantitative probability of such an accident mode involving unusual operator action is virtually impossible to establish. Because of this uncertainty, a fire involving the total contents of the BLTC drip pan sodium (1.8 gal at 500°F, $6.8 \times 10^{-3} \text{ m}^3$ at 260°C) and 0.15 gal ($5.7 \times 10^{-4} \text{ m}^3$) film on the fuel assembly is assumed to occur and the consequences are discussed in the following section. The consequences are evaluated against the criteria for extremely unlikely events.

15.7.3.2 Consequence Analysis

If 1.95 gal ($7.4 \times 10^{-3} \text{ m}^3$) of sodium at 500°F (260°C) (14.6 lb, 6.6 kg) is completely burned in the FSF building, 1520 g of sodium aerosol would be released into the high bay area. The radiation content of the BLTC

sodium is estimated to be 10.8 $\mu\text{Ci/g}$ based on the EOL contamination levels predicted for the IDS. The estimated maximum offsite boundary sodium aerosol concentration is 1.3 mg/m^3 . The estimated maximum site boundary radiation dose is 0.59×10^{-4} mrem from gamma radiation and 0.13×10^{-4} mrem from beta radiation. The internal dose to the lungs was 0.44 mrem.

The offsite radiation doseage is of the order of 10^{-6} of the limits imposed by 10CFR72 (proposed) or 10CFR100, and FSF building and onsite doses are within the supplementary guidelines of 10CFR20 and ERDAM 0524. The off-site sodium aerosol concentration is less than the one-hour threshold limit of 2 mg/m^3 set by 29CFR17 (Reference 19). This is based upon the conservative assumption of 21 mg/sec consumption of sodium in the fire. The total aerosol release time is 5.25 minutes.

15.8 LOSS-OF-SHIELDING ACCIDENTS

The loss-of-shielding accident group includes those accidents which may represent a radiological hazard due to inadequate shielding between the stored fuel assemblies and an occupied area. Due to the below grade position of the storage vessel, the dose limits imposed by 10CFR72 (proposed) or 10CFR100 at the site boundary can be met for any accident due to the great distance from the facility and the tremendous attenuation of the soil. The appropriate guideline for evaluation of this group would, therefore, be the doses allowed by 10CFR20 and ERDAM 0524 for on-site exposure to operating personnel.

15.8.1 Sodium Leak from Vessel

A leak in the FSF storage vessel may cause a drop in the sodium level. The sodium constitutes a portion of the neutron shielding between the stored fuel and operating floor. This would, therefore, be a loss of shielding accident.

15.8.1.1 Identification of Causes

The causes for a leak in the storage vessel and an estimate of the probability of the accident were presented in paragraph 15.6.3, "Liquid Metal Leaks." This is an unlikely event.

15.8.1.2 Analysis of Effects and Consequences

The shielding consequences of a lowered sodium level were analyzed for the case of minimum sodium level and a full turntable loading of fuel. This is the worst configuration that could result from a credible event for the following reasons. First, lower sodium levels are not credible because the guard tank prevents further leakage. Failure of the guard tank, in addition to a storage vessel leak, has been shown to be not credible in paragraph 15.6.3. Second, fuel loading that would provide greater source strengths or multiplication factors are not credible because only FFTF fuel will be stored in the FSF, and the shielding calculations take into account the worst load configuration multiplication factors and source terms available within the set of fuel assemblies to be stored. Dose rates associated with this accident would rise from a normal level (Reference 20) of 0.115 mrem/hr gamma and 0.0012 mrem/hr neutron to an accident level of 0.35 mrem/hr gamma and 0.029 mrem/hr neutron in the vicinity of the striker plate. This dose would exceed the FSF design goal for normal operation (≤ 0.2 mrem/hr) but not ERDAM 0524 limit for unlimited access of < 2 mrem/hr.

15.8.2 Loss of Borated Polyethylene Shielding Above Closure Head

15.8.2.1 Identification of Causes

It has been hypothesized that a fire caused by a spill of sodium from the BLTC catch pan could cause melting and redistribution of the borated polyethylene neutron shielding between the striker plate and closure head. Such an effect would be limited in extent due to the small amount of sodium available and the presence of a "can" around the polyethylene. There would,

however, be a small area where neutron dose rates would be greater. The BLTC sodium fire (15.7.3) is itself extremely unlikely. Thus a loss of the polyethylene shielding is also extremely unlikely.

The causes and probability of a fire of this sort were addressed in 15.7.3.1 and the BLTC sodium spill and fire was treated as a credible accident.

15.8.2.2 Analysis of Effects and Consequences

As a part of the analysis in the design of the closure/head striker plate, a comparison was made between the striker plate surface dose rates with and without the borated polyethylene shielding in place. This provides an estimate of the upper limit on the possible consequences of this sort of loss of shielding accident. The analysis was carried out using a source term based on the maximum anticipated fuel loading. Dose rates with the shielding in place (Reference 20) were 0.115 mrem/hr gamma and 0.005 mrem/hr neutron. Dose rates without the four inches of polyethylene shielding in place were 0.26 mrem/hr neutron and 0.32 mrem/hr gamma. This dose rate would exceed the FSF design goal for normal operation (≤ 0.2 mrem/hr) but not the ERDAM 0524 limit for unrestricted access (2 mrem/hr) and it would not pose an unacceptable hazard to personnel in the area.

15.9 NATURAL DISASTERS

The occurrence of accidents classified as natural disasters cannot be prevented by facility design features. The object of design for these events is, therefore, limited to mitigation of consequences given that the event occurs. The postulation of design basis natural disasters establishes a set of criteria for design of the facility to assure safe operation. The selection of design basis natural disasters for this purpose is such that phenomena which exceed them in severity are not considered credible. Because of the diversity of events covered in this group, the safety criteria are given on an individual basis rather than this introduction to the group.

15.9.1 Design Basis Earthquake (DBE)

The DBE for the FSF has been selected based on the seismic studies made for the FFTF reactor. The ground motion models generated for the design of that facility are those used for the design of the FSF. These ground motions were selected on the basis that earthquake generated ground motions in excess of the DBE were not considered to be credible for the HEDL 400 Area site. Critical FSF components are, therefore, specifically designed to withstand the debris and ground motion associated with the DBE.

15.9.1.1 Analysis of Effects and Consequences

The safety consequences of this event are evaluated with respect to the requirements of 10CFR100 "Reactor Site Criteria". The FSF must perform within the limits for extremely unlikely events after a DBE. In addition, several supplemental criteria were established for the evaluation of this event. Specifically:

- 1) Failure of the NaK heat transfer loop pressure boundary shall not occur.
- 2) Failure of the sodium pressure boundary inside the storage vessel vault or of components inside that boundary shall not occur.

15.9.1.2 Methods of Analysis

A stress analysis was performed on the below-grade building structure, the NDHX vaults, the storage vessel assembly, and the components of NaK heat transfer loops. The design of these components was found to be adequate to

withstand the forces induced by the DBE accelerations. The FSF stress report (Reference 11) concluded that:

- 1) The FSF turntable storage tubes and grid plates remained undistorted, thus maintaining the proper subcritical lattice spacing for the stored fuel.
- 2) The cover gas, NaK, and sodium pressure boundaries were intact.

Components of the FSF which are not designed to be operational after the DBE and which can be assumed to fail in this situation are:

- the FSF above grade structures excluding the NDHX vaults, the bridge crane, and the BLTC when mated to the closure head
- offsite power supplies (assumed lost)
- the NDHX stacks
- the sodium purification loop

After a design basis earthquake, those components which are seismically hardened are intact. All other components are assumed to fail. Under such conditions, it is still possible to assure that the fuel is stored in a safe, cooled, subcritical array.

Cooling is assured by providing the NDHXs with backup fans, generator and fuel supply for operation independent from offsite power. Moreover, it is possible to maintain air flow at a reduced but sufficient rate to prevent the sodium temperature from exceeding 800°F (427°C), even without the blower or the major portion of the NDHX stack; there is sufficient natural draft created by the seismically hardened portion of the NDHX to reach thermal equilibrium below 800°F (427°C) even at the maximum heat load and summer ambient temperatures.

Figure 15.9-1 shows the heatup transient of the sodium during such a one-loop, short-stack, design-heat-load accident. Reference 12 supplies further

FIGURE 15.9-1. Storage Vessel Temperature After a Seismic Accident.

details on loss of cooling accident. Such procedures (see Appendix B) may be carried out on either one of the NDHXs since only one is necessary to cool the stored fuel. It is estimated that the NDHX structures could be cleared in less than one day. The heat removal system design, therefore, allows for adequate decay heat dissipation even in the event of an earthquake which topples the NDHX stacks and temporarily interrupts off-site power.

Earthquake radiological safety is assured in the design of the FSF. Even if the pressure boundary were to fail there will be no significant leakage of fission gas from the system. The design of the heat removal system assures that the FSF can be placed in a stable thermal condition following the DBE thus preventing fuel pin thermal failure and the possible release of fission products to the cover gas. The design of the storage tubes preserves the subcritical positioning of the stored fuel assemblies and prevents mechanical damage to them. The exclusion of sodium purification system components from the Seismic I category means that sodium leakage may occur in the cold trap vault. In addition, failure of the above-grade vault cooling system air ducts may allow additional air to enter the cold trap vault to continue burning the spilled sodium. An earthquake sensitive relay system will shut off the purification loop isolation valves, thus limiting any sodium fire consequences to those calculated for the accident discussed in 15.2.4.1, "Sodium Spills and Fires". The doses expected from that incident were calculated to fall within the 10CFR100 limits. The occurrence of a DBE during a fuel transfer operation could conceivably lead to stalling of the fuel assembly in mid-transfer. The consequences of such an incident are at worst a stuck fuel assembly accident (15.5.4). Damage to a fuel assembly in the BLTC during the DBE was found to be negligible.

In conclusion, the 10CFR100 requirements for off-site dose are met by the FSF even under the conditions of the DBE. The supplementary criteria which apply to this event are also met.

15.9.2 Tornado

15.9.2.1 Identification of Causes

The DBT for the FSF is identical to that used in the design of other nuclear facilities at the HEDL 400 area. The tornado projectiles and wind forces used reflect an extreme weather condition which is considered to be extremely unlikely. Tornado-generated missiles or wind loads greater than those used for this design basis are not considered credible for the plant site. Tornado or extreme weather damage to components of the FSF which are specifically designed to withstand the DBT missiles and wind loads is, therefore, not credible. The DBT is extremely unlikely.

15.9.2.2 Analysis of Effects and Consequences

15.9.2.2.1 Safety Evaluation Criteria

The Safety consequences of this event are evaluated with respect to the requirements of 10CFR100, "Reactor Site Criteria." In addition, several supplemental criteria were established for the evaluation of this event. Specifically:

- 1) The cover gas pressure boundary shall remain intact (this includes the BLTC if it is mated at the time of the DBT).
- 2) The NaK heat transfer loop pressure boundary shall remain intact.

15.9.2.2.2 Methods of Analysis

Several stress analyses were performed on the closure head area of the FSF. This is the only portion of the FSF cover gas pressure boundary which may be exposed to tornado missiles and debris. (The BLTC was analyzed for tornado missile protection as a part of its design.) All other portions of the cover gas pressure boundary and the NaK and sodium pressure boundaries are

protected by the reinforced concrete, tornado resistant below-grade, and NDHX structures. None of the standard tornado missiles or anticipated debris were found to damage the cover gas pressure boundary. This is due to the energy absorbing characteristics of the striker plate and the massive construction of the closure head.

Components of the FSF which are not designed for the tornado and which must be assumed to fail in this situation are:

- the FSF above-grade structures excluding the NDHX vaults and the bridge crane
- off-site power supplies (assumed lost)
- the NDHX stacks (crimped shut due to tornado missile impact)

After the DBT, one must assume that only those components which are below-grade, housed in the NDHX vaults, or specifically tornado hardened are intact. All other components are assumed to fail. Under such conditions, it is still possible to assure that the fuel is stored in a safe, cooled, subcritical array.

Subcriticality is assured by positive positioning of the fuel assemblies in the storage tubes. The turntable and storage tubes are protected from the DBT by their position within a pressure boundary which is designed to remain unharmed during the DBT.

Assuming that the FSF is operating under the design basis heat load of 205 kW before the tornado and that the vault cooling system is rendered inoperable due to power outage and/or equipment damage, 1.3 days are available in which to clear debris from the NDHX structures and activate the generator and backup fans before the sodium temperature reaches 650°F (343°C). Three days are available before the temperature reaches 800°F (427°C). Those temperatures for faulted conditions have been determined to be satisfactory (paragraph 3.9.2). Reference 12 supplies further details on loss of cooling accidents. Such procedures may be carried out for either

one of the NDHX's since only one is necessary to cool the stored fuel. The heat removal system design, therefore, allows for adequate decay heat dissipation even in the event of a tornado that topples the NDHX stacks and temporarily interrupts off-site power.

Additional cooling is assured by providing the NDHXs with backup fans, generator, and fuel supply for operation independent from off-site power.

Radiological safety is assured in the design of the FSF by protecting the entire cover gas pressure boundary from tornado missiles and wind forces. With this pressure boundary intact, there will be no increased leakage of radionuclides from the system. The design of the heat removal system assures that the FSF can be placed in a stable thermal condition following the DBT thus preventing the release of fission gas to the cover gas due to fuel pin thermal failure. The design of the storage tubes preserves the positioning of the various stored fuel assemblies and prevents mechanical damage to them. The occurrence of a DBT during a fuel transfer operation could conceivably lead to stalling of the fuel assembly in mid-transfer when BLTC power failed. The consequences of such an incident are at worst the stuck fuel assembly condition (paragraph 15.5.4).

In conclusion, the 10CFR100 requirements for off-site dose are met by the FSF even under the conditions of the DBT. No abnormal release is expected from this event.

15.9.3 Flood

The FSF operating floor elevation at 550 ft (167.6 m) above sea level has the following flood margins for various river levels:

- 210 ft (64 m) above low water level (see paragraph 2.4.2)
- 180 ft (55 m) above standard project flood level (see paragraph 2.4.2)

- 160 ft (49 m) above probable maximum flood level (see paragraph 2.4.2)
- 160 ft (49 m) above hypothetical dam failure (Grand Coulee Dam) (see paragraph 2.4.2).

Based on these margins, the general flood protection provided by choice of the 400 area as the FSF site is sufficient. The FSF does not make any special provision for flooding, as such an event is considered not credible.

15.9.4 Loss of Off-Site Power

Loss of off-site power is treated here as a natural disaster because it is, like the weather and geology, not susceptible to modification by plant design features.

15.9.4.1 Identification of Causes

This accident involves the loss of off-site power to the FSF. It has already been assumed that such loss occurs during the DBE and DBT. Other initiators for this accident (usually consisting of local power grid faults) may cause loss of power only and have no other facility impact.

15.9.4.2 Analysis of Effects and Consequences

15.9.4.2.1 Safety of Evaluation Criteria

The safety consequences of this event are evaluated with respect to the requirements of 10CFR100, "Reactor Site Criteria." In addition, the consequences are compared with the supplementary guidelines of 10CFR20 and ERDAM 0524. Since this is an anticipated event, the criterion that no plant damage occurs must also be applied.

15.9.4.2.2 Method of Analysis

The FSF Design was examined to determine which components and systems would be unable to operate in a loss-of-power event. These were primarily:

- Trace heating
- Sodium purification EM pump and cold trap fans
- Vault cooling system pump and fans
- Normal facility utilities (lighting, HVAC, etc.)
- Turntable position system
- Noncritical instrumentation.

The consequences of trace heating loss at low decay powers are discussed in paragraph 15.10.1 and are found to be negligible from the safety standpoint.

Loss of circulation in the cold trap vault does not present any safety hazard. The safety consequences of a loss of vault cooling are analyzed in paragraph 15.6.2 and are found to be minimal when the natural convection heat removal system remains in operation.

Normal facility lighting will be inoperative but battery-powered emergency lighting is provided to allow some visibility in the FSF building. Inability to move the turntable during the loss of power will not be a handicap since fuel handling will be suspended for the duration of the outage.

Conversely, several safety-related systems would retain full operability during the loss of power, providing the following:

- 1) The cover gas pressure boundary remains intact.
- 2) The natural convection heat removal system would remain in full operation.

- 3) If the BLTC is engaged in a fuel assembly transfer at the time of the outage, the transfer will continue since the BLTC grapple power is provided from a floor station hooked into the FFTF backup power system.
- 4) Critical instrumentation such as radiation monitors, leak and smoke detectors, and selected process instrumentation will remain in operation, being supplied by a backup battery power source.

The presence of such design features assures that there will be no abnormal release of radioactive species from the facility and all radiation levels will remain within nominal values.

The criteria of 10CFR100, 10CFR20, and ERDAM 0524 are all met with large margins during such an incident, and there is no adverse effect on any of the plant components.

15.10 OTHER ACCIDENTS

There are several accident conditions covered in this section, all of which have minor consequences. These are:

- excessive vessel cooling
- closure head impact loads
- abnormal pressures in NaK loops

Again, the ultimate safety criterion for these events is that they do not lead to release of fission products that exceed dose limits in 10CFR100.

15.10.1 Excessive Vessel Cooling

15.10.1.1 Identification of Causes

Because sodium has a melting point higher than the surrounding temperatures, it is plausible that the storage vessel sodium might freeze. This event might occur after one of the following initiators.

- 1) Failure of the heat trace system on the vessel and the failure of the sodium in-line heater. This event has significance only if the vessel heat losses exceed the fuel decay heat generation rate. These heat losses can be reduced to less than 10 kW at sodium temperatures less than 300°F (149°C) by shutting the louvers and dampers in the NDHX and by shutting down the vault cooling system. Taking into account only the sodium mass, the temperature would drop at a rate of only 0.5°F/hr (0.3°C/hr) with a constant heat loss of 10 kW.
- 2) Too much heat removed by the heat removal and vault cooling systems. This could be caused by a drop in ambient temperatures or by too much air flowing through the NDHX. Removal of a significant portion of fuel without compensating in the NDHX might also lead to too much heat removal.

These incidents tend to be, in part, self-correcting. As the sodium temperature decreases, the driving force for the circulation of NaK diminishes, thus reducing the heat removal rate automatically.

- 3) Too much heat removed by the cold trap. Depending upon temperature and flow rate, the sodium cold trap can remove heat at a rate upwards of 40 kW.

This is an unlikely event since the occurrence of a power outage of sufficient duration to allow sodium freezing is also unlikely.

15.10.1.2 Consequence Analysis

After any of the events leading to this accident there is a relatively long period of time before anything serious could happen. If the sodium temperature drops below 270°F (135°C), material dissolved in the sodium might begin to precipitate on metal surfaces. Below 208°F (98°C), the sodium will freeze. If this accident should happen, there would be no damage to the storage vessel or argon cover gas piping, and there would be no release of radioactive material. Fuel pin clad temperature in the stored assemblies and canisters would not rise above the 1500°F (816°C) emergency temperature limit as shown in Reference 21. The storage vessel heater zone arrangement is designed to allow successive meltout of the sodium to a free surface. The feature allows recovery from an event in which the sodium freezes solid.

15.10.1.3 Results of Analysis

Because this accident leads to no release of vessel cover gas, there is no release of fission products and the safety criterion is met. The impact to plant operation is operational only, and there will be no significant damage.

15.10.2 Abnormal Pressures in NaK Loops

15.10.2.1 Identification of Causes

Low pressures (atmospheric) in the NaK loops could result from a depleted argon supply combined with a leak through the rupture discs. Overpressurization could result from several sources.

- 1) A failure or leak in two blocking valves between the argon source and the expansion tanks. The 30 lb/in² gage (206 kPa) source argon is normally throttled manually to 5 lb/in² gage (35 kPa), and then the expansion tank is isolated from the source. Any pressure over 15 lb/in² gage (103 kPa) requires the concurrent failure of the loop's disc and the pressure relief valve.
- 2) Any increase in the average temperature of the NaK in the loops. This will occur gradually as the total decay power in the vessel increases, but this can be vented from time to time to the atmosphere. It might also occur if there were a sudden change (hotter) in the ambient temperature and no compensating changes were made in the NDHX.
- 3) Lastly, it could occur if the one loop were to fail.

Any pressure increase will normally be limited to 35 lb/in² gage (100 kPa) by the pressure relief valves on the NaK expansion tanks.

15.10.2.2 Consequence Analysis

Any pressure in the expansion tanks up to the system relief pressure would not cause the pressure in the NaK loops to exceed the piping design pressure anywhere. Any pressure up to the supply headed pressure (30 lb/in² gage, 250 kPa) will not cause any NaK loop component to reach its rupture pressure. Credible pressure variations in the NaK loops would, therefore, not pose a threat of release. This accident thus complies with the radiological safety criteria.

15.10.3 Closure Head Impact (Dropped Loads)

Dropped loads or other impacting objects might cause damage to the closure head, initiating a release sequence.

15.10.3.1 Identification of Causes

Several possible causes for dropped loads or other mechanical impact on the closure head have been identified. These are involved with tornado missiles and crane faults. Tornado missiles were discussed earlier, in Paragraph 15.9.2, and are not considered here. The FSF overhead crane requirements, Reference 22, are designed to eliminate or minimize crane accidents, particularly dropped load accidents, resulting from mechanical failure, single control system failure and operator action. Thus, dropped load accidents are considered to be unlikely. Because of known cases of dropped load accidents, however, they are assumed to be credible.

15.10.3.2 Consequence Analysis

The heaviest load that the FSF Crane will move is the FFTF floor valve. A special lifting sling prevents the 17,000-lb (7711-kg) floor valve from being lifted more than 2 ft above the striker plate level. None of the other loads to be handled by the crane represent as much energy as a drop of this floor valve from 2 feet.

A stress analysis was performed to determine the effects of a drop of the maximum crane load (floor valve) from the maximum height 2 ft (0.6 m) onto the striker plate. Several conservative assumptions were made including impact on an edge (the worst possible point of impact) and zero energy absorption by the floor valve. The analysis showed that no damage would occur to the closure head or to the pressure boundary seals. The striker plate was deformed and would probably need replacement, but it had served its purpose.

A similar stress analysis was done for the event where the drop took place over the floor valve adapter. The results were essentially the same, with no damage to the closure head or pressure boundary, and local deformation of removable/repairable components.

Since none of the other drop loads have greater energies or greater penetrating power than the floor valve edge drop, it was concluded that the consequences of these events would be less severe.

In the absence of any damage to the pressure boundary, there would be no abnormal leakage of radionuclides and thus no radiological hazard. The

criteria of 10CFR100, 10CFR20, and ERDAM 0524 are met, and this accident is found to be without significant safety consequences.

15.10.4 Conventional Fires

Throughout the FSF, care is taken to prevent accumulation of combustible materials and to prevent such materials from being exposed to an ignition source. However, some combustible supplies are required to operate the FSF.

The presence of combustible materials in an air atmosphere does present some fire hazard, and fire protection systems have been installed in the FSF to permit use of some additional convenience materials. The materials which may fuel a conventional fire have been identified. They are:

- control room library and supplies (manuals, desks, papers, upholstery, etc.)
- protective clothing and supplies
- waste bin contents
- electrical insulation
- mops, rags, and janitorial supplies
- air conditioner oil and NDHX generator fuel

The presence of only small amounts of combustible materials and the general absence of potential ignition sources make a fire in the FSF unlikely. The presence of fire control systems in the FSF make the spread of a fire improbable.

15.10.4.1 Area Isolation

Certain areas are effectively isolated from the rest of the FSF by having their own air ventilation system and an effective electrical fire barrier in the form of conduit runs. The following areas are effectively isolated from the remainder of the FSF.

- control room
- heat exchanger
- motor control room
- argon vault room
- vault cooling room

15.10.4.2 Consequence Analysis

The consequence of a fire in each area has been subjected to analysis with an estimate of the maximum amount of combustibles in each area. The results of this analysis are presented.

Roof Fire -- A fire in the composition roof would be effectively limited to one of five sections. Complete destruction of the roof in any section would result in suspension of further fuel transfer operations in the FSF but would not result in the failure of a safety related component nor cause a public hazard condition. A roof fire is readily accessible to the fire department.

Vestibule Fire -- A vestibule fire will be detected and suppressed by the automatic sprinkler system. The combustible should be limited to 10 lb (4.5 kg) of plastic products and 10 lb (4.5 kg) of paper and wood products. The worst condition for fire would combine unattended operation with a failure of the sprinkler system. The results of a fire in this limited amount of combustibles will be contained in the vestibule area without affecting the operation.

Janitor Closet -- A janitor closet fire will be detected and suppressed by the automatic sprinkler system. The combustibles should be limited to 10 lb (4.5 kg) of paper, cloth, and wood products as well as two gallons of various cleaners, which should be considered flammable, in various containers. The worst conditions for fire would combine unattended operation with a failure of the sprinkler system. This fire is not expected to extend beyond the closet and should not affect any operation.

- Toilet Room -- A fire in the toilet room will be detected and suppressed by the automatic sprinkler system. The combustibles should be limited to 10 lb (4.5 kg) of paper, wood, and plastic products. The worst condition for fire would combine unattended operation with a failure of the sprinkler system.
- The result of this fire would not affect operation and should only result in smoke damage.

Control Room -- A fire in the control room will be suppressed by the automatic Halon system. It is conceivable the control room may contain 125 lb (57 kg) of paper and wood products along with 250 lb (113 kg) of electronic PCB and insulation. Another 120 (54 kg) lb of plastic furniture and floor covering may be installed. It appears sufficient combustible material is present that a fire with a defective fire suppression system would not only make the control room inoperative but may spread to the roof. Without a control room, continued fuel transfer operations would be suspended, but the FSF would continue to reject heat without a failure of a safety related component and would not cause a public hazard. FSF heat removal system control would be by local manual mode with local readout of temperature.

Natural Draft Heat Exchanger -- This is essentially an isolated area without automatic fire protection. This area contains two sources of combustibles for a conventional fire -- electrical insulation estimated at 20 lb (9 kg) and the fuel for the gasoline engine on the backup electric power supply. The worst conventional fire would result from engine fuel system leaks or spillage during the filling of the fuel tank. The maximum amount of the fuel spillage should be limited to the 2-gal. (7.6-litre) container used to replenish the 1.5-gal. (5.7-litre) engine fuel tank. The fire from the fuel spill could ignite the insulation in the area disabling the standby cooling system. As discussed in paragraph 3.1.10, this NDHX is a part of a 100% redundant system and the removal of one NDHX from the system cooling of the FSF would not result in a public hazard.

The main fuel supply for the NDHX standby cooling system will be stored at a safe distance from the FSF in compliance with the NFPA requirements for fuel. Administrative control will limit transfer of this fuel to the NDHX to a one 2-gal. (7.6-litre) safety container.

Motor Control Room -- A motor control room fire will be detected and suppressed by the automatic sprinkler system. The only combustible in this isolated area is electrical insulation estimated to be less than 100 lb. (45 kg). An electrical fire in this area, with or without fire protection will be cleared by automatic load protection, and would result in the same effect as a loss of power covered in paragraph 15.2.6.4. It is expected this fire would suspend additional fuel transfer, but since cooling and manual control will still be available, no public hazard would result.

Mechanical Equipment Room -- The mechanical equipment room houses the heating, ventilation and air conditioning equipment and is protected by automatic sprinklers. It is estimated the maximum combustibles will be approximately 100 lb (45 kg) of electrical insulation and about 3 gal. (11.4 litres) of oil in the hermetically-sealed compressors.

A fire in this area, at the most, would disable the HVAC system and the attendant power distribution. The loss of the HVAC system is discussed in paragraph 3.11.4 and the loss of power is discussed in paragraph 15.9.4. The loss of these systems do not affect the performance of a safety related component and a public hazard will not result. The FSF manual control will not be affected.

Vault Cooling Equipment Room -- The vault cooling equipment room is protected by automatic sprinklers. The combustibles contained in this room are estimated at less than 70 lb of electrical insulation. A fire in this room, at the maximum, would disable the vault cooling system. The effect of the loss of the vault cooling system is described in paragraph 15.6.2 and does not create a public hazard. Since this is an isolated area, a fire would be easily contained.

Electrical Equipment Room -- The electrical equipment room is protected by an automatic sprinkler system. Combustibles in this room are estimated at less than 40 lb (18 kg) of insulation. A fire in this room, with or without fire protection would be similar to a loss of power failure as described in paragraph 15.9.4, except a fire in this room may damage the emergency power system. If the emergency power was lost, adequate power would still be available through the normal system from the motor control room. The loss of this room's functions would not cause a public hazard, as all manual control would still be functional along with most normal control systems.

Argon Vault, Stairwells, and Passage -- These three areas are considered together because no identifiable combustibles are stored or installed there, and a conventional fire is not anticipated. Stairwells and passages are protected by an automatic sprinkler system, while the argon vault is protected by portable fire extinguishers.

High Bay (Process) Area -- Fire protection in this area is provided by portable extinguishers. The fire most difficult to control will result from liquid metal spills from the BLTC (covered in paragraph 15.7.3). Other combustibles in this area are an estimated 200 lb (91 kg) of protective clothing and supplies, and an estimated 300 lb (136 kg) of electrical insulation found in motors and cable. Protective clothing will be stored in the provided storage cabinets and is not a significant conventional hazard. A fire in the floor valve motor or crane motor may cause the suspension of continued fuel transfer operations but would not affect a critical component nor create a public hazard.

15.10.4.3 Fire Detectors and Alarms

Detectors are installed in the FSF and connected in zones or areas to provide early detection of a conventional fire. The signal from a detector is displayed as a zone on a building annunciator, and in the FSF control room.

Upon receipt of a detector signal, the annunciator activates the local alarm system, starts the automatic fire suppression system, and sends a signal to the FFTF Control Room and to the fire protection loop.

A separate alarm system, through which detailed instructions can be issued, can be activated from the control room.

The detector and alarm systems installed in the FSF are:

- smoke detectors
- infra-red detectors
- flow/pressure detectors
- manual alarm stations
- ionization type detectors

With administrative control limiting the amount of combustibles in the FSF, the installed fire detection and automatic suppression systems provide adequate coverage of conventional fires. A fire is not expected to remain undetected but limiting the combustibles prevents any conventional fire from damaging critical components or causing a public hazard.

15.10.5 Accidents at Other Facilities

The presence of other facilities in the vicinity of the FSF creates the possibility of accidents which affect FSF safety but which do not originate within the FSF.

15.10.5.1 Identification of Causes

Two facilities in the FSF vicinity were identified as being capable of affecting FSF operations because of accident-induced hazards. These are the WPPSS power plant and the FFTF test reactor. The worst accident associated with the WPPSS power plant is the release of a chlorine gas cloud. The worst accident associated with the FFTF is the cladding failure of fuel assemblies in the IEM cell and subsequent release of the fission products.

15.10.5.2 Consequence Analysis

During normal operation, the FSF is an unattended facility and is capable of extended periods of operation without operator intervention. The FSF could therefore be evacuated without adverse safety effect until the accident hazard had passed. However, during the period immediately following an earthquake or tornado it may be necessary for personnel to engage in recovery operations to assure continued heat removal. Completion is expected within the first 24 hours following an accident. The actual procedures would involve no more than 4 hours of onsite work. Such operations are safety related, and the impact of accidents at other facilities were evaluated with respect to these activities.

WPPSS Chlorine -- A chlorine gas cloud produced by a WPPSS reactor accident would not prevent operations at FSF, even subsequent to a concurrent FSF accident, because personnel could be equipped with respiration protection and protective clothing to allow recovery activities to continue. Such protection is available from a central location at the 400 area, and site personnel would have access to it for FSF recovery operations.

IEM Cell Radiation Release -- The most serious effect on FSF from an FFTF accident would be fuel cladding failures in the IEM Cell subsequent to an earthquake. If this event occurred, there would be recovery operations needed at FSF such as clearing collapsed NDHX stacks. The FSF recovery operations are expected to take less than 24 hours. The release of fission products from the IEM Cell is limited by either inventory or containment (FSAR for the FFTF) and will not interfere with the shutdown and isolation of the plant or prevent FSF recovery activities.

15.10.5.3 Results of Analysis

None of the offsite accidents was found to prevent safe operation of the FSF. The nature of the FSF design permits unattended operation, allowing

personnel to evacuate the FSF without compromising safe operation. Recovery operations that may be necessary after earthquakes or tornados would not be prevented by the simultaneous occurrences of an accident at another facility.

15.10.6 Asphyxiation

15.10.6.1 Identification of Causes

Argon with less than the required oxygen content to support human respiration is used within the FSF. Argon can present a hazard for asphyxiation only if it should leak into inhabited areas and dilute the oxygen content to asphyxiation levels. The location of significant contained volumes of argon, except for main argon supply bottles located in the open air, are below grade. The argon volumes are in the storage vessel cover gas system, the backup argon storage bottles, and the NDHX cell NaK expansion tanks (small volume), which are above grade.

15.10.6.2 Consequence Analysis

The argon cover gas in the storage vessel is contained in a high-integrity sealed system within a tornado-protected boundary. There is no pathway for human entry into the cover gas space and little potential for significant leakage other than planned releases through the FSF stacks. The ports to the system are O-ring sealed; if leakage of these seals occurred the gas would leak into the large volume of the crane bay where its mixing with the air would not cause an asphyxiation problem.

All relief valves are exhausted to the NDHX stacks for dispersion or directly to the outside.

The NaK cover gas boundary is normally isolated from the rest of the argon system and is seismically qualified to Category I requirements.

There is the possibility that argon could leak into below-grade areas of small enough volume to dilute the oxygen levels significantly. To minimize the risk of asphyxiation, all normally occupied areas* are monitored continuously with alarmed oxygen monitors. The alarms indicate locally and at the FSF and FFTF control rooms. All normally unoccupied areas, such as pipeways, require entering personnel to have portable oxygen monitors with them; all entrances to these areas are posted as to entry requirements.

15.10.7 Effects of FSF Accidents on Other Nearby Facilities

15.10.7.1 Identification of Causes

There are two potential results of accidents that could effect other nearby facilities. These are radiological and/or chemical.

15.10.7.2 Consequence Analysis

Radiation, direct or from internally deposited radioactivity, from FSF sources does not effect the operation of other nearby facilities based on the dose levels listed in Table 15.1-1 for on-site locations.

The worst case chemical release would be sodium and potassium oxide releases from alkali metal fires. Near the FSF, the aerosol concentrations may reach an order of magnitude above the accepted TLV value.** These levels could cause an area evacuation but would not cause unsafe operations of the FTR (Fast Test Reactor). The control room of the FTR has been designed to be operated with larger alkali oxide aerosol concentrations in the outside atmosphere.

*The facility is normally unmanned, but this term is used for free access areas where personnel may enter on a fairly routine basis.

**The "Safe occupational" threshold limit value for alkali metal aerosols is 2 mg/m^3 for continuous exposure during a normal workday. The aerosol would not exceed the 8 hour time weighted average of 2 mg/m^3 as specified in 29CFR1910.

15.11 REFERENCES

1. Code of Federal Regulations, 10 CFR 100, "Reactor Site Criteria," USGPO, Washington, DC.
2. Code of Federal Regulations, 10 CFR 20, "Standards for Protection Against Radiation," USGPO, Washington, DC.
3. ERDA Manual, "Standards for Radiation Protection," Chap. 0524, US Department of Energy, Washington, DC, March 1977.
4. Fast Flux Test Facility Final Safety Analysis Report, HEDL-TI-75001, Hanford Engineering Development Laboratory, Richland, WA, December 1975.
5. "Reactor Safety Study, Appendix III," WASH 1400, US Nuclear Regulatory Commission, Washington, DC, October 1975.
6. A. C. Karkau, Probabilities of NaK and Na Leaks, N106T1230004, Atomics International, Canoga Park, CA, March 1977.
7. ERDA Manual, "Nuclear Criticality Safety," Chap 0530, US Department of Energy, Washington, DC, December 1976.
8. AEC Manual, "Safety of AEC-Owned Reactors," Chap. 0540, US Department of Energy, Washington, DC, January 1975.
9. ASME Boiler and Pressure Vessel Code, The American Society of Mechanical Engineers, New York, NY.
10. Criticality Calculations for the Final Design of the FSF Storage Vessel, N106T1210005, Atomics International, Canoga Park, CA.
11. W. L. Propes, Final Stress Analysis Report of the Fuel Storage Facility Vessel Subsystem, N106SR220002, Rev. NC-1, Atomics International, Canoga Park, CA, September 1977.
12. D. Fong and K. Weston, NDHX Emergency Heat Rejection Study, N106T1310009, Rev. A, Atomics International, Canoga Park, CA, June 1978.
13. Fuel Storage Facility System Design Description, N106SDD210001, Rev. 1, Atomics International, Canoga Park, CA, April 1978.

REFERENCES (Cont'd)

14. K. Weston and T. Nakae, FSF Sodium Leaks and Potential Consequences, N106T1230010, Atomics International, Canoga Park, CA, June 1978.
15. R. Houston, Determination of Minimum Leak Level for FSF Storage Vessel Subsystem, N106T1220005, Rev. C, Atomics International, Canoga Park, CA, June 1978.
16. K. Weston, Alkali Metal Fire and Dose Analysis for FSF Preliminary Safety Analysis Report, N106T1230006, Rev. A, Atomics International, Canoga Park, CA, October 1977.
17. Code of Federal Regulations, 29 CFR 1910, "Occupational Safety and Health Standards," USGPO, Washington, DC.
18. S. Berger, Miscellaneous FSF Shielding Analysis, N106T1210004, Rev. A, Atomics International, Canoga Park, CA, October 1977.
19. Code of Federal Regulations, 29 CFR 1910, OSHA Safety and Health Standards, USGPO, Washington, DC, November 1978.
20. D. B. Barr, Dose Rates Above the Top Shield of the FSF, N106T1210007, Rev. A, Atomics International, Canoga Park, CA, June 1977.
21. C. Kudiga, Sodium Freeze Accident Scoping Analysis, N106T1200001, Atomics International, Canoga Park, CA, April 1978.
22. Fuel Storage Facility Title I Engineering Final Report, FSF 8954-1/2, Appendix C, Section 14A, Norman Engineering Co., Los Angeles, CA, September 1977.

A P P E N D I X A

LOSS OF HEAT REMOVAL SYSTEM ACCIDENTS

APPENDIX A

LOSS OF HEAT REMOVAL SYSTEM ACCIDENTS

While analyzing accidents in the loss-of-HRS category, it is important to consider if the accident can be remedied either by repairs or by change of operating conditions or if the accident involves the permanent, inoperable loss of a loop.

Those accidents in the first category (reparable) include damage to any piece of equipment not constituting part of the liquid metal boundary (NDHX, the stack, electrics). It includes NaK piping failures (rupture weld failures) that occur outside the storage vessel vault. Accidents (pipe failures) below the sodium level also are included and are discussed later. Electrical losses or failures have little impact upon the system because there are no electrical devices required for operation. Flow and temperature measurement are not important except over a long period of time.

The NaK heat removal loops are designed and fabricated to the ASME Section III, Class 3, Boiler and Pressure Vessel Code. This high quality piping is operated at least 20 lb/in.² gage (140 kPa) below the design pressure, and the loops are designed to withstand DBE seismic and tornado forces. Nevertheless, potential weld or piping failures represent credible failure modes that could cause permanent loss of a loop. In analyzing these potential failures, the following variables are considered.

- 1) The decay heat load present in the vessel. The normally expected maximum heat load is 144 kW. A load of 205 kW (Figure 15-A-1) is the upper bound on any credible decay heat load; this could occur only after an FFTF core dump with space in the storage vessel for exactly 75 fuel assemblies dumped. In any analysis, it is unrealistic to hypothesize that 205 kW would be accepted into FSF with one heat removal loop already inoperable, because the normal maximum acceptable decay power

FIGURE 15-A-1. Estimated Decay of Thermal Power in FSF Storage Vessel.

will be administratively controlled to a level where the vault cooling system alone would provide adequate heat removal capability if the remaining loop is lost. Thus any analysis with both loops failed and a decay power of 145 kW to 205 kW requires nearly simultaneous failures (0-4 months apart) of both loops because 205 kW will decay to below 144 kW by the end of four months.

- 2) The location of the heat removal system failures. Failures in the piping that occur outside of the storage vessel vaults can be considered repairable though that loop may be temporarily shut down (repairs will take ≤ 30 days). Failure occurring inside the storage vessel vault must be considered irreparable. If the failure(s) occurs below the sodium level in the storage vessel, there exists a possibility of retaining the heat removal capability of the failed loop(s). By pulling a vacuum on the NaK cover gas, sodium can be drawn into the heat removal loop for use as a working fluid. By this means, the coolant level in the loop may be maintained, though the coolant composition will change. In any case, the heat removal loop would remain functional even with purge sodium as a heat transfer fluid.

With the above considerations in mind, the following special cases are postulated to bracket potential accidents involving loss of the heat removal system.

	<u>Initial Decay Power at the time of Accident</u>	<u>Accident Description</u>
Case A	205 kW	One loop failed (anywhere)
Case B	144 kW	Both loops failed (anywhere)
Case C	145-205 kW	Both loops failed anytime within 4 months following an FFTF core dump

	<u>Initial Decay Power at the time of Accident</u>	<u>Accident Description</u>
Case D	144 kW	Both loops failed in the vaults. (i.e., in an inaccessible area)
Case E	144 kW	Both loops completely inoperable (failure in vault and <u>above</u> the sodium level)

The probability analysis of all the above events assumes that no effort is made to repair or to incorporate any emergency cooling procedures.

A study (Reference 2) made to evaluate the probabilities of various loss of heat removal system accidents produced data tabulated below.

	<u>Event Description</u>	<u>Mean Time Between Failures (MTEF)</u>
1.	Weld or pipe failure in loop, 0-205 kW (Case A)	132 years
2.	Failure below the vessel sodium in either loop, 205 kW	1,441 years
3.	Failure, other than (2), in the vessel vault in either loop, 205 kW	1,297 years
4.	Failure outside the vessel vault (i.e., an accessible area)	163 years
5.	Both loops failed, 144 kW (Case B)	69,444 years
6.	Both loops failed, 145-205 kW (Case C)	2.78×10^6 years
7.	Both loops failed in the vaults, 144 kW (Case D)	1.87×10^6 years
8.	Both loops permanently inoperable, 144 kW, (Case E)	6.57×10^6 years

The MTBF for Event 6 (Case C) indicates that failure of both loops at heat loads greater than 144 kW is not credible and need not be evaluated. Similarly, Events 7 and 8, which respectively correspond to inaccessible and permanently disabling failures of both loops are also not credible.

The worst credible failure mode (Event 5) corresponds to failure of both loops at heat loads of up to 144 kW with at least one of the loops repairable within 30 days. The consequences of this accident were therefore evaluated in paragraph 15.6.1 and a thermal analysis (Reference 4) was performed in order to demonstrate compliance with the loss of HRS safety criteria called out in Section 15.6.

A P P E N D I X B

DBE AND DBT HEAT REMOVAL SYSTEM RECOVERY

APPENDIX B

DBE AND DBT HEAT REMOVAL SYSTEM RECOVERY

The NDHX stacks may be damaged by a DBE or DBT. These stacks are usually necessary for dissipating the decay heat by providing a driving force for convection across the NaK tube bundle. If both loops lose this natural convection flow, the NDHX room must be entered and the backup motor-generator-fan set must be activated to provide forced convection flow. Debris blocking the NDHX room door or NDHX inlet or outlet ducts may complicate matters, but it is a problem capable of immediate brute-force solution (i.e., bulldozer or crane). Such activities may be carried out for only one of the loops since a single loop is sufficient to reject the entire heat load. Two possible hazards exist when attempting to perform these operations.

First, accidents at other facilities arising from the DBE may cause a rise in radiation levels which would force personnel engaged in recovery activities to use protective measures (respirators, Beta suits).

Second, in the unlikely event that there is a sodium fire in the FSF purification vault or if a 1.4 kW assembly was involved in a fuel transfer and stuck opposite the floor valve adapter seals, there is the possibility of radioactivity release. In such a case, workers would have to use protective clothing and respirators to minimize doses. A calculation of the dose rate for the maximum fire after a DBE was made in Reference 6 (paragraph 15.11). The whole body dose to a worker standing directly in the aerosol plume at its source was conservatively calculated to be 400 mrem in the first hours. The allowable dose for a worker undertaking such a recovery task is 12 rem. This would allow a minimum of one shift (8 hrs) to perform the recovery task. The recovery could thus be performed within acceptable emergency guideline doses for all parties involved. In order to place the probability

of dosing a worker to 12 rem in its proper perspective, however, the following conservatisms must be cited:

- 1) Rupture of purification loop and nitrogen duct is extremely unlikely.
- 2) It is unlikely that a pool of sodium would ignite at the operating temperatures present.
- 3) The plume dose is calculated based on complete combustion of oxygen component in the air (an unlikely event).
- 4) The worker would have to situate himself directly in the plume during his work (unlikely).

The probable dose to a recovery worker would thus be substantially less than the maximum dose of 400 mrem per hour.

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CHAPTER 16
TECHNICAL SPECIFICATIONS

16.0 TECHNICAL SPECIFICATIONS

The technical specifications for the Fuel Storage Facility are included with the Technical Specifications for the Fast Flux Test Facility (FFTF) in the FFTF Final Safety Analysis Report, HEDL-TI 75001, Hanford Engineering Development Laboratory, Richland, WA.

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CHAPTER 17
QUALITY ASSURANCE

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17.0 QUALITY ASSURANCE

The Fuel Storage Facility is managed by the FFTF Project Office with separate contractors to perform design, construction and operation as shown in Figure 17.0-1. Respective contractors prepared the individual QA sections which describe their own QA programs. As part of the project manager's function, FFTFPO reviewed all contractor QA programs and performed several audits. In this FSAR, HEDL did not revise the sections submitted by other contractors.

17.1 QUALITY ASSURANCE DURING DESIGN AND CONSTRUCTION

Quality Assurance during the Title I and Title II design of the FSF is provided by both Atomics International (AI) and Norman Engineering (NECO). NECO is responsible for the design of the building site, structures, and utilities. AI is responsible for the design of the sodium and supporting system of the FSF. AI will also provide Quality Assurance during the procurement of long-lead components to be used for FSF construction.

Quality Assurance during construction of the FSF will be provided by J. A. Jones and Vitro Engineering. The Hanford Engineering Development Laboratory, which will operate FSF, will provide Quality Assurance during operation.

17.1.1 Organization

17.1.1.1 AI Organization

The AI Quality Assurance Department is an independent organization reporting directly to AI's President. Its organizational structure is shown on Figure 17.1-1. Depicted on Figure 17.1-2 is the relationship of the QA Department with the various functions involved in the FSF program, such as the Program Office, Engineering and Procurement. While providing quality-related services to the various functions, the QA Department maintains its independent status by its line of authority.

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FIGURE 17.0-1. Quality Assurance Organization.

FIGURE 17.1-1. Atomics International QA Organization Chart.

FIGURE 17.1.2. AI Organization for FSF Program (Existing During Title I and II Engineering).

Quality Assurance policy originates with the President of Rockwell International, through the issuance of a Corporate Policy statement covering Product Quality, Reliability and Safety. The Quality Policy is issued to each division of Rockwell International with a Corporate Directive, which is prepared and authorized by the Senior Vice President, Corporate Staffs, and which directs each division to take action to implement the Corporate Quality Policy. The President of the AI Division implements the Corporate Quality Policy Directive through Standard Operating Policies, which provide quality assurance direction consistent with Corporate Policy, as well as the Quality Assurance Program requirements applicable to AI business objectives and contract requirements. The overall Quality Program is implemented in the operating manuals of the quality-affecting organizational units by the managers of these units. The Quality Assurance Director, who reports directly to the AI President, verifies compliance of the quality-affecting organizations to the Quality Program under the authority granted in the Standard Operating Policies.

Quality assurance functions implemented within AI are defined in Standard Operating Procedures. All functional organizations (Program Offices, Engineering, Purchasing, Quality Assurance, and Manufacturing) are assigned responsibility for:

- the preparation and issuance, in their operating manuals, of written instructions and procedures
- the indoctrination and training of their personnel in these procedures, as applicable to their work assignments

In addition, the Quality Assurance Director is assigned the following specific quality assurance functions:

- identifying those procedures which cover the performance and verification of quality-related activities
- conducting audits of the implementation of such procedures

- identifying quality deficiencies and problems in the Program and reporting them, with any recommendations, to the responsible AI executive, functional, and program managers
- verifying that solutions to reported quality problems or deficiencies are achieved
- stopping nonconforming work, and controlling further processing, fabrication, and delivery of nonconforming items
- submitting overall status reports on the AI Quality Assurance Programs to the AI President, as well as to concerned program and functional managers

The division of functions within the Atomics International Quality Assurance Department are as follows.

Quality Audits and Procedures Staff -- The AI audit program responsibilities of the Quality Assurance Director are implemented through his Audit Staff. Procedures Staff is responsible to the Quality Assurance Director for development, issuance, control, and revision of Quality Assurance Manuals and procedures. Additionally, the Procedures Staff reviews procedures, and revisions thereto, prepared by other quality-affecting organizations, to assure compatibility with overall AI Quality Assurance Program requirements.

Instrument and Technical Support Unit Manager -- The Instrumentation and Technical Support Unit Manager provides the following:

- nondestructive examination technical support and consultation to AI organizations and suppliers
- chemical, physical, and mechanical property testing services to support other Quality Assurance Department units

Procurement Quality Assurance and Planning Unit Manager -- The Procurement Quality Assurance and Planning Unit Manager provides the following:

- Procurement document review and supplier quality surveys for materials and fabrication services, and maintenance of the approved list of such suppliers

- source inspection, planning, surveillance, and audits of suppliers of materials and fabrication services
- receiving inspection planning
- a quality data and records collection and storage system for procured items
- Quality Assurance Program administration for FSF activities, to monitor and assure effective implementation of quality requirements, from design through procurement and installation
- Quality Engineering support for change control boards, design reviews, and design document review and approval for the FSF program
- review and evaluation of bid invitations and returns
- participation on capability evaluation teams for prospective suppliers of major items

Inspection and Test Unit Manager -- The Inspection and Test Unit Manager is responsible for the following:

- performing inspection and testing services for all products manufactured by AI
- performing receiving inspection of procured items and services, identifying and documenting nonconforming conditions of these items and services, and assuring prompt conformance to the established nonconformance dispositions
- performing inspection of packaging, preservation, and identification of items prior to shipment

Quality Assurance Engineering Unit Manager -- The Quality Assurance Engineering Unit Manager provides quality assurance engineers to support the FSF Quality Assurance Program Administrator. Quality Engineering personnel perform the following activities:

- nonconforming item review board coordination
- development and implementation of statistical test programs and analyses
- evaluation of inspection and test data and reporting of quality trends

Qualification Requirements for AI QA Management Positions -- The minimum qualification requirements for the various AI Quality Assurance Department management positions and the program administrators are as follows.

- Quality Assurance Director shall have:
 - (a) A Bachelor of Science degree in Engineering, Science, or Technology from an accredited college or university;
 - (b) 15 years of experience in quality assurance or engineering in an advanced technology industry, of which at least five years will be in quality assurance; and, of this five years, at least two years will be in the nuclear area;
 - (c) experience in the direction of personnel and the planning and management of resources needed to conduct a Quality Assurance Program;
 - (d) a knowledge of industry and government codes, standards, and regulations defining quality assurance requirements and practices; quality assurance administrative methods and technology and their application; and, experience in planning, defining, and performing quality assurance practices and application of procedures.
- FSF-QA Program Administrator shall have:
 - (a) Bachelor of Science degree in Engineering, Science, or Technology, or equivalent training plus experience in quality technology
 - (b) five years experience related to the design or manufacture of complex components, of which at least two years will have been as a quality engineer involved in fabrication of nuclear pressure boundary components.
- Quality Assurance Engineering, Procurement Quality Assurance and Planning, and Instrumentation and Technical Support Unit Managers shall have:
 - (a) Bachelor of Science degree in Engineering, Science, or Technology from an accredited college or university, or equivalent training plus experience;
 - (b) five years experience in or related to the field of educational major, of which at least two years will have been in quality engineering or technology;

- (c) a knowledge of at least two of the following areas of specialty: statistics/reliability, nondestructive examination, physical/mechanical properties measurement, metal fabrication, measurement technology, instrument and control fabrication and testing, chemical processing and analysis, failure analysis, and quality program development and implementation.
- Inspection and Test Unit Manager shall have:
 - (a) Ten years experience in a manufacturing industry, of which five years will have been in quality control/assurance;
 - (b) a general knowledge of manufacturing and inspection methods and techniques including dimension and electrical measurements, nondestructive examination, quality planning, and fabrication and assembly methods.

17.1.1.2 NECO Organization

The responsibility for NECO QA activities is assigned to the QA Manager who reports to the NECO President. The QA Manager is responsible for preparation and maintenance of the Project Quality Assurance Documents, and the coordination of all QA activities on the Project. The QA Manager has the authority and freedom to initiate the necessary procedures for quality-related activities, to identify quality problems, and to recommend corrective action.

The QA Manager does not have prime responsibility for schedule or cost, but does have the authority to stop work pending resolution of quality matters and the freedom to do the following:

- identify quality problems
- initiate, recommend, or provide solutions through designated management channels
- verify implementation of solutions
- control further processing, delivery, or installation of a nonconforming item, deficiency, or unsatisfactory condition until proper dispositioning has occurred

Overall contract responsibility for the facility design is assigned to the Project Manager. He provides the focal point for communications among the various disciplines and DOE. The QA Manager maintains communications on quality-related matters with DOE and provides advice of such discussions to the NECO Project Manager.

17.1.2 Quality Assurance Program

17.1.2.1 AI Quality Assurance Program

The quality assurance program described herein complies with the requirements of Reference 1 for the AI scope of work. The compliance of this FSF program to these requirements by invoking selected AI policies and procedures is shown in Appendix A to this Chapter.

This program is issued and made mandatory by direction of the President of Atomics International by Standard Operating Policies that require the issuance of operating procedures, and provides verification of their enforcement through a system of quality program audits. AI delegates execution responsibility of appropriate quality assurance program elements to suppliers of material, equipment, and services, but retains responsibilities for their implementation by these suppliers.

Management assessment of the scope and effectiveness of the quality assurance program is accomplished by two independent audits conducted by personnel outside the Quality Assurance Department. One of these is performed by an audit team assigned by the AI President, at yearly intervals, and specifically addresses the 10CFR50, Appendix B, requirements as they are implemented through that portion of the quality assurance program that addresses Section NA-4000 of Section III of the ASME Boiler and Pressure Vessel Code (Reference 2). The second audit occurs at 18-month intervals and is conducted by the office of the Rockwell International Corporation Director of Quality

Assurance. This latter audit is to assure compliance with contractual and statutory quality assurance requirements.

Continuing involvement of the AI President in quality assurance matters is achieved by three routinely scheduled interactive associations with the Quality Assurance Director. These are: (1) periodic staff meetings, during which each member of the President's staff, which includes the Quality Assurance Director, must report on significant problems, accomplishments, and status of activities; (2) periodic Executive Council Meeting, in which formal and in-depth reports are presented by Program Managers, and during which time the Quality Assurance Director addresses significant quality problems, with recommendations for corrective action; and, (3) submission of a monthly quality status report to Executive Management that covers quality progress, accomplishments, problems, and audit results, and to the customer as required by contract.

Quality policy originates at Rockwell International, with the issuance of a "Product Quality, Reliability, and Safety" policy statement, in which the President of the Corporation states: "It is the policy of the Corporation that its products will meet or exceed applicable standards and requirements for quality, reliability, and safety." The Senior Vice President, Corporate Staffs, issues a directive, applicable to all Division Presidents of the Corporation, which requires actions to be taken to implement this corporate policy, including the following:

- provisions for a competent and professional quality organization for each manufacturing activity, and reliability and product safety organizations at the appropriate levels
- evaluation of existing and new products, with respect to quality, reliability, and safety
- attainment of quality, reliability, and safety, through proper design, development, materials, and parts procurement, manufacturing, testing, product inspection, marketing, and servicing of the product

- timely and effective investigation and analysis of any accident claims and complaints, and maintenance of appropriate records of same
- utilization of proper descriptions of products, and their application in the following: (a) advertising and sales literature, (b) proposals and contracts, (c) service, instruction, and technical manuals, (d) product labeling, and (e) customer advisories

The Corporate Quality Policy is implemented at AI through Standard Operating Policies issued by the President - Atomics International Division. This Division policy states: "The Managers of Engineering, Material (Purchasing), Manufacturing, Quality Assurance, and Program Offices will be responsible for:

- 1) The preparation and issuance, in their operating manuals, of written instructions and procedures which establish the methods and responsibilities for performing quality-related activities and for verifying satisfactory performance of such activities.
- 2) The indoctrination and training of their personnel in these procedures, as applicable to their work assignments.
- 3) The instructions and procedures covering quality-related activities will meet the Quality Assurance Program requirements of the applicable government regulations and/or contract provisions.
- 4) Each individual is responsible for performing quality-related activities in accordance with the applicable instructions and procedures."

Based on the previously described quality policy, the department managers provide procedural coverage in their department manuals for quality-affecting activities.

The Quality Assurance Director has overall responsibility for assuring conformance to the procedures of the Quality Assurance Program Manual. He has the further responsibility, authority, and organizational freedom to stop nonconforming work, and to control further processing, fabrication, and delivery of nonconforming items. If differences of opinion occur that cannot be resolved, these are referred to the President of AI for final resolution. Changes to department manuals may be proposed by any individual or organization, but final review and approval rests with the department manager. Changes to AI Code Manuals (ASME Section III, ASME Section VIII, and the basic Quality Assurance Department Manual) receive final review and approval by the Quality Assurance Director.

Provisions for controlling the distribution of Department and ASME Code Manuals are addressed in each manual. These provisions provide for serialization of each manual in use, and for a record to be maintained of the recipients of each manual. Revisions of procedures in the manuals are distributed to each manual holder of record, along with an updated table of contents.

The FSF Quality Assurance Program Administrator identifies the procedures from Department and ASME Code Manuals that constitute the quality assurance program for the FSF scope of work. These procedures, along with other policies, definitions, and instructions for the preparation (including the QA and QC requirements of materials and components), distribution and control of these procedures and other documents such as drawings, specifications, etc., are compiled into the FSF Quality Assurance Program Index (QPI).⁽³⁾ This QPI and any subsequent changes thereto require the approval signature of the AI QA Director and the FSF Program Manager. Released copies are distributed to all cognizant FSF personnel and their managers to be used as the requirements for all engineering and procurement activities. The content of the FSF QPI is summarized in Appendix B.

Personnel performing quality-affecting functions receive formal training and indoctrination into the purpose, scope, and implementation of the policies and procedures in accordance with defined requirements. These personnel are

assigned to quality-affecting activities based on job requirements that specify minimum levels of education and experience in specific fields. Also, before new personnel are assigned to activities affecting quality, first-level managers conduct qualification interviews to assure adequate knowledge of the principles and techniques involved in these activities. Individuals are given yearly proficiency reviews by their managers.

Contractors for component designs and/or fabricated items are required to submit their quality assurance program descriptions for these items for AI review and approval. This review is made to assure that the suppliers' programs comply with AI requirements. Periodic audits of suppliers' Quality Assurance program activities are conducted by AI. The requirements for quality assurance program description submittal, and notification of the right of audit, are contained in quality assurance administrative requirements which are made part of each component contract.

In addition to the aforementioned training and indoctrination requirements specific to the conduct of FSF activities, selection of personnel performing quality-related activities is based on job descriptions for specific areas of responsibility which require minimum levels of education in specific fields and minimum levels of experience in related fields. Job assignments are made on the basis of these background criteria, as well as the judgement of the responsible manager as to each individual's ability to satisfactorily perform in that field. In the area of nondestructive examination, personnel are trained, qualified, and certified in accordance with American Society for Non-Destructive Testing Practice SNT-TC-1A. All personnel performing quality-related activities receive annual performance reviews.

17.1.2.2 NECO Quality Assurance Program

The quality assurance program implemented by NECO during the Title I and Title II design of the building site, structures, and utilities met the requirements of RDT F 2-2, defined in NECO's Quality Assurance Plans for the FSF. This document identifies, invokes, and describes the applicable sections of the

NECO Quality Assurance Manual for Engineering and Construction and the NECO Quality Assurance Procedures Manual. The former document defines policies and responsibilities, establishes minimum requirements, and provides guidance for implementing a comprehensive and total quality assurance program for engineering and construction of DOE facilities. The latter document contains detailed procedures to be implemented by personnel assigned to quality assurance responsibilities.

17.1.3 Design Control

17.1.3.1 AI Design Control

AI uses a Project Engineer for accuracy, completeness, and technical excellence of the Engineering work, and for coordinating the efforts of the engineering functional organization in the FSF program. This responsibility includes the coordination and integration of all activities related to systems requirements definition, system engineering, component design, interface control, change control, budget control within the Research and Engineering Department, and functional responsibility for achieving all Engineering milestones and schedule commitments. The Project Engineer is supported in this effort by Cognizant Engineers assigned within the Research and Engineering Department. Each of the Cognizant Engineers is assigned to a specific line organization, unit, and group, reflecting the particular discipline that represents his overall expertise. The Cognizant Engineer draws the necessary staff from his line reporting organization to perform the FSF design and engineering activities within the range of his particular discipline. He obtains support and interface information from any and all of the organizations within the Research and Engineering Department. Within their assigned disciplines, the Cognizant Engineers coordinate and prepare the technical reports and specifications necessary to complete the FSF overall design package.

The System Design Description (SDD)⁽⁴⁾ provides the principal means of design definition and control for the FSF. The SDD is used to define the various technical, operational, and safety considerations involved, to identify interfaces, and serves as the basic technical document for the system.

The control of the design interfaces is accomplished by means of the ICD's (Interface Control Drawing) and Component Specification. The ICD's and specifications identify the physical interface characteristics between systems and part of a system to assure overall compatibility.

The ICD's and specifications are controlled documents which are approved by the interfacing organizations and the FSF management and issued to affected functions. Changes to these require approval of the interfacing organization and the FSF management prior to being officially released.

Component specifications are prepared to define the requirements for the design, including interfaces, fabrication, quality assurance, testing, handling, and shipping of components and structures in compliance with the SDD and all approved baseline documents.

Engineering drawings are developed to meet the requirements of the SDD, approved baseline documents, and component specifications, and to further define and establish engineering parameters, characteristics, and design functions.

Each of these engineering documents is reviewed by Quality Assurance before release to ensure that the appropriate quality standards and requirements are specified. If deviations or changes from these quality standards are found in these documents, corrective actions are taken.

The Project Engineer is responsible to assure that suitable design controls are applied to such disciplines as seismic, stress, thermal hydraulic, radiation, and material compatibility. Design reviews are held at various design

milestones to verify design adequacy, and to insure that Design characteristics can be controlled, inspected, and tested; and that inspection and test criteria are identified.

These reviews are conducted by an Ad Hoc Design Review Board, consisting of a Chairman, Administrator and technical specialists for each of the disciplines involved in the design. These technical members must be qualified in their respective fields, but will not have been directly involved in the design being reviewed. The design in forms of layouts, schematics, block diagrams, specifications and the analyses supporting the design are reviewed in depth. All technical questions must be answered to the satisfaction of the respective technical member. Each question not resolved at the review is assigned as an Action Item to a member of the engineering team presenting the design. Minutes of the design review, including the summaries and listing of the Action Items, assignees, and the date for resolution, is prepared and formally released. Action Items must be cleared in writing to the satisfaction of the pertinent technical member. When all Action Items have been so resolved, the administrator reissues the Minutes indicating that all Action Items have been cleared with copies of all written responses to the Action Items attached. The design is approved when the final issue of the Design Review Minutes is published.

After the design approval by the Design Review Board, the drawings, specifications and other design documents are completed, checked, and prepared for the official release. The procedures for the release of these documents are defined in the Engineering Management Procedures (EMP). Approval signature requirements for the various documents are defined in the Release Plan of Action (RPA), which is prepared and issued for each program or project. The RPA is updated periodically as changes in the personnel responsible for the various functions occur.

Once initially issued, changes to the engineering documents can only be changed by the function which originated the document. The release of the changes require the same level of approval signature as the original release.

17.1.3.2. NECO Design Control

The Project Manager is responsible for identifying, to engineering discipline managers and the QA Manager, the technical requirements of the facility.

This identification is formal and documented. The distribution of this technical data constitutes the initiation of design activity for the facility. Modifications to this information, once it has been distributed, will occur only with the Project Manager's approval.

Upon receipt of this information, which identifies all the technical parameters of the facility, the engineering disciplines proceed to design the facility incorporating the necessary functional, safety, economic, and schedule requirements.

Specifications and design drawings, including interface drawings, are reviewed by the Quality Assurance group during the development of the specifications and drawings. Quality Assurance reviews are scheduled. The QA Group marks up a review set of specifications or prints indicating reasons for changes and returns them to the FSF Project Manager. The recommendations of the QA Manager are incorporated into the design under the direction of the Project Manager.

In addition to the scheduled review of specifications and drawings, the Quality Assurance Manager may schedule special meetings to address the requirements of critical systems, subsystems, or components.

The design control system administered by the Project Manager under the Project Work Plan is audited by the QA Manager.

All documents, after final release, are collected and maintained in a systematic and controlled manner.

17.1.4 Procurement Document Control

17.1.4.1 AI Procurement Document Control

AI uses a system of procedures that describe the sequence of actions to be taken in preparing, reviewing, approving, and controlling procurement documents. The basis for all procurement actions is the Purchase Requisition, which is prepared by the organization requesting the material, service, or component being purchased. The Purchase Requisition not only lists the hardware or services being requested, but also lists, defines, and schedules the submittal of documents to be provided as part of the procurement. These documents may include design drawings; design reports; manufacturing, inspection, and test plans; QC procedures; cleaning procedures, packaging and shipping plans; test and inspection reports; material certificates of compliance, etc. These documents are to be submitted either for approval or information, as defined in the Purchase Requisition. In the case of ASME-coded components, the code requirements for the maintenance of various records by the supplier are also noted. In addition, the surveillance and evaluation activities to be exercised by AI QA representatives are defined in the Purchase Requisition. The Purchase Requisition, including the drawings, specifications, design reports, and other documents incorporated or referenced therein, is reviewed and approved by FSF Program Manager and qualified Quality Assurance Department personnel to assure that correct complete quality requirements and the current issue of the applicable drawings and other documents are stated or referenced. The requirements of the Purchase Requisition are eventually transferred to a Purchase Order which is negotiated with the selected supplier. Purchase Orders are reviewed by Quality Assurance Department personnel to assure that the quality requirements are consistent with the original Purchase Requisition.

Purchase Orders for structures, systems, and components identify appropriate requirements, which must be addressed in the supplier's quality assurance program description. The supplier's program is reviewed by survey, and approved

by qualified Quality Assurance Department personnel, before start of activities affected by the Quality Assurance Program. Provision is made in the Purchase Order to ensure AI's right of access to the supplier's facilities and records for source surveillance and visits.

The Purchase Requisition - Purchase Order cycle described here is also used to process changes and revisions to the contract. The same review and approval is required of changes as that which is required of the original Purchase Requisition and Purchase Order. Procurement documents pertaining to spare or replacement parts are treated in the same manner as that used for initial production parts.

17.1.4.2 NECO Procurement Document Control

(Not applicable since NECO is not involved in any procurement.)

17.1.5 Instructions, Procedures, and Drawings

Both AI and NECO policy, procedure, and instruction documents prepared to cover management, design, engineering, procurement, quality assurance, and manufacturing, are described in Section 17.1.2, Quality Assurance Plans.

17.1.6 Document Control

17.1.6.1 AI Document Control

Documents, such as design specifications, design drawings, equipment specifications, and material processing specifications are prepared, reviewed, approved, issued and distributed in accordance with written procedures contained in the SOP and EMP. Changes to these documents must be processed similarly to the original documents before changes become effective.

Originals, prints, and/or reproducibles of these documents are controlled by the Engineering organization, which releases, distributes, stores, and maintains files and records of these documents. Document changes including obsolescence, are prepared, reviewed, approved, and distributed in accordance with applicable procedures, but only by or with the approval of the organization or function that prepared, reviewed, and approved the original. A list of the latest issues of drawings, specifications, and design-basis documents is also periodically issued by Engineering to all cognizant FSF program personnel to assure that only current issues are being used. Periodic audits are conducted to verify that active documents are in use, and obsolete issues have been removed from use.

AI Quality Assurance Manual, ASME Code Manuals, and department operating procedure manuals are prepared, approved, distributed and controlled in accordance with a procedure in each manual. AI manuals and procedures applicable to the FSF program are listed in the periodically updated FSF Quality Assurance Program Index (refer to 17.1.2.1). As noted in Appendix B, procedures pertaining to inspection, tests and nonconformances are included in the QPI.

Assurance that quality source surveillance is performed to the latest purchase order change is achieved through a system that routes purchase requisitions and orders and changes thereto to the QA Department Planning Function. At the time that a change is received by this organization, it is reviewed for quality requirements and the source surveillance instructions revised as necessary. Copies of revised instructions and the change orders are sent to the Procurement Quality Assurance Representative.

17.1.6.2 NECO Document Control

Document Control Procedure (QPI6) is implemented to achieve the following goals.

- 1) All documents are correct, complete, and clearly understandable.
- 2) Documents are issued only after adequate review and proper approval.
- 3) Comments are resolved and approvals are obtained from appropriate organizations.

Formal review of all documents is scheduled in the Project Work Plan. In order of sequence, the reviews are scheduled by the engineering discipline members, the QA Manager, and the Project Manager before a formal submittal to DOE.

Release of documents occur only at the direction of the Project Manager and only after completion of all of the formal reviews. Documents are distributed as noted in a distribution schedule.

The Project Manager maintains current schedules for all documentation including drawings and specification indices and schedules of other special studies and documents. These lists contain as a minimum the proper title, date of scheduled completion and revision, and date if currently released. These current schedules are distributed to all engineering discipline managers and the QA Manager.

17.1.7 Control of Purchased Materials, Equipment, and Services

17.1.7.1 AI Control of Purchased Materials, Equipment, and Services

The control of purchased items is based on placing purchase orders only with qualified sources, QA surveillance during the procurement, receiving inspection, and review of required vendor documentations.

Each potential supplier of materials, structures, systems, and components is evaluated to assess his capability to provide acceptable services and products. Evaluation of major-items suppliers is performed by a team, consisting

of representatives of Purchasing, the Program Office, Quality Assurance, and Manufacturing Departments. Representatives of Design Engineering, Materials, and Processes Engineering, and other units of the Engineering Department participate in the evaluation as necessary. The details of the evaluation include reviews of past performance, evaluation of procedures and capability descriptions provided by the supplier, surveys of the supplier's facility and Quality Assurance Program in operation, and/or experience of other AI procurement programs with the supplier. The evaluation considers the supplier's capability to supply a product which satisfies all requirements. Results of this evaluation are documented and retained on file at AI.

AI Quality Assurance Department personnel perform surveillance of suppliers during fabrication, processing, inspection, testing, and shipment of products. These surveillance activities are planned and performed in accordance with written procedures. The plans provide instructions that specify the characteristics or processes to be witnessed or verified, the documentation required, and the acceptance criteria that must be met. Sufficient surveillance is performed to verify that quality is achieved in items that cannot be inspected upon receipt. This surveillance ends with written approval given by appropriate Quality Assurance Department personnel to ship the item to the construction site.

Receiving inspection is performed on products delivered to AI to assure acceptability before use. This inspection is carried out in accordance with written inspection plans. The product is evaluated to determine that it is properly identified, that it meets inspection criteria, that necessary inspection and testing records are included with the product, and that the accepted product is identified as to its acceptability before being released for use or storage. Nonconforming items are segregated, controlled, and clearly identified pending proper disposition. AI Quality Assurance Department personnel provide data that is necessary for the recipient to plan and/or prepare procedures to handle, inspect, and store items purchased by AI and delivered directly to the construction site from the supplier.

AI requires that the supplier furnish, as a minimum, certifications that identify the product and the specific requirements met by the item. The supplier is further required to submit a report identifying any requirements which have not been met and indicating his disposition of such nonconformances.

Certifications and test reports are reviewed and approved by appropriate Quality Assurance Department personnel.

The same controls are used for spares and replacements that are used for initial FSF items.

Suppliers' certificates of conformance are validated by an established program of audits, surveillance, and overchecks. This is done by using itinerant or resident Quality Assurance site representatives or source inspectors, hold point release, and supplier audits. Additionally, fabrication specifications require supporting technical data for certificates of conformance, and this data is reviewed for completeness before use of any item.

17.1.7.2 NECO Control of Purchased Materials, Equipment, and Services

(Not applicable since NECO is not involved in the purchase of any materials, equipment, or services for the construction of FSF.)

17.1.8 Identification and Control of Materials, Parts, and Components

17.1.8.1 AI Identification and Control of Materials, Parts, and Components

Identification requirements are defined on the engineering drawing or fabrication specification for each part, subassembly, and assembly. These requirements include the location, method of identification (nameplates, impression stamp, electrochemical etch, etc.) and the identification data (part no., serial no., tag no., etc.). Detail parts and subassemblies making up the higher assemblies are called out by their identification numbers in the assembly drawing. This assures that proper parts and components are used in the completed assemblies.

For procured items, identification verification methods are included in the evaluation of the supplier's manufacturing and quality assurance procedures and processes. Particular attention is focused during the review on the proposed methods to control material, parts, and fabrication and inspection instructions during the total manufacturing process. This includes the physical control of the hardware, manufacturing, and inspection documents and records from the initial material procurement, through fabrication and inspection of individual parts, to the final assembly and test. Periodic in-plant surveillance checks are made to assure compliance with the submitted plans.

17.1.8.2 NECO Identification and Control of Materials, Parts, and Components

(Not applicable since NECO is not involved in the fabrication or procurement of any material, part, or component for FSF construction.)

17.1.9 Control of Special Processes

17.1.9.1 AI Control of Special Processes

Procedures, equipment, and personnel performing special processes on procured items are qualified or certified by the supplier or his subcontractor who performs the processes. Qualification is accomplished by satisfactorily completing the various performances defined in the check sheet prepared in accordance with requirements of the applicable codes, standards, specification or process. Qualification processes of individual procedures and equipment are witnessed and approved by the supplier's quality assurance.

Qualification records of procedures, equipment, and personnel for performing special processes are established, filed, and maintained current in compliance with supplier QA procedures. Periodic audits of these records are performed to ensure their adequacy.

The capability of the supplier to meet the above requirements is included in the supplier qualification review before placing the procurement. AI QA conducts onsite surveillance of the qualification procedures and review of the qualification records to verify supplier conformance to these requirements.

17.1.9.2 NECO Control of Special Processes

(Not applicable since NECO is not involved in the fabrication or procurement of any material, part, or component for FSF construction.)

17.1.10 Inspection

17.1.10.1 AI Inspection

For a purchased item, AI delegates the execution responsibility for inspection activities to the supplier, but only after AI has judged that the supplier's QA procedures are acceptable. This judgment is based on the initial vendor qualification review and the review of manufacturing, inspection, and test plans submitted by the supplier for the specific item. These plans describe all major activities and milestones, from the initial planning to the final delivery. Each inspection point, along with the inspection procedure or method and acceptance criteria, is also identified. Hold points for onsite witnessing by AI QA representatives, Code Inspectors, or DOE representatives are established and noted on the manufacturing and QA plans. AI surveillance of the QA processes and QA records is conducted to verify supplier compliance to applicable procedures. Supplier's Certificates of Compliance are reviewed before final acceptance, and filed for future reference.

17.1.10.2 NECO Inspection

(Not applicable since NECO is not involved in the fabrication or procurement of any material, part, or component for FSF construction.)

17.1.11 Test Control

17.1.11.1 AI Test Control

Procedures for test on procured components are prepared by the supplier and reviewed by AI for compliance to the drawing or specification requirements. The test procedures provide detailed description of the following as applicable:

- test setup, including the major test equipment
- test environments and acceptable limits
- test instruments and their calibration or qualification procedures and record
- qualification procedures or record of specialized skill of the test operator
- test parameters to be measured and recorded
- test acceptance criteria
- precautions and limits
- posttest disassembly procedures

The actual tests are conducted by the vendor under the surveillance of his Quality Assurance personnel. Critical tests are witnessed by AI Quality Assurance and/or Engineering personnel.

Test reports containing the actual test history and the test data are prepared by the supplier, approved by the Supplier's Quality Assurance Department, and submitted to AI. Test reports are reviewed by AI to verify compliance with the drawing and/or specification requirements and included in the final QA Data Package.

17.1.11.2 NECO Test Control

(Not applicable since NECO is not involved in the fabrication or procurement of any material, part, or component for FSF construction.)

17.1.12 Control of Measuring and Test Equipment

17.1.12.1 AI Control of Measuring and Test Equipment

Measuring and test equipment is identified, described, and the applicable qualification and/or calibration status is indicated in the test procedure prepared by the supplier of the procured item requiring tests. Assessment of the equipment capability to meet the test requirements is included in the AI review of the test procedure. The responsibility for the proper use of the indicated equipment, including accuracy of the instrumentation, is delegated to the supplier. Surveillance of test information and data, and the review of the test results and qualification and calibration records, are conducted by AI.

17.1.12.2 NECO Control of Measuring and Test Equipment

(Not applicable since NECO is not involved in the fabrication or procurement of any material, part, or component for FSF construction.)

17.1.13 Handling, Storage, and Shipping

17.1.13.1 AI Handling, Storage, and Shipping

For procured components, the basic handling, cleaning, storage, packaging, preparation for shipping, and shipping requirements are defined in the procurement package, either as part of the component specification or as fly-sheets. Suppliers are required to prepare and submit to AI the proposed procedures or instructions covering these operations. AI review of these procedures and instructions includes the assessment of the adequacy to protect the component from damage or deterioration during the fabrication, shipment to the delivery point, and storage before component installation or use.

The component supplier is responsible for the implementation of these procedures and instructions during all phases of the procurement.

Periodic AI surveillance onsite verifies compliance during the in-plant phase. Where necessary, the packaging and loading for shipment are witnessed before delivery.

For components requiring special handling at the delivery site and for installation, similar processes are carried out with rigging and installation contractors.

17.1.13.2 NECO Handling, Storage, and Shipping

(Not applicable since NECO is not involved in the fabrication or procurement of any material, part, or component for FSF construction.)

17.1.14 Inspection, Test, and Operating Status

17.1.14.1 AI Inspection, Test, and Operating Status

Each major supplier submits monthly progress reports indicating the work status in relation to the activities and/or milestones shown on the AI reviewed manufacturing, inspection, and test plan (refer to 17.1.10). The report also covers any actual or anticipated problems and the actual or recommended solutions. AI remains cognizant of the fabrication status by evaluation of these reports and by more frequent telephonic contact. Periodic onsite visits are made by AI procurement and quality assurance personnel. Special visits are made by engineering and other personnel as appropriate to resolve technical problems.

17.1.14.2 NECO Inspection, Test, and Operating Status

(Not applicable since NECO is not involved in the fabrication or procurement of any material, part, or component for FSF construction.)

17.1.15 Nonconforming Materials, Parts, or Components

17.1.15.1 AI Nonconforming Materials, Parts, or Components

The responsibility for resolving the nonconformance of materials, parts, or components is delegated to the supplier of the item. Normally, the nonconformance is cleared by appropriate supplier action (replace, repair, refabricate, etc.) The adequacy of the supplier's methods for nonconformances will have been established during the AI survey and evaluation of the supplier's QA system. Where the nonconformance does not adversely affect safety or function, a request for acceptance of the nonconformance conditions may be submitted to AI by the supplier on an AI Supplier Nonconformance Report. The nonconformance report is reviewed and evaluated by the AI FSF Material Review Board. Based upon the results of the technical evaluation, the board may accept or reject the request and process the request. In either case, the Supplier Nonconformance Report, after processing, is returned to the supplier, who is identified on the form. Copies of all Supplier Nonconformance Reports are included in the QA Data Package submitted by the supplier and included in the final data package for the particular component after AI review.

17.1.15.2 NECO Nonconforming Materials, Parts, or Components

(Not applicable since NECO is not involved in the fabrication or procurement of any material, part, or component for FSF construction.)

17.1.16 Corrective Action

17.1.16.1 AI Corrective Action

Corrective action for nonconformances to technical requirements and technical procedures is conducted within AI under the control of the Quality Assurance Department in accordance with established procedures (SOP M-48). Technical requirements are those contained in design drawings, specifications and

procurement documents. Technical procedures are those which guide the general process of documenting and disseminating technical information.

Corrective action is initiated because of product nonconformance or audit findings. The source of the nonconformance is established and the corrective action is assigned to the function responsible for the source of the nonconformance. The time period for instituting the corrective action is set, and progress is monitored by the Quality Assurance Department.

The corrective action is closed out when completion of the required action has been verified by QA. AI management is made aware of the initiation of the corrective action, conditions to be corrected and the implementation of the identified corrections.

The supplier corrective action methods are included in the AI survey and evaluation of the supplier's QA system and are judged to be adequate as is or with some modifications. In general, the corrective action methods must fulfill the following:

- identify the source of the nonconformance
- identify the corrective action required and the responsible function or person
- monitor and record the corrective action from initiation to closeout
- notify various management levels of the problem and solution

The adherence of the supplier to his established corrective action methods is verified by onsite surveillance of performance and evaluation of his records.

17.1.16.2 NECO Corrective Action

Corrective Actions involving technical procedures during the NECO Title I and Title II design are covered under 17.1.18.2, NECO Audits.

17.1.17 Quality Assurance Record

17.1.17.1 AI Quality Assurance Record

AI policies and procedures identify records which are maintained to provide documentary evidence of the quality of items and activities affecting quality. These procedures provide specific organizational responsibility for record collection and retention.

Originals of the AI engineering documents (drawings, specifications, technical reports, etc.), are retained until the termination of the affected program in a secured, fire-resistant vault in the Engineering Department. Subsequently, these documents are sent to central Rockwell storage. All documents required from the vendor as a line item of the purchase order are processed by Engineering, and a copy of each document is similarly stored. Microfilm records of all engineering documents are permanently stored in another secured, fire-resistant vault.

All procurement documents (initial procurement requests, purchase order change notices), and copies of all vendor-submitted documents are compiled into a case file and stored for 3 years in the secured Purchasing Department record room. Subsequently, these are also sent to central Rockwell storage.

All quality-related documents, whether AI-generated or supplier-furnished, are also compiled into a case file and retained for 3 years in the QA Department secured record room. Subsequently, these documents are also transferred to corporate storage. On the FSF program, all documents which make up the Data Package of each component are transmitted to the FSF Construction Manager. These include: original or copy of certification; inspection and test reports; other documents to verify that all requirements of the respective drawings and/or specifications have been met; and, other documents such as parts lists, operations and maintenance manuals, etc.

All vaults and record rooms indicated above are located within the fenced and secured area of the AI headquarters location in Canoga Park, California.

Central corporate storage is a secured, fire-resistant facility located in El Segundo, California. This facility is maintained on a 40-hour-per-week basis and is locked when not occupied.

17.1.17.2 NECO Quality Assurance Record

During the Title I and Title II design, records to furnish documentary evidence of quality assurance activities performed are prepared, collected, processed, analyzed, and distributed within NECO. These records are consistent with the quality requirements of applicable codes, specifications, and contract.

Quality assurance records include, but are not limited to, the following: drawings, specifications, engineering change orders and modifications, and other contract documents; quality assurance program manuals, program plans, procedures, and instructions; design analyses and stress reports of critical items; the results of reviews and audits; and, monitoring of work performance.

Records are identified, indexed, filed, and maintained in a manner which will permit prompt retrieval and review. Records provide sufficient information to allow positive identification between the record and the specific item or activity to which it applies.

Each organization responsible for maintaining records implements a records receiving and control system. The system applies to the receipt and control of records into a temporary working file as well as to the permanent storage file. As a minimum, the system includes:

- a records checklist which identifies the types of records and their retention periods
- an index of all records received and their disposition

- instructions for receipt, inspection, distribution, and disposition of all incoming records

Each control system is structured to permit a current and accurate assessment of the status of records and documents.

17.1.18 Audits

17.1.18.1 AI Audits

Quality assurance audits are conducted during the Title I and Title II design and long-lead equipment phases. These audits are conducted in accordance with established procedures, checklists, and schedules by personnel from the Quality Assurance Department; or, in the instance of team audits, they are conducted under the direction of an experienced auditor from the QA Department. Team audit personnel are selected to prevent their having direct responsibilities in the areas being audited.

Auditors document their findings, and these findings are reviewed with managers having responsibility for the area audited. At the time of this review, the affected manager accepts a commitment to implement corrective action for deficiencies and a specified date when implementation will be complete. Upon notification of completion of a corrective action commitment, that area is reaudited to assure the corrections have been accomplished.

Audits for the FSF Program are conducted of engineering, procurement, and quality assurance policies and procedures. These are first evaluated against code, standard, and contract requirements; and then the effectiveness of their implementation to ongoing work effort is established during audits. A review of documents and records is an integral part of all audits.

AI also audits suppliers of structures, systems, and components. Quality Assurance Department personnel perform audit planning, scheduling, audit team selection, audit coordination and contact, report issuance, and followup to verify implementation of effective corrective action. Audits are scheduled, but unscheduled audits may be performed when deemed necessary. Audits are scheduled based on supplier activity status. Audits are performed to evaluate the effectiveness of supplier Quality Assurance Programs. Checklists are prepared to guide the conduct of audits.

Each major supplier for the FSF program conducts planned and periodic audits to assure compliance to all aspects of and the effectiveness of their quality assurance program, previously reviewed and accepted by AI. The supplier audit is conducted in a manner similar to that conducted within AI. The supplier also conducts onsite audits of his subcontractors as defined in his Quality Assurance Program. The supplier submits the results of the audits to AI.

17.1.18.2 NECO Audits

A comprehensive system of planned and documented audits is implemented to verify compliance with all aspects of the quality program during the Title I and Title II Designs.

Audits are performed:

- to provide an objective evaluation of compliance with established requirements, methods, and procedures
- to assess progress in assigned tasks
- to verify implementation of recommended corrective actions

Audits include evaluation of work areas, activities, processes, and items as well as a review of documents and records.

Audits are performed in accordance with checklists for each activity or item being audited, and by appropriately trained personnel not having direct responsibility in the areas being audited.

During the audit, objective evidence is examined for compliance with the quality program requirements. Selected elements of the program are audited to the depth necessary to determine whether they are being performed effectively. When any nonconformance is found by an audit, further investigation is conducted in an effort to identify the basic cause of the nonconformance. Elements found to be nonconforming are acknowledged by a member of the audited organization.

An effort is made to clarify misunderstandings and to reach agreement on findings which constitute quality program nonconformances. A tentative course and schedule of corrective action for each nonconformance is established.

Quality management personnel will take necessary action to assure that any deficiency is corrected. Deficient areas will be reaudited as frequently as necessary until the required corrections have been accomplished.

The results of each audit will be documented, and reported to and reviewed by management. Summaries of audit findings with corrective actions will be prepared and submitted to top management, the Norman Engineering Company Quality Assurance Head, and the contracting officer.

17.2 CONSTRUCTION AND INSTALLATION

17.2.1 Scope

This section describes the quality assurance measures to be taken by J. A. Jones during construction and/or installation work on the FSF Project. Such work shall include construction of the RSBE/FSF facilities, including Building Utilities and Sodium System Installation.

Construction and major installation work will normally be performed by a fixed-price subcontract, and J. A. Jones will serve as the contractor performing construction management (CM).

Onsite CPAF* (J. A. Jones) or outside CPAF* or CPFF** contractors will be used to perform work as required. All work will be performed under J. A. Jones direction and control.

HEDL personnel will perform installation work during startup testing as required and as permitted by the Bacon-Davis Act.

Table 17.2-1 identifies the J. A. Jones documents by which the requirements of this section will be implemented. The J. A. Jones CM Plan and approved Standard Operating Procedures meeting the RDT F2-2 requirements shall be utilized.

17.2.2 Organization

The J. A. Jones QA Manager reports directly to the Vice President and Manager of J. A. Jones Hanford Operations. His department's responsibilities include quality engineering, quality control, inspection, auditing and records keeping.

*Cost plus award fee

**Cost plus fixed fee

TABLE 17.2-1

CM FSF QA PROGRAM REQUIREMENTS

<u>APPLICABLE F2-2 PARAGRAPH</u>	<u>PROGRAM ACTIVITY/ REQUIREMENT</u>	<u>REMARKS (REFERENCE DATA)</u>
2.0	<u>MANAGEMENT AND PLANNING</u>	"JA Jones CM Plan for FSF Construction" dated 5/27/79.
2.2	Quality Assurance Program	
2.3	Organization	JAJ CM Plan for FSF, Exhibit #1, "Organizational Matrix"
2.5	Audits and Reviews	JAJ Quality Assurance Procedure, JAJ/RL-QA-1261, "Audits"
2.6	Corrective Action	JAJ Quality Assurance Procedure, JAJ/RL-QA-1241, "Corrective Action"
2.7	Engineering Holds	Per design documents and refer- ence standards. FPSC shall define in QA Plan.
2.8	Unusual Occurrence	JAJ Engineering Procedure JAJ- RL-E-209, "Unusual Occurance Procedure" NOTE: Not Issued

NOTE: Unless otherwise specified, F2-2
paragraph references include all
subparagraphs.

TABLE 17.2-1 (cont'd)
CM FSF QA PROGRAM REQUIREMENTS

<u>APPLICABLE F2-2 PARAGRAPH</u>	<u>PROGRAM ACTIVITY/ REQUIREMENT</u>	<u>REMARKS (REFERENCE DATA)</u>
4.10	Control of Nonconforming	JAJ Quality Assurance Procedure #JAJ/RL-QA-1231, "Nonconformance Reporting" JAJ CM Plan, Exhibit IV
4.11	Control of Received Items	JAJ FSF CM Procedure #13, JAJ/RL-QA-1081
4.13	Alloy Verification	JAJ FSF CM Procedure # CM 1 and CM 12

NOTE: Unless otherwise specified, F2-2 paragraph references include all subparagraphs.

TABLE 17.2-1 (cont'd)

CM FSF QA PROGRAM REQUIREMENTS

<u>APPLICABLE F2-2 PARAGRAPH</u>	<u>PROGRAM ACTIVITY/ REQUIREMENT</u>	<u>REMARKS (REFERENCE DATA)</u>
5.0	<u>MANUFACTURING FABRICATION AND ASSEMBLY</u>	Requirements of the Subcontractor or Supplier
5.2	Planning	Requirements will be detailed in the Special Conditions of the Contract.
5.3	Inspection and Test Plan	JAJ FSF CM Plan Procedure # CM 1
5.4	Material Identification and Control	JAJ FSF CM Plan Procedure # CM 12
5.5	Control of Processes	Standard Operating Procedure JAJ/RL-E-206A
5.6	Inspection and Tests	FSF CM Plan and Standard Operating Procedure JAJ/RL-QA 1067 Procedure (CM) #13
5.7	Document Control	JAJ FSF CM Procedures CM 6 and CM 8
5.8	Measuring & Test Equipment Calibration and Control	Subcontractor's Approved QA Program - Title III Inspection Plan
5.10	Control of Nonconforming Items	JAJ FSF CM Plan Exhibit IV "Change Control Procedure and JAJ CM Procedure # CM 11

NOTE: Unless otherwise specified, F2-2 paragraph references include all subparagraphs.

TABLE 17.2-1 (cont'd)

CM FSF QA PROGRAM REQUIREMENTS

<u>APPLICABLE F2-2 PARAGRAPH</u>	<u>PROGRAM ACTIVITY/ REQUIREMENT</u>	<u>REMARKS (REFERENCE DATA)</u>
5.11	Corrective Action	Standard Operating Procedure JAJ/RL-QA-1241
5.12	Handling, Preservation, Packaging, Storage, and Shipping	JAJ FSF CM Plan - Operating Procedure #CM 13 - Standard Operating Procedure JAJ/RL- QA-1067
5.13	Quality Records	JAJ FSF CM 1 CM will require submittal data and shall perform in-process audits. JAJ FSF CM Plan and JAJ/RL-QA-1064
5.15	Alloy Verification	JAJ FSF CM 1 JAJ FSF CM 12 Title III Inspection Plan

NOTE: Unless otherwise specified, F2-2 paragraph references include all subparagraphs.

TABLE 17.2-1 (cont'd)
CM FSF QA PROGRAM REQUIREMENTS

APPLICABLE F2-2 PARAGRAPH	PROGRAM ACTIVITY/ REQUIREMENT	REMARKS (REFERENCE DATA)
6.0	<u>CONSTRUCTION AND INSTALLATION</u>	JAJ FSF CM Plan
6.2	Organization	Reference Exhibit I and II
6.3	Contract Review	Standard Operating Procedure JAJ/RL-P-600
6.4	Construction Planning	JAJ FSF CM Plan
6.5	Control of Configuration	JAJ FSF CM Plan
6.6	Construction Procurement	JAJ/RL-P-600 CM FSF Plan
6.7	Construction and Installation	JAJ FSF CM Plan
6.8	Inspection	JAJ FSF CM Plan
6.10	Quality Assurance Records	Reference 2.4 and 5.7 of this Table 16.6.1

NOTE: Unless otherwise specified, F2-2 paragraph references include all subparagraphs

NOTE: Submittal data issued by the contractor shall define his role associated with construction and installation.

J. A. Jones will contractually require any outside construction subcontractor working on the FSF to establish a quality assurance program in accordance with the requirements defined in the specifications, including RDT F2-2 where applicable.

17.2.3 Contract Reviews

Contractual and work order packages establish the requirements to be met by subcontractors or J. A. Jones CPAF forces, including needs for special controls, processes, equipment, and personnel training. The documents which comprise these packages are reviewed within J. A. Jones CM and FFTFPO prior to release for use. Orientation meetings are held with the performing company to assure full understanding of the job requirements and capability to meet these requirements before contracts are placed, before work begins, and periodically during the course of the work.

17.2.4 Construction Planning

J. A. Jones prepared a Project Plan (PP) which defines all work packages required, long-lead procurement items, planned start and completion dates for scheduled activities, and the need date for design documents to support construction, as well as definition of interface requirements. Controls to be applied during construction include development of detailed procurement, fabrication and construction/installation schedules, a CM Quality Assurance Program, and performance of periodic audits to assure compliance. Acceptance testing, inspection, and records keeping shall be per J. A. Jones CM Plan.

17.2.5 Control of Configuration

J. A. Jones maintains control over the configuration of construction and installed work in accordance with the Change Control Procedure described in the J. A. Jones CM Plan. Design/Field Changes are used to record all changes to approved design documents. Change data is transmitted to the subcontractor via modifications to the original contract. Revised drawings are not issued

to the subcontractor except through formal contract modification, and the Design/Field Changes are used for producing as-built documentation. The subcontractor is required to provide a set of as-built documents reflecting configuration changes. Travelers will be used where required, and J. A. Jones CM will assure that travelers are at the work location before work commences.

17.2.6 Construction Procurement

Engineered components are procured by AI and J. A. Jones as required. J. A. Jones procurements shall be made in accordance with approved Standard Operating Procedures as defined in the J. A. Jones CM Plan. Any revisions to these procedures are reviewed and approved by DOE. Any procurements made by outside subcontractors under contract to J. A. Jones will be required to provide similar procedures and schedules for J. A. Jones's review.

17.2.7 Construction and Installation Control

17.2.7.1 Material Handling and Cleaning Control

J. A. Jones has established Standard Operating Procedures, which will be implemented on the FSF work as applicable, for material identification and control, storage, and cleanliness control. Work performed by subcontractors shall be in accordance with requirements defined in the specifications and/or shipping and storage procedures. J. A. Jones CM shall perform audits to assure compliance to requirements.

17.2.7.2 Special Process Control

Welding and nondestructive testing processes shall be in accordance with specifications and approved submittal data requirements. J. A. Jones shall perform periodic audits to assure compliance.

17.2.7.3 Interface Control

J. A. Jones will be responsible for accomplishing the work described in 17.2.1 with minimal impact to the FFTF operations. Interface requirements will be controlled as per the interface matrix within the CM plan.

Construction schedules shall include definition of interface requirements, and task studies shall be procured to allow advance planning for interface requirements.

17.2.7.4 Measuring and Test Equipment

The fixed-price subcontractor or CPAF contractor will be responsible for controlling his measuring and test equipment per approved submittal data. Periodic audits shall be performed by J. A. Jones QA to assure compliance. The Government's Inspection Agency (Vitro) shall be responsible for witnessing tests and verifying operability of equipment.

17.2.7.5 Control of Nonconforming Items

J. A. Jones CM has primary responsibility for the control of nonconformances. The Government's Inspection Agency, as well as J. A. Jones, FFTFPO, HEDL, or the Subcontractor's QA organization can issue a nonconformance report which will be dispositioned by J. A. Jones CM in accordance with the Change Control Procedure defined in the CM Plan. Rework deficiencies shall be identified and corrected or repaired and reinspected or retested by the same methods as for the original test or inspection. Major nonconformances are documented and processed as Design/Field Changes.

17.2.7.6 Statistical Quality Control and Analysis

Not applicable to FSF work.

17.2.7.7. Indication of Status of Components and Systems

Subcontractors will be required to establish and implement systems for indicating status. These systems will be subject to J. A. Jones CM review and concurrence. Travelers will be used where required for "stamped" work in accordance with an approved Quality Assurance program.

17.2.8 Inspection

Systems have been established by FFTFPO and by J. A. Jones for the inspection of installation and construction work performed during the FSF construction. Vitro shall serve as the Government's Inspection Agency, and J. A. Jones CM will have audit authority over subcontract work, as well as approval authority over his QA Program. An Inspection Plan shall be produced by Vitro which includes hold points which may not be bypassed during construction. Subcontractors will be required to inspect their own work and to implement approved programs to assure quality. Travelers will be prepared when necessary.

17.2.8.1 Source Inspection

Source inspection shall be periodically performed by Vitro and/or J. A. Jones CM QA representatives. Surveillance shall include inspection for compliance with approved procedures and referenced standards. Inspection shall assure that a qualified QA Program can be and is being implemented and that hold points are being observed.

17.2.8.2 Receiving Inspection

Receiving inspection performed by J. A. Jones shall be in accordance with approved Standard Operation Procedures. Any outside subcontractor who procures materials with quality requirements will be required to establish receiving inspection systems subject to J. A. Jones CM and A/E approval. J. A. Jones CM shall perform periodic audits to assure compliance with

Operating Procedures. The Government's Inspection Agency shall verify product quality during construction and installation activities.

17.2.8.3 Installation Inspection

Inspection of construction and installation work will be in accordance with the Government's Inspection Agency's Inspection Plan. J. A. Jones CM shall perform periodic audits for compliance with Operating Procedures.

17.2.9 Testing, Operation, and Maintenance

J. A. Jones CM shall assist HEDL during startup operations as an advisor only. HEDL shall provide the necessary Quality Assurance Program.

17.2.10 Quality Assurance Program Records

The QA records, including those for purchased components and construction tests, will be maintained by responsible organizations as defined in their work statements. All QA documents will be transmitted to J. A. Jones FSF CM by respective organizations prior to termination of their respective representatives. J. A. Jones FSF CM shall prepare a turnover package which will be taken over by HEDL Information Services for long-term maintenance. The records are protected and indexed for ready retrievability. Records of acceptance (preoperational) tests are accumulated, filed, and indexed by the testing agency and will be turned over to Information Services upon completion of the test program. Radiographs are stored in the DOE records holding area which serves the entire Hanford complex. This area meets the requirements of Federal Property Management Requirements (FPMR) 101-11.5, with respect to fireproof construction, sprinkling, alarms, and humidity and temperature control. Ultimately, the radiographs are shipped to the Federal Records Center in Seattle, Washington, which also meets these requirements. Retention periods by HEDL will be in accordance with ERDAM 0230, Annex C-12 (Reference 5).

17.2.11 Quality Audits

Quality audits shall be conducted in accordance with approved Operating Procedures as defined in Table 17.2-1, item 2.5.

17.3 FACILITY OPERATION

17.3.1 Scope

This section describes the quality assurance program to be implemented by Westinghouse Hanford Company, the operating contractor for the Hanford Engineering Development Laboratory (HEDL) and for the Fuel Storage Facility (FSF) during the operation, maintenance and modification of this facility beginning with the acceptance of systems and facilities from the construction manager (J. A. Jones). The criteria for the quality assurance program are described and a broad overview of the method of implementation is provided. The documents by which the quality assurance program will be implemented, both by the HEDL Quality Assurance Department (HEDL QA) and by other HEDL organizations which perform or direct activities potentially affecting the safety and quality of the FSF, are listed in Reference 6.

17.3.2 Applicability

The QA program described in this chapter applies to all HEDL activities which could affect the safety and reliability of the FSF during its operational phase, including the methods by which HEDL assures that the quality-related activities of suppliers and contractor services to the FSF will be adequately controlled. The QA program applies to the entire FSF/FFTF.

17.3.3 Responsibility

HEDL QA (see Reference 6, Chapter 16.2) is responsible to develop and enforce a QA program which assures the safe and reliable operation of the FSF by effectively implementing the requirements of RDT F2-2 and other applicable codes and standards. This program is subject to the concurrence of DOE.

17.3.4 Relation to Other Requirements

The HEDL QA program is consistent with the eighteen criteria of Appendix B of 10CFR50 (see FFTF FSAR Chapter 16). Insofar as HEDL's work involves the design, procurement, fabrication, installation, and repair of pressure boundary equipment and systems, the QA program also satisfies the applicable requirements of the ASME Boiler and Pressure Vessel Code, as evidenced by appropriate ASME Certificates of Authorization.

17.3.5 Purchaser (Owner) Actions

The QA programs and quality-related activities of HEDL and HEDL contractors and suppliers are subject to review, concurrence, and audit by DOE or its designated representatives at all times and in all places where such activities are being formed. All records and information developed in such activities are available to DOE or its designated representative upon request.

17.3.6 Operation, Maintenance, and Modification

Several HEDL departments, including FFTF Project Engineering, Administration and Services, Quality Assurance, and Safety Analysis are involved in operation, maintenance, and modification of the FSF. For additional discussion see Chapter 16.7 of the FFTF FSAR. For acceptance testing, QA Program see Chapter 16.6.9 of the FFTF FSAR.

17.3.7 Procurement

Chapter 16.4 of the FFTF FSAR describes the QA program relative to procurement of replacement parts of the FSF.

17.3.8 Manufacturing, Fabrication, and Assembly

All manufacturing, fabrication, and assembly required to operate the FSF will be conducted under the QA guideline described in Chapter 16.5 of the FFTF FSAR.

17.3.9 Audits

Internal quality assurance audits covering FSF operation will be conducted under the guidelines of FFTF FSAR Chapter 16.8

17.4 REFERENCES

1. Code of Federal Regulations, 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants," USGPO, Washington, DC.
2. ASME Boiler and Pressure Vessel Code, The American Society of Mechanical Engineers, New York, NY.
3. FSF Quality Assurance Program Index, N106QPI530001, Rev. A, Atomics International, Canoga Park, CA, October 1977.
4. Fuel Storage Facility System Design Description, N106SDD210001, Rev. 1, Atomics International, Canoga Park, CA, April 1978.
5. ERDA Manual, Chap. 0230 Appendix, "Records Management Handbook," Annex C-12, US Department of Energy, Washington, DC, September 1976.
6. Fast Flux Test Facility Final Safety Analysis Report, HEDL-TI-75001, Chap. 6, Hanford Engineering Development Laboratory, Richland, WA, December 1975.

A P P E N D I X A

AI COMPLIANCE TO 10CFR50, APPENDIX 'B' CRITERIA

10 CFR 50 APPENDIX 'B' CRITERION	AI IMPLEMENTING DOCUMENT OR PROCEDURE	
	NUMBER	TITLE
I ORGANIZATION	SOP M-10	PROGRAM MANAGEMENT
	SOP M-12	PERFORMANCE MEASUREMENT SYSTEM (PMS)
	SOP M-48	CORRECTIVE ACTION SYSTEM
	SOP M-78	AI QUALITY ASSURANCE PROGRAM
	QAOP N1.21	QUALITY ASSURANCE PLANS
II QUALITY ASSURANCE PROGRAM	SOP A-01	AI POLICIES AND PROCEDURES
	SOP M-78	AI QUALITY ASSURANCE PROGRAM
	SOP N-50	QUALITY ASSURANCE (QA) - PROGRAM SUPPORT FUNCTIONS
	SOP N-52	QUALITY PROGRAM AUDITS
	SOP N-54	AI QUALITY RECORDS
	EMP 3-1	ENGINEERING DOCUMENTATION PROCESS
	AIMP 2.126.1	CASE FILE PREPARATION
	QAOP N1.00	PREFACE TO QUALITY ASSURANCE MANUAL
	QAOP N1.01	QUALITY ASSURANCE DEPARTMENT FUNCTIONS
	QAOP N1.03	VISION REQUIREMENTS FOR QUALITY ASSURANCE PERSONNEL
	QAOP N1.21	QUALITY ASSURANCE PLANS
	QAOP N1.23	QUALITY STATUS REPORTS
	QAOP N6.02	
	QAPP N7.1	QUALIFICATION AND CERTIFICATION OF NONDESTRUCTIVE EXAMINATION PERSONNEL
	QCPP U7.1	

10CFR50		AI IMPLEMENTING DOCUMENT OR PROCEDURE	
APPENDIX 'B' CRITERION		NUMBER	TITLE
II	QUALITY ASSURANCE PROGRAM (CONT.)	QAPP N-1.3	TRAINING
		QAPP N-14.1	RECORDS AND DATA REPORTS
		QAPP N-15.1	EXHIBIT INDEX
			STATEMENT OF AUTHORITY AND POLICY FOR ASME CODE SECTION III MANUAL
			PREFACE TO QUALITY ASSURANCE MANUAL FOR ASME CODE SECTION VIII MANUAL
III	DESIGN CONTROL	SOP M-20	CONFIGURATION SUMMARY REPORTS
		EMP INDEX	PREFACE AI ENGINEERING MANAGEMENT PROCEDURES MANUAL
		EMP 2-8	ENGINEERING STUDIES
		EMP 2-9	DESIGN AND ACCEPTANCE CRITERIA
		EMP 3-5	STANDARD RELEASE SYSTEM

10CFR50		AI IMPLEMENTING DOCUMENT OR PROCEDURE	
APPENDIX 'B' CRITERION		NUMBER	TITLE
III DESIGN CONTROL (CONT.)		EMP 3-24	CONTROL OF ENGINEERING DRAWINGS
		EMP 3-25	ENGINEERING ORDERS
		EMP 3-26	PREPARATION AND CONTROL OF SUPPORTING DOCUMENTS
		EMP 3-42	ENGINEERING MANAGEMENT SYSTEM FOR SPECIFICATIONS
		EMP 3-46	DOCUMENT RELEASE AND CONTROL SYSTEM
		EMP 3-48	MATERIAL PROCESSING PROCEDURES
		EMP 3-50	ENGINEERING HOLDS
		EMP 3-51	WELDMENT CHECK LIST
		EMP 3-52	ENGINEERING RELEASE PLAN OF ACTION
		EMP 3-54	PROGRAM DATA LISTS
		EMP 3-58	ENGINEERING WORK PLANS
		EMP 5-3	DESIGN REVIEWS
		EMP 5-17	CHECKING OF ENGINEERING DRAWINGS
		EMP 5-21	MATERIALS AND PROCESSES CONTROL SYSTEM
		EMP 5-24	APPLICATION OF STANDARDS
		QAPP N-2.1	DESIGN AND DOCUMENT CONTROL
		QCPP U-2.1	DESIGN DOCUMENTATION CONTROL

10 CFR 50 APPENDIX 'B' CRITERION		AI IMPLEMENTING DOCUMENT OR PROCEDURE	
		NUMBER	TITLE
IV	PROCUREMENT DOCUMENT CONTROL	SOP J-12	PREPARATION AND PROCESSING OF THE
		SOP J-13	PURCHASE REQUISITION
		CMP 1.1	REQUEST FOR WRITTEN QUOTATIONS &
		CMP 2.125	PROPOSALS FROM SUPPLIERS
		QAOP N4.00	PROCUREMENT POLICY
		QAPP N-3.2	CHANGES TO PURCHASE ORDERS AND OTHER
		QAPP N-3.7	DIRECTION TO SUPPLIERS
V	INSTRUCTIONS, PROCEDURES AND DRAWINGS	QCPP U-3.1	PROCUREMENT DOCUMENT REVIEW
			PROCUREMENT DOCUMENTS
			PROCUREMENT AS AN ENGINEERING
			ORGANIZATION
			PROCURED MATERIALS, PARTS, AND
			SERVICES
		SOP A-01	AI POLICIES AND PROCEDURES
		SOP M-78	AI QUALITY ASSURANCE PROGRAM
		SOP M-80	UNUSUAL OCCURENCE REPORTS - RRD
		SOP N-50	PROGRAMS
		SOP N-54	QUALITY ASSURANCE (QA) - PROGRAM
		EMP 2-9	SUPPORT FUNCTIONS
		EMP 3-1	AI QUALITY RECORDS
		EMP 3-5	DESIGN AND ACCEPTANCE CRITERIA
			ENGINEERING DOCUMENTATION PROCESS
			DRAWING PREPARATION - STANDARD
			RELEASE SYSTEM

10CFR50		AI IMPLEMENTING DOCUMENT OR PROCEDURE	
APPENDIX 'B' CRITERION		NUMBER	TITLE
V	INSTRUCTIONS, PROCEDURES AND DRAWINGS (CONT.)	EMP 3-9	PREPARATION AND CONTROL OF SPECIFICATIONS
		EMP 3-12	SYSTEM AND SUBSYSTEM SPECIFICATIONS
		EMP 3-17	COMPONENT SPECIFICATIONS
		EMP 3-18	COMPONENT TRACEABILITY
		EMP 3-34	SYSTEM DESIGN DESCRIPTIONS
		CMP 1.126.1	CASE FILE PREPARATION
		QAOP N1.21	QUALITY ASSURANCE PLANS
		QAOP N1.23	QUALITY STATUS REPORTS
		QAOP N6.02	QUALIFICATION AND CERTIFICATION OF NONDESTRUCTIVE EXAMINATION PERSONNEL
		QAPP N-7.1	
		QCPP U-7.1	
		QAPP N-11.1	AUTHORIZED INSPECTOR
		QAPP N-14.1	RECORDS AND DATA REPORTS
VI	DOCUMENT CONTROL	SOP J-12	PREPARATION AND PROCESSING OF THE PURCHASE REQUISITION
		EMP 3-9	PREPARATION AND CONTROL OF SPECIFICATIONS
		EMP 3-21	CHANGE CONTROL
		EMP 3-24	CONTROL OF ENGINEERING DRAWINGS

10 CFR 50 APPENDIX 'B' CRITERION	AI IMPLEMENTING DOCUMENT OR PROCEDURE	
	NUMBER	TITLE
VI DOCUMENT CONTROL (CONT.)	EMP 3-25	ENGINEERING ORDERS
	EMP 3-26	PREPARATION AND CONTROL OF SUPPORTING DOCUMENTS
	EMP 3-46	DOCUMENT RELEASE AND CONTROL SYSTEMS
	EMP 3-52	ENGINEERING RELEASE PLAN OF ACTION
	CMP 2.125	CHANGES TO PURCHASE ORDER AND OTHER DIRECTIONS TO SUPPLIERS
VII CONTROL OF PURCHASED MATERIAL, EQUIPMENT AND SERVICE	QAPP N-2.1	DESIGN AND DOCUMENT CONTROL
	QCPP U-2.1	DESIGN DOCUMENTATION CONTROL
	SOP J-58	RECEIVING AND INSPECTION OF INCOMING MATERIAL AND EQUIPMENT
	SOP J-59	WAREHOUSING OF DIRECT CHARGED PURCHASED MATERIALS BY TRAFFIC AND WAREHOUSING
	SOP J-60	HANDLING AND STORAGE OF PROJECT CRITICAL HARDWARE
	SOP L-40	PROCUREMENT AND CONTROL OF SUPPLIER DATA
	CMP 2.30	COMMITMENTS TO SINGLE, SOLE, PROPRIE- TARY AND MANDATORY SOURCE
	CMP 3.121	SOURCE SELECTION

10 CFR 50 APPENDIX 'B' CRITERION		AT IMPLEMENTING DOCUMENT OR PROCEDURE	
		NUMBER	TITLE
VII	CONTROL OF PURCHASED MATERIAL, EQUIPMENT AND SERVICE (CONT.)	CMP 4.103	SUBCONTRACTOR COST OR PRICING DATA
		CMP 4.7	CONSTRUCTION CONTRACTS
		CMP 4.9	ARCHITECTURAL & ENGINEERING SERVICES
		AIMP 2.202	REVIEW OF CASE FILE PRIOR TO MANAGEMENT APPROVAL
		AIMP 2.226	SELECTION OF CONTRACTORS BY BOARD PROCESS
		QAOP N4.01	SUPPLIER EVALUATION AND APPROVAL
		QAOP N4.02	PROCUREMENT QUALITY ASSURANCE PLANNING
		QAOP N4.03	PROCUREMENT QUALITY ASSURANCE - SOURCE INSPECTION
		QAOP N4.04	PROCUREMENT QUALITY ASSURANCE - RECEIVING INSPECTION
		QAPP N-3.1	APPROVED PROCUREMENT SOURCES
		QAPP N-3.2	PROCUREMENT DOCUMENTS
		QAPP N-3.3]	PROCUREMENT INSPECTION INSTRUCTIONS
		QCPP U-3.2]	
		QAPP N-3.4]	PROCUREMENT INSPECTION
		QCPP U-3.3]	
		QAPP N-3.5	IDENTIFICATION & CONTROL OF MATERIAL AND ITEMS
		QCPP U-3.4	

10CFR50		AT IMPLEMENTING DOCUMENT OR PROCEDURE	
APPENDIX 'B' CRITERION		NUMBER	TITLE
VIII IDENTIFICATION AND CONTROL OF MATERIALS, PARTS AND COMPONENTS		QAOP N4.02	PROCUREMENT QUALITY ASSURANCE - INSPECTION PLANNING
		QAOP N10.00	NONCONFORMING MATERIALS, PARTS, AND COMPONENTS
		QCPP U9.1	
		QAPP N-3.2	PROCUREMENT DOCUMENTS
		QAPP N-3.3]	PROCUREMENT INSPECTION INSTRUCTIONS
		QCPP U-3.2]	
		QAPP N-3.4]	PROCUREMENT INSPECTION
		QCPP U-3.3]	
IX CONTROL OF SPECIAL PROCESSES		EMP 5-21	MATERIALS AND PROCESSES CONTROL SYSTEM
		QAOP N6.02]	QUALIFICATION AND CERTIFICATION ON NONDESTRUCTIVE EXAMINATION PERSONNEL
		QAPP N-7.1]	
		QCPP U-7.1	
X INSPECTION		QAOP N1.21	QUALITY ASSURANCE PLANS
		QAOP N4.02	PROCUREMENT QUALITY ASSURANCE - INSPECTION PLANNING
		QAOP N4.03	PROCUREMENT QUALITY ASSURANCE - SURCE INSPECTION

10 CFR 50 APPENDIX 'B' CRITERION	AI IMPLEMENTING DOCUMENT OR PROCEDURE	
	NUMBER	TITLE
X INSPECTION (CONT.)	QAPP N-3.3 QAPP N-3.4 QCPP U-3.3]	PROCUREMENT INSPECTION INSTRUCTIONS PROCUREMENT INSPECTION
XI TEST CONTROL		NO TESTS SCHEDULED AT AI
XII CONTROL OF MEASURING AND TEST EQUIPMENT		NO FABRICATION AT AI
XIII HANDLING, STOR- AGE AND SHIPPING	SOP J-60	HANDLING AND STORAGE OF PROJECT CRITICAL HARDWARE
XIV INSPECTION, TEST AND OPERATING STATUS	SOP N-54 QAOP N1.21 QAPP N3.4] QCPP U3.3] QAPP N-11.1	QUALITY RECORDS QUALITY ASSURANCE PLANS PROCUREMENT INSPECTION AUTHORIZED INSPECTOR

10 CFR 50 APPENDIX 'B' CRITERION	AI IMPLEMENTING DOCUMENT OR PROCEDURE	
	NUMBER	TITLE
XV NONCONFORMING MATERIALS, PARTS OR COMPONENTS	SOP N-54 QAOP N10.00	AI QUALITY RECORDS NONCONFORMING MATERIALS, PARTS, AND COMPONENTS NONCONFORMING MATERIALS AND ITEMS RECORDS AND DATA REPORTS
	QAPP N-9.1 QAPP N-14.1 QCPP U-12.1	
XVI CORRECTIVE ACTION	SOP J-58	REVEIVING AND INSPECTION OF INCOMING MATERIAL AND EQUIPMENT CORRECTIVE ACTION SYSTEM UNUSUAL OCCURRENCE REPORTS - RRD PROGRAMS PROCUREMENT QUALITY ASSURANCE - SOURCE INSPECTION NONCONFORMING MATERIALS, PARTS, AND COMPONENTS CORRECTIVE ACTION FOR NONCONFORMING PRODUCTS CORRECTIVE ACTION
	SOP M-48 SOP M-80	
	QAOP N4.03	
	QAOP N10.00 QCPP U-9.1	
	QAOP N14.00	
	QAPP N-13.1	
XVII QUALITY ASSURANCE RECORDS	SOP N-54 EMP 3-1	AI QUALITY RECORDS ENGINEERING DOCUMENTATION PROCESS

10 CFR 50 APPENDIX 'B' CRITERION	AI IMPLEMENTING DOCUMENT OR PROCEDURE	
	NUMBER	TITLE .
XVII QUALITY ASSURANCE RECORDS (CONT.)	CMP 2.126.1 QAPP N-14.1 QCPP U-12.1	CASE FILE PREPARATION RECORDS AND DATA REPORTS
XVIII AUDITS	SOP N-52 QAOP N1.04 QAPP N-12.1	QUALITY PROGRAM AUDITS QUALITY ASSURANCE AUDITS AUDITS

A P P E N D I X B

SUMMARY DESCRIPTION OF AI POLICIES AND PROCEDURES APPLICABLE TO
THE FSF PROGRAM (FROM REFERENCES)

A. STANDARD OPERATING POLICIES (SOPs)

SOP A-01 AI Policy and Procedures Manuals

This SOP defines the types of AI administrative policies and procedures authorized, and establishes minimum format and distribution requirements for such policies and procedures.

It identifies the highest level of management, corporate or otherwise, responsible for establishing Quality policies, goals, and objectives. A clear path of communication between Quality Assurance organization and Corporate management is defined.

Positions and groups responsible for defining both content and changes to the Quality Assurance Program and manuals are identified, in addition to the management level responsible for the approval of the Quality Assurance Program and manuals. Provisions are established for controlling and distribution of Quality Assurance manuals and revisions.

SOP J-12 Preparation and Processing of the Purchase Requisition

This SOP establishes methods and policies applicable to the preparation and processing of the Purchase Requisition (Form N25-R-2). The requisition is used for authorizing procurement, through Purchasing, of materials, equipment, and services from suppliers.

Procedures are established that delineate the sequence of actions to be accomplished in preparation, review, approval, and control of the Purchase Requisition.

SOP J-59 Warehousing of Direct-Charged Purchased Materials by Traffic and Warehousing

Methods are specified to identify and control materials. Verification of correct identification of material, prior to release, is required. Material shall be protected against loss, damage, and deterioration from environmental conditions.

SOP J-60 Handling and Storage of Project Critical Hardware

Special handling, preservation, storage, packaging, and shipping requirements are specified, and performed by qualified personnel, under predetermined instructions.

SOP J-58 Receiving and Inspection of Incoming Material and Equipment

Receiving inspection of supplier-furnished material and equipment is performed in accordance with the following. The material is properly identified, and corresponds with receiving documentation. Inspection is performed and judged acceptable, in accordance with predetermined instructions, prior to use. Items accepted and released are identified as to their inspection status, prior to release. Nonconforming items are segregated, controlled, and identified until proper disposition is made.

SOP L-40 Procurement and Control of Supplier Data

Procedures are established for preparation, review, and control of instructions, procedures, drawings, and changes thereto. These documents and changes thereto are procedurally controlled to assure adequacy. Provisions are established, identifying the personnel responsible for these activities. Changes are reviewed by the same organizations that performed the original review, unless delegated by the applicant to qualified responsible organizations. Approved changes are promptly included in the appropriate documents.

SOP M-10 Program Management

This SOP sets forth principles and guidelines for the management of Atomics International Business Programs. The Guidelines include organizational framework, program management processes, performance monitoring, and reporting systems.

SOP M-12 Performance Measurement System (PMS)

This SOP identifies the AI Performance Measurement System (PMS) and establishes responsibilities for implementing and maintaining the system on AI programs.

The basic purpose of PMS is to provide traceable and integrated cost, schedule, and work status information to the customer and AI management as a basis for making decisions to guide the future course of a program.

PMS consists of several subsystems such as: planning (work definition, work breakdown structure, scheduling and budgeting), work authorization, performance reporting, variance analysis, corrective action, and performance baseline change control.

SOP M-20 Configuration Summary Reports

This SOP establishes the policies, methods, and responsibilities for the preparation, issuance, and use of Configuration Summary Reports.

The primary purpose of this report is to aid the Manufacturing, Quality Assurance, and Engineering functions in determining configuration and effectivity requirements for product hardware.

SOP M-48 Corrective Action System

This SOP establishes methods for assuring that action is taken for correction of conditions causing nonconformances to technical requirements and technical procedures.

Evaluations of nonconformances, and determination of the need for corrective action, follow established procedures. Prompt corrective action is initiated,

following the determination of nonconformance to procedural or technical requirements. Adverse conditions significant to quality, their causes, and corrective actions, are reported to the appropriate levels of management.

SOP M-78 AI Quality Assurance Program

This procedure defines the Quality Assurance Program to be applied to all AI products and services, in compliance with applicable contract, federal, or state requirements. Management (above or outside of Quality Assurance, and to the highest Corporate level) regularly assesses the Quality Assurance Program effectiveness. The establishment of indoctrination and training progress review is specified.

SOP M-80 Unusual Occurrence Reports - RRD Programs

This SOP establishes methods and responsibilities for reporting to the customer of unusual occurrences affecting AI programs under the cognizance of the Division of Reactor Research and Development of ERDA. It implements the requirements of RDT Standard F 1-3T.

SOP N-50 Quality Assurance Program Support Functions

This procedure establishes policy on the utilization of AI Quality Assurance Department functions on AI programs, and describes the Quality Assurance Department functions and interfaces with other AI departments.

It summarizes the provisions for resolving disputes arising from a difference of opinion between Quality Assurance-Quality Control and other department personnel.

The procedure outlines the safety-related structures, systems, and components controlled by the Quality Assurance Program, and the respective organization executing Quality Assurance-related functions on these items during the design, engineering, procurement, inspection, manufacturing, construction, and testing phases. Quality-related activities (inspection and test, etc.) performed, with appropriate equipment and under suitable environmental conditions, are described.

SOP N-52 Quality Program Audits

Procedures and responsibilities for assuring the adequacy and effectiveness of the AI Quality Assurance Program, through audits of procedures, standards, methods, and practices used in producing AI hardware or software products, are established by this SOP.

Audits are performed in accordance with pre-established written procedures or checklists, and are conducted by trained personnel not having direct responsibilities in the areas being audited. The audits include an objective evaluation of quality-related practices, procedures, and instructions, and the effectiveness of implementation and conformance with policy directives.

Audit data are analyzed, and reports indicating quality trends and the effectiveness of the Quality Assurance Program are provided to management. The audit results are documented, and then reviewed with management having responsibility in the area audited. Subsequently, responsible management takes the necessary action to correct the deficiencies revealed by audit.

SOP N-54 Quality Records

AI Quality Records are defined, and responsibilities for their retention are established, by this SOP. Its purpose is to establish standards for meeting AI and customer contract requirements for filing, storing, and retrieving of quality history information on AI products and services.

Quality Assurance records include: (1) operating logs; (2) results of reviews, inspections, tests, audits, and material analyses; (3) monitoring of work performance; (4) qualification of personnel, procedures, and equipment; and (5) other documentation, such as drawings, specifications, procurement documents, calibration procedures and reports, and nonconforming and corrective action reports. The records are to be readily identifiable and retrievable.

Requirements and responsibilities for record transmittals, retention and maintenance subject to work completion must be consistent with applicable codes, standards, and procurement documents.

Record storage facilities are to be constructed, located, and secured to prevent loss or destruction of the records, or their deterioration by environmental conditions.

B. QUALITY ASSURANCE MANUALS* PROCEDURES

QAOP N1.00 Preface to Quality Assurance Manual

The preface to each Quality Assurance Manual delineates the purpose and authority of the manual.

QAOP N1.01 Quality Assurance Department Functions

This document outlines the functions of the individual groups within the Quality Assurance Department.

QAOP N1.03 Vision Requirements for Quality Assurance Personnel

This procedure establishes vision standards for Quality Assurance Department personnel, and defines responsibilities for administering an eye examination program.

QAOP N1.04, QAPP N-12.1 Quality Assurance Audits

These procedures outline the Quality Assurance responsibilities for implementing and maintaining an audit program to determine the overall effectiveness of the AI and supplier quality programs, and to identify areas where corrective prevention action is required.

QAOP N1.05 Qualification of Audit Personnel

This procedure defines AI's system for qualifying and documenting the qualification of lead auditors.

QAOP N1.21 Quality Assurance Plans

This procedure defines Quality Assurance Department responsibilities for participating in the preparation of Quality Assurance Program Plans or Quality Assurance Program Indexes, and for preparing Quality Assurance Functional Plans.

QAOP N1.23 Quality Status Reports

This procedure establishes Quality Assurance Department requirements and responsibilities for preparation of periodic Quality Assurance Program Status Reports, and for submittal of the reports to Atomic International customers.

*Atomic International Quality Assurance Department Procedures (QAOP)
Atomic International ASME Code Section III Manual (QAPP)
Atomic International ASME Code Section VIII Manual (QCPP)

QAOP N4.00, QAPP N-3.2, QCFP U-3.1 Procurement Documents

These procedures define requirements and responsibilities for preparation, review, and approval of procurement documents associated with the purchase of materials, parts, and services.

QAOP N4.01, QAPP N-3.1 Approved Procurement Sources

These procedures define Quality Assurance Department requirements for evaluation and approval of procurement sources (suppliers) of material, parts, and services used in AI products.

QAOP N4.02, QAPP N-3.3, QCFP U-3.2 Procurement Inspection
QAPP N 3.4, QCFP U-3.3 Instructions

These procedures define Quality Assurance Department requirements and responsibilities for preparing inspection instructions applicable to procured items and services.

QAOP N4.03 Procurement Quality Assurance - Source Inspection

This procedure defines Quality Assurance Department requirements and responsibilities for inspection of procured items and services at a supplier's facility.

QAOP N4.04 Procurement Quality Assurance - Receiving Inspection

These procedures define Quality Assurance Department requirements and responsibilities for inspecting and testing incoming procured items and services.

QAOP N6.02, QAPP N-7.1, QCFP U-7.1 Qualification and Certification
of Nondestructive Examination
Personnel

These procedures establish requirements and responsibilities for the training, examination, qualification, and certification of Atomic International personnel engaged in the following nondestructive examination processes on Code materials and items:

Radiographic
Magnetic Particle
Ultrasonic

Liquid Penetrant
Eddy Current
Helium Leak

QAOP N10.00, QAPP N-9.1, QCPP U-9.1 Nonconforming Materials,
Parts, and Components

These procedures define requirements for the review, control, and disposition of nonconforming materials, parts, and components. Nonconforming materials, parts, and components are defined as those having any characteristic which deviates from requirements of drawings, specifications, codes, standards, or other applicable documents.

QAPP N-13.1 Corrective Action for Nonconformance
Products

These procedures establish requirements for taking action to correct conditions causing nonconforming material, parts, and components. Its purpose is to provide increased assurance that AI products will meet design, configuration, and performance requirements.

QAPP N-1.3 Training

This procedure defines requirements and responsibilities for training and indoctrination of personnel performing activities affecting quality or Code compliance, as necessary, to assure that suitable proficiency is achieved and maintained.

QAPP N-2.1, QCPP U-2.1 Design Documentation Control

These procedures establish the requirements and responsibilities as an Owner's Agent, and for the control of design activities and documents associated with items being constructed in accordance with the requirements of the Code.

QAPP N-3.5, QCPP U-3.4 Identification and Control of Materials and
Items

These procedures define requirements for identifying and controlling materials and parts that are required to comply with Code requirements.

QAPP N-3.7 Procurement as an Engineering Organization

This procedure defines the requirements and responsibilities when Atomics International, as an Engineering Organization, assumes responsibilities, in addition to the design, for purchasing and contracting activities. It establishes quality assurance interfaces with the Manufacturer, Material Manufacturer, Material Supplier, or Installer involved in the activities of the Engineering Organization.

QAPP N-14.1, QCPP U-12.1 Records and Data Reports

These procedures define requirements and responsibilities for accumulating records generated during design and/or fabrication of Code items at Atomics International, transmitting records to the Owner or Customer, and retention of records by Atomics International.

QAPP N-15.1, QCPP U-14.1 Exhibits

These exhibits contain samples of forms and documents described in the QAPP and QCPP Procedures.

QAPP N-11.1, Authorized Inspector

These procedures define Atomics International requirements and responsibilities for assisting the Authorized Inspector in performing his duties, in accordance with Code requirements.

C. ENGINEERING MANAGEMENT PROCEDURES (EMP's)

EMP INDEX Preface to the Engineering Management
Procedures Manual

This procedure describes the scope of the Engineering Management Procedures (EMP) Manual.

EMP 2-8 Engineering Studies

This procedure establishes the requirement for conducting studies to establish that the design meets the design criteria, is based upon proven practices or analysis, and is adequate for the intended service. It describes the method for preparing, releasing, and controlling Engineering Studies.

EMP 2-9 Design and Acceptance Criteria

This procedure delineates the need for design and acceptance criteria to be defined and published in the appropriate design basis documents.

EMP 3-1 Engineering Documentation Process

This procedure describes the scope of procedures which control the preparation, release, and control of specifications, drawings, and reports by Engineering.

EMP 3-5 Standard Release System

This procedure provides instructions for the preparation, numbering, release, and control of drawings for the Standard Release System, and provides guidelines for application of the standard release.

EMP 3-9 Preparation and Control of Specifications

This procedure describes the format and methods of preparation, numbering, and control of specifications.

EMP 3-12 System and Subsystem Specifications

This procedure gives the format for System and Subsystem Specifications.

EMP 3-17 Component Specifications

This procedure gives the format for Component Specifications.

EMP 3-18 Material Specifications

This procedure gives the format for Material Specifications.

EMP 3-19 Process Specifications

This procedure establishes the format for Process Specifications.

EMP 3-21 Change Control

This procedure defines the method for requesting, evaluating, approving, and executing engineering changes.

EMP 3-24 Control of Engineering Drawings

This procedure describes the methods for control of drawing originals and prints, released by both the Standard or Limited Release Systems.

EMP 3-25 Engineering Orders

This procedure describes the preparation and use of an Engineering Order to release drawings or specifications, and defines requirements.

EMP 3-26 Preparation and Control of Supporting Documents

This procedure establishes the types of supporting documents, and defines the requirements for their preparation, release, and change.

EMP 3-34 System Design Descriptions

This procedure provides instructions for the implementation of RDT F1-2 for preparation, use, maintenance, and control of System Design Descriptions at Atomics International.

EMP 3-44 Nuclear Item Design Specifications

This procedure establishes the format for a Nuclear Item Design Specification and contains a description of the type of information to be included within each section of the specification.

EMP 3-46 Document Release and Control Systems

This procedure describes the Engineering Order Release System, the Supporting Document Release System, and the Technical Data Transmittal System.

EMP 3-47 Material Processing Specifications

This procedure gives the format for Material Processing Specifications.

EMP 3-48 Material Processing Procedures

This procedure gives the format for Material Processing Procedures.

EMP 3-51 Weldment Checklist

This procedure provides the checklist to be completed for critical weldments, and the system for its implementation.

EMP 3-52 Engineering Release Plan of Action

This procedure gives the format and requirements for a plan describing the means of preparation and release and approval of program documents.

EMP 3-54 Program Data Lists

This procedure gives the requirements for preparation and maintenance of Program Data Lists used as a means of controlling and scheduling program documentation.

EMP 3-58 Engineering Work Plans

This procedure describes the responsibilities and format for preparation of an Engineering Work Plan. The Engineering Work Plan documents and schedules the planned engineering work assignments to be performed in order to fulfill the requirements of the work package.

EMP 5-3 Design Reviews

This procedure establishes the requirements for independent design reviews, and the means of their scheduling, conduct, and reporting.

EMP 5-17 Checking of Engineering Drawings

This procedure establishes the responsibilities for checking of all engineering drawings.

EMP 5-21 Materials and Processes Control System

This procedure establishes the policy and responsibilities for control of materials and processes.

EMP 5-24 Application of Standards

This procedure provides guidance and direction for the application of codes and standards. It categorizes various types of standards, and establishes responsibilities for their collection and application.

D. MATERIAL DIVISION (PURCHASING) MANUAL PROCEDURES (MD:

CMP 1.1 Procurement Policy

This procedure describes the procurement policy of Rockwell International, and supplements it to cover procurement reflecting ERDA requirements.

AI MP 2.125 Changes to Purchase Orders and Other Directions
to Suppliers

This procedure establishes standards for accomplishing changes to purchase orders, and effecting other direction to suppliers.

CMP 2.126.1 Case File Preparation

This procedure establishes the documentation required to be accumulated in procurement case files.

CMP 2.30 Commitments to Single, Sole, Proprietary, and
Mandatory Sources

This procedure establishes controls over procurement from single, sole, proprietary, and mandatory sources.

CMP 3.121 Source Selection

This procedure defines Rockwell International's practice concerning selection of procurement sources and making commitments.

CMP 4.103 Subcontractor Cost or Pricing Data

This procedure provides instruction for obtaining, evaluating and documenting supplier cost or pricing data.

CMP 4.7 Construction Contracts

This procedure establishes guidelines for preparation and use of construction contracts.

CMP 4.9 Architectural and Engineering Services

This procedure establishes the guidelines for the procurement of architectural and engineering services.

AI MP 2.202 Review of Case File Prior to Management Approval

This procedure establishes the requirements for pre-approval review by a Cost Analyst and Contract Analyst of procurement contracts and related documentation for compliance with contractual, procedural, and policy requirements.

AI MP 2.226 Selection of Contractors by Board Process

This procedure defines Atomics International's practice for the selection of contractors by Selection Board process.

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