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**FFTF REFERENCE CONCEPT
SUMMARY DESCRIPTION**

JANUARY 1969

**AEC RESEARCH &
DEVELOPMENT REPORT**

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JANUARY 1969

COMPILED BY

R. C. Walker
R.C. Walker

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W. B. McDonald
W.B. McDonald

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THIS REPORT WAS PREPARED BY
THE FFTF INTEGRATED REACTOR CONCEPT TASK FORCE
W. B. McDONALD, BATTELLE-NORTHWEST, CHAIRMAN

DURING THE DEVELOPMENT OF THIS CONCEPT, MAJOR CONTRIBUTIONS WERE PROVIDED BY WESTINGHOUSE ELECTRIC CORPORATION, ADVANCED REACTORS DIVISION, AND ATOMICS INTERNATIONAL, A DIVISION OF NORTH AMERICAN ROCKWELL CORPORATION. THIS REPORT WAS INITIALLY PREPARED SPECIFICALLY FOR PRESENTATION TO THE ATOMIC ENERGY COMMISSION, DIVISION OF REACTOR DEVELOPMENT AND TECHNOLOGY, ON DECEMBER 17-18, 1968. SUBSEQUENTLY, THE REPORT HAS BEEN RE-EDITED FOR IMPROVED CLARITY AND CONTINUITY.

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SECTION 1.0 INTRODUCTION

During meetings at AEC-Headquarters on July 17-18, 1968^{1,2}, a plan of action for completion of the FFTF concept selection was developed. Basically, this plan provided that, by January 1, 1969, BNW would achieve RDT concurrence on an integrated reference concept. It was agreed that, during the interim, Westinghouse would participate in development of the concept in such a manner that they would be in basic agreement with the concept recommended to RDT.

Subsequent to the above mentioned meeting, Westinghouse as well as Atomics International personnel have been integrated into the BNW task force activities and both organizations have made substantial contributions to the evaluation of alternative concepts considered as well as to the reference FFTF concept presented herein.

A presentation of this concept was made to RDT on December 17-18, 1968 at which time BNW judged the concept to be responsive to the overall objectives, requirements and guidelines established as bases for concept selection and recommended that the basic reactor plant concept presented herein be adopted as the basis for completion of related Conceptual Design Descriptions as well as for initiation of Preliminary Design³. Westinghouse and Atomics International concurred with this recommendation.

Subsequent to this presentation, RDT advised that they are in agreement with this recommendation⁴.

-
1. Letter, M. Shaw to D. G. Williams, Transmittal of Conference Report, dated August 6, 1968. (RDT:RE:D:681)
 2. Letter, M. Shaw to D. G. Williams, Transmittal of Conference Report, dated October 2, 1968. (RDT:RE:D:768)
 3. Letter, E. R. Astley to J. M. Shivley, Transmittal of Summary Concept Description, dated December 15, 1968. (PNL/FFTF/681573)
 4. Letter, M. Shaw to D. G. Williams, FFTF Integrated Reactor Concept Selection, dated December 19, 1968. (PNL/FFTF/68723)

SECTION 2.0 PRINCIPAL OBJECTIVES, REQUIREMENTS AND GUIDELINES

Subsequent to the July 17-18, 1968 BNW/W/RDT meeting at AEC Headquarters^{1,2}, BNW compiled a summary of principal FFTF Objectives and Requirements. This summary did not attempt to restate all of the detailed requirements which appear in Conceptual Design Description documents. The intent was to select and compile those objectives and requirements which were expected to be most significant with respect to concept selection.

During the BNW/W/RDT conferences held at Waltz Mill Site on August 27-30, 1968, this summary was reviewed in detail³ and, with the inclusion of a few additional comments subsequently submitted by RDT⁴, was established as a firm basis for evaluation of alternate concepts.

As a result of three subsequent BNW/W/RDT conceptual design review meetings^{5,6,7} (last two attended by AI), several guidelines were established which were intended to supplement or clarify the previously agreed to "Principal Objectives and Requirements". The following represents the current list of objectives, requirements and guidelines.

2.1 DEFINITIONS

- . Objective - A concise statement of a proposed accomplishment that is specifically related to the needs of the FFTF Program.
- . Functional Requirements - A qualitative statement of what will constitute minimum compliance with an objective.

1. Letter, M. Shaw to D.G. Williams, Transmittal of Conference Report, dated August 6, 1968 (RDT:RE:D:681).
2. Letter, M. Shaw to D.G. Williams, Transmittal of Conference Report, dated October 2, 1968 (RDT:RE:D:768).
3. Letter, E.R. Astley to J.M. Shivley, Transmittal of Conference Report, dated September 5, 1968 (PNL/FFTF/68865).
4. Letter, E.E. Kintner to Manager, RL, PNL/W/RDT Conference Report Agreements and Conclusions, dated October 22, 1968 (RDT:RE:D:799).
5. Letter, M. Shaw to D.G. Williams, Meetings on FFTF Conceptual Design Work, AEC Headquarters, September 20-21, 1968, dated September 27, 1968 (RDT:RE:050).
6. Letter, M. Shaw to D.G. Williams, FFTF Reactor Conceptual Selection Meeting of October 26-28, 1968, dated November 25, 1968 (RDT:RE:D:863).
7. Letter, E.E. Kintner to Manager, RL, FFTF Reactor Conceptual Design Review Meeting of November 25 and 26, 1968, dated December 11, 1968 (RDT:RE:H:463).

- . Specific Requirement - A quantitative statement of what will constitute minimum compliance with an objective.
- . Concept Development Guideline - A clarification, amplification, or modification of an objective or requirement.

2.2 OBJECTIVES

- A. An adequately controlled and instrumented environment in a high fast flux, representative of LMFBR reactors, for testing instrumented fuel specimens, fuel rods, fuel subassemblies, and cladding material.
- B. Capabilities to test fuel up to and including failure (short of planned meltdown) in dynamic sodium; however, the design shall provide for containing inadvertent meltdown of the experiment.
- C. Reliable plant performance (high and predictable plant factors).
- D. Means to test fuel for short time periods.
- E. Interim non-destructive fuel examination capability.
- F. Utilization of existing technology wherever possible.
- G. A facility servicing all U.S. fast flux requirements.
- H. When not in conflict with other objectives, to provide experience in the design, fabrication, testing and operation of sodium components and systems and fuel handling, which is applicable to the needs of the LMFBR Program.

2.3 FUNCTIONAL REQUIREMENTS

- A. Closed loops shall accommodate test specimens which extend into the space above and below the core.
- B. An axial positioner shall be available for use with selected tests in a single closed loop, and another positioner for a single open test position shall be available to permit precisely controlled "steady-state" irradiations, cyclic testing, and transient testing.
- C. The future use of alternative (non-sodium) coolants in a single closed loop and a single package loop shall not be precluded.
- D. Closed loops shall be accessible and replaceable.
- E. The design shall provide for the installation of capsules and package loops in alternate positions with a high degree of flexibility.

- F. The FTR shall be capable of accommodating short-term test facilities, as required for various programmatic purposes, in the first row reflector and in-core peripheral positions.
- G. Flux available for closed loop testing shall be the highest attainable, consistent with testing allocations, reactor safeguards, driver fuel and total core power limitations.
- H. Control and safety margins shall be provided by movable poison elements to meet all operating, shutdown, and accident requirements.
- I. A primary shutdown and secondary or backup reactivity shutdown system shall be provided based on operational principles which differ from each other as much as possible.
- J. Containment shall be provided for a DBA consisting of a prompt nuclear excursion leading to a core disassembly. The design shall be consistent with utilization of a two-barrier containment system. The design shall include features to protect the public by mitigating the effects of a shutdown DBA (during refueling) or by incorporating demonstrably positive means of preventing such an accident. (NOTE: Modified by 2.5 A and B).
- K. A sufficient number of control rods or other positive means shall be incorporated to ensure sufficient negative reactivity margins during refueling.
- L. Reactor core components shall be designed for accessibility for maintenance, surveillance, in-place testing and replaceability.
- M. The minimum number of driver fuel assemblies in the core shall be dependent upon reactor safeguards single-channel meltdown criteria. The number of fuel assemblies shall be selected on the basis of first core, as well as advanced core, requirements.
- N. Driver fuel and nuclear control assemblies shall be designed to provide redundant restraint against accidental movement from their core positions. At least one of the hold-down restraints shall not require the actuation of mechanical devices.

- O. The design shall provide for maximum flexibility in the interchangeability of driver fuel and closed loop positions (NOTE: Modified by 2.5 J).
- P. The reactor vessel, exclusive of the cover, shall be designed to accept a backup core without structural modification.
- Q. The fuel handling system shall be designed to be operated in a remote mode. (NOTE: Modified by 2.5 C and T).
- R. The fuel handling system shall be designed to preserve the metallurgical, mechanical and chemical conditions acquired in the reactor for the experimental specimens.
- S. The fuel handling system shall be capable of handling closed-loop test specimens that have been tested to planned failure, with proper safeguards for personnel, equipment and facility (NOTE: Modified by 2.5 X).
- T. The fuel handling system shall have the capability of placing, removing and replacing core assemblies, including test assemblies, in a pre-selected rotational orientation.
- U. The fuel handling system shall be designed to change out driver fuel and experimental specimens at a rate that supports high plant availability.
- V. Remotely controlled equipment shall be provided for performing the interim and final examination operations on fuels, materials and components irradiated in the FTR.

2.4 SPECIFIC REQUIREMENTS

- A. The core shall provide the capability for the installation of at least six closed loops, at least three of which shall be plumb vertical (NOTE: Modified by 2.5 N).
- B. Each closed loop position shall have the capability to accept a closed loop test specimen with a major cross-sectional dimension of at least 2.5 inches.
- C. 1. For the purpose of evaluating core performance and control considerations, the design shall provide for the installation of at least twelve open test positions within the core (NOTE: Modified by 2.5 Z).

2. The design shall provide flexibility for positions with driver fuel positions or first row reflector positions
(NOTE: Modified by 2.5 J).
- D. The initial core shall provide a peak flux (ϕ) of at least 7×10^{15} n/cm²-sec to at least one closed loop position. The ultimate design goal for advanced cores is a peak flux equal to or greater than 10^{16} n/cm²-sec.
- E. The facility shall be designed for a plant availability goal of 75%.
- F. Each driver fuel subassembly shall be instrumented for bulk coolant outlet temperature and channel flow. Flow instrumentation shall provide clear threshold indication of at least 60% normal flow (NOTE: Modified by 2.5 K and Q).
- G. Instrumentation shall be provided for each driver fuel position to promptly detect a fuel pin failure and to locate the affected subassembly (NOTE: Modified by 2.5 K).
- H. Each closed loop, package loop, capsule and at least six open loops to be instrumented for inlet¹ and outlet coolant temperatures, coolant flow and prompt fuel failure detection (NOTE: Modified by 2.5 K).
- J. All core assemblies shall conform to a uniform core lattice dimension and geometry.
- K. The reactor vessel and heat transport system shall be designed for maximum steady-state operating conditions of 1200 °F bulk reactor vessel outlet coolant temperature and 900 °F bulk reactor vessel inlet coolant temperature.
- L. The vessel and cover shall be designed for a service life of 20 years including all predictable operating cycles. Vessel components and internals for which a 20-year life expectancy cannot be ensured, shall be designed to permit replacement without structural cutting and welding (NOTE: Modified by 2.5 Y).

1. Bulk reactor inlet temperature for open test positions.

- M. The fuel handling system shall be designed to maintain the driver fuel cladding surface temperature below 1000 °F. (1500 °F under emergency conditions).
- N. The fuel handling system shall be capable of handling core assemblies, including experiments, 24 hours after reactor shutdown (NOTE: Modified by 2.5 G, P, R, S and V).
- O. The Interim Examination facilities shall be capable of performing the operations listed in Table 2-I.
- P. Capabilities, systems, components and features of the FFTF shall be designed in such a manner that the Project can be completed within authorized capital cost.
- Q. Design of the FFTF shall be in such a manner as to attain safe, predictable reactor criticality by late 1973 - early 1974, including allowances for required development programs.

2.5 CONCEPT DEVELOPMENT GUIDELINES

- A. The possibility of a Design Basis Accident during refueling will not be assumed to be a design restraint.
- B. The concept studies must provide means for accommodating the consequences of expulsion of sodium from the reactor vessel as a result of the operating DBA.
- C. Design concepts which are based on use of a sealed hot cell (one in which all operations for the life of the reactor are assumed to be carried out remotely in an inert atmosphere) are not acceptable.
- D. A routine capability must be provided for contact maintenance in an air atmosphere in the reactor head area during shutdown.
- E. The concept design should place no reliance on use of space suits for operations or maintenance.
- F. The concept design approach may assume that limited, pre-selected top head activities can be accomplished in an inert atmosphere. The specific operations to be accomplished in this mode and major pieces of equipment required should be identified.

- G. Direct fuel withdrawal shall be utilized for refueling. Fuel handling machines which are gas cooled will be utilized in the designs unless concept studies indicate that this mode of cooling is unworkable. Fully shielded machines may be considered. The use of fuel manipulation devices such as pantographs are not acceptable (NOTE: Modified by 2.5 V and W).
- H. The reactor head plug shall be fixed.
- I. Either diverged conical cores or skewed cores are mechanically feasible and manufacturable.
- J. Concept design efforts shall have flexibility in locating open and closed loops within the core.
- K. Assumptions of the vertical core instrumentation team¹ regarding instrumentation size and number shall be utilized for the remainder of the concept development work. Instrumentation which requires the following features is not acceptable:
 - 1. Significant deflection or flexing of instrument leads within the bayonet assembly.
 - 2. Driver fuel instrumentation sensors so located as to sense an environment which is nonrepresentative of average fuel assembly flow and outlet temperature during all normal reactor operating conditions.
 - 3. Instrumentation sensors which are not located in fixed transverse relationship to the subassembly coolant outlet region during all normal reactor operations.
 - 4. Mechanical design of bayonet assembly does not adequately provide for flow induced vibration or for rigors of installation and removal.
- L. No allowance for a separate Emergency Core Cooling System (ECCS) space envelope need be included in the concept provided that a sufficient sodium head is allowed for in the reactor vessel, consistent with overall ECCS requirements.

1. Informal Report, N. E. Todreas, et al., Instrumentation Capabilities of Vertical Cores, September 1968, ICD 18.

- M. An acceptable design approach relative to stainless steel swelling phenomena is to proceed initially on the basis of a nominal swelling allowance and shorter initial core life if further data dictate.
- N. The driver fuel, open and closed loops will be vertical.
- O. The overall fuel subassembly length shall not exceed 15 feet.
- P. Driver fuel movement will be made using three major ports.
- Q. The refueling port plug which supports the rotating instrument plate shall be designed for removal prior to refueling.
- R. The in-vessel driver refueling machine shall be designed to be removed from the reactor prior to reactor operation.
- S. Transfer of driver and test fuel from the reactor shall be performed using a shielded cask operating in air.
- T. Routine manned access to the reactor head area shall be possible immediately after shutdown and during refueling.
- U. Closed loop branch arm piping shall be routed off the reactor between the operating floor and the primary reactor pressure boundary.
- V. Provision of a capability for in-vessel decay storage is considered to have substantial merit and will be investigated during the current concept development period.
- W. Direct gas cooling shall not be used for transfer of driver or test fuel from the reactor.
- X. Routine capability shall be provided for direct refueling of closed loops without loop inpile tube section removal from the reactor.
- Y. BNW/W agreements (Ref. MMR8, dated November 20, 1968) on fluence limits (based on 10% total elongation at end of life) will be used as the basis for long life reactor structural components design.
- Z. Six closed loops and at least two contact instrumented, full subassembly size open loops will be provided by the design.

TABLE 2-I. INERT GAS CELL EXAMINATION FACILITY OPERATIONS

<u>Operation</u>	<u>Limited Interim Examination^(a)</u>	<u>Extended Interim Examination^(b)</u>	<u>Final Non-Destructive Examination</u>
Duct removal	X	X	X
Disassembly		X	X
Reassembly		X	
Visual inspection	X	X	X
Photography	X	X	X
Overall dimensional measurement	X	X	X
Interpin spacing measurement	X	X	X
Length, warp and cross- section measurement of pins, specimens and hardware		X	X
Fuel pin defecting		X	
Neutron radiography	X	X	X
Gamma autoradiography	X	X	X
Gamma scanning	X	X	X
Cleaning of fuel and materials test sub- assemblies	X	X	X

- a. Limited Interim Examination Operations are performed on those selected fuels materials and components irradiated in the FTR whose decay heat output, inspection requirements and design permit inspection and return to the reactor to be completed during a scheduled reactor outage.
- b. Extended Interim Examination Operations are performed on those fuels, materials and components irradiated in the FTR whose decay heat output, of one or more reactor operating cycles before return to the FTR. Replacement specimens can include the original specimens, nonirradiated specimens or specimens previously irradiated at the FTR or other sites.

TABLE 2-I (continued)

<u>Operation</u>	<u>Limited Interim Examination^(a)</u>	<u>Extended Interim Examination^(b)</u>	<u>Final Non-Destructive Examination</u>
Instrumentation tests	X	X	X
Instrumentation repair		X	
Replace duct	X	X	
Weighing of capsules, pins and specimens		X	X
Leak testing		X	X
Bond testing		X	X
Fission gas collection and measurement		X	X
Repressurizing and sealing after fission gas collection	X		
Milling and cutting of capsules, hardware and irradiated components			X
Preparation of specimens for shipment			X

-
- a. Limited Interim Examination Operations are performed on those selected fuels, materials and components irradiated in the FTR whose decay heat output, inspection requirements and design permit inspection and return to the reactor to be completed during a scheduled reactor outage.
- b. Extended Interim Examination Operations are performed on those fuels, materials and components irradiated in the FTR whose decay heat output, inspection requirements and design require the additional processing time of one or more reactor operating cycles before return to the FTR. Replacement specimens can include the original specimens, nonirradiated specimens or specimens previously irradiated at the FTR or other sites.

SECTION 3.0 SUMMARY CONCEPT DESCRIPTION

The FFTF Concept is a fixed reactor head, vertical core design with individual nozzle access primary testing facilities. The design features location of closed loops and open test positions along three near-radii of the core at 120-degree intervals. Driver fuel assemblies are installed and removed utilizing three major circular ports located in the trisected regions between the test facility rows. The reactor is located below the operating floor level within a containment structure which contains an air atmosphere in normal operating areas. An air-filled machinery dome is installed over the reactor to provide missile protection and protect the final containment system from the consequences of a major sodium-air reaction.

The reactor core is made up of hexagonal assemblies in a 91-site array. The driver fuel assemblies feature mixed $\text{PuO}_2\text{-UO}_2$ fuel cladding in 0.230-in. diameter, Type 316 stainless steel arranged in a 217-pin array. The driver fuel assemblies are 14 ft long including approximately two feet of length to accommodate fuel pins which require fission gas plenums longer than those of initial core driver fuel. Approximately three scram-assisted safety rods and six scram-assisted control rods are located within the core. These are supplemented by approximately 15 gravity-drop control rods located in the first of the three reflector rows. All control rods utilize enriched B_4C as the neutron absorber. All control rods in the concept are driven by inline, top mounted drives.

Six sodium cooled closed loops and two contact instrumented, subassembly size, open test positions are located along the three test position radii. Further study will be necessary to ensure means for accommodating the trail cable type short term test facility. An axial positioner is provided for selected tests in an open test position as well as one axial positioner for a closed loop. Two closed loops are located at essentially the core center, one closed loop and two open test positions are located at approximately core mid-radius and three closed loops are positioned at the core periphery. Closed loop

positions can accommodate open tests or driver fuel and additional flexibility for relocation of closed loops into open test positions is provided.

In the event that driver fuel is located in any of these test positions, instrumentation and backup holddown is provided utilizing a special assembly inserted through the test position nozzle. One special proximity-instrumented open test position is provided at essentially the core center. This position can also be utilized for a variety of special test assemblies such as reduced diameter open loops, capsules or materials tests.

All driver fuel positions are instrumented for outlet temperature, flow and failure location utilizing an instrument plate assembly which also provides a mechanical backup to the full hydraulic balance holddown. The instrument plate assembly in the reference concept arrangement also is utilized to radially position driver fuel assemblies prior to actuation of the radial core restraint system. Instrumentation package sensors (thermocouples and flowmeter) are located in the instrument plate; these instruments are installed in a module which can be replaced from the top of the support column for the instrument plate. Passive instrumentation components (flow conditioning components for temperature, flow and FEDAL measurements) are also located in the instrument plate.

The instrument plate assembly is supported by a rotating plug located in the approximately 4-ft diameter refueling port for each reactor sector. For the reference concept arrangement, the rotating plug also contains the in-vessel driver fuel handling machine. The lower end of the instrumentation plate support column is guided in a socket which also serves as the connection plug for the sodium sampling FEDAL system.

Driver refueling is accomplished by use of a fixed-length arm grapple assembly positioned by means of rotation of a plug located in the refueling port. This fuel handling machine transfers driver fuel under sodium to finned pots located at the core elevation but radially outside the core. In the reference reactor arrangement, in-vessel storage

of driver fuel is provided utilizing a lazy susan. Driver fuel is transferred into and out of the reactor vessel through transfer ports utilizing the finned pots and a shielded ex-vessel fuel handling machine.

Refueling of closed loop and open test facilities is accomplished by means of a shielded, sodium cooled, fuel handling machine to ensure safe transfer of experiments without damage and to ensure maximized plant availability due to test facility refueling.

The concept employs a three loop primary heat transport system with hot leg pumps to supply coolant to the reactor. The reactor vessel may be either of the piped downcomer or annular design; this concept feature is not yet established.

SECTION 4.0 SUMMARY OF PRINCIPAL CONCEPT
CHARACTERISTICS & DATA

<u>A. General Plant Data</u>	<u>Units</u>	<u>Values</u>	<u>Status*</u>
1. Core arrangement		vertical	1
2. Design life	years	20	1
3. Peak flux	n/cm ² /sec	7.3×10^{15}	2
4. Total power	MWt	400	1
5. Reactor coolant		sodium	1
6. Core volume	liters	1025	2
7. Reactor coolant flow	lb/hr	1.5×10^7	2
8. Reactor pressure drop			
design maximum	psi	100	2
9. Reactor bulk inlet temp.			
initial core	°F	600	2
design maximum	°F	900	1
10. Reactor bulk outlet temp.			
initial core	°F	900	2
design maximum	°F	1200	1
11. Core temperature rise avg.			
initial	°F	300	2
design maximum	°F	400	1
12. Reactor cover gas		Argon	1
<u>B. Reactor Vessel</u>			
1. Diameter	feet	~18	2
2. Height	feet	~51	2
3. Wall thickness	in	2	2
4. Wall fluence, total	avt	10^{21}	2
5. Material		304 SS	1
<u>C. Core Design</u>			
1. Number of core lattice posi- tions		91	1
2. Number of driver fuel sub- assemblies		73	2

* 1. Firm requirement or definite concept characteristic
2. Current reference design value

	<u>Units</u>	<u>Values</u>	<u>Status</u>
3. Number of closed loops		6	2
4. Number of contact instrumented in-core open test positions		2	2
5. Number of proximity instrumented open test positions		1	2
6. Number of in-core safety rods		3	2
7. Number of in-core control rods		6	2
8. Number of peripheral control rods		15	2
9. Equivalent core diameter	inches	~47	2
10. Active core height	inches	36	2
11. Reflector material		nickel	1
12. Reflector thickness			
axial	inches	6	1
radial	rows	3	2
13. Fuel pin heat transfer area	sq. ft.	2,800	2
14. Core coolant velocity, maximum	ft/sec	30	2
15. Direction of coolant flow		upward	1

D. Driver Fuel

1. Fuel composition		20-30 vol % PuO ₂ 70-80 vol % UO ₂	2
2. Cladding material		316 SS	1
3. Linear heat generation rate, average	kW/ft	7.85	2
4. Overpower factor		1.25	2
5. Engineering hot channel factor		1.13 to 1.25	2
6. Peak linear heat generation at overpower	kW/ft	18.0	2
7. Target burnup, average	MWD/Tonne (metal)	45,000	2
8. Cladding heat transfer coefficient	Btu/hr-°F-ft ²	40,000	2

	<u>Units</u>	<u>Values</u>	<u>Status</u>
9. Max fuel temperature	°F	4,200	2
10. Fuel assembly length	ft	14	2
11. Fuel geometry		Hexagonal pin cluster	1
12. Pin diameter	inches	0.230	2
13. Spacer wire diameter	inches	0.056	2
14. Number of pins per assembly		217	1
15. Subassembly cross-section outside dimension (across flats)	inches	4.615	2
16. Lattice spacing	inches	4.715	2
17. Duct wall thickness	inches	0.140	2

E. Physics Data

1. Delayed neutron fraction		0.003	2
2. Neutron lifetime	seconds	3.5×10^{-7}	2
3. Doppler	$T(\frac{dk}{dt})$	-0.004	2
4. Power density	MW/liter	0.4	2
5. Power distribution (peak/average)			
radial		1.35	2
axial		1.24	2
total		1.68	2

F. Test Facilities

1. Closed loops			
- number		6	2
- location		2 adjacent to core center	2
		1 @ mid-radius	2
		3 @ core periphery	2

	<u>Units</u>	<u>Values</u>	<u>Status</u>
- power handling capability	MW	4	1
- test flow rate	gal/min	30-300	1
- test section outlet temp.	°F	1,400 (bypass flow permitted)	1
- test section length	inches	36	2
- test section diameter	inches	2.5	1
- pumping head-primary	lb/in ²	250	2
- test section pressure drop, max.	lb/in ²	90	2
- material (in-core tube)		316 SS	1
2. In-core open test positions			
- number			
with contact instrumentation		2	2
with proximity instrumentation ^(a)		1	2
- power		}	Same as driver fuel subassembly
- coolant flow rate			
- test assembly length			
- test assembly cross section			
- coolant			
3. Short term irradiation facility			
- type		trail cable	1
- number		1	2
- location		core periphery	1
- minimum irradiation time at constant flux	minutes	1	2
- sample length	inches	24	2
- sample cross section, maximum	inches	2.0	2

(a) One proximity instrumented test position will have superior instrumentation accessibility. Driver fuel positions can be used as open test positions with standard driver instrumentation.

	<u>Units</u>	<u>Values</u>	<u>Status</u>
<u>G. Heat Transport System</u>			
1. Primary loops			
- number		3	1
- primary loop material		304 SS	1
- primary loop flow (per loop)	lb/hr	5.0×10^6	2
2. Primary pumps			
- number		3	1
- design pump head	feet	500	2
- net positive suction head	feet	43	2
- design temperature	°F	1,050 ^(a)	1
- speed control		wound rotor motor with liquid rheo- stat	1
- motor power			
brake	HP	1,665	2
rated	HP	3,000	2
3. Intermediate heat exchangers			
- number		3	1
- type		vertical shell & tube	1
- LMTD			
initial	°F	75	2
ultimate	°F	100	2
- capacity			
initial	MW	133	2
ultimate	MW	177	2
<u>H. Shielding</u>			
1. Within reactor vessel			
- material		stainless steel	2

(a) Pump tank will be designed for 1200 °F

	<u>Units</u>	<u>Values</u>	<u>Status</u>
- configuration		hexagonal assem- blies	2
- thickness	inches	24 (5 concentric rows)	2
2. Cover shield			
- material		Low alloy steel	2
- thickness	feet	4	2
- coolant		Argon, helium or Na	2

SECTION 5.0 CONCEPT FEATURE DESCRIPTION

5.1 REACTOR CORE

5.1.1 Core Arrangement

5.1.1.1 Description

The reactor core arrangement for this concept is composed of a 91 element vertical hexagonal array. The test positions are located in a "Y" shaped array with the center of the "Y" at the center of the core and the arms extending to the center of the flats of the overall pattern. The tri-sected design was selected primarily for compatibility with the refueling approach which utilizes offset refueling arms each of which services a one-third (120-degree segment) section of the core. With this type of refueling scheme, it is desirable to "clear" the region between the top of the fuel ducts and the bottom of the reactor top cover of all obstructions within a section being refueled so that refueling equipment will have free access to the 120-degree segment being serviced. Since it is desirable to be able to leave the open test positions and closed loops in place during refueling, the test positions should be placed along the radial divisions separating the three refueling zones.

Another consideration in determining the placement of the test positions is the routing of the closed-loop piping away from the upper nozzles. The ideal routing is a straight corridor extending radially outward from the center of the reactor. To prevent interferences with the in-core control rods and the closed and open test nozzles, it was found through studies of various alternative layouts that the most efficient arrangement placed the closed loops on one side of the corridor (actually one row off of a true radial pattern) with the piping corridor extending down one side of the test positions. This arrangement leaves a driver position in the center of the core.

With this type of arrangement, each piping trench will accommodate one closed loop located at the edge of the core with piping running radially outward and one closed loop position located either near the center of the core or in a mid-core position with the piping extending along one

side of the outboard positions. An individual corridor will not accept two closed loops at inner positions unless branch arm piping is stacked because of the width of the closed loop chaseway.

The requirement for in-core and peripheral controls are covered in Section 5.3, the main requirement from a core arrangement standpoint is to maintain adequate control and a symmetrical geometry. Because of the requirement for nine in-core control positions with eight inch drive mechanism envelopes and the prior test loop position arrangement, control rod positions appear to be restricted to those shown on SK-3-14567; however, one additional position per sector might be available.

Outboard of the active core there are three rows of reflector and four to six rows of shielding. The first reflector row contains the peripheral control rods and selected test assemblies. The second row contains the radial restraint actuators, and the third row is utilized for reflectors or uninstrumented material test. The first two reflector rows are totally accessible with the fuel handling machine. Nine positions in the third row and a number of shielding positions will require special maintenance procedures for removal. The shielding shown, along with the core barrel and sodium annulus, provides sufficient shielding to predict a vessel life of 20 years utilizing present damage criteria.¹

5.1.2 Fuel Assembly

5.1.2.1 Description

The fuel assembly is shown on SK-3-14533. The assembly is made up of a hexagonal duct of varying wall thickness which contains axial shielding, the driver fuel pin bundle, an inlet for primary sodium and an upper outlet region. The upper outlet region adapts to the instrumentation lance and incorporates attachment slots for the interface with the fuel handling machine.

1. R.A. Moen, Fluence Limit for FTR Vessel, Core Barrel, and Core Support Structure, Battelle-Northwest Laboratory, Richland, Washington, BNWL Document No. MMR-8, November 20, 1968.

In order to provide a structure compatible with the radial restraint system (see Section 5.1.3) the driver fuel duct has 0.050-in. thick contact pads located just above and below the active core region. The duct wall thickness below the core is approximately 0.183-in. thick, the in-core duct wall is 0.140-in. thick and above the core, the duct wall is reduced to 0.100 inch. The thicker section below the core stiffens the lower region of the duct to provide the proper deflection loading by the radial restraint system. The reduction in thickness of the in-core and upper walls provides flexibility while maintaining its pressure containment function.

The outside across flats dimension is 4.615 in. Maximum dimension across flats at the pads is 4.715 in. The 0.050-in. clearance on each side allows for thermal bow and tolerance buildup. The duct contains a 217-pin hexagonal array of 0.230-in. OD fuel pins on a 0.286-in. triangular lattice. With the shielding pin array of 0.375-in. OD stainless steel pins shown, projected tubesheet life is 10 years with a 500 °F bulk inlet temperature.

Height increments for the various positions of the core are outlined below.

Tabulation of Fuel Assembly Length Increments

<u>Increment</u>	<u>Value (inches)</u>	<u>Comment</u>
Inlet section	18	Length for nose, upper and lower seal and inlet slots
Tubesheet to Active Core	42	Provide shielding to prevent tubesheet damage Includes shielding and 6 inch lower reflector
Active Core	36	
Upper Reflector	6	
Initial Gas Plenum	36	

(contd)

<u>Increment</u>	<u>Value (inches)</u>	<u>Comment</u>
Gas Plenum Contingency	24	Provides additional plenum space for high burnup fuels
Grapple and Instrumentation Interface	6	Connection point for the FHM and instrument lance
Total Height	<hr/> 168 = 14 feet	

5.1.2.2 Alternative Design Options

The primary alternative areas (assuming the number of fuel pins as a fixed quantity) lie in the length increments. The below-core increment could be lengthened, the shielding removed, and the intervening sodium utilized for shielding. This would reduce the pressure drop in this region. Alternatively this increment could be shortened by incorporating B₄C shielding.

Above the core, the only major length variable is the plenum contingency. Total elimination of this length would reduce the overall height to 12 feet.

5.1.3 Core Restraint and Holddown

5.1.3.1 Axial Holddown

- A. Criteria - The holddown system for the in-core assemblies must ensure that axial motion under all loading conditions short of the DBA results in a net negative reactivity effect. To ensure that the above criteria can be accomplished, a primary holddown system with a backup minimum travel system is required. The desired design should minimize the number of in-sodium mechanical devices required. The probability of a single failure causing total core lifting should also be minimized.
- B. Description of Design - Primary holddown for this concept utilized a combination of fuel assembly weight and hydraulic balance. The hydraulic balance system chosen is shown in SK-3-14400. A low-pressure region is created below the locating nose of the fuel assembly by small

bypass tubes located adjacent to the fuel assembly duct walls which connect into the low-pressure plenum area (see SK-3-14533). Failure by plugging the vent holes would be limited to a single channel with this concept. This design also introduces an additional compliant member between the tubesheet and the duct.

The hydraulic balance leaves the assembly in a floating mode. The assembly is then held in position by its own weight which also provides a contingency holddown effect to offset pressure surges.

Backup holddown is afforded by stops on the instrument lances. The positioning of the stop is based on allowances required for tolerance buildup, thermal expansion, and radiation-induced expansion effects.

5.1.3.2 Radial Restraint

- A. General - Radial, like axial restraint, must provide a predictable method for determining core motion during all phases of FTR operation. Initial restraint studies performed on both vertical and dispersed cores has considered the effect only of thermal camber^{1,2}. Recent analysis of radiation-induced effects have shown that the thermal camber effect is almost a second order phenomena in comparison with radiation induced effects with advanced core.

Effects considered for the initial core will be limited to the thermal bow phenomena and possibly primary radiation induced creep by limiting the bulk outlet temperature to 800 °F and a burnup consistent with acceptable irradiation distortion.

- B. Description of Design - The radial restraint design is shown on SK-3-14501 and SK-3-14504. The design utilizes the second reflector row for applying a compliant load, utilizing beam deflection, to the outer periphery of the core. Initial positioning and alignment of the top of the ducts is provided by the instrument tree which sequentially positions each row of ducts. During reactor refueling each

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1. M. Jakus and N. Sutherland, Core Radial Constraint Theory and Application to the FTR, (to be published).
 2. Summary Report on Task II, Core Radial Restraint, Westinghouse (to be published).

row of ducts may be deflected a maximum of 0.1 inch per row. The tree must, therefore, be capable of deflecting the top of the outer fuel row a maximum of 0.5 inches.

Once the assemblies are in position, the beams noted above (see SK-3-14568) are deflected inward loading the assemblies at a pad located just below the active core. An additional pad located just above the active core provides an additional contact point on the fuel assemblies. The loading is applied preferentially by variable rate flat springs located on the loading clamp. By utilizing a stiffer spring for the central position on each clamp, the core would initially be loaded across points of the overall core map. As deflection continues, the next row away from the points begins to load and so on to create a triangular loading pattern.

Two secondary loading beams attached to a rotary cam provide the actuating mode for the backing plate. The cam actuator is driven by a pneumatic cylinder. The cam device is removable for maintenance through a special port in the reactor cover. Six actuators are required for the total core.

- C. Alternative Designs - Alternative actuation systems as well as different duct designs have been investigated.

For advanced cores, flexible ducts with multiple loading points may be required. This would require replacement of the loading beams only.

Actuation systems investigated included:

- . Hydraulic
- . Thermal
- . Mechanical

The hydraulic and thermal designs were found to be unacceptable because of their potential for feedback and problems with applying loads at low power.

Several alternative mechanical designs were also investigated. These include a pantograph actuated device shown on SK-3-14502. This design is similar to the selected design and incorporates approximately the same features.

A second alternative utilizes a multiple leaf spring loading design. This system would be actuated by the fuel handling machine grapple as shown on SK-3-14503. The principal advantage of this design is that additional cover penetrations are not required. The time required to actuate this number of devices, however, may be unacceptable.

5.2 Reactor Vessel and Head

5.2.1 Vessel

5.2.1.1 Description

The reference reactor vessel and arrangement of internals for this concept is shown on drawing SK-3-14540. The material of fabrication of the wall and structural components of the assembly are of solid 304 SS. Nominal vessel dimensions are as follows:

Barrel, thickness	2 inches
Barrel, diameter	18 feet
Height	51 feet
Outlet Nozzles (3), diam	26 inches
Inlet Nozzles (6), diam	12 inches

The vessel configuration shown is suitable for a mixed mean coolant discharge temperature of 1200 °F at the outlet nozzles. The thin-walled construction is possible because of the low vapor pressure of the sodium at the outlet temperature. Nominal design pressure is 30 lb/in²-g although operating vessel cover gas pressure will normally be close to 15 lb/in²-a. The thin wall of the vessel pressure boundary has the advantage of readily yielding and failing under excessive sustained pressure forces. This rapid increase in volume relieves accelerating forces existing under the head immediately following the sodium slug impact, a postulated consequence of the DBA. The 2-inch thickness has been determined to be sufficient to apply standard shape fabrication techniques in maintaining normal tolerances, and to withstand primary piping end forces and moments.

The coolant enters the vessel through six 12-inch inlet lines which contain removable check valves. The inlet lines serve as inspection entry ports for TV cameras to observe pressure boundary surfaces and for weld examination. The inlet flow is mixed in the large plenum below the core from whence it passes through a flow distribution cup which shreds large vortices and delivers a smooth flow of coolant to

the hydraulic hold-down stems which are fixed in a stiff grid plate structure. This arrangement makes it virtually impossible to starve any channel of coolant flow because of foreign objects in the primary system.

It is not considered feasible to operate the vessel wall uncooled at a temperature of 1200 °F. Accordingly, the reference arrangement utilizes approximately a 10% bypass flow to cool the barrel and outlet nozzles. At this flow, the nozzle region is approximately 950 °F maximum and the upper barrel approximately 1000 °F at the point where the bypass flow is returned to the discharge plenum. The bypass baffle is set some three inches away from the pressure boundary to permit its inspection for most surveillance purposes without baffle removal. The radial distance allows ready removal and replacement of the baffle should this prove necessary.

The fluence limit for all structural components (these being solid, Type 304 SS) shall be sufficiently low to ensure predictable behavior at EOL, i.e., sufficient ductility to exhibit a residual total elongation of 10%. Data presently available indicate that the total neutron fluence corresponding to this limit is 10^{21} nvt at the temperature under consideration for the vessel wall and other 20-year life structural components. For this concept, assuming a central total flux of 10^{16} nv, three rows of reflectors and five rows of 304 shielding bars are required for an 18-ft diam vessel. The 18-ft diameter permits the use of readily removable, rotating, fuel storage racks..

The vessel is top-mounted by suspension from an upper ring flange. The ring flange rests on a support skirt which sets in hangers supported by ring girders mounted in the building structure. The support skirt arrangement accommodates radial ring flange expansion and allows the heavy section to operate in a relatively low temperature (~ 450 °F) isothermal region. The primary seal is by metallic "O" rings in the

top face of the ring flange. The temperature disparity between the ring flange and the upper vessel wall (450 to 1000 °F) is taken without overstress along the approximate 7-ft length between the sodium level and the flange.

The complete assembly is surrounded by a closely coupled 304 SS guard vessel. The latter is mounted from the cavity floor by a bottom support skirt and has no penetrations, its purpose being to provide core coverage at all times for assurance of ECCS operation. In this regard the discharge nozzles are shown with internal dip legs. The same HTS system protection (reserve coolant drawdown in event of primary piping failure) can be provided by lower outlet nozzles with external dip legs. In the latter case the envelope of the guard vessel would be somewhat more complicated.

5.2.1.2 Alternatives

Currently under consideration is an alternative cooled-wall vessel concept. In this alternative concept, the inlet coolant enters the vessel at the same elevation as the outlet and flows downward through an annulus to the lower plenum. Some of the inlet coolant also flows upward in the annulus for cooling the upper vessel wall. Although no firm decision has been made relative to the alternative vessel concept, it is presently judged that both are basically feasible and that either could be designed to be compatible with the basic, overall reactor concept.

It may also be possible to move the periphery of the lazy susans into the radial shield. Such a move might decrease the diameter of the vessel but would require the interposing of boron carbide shadow shielding and additional cooling of the storage component but this innovation would gain as much as a foot decrease in vessel radius. For a vessel diameter of 16 feet, however, the shielding requirement increases

to seven courses of radial shielding. For an 18-foot vessel, shadow shielding would be required between the lazy susans and the vessel wall.

5.2.2 Head

The reference head for this Concept is shown on Drawing SK-3-14541 in plan and section; partial plans of the head are shown on SK-3-14519. The basic structural material is low-alloy steel which is clad with type 304 SS. Nominal head dimensions are as follows:

Diameter, overall	22 feet
Diameter, in-vessel	18 feet
Depth, overall with plugs	10 feet
Depth, without plugs	7 feet
Depth, structural	4 feet
Diameter, refueling ports	~ 4 feet
Weight, assembled	~ 300 tons

The primary purpose of the reactor head is to complete the pressure vessel boundary, provide positioning and support for reactor instrumentation, and provide refueling access to the core. The head also ensures the following secondary requirements:

- . Gamma shielding and load support
- . Thermal barrier
- . Sodium vapor deposition barrier
- . DBA blast retention.

5.3 REACTOR NUCLEAR CONTROL

5.3.1 Description

The control and safety rod drive mechanisms evaluated for prior concepts will also be used in this concept. For this reason, descriptions of the actual drive mechanisms and the reasons for selection can be found in prior work¹ and will not be reiterated here. For simplicity, all drawings in this report will show the roller nut drive mechanism as a standard drive.

The adaptation of these drives to this concept requires a hanger rod which is detachable at the drive and poison subassembly ends. The refueling mode for this concept requires that the top head be cleared of protuberances to permit change out of test assemblies and drivers. The proposed design permits removal of the drive mechanism under air-contact maintenance modes. Release of the poison subassembly is accomplished from the top of the reactor (see SK-3-14530). Removal of the hanger rod must be accomplished with a sealed machine which can withdraw the sodium wetted shaft and install a plug in the vessel cover.

The drive mechanism is mounted on the reactor vessel cover. The mechanism drive shaft connects with a hanger rod by use of a breach lock mechanism. The hanger rod passes through a seal system in the bottom of the vessel cover and extends to the poison section. A collet mechanism is located at the bottom of the hanger rod and is used to connect to the poison section. The collet can be operated from the area of the breach lock after the drive has been removed.

The use of a separable hanger rod and drive mechanism makes location of the deceleration device above the reactor vessel mandatory. If the decelerator were located beneath either the breach lock or collet,

1. J. C. Noakes, Control Mechanisms Review Study, (to be issued), and letter, E. R. Astley to J. M. Shivley, "Reactor Nuclear Control Components (System 33) Planning," August 22, 1968. (PNL/FFTF-68788)

deceleration loads on these joints would be unacceptable. The decelerator selected for this concept consists of a hydraulic snubber, using oil as a working fluid, which is integral with the drive mechanism.

In this concept, a scram-assist mechanism is also mounted on the top of the drive mechanism. The scram-assist is provided by a spring, although pneumatic pistons could be used. Mounting the scram-assist mechanism on top the reactor cover permits the mechanism to be cocked by a separate pneumatic cylinder, thus reducing the load on the drive mechanism. Pneumatic pressure is only required during the cocking operation.

Scram-assist devices for obtaining initial accelerations of 2g on the scramble rods will impose a buckling load on the hanger rod. The arrangement in this concept differs from that of Concept V in that the hanger rod was guided by the instrument tree to a distance of 4 to 5 feet from the top of the core. In this concept, the hanger rods are unsupported, except for possible short bearings in the instrument plate. Preliminary calculations indicate that the buckling limit for the hanger rod is in excess of 4000 pounds. Previous investigations into the forces required for achieving 2g acceleration indicate that the force required would be on the order of 1000 to 2000 pounds.

The drive mechanism with attached spring-assist mechanism and decelerator is shown on SK-3-14531. The poison subassembly for a typical safety rod is shown on SK-3-14529 and shows a 61-pin B_4C bundle.

The interface between the refueling plug and the control rods located at the corners of the hex pattern is of critical nature because of possible interference, and must be very closely monitored during design. The task force has investigated the cases for conceptual feasibility and this interface is shown on SK-3-14532. As the plug

diameter is increased, the peripheral rod located outboard of the plug becomes limiting and as the diameter decreases, the in-core control rod located near the plug becomes limiting.

5.3.2 Operating Philosophy and Problem Areas

All drives and hanger rods are designed to be removed prior to each refueling operation. The drive mechanism along with the lead screw is removed by unbolting the flange from the top cover and rotating the entire mechanism to release the breach lock mechanism. Release of the poison subassembly is then accomplished by a special tool that is inserted into the lead screw cavity. The hanger rod can now be removed by the casks used to remove driver fuel instrumentation modules. Complete removal is required for all sectors that will require refueling. It may be possible to leave the hanger rods in place in sectors that are not to be refueled. The capability to leave selected hanger rods in place during refueling will depend upon final configuration of the core clamping system.

The current refueling machine concept does not envision changeout of grapple heads on the refueling arm. Therefore, poison sections will be removed with the same grapple head used for driver fuel. Removal and replacement of poison subassemblies by use of a common grapple head will require an interlock to prevent errors in refueling. One such method suggested would be to install the pickup point for the poison subassembly deeper in the duct than that for the fuel subassembly. In this way, an "over travel" permissive switch must be unlocked to permit the fuel handling arm to lower far enough to pick up a poison subassembly.

5.3.3 Control Rod Design Options

In an attempt to reduce the number of top-mounted control/safety rod drives and thus enhance plant availability, the design option of bottom operated control mechanisms has been investigated. The concept examined represents a combination of the EBR-II safety rod beam mount

and a pivoted DFR arrangement. The system shown in SK-3-14542 arranges five adjacent rods under each rotating plug in a single drive unit. The bottom of the rods is ganged to a beam which is then extended by a box frame at right angles to a point in the periphery of the vessel. The box frames are driven by control drives about pivot points in such a manner that downward movement of two control drives results in upward (out-of-the-core) movement of the five rods. The two rods adjacent to the plugs but still in the reflector row are individually driven, as are all the in-core rods. The resultant arrangement consists of six individually driven peripheral rods and three banks of five ganged rods driven by offset drives, in addition to the nine individually driven in-core rods.

In order to minimize the number of penetrations through the high-pressure plenum, a toroidal inlet chamber is used, designed so the drive shafts can pass to the bottom of the vessel outboard of the high-pressure section. Penetration of the push rods through the bottom high-pressure section of the core will require seal sleeves. Linkages required to allow vertical movement of the drive and driven shafts are shown as sliding two-way pin/slot arrangements. Vessel length will increase by the rod stroke length plus linkage and bar requirements or in the range of six feet. Vessel diameter is maintained at 18 feet. However, no provisions are made for removal and installation of the linkage mechanisms without core disassembly. Additional disadvantages are the use of under sodium linkages, high-pressure to low-pressure plenum sliding seals, and rod insertion rates that are limited by drive motor speeds since the drive mechanism must go up for the rod to go down. Advantages include the addition of six peripheral rods and the reduction of nine drive mechanisms and hanger-rod shafts that must be removed for refueling and reactivity checks. This conclusion is based on the assumption that the six drive mechanisms for the 15 ganged rods need not be removed during refueling.

5.4 REACTOR REFUELING SYSTEM

5.4.1 Driver Fuel and Core Component Handling

5.4.1.1 Refueling Plug

Three refueling plugs provide access for removable instrumentation to each driver fuel subassembly and provide access for refueling the reactor. Each refueling plug services a one-third sector of the core. The plugs are removable for maintenance and require rotation for positioning the instrumentation probes and for operating the reference in-vessel fuel handling machine.

The plug supports the instrumentation and holddown plate assembly and the in-vessel fuel handling machine (IVHM) as shown on Drawing N68181267. The IVHM can be removed from the refueling plug for maintenance. The refueling plug must be removed for maintaining the instrumentation hold-down plates for other than individual thermocouple and flow meter instrumentation maintenance. The IVHM must be removed before the refueling plug can be removed.

5.4.1.2 In-Vessel Handling Machine

The in-vessel handling machine (IVHM) is shown on Drawing N68181249. The IVHM is used to transfer driver fuel assemblies, control elements and reflectors between the reactor core and the in-vessel lazy susan storage facility.

The IVHM contains an offset grapple that can be positioned over the center of the core and over all positions in a core sector. Operation of the IVHM is accomplished by simultaneous rotation of the offset arm assembly and the refueling shield plug. The offset arm lifts the fuel subassembly or other core components by means of a screw-type jack and is guided by rails on the main IVHM support column.

The grapple is operated only in the down position, where it picks up or releases a fuel subassembly in either the core or the lazy susan. A fuel subassembly holddown foot is located at the bottom of the IVHM column, about one inch above the top of the subassembly ducts which prevents adjacent subassemblies from being raised with the subassembly being removed. As the grapple reaches its full "up" position, the hold-

down foot automatically pushes down on adjacent subassemblies to ensure proper position in the core.

The IVHM also contains a subassembly orientation device that rotates the subassembly about its axis as desired for repositioning in the core. The grapple with a subassembly attached is raised to the full "up" position and positioned under the refueling plug in contact with the orientation device. The subassembly then may be rotated through any desired arc.

The drive unit (motors, switches, controls) is located on the top of the shield plug, as shown in Drawing N68181248. When this drive unit is removed, the top shield is flush in this region. The drive shafts are connected to the driven shafts by single-orientation splined joints.

When the drive unit is removed, the shafts are held in place by lock-bars that prevent shaft rotation. The IVHM is operated from a control console located remotely from the IVHM itself.

5.4.1.3 Instrumentation and Holddown Plate Assembly

The instrumentation and holddown plate shown in Drawing N68181255 provide instrumentation probes to each fuel subassembly in a sector, serves to compact the fuel subassemblies toward the center of the core, provides back-up fuel subassembly holddown, and guides the control rod driveline assemblies. The column supports and positions the instrumentation plate and the holddown plate. The lower end of the support column is offset and is supported laterally by a bearing and structure.

To prepare for fuel handling, the control rod drivelines are removed, and the instrument plate raised until all probes are above the top of the fuel assemblies. The support column is raised until it is removed from and is above the lateral bearing. The holddown plate ensures that the instrument probes withdraw from the ends of the subassemblies without lifting any adjacent subassemblies out of the core. The

instrument tree, in the raised position, is rotated away from the core by a dual rotation of the instrument tree support column and the shield plug.

All actions of the instrument tree are powered by electric motors located in the shield plug. The motors are controlled from a console located remotely from the shield plug.

5.4.1.4 In-Vessel Lazy Susan Storage Facility

A lazy susan is located adjacent to the shielding outboard of each sector of the core as shown on drawings SK-3-14459 and SK-3-14540.

Each rotor has nine positions; each position being capable of accepting a finned pot containing a driver subassembly or other core component. Fuel or core components are transferred from the core to the storage position in the lazy susan via the IVHM. The lazy susan may be indexed as required by electric motors geared to the centrally located drive shaft. To transfer a driver subassembly or other core component out of the reactor, the storage position containing the desired finned pot is oriented under the transfer port. The subassembly or core component may then be withdrawn through the transfer port into the ex-vessel handling machine.

5.4.1.5 Ex-Vessel Handling Machine

The ex-vessel handling machine (EVHM), shown on drawing N68181217, transports driver fuel subassemblies and core components from in-vessel storage through the transfer port to either interim storage or the transfer cell. The EVHM returns new driver fuel subassemblies and core components to the in-vessel storage positions.

The EVHM is shielded and carries the subassemblies in sodium-filled finned pots; the EVHM can handle and cool up to three subassemblies. The decay heat from the subassemblies is transferred by conduction and convection through the sodium to the finned pots. The pots are cooled by forced convection and radiation to the cask walls. Argon is used as the coolant media for the pots. The gas flows out of the machine

main body through the shielded sodium vapor trap, the circulator, the air blast heat exchanger, the heater for preheat and temperature control, and back to the machine. High pressure gas storage tanks provide emergency cooling (blow down) in case a pot sticks in the reactor head, or shield. Redundancy of cooling systems provides emergency cooling of the fuel.

The machine uses one grapple and drive assembly, and a three-position rotating barrel. Three pots are contained in the rotating barrel. A grapple-operated guide sleeve is used to protect the machine drip pan valve and closure valve, and the reactor floor valve from sodium drippage during pot transfers.

5.4.1.6 Plug Handling Cask

The plug handling cask is a shielded machine capable of transporting a shield plug containing an instrument tree. It has the capability of handling the IVHM, the lazy susan shield plug, and the lazy susan storage rotor. The plug handling cask may be the same maintenance cask used to handle the primary pumps and the IHXs.

The plug handling cask is comprised of a lower closure valve (which mates with the floor valve), the drip pan, the main barrel, the hoisting and guiding system, and the environmental control system for maintaining an inert atmosphere at the required temperature.

5.4.1.7 Control Rod Driveline and Instrument Cask

The control rod driveline and instrument cask, shown in Drawing N68181268, is a dual purpose cask used to replace the control rod driveline and guide from the reactor and the fuel assembly instrumentation tree. The cask is used to remove the control rod driveline from the reactor after the driveline has been disengaged from the absorber section. After removal of the control rod driveline, a seal and shield plug are inserted into the reactor opening prior to removing the cask.

When used for instrument maintenance, the cask is positioned over the instrument probe to be removed (see Drawing N68181250) and the selected

probe is withdrawn. A new probe is inserted into the vacant hole. Glove ports and lead glass windows are provided at the top and bottom of the cask to assist in the instrument probe exchange. The lower glove ports are provided with shielding for the actual pulling of driveline or instrument probes.

5.4.2 Open and Closed Loop Handling

5.4.2.1 Loop Preparation Cask

The loop preparation cask, shown on Drawing N68181185 is used to prepare both open and closed loops for test removal from the reactor. The instrumentation connector plug and leads are removed by hand prior to the use of the loop preparation cask.

A shielded floor valve is placed over the loop experiment after the instrumentation leads have been removed. The loop preparation cask is placed on the floor valve. The loop preparation cask is a shielded machine containing: (1) a weld cutter and a vacuum cleaning system to pick up chips formed during cutting; (2) a grapple for removing the top loop shield seal-plug; and (3) a weld head for rewelding the seal-plug to the loop nozzle.

5.4.2.2 Loop Handling Machine

The loop handling machine (LHM) shown on Drawing N68181222, is a shielded sodium-cooled machine for handling open loop test assemblies, closed loop test assemblies, and the in-reactor assembly of the test loops. The LHM is used to remove these assemblies from and insert them into the reactor, and to transport them from the reactor to either interim storage or the transfer cell. The sodium coolant is obtained from the reactor bulk coolant system if handling open loop tests, and from the closed loop system if handling closed loop tests. The LHM secondary cooling system uses NaK as coolant.

A snorkel, attached to the wet valve, is socketed into the test position nozzle to seal the machine to the port or nozzle and to prevent sodium leakage when the sodium is raised. Sodium coolant is retained in the internal assembly of the LHM by a wet valve located at the bottom of the internal assembly. The primary mode

of sealing the wet valve is metal-to-metal contact. The secondary or backup mode of sealing is freezing the sodium between the valve and valve seat. After the test assembly is inside the machine, the wet valve is closed and the sodium in the snorkel drains back to its source. The residual sodium on the inside wall of the snorkel is frozen to control drippage. The snorkel is then withdrawn into the LHM; a drip pan is positioned under the snorkel; and the shield valve seals the internals of the machine from the atmosphere.

5.4.3 Refueling System Design Options

A. Separate plugs for instrument plate and in-vessel refueling machine.

The reference design includes a single rotating plug with both the instrument plate and refueling machine installed in the plug. An alternative approach would be the use of separate plugs for refueling and the instrument plate. For refueling with this alternative, the plugs would be exchanged through use of a floor valve and plug cask. Maintenance on the instrument plate and on the grapple would be performed while out of the reactor. This alternative was not selected for the following reasons:

- . If the combined plug approach could not be designed, it would always be possible to revert to separate plugs.
- . The ex-reactor handling and maintenance facility requirements are more extensive for the separate plug approach significantly increasing plant capital cost.
- . The reactor availability is higher with a combined plug approach.

B. Multiple, Fixed Position, In-Vessel Storage

Fixed position, in-vessel storage positions were considered as an alternative to the lazy susan approach shown in the reference design. Advantages of fixed position compared with the lazy susan storage include:

- . Elimination of the in-vessel operating mechanisms of the lazy susan.

- . Elimination of the 4-foot diameter cover penetration for lazy susan maintenance access. (Alternatively, maintenance access could be provided through use of the refueling machine port but this would require offset equipment as well as lifting and translation of the lazy susan for removal.)
- . Potential reduction in vessel diameter.
- . Easier mechanical accommodation of the dropped fuel pot accident.

Use of fixed position, in-vessel storage locations requires that these positions be located within the reach of the in-vessel fuel handling grapple unless an additional mechanism is mounted in the reactor to transfer fuel from the grapple to the storage positions. With fixed storage positions located within grapple reach, and either single or multiple fuel exit ports, the location of the exit ports is such that there is a small (~6 inch) overlap between the reference 3-barrel, ex-vessel machine and the rotating plug and in-vessel machine support shaft when the ex-vessel machine is positioned on the exit port(s). This problem can be accommodated by:

- . Offsetting the refueling grapple drive mechanism.
- . Burying the in-vessel machine grapple drive to a level below the ex-vessel machine flange.
- . Operation of the in-vessel and ex-vessel machines at the exit port(s) locations at different times.

Availability studies indicate that there is a small incentive for a lazy susan compared with fixed position storage where the in-vessel machine and ex-vessel machine can work simultaneously. The incentive arises from the capability with a lazy susan to remove fuel from the reactor while the refueling grapple is removing fuel from the core. In the fixed storage case where a single exit port is used, the in-vessel machine is needed to transfer fuel from storage positions to the exit port. The incentive for simultaneous operation of both in-vessel and ex-vessel machines is significant

(two to three days per core sector per refueling cycle). The design of the ex-vessel machine to minimize overlap and design of the drive package for the in-vessel machine to either bury it below the operating floor or to offset it sufficiently to clear the ex-vessel machine would be required.

C. Single Transfer Site at Exit Port

This option permits only loading core components into finned pots at the exit port for direct withdrawal from the reactor vessel. The reduced heat load on the ex-vessel FHM that results from in-vessel decay storage is not achieved with this option. Studies should be performed to determine how much ex-vessel interim storage is needed when in-vessel storage is provided, where such ex-vessel interim storage should be located, how core components might be transferred to ex-vessel interim storage and then to other fuel handling facilities outside of containment, and how test specimen and driver fuel handling might be combined to minimize the total fuel handling and decay storage systems.

D. Ex-Vessel FHM with More than Three Barrels

The reference design has a 3-barrel, ex-vessel FHM on the basis that efficiency gains can be realized with more than one barrel, that these barrels require the next minimum increase in size of the machine, and that increases in size, complexity, and cost for more than these barrels could not be justified in the time available to study this concept. Previous multi-barreled FHM studies had established the foregoing relationships for single nozzle access reactors. Additional studies are required to determine the optimum number of barrels to be incorporated in the ex-vessel FHM for the reference design.

E. Test Facility Refueling Equipment

The reference concept utilizes a sodium-cooled fuel handling machine (SCFHM) for refueling the contact instrumented open loops and for handling of closed-loop test specimens. No design alternative to

to this wet machine has been identified for these testing capability requirements. Only if contact instrumented open loops were eliminated and a workable design approach could be devised for routine removal of the test specimen with the closed-loop thimble, could an alternative to a sodium-cooled test facility machine be considered. The wet valve machine has been selected for the reference approach because the sodium lift required for the taller internal dam design test facility machine would require reactor or closed loop cover gas pressurization in routine refueling.

5.4.4 Fuel Handling Equipment Arrangement

Preliminary layouts of the ex-reactor fuel and test handling equipment are shown on SK-3-14496 and SK-3-14497.

5.5 IRRADIATION TEST FACILITIES

5.5.1 Test Facility Location and Type

The reference core provides space for nine test facilities along the 120 degree tri-secting axes of the core. One of the three positions adjacent to the central driver location can only have proximity instrumentation in order to provide refueling access to the central position.

The remaining eight positions can accommodate six closed loops and two open tests using contact or proximity instrumentation, eight open tests with contact or proximity instrumentation, or any combination thereof.

All nine experiment locations are instrumented through independent ports penetrating the vessel cover. However, any test position can be reached by the driver fuel handling machine and thus, all proximity instrumented open tests can be changed out using the driver handling sequence.

The core map does not prescribe a position for a short-term facility (trail cable). It appears that a trail cable may be located in any one of the three peripheral experiment locations (next to first reflector row). Task force effort did not examine the vessel head routing for the access tube and associated changeout mechanism for this facility.

5.5.2 Closed Loop Description

5.5.2.1 Closed Loop Branch Arm Piping-Drawing SK-3-14456, Sht. 1

The design provides for 2-inch OD Sch. 40 coolant piping enclosed by a 4-inch ID insulation envelope. Insulation consists of an argon gas gap, reflective metal wrap and 5/8 of Mini-K fiber insulation. Further optimization may be achieved by providing multiple layers of stainless foil with fiber insulation between layers all contained inside an evacuated space.

Up to 15 in. of high density radiation shielding can be applied above the coolant piping. The instrument cable is routed along the coolant piping above the shielding but below the operating deck.

The branch arm piping has been doubly encapsulated to permit decontamination (by replacement) of sodium spills into the branch arm pipe chaseway and to contain sodium from a fractured branch arm pipe. Handling sequences for changeout and replacement are illustrated on Drawing SK-3-14456, Sheet 2.

5.5.2.2 Closed Loop Nozzle Detail - Drawing SK-3-14457

The design provides a spool piece and closure arrangement utilizing welds as the final seal between reactor cover gas and containment air atmosphere and between loop sodium and containment air atmosphere. K-gaskets are used as primary seals between tube inlet and outlet coolant and as an initial barrier between loop sodium and reactor cover gas and between loop sodium and the containment air atmosphere. The closure design permits routine removal of the experiment separate from the outer tube and non-routine removal of the experiment and outer tube together.

5.5.2.3 Closed Loop In-Core Tube - Drawing SK-3-14515

The tube provides double walled inner and outer tubes with argon in the gas gap. Along the in-core portion of the test, sintered stainless steel replaces the gas gap because it is a more conductive surface for decay heat removal. The circular cross section of the outer tube is adapted to fit the hexagonal core array. The test space can accommodate 37 pins of 0.25-in. diameter or 19 pins of larger diameter.

This drawing and other test positions drawings are based on a 7-in. OD spool piece assembly for open and closed loops. This dictates a 5-in. ID clearance hole in the reactor cover and spool piece which limits the diameter across the hex of open loop ducts top less than 5-in. OD. This would limit the experiment to less than full driver size.

Further study of the problem indicated that the spool assemblies could be increased to approximately 7 5/8-in. OD which provides a clearance hole of 5 1/2-in. ID. This is compatible with driver duct dimensions. Principles of operation and maintenance appear to be essentially unchanged. Because of the limited time available the drawings were not revised.

5.5.3 Open Test Position (With Contact Instrumentation)

5.5.3.1 Open Test Position Nozzle Detail - Drawing SK-3-14458

This nozzle utilizes the same welded closure configuration as that used for the closed loop and, therefore, the same welding and cutting equipment may be utilized. The opening through the nozzle allows withdrawal of an adapted driver size open test assembly similar to the closed loop. An extension is provided between the nozzle and the reactor sodium pool to permit wet withdrawal of a test utilizing the closed loop handling machine.

5.5.3.2 Open Test In-Pile Section - Drawing SK-3-14514

The design provides for a driver size test assembly. As previously noted, eight positions can accommodate contact instrumentation. Standard instrumentation includes thermocouples, pressure sensors, gas sampling line, flux thimble, and a Fedal sampling system. The latter involves a vortex generator and sodium sampling line to detect fission gas in the test channel.

5.5.4 Open Test, Closed Loop and Control Rod Integuments-Dwg. SK-3-14513

The drawings show envelopes to define handling methods.

5.5.5 Design Options for Irradiation Test Facilities

The reference closure designs for both closed loops and open test positions involves remote welding and cutting operations. This choice is based on the great incentive for providing ensured no-leakage boundaries. If development efforts now planned and in progress on mechanical seals result in proven designs, then an availability advantage may result from the use of mechanical clamps on closed loops and open tests.

5.6 REACTOR INSTRUMENTATION

5.6.1 Instrument Support Assembly (See Drawing N-68181255)

The primary functions of the instrument plug(s) are to:

- . Provide access to the driver assemblies so that they may be instrumented for flow, temperature and fuel failure location.
- . Provide backup mechanical holddown for the core.
- . Provide a method for assembling the core prior to applying radial restraint.

Each of the three instrument plugs provides access to approximately one-third of the core and consists of:

- A. An instrument plate that supports the driver instrument packages, the core assembly pieces and the mechanical holddown foot.
- B. The driver shaft which permits translation and rotation of the instrument plate.
- C. The FEDAL line plug-in connection grid.
- D. The guide tubes which permit replacement of the flow and temperature sensors with the plug in or out of the reactor.
- E. The shielding needed in the top head region.
- F. All hardware required for the function of the instrument plug such as drives, bearings and seals.

5.6.2 FEDAL Arrangement

5.6.2.1 Basic Concept Description and Rationale

The technical bases document¹ for FEDAL indicates that fuel failure detection must be accomplished rapidly so that action can be taken to prevent continuation of the damage sequence which is not fully understood. This reasoning and other considerations dictate the following guidelines for the FEDAL detection-location system.

1. J.J. Regimbal, Technical Bases for FFTF Driver Fuel Instrumentation, BNWL-916, pp 6-8, Pacific Northwest Laboratory, Richland, Washington. October 1968.

- . In the early stages of FTR operation, the reactor will not be operated intentionally with failed fuel.
- . The fuel failure must be detected rapidly; the detection time has not been established.
- . Fuel will be considered to be failed if the cladding has been violated at any location including the gas plenum region.
- . Failed fuel location should be possible within the refueling time period for the reactor.

Three FEDAL location systems were considered for the location system concept. These were:

- . Detection of released fission gases.
- . Detection of delayed neutrons.
- . Detection of released tag gases.

The detection of released fission gases was selected as the reference concept for the following reasons:

- . This system has the potential for locating a failed element from which only gas is released either continuously or in a burst.
- . It has the potential for locating an element with the reactor shut down.
- . It has the potential for identifying single or multiple elements which may fail at the same time.

To ensure that the sodium sample stream can obtain gas from any fuel pin in the subassembly and that the amount of gas in the sample stream be maximized, a fission gas concentrator is required. The concentrator (also known as a vortex generator, or a fission gas separator) is located above the fuel pins in the subassembly outlet coolant flow. Gas bubbles are forced to the center of the concentrator by centrifuge action where a perforated tube serves as a collector and the starting point for the sodium sample line.

These sodium samples are removed from the reactor vessel to shielded cells by means of pumps located in the rupture detection cell. Here the gas is separated from the sodium by means of stilling chambers. The sodium is returned to the sodium purification surge tank and the separated gas is delivered to a monitoring location shielded from the sodium where the GM detectors monitor for an increase in gas activity.

A conceptual arrangement for the driver location system is shown in Drawing SK-3-14548. Other details of the conceptual FEDAL system are shown on the following drawings:

- . N-68181251 FEDAL Tube Configuration at Stationary Top Shield
- . N-68181255 Instrument Support Assembly
- . N-68181256 Instrumentation Assembly and Driver Fuel Alignment
- . SK-3-14519 Reactor Cover Assembly
- . SK-3-14540 Reactor Elevation
- . SK-3-14541 Reactor Head Plan and Section

5.6.2.2 FEDAL for Open and Closed Loops

Detection of closed loop test section failure will be accomplished by monitoring in the closed loop cell. For the open loops with fuel tests, the sodium sample line will exit near the top of the open loop stalk where the sample is transferred from the stalk to exit tubing in the reactor cover shield (see SK-3-14458). This is accomplished by means of a plug-in socket on a shoulder of the open loop. This connection is broken when the open loop is withdrawn. The exit tubing in the shield must be accessible for unplugging or replacement.

The exit method for open loops is not ideal. A leak at the transfer point during operation will cause loss of flow. During each removal of the open loop, both ends of the line (the end in the stalk and

the end of the line in the shield) are potentially exposed to oxygen which could result in oxide plugs when the open loop is re-installed. Therefore, alternatives such as locating this connection under the sodium pool must be considered during preliminary design. Two ways of doing this are to remove the sample line from the bottom of the open loop (bottom of the core) or provide egress through a gland located beneath the shield and under the sodium.

5.6.3 Driver Fuel Subassembly Instrument Package

The driver fuel instrumentation consists of four main assemblies:

- . The removeable sensor assembly (shown on Dwg. N68181256) which consists of an eddy current flowmeter, five thermocouples, the lead wires associated with the sensors, a push-pull support rod, and the male half of a ring type electrical connector.
- . The flow conditioner-neutron shield assembly (shown on Dwg. N68181256) and the fission gas concentrator are located in a flow duct. A lead out-guide is welded into the top of flow duct.
- . The maintenance access assembly (shown on Dwg. N68181250) including the female half of the ring type electrical connector mentioned in the first bullet), several plug type electrical connectors, the necessary seals, and the mechanical support structure required.
- . The FEDAL system including in-vessel and out-of-vessel equipment. This system is discussed separately in Section 5.6.2.

Each of these will be discussed in sufficient detail to supplement the material presented in the drawings.

The total pressure drop across the instrumentation package is estimated to be between 10 and 15 lb/in².

- A. The removable sensor package provides for the monitoring of sub-assembly coolant flow and temperature. Provision has been made for five thermocouples assuming a 2-of-4 temperature safety trip plus one used for data gathering and/or control functions. With the neutron shielding presently envisioned, the thermocouple expected life is three years.

The flowmeter should provide acceptable ($< \pm 20\%$ of the full flow as the error) flow measurement over a 20-120% flow range. Readings will be available at lower flows but the accuracy may be questionable.

The leads (two conductor) are mineral insulated and sheathed. These leads are 0.125-in. nominal OD and are attached to the push-pull support rod by the spherical balls as shown in Dwg N68181256. The spherical balls are included to provide guidance and support, and to prevent galling or fretting of the wires with the guide tube.

The male half of the electrical connector begins immediately above the radiation shield (Dwg N68181250). A transition piece is provided for each sheathed lead to allow the use of flexible, comparatively larger wires from this point to the rear of the connector. The connector proper consists of a hollow cylinder which has electrically conductive surfaces separated by insulator material and has the lead terminations attached to these conductive surfaces from the inside of the cylinder. This type of connector has been successfully used in hot cell operations.

There is a structural piece with a grapple ring below the connector proper to provide for attachment of the device which will be used to insert or remove the sensor assembly. The push-pull support rod is attached to this structure through the solid radiation shield piece. This assembly minimizes the force exerted on the electrical leads during insertion.

The removable sensors package is to be completely bench assembled using hands-on assembly procedures. The entire assembly is to be inserted into the guide tube. This procedure provides a more reliable and economical assembly than a procedure which involves making up connectors after insertion.

- B. The mechanical support and flow conditioner-neutron shield serves to reduce the physical or radiation damage to the active sensors. The flow guide tube serves to channel the flow from a single assembly past its instrumentation, provides the structural support for all of the instrumentation, and serves as part of the core gathering agency. The leadout guide tube provides the continuous path for the sensor package to follow during insertion and withdrawal. The bends in the leadout guide tubes are sized to allow relatively easy passage of the sensor package.

Only the flow straightener and neutron shielding portion of the flow conditioning equipment are discussed in this section. The remainder of this equipment is discussed in the FEDAL Description Section. The neutron shield is included to reduce the fluence in such a manner that a 3-year lifetime goal for the thermocouples is considered reasonable. The flow straightener provides a flow profile to enhance the accuracy of the eddy current flowmeter.

- C. The design of the maintenance access port provides the capability for replacing the removable sensor packages at periodic intervals. Four plug-type connectors are provided to permit easy removal of the electrical jumper leads when required by the fuel handling procedures. The lower half of the plug type connector and the female half of the ring connector are attached to a structural plate which includes a depleted uranium shielding structure. Removal of this connector assembly can be performed using hands-on access techniques. The male halves of the ring type connector are all exposed after the connector assembly is removed.

The hold-down plate restrains the sensor packages from motion induced by loading at the lower end. This hold-down plate is removed when sensor packages are to be replaced. Note that after the holddown is removed, the double seals on each sensor package are still maintained.

5.6.4 Experimental Facility Instrumentation

5.6.4.1 Open Test Facility Instrumentation SK-3-14514

The open test facility process instrumentation is similar to the driver fuel instrumentation described in Section 5.6.3 and consists of the following:

- . One eddy current flowmeter
- . Five outlet coolant thermocouples
- . One vortex generator and sodium sample line. The sodium sample line exits through a connection in the nozzle area.
- . Provision for 0.50-in. OD thimble for flux mapping during startup testing.

Experimenter's Instrumentation - Drawing SK-3-14514 shows the capability for making the following measurements which are in addition to the process instrumentation requirements:

- . Two fuel pin meat thermocouples
- . Four fuel pin cladding thermocouples
- . One fission gas pressure measurement using a NaK capillary to electromechanical transducer
- . One fission gas pressure measurement using a null balance pressure transducer.
- . Many other combinations of instruments or test arrangements are possible.

Nozzle connector and instrument leadout details are show on SK-3-14546 and SK-3-14547.

5.6.4.2 Closed Loop

Instrumentation has two major parts, process surveillance and experimenter instrumentation. Reference Drawing No. SK-3-14515.

A. Process Surveillance

- . Inlet coolant temperature - three thermocouples mounted below the test section.
- . Outlet coolant temperature - three thermocouples mounted above the test section before bypass coolant is introduced.
- . Coolant flow is measured in closed loop piping external to the reactor.
- . Failed fuel detection is performed outside the reactor. Failed fuel location is of course not required.
- . Differential pressure across the test section is measured by two bellows - NaK capillaries. One is installed below the test section and the other above the test section. The two capillaries are led to a differential pressure electromagnetic transducer located in the nozzle.
- . Flux monitor thimble. A 0.25-in. OD thimble is shown penetrating to the bottom of the test section. Installation of the thimble and its size depends upon the needs of the experimenter. Space for a 0.5-in. diam. thimble is shown through the nozzle connector (Ref. Dwg No. SK-3-14547). The thimble may also be used separately during early startup for flux mapping.

B. Experimenters Instrumentation

- . For a fuel test, Dwg SK-3-14515 represents the manner of connecting experimenters lines. The following instrumentation capability, in addition to surveillance instrumentation, is possible. Alternative combinations of instruments are possible at the experimenters discretion.

Fuel pin internal temperature - 2 TC's

Fuel pin cladding temperature - 4 TC's

Fuel pin internal pressure - using the pressure balance method, a tube is led to the nozzle connector from a diaphragm in the fuel pin.

- . For an alternative material test, instrument lines are available for the following representative tests:

Specimen temperature - 3 TC's

Strain gages (when developed) - 2

Nozzle connector and instrument leadout details are shown on Dwgs. SK-3-14546 and SK-3-14547.

5.6.5 Cable Routing - Reactor Cover Assembly

- . The connectors that pick up the instrumentation leads as they emerge at the top of the reactor cover shield are shown on Drawing N68181250 for driver instrumentation and SK-3-14547 for open and closed test loops.
- . The location of each connector and cable routing over the reactor cover shield is shown on Drawing SK-3-14519.

The cable is assembled into harnesses which have a quick disconnect at each end. The driver fuel element instrument harnesses are stored below the reactor cover floor plate during fuel handling. The test loop harnesses are also stored under the floor plate but would be removed during closed loop pipe handling.

The harness assemblies connect to junction boxes located on the circumference of the reactor top shield assembly.

5.6.6 Low Level Neutron Monitors

5.6.6.1 Basic Requirements and Rationale

The flux monitor system is required to monitor the neutron flux level at all times. The most difficult situation to monitor occurs several minutes after the reactor has been shutdown from full power operation. The Low Level Neutron Monitor (LLNM) must be capable of monitoring the

shutdown flux, which may be as low as a factor of 2×10^{11} below full power flux. A minimum acceptable count rate for a low level neutron monitor has been established at 5-10 counts/second.

5.6.6.2 Detector Location

The application of recently established fluence limits on the vessel wall were used to calculate the reaction rates of candidate sensors materials, boron and U-235, located in the neutron shield external to the vessel. These reaction rates indicate that the minimum count rate values cannot be established with state-of-the-art sensors located in the external shield and circuitry when the reactor is in its most subcritical state.

For this reason, an acceptable location for a low level neutron sensor was sought within the reactor vessel. Studies indicate that an acceptable detector location would be at approximately the interface of the nickel reflector and the in-vessel shield.

5.6.6.3 Detector Response

The estimated neutron shutdown count rate for a commercial 2-in. diameter fission chamber containing about 1.8 grams of U-235 (2 mg/cm^2) when operating in a gamma field of 10^5 R/hr is about 10 counts/second. This estimate takes into consideration a decrease of 2×10^{11} in reactor power, and the local effect induced by the insertion of the peripheral rods.

The background gamma immediately after shutdown will be about $1-2 \times 10^6$ R/hr because of the decay gammas, predominately from manganese, in the stainless steel. This will reduce the available neutron count rate to values below 10 counts/second for a period of 8 to 10 hours after shutdown.

5.6.6.4 Mechanical Features

Drawing SK-3-14519 shows the location selected for the shutdown monitor. It is expected that three in-vessel monitors will be required to ensure that two are normally available. A mechanical interference exists between the fuel handling-instrument plug and the flux monitor thimble

during refueling. For this reason the flux thimble containing the neutron sensor and driver must be removed from one cover sector before refueling in that sector can be accomplished. The removal procedure for the chamber drive and thimble will be similar to that for the control rods driveline.

The flux monitor assembly will consist of the protective thimble which contains the fission chamber, some inorganic moderator, the driver extension, a gas cooling system and shielding. The chamber positioning mechanism will be similar to a rod drive and will be capable of raising the chambers approximately 42-in. above its fully inserted position. The neutron flux in the retracted position will be sufficiently low to permit a chamber life of several years if based only on this consideration.

The chamber cooling system must maintain a chamber environment of about 600 °F or less. However, the cooling system need not be operative during reactor shutdown.

A supplementary LLNM system located in the shield external to the reactor vessel will be required to obtain the necessary overlap with the intermediate range monitor system.

5.6.7 In-Vessel Out-of-Core Instrumentation

5.6.7.1 Coolant Level

Coolant level will be sensed through closed thimbles by means of induction coils similar to those used at Hallam. Three vessel cover penetrations of about 5-in. diam will be required.

Three penetrations are required to provide the necessary redundancy and backup required for scramming and other purposes. The thimble locations are out of the core area, over core shield pieces and out of the way of fuel handling operations.

5.6.7.2 Inlet, Outlet and Cover Gas Temperature

The inlet coolant temperature will be measured with thermocouples installed through a thimble which penetrates the vessel cover and a core shield piece, and with access to the inlet plenum. This same

thimble (perforated) may be used for the thermocouples which measure the temperature of the three outlet coolant streams. The cover gas temperature is measured with thermocouples located between the pool level and the shield. Three locations for these thimbles are necessary to provide redundancy of measurement as well as to provide temperature measurements at the points necessary due to the arrangement of the inlet and outlet streams.

5.6.7.3 Inlet Pressure

Inlet coolant plenum pressure will be measured with a bellows, a NaK-filled capillary, and an electromechanical transducer located out of the high-temperature region. The bellows will be mounted on a long stalk inserted in a thimble which penetrates the vessel cover and vessel shield piece to be exposed to the inlet pressure.

5.6.7.4 Cover Gas Pressure

Because the cover gas pressure will be low (near atmospheric) a large diaphragm is required to obtain adequate sensitivity. It is probably impossible to find enough space inside the vessel or in the top shield for this sensor. Therefore, the cover gas pressure will probably be measured out of the vessel by means of such a large sensor connected to a large (about 2-in. diam) pipe connected to the reactor cover gas space.

In summary the overall plan for making these in-vessel bulk measurements requires six 5-in. diam penetrations of the vessel cover and shield. Refer to SK-3-14540 and 41 for the proposed location of the top shield penetrations.

5.7 CONTAINMENT

The overall containment concept is shown in SK-3-14425 and SK-3-14426 and utilizes a double barrier to maintain radionuclide leakage to the atmosphere to concentrations specified by 10 CFR 100 criteria for a Design Basis Accident. The outer and final barrier to the environs is a steel pressure vessel either cylindrical or spherical in shape with a leakage rate of less than one volume percent per day at design pressure. For the diameters required, a cylindrical vessel is limited to about 20 lb/in.²g and a spherical vessel to about 35 lb/in.²g before stress relieving in the field is required. The vessel is designed in accordance with Section III of the Nuclear Pressure Vessel Code as a Class B vessel and is to be tested initially to determine its leak rate and at periodic intervals thereafter to ensure its containment capability.

Personnel and equipment access through the containment vessel is normally via air locks. A large equipment access door, normally sealed, is available during special controlled shutdown conditions. Penetrations for piping and wiring are sealed in a positive manner to prevent leakage. Ventilation and piping are provided with double valves.

The inner containment barrier is the reactor machinery dome (See SK-3-14495). The steel liner of the reactor cavity vault and the Heat Transport System cells offer protection from accidents which may originate in the reactor core and primary sodium system. Accidents in a closed loop cell are isolated to that particular cell.

The reactor machinery dome is a steel pressure vessel which is positioned over the top face of the reactor during reactor operation which has an air atmosphere. Its purpose is to contain energy released from the reactor vessel via the vessel head as a result of an accident and to contain sodium air reactions which may result from sodium injection through the head. It also provides missile protection to the outer containment vessel with its energy absorbing materials inside the dome. The reactor machinery dome is hemispherical in shape and about 33 feet in diameter. It is removed during shutdown for maintenance and refueling operations. A manhole access provides entry to the reactor top face without removing the structure.

The reactor cavity vault is a cylindrical concrete structure which houses the reactor vessel. It is steel lined to prevent leakage and designed for high pressure, similar to the reactor dome. The reactor cavity vault is common to the three Heat Transport System cells; temperature and pressure originating in the vault equalize with these adjoining cells. The atmosphere in the reactor cavity vault and the HTS cells is normally nitrogen so as to minimize the possibility of a sodium-air reaction. The concrete is thermally insulated and will have embedded cooling coils to supplement the inert gas cooling and limit temperatures within the structural concrete.

Instrumentation and Controls for the Containment System shall operate as follows:

- . They shall monitor the temperatures and pressures within the containment vessel and the radionuclide concentration in the air leaving the vessel, and provide signals to initiate containment in the event of an abnormal energy or radioactivity release.
- . Containment shall be initiated as either ventilation containment or total containment.
- . A reactor shutdown shall be initiated by either the ventilation containment actuation signal or the total containment actuation signal, whichever occurs first.
- . Exhaust and supply ventilation valves shall close simultaneously upon indication of radionuclide concentrations in the exhaust air which exceed a preset limit.
- . All containment valves not essential for safe reactor operation shall close automatically upon an increase in pressure or temperature beyond a preset limit.
- . Individual cells within the containment vessel, in which a potential for a serious contamination incident exists, shall be isolated and shall be monitored for temperature, pressure and airborne radionuclide concentration.

SECTION 6.0 NUCLEAR SAFETY ASSESSMENT

6.1 OPERATIONAL ASSESSMENT

6.1.1 Accident Prevention/Mitigation

6.1.1.1 Effect on Core Parameters

The core characteristics related to safety have not been analyzed in detail for the vertical concepts. Two of the most important parameters are expected to improve or change in the direction of increased safety: The Doppler coefficient will increase somewhat for the vertical concepts since the enrichment can be lowered slightly, and the sodium void effect is expected to be slightly less positive for voiding in the central region of the core because of the increased leakage resulting from the smaller core volume.

6.1.1.2 ECCS Requirements for In-Vessel Storage

In-core fuel storage for this concept will place an additional, but essentially insignificant, burden on the ECCS system in the event that the normal cooling capability is lost. The ECCS system must be capable of removing the decay heat in both the core and the fuel stored in the lazy susans.

Approximately 20 fuel subassemblies will be removed from the core with goal exposure and stored in the lazy susans until the next refueling shutdown. If a refueling shutdown of 20 days is assumed and the fuel removed corresponds to 100 MW of goal exposure FTR fuel, the decay heat produced will be 100 kW at the time of startup. The ECCS system must be sized to contend with approximately 12 MW of decay heat which is produced in the core shortly after shutdown. The added heat load of the fuel stored in the lazy susan acquired from its previous in-core exposure is, therefore, insignificant.

Additional heat will be generated in the lazy susan fuel resulting from coupling with the core. The amount of power produced by this effect has been calculated at 30 kW per element. For 20 elements this gives a heat source of 600 kW during full power operation. If it

is then assumed that the decay heat produced immediately after shutdown is equivalent to that of the core (3%), the ECCS system would be required to remove an additional $0.03 \times 600 \text{ kW} = 1.8 \text{ kW}$ of heat. Again, this is insignificant compared to the ECCS design capability of approximately 12 MWt.

6.1.1.3 Subassembly Hydraulic Balance

This concept shows a hydraulic balance design which incorporates individual low pressure plenums for each driver fuel assembly. This design has the advantage that failure of any holddown feature caused by plugging or mechanical deformation will affect only a single assembly and not all assemblies simultaneously. However, detection of failure of any one, or limited number of holddown features constituting a "Single Unlikely Fault" will require further evaluation during preliminary design. Instrumentation to signal the first failures and thus initiate corrective action before the occurrence of additional failures is required to demonstrate an independence of the failures.

During operation, redundant holddown is provided mechanically through the upper instrumentation support plate. Failure of any hydraulic balance feature will allow the fuel assembly to move upward until it reaches this mechanical restraint. Any upward movement would result in a loss of reactivity which could potentially be gained by subsequent downward movement upon some flow reduction. The number of assemblies which may move upward before exceeding the limits of a "Single Unlikely Fault" depends upon the reactivity worth per element and the allowable distance to the upper restraint. On the average, movement of a single element will be worth $\sim 5\text{¢}/\text{in}$. If assemblies were allowed to move without some positive indication (other than a slow drift in the core reactivity), the design safety criterion requiring two independent holddown features would not be fulfilled since any subsequent failure of the mechanical restraint would release fuel.

Although this particular design is shown, it is not believed to be concept dependent. Application of a common low pressure plenum for all assemblies has the disadvantage in that conditions which affect the pressure in the low pressure plenum would affect all assemblies simultaneously; however, such a design would enhance fast response instrumentation for signaling the initial fault condition.

6.1.1.4 Control Drives Through Top Head Plugs

The top head design for this concept shows control rod penetrations through the refueling port plugs. Since the reactivity-control element drives are supported by the top head, any movement of the head relative to the core midplane would result in effective control withdrawal. Movement of this type may be quite significant - up to $\sim 1\%$ of movement for several rods.

It is questionable whether the top head plugs can be designed to resist upward movement under impact loads to the same extent as the top head itself. Therefore, to minimize any additional potential reactivity feedback by movement of the plugs during severe accident conditions, it is recommended that, if possible, none of the reactivity-control element drives be supported from the plugs in the top head, but by the top head itself unless it can be shown that plug movement does not make a significant contribution to the energy release from an accident.

6.1.2 Design Basis Accident Considerations

6.1.2.1 In-Vessel Fuel Storage

The energy release resulting from the DBA is not expected to increase because of the in-vessel storage of approximately 20 fuel assemblies.

Calculations of the number of FFTF fuel pins required to produce a critical mass show that at least nine fuel subassemblies must be arranged in a hexagonal array to achieve a critical mass with a 56 mil pin spacing. If the spacing between pins is decreased to zero, approximately seven subassemblies are required to total 1535 fuel pins.

Each lazy susan is expected to store 6 to 7 fuel subassemblies during reactor operation. Thus, in order for the lazy susan to obtain a critical mass during a DBA, the blast must rearrange the fuel subassemblies from an annular array to a closed packed hexagonal array and, in addition, the fuel pin spacing between all pins must be decreased to zero. It should not be difficult to design the lazy susan to prevent this occurrence.

6.1.2.2 Blast Resistance of Top Head

A major containment problem for the FFTF is restraining the vessel cover from becoming a missile during a DBA. A small percentage of the DBA theoretical work can be transmitted to the cover by a coolant slug which has been accelerated by a fuel and sodium vapor bubble generated in the core region from a DBA type accident. The cover will experience peak loadings from the slug impact followed by a long time, small pressure loading from the gas bubble.

Prior work has demonstrated that the cover can be restrained after displacement of 1 to 2 ft with some combination of certain features incorporated in the vessel design. Some of these design features include: (a) a vessel wall that yields significantly under DBA conditions, (b) a vessel wall attached to the cover, (c) an outer ring of holddown bolts attached to the foundation, and (d) a vessel which is not prevented from accelerating downward. The necessity of all of these features has not been shown; however, as each solves specific problems.

A vessel wall which yields significantly provides expansion volume in the core region which decreases the slug driving pressure. Also, when the slug impacts the cover, wall deformation in the upper vessel absorbs a major portion of the slug kinetic energy.

The outer ring of holddown bolts attached to the foundation provides restraint and energy absorbing capability for the energy transferred to the cover by the slug impact.

This concept, as presently shown, appears to satisfy the nonrigid wall feature and the cover is attached to the foundation. The vessel is not bolted directly to the cover, however, this is not considered a concept-related item and could be changed on subsequent designs.

The main concept-related feature of concern is the vessel cover with its large ports and heavy plugs. The strength of the cover must be carefully analyzed because of the large holes with smaller openings interspersed. Whether or not it can withstand a DBA must be evaluated further as the design is refined.

The heavy plugs are potential missiles. The present design does not detail holddown devices that will absorb potential DBA energies. Again, design changes to plug, holddown devices, along with further definition of energy transfer from the slug to the head could alleviate this problem.

A missile barrier placed inside the machinery dome over the core could provide final protection against dome penetration by small and large missiles. It is considered doubtful, however, if the barrier can withstand impact of three massive top head plugs and it should be designed in such a manner that the plugs remain rigidly fastened to the cover throughout the DBA conditions.

6.1.2.3 Sodium Expulsion/Fire Resistance

This concept shows the region above the upper head enclosed within a machinery dome during reactor operation. The gas region under the machinery dome may be rendered inert ($< 2\% O_2$) to minimize any potential sodium interactions for severe accident conditions in which sodium may spray into this region. Even if air is maintained in this region, the machinery dome provides a barrier against further sodium release into the containment regions and the design pressure of the machinery dome would be determined by the DBA and any associated sodium fires. If the reactor vessel head and refueling covers remain fixed, then the vessel itself will fail, spilling sodium into the reactor cavity. This volume is also rendered inert so that this sodium fire hazard is minimized.

6.1.2.4 Radiological Aspects

The final barrier against potential release of fission products or vaporized fuel consists of a cylindrical steel vessel surrounding the reactor and primary coolant system. In addition, this concept provides for a machinery dome over the reactor to guard against possible release of sodium to the containment vessel air atmosphere in case of a core disruptive accident of an explosive nature. Preliminary radiological analyses for the 300 N site have indicated that a single shell containment vessel with a 0.1%/day leak rate at design pressure would provide adequate protection for the general public in the event of a disruptive accident to the reactor core. Equivalent protection would be provided by a 10%/day leak rate of the inner barrier and a 1%/day rate in the final barrier.

The inclusion of a machinery dome over the reactor, as depicted in this concept, serves to reduce the pressure requirements of the containment vessel by eliminating the potential release of sodium to the air atmosphere of the containment vessel. In addition, the machinery dome constitutes an additional barrier against the potential release of radioactive material in the event of a disruptive accident.

Evaluation of the containment system is based on the capability of the system to limit the potential radiation exposure of the general public to the criteria of 10 CFR 100 in the event of a disruptive accident.

The Hanford reservation boundaries are approximately three miles to the east and five miles to the south of the selected FFTF site. At these points of interest, the potential dose from release of radionuclides from the containment vessel are substantially less than the criteria of 10 CFR 100. In conclusion, it is felt that the containment choice for this concept is adequate to protect the general public from accidental release of radionuclides as a result of a disruptive accident to the reactor core.

6.2 SHUTDOWN ASSESSMENT

6.2.1 Core Disconnect Problems

6.2.1.1 Disconnect of Reactivity-Control Elements

A safety criterion which was recommended early in the conceptual design phase of the FFTF was that provisions would be made to allow either the capability for cocked safety rods during refueling or the capability for periodic reactivity checks. This concept requires disconnection of all reactivity-control element drives during some refueling operations and disconnection of at least 1/3 of the rods during other operations. Some problems associated with rod disconnection are:

- Possibility of a sticky disconnect and withdrawal of the rod with removal of its drive or extension.
- Problem of location and reliability of a rod full-in sensor which must remain operable throughout refueling to verify its position.
- The mechanical complexity of the control rod drive is sufficient to cause concern about a drive's operability if frequent disassembly-assembly operations are necessary. From the safety standpoint reliable rod operation is of utmost importance and this will be difficult to achieve without a significant sacrifice of availability for testing, maintenance, and repair.
- Removal of the rod drive and/or extension constitutes loss of one of the two independent rod holddown features. However, during refueling, both accidental increase of the coolant flow rate and failure of flow orificing to the rod must occur simultaneously to initiate rod expulsion from the core.

The reactivity worth of individual rods may be 1-2% $\delta k/k$. Therefore, expulsion of any rod would constitute a major reactivity increase and, if not initiating criticality, would significantly affect the shutdown margin. The conceptual requirement for disconnecting reactivity-control rods during refueling constitutes a significant drawback of this conceptual design and further design work should be directed toward improving the safety aspects of this concept feature.

6.2.1.2 Loss of Backup Mechanical Holddown

Removal of the instrumentation support plate above the fuel assemblies for refueling removes the mechanical backup holddown for the subassemblies in that sector. However, even with failed hydraulic balance features and the instrumentation package removed, an increase in the shutdown coolant flow rate from 10 to 20% of full-power flow to greater than 50% would be required to lift any fuel assemblies. The lack of adequate instrumentation to sense plugged balance tubes could result in major core movements with only a single fault -- that of increasing the shutdown flow rate. Therefore, means of detecting loss of hydraulic balance is required to ensure that multiple-independent faults must occur before core movements which affect reactivity.

6.2.1.3 Instrumentation Movement

This concept requires that all in-core instrumentation be moved during a refueling shutdown. From the safety standpoint this is not desirable since the more instrumentation that is disturbed the greater the possibility of misplacement and/or damage.

The movement of the weight and bulk of a plate involving 1/3 of the in-core instrumentation may make it difficult to ensure that no damage has been done to the instrumentation tree or the in-core components. Presumably, it can be ensured that all instrumentation is in proper working order prior to startup, with some sacrifice of availability, but assurance that no damage has occurred to other in-core components may be difficult.

Some of the instrumentation above the core may be protective and thus connected into the scram system, i.e., thermocouples and perhaps, flowmeters. However, if the instrumentation fails in any one channel and must be replaced, this concept provides the capability for replacement without disturbing the complete instrument tree.

6.2.1.4 Radial Clamping Effect

This concept requires relief of the core radial clamping device before refueling and, therefore, the core must be clamped together following

refueling and prior to criticality. During the clamping procedure, a gradual reactivity addition will be effected by reduction of the core radius. It is estimated that radial compaction of the core of about 0.7 in. will increase the reactivity about 4.5 to 5.5\$. This is a major reactivity increase which occurs after core reloading operations, and is an important consideration for maintaining reactivity shutdown margins. Consideration should be given to limiting the use of the core clamping device to times when the machinery dome is in place.

6.2.2 Under-Sodium Fuel Transfer

One significant advantage of this concept related to fuel movements is that the transfer of assemblies from the core to initial storage is performed with the fuel continuously submerged under sodium. This alleviates any cooling problem during transfer from one media to another.

6.2.3 Shutdown Criticality Accident

The recent designs of the FFTF have been performed based on preclusion of the shutdown DBA. Since the occurrence of a shutdown DBA will have severe consequences, possibly including failure of the outer containment barrier resulting from a sodium spray fire combined with the DBA blast, it is imperative that adequate means be provided to ensure that the reactor is subcritical during fuel handling operations, initially through the use of intermediate criticality checks.

Subsequently, it may be possible to conduct refueling operations without criticality checks once the following criteria have been satisfied:

- (a) A highly reliable subcritical monitoring system is operational.
- (b) The NPTF has been built and is operational to decrease uncertainties in rod, fuel, and experiment worths prior to installation in the FTR.
- (c) The shutdown margin has been verified experimentally and shown to be of a sufficiently large magnitude to override reasonable cumulative tolerances and the occurrence of multiple "Single Unlikely Faults".

It is difficult to ensure that (a), (b), and (c) will be provided during refueling operations; especially with large amounts of reactivity being moved.

At each refueling shutdown of the FTR, large numbers of in-core components will be moved or replaced involving a large amount of reactivity. The total reactivity moved during a normal shutdown, including fuel element shuffling, test loop changeout, and control and safety rod replacement, is estimated to exceed 100\$. In addition, the possibility of fuel misplacement must be considered. On the average, an error resulting from the placement of a high-enrichment fresh element in the center zone rather than in the outer zone results in a decrease in the shutdown margin of 1\$.

Because of the large amounts of reactivity being moved, the possibility of errors resulting from misplacement of fuel, and the uncertainties in methods that might be available to verify subcriticality, it must be presently assumed that intermediate refueling criticality checks will be performed.

During later stages of design, consideration should be given to maximizing the shutdown margin by design of high worth safety rods. The under sodium fuel handling devices must also be designed to minimize the possibility of errors resulting in misplacement of fuel and/or control rods.

6.2.4 Sodium Fires During Shutdown Operation

The possibility of a large sodium spray fire occurring during shutdown, when the machinery dome is removed, appears to be remote. Design measures must be taken to ensure that sodium spray fires cannot occur during refueling which would cause a pressure rise above the design pressure of the outer containment. The loop handling machinery (LHM), which is a sodium cooled machine, appears to be the most hazardous from the sodium fire standpoint.

The occurrence of a sodium pool fire in one of the primary heat transport cells during maintenance will have to be designed against, but this appears to be non-concept dependent, affecting only the containment design.

SECTION 7.0 OPERATIONS AND MAINTENANCE EVALUATION

7.1 INTRODUCTION

A maintenance and operations evaluation of the top head area and associated servicing machines for this concept was made to identify problem areas, assess operability and maintainability capabilities, recommend solutions, and to determine if there are any major operational or maintenance problems with respect to this concept.

This portion of the report has used the same assumptions as described in Section 8.0 Availability Appraisal in developing the sequential steps and time increments involved in top head activities.

7.2 SUMMARY

This study resulted in the identification of a number of potential problem areas or areas of concern most of which resulted from a lack of design details. There were no problem areas or concept feature complications identified that were considered nonsolvable or that would render the concept unacceptable from an operations and maintenance standpoint.

7.2.1 Refueling Sequence

7.2.1.1 Driver Fuel and Removal Core Components

The basic steps required to accomplish a refueling outage for the cycle reference concept are shown in Drawing SK-3-14060. In developing this cycle, detailed listing of the sequential steps involved and a time study were prepared. A similar drawing, SK-3-14061, is shown for the concept alternative which utilizes separate instrument and refueling plugs. Both options are operable and maintainable.

7.2.1.2 Open and Closed Loop Refueling

A separate fuel handling machine is required for replacing the open and closed loop experiments or for changing out the in-core closed loop tube. Separate floor valves are also required for the open and closed loops.

The closed loop cross arm piping change-out discussed earlier in Section 5.0 of this report, appears to be the greatest maintenance problem

identified. It is expected that satisfactory solutions to this problem can be attained during detailed design.

7.2.2 Maintainability of Reactor Mechanisms

The maintainability of this concept has been analyzed in terms of the preliminary nature of the drawings prepared, recognizing that judgment of the maintainability of the concept should try to highlight and compare problems inherent in the overall design approach rather than in features as shown on the preliminary concept drawings. Consideration has been given only to maintenance problems of the equipment at the reactor itself. Secondary heat removal systems and other ancillary support systems (both radioactive and non-radioactive, with and without sodium or other contaminants) have not been discussed in this analysis because in general their problems will be common to any concept.

The relative maintainability of the various items of equipment in this concept is highlighted in terms of the red triangle, orange square and green circle display technique. This method was chosen because it permits general characterization of the elements of the concept without requiring listing of maintenance problems, resulting in exactness that is inappropriate because of the preliminary nature of the drawings. The legend of Figure 7.1 explains the criteria used to denote which category the maintenance problems of a particular piece of equipment might fall within.

The maintainability analysis includes consideration of the extent to which maintenance facilities are affected by the concept, in that a conceptually complex machine, or one that includes features clearly beyond the state of the art, will contribute to the amount of maintenance work that must be done, and will probably affect the scope of the facilities required to do that work. Thus, the complex gear train shown working under sodium in the in-vessel FHM helps to characterize that piece of equipment as one with relatively poor reliability and high failure rates, and one that will require extensive and costly facilities for its repair. Another design might show some more

V-A MAINTAINABILITY

- 1 FULL REMOTE MAINTENANCE SYSTEM REQUIRED
- 2 MAXIMUM IMPACT ON PLANT CAPITAL COSTS
- 3 EXTENSIVE MAINTENANCE DEVELOPMENT PROGRAM REQUIRED
- 4 EXTENSIVE DEVELOPMENT OF SPECIALIZED PROTOTYPE MAINTENANCE EQUIPMENT REQUIRED
- 5 MAJOR IMPACT ON PLANT AVAILABILITY BECAUSE OF MAINTENANCE CYCLE REQUIREMENTS
- 6 SEVERE CONTAMINATION CONTROL PROBLEMS

- 1 FULL DECONTAMINATION REQUIRED, BUT CONTACT MAINTENANCE POSSIBLE
- 2 NOMINAL IMPACT ON PLANT CAPITAL COSTS
- 3 MINIMUM MAINTENANCE DEVELOPMENT PROGRAM REQUIRED
- 4 SPECIAL PURPOSE ENGINEERED EQUIPMENT REQUIRED FOR MAINTENANCE
- 5 SIGNIFICANT IMPACT ON PLANT AVAILABILITY BECAUSE OF MAINTENANCE CYCLE REQUIREMENTS
- 6 RIGID CONTAMINATION CONTROL REQUIREMENTS

- 1 MANUAL ACCESS CONTACT MAINTENANCE IN SITU POSSIBLE
- 2 MINIMAL IMPACT ON PLANT CAPITAL COSTS
- 3 NO MAINTENANCE DEVELOPMENT PROGRAM REQUIRED
- 4 NORMAL MAINTENANCE TOOLING REQUIRED
- 5 ACCEPTABLE AVAILABILITY ACHIEVABLE IN TERMS OF MAINTENANCE CYCLE REQUIREMENTS
- 6 NO UNUSUAL CONTAMINATION CONTROL PROBLEMS

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FIGURE 7.1

reliable and less complex method of transmitting the forces and motions required. In anticipation of this, then, the display technique only denotes an overall trend toward greater or lesser maintenance problems that might finally result with equipment of the general kind required for service in this concept.

7.2.3 Maintainability of Reactor Top Head Area

Top head features related to maintenance have been considered by the same method as previously described in Section 7.2.2 for the reactor refueling and instrumentation equipment.

Top head maintenance features will be improved as the design is further developed, and such improvement will affect operations, plant availability, and overall maintainability. The present concept is predicated upon the assumption that contact maintenance modes can be used for all of the drive mechanisms and control hardware located above the reactor top head. Contamination control is an inherent problem in this design which must be considered in all operations and maintenance tasks that breach the reactor vessel, the closed loops and other radioactive systems. No definitive solutions to these problems have been shown, but it is expected that as the design develops and the details of maintenance work are exposed, adequate contamination control measures can be found that will not jeopardize the overall plant operations.

7.2.4 Instrumentation (Maintenance)

7.2.4.1 Reactor Core

The instrumentation support assembly shown in Drawing N68181255 consists of a rotating plug in the reactor cover, a shaft from the plug to a bottom guide located near the top of the fuel subassemblies, and a platform attached to the shaft above the fuel subassemblies from which the instrumentation probes extend into the fuel subassemblies. One-third of the reactor core is serviced by each of the three instrumentation support assemblies which are designed to be rotated and lowered into the fuel subassemblies prior to core clamping.

- A. Reactor Core Instrument Support Assembly - It appears that the design concept can be made to work with proper design and development. The problems of alignment and positioning for core clamping have not been fully evaluated. Any repair of the instrument probes and support plate will require that the instrument support assembly be removed from the reactor.
- B. FEDAL System - Satisfactory repair and replacement of any damaged FEDAL line can be accomplished by replacing the particular damaged assembly in the reactor or trench. All three of the FEDAL assemblies are removable by use of fuel handling methods.
- C. Driver Fuel and Test Assembly Instrument Packages - It appears that the instrumentation design concept can be made to work by proper design and development. Replacement of the reactor core instrumentation for the individual driver ducts is possible without removing the entire instrument support assembly from the vessel. If removal problems occur in removing the instrument bundle, the entire instrument support assembly can be removed. Replacement of an open or closed loop instrument requires replacement of the entire open or closed loop assembly.
- D. Nuclear Instrumentation - Repair and replacement of the flux monitor sensors is possible for an electrical read-out instrument system and is not exceedingly difficult. Radiation shielding of the flux monitor cover ports can be accomplished with state-of-the-art techniques. Movement of the sensors will be accomplished by standard methods and presents no major maintenance problem. Replacement of the flux monitor encapsulating tube can be accomplished, but, in the case of the open and closed loops, requires replacement of the complete open or closed loop assembly.
- E. Reactor Cover Instrumentation - All reactor cover instrumentation connectors can be manually repaired with present state-of-the-art methods and without exposing personnel to sodium vapor. Connector contact resistance can be maintained by standard techniques. Removal of a driver duct instrument bundle requires special handling because of the radiation levels, containment requirements, and temperatures encountered. It

appears that this removal concept can be made to work by proper design and development.

7.2.5 Contamination Control of Head Area

The basic approach to providing contamination control for the head area is through design features incorporated into the top head area and top head servicing equipment such as:

- . Removable floor plates
- . Removable catch pans provided in area of the Closed Loop Cross Piping (SK-3-14456)
- . Providing gas seals at all penetrations
- . Floor valves sealed to the top face penetration and sealed to the core component handling devices
- . Drip pans on all fuel and component handling machines
- . Closure valves on all fuel and component fuel and component handling devices
- . Gas coolant pressures and exhaust control on all component and fuel handling machines.

All of the above considerations are shown in the conceptual design for the top head area. Obviously, they are not in final detail and extensive attention must be given to these areas as detailed design progresses. Contamination control considerations are included in the discussion of each portion of the Section 7.0 report.

A major area of concern exists in regard to the FEDAL system as shown on Drawings N68181251, SK-3-14519, and SK-3-14540. It appears that extensive design refinement will be required to provide adequate contamination control capability for removal and replacement of these lines or for the repair or unplugging of these lines. The use of removable pans in the FEDAL trenches, the re-routing of the lines or relocation of the top reactor FEDAL plugs might be considered in future design layouts.

In addition to the above-mentioned design features, certain operational and maintenance procedures must be employed to minimize contamination spread in the top face area. Examples are:

- . Incorporation of "bagging" procedures wherever possible to confine immediately any contamination inside a suitable enclosure as the FEDAL lines are removed from the reactor.
- . The use of floor covering for all areas immediately adjacent to the work zone with suitable plastic to prevent the deposition of contamination on these areas.
- . Proper ventilation control (facility pressure differential control, air flow control, etc.) in the area of potential contamination release in such a manner that the spread is limited to small areas.
- . Incorporation of proper preventative maintenance procedures to ensure that all contamination control features are in proper condition to afford the protection required.
- . Personnel trained in contamination control and cleanup procedures.

In summary, it appears that contamination control can be adequately provided by proper design incorporated with procedural control. It is, however, an area of major concern and will require constant attention, especially during the development and design period.

SECTION 8.0 AVAILABILITY APPRAISAL

8.1 INTRODUCTION

The major effect of any reactor concept on plant availability is the ability to perform fuel, test specimen and reactor component change-out in a minimum of reactor outage time. The effect of reliability of other plant process equipment, systems or control is not directly dependent on the reactor concept so have not been considered in this appraisal. The assessment of plant availability has, therefore, been limited to the time required to perform the normal fuel and in-core component replacement operation, routine reactor maintenance and the effect of those expected incidents causing unplanned outages. This report does not include assessments for unusual maintenance items such as stuck fuel, grid replacement, etc.

The availability values in this report are not to be construed as an indication of true plant availability, but are numbers to assess the concept relative to other concepts previously evaluated.

8.2 SUMMARY

The relative plant availability for this concept design incorporating three rotating plugs for refueling is 61.6%. This base case and the effect of alternative considerations are summarized in Table 8.I.

This concept has the advantage of permitting other work to be performed on the reactor head simultaneously with the fuel handling operation in another sector. It permits in-vessel storage of used fuel to allow decay during an operating period. Removal and replacement of fuel from the lazy susan is independent of the in-core refueling operation.

The concept is penalized by the time required to clear the reactor head to prepare for the refueling operation and the alternative replacement and removal cycles required to perform intermediate criticality checks, although only two intermediate checks are currently planned. These will be performed after refueling operations in each sector with the last sector being done during

reactor startup. Previous studies included three intermediate criticality checks: (1) after rod change-out, (2) after test specimen change-out, and (3) midway in replacement of the driver fuel. If the two intermediate approaches to criticality prove unnecessary (Case C), the plant availability will be increased to 65.0% since six days are assessed to this operation.

If fuel experience permits extending the operating cycle to three per year, i.e., 84 days per operating period (Case D), the availability would be 62.1% assuming critical checks. Changing out the center located loops each operating cycle reduces the availability about 2%. The use of three removable plugs (Case F) reduces the availability to 57.5% for the 63-day operating cycle.

8.3 ASSUMPTIONS

The relative availability for this concept has been based on the following assumptions which are consistent with those used in previous studies.

- A. The NPTF will be operational and all core components except the reflectors will be checked, calibrated, and their reactivity worth measured prior to introduction into the FTR core.
- B. All core components subjected to the core flux must be replaced after approximately 260 days of exposure because of current fluence limits.
- C. An average operating period will be 9 weeks (63 days). The average refueling cycle will consist of:
 - a. Replacement of 19 fuel assemblies
 - b. Shuffling of 19 fuel assemblies
 - c. Replacement of 15 reflectors
 - d. Replacement of six control or safety rods
- D. Critical measurements will be made following core component change-out and shuffling in each section of the trisected core and the loop change being performed concurrently.

- E. No allowance is made for equipment problems during the outage operation.
- F. Personnel are available and efficiency is considered at 100%.
- G. Thirty scrams will occur per year. Each of these will require an average of four hours for recovery. This will average 7 1/2 scrams per operating period with the attendant loss of 30 hours operating time if four cycles per year are assumed.
- H. There will be seven to 10 operational interruptions because of experimental variables, either experimental failure or the experimenter's need to inspect his experiment. Some of these interruptions will initiate a normal core component change-out outage, but some will not. For the purpose of this appraisal, it is assumed that operation is interrupted five times during a year, or an average of 1 1/4 operational interruptions which will require access to the top face during an operating period.
- I. A machinery dome will be required to be in place for all reactor approaches to critical. It is assumed that removal and replacement can be accomplished in a 24-hour period.
- J. The reactor is considered operational any time the power level is greater than zero, since test information may be generated at any power level greater than "0".
- K. Cover gas for fuel and core component handling equipment is the same as the reactor cover gas.
- L. Separate FHM or cask will be required to handle closed and open loop components. It is assumed that the driver fuel and control rod handling mechanism are not tied to loop component handling.
- M. Because of nozzle damage from repeated seal welding, the cross-arm piping will require a change-out every 2 1/2 years. For the six loops a change-out will be required every other outage period.

- N. Loop cross-arm piping and electrical leads are below deck and need not be disturbed for the discharge-charge-shuffle operation.
- O. The FHM machine contains three barrels capable of handling finned pots. It will move only between the lazy susan port and decay storage or radioactivity maintenance ports.
- P. It is assumed that there will be adequate crew size (manpower) available to simultaneously work a loop D-C and core component discharge-charge-shuffle operation.

8.4 CASE STUDIES

In order to fully evaluate this concept based upon the foregoing assumptions, alternative operating conditions are considered. These include the following:

- Case A - 63 day operation - No center loop change-out.
- Case B - 63 day operation - All three center loops changed-out each outage.
- Case C - 63 day operation - No center loop change-out and no intermediate approaches to critical.
- Case D - 84 day operation - No center loop change-out.
- Case E - 84 day operation - All three center loops changed-out each outage.
- Case F - 63 day operation - Removable refueling plugs.

Major sequential steps and estimated times for the base case for each of the above are listed in attached Tables; 8.II, 8.III, and 8.IV.

The cases for change-out of the three center loops are considered since this operation must be independent of other refueling work because of machine interferences.

8.5 NORMAL REFUELING OUTAGE

The refueling operation includes replacement of an equal number of core components in each sector of the trisected core arrangement. In the time study, it has been assumed that equipment design and manpower availability will permit certain simultaneous operations rather than performing all work sequentially.

On this basis, the normal refueling outage is 552 hours or 23 days. Major sequential steps and estimated times are listed in Table 8.II.

8.6 OUTAGE MAINTENANCE

Outage maintenance operations are assumed to include those items that can be identified as requiring replacement or repair on a routine basis. The effect on plant availability is estimated at 92 hours per cycle.

Items considered in this category include:

- (a) Replacing control rod drives
- (b) Replacing instrument jumpers
- (c) Replacing closed loop cross-arm piping
- (d) Replacing driver instrumentation

8.7 UNPLANNED OUTAGE

Certain unplanned conditions are identified as having an effect on availability. These conditions average 300 hours of outage for each operating cycle and include (per outage):

- . Scram Recovery - 30 hours
- . Interim Examination - 180 hours
- . Closed Loop System - 90 hours

8.8 UNUSUAL MAINTENANCE

No downtime is considered in this study for this type of outage item. It is believed that this reactor concept lends itself to unusual maintenance repairs if provisions can be made to inert and shield the reactor head for this activity.

TABLE 8.1
RESUME AVAILABILITY APPRAISAL

CASE		A	B	C	D	E	F
OPERATING PERIOD	HOURS	1512	1512	1512	2016	2016	1512
	DAYS	63	63	63	84	84	63
CORE COMPONENT CHANGE	HOURS	552	660	420	708	816	728
	DAYS	23	27.5	17.5	29.5	34	30.3
MAINTENANCE OUTAGE	HOURS	90	90	93	127	127	90
	DAYS	3.75	3.75	~ 4	5.3	5.3	3.75
UNPLANNED OUTAGE	HOURS	300	300	300	396	396	300
	DAYS	12.5	12.5	12.5	16.5	16.	12.5
TOTAL	HOURS	2454	2562	2325	3247	3355	2630
	DAYS	102	107	97	135	140	110
RELATIVE PLANT AVAILABILITY PERCENTAGE		61.6	59.0	65.0	62.1	60.0	57.5

Case A 63 days operation - no center loop change out

Case B 63 days operation - all three center loops changed out each outage

Case C 63 days operation - no center loop change out and no intermediate approaches to critical

Case D 84 days operation - no center loop change out

Case E 84 days operation - all three center loops changed out each outage.

Case F 63 days operation - removable refueling plugs

TABLE 8.II ROTATING PLUGS (9 WEEKS OPERATING CYCLE)

TABLE 8.111 ROTATING PLUG 84 DAYS OPERATING TIME

END OF NORMAL DOWNTIME
BASIS (12) WEEK OPERATING PERIOD

TABLE 8.IV ROTATING PLUGS (9 WEEKS OPERATING CYCLE) NO APPROACHES TO CRITICAL DURING CORE COMPONENT CHANGES

SECTION 9.0 ASSESSMENT OF DEVELOPMENT IMPACT9.1 GENERAL

Overall review of the development requirements for this concept indicates the following broad impact:

- A. Greater development and testing will be required for the several in-vessel equipment items which must operate reliably in sodium. This equipment includes the instrument plate assembly and components, the in-vessel fuel handling machine, the lazy susan, and the core restraint mechanism.
- B. Driver fuel instrumentation components are substantially smaller than in previous concepts. As a result, substantial effort will be required to develop reliable, compact instrument sensors, connectors and electrical systems.
- C. The characteristic of the concept which requires close alignment of equipment installed in the reactor cover with the core; for example, the instrument plate and refueling machine will require greater emphasis on mockup testing on full scale, simulated thermal environment reactor equipment. A full scale, 1/3 segment model of the reactor vessel internals and cover is envisioned to be necessary.
- D. Extensive development and testing of the reactor cover and components is anticipated to provide reliable rotating plugs, bearings, seals and holddown mechanisms.
- E. Other currently identified areas requiring development not previously of equivalent emphasis include:
 - . Sodium sample FEDAL system, including the in-vessel multiple connections.
 - . Higher temperature application of B_4C control materials.
 - . Control rod drive and drive train mechanisms.

More comprehensive evaluation of detailed development requirements is planned.

APPENDIX ADRAWINGS

<u>No.</u>	<u>Title</u>
SK-3-14060	Operational Sequence, Three Rotating Plugs - FTR Concept
SK-3-14061	Operational Sequence, Removable Plugs - FTR Concept
SK-3-14400	Flexing Grid Plate Assembly, Hydraulic Balanced
SK-3-14425	Building Plan XX, Alternate Plan at Elevation 0'.0"
SK-3-14426	Building Plan XX, Alternate Section
SK-3-14456	Sheet 1 - Closed Loop, Double Containment and Encapsulation System
	Sheet 2 - Closed Loop, Double Containment and Encapsulation System
SK-3-14457	Closed Loop Spool Piece Assembly, Welded Closure
SK-3-14458	Open Loop Spool Piece Assembly, Welded Closure
SK-3-14459	Carrousel, In-Vessel Storage Arrangement
SK-3-14495	Reactor Machinery Dome
SK-3-14496	Handling Equipment - Arrangement Plan
SK-3-14497	Handling Equipment - Arrangement Section
SK-3-14501	Core Restraint System, Cam Actuated
SK-3-14502	Core Restraint System, Toggle Actuated
SK-3-14503	Core Restraint System, Leaf Spring
SK-3-14504	Core Restraint System, Cam Actuated; Elev.
SK-3-14513	Open and Closed Loop Tests and Control Rod Ingeguments
SK-3-14514	Open Test Loop-Immersed Section
SK-3-14515	Closed Loop - In-Pile Tube Immersed Section

<u>No.</u>	<u>Title</u>
SK-3-14519	Reactor Cover Assembly
SK-3-14529	Poison Tip Section Control/Safety Rods
SK-3-14530	Rod Hanger Shaft and Gripper Assembly
SK-3-14531	Control/Safety Rod Driver Mechanism
SK-3-14532	Control/Safety Rod Head Arrangement
SK-3-14533	Driver Fuel Assembly
SK-3-14540	Reactor Arrangement
SK-3-14541	Reactor Head Plan and Section
SK-3-14542	Reactor Arrangement W/Bottom Entry Rod Option
SK-3-14546	Open and Closed Loop Instrument Connector
SK-3-14547	Open and Closed Loop Instrument Connector
SK-3-14548	FEDAL Location System Arrangement for Drivers
SK-3-14567	Reference Core Map
SK-3-14568	Primary Loading Beam, Radial Restraint
N68181185	Loop Preparation Cask
N68181217	Indirect Gas Cooled Fuel Handling Machine
N68181222	Sodium Cooled Closed Loop Handling Machine
N68181248	IVHM Removable Drive Unit
N68181249	In-Vessel Handling Machine, Grapple Assembly
N68181250	Instrumentation Shield Plug Region, FFTF Three Plug Concept
N68181251	FEDAL Tube Configuration at Stationary Top Shield
N68181255	Instrument Support Assembly

<u>No.</u>	<u>Title</u>
N68181256	Instrumentation Assembly and Driver Fuel Alignment
N68181267	FFTF Instrumentation and IVHM Plug, Plan View
N68181268	Instrumentation Removal Cask

1. SHUTDOWN AND PREPARE FOR FUEL HANDLING

- COOLDOWN
- REMOVE MACHINERY DOME
- DISCONNECT ELECTRICAL JUMPERS
- REMOVE CONTROL ROD DRIVES & SHAFTS AS REQUIRED
- RAISE INSTRUMENT PLATE AND ROTATE TO "UNDER PLUG" LOCATION
- RELEASE CORE BUNDLING

2. PLUG EXCHANGE

- INSTALL FLOOR VALVE
- POSITION PLUG CASK & REMOVE INSTRUMENTATION PLUG
- CLOSE FLOOR VALVE
- PLACE INSTRUMENT PLUG IN STORAGE
- PICK UP FUEL HANDLING PLUG & PLACE IN THE REACTOR
- REMOVE FLOOR VALVE FOR FUEL HANDLING OPERATION

3. CHANGE OUT CONTROL RODS

- REMOVE AND INSTALL CORE COMPONENTS AND TEST SPECIMENS IN ZONE I

NOTE: CORE COMPONENTS INCLUDE DRIVER FUEL ASSEMBLIES, REFLECTOR ELEMENTS, CONTROL ELEMENTS, PACKAGE LOOPS, ETC.

4. CRITICALITY CHECK

- REMOVE THE FUEL HANDLING PLUG AND REPLACE THE INSTRUMENT PLUG (SEE PROCEDURE #1)
- LOCATE AND LOWER INSTRUMENTATION INTO OPERATING POSITION
- RETURN THE REACTOR TO NORMAL TO PERFORM APPROACH TO CRITICAL
 - INSTALL ALL CONTROL RODS
 - RECONNECT ELECTRICAL JUMPERS
 - BUNDLE THE CORE
 - INSTALL MACHINERY DOME
- PERFORM APPROACH TO CRITICAL

5. REFUEL DRIVER AND TEST ZONES II & III

- REPEAT PROCEDURES #1, #2, #3 AND #4 FOR ZONES II AND III

CORE PLAN

6. RETURN TO OPERATION

- FOLLOWING THE APPROACH TO CRITICAL FOR THE FINAL ZONE
 - COMPLETE START-UP CHECKS
 - PROCEED TO POWER

CONCEPTUAL

3 RD ISSUE 12-12-68
2nd issue 12-4-68
1st issue 11-22-68

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DATE		F.T.R. CONCEPT
SCALE		FAST FLUX TEST FACILITY
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APPD			FLEXING GRID PLATE ASSEMBLY	
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2	12/1/68	FOR CONFORM	W. T. DEB	12/1/68	W. T. DEB	W. T. DEB	REFERENCE DRAWINGS

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SECTION B-B

SECTION A-A

SECTION C-C

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DRAWN BY J. W. HOLB		DATE 12-6-68	CLOSED LOOP-DOUBLE CONTAINMENT AND ENCAPSULATION SYS. FOR B-A
CHECKED BY G. SMITH		DATE 12-6-68	FAST FLUX TEST FACILITY
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FOR DETAILS - SEE SHEET 1

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CLOSED LOOP-DOUBLE
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ENCAPSULATION SYS FOR 6A

FAST FLUX TEST FACILITY

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REACTOR MACHINERY DOME

DESIGN PRESS 100 PSIG
NET VOLUME 8600 CU FT
LIFTING WEIGHT 60 TONS

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DETAIL B
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BY: C.C. STEELE
CHKD: C.C. STEELE
DATE: 12/14/68

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PROJECT: FAST FLUX TEST FACILITY

DWG NO: 300-6

INDEX: 880101-990201

CLASSIFICATION: NONE

DWG NO: SK-3-14496

SHEET NO: 1

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CLASSIFICATION: NONE

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ACTUATING ROD

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3/8 INCH = 1 FOOT

CONCEPTUAL

SECTION AT A-A

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13	13	12-12-68	NOTED	JJ DARBY	12-12-68

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PARTIAL PLAN VIEW
SCALE - QUARTER

FUEL HANDLING MACHINE
SEE A-1 DWG N 68181249

TOP OF CORE SUBASSEMBLIES

ELEVATION AT B-B
SCALE QUARTER

CORE BARREL

FUEL HANDLING
PLUG

RELEASED MODE

DIRECTION OF
MOTION UNDER
CONSTRAINT LOAD

ACTUATED MODE

SECTION A-A
SCALE - FULL

ARRANGEMENT
SCALE - HALF SIZE

CONCEPTUAL

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APPROVED FOR CONFORMANCE BY TH DESIGN CRITER A	DATE	U S ATOMIC ENERGY COMMISSION
FOR		RICHLAND OPERATIONS OFFICE
APPRO		PACIFIC NORTHWEST LABORATORY
		OPERATED BY BATTTELLE MEMORIAL INSTITUTE
DESIGN G WAYMIRE	12-13-68	FTR CONCEPT V-A
OPTS APPRO		CORE RESTRAINT SYSTEM
CHECKED		LEAF SPRING
DRAWN DARB	12-13-68	FAST FLUX TEST FACILITY
SCALE NOTED		300 GEN
CLASIFIED		INDEX NO 1500.01
CLASIFICATION		DWG NO SK-3-14503
NONE		SHEET NO 1

BY DATE	FOR	BY DATE	FOR	REV	DESCRIPTION	REV	DATE
APPRO FOR CONFORM	W TH DES CRIT A	APPRO	DATE	DATE	REVISIONS	DATE	CHECK PRT ISSUE NO
REFERENCE DRAWINGS		COMMENT PRT ISSUE NO		DATE	DRAWING STATUS		
NEXT USED ON

NOTE:
THIS ASSEMBLY IS TYPICAL OR 6 POSITIONS.
SEE SK-3-14501 FOR PLAN VIEW, & SK-3-14540
(REACTOR ARRG'T ELEV) FOR O'ALL INSTALL-
ATION.

C O N C E P T U A L

[illegible]

15'-6"
FIRST SECTION
REMOVAL ACCOMPLISHED
WITH PERSONNEL
CONTACT ASSISTANCE

26'-8"
SECOND SECTION
REMOVAL ACCOMPLISHED
WITH SHIELDED CASK

2" O D ROD

5 3/8" HEX ACROSS CORNERS
OF FLOW CHANNEL WHICH
IS MOVED INTEGRALLY WITH
THIRD SECTION

10'-1"
THIRD SECTION REMOVED AND
TRANSFERRED TO LAZY SUSAN
UNDER REACTOR COVER

11'-6"

CONTROL ROD
REF SK-3-14484

3" O D INST STALK

4 7/8" HEX ACROSS CORNERS

14'-0" TEST DIA SAME
AS DRIVER

OPEN LOOP
REF SK-3-14514

NORMAL SODIUM
OPERATING LEVEL

43'-0" 20'-0" 4 1/2" DIA

4 7/8" HEX ACROSS CORNERS

2 1/2" TEST DIA

CLOSED LOOP
REF SK-3-14515

CONCEPTUAL 1ST ISSUE 12 12 68

APPROVED FOR CONFORMANCE	DATE	U. S. ATOMIC ENERGY COMMISSION
BY		RICHLAND OPERATIONS OFFICE
FOR		PACIFIC NORTHWEST LABORATORY
APPROVED		OPERATED BY BATTTELLE MEMORIAL INSTITUTE
BY		OPEN & CLOSED LOOP
DATE		TESTS, & CONTROL ROD
CHECKED		INTEGUMENTS
BY		CONCEPT V-A
SCALE	1/2" = 1'-0"	FAST FLUX TEST FACILITY
CLASSIFICATION	NONE	300 GEN 2200 01
		SK-3-14513 111

SECTION A-A

SECTION B-B

SECTION C-C

SECTION D-D

NOTE
HEX CONFIGURATION SHALL BE CONSTRUCTED TO CLEAR 5" DIA NOZZLE OPENING IN REACTOR COVER. TO ACCOMMODATE THIS REQUIREMENT SHROUD LATCH & INSTRUMENT LEADS MAY BE MODIFIED AS NECESSARY.

INSTRUMENT LEADS INCLUDE:
2 - FUEL PIN CLAD TEMP T/C's
4 - FUEL PIN TEMP T/C's
1 - FUEL PIN PRESS GAS BALANCE LINE (PRESS BALANCE DIAPHRAGM LOCATED IN TOP OF SELECTED PIN)
1 - FUEL PIN PRESS TUBE TO PRESS BELLOWS LOCATED ABOVE

DETAIL I
SCALE 1/4" = 1'-0"

CONCEPTUAL		121 ISSUE 12-12-68	
APPROVED FOR CONFORMANCE WITH DESIGN CRITERIA		DATE	
BY		U. S. ATOMIC ENERGY COMMISSION	
FOR		RICHLAND OPERATIONS OFFICE	
APPRO		PACIFIC NORTHWEST LABORATORY	
APPRO		OPERATED BY BATTTELLE MEMORIAL INSTITUTE	
ENGR K O CREEK		OPEN TEST LOOP	
DFTS APPRO		IMMERSED SECTION	
CHECKED		CONCEPT V-A	
DRAWN		FAST FLUX TEST FACILITY	
SCALE FULL		FILE NO 300 GEN	
CLASSIFIED BY		INDEX NO 2200 01	
CLASSIFICATION		NONE	
SK-3-14514		111

SECTION A-A

SECTION B-B

SECTION C-C

SECTION D-D

SECTION E-E

CONCEPTUAL		L 187 ISSUE 12 12 68	
APPROVED FOR CONFORMANCE WITH DESIGN CRITERIA		DATE	
BY	FOR	U. S. ATOMIC ENERGY COMMISSION	
APPROVED	MAVOGEL	RICHLAND OPERATIONS OFFICE	
CHECKED		PACIFIC NORTHWEST LABORATORY	
SCALE	FULL	OPERATED BY BATTHELLS MEMORIAL INSTITUTE	
CLASSIFIED BY	2200 01	CLOSED LOOP IN-PILE TUBE	
DECLASSIFIED BY	2200 01	IMMERSED SECTION	
NONE	SK-3-14515	CONCEPT V-A

CONCEPTUAL		1ST ISSUE 12-5-68	
U. S. ATOMIC ENERGY COMMISSION		RICHLAND OPERATIONS OFFICE	
PACIFIC NORTHWEST LABORATORY		OPERATED BY BATTTELLE MEMORIAL INSTITUTE	
POISON TIP SECTION		CONTROL/SAFETY RODS	
CONCEPT V(a)		FAST FLUX TEST FACILITY	
SCALE NOTED		300 GEN 1901.01	
5K-3-14529		11	

REV	DATE	DESCRIPTION	BY	CHKD	DATE	REV	DATE	DESCRIPTION	BY	CHKD	DATE
1	12/5/68	CONCEPT V(a)	W. JACKSON	NOTED	12/5/68	1	12/5/68	CONCEPT V(a)	W. JACKSON	NOTED	12/5/68

REFERENCE DRAWINGS	COMMENT PRT ISSUE NO	DATE	CHECK PRT ISSUE NO	DATE	LAST REV
NEXT USED ON	DRAWING STATUS

COLLET HANGER ROD

HANGER ROD WITH DRIVE DISCONNECTED

CONCEPTUAL						JUL 1970 ISSUE 12-14-68	
APPRO FOR CONFORMANCE WITH DESIGN CRITERIA			U. S. ATOMIC ENERGY COMMISSION RICHLAND OPERATIONS OFFICE				
BY _____ FOR _____ APPRO _____ APPRO _____			PACIFIC NORTHWEST LABORATORY OPERATED BY BATTELLE MEMORIAL INSTITUTE				
ENGR D. MARINOS <i>2/14/68</i>			ROD HANGER SHAFT & GRIPPER ASSEMBLY				
DRAFT APPRO _____ CHECKED _____			CONCEPT II (α)				
DRAWN BY J. JACKSON <i>2/14/68</i> SCALE NOTED			FAST FLUX TEST FACILITY				
CLASSIFICATION _____			PLANS NO. 300 GEN		INDEX NO. 1901-01		
CLASSIFICATION <i>SECRET</i> DATE _____			ENGINE NO.		SHEET NO. SHEETS		
NONE			SK-3-14530 11

CONCEPTUAL

1ST ISSUE 12-14-68

APPRO. FOR CONFORMANCE WITH DESIGN CRITERIA		DATE	U. S. ATOMIC ENERGY COMMISSION RICHLAND OPERATIONS OFFICE	
BY	FOR		PACIFIC NORTHWEST LABORATORY	
APPRO			OPERATED BY BATTELLE MEMORIAL INSTITUTE	
APPRO			CONTROL / SAFETY ROD	
INSTR. MARINOS	12-14-68		DRIVE MECHANISM	
DRAFT APPR			CONCEPT ∇ (a)	
CHECKED				
DESIGN W. H. DAVENPORT	12-14-68		FAST FLUX TEST FACILITY	
SCALE	NOTED			
CLEARANCE	12-14-68	BLDG NO.	300	INDEX NO. 1901.01
CLASSIFICATION		DWG NO.		
LAST REV.	1	SK-3-14531		SHEET NO. 11
CLASSIFICATION	NONE			

[illegible]

REACTOR ELEVATION

0 1 2 3 4 5 6 7 8 9 10 FEET

REV	DATE	BY	CHKD	DESCRIPTION
1	10/1/68	W. J. L. L.	W. J. L. L.	INITIAL DESIGN
2	10/1/68	W. J. L. L.	W. J. L. L.	REVISED DESIGN
3	10/1/68	W. J. L. L.	W. J. L. L.	REVISED DESIGN
4	10/1/68	W. J. L. L.	W. J. L. L.	REVISED DESIGN
5	10/1/68	W. J. L. L.	W. J. L. L.	REVISED DESIGN
6	10/1/68	W. J. L. L.	W. J. L. L.	REVISED DESIGN
7	10/1/68	W. J. L. L.	W. J. L. L.	REVISED DESIGN
8	10/1/68	W. J. L. L.	W. J. L. L.	REVISED DESIGN
9	10/1/68	W. J. L. L.	W. J. L. L.	REVISED DESIGN
10	10/1/68	W. J. L. L.	W. J. L. L.	REVISED DESIGN

REV	DATE	BY	CHKD	DESCRIPTION
1	10/1/68	W. J. L. L.	W. J. L. L.	INITIAL DESIGN
2	10/1/68	W. J. L. L.	W. J. L. L.	REVISED DESIGN
3	10/1/68	W. J. L. L.	W. J. L. L.	REVISED DESIGN
4	10/1/68	W. J. L. L.	W. J. L. L.	REVISED DESIGN
5	10/1/68	W. J. L. L.	W. J. L. L.	REVISED DESIGN
6	10/1/68	W. J. L. L.	W. J. L. L.	REVISED DESIGN
7	10/1/68	W. J. L. L.	W. J. L. L.	REVISED DESIGN
8	10/1/68	W. J. L. L.	W. J. L. L.	REVISED DESIGN
9	10/1/68	W. J. L. L.	W. J. L. L.	REVISED DESIGN
10	10/1/68	W. J. L. L.	W. J. L. L.	REVISED DESIGN

CONCEPTUAL

U. S. ATOMIC ENERGY COMMISSION
PACIFIC NORTHWEST LABORATORY
OPERATED BY AVIATION TECHNOLOGICAL INSTITUTE

REACTOR ARRANGEMENT
FOR CONCEPT I

FAST FLUX TEST FACILITY

SK-3-14540

0 1 2 3 4 5 FT

CONCEPTUAL 13 ISSUE 12-14-68

U S ATOMIC ENERGY COMMISSION
RICHLAND OPERATIONS OFFICE
PACIFIC NORTHWEST LABORATORY
OPERATED BY BATTTELLE MEMORIAL INSTITUTE

REACTOR HEAD
PLAN & SECTION A-A
FOR CONCEPT 7a
FAST FLUX TEST FACILITY

SCALE 1/4" = 1'-0"

DATE 12/14/68

SK-3-14541

REV	DATE	DESCRIPTION
1	12/14/68	ISSUE 1

REV	DATE	DESCRIPTION
1	12/14/68	ISSUE 1

REFERENCE DRAWINGS

COMMENT PRT ISSUE NO

CHECK PRT ISSUE NO

DRAWING STATUS

SECTION A-A

INST & VHM PLUG IN
REACTOR HEAD
REF A I DWG N68181249
TYP FOR 3

PLAN

SECTION B-B

REACTOR MACHINERY DOME
REF SK 3 14495

VALVE MAINTENANCE & INSPECTION PORT

12 REACTOR INLET LINE
TYP FOR 6

CONTROL ROD DRIVE
REF SK 3 4531

REACTOR HEAD REF DWG SK 3 14551

OPERATING FLOOR EL 0 0 REF DATUM

RING GIRDER

18 0 OD REACTOR VESSEL

Na LEVEL EL (-) 11 6

26 0 D REACTOR SODIUM
OUTLET LINE TYP FOR 3

EL (-) 18 0

SAFETY STANDPIPE

SST INSULATING LINER

REACTOR VESSEL

GUARD VESSEL

VESSEL WALL COOLING-
ANNULUS

ROD DRIVE MECHANISM
SEE ENLARGED VIEWS
ABOVE

FUEL ZONE
EL (-) 38 7

32 0 DIA
REACTOR CAVITY

20 0

14 0

EL (-) 57 6

REACTOR ARRGT - ELEVATION-CONCEPT V-A
OPTION

REV	NO	DATE	DESCRIPTION
1	0	12/14/60	CONCEPT V-A

CONCEPTUAL

U S ATOMIC ENERGY COMMISSION
PACIFIC NORTHWEST LABORATORY
OPERATED BY BATTELLE MEMORIAL INSTITUTE

REACTOR ARRANGEMENT
W/ BOTTOM ENTRY ROD OPTION
CONCEPT V-A

FAST FLUX TEST FACILITY

SK-3-14542

CONCEPTUAL 1ST ISSUE 12-13-68

APPRO. FOR CONFORMANCE WITH DESIGN CRITERIA		DATE	U. S. ATOMIC ENERGY COMMISSION RICHLAND OPERATIONS OFFICE PACIFIC NORTHWEST LABORATORY OPERATED BY BATTTELLE MEMORIAL INSTITUTE	
BY	FOR		OPEN AND CLOSED LOOP INSTRUMENT CONNECTOR	
APPRO.	APPRO.		CONCEPT-5A	
DESIGNED BY	DATE		FAST FLUX TEST FACILITY	
SCALE	DATE		BLDG. NO. 300 GEN. 2200-01	
CLASSIFIED BY	DATE		SK-3-14546	
CLASSIFIED BY	DATE		1 1	

REV. NO.	REV. DATE	DESCRIPTION	BY	FOR	DATE
1	12-13-68	CONCEPT-5A	J.W. CRIGLER		
REVISIONS					
REFERENCE DRAWINGS					
NEXT USED ON					

CLOSED LOOP INSTRUMENT CONNECTOR

OPEN LOOP INSTRUMENT CONNECTOR

SECTION 'A-A'

SECTION 'B-B'

SECTION 'C-C'
FLUX MONITOR THIMBLE CONNECTION

CLOSED LOOP INSTRUMENT CAPABILITY

PROCESS SURVEILLANCE AND SAFETY		NO OF PINS
3 OUTLET TEMPERATURE T/C		6
3 INLET TEMPERATURE T/C		6
1 DIFFERENTIAL PRESSURE ELECTROMECHANICAL TRANSDUCER		5
SPARES		1
EXPERIMENTERS		18 TOTAL
FUEL TEST		
6 FUEL PIN CLAD OR FUEL TEMPERATURE T/C		12
1 FUEL PIN PRESSURE ELECTROMECHANICAL TRANSDUCER		5
SPARE		1
MATERIAL TEST		18 TOTAL
3 TEST SPECIMEN TEMPERATURE T/C		6
2 STRAIN GAUGES (When Developed)		6
SPARES		6
TUBING		18 TOTAL
1 FLUX MONITOR THIMBLE 1/2" O D		
1 FUEL PIN PRESSURE MEASUREMENT TUBE 1/8" O D		
2 TRANSDUCER COOLANT TUBES		

OPEN LOOP INSTRUMENT CAPABILITY

PROCESS SURVEILLANCE AND SAFETY		NO OF PINS
5 OUTLET TEMPERATURE T/C		10
1 EDDY CURRENT FLOWMETER		6
SPARES		10
		26 TOTAL
EXPERIMENTERS		
FUEL TEST		
6 FUEL PIN CLAD OR FUEL TEMPERATURE T/C		12
1 FUEL PIN PRESSURE ELECTROMECHANICAL TRANSDUCER		5
SPARE		1
MATERIAL TEST		18 TOTAL
3 TEST SPECIMEN TEMPERATURE T/C		6
2 STRAIN GAUGES (When Developed)		6
SPARES		6
TUBING		18 TOTAL
1 FLUX MONITOR THIMBLE 1/2" O D		
1 FUEL PIN PRESSURE MEASUREMENT TUBE 1/8" O D		

CONCEPTUAL

1ST ISSUE 12/19/68

APPROVED FOR CONFORMANCE WITH DESIGN CRITERIA		DATE		U. S. ATOMIC ENERGY COMMISSION	
BY				RICHLAND OPERATIONS OFFICE	
FOR				PACIFIC NORTHWEST LABORATORY	
APPROVED				OPERATED BY BATTTELLE MEMORIAL INSTITUTE	
DRAWN		DATE		OPEN & CLOSED LOOP INSTRUMENT CONNECTOR CONCEPT 5A	
CHECKED				FFTF	
SCALE				300 GEN	
CLASSIFIED BY		DATE		INDEX NO 7200 01	
REVISIONS				SK-3-14547	
COMMENT PRT ISSUE NO		DATE		SHEET NO 1	
CHECK PRT ISSUE NO		DATE			
DRAWING STATUS

IDENTIFICATION

- 1 INSTRUMENT PACKAGE WITH FISSION GAS CONCENTRATOR.
- 2 LOWER PLUG-IN GRID STRUCTURE, SEMI-PERMANENT.
- 3 VERTICAL GROUP OF 26-3/4" SAMPLE LINES - TYPICAL OF 3.
- 4 SHIELDING ACCESS PLUG FOR LINE CLEANOUT AND REPLACEMENT
- 5 REMOVABLE SHIELDING FOR REPLACEMENT OF HORIZONTAL SAMPLE LINES IN PIPING TRENCH.
- 6 SODIUM PUMP, MAGNETIC.
- 7 FLOWMETER FOR SAMPLE FLOW MONITORING.
- 8 GAS SEPARATION TANK.
- 9 REFLUXING SODIUM TRAP.
- 10 GAS COIL OR TANK AND GM DETECTOR WITH INDIVIDUAL AND REMOVABLE SHIELDING.
- 11 VAPOR PUMP.
- 12 ARGON PRESSURE AND VACUUM LINE.
- 13 PRIMARY SHIELDED CELL FOR Na.
- 14 SECONDARY SHIELDED CELL FOR FISSION GAS DETECTION.

CONCEPTUAL 1ST ISSUE
12/12/68

U. S. ATOMIC ENERGY COMMISSION RICHLAND OPERATION OFFICE			
SCALE NONE		BATTELLE-NORTHWEST	
DRAWN BY R.C. ROBERTS	DATE 12-9-68	FEDAL LOCATION SYSTEM ARRANGEMENT FOR DRIVERS CONCEPT V-A	
CHECKED BY	DATE		
RESPONSIBLE ENGR. W. DAL OS	DATE 12/9/68		
APPROVED	DATE		
APPROVED	DATE	PROJ. NO. FFTF	
CLASSIFIED BY R. J. M... DATE 12-11-68	DATE	BLDG. NO. 300 GEN	INDEX NO. 5800
CLASSIFICATION NONE		DWS. NO. SK-3-14548	SHEET NO. 1
			SHEETS 1

SCALE: 1" = 6 IN

SECTION A-A

SECTION B-B

SECTION C-C

SECTION D-D

SCALE: FULL-SIZE

CONCEPTUAL		1 ST ISSUE		12-14-68	
U. S. ATOMIC ENERGY COMMISSION					
RICHLAND OPERATIONS OFFICE					
PACIFIC NORTHWEST LABORATORY					
OPERATED BY BATTELLE MEMORIAL INSTITUTE					
PRIMARY LOADING BEAM					
CONCEPT V _a					
RADIAL RESTRAINT					
FAST FLUX TEST FACILITY					
300 GEN					
1900					
SK-3-14568					

REV NO	DATE	DESCRIPTION	BY	CHKD	APPD
1	12/14/68	CONCEPT V _a	J. JACKSON		
2	12/14/68	CONCEPT V _a	G. R. WAYMIRE		
3	12/14/68	CONCEPT V _a			
4	12/14/68	CONCEPT V _a			
5	12/14/68	CONCEPT V _a			
6	12/14/68	CONCEPT V _a			
7	12/14/68	CONCEPT V _a			
8	12/14/68	CONCEPT V _a			
9	12/14/68	CONCEPT V _a			
10	12/14/68	CONCEPT V _a			
11	12/14/68	CONCEPT V _a			
12	12/14/68	CONCEPT V _a			
13	12/14/68	CONCEPT V _a

LOOP PREPARATION CASK

SHOWN HANDLING LOOP PLUG
WGT ~ 25,000#

ATOMIC INTERNATIONAL	DESIGNED BY	DATE
A DIVISION OF ROBERT ANDERSON, INC.	BY	10/1/85
10000 1ST AVE. S.W.	CHKD BY	10/1/85
SEATTLE, WA 98148	DATE	10/1/85
	BY	10/1/85
	DATE	10/1/85
	BY	10/1/85
	DATE	10/1/85

REVISIONS	DATE	APPROVED
1	10/1/85	
2	10/1/85	
3	10/1/85	
4	10/1/85	
5	10/1/85	
6	10/1/85	
7	10/1/85	
8	10/1/85	
9	10/1/85	
10	10/1/85

ATOMIC INTERNATIONAL
SODIUM COOLED CLOSED LOOP
HANDLING MACHINE
DESIGN NO. 01571
REV. 10/61

ATOMIC INTERNATIONAL
SODIUM COOLED CLOSED LOOP
HANDLING MACHINE
DESIGN NO. 01571
REV. 10/61

NOTICE
ATOMIC INTERNATIONAL
SODIUM COOLED CLOSED LOOP
HANDLING MACHINE
DESIGN NO. 01571
REV. 10/61

REVISIONS			
SYN	ZONE	DESCRIPTION	DATE

ATOMICS INTERNATIONAL A DIVISION OF NORTH AMERICAN ATOMICS INC 2200 BENTLEY AVENUE, CHANOGA PARK, CALIF.			
INSTRUMENTATION SHIELD PLUG			
REGION - FFE 3 PLUG CONDUIT			
DRAWN CHECKED ENGINEER APPROVED	DATE SIZE SCALE SHEET NO. OF	DWG NO E NG631250	DATE APPROVED

REVISIONS			
SYN	ZONE	DESCRIPTION	DATE APPROVE

SECTION A-A
ROTATED 90° C/CLOCKWISE

SECTION 13 13
FULL SIZE

ATOMICS INTERNATIONAL
A DIVISION OF NORTH AMERICAN AVIATION, INC.
4900 DEWITT AVENUE, CAROLAN PARK, CALIF.

**FEDAL TUBE CONFIGURATION
AT STATIONARY TOP SHIELD**

DESIGN	DATE	BY	NO.
CHECKED			
ENGINEER			
APPROVED			

SCALE: FULL SIZE
SHEET NO. 1 OF 1

ATOMIC INTERNATIONAL
A Division of Atomic Energy Research Corporation
P.O. Box 1000, Santa Barbara, California 93103
TELEPHONE (805) 964-1000

DESIGNED BY	REVIEWED BY	DATE
1	2	3
INSTRUMENT SUPPORT ASSY.		
J 09974 163181255		
SHEET 1 OF 1		

REVISIONS			
SYN	ZONE	DESCRIPTION	DATE

ATOMICS INTERNATIONAL <small>A DIVISION OF NORTH AMERICAN AVIATION INC 1900 DELICATO AVENUE CANOGA PARK, CALIF.</small>	
INSTRUMENTATION ASSEMBLY DRIVER FUEL ALIGNMENT	
DRAWN CHECKED ENG'G LEP APPROVED	FILE SIZE E SCALE SHEET NO. 01 <small>PRINTED IN U.S.A.</small>

REVISIONS			
SYM	ZONE	DESCRIPTION	DATE

ATOMICS INTERNATIONAL <small>A DIVISION OF NORTH AMERICAN AVIATION INC</small> <small>8900 DESOTO AVENUE CANOGA PARK, CALIF.</small>			
FFTF INS & IVHM PLUG, PLAN VIEW, REF CONCEPT			
DRAWN CHECKED ENGINEER APPROVED	C. R. HARRIS J. R. HARRIS J. R. HARRIS J. R. HARRIS	SEE E SOLE J. R. HARRIS	N68181267 SHEET NO. 1 OF 1

REVISIONS			
SYM	ZONE	DESCRIPTION	DATE APPROVED

ATOMICS INTERNATIONAL	
A DIVISION OF NORTH AMERICAN AVIATION, INC.	
8750 REDWOOD AVENUE, CANOGA PARK, CALIF.	
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R.J. Hennig	J.E. Mills	G.H. Strong
P.L. Hofmann	W.A. Muller	C.D. Swanson
B.M. Johnson	C.A. Munro	J.C. Tobin
H.G. Johnson	C.R. Nash	K.G. Toyoda
E.M. Johnston	C.L. Peckinpaugh	R.C. Walker
F.J. Kempf	R.E. Peterson	G.R. Waymire
J.W. Kolb	O.W. Priebe	J.H. Westsik
G.A. Last	W.D. Richmond	J.F. Wett
H.D. Lenkersdorfer	W.E. Roake	N.G. Wittenbrock
T. Lillie	R.L. Roske	T.W. Withers
W.W. Little, Jr.	E. Ruane	M.R. Wood
C.E. Love	D.P. Schively	F.W. Woodfield
M.K. Mahaffey	F.H. Shadel	FFTF File (10)
D.E. Mahagin	D.W. Shannon	Tech. Info Files (5)
B. Mann	D.E. Simpson	Tech. Pubs (3)
C.W. Higby	D.J. Oakley	D.R. de Halas
	L.J. Defferding	Legal (2)