ALTERNATE REACTOR CONCEPT DESIGN STUDY

FOR THE

FAST FLUX TEST FACILITY

FIXED SHIELD PLUG, VERTICAL CORE REACTOR CONCEPT

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J. G. Duffy, Project Engineer
J. W. Hess
J. Matte
R. E. Mueller
J. R. White

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ATOMIC POWER DEVELOPMENT ASSOCIATES
1911 First Street
Detroit, Michigan

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PREFACE

The alternate reactor concept design study reported here has been performed by APDA to provide assistance to Pacific Northwest Laboratories in connection with the design of the Fast Flux Test Facility (FFTF).

APDA performed this study under Project Agreement 16 of AEC Contract No. AT(11-1)-865 in direct support of the AEC program to develop a facility for the irradiation testing of fast reactor fuels and materials.

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P. V. Bakre
R. A. MacLeod
K. W. Schang

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I. INTRODUCTION

This is a report on an APDA design study of a fixed shield plug, vertical core reactor concept for the Fast Flux Test Facility (FFTF). APDA performed this study under AEC contract during the period February through April, 1966 to provide Pacific Northwest Laboratories (PNL) with one of a number of alternate test reactor concepts which are to be considered in the development of the preliminary design of FFTF. A parallel APDA design study of an alternate skewed core reactor concept was also performed during this period of time and is separately reported in report No. APDA-186. These studies are an extension of preliminary work on alternate test reactor concepts which was completed by APDA on November 30, 1965 and described in interim technical report material which was issued to PNL in an APDA document numbered MD-65-625.

Major features of the concept reported here include a sodium-cooled vertical core of free thermal bowing subassemblies, and in-reactor fuel handling by three pantograph-type offset handling mechanisms which penetrate the fixed shield plug of the reactor. The major objectives of this report are to identify the significant features and characteristics of this concept and to compare them to those of the FFTF interim reference reactor design concept as defined in report number BNWL-CC-400.

The cost estimate for the interim reference concept has not been made available for this study and, therefore, those features and conditions which represent cost factors are identified and compared on a qualitative judgment basis. The cost factors which are considered include reactor power level, equipment size and relative complexity, relative fabricability of equipment components, degree of precedent or innovation in equipment design,

and the consequent relative magnitude of development effort that is judged to be required. The comparison also includes consideration of those factors which are the measure of the desirability of each concept from a functional performance standpoint. These functional performance factors include the number and size of test loops that can be accommodated, test hole flux levels, flexibility and accessibility considerations, and reliability and load factor considerations. Each concept can be characterized by relative comparison of cost factors, functional performance factors, and confidence considerations. In comparison to the interim reference concept, the alternate concept reported here was developed with the broad objectives of achieving lower cost and a higher level of confidence in the reactor design, and with acceptance of more limited accessibility and greater restriction on the arrangement and number of test loops that can be accommodated in the reactor.

The layout design work on this concept has been focussed on those features which are pertinent to the comparative evaluation. The layout studies therefore cover the arrangement of the reactor internals and overhead shielding, the control and safety rod drives, the in-reactor components of closed test loops and their coolant line connections above the reactor, and the equipment for handling driver fuel, test fuel specimens, shield plugs, loop tubes, and rod drives within the reactor, and above-floor within the reactor building. It is expected that the remainder of the reactor plant, including the external test loop facilities, will be smaller in overall size and cost as a result of the smaller size and power level of this alternate concept reactor and its greater restriction on the number of test loops that can be accommodated.

The reference cermet driver fuel material, fuel pin diameter, and pin spacing of the interim reference concept have been retained for this

alternate concept study. The published design data on the hydraulic, thermal, and nuclear characteristics of this core composition have therefore been generally applicable to the alternate concept, with the result that both concepts are on a closely comparable basis in these areas. It is expected that changes in the fuel material or dimensional arrangement, if applied to both concepts, will not significantly affect the comparison.

The organization of this report generally parallels the pertinent portions of the interim reference concept report, BNWL-CC-400. The features and characteristics of the alternate concept are compared to those of the interim reference concept throughout this report as they are described and discussed. A summary comparison follows in a separate section near the end of the report.

II. DESIGN CRITERIA

The alternate concept has been designed to meet the objectives and requirements of the design criteria specified by PNL for this study. These criteria are identified as Attachment 2 of Volume 1 of report No. BNWL-CC-175. The following list covers those criteria items which had significant influence on the development of the alternate concept:

On-line availability - the design value remains to be determined by cost vs. reliability trade-offs, early target is 75%; value is proportional to availability.

Proven components and concepts - to be used where feasible.

Closed loops - 4-8 instrumented, capability for 6-inch loops.

Open loops - 6-8 instrumented.

Test holes and facilities - must include a rabbit facility.

Driver fuel - cermet as reference driver fuel, mixed oxide fuel primary backup. Secondary backup to be recommended.

Total peak flux - $\geq 10^{16}$ n/cm²-sec minimum with reference cermet fuel.

Coolant - sodium coolant in the primary and closed loop systems with capability for using other coolants in the closed loops to meet potential needs.

Test fuel pins -

fuel length - 1.5 to 4 ft.

diameter - 0.2 to 0.4 inch; not limited.

overall length - ~8 ft.

In-reactor - design shall not preclude use of any in-pile instrument in open or closed loops; direct-instrument

connection between experiments and the reactor exterior shall be assumed. As a minimum, closed loops will be provided with coolant temperature and fission gas instrumentation and the design should readily permit additional instrumentation which is successfully developed to meet experimental needs. Potential measurements include internal temperatures and pressures of experimental objects.

Additional criteria for the APDA concept studies, covering fuel specimen removal times and decay heat rates, were specified by a letter from L. M. Finch to A. Amorosi, dated February 4, 1966. The substance of the requirements given is that it must be possible to remove any fuel specimen from the reactor core or from any loop within 8 hours after shut-down, and that the removal equipment must be able to handle decay heat levels as high as 100 Kw and control specimen temperature within a 50° F range for all decay heat loads down to 0 Kw. These requirements can be accommodated by the alternate concept described in this report, although the need for early removal of large, high burn-up subassemblies which are at a high decay heat level is debatable. If early removal of a high burn-up test specimen is deliberately planned for experimental purposes it is reasonable to expect that only a small diameter loop--i.e., a small sample--will be used for such purpose. If a large diameter loop is to be used to carry a large test pin bundle to a burn-up level which results in a high initial decay heat level, the loop can be designed with provision for decay storage of the fuel in the above-core section of the loop, as is proposed in this alternate concept. The alternate concept also provides for decay storage of the core subassemblies in storage positions in the reactor near the vessel wall.

III. IRRADIATION TESTING CAPABILITY

The testing capability of the vertical core concept is indicated by the core midplane section drawing, Figure III-A. The core lattice, as shown, consists of lattice hexagons which are 3-1/2 inches across flats. This hexagon dimension may be subject to small variations (approximately $\pm 1/4"$) in final design as a function of final selection of driver fuel pin diameter and pin spacing, and this may, in turn, have a small effect on the diameters of loops that can be accommodated.

The principal test facilities in this concept consist of the large center test hole, which occupies the space of 7 lattice positions, a pair of 6-inch diameter and two 8-inch diameter peripheral closed loops, and a rabbit tube facility which is also located at the core edge. The center test hole is shown occupied by a cluster of three vertical 3-1/2 inch diameter loops. These small loops could be replaced by a single vertical 8-inch diameter loop. The peripheral loops and rabbit tube are tilted outward at a 3-1/2° angle from vertical to provide space for access to them at operating floor level.

Test subassemblies can also be located in driver core lattice positions for "open loop" testing in the primary coolant sodium. During reactor operation, each test subassembly in a core lattice position will have an instrument finger located immediately above it. Each finger will contain thermocouples to monitor the sodium coolant temperature as it flows out of the subassembly. The fingers could also contain one or more small diameter tubes which could be used for purposes such as taking samples of this sodium. The arrangement and use of these instrument fingers is described in greater detail in Section VI of this report.

Table III-(1) lists all of the experimental facilities, including the possibility of using the instrument tube position and the oscillator rod position for test purposes. Total fluxes are estimated to range from $1.29 \times 10^{16} \text{ n/cm}^2\text{-sec.}$ at the reactor center to $0.70 \times 10^{16} \text{ n/cm}^2\text{-sec.}$ at the core edge when the reactor is operated at 360 Mw. The use of safety rod nose piece positions for irradiation of structural materials samples, as proposed in BNWL-CC-400, page 2.0-2, has not been specifically investigated in this study, but it appears to be feasible in this alternate concept as it is in the interim reference concept. Chemical, thermal, and hydraulic environment considerations and test fuel composition and location considerations are essentially the same for the alternate concept as those stated for the interim reference concept.

The alternate concept closed loop arrangement can be compared favorably with the initial loading case of the reference concept. However, the alternate concept design was developed with acceptance of limitations on the number and flexibility of its test facilities in favor of achieving low capital cost and high operational reliability objectives. It therefore does not have the flexibility to accommodate a larger number of closed loops, and a packaged loop can be located only in a closed loop position. Closed loop instrumentation is the same for both the alternate and reference concepts, but subassemblies located in driver core positions for "open loop" testing in the alternate concept can not be as completely instrumented as "open loop" specimens in the reference design. The alternate vertical core concept does have the advantage of being able to accommodate an 8-inch diameter vertical loop in the center of the core. In addition, the question of gravity effects on test specimens in a tilted condition is of less concern in this alternate concept, since the peripheral loop tilt angle is only $3\text{-}1/2^\circ$ in comparison to tilt angles which are in the range between 15° and 20° in the reference concept.

TABLE NO. III-(1)Experimental Facilities - Vertical Core Concept

<u>Facilities</u>	<u>Number</u>
Closed Loops for Fuel Tests, 8 in. dia.	2 (Note A)
Closed Loops for Fuel Tests, 6 in. dia.	2
Closed Loops for Fuel Tests, 3-1/2 in. dia.	3 (Note A)
"Open Loop" Fuel Test Positions	Note B
Packaged Loop Material Tests	Note C
"Open Loop" Materials Test Positions	Unlimited (3 shown)
Rapid Transit (Rabbit) Tube	1
Instrument Tube into Core Region	1
Oscillator Rod Position	1

Note A - One 8" closed loop can replace the three 3-1/2" loops.

Note B - Any of the driver fuel positions, with number and location dependent on neutronics of test specimen.

Note C - Only fuel test closed loops are available, to the extent not needed for fuels testing.

IV. SUMMARY DESCRIPTION OF DESIGN

The fuels testing reactor concept described in this report is an alternate to the reference concept for the FFTF, as described in BNWL-CC-400. The scope of this concept study has been limited to those portions of the alternate concept which represent major departures from the reference concept. The principal difference between the two concepts is the use in this alternate concept of a vertical core reactor of free thermal bowing subassemblies, with in-reactor fuel handling mechanisms operated through a fixed shield plug, in contrast to the lattice of skewed tubes employed in the reference reactor. This major difference requires or permits significantly different concepts for the reactor internals and overhead shielding, the control rod drives, the handling methods and equipment, the containment vessel size and arrangement, the design of fuel, test loops, and numerous other reactor internal details. These concept differences are described and discussed in this report, but little or no attention is devoted to peripheral or ancillary equipment and facilities common to the two concepts, even where it is evident that these facilities would be smaller and less costly for this alternate concept.

The reactor consists of a free standing vertical core of free thermal bowing subassemblies supported and contained within a cylindrical reactor vessel, as shown in Figure IV-A. The successive layers of containment and shielding surrounding the reactor vessel were not studied or shown, since they would carry out the design functions to the same degree as in the reference concept. The upper reactor containment vessel is a 75-foot diameter cylindrical and hemispherical steel pressure vessel joined and anchored to the concrete lower reactor containment vessel. This dome carries a circular girder on which is installed a polar bridge crane which carries a fuel handling cask, a plug

handling cask, and an equipment hoist for erection and maintenance. The reactor is closed at the top by a fixed shield plug.

Closed loops extend down through the plug and the core into the support plates, as shown in Figure IV-A. These consist of two 8-inch and two 6-inch loops located on the edge of the core, and three 3-1/2 inch loops located in the center of the core. Other equipment located within the reactor vessel include three fuel handling mechanisms, an instrument plate which positions instrument fingers in close proximity to the top of all driver subassembly positions, and the safety and control rod extensions.

The reactor primary coolant system is assumed to be basically identical to that of the reference concept except that the lower power level will permit smaller flow rates and correspondingly smaller equipment. One minor difference is that the operating level of the sodium surface in the vertical core reactor suggests that long-shaft sodium pumps be used, similar to the successful units at EFAPP. Closed loop coolant system arrangements are similar to those of the reference concept. Sufficient shielding is provided around the loop coolant piping to allow loop removal operations on one loop while adjacent loops (and the reactor) remain in operation.

Handling of driver fuel and "open-loop" test specimen subassemblies is done by one of the fuel handling mechanisms operating within the reactor, shown in Figure IV-A. The in-reactor fuel handling mechanism transfers fuel from the core to fuel storage locations or to the transfer pot of the above-reactor cask when it is lowered through an exit port in the fixed shield plug. The above-reactor fuel handling cask contains the NaK-cooled transfer pot and also a gripper. Both gripper and transfer pot are used for removing test subassemblies from the outer loops. The gripper is also used for removing loops and safety or control rod extensions from the reactor. Removal of test

specimens from a center loop is done by removal of the loop from the reactor to a maintenance pit where suitable equipment for the balance of this operation is provided.

A significant feature of this concept is the ability to place any desired test or driver subassembly in a decay storage area or position so that it can be held in the reactor until it has reached a conveniently low level of activity for removal from the reactor. This will simplify the majority of handling operations and thus improve the reliability of the facility, since only occasionally will a test program require removal of fuel or specimens promptly after shut-down. The fuel handling mechanisms, shielding, and heat rejection equipment are sized to handle prompt removals, however.

In addition to provision for the closed loop and open position test arrangements required by the criteria, this concept incorporates a number of maintenance penetrations through the fixed shield plug. These offer not only accessibility for maintenance and repair, but also opportunities to experimentalists for future innovations in test apparatus and instrumentation.

V. PHYSICS DESIGNSummary

The conceptual nuclear design of the fixed shield plug, vertical core reactor concept has been formulated to meet the three major nuclear design objectives of the Fast Flux Test Facility (FFTF).

- (1) The reactor should have the ability of sustaining very high fast neutron flux levels of the order of 10^{16} n/cm²/sec above 0.1 Mev.
- (2) Ample control margins must be available to accommodate the presence of low reactivity test assemblies and to allow for substantial driver fuel burnup per operating cycle.
- (3) In-core loop design must be such that failure of test assemblies can be tolerated without causing damage to the reactor.

The 480 liter alternate vertical core, fueled with PuO₂-SS cermet subassemblies and controlled by six enriched B₄C rods, essentially meets these objectives.

The physics design of the 480 liter alternate vertical core reactor concept is not significantly different than that of the 600 liter FFTF interim reference reactor design described in Report No. BNWL-CC-400. Although the vertical core's fissile plutonium concentration is higher than that of the interim reference core, it is still relatively low and results in a high flux to power ratio. The vertical core's flux to power ratio is only about 10% less than that of the interim reference core. Consequently, the vertical core requires a specific power level only about 10% larger than that of the interim reference reactor in order to achieve essentially the same flux levels

as in the interim design. However, since the higher fuel concentration in the vertical core results in a proportionately larger decrease in critical core volume than increase in specific power, the total reactor power required for the same flux levels is reduced to 360 Mw (ultimate design) compared to the 400 Mw ultimate design power for the interim reference core.

The interim reference reactor's cermet fuel is also being used in the alternate vertical core concept design. Therefore, the same fast acting negative fuel expansion coefficient also exists in this design. Furthermore, due to the vertical core's smaller size, the overall sodium void coefficient is probably more negative than that of the interim reference design. Thus, the same important safety features for the reactor due to negative coefficients for sodium and fuel also exist in the vertical core design.

The control requirements to account for burnup, temperature defect, and shutdown margin in the interim reference FTR design are large (10-12% Δk). For the purpose of the alternate vertical core concept studies, it was assumed that the control requirements would not be significantly different. The temperature defect may be somewhat larger due to the smaller core size whereas the requirements for burnup will probably be a bit smaller because of a need for more frequent refueling. Six enriched $B_{10}C$ rods, three shim rods and three safety rods, are used to meet the control requirements in the alternate vertical core concept design.

No calculations have been made to examine the consequence of test assembly failures on the overall integrity of the vertical core reactor. From cursory examination, it is not expected that the failure of test assemblies in the closed test loops would damage the reactor any more severely than in the interim reference design. However, the three closed loops in the vertical

core design which are located together at the core center should be looked at in more detail in future studies.

In the following sections, the basic nuclear design aspects of the 480 liter PuO_2 -SS cermet fueled vertical core are described and discussed in more detail.

Reactor Statics

The 480 liter vertical core is shown in Figure III-A. Although the core is not at present zoned to obtain spatial power flattening, this could easily be done if it proved necessary. In any event, fuel zoning would have relatively little effect on the total fuel inventory and the spatial neutron flux distribution. For the purpose of making the nuclear design calculations for the alternate vertical core concept, the same ground rules were used as those in the interim reference design physics studies (BNWL-CC-400). These represent idealized physics calculations.

For the purpose of making the nuclear design calculations, the irregular core shape has been ignored. The equivalent homogenized cylindrical core model was used. The cylinder had an L/D ratio of unity. It contained approximately 34 v/o sodium, 59.74 v/o stainless steel, and 6.26 v/o PuO_2 . This composition corresponds to the vertical core fuel pin density shown in Figure V-A. The PuO_2 density in these idealized calculations was 100% of theoretical density (11.46 gms/cc). The plutonium composition was taken to be 90 a/o Pu-239 and 10 a/o Pu-240. The radial reflector was 30 cm thick and consisted of 80 v/o Inconel and 20 v/o sodium. End reflectors were represented by 11 cm reflector savings at each end. The calculations represent clean core calculations and they were done for a value of $k_{\text{eff}} = 1.05$. The 5% excess reactivity value used is the same as used in the interim reference reactor design calculations. It was assumed that this is sufficient to account for

the control requirements for burnup and temperature defect, the heterogeneity effects of the control channels, loops, etc. and other calculational uncertainties.

The reactor statics calculations for the vertical core were made using a modified version of the 24-group British Cross-Section Set and the one-dimensional version of the UKAEA's Cram diffusion theory code. The use of this code and cross-section set was first checked by calculating the FFTF interim reference reactor design. The results of these calculations compared very closely with the physics values reported in BNWL-CC-400. The corresponding calculations for the alternate vertical core reactor concept, on the basis of the model described above, gave a critical core height and diameter of 84.5 cm.

The other nuclear characteristics of the PuO_2 -SS vertical core are summarized and compared with the interim reference design in Table V-(1). The reactor power for the vertical core design was chosen so as to give the same average core flux as in the interim reference reactor design. The slightly smaller flux at the core center in the case of the vertical core design results from its slightly smaller maximum-to-average power ratio.

Vertical core calculations of β_{eff} , sodium worth, fuel and sodium expansion coefficients, Doppler coefficient, and reactivity lifetime were not made. However, it is not expected that any significant differences in these parameters, affecting feasibility of the vertical core concept, will exist compared to the interim reference design values.

It should be pointed out again, that the above calculations pertain only to the homogenized clean core design. However, the control margin allotted for burnup ($\sim 4\% \Delta k$) will only accommodate about 5.5% average burnup of the fuel per operating cycle, whereas the average removal burnup of the fuel may be as high as 30%. Therefore, after long term operation the core will eventually attain an equilibrium condition in which the average burnup

will be approximately 15%. To accommodate the reactivity loss resulting from 15% average burnup throughout the core, some increase in core size can be anticipated if driver fuel is added at the core edge.

Control statics

It was assumed that the static control requirements for the alternate vertical core design were not significantly different than those of the interim reference design. However, these requirements are large and amount to approximately 10-12% Δk . In the vertical core design three enriched $B_{14}C$ shim control rods are provided to account for fuel burnup effects and temperature defect ($\sim 6\% \Delta k$). The rods are located near the edge of the core (See Figure III-A) rather than near the core center in order to avoid creating flux depressions in the high flux central loops. Each of the three shim control rods has a reactivity worth of about 2% Δk per rod. Conservative estimates indicate that in order to achieve this worth for rods in this location, the boron enrichment of the $B_{14}C$ must be of the order of 55w/o B-10. This value is based upon a rod composition of 20.5 v/o $B_{14}C$, 64.5 v/o sodium, and 15 v/o stainless steel in the vertical core's hexagonal rod lattice positions. Cooling of the shim rods with this design should not be a difficult problem.

The amount of control margin required for shutdown is largely a safeguards question. In the vertical core design, three enriched $B_{14}C$ safety rods are used to attain the same safety control margin used in the FFTF interim reference reactor design (6% Δk). The three safety rods are located near the core center (Figure III-A); however, since they will be withdrawn during operation they will have no appreciable effect on the flux in the centrally located loops. The worth of each of the three safety rods is about 2% Δk per rod. In order to achieve this worth with the rods located

as shown, the $B_{10}C$ enrichment must be 21 w/o B-10 in boron, assuming the same safety rod composition as used for the shim rods above.

Discussion

The physics design of the alternate vertical core reactor concept does not vary greatly from that of the interim reference reactor design. This is mainly because both reactors use the same fuel. However, because of the higher fuel density achievable in the vertical core, a reduction in core volume, critical mass, and reactor power results. The other physics parameters are essentially the same.

It should be pointed out that in order to achieve the required clean critical loading of 270 kgs of Pu-239 in the vertical core design shown in Figure III-A, it must be assumed that each of the closed fuel loops contains half of the fuel loading of a normal driver subassembly, or its equivalence, and that each of the open loops contains the equivalence of the fuel loading of a full driver subassembly.

One can think of instances, however, where the loops might not be so highly loaded. This would require additional driver subassemblies to be loaded at the core edge and result in a corresponding increase in core size. Furthermore, there are indications that calculations tend to overestimate by quite a bit the reactivity worth of nickel reflected fast reactor cores. It is therefore conceivable that the effects of partial loop filling and nickel reflector calculational uncertainties, combined with the effects of heterogeneity reactivity losses, fuel zoning if it is used, and other calculational uncertainties, could result in a considerably larger excess reactivity requirement than allowed for in the idealized calculations. It is estimated that as much as 7% more excess reactivity than the 5% value used might be required in the idealized homogenized calculations, to adequately

describe the actual reactor. For the case of edge loaded fuel, this further reactivity requirement would result in an increase in core size which would be in addition to any core size increase required to account for equilibrium conditions. The possible implications of this on the design and operation of this alternate concept reactor would have to be investigated in more detail in future studies.

TABLE V-(1)

COMPARISON OF NUCLEAR CHARACTERISTICS OF THE
VERTICAL CORE FTR VS. THE INTERIM REFERENCE FTR*

<u>Item</u>	<u>Interim Core Value</u>	<u>Vertical Core Value</u>
K_{eff}	1.05	1.05
Burnup	Clean	Clean
Zoned	No	No
L/D	1	1
D(L), cm	91.5	84.5
Volume, liters	600	480
v/o PuO ₂ (100% dense, 90/10)	5.50	6.26
Mass of Pu-239, kg	300	270
Mass of Pu-240, kg	33.5	30
Power, Mw	400	360
(P_{max}/P_{avg}) axial	1.33	1.33
(P_{max}/P_{avg}) radial	1.27	1.25
P_{max}/P_{avg}	1.69	1.66
$\bar{\phi}$ (average), n/cm ² /sec	$.78 \times 10^{16}$	$.78 \times 10^{16}$
ϕ_0 (center), n/cm ² /sec	1.32×10^{16}	1.29×10^{16}
ϕ (> 0.1 Mev)/ ϕ total	0.71	0.71
Number of Shim Rods	3	3
Number of Safety Rods	3	3
Rod Worth, Δk per rod	0.02	0.02

* Results of idealized physics calculations. The calculations do not consider reactor design parameters such as zoning, heterogeneity effects, etc.

NOTE: Pin density does not include
3 pin positions used for corner
posts in each subassembly.

FUEL PIN DENSITY vs DISTANCE FROM CORE MIDPLANE
FOR
FFTF REACTOR CONCEPTS

VI. REACTOR DESIGN

This section describes the reactor of the vertical core alternate concept, with particular emphasis on the reactor vessel and core, since the other components of the reactor complex are treated extensively in following sections of the report. As shown in Figure VI-A, the reactor is a 13'-6" I.D. cylindrical vessel with an elliptical bottom head and internal structures serving as coolant plenums and fuel subassembly support. Because the coolant pressure drops through the core to approximately atmospheric at the sodium service, the reactor is constructed of the relatively thin wall typical of sodium-cooled reactors.

The lower portion of the reactor vessel contains coolant distribution plenums and thermal shielding. This structure also holds the fuel subassembly support plates. The parallel, vertical core consists of hexagon subassemblies supported by cylindrical lower nozzles inserted through the support plates. The subassemblies are free standing, free thermal bowing, and held in the support plate by hydraulic hold-down, an arrangement which is essentially the same as the EBR-II core design. The closed loop tubes also are aligned by penetrating through the support plates. Surrounding the core subassemblies is a reflector consisting of hexagonal bars installed in the support plate, the large-diameter closed loop tubes, a rabbit tube, an instrument tube and an oscillator rod. Open position materials testing subassemblies can be substituted for reflector bars, as required. The walls and outlet nozzles of the reactor vessel are lined with shielding plates to provide protection to the vessel from neutron damage and from thermal shock. The vessel is suspended on its top flange from the anchor ring in the reactor building floor, as in the reference design.

The upper end of the reactor vessel is closed by a fixed-location removable shield plug which acts as part of the biological shield. The gas-tight barrier includes the shell and uppermost thermal shield plate of the plug, permitting the use of low-cost shielding material within the plug without contamination of the reactor cover gas or sodium. Mechanisms which penetrate the shield plug will incorporate within themselves suitable shielding materials to replace the plug shielding displaced. The space between the top of the fixed shield plug and the operating floor contains the closed loop coolant piping (shielded), various sealing adapters, the lower portion of fuel handling mechanism drives, control and safety rod drives, and the instrument plate drives, shown in Figure VI-B. Also shown in the figure are the exit ports (with plugs removed), and the corresponding fuel transfer pot guides on the support plate. Further discussion of the fuel unloading will be found in Section IX.

The fixed shield plug has space for additional plugged penetrations which can be used for maintenance operations and repairs, or for special experiments or additional instrumentation. Seven such maintenance ports are shown in suggested locations in Figure VI-C.

The general arrangement of the core, the reflector, the safety and control rods, and the closed loop tubes and other experimental facilities is presented in Figure VI-D. The central space can be used either for three 3-1/2 inch diameter closed loops, as shown, or for other loop arrangements, including a single 8 inch diameter closed loop. Since the reflector can be handled like fuel subassemblies, open position materials irradiation subassemblies can be substituted at any reflector position. Similarly, any driver fuel position can be utilized for open position fuel specimen testing, to the extent that perturbations of core neutronics, hydraulics, temperature, etc. caused by the substitution are acceptable.

A general description of driver fuel subassembly dimensions is given in Figure VI-E. The subassembly is a regular hexagon in cross section, 3-1/2 inches across flats and 96 inches long. Because hydraulic hold-down, rather than mechanical, is employed, the nozzle is of one-piece construction without a compression spring.

This reactor configuration, by its compact dimensions and simple configuration, offer a reactor facility which is smaller and less costly than that proposed by the reference concept, with only a modest reduction in testing positions incurred by this reduced reactor size. The more conventional configuration and component size of this alternate concept offer a higher degree of confidence that the design and construction can be completed within a predetermined budget and schedule, and with a smaller developmental effort. The limiting of tilt of the peripheral loops to 3-1/2 degrees minimizes gravity effects on test specimens which might be considered objectionable in some cases.

In general, this concept employs features and components already proven in EBR-II and EFAPP to a much greater extent than the reference concept, leading to expectation of reduced developmental costs and higher on-line availability of the facility. Design, developmental, fabrication and erection costs and times should be substantially smaller for this alternate concept, and problems such as transportation of the reactor vessel to a Hanford site may be considerably reduced.

A smaller degree of flexibility in number and location of loops and a lower level of accessibility for internal maintenance operations is accepted to achieve the above advantages.

FIGURE VII-B

LTR	ZONE	CHARGE	MADE BY	CHK'D	DATE	APP'D	DATE
FFTF REACTOR CONCEPT-VISIT CORE - ELEV. SECTION THROUGH INSTRUMENT PLATE DRIVE & RABBIT PORTS							
DESIGNED BY	BRADLEY	A-8	CA				
DRAWN BY							
CHECKED BY							
APPROVED BY							
DATE	11-0						
ATOMIC POWER DEVELOPMENT ASSOCIATES, INC.							RELATIVE NUMBER
USE DIMENSIONS ONLY							6XN-6273

[illegible]

FIGURE VI-D

DATE	BY	CHANGE	DATE	DATE	DATE
1/1/68	W	1	1/1/68	1/1/68	1/1/68
TITLE: FUEL HANDLING MECHANISM DESIGN AT THE OF FUEL SUB ASS. S. 3/1/68					
ATOMIC POWER DEVELOPMENT ASSOCIATES INC. 1/1/68 6XN 6271					

VII. REACTOR CONTROL MECHANISMS

Control mechanisms include safety, shim and regulating rods occupying core positions as shown in Figure III-A. The safety rod concept, shown in Figure VII-A, consists of a poison rod which is raised or lowered within a guide tube by a drive system attached to the upper end of the tube. A magnetically latched gripper connects the poison rod to the drive. Reactor scramming is accomplished by releasing the gripper and driving the poison rod into the core. A hydraulic damper absorbs the kinetic energy of the rod after its scram insertion. Shim and regulating rods would be similar in design but without the scram feature.

The objective of this concept is to provide a sealed, self-contained unit with high reliability and maintainability. In satisfying these requirements, existing components are used wherever possible. The magnetically latched gripper, hydraulic damper and drive system are patterned directly on the Fermi safety rod design. Also, the drive system is located in an accessible area such that adjustments can be performed without removing the drive system or safety rod from the reactor. Experience at both Fermi and EBR-II has indicated the desirability of this feature. Scram assurance is provided by the fast rod drive motor. This motor will drive the poison rod into the core at a rate of 120 inches/min. and exert a force of 1500 lbs. in the event of a gripper failure or stuck rod during scram. To insure leaktight containment of the reactor, the drive system housing is sealed to the reactor shield plug.

Other design features include the use of NaK coolant within the safety rod containment tube to transfer heat from the poison section to the reactor coolant and provide a more controlled environment for the mechanism

while it is installed or being removed from the reactor. Another feature is the use of pneumatic pressure, continuously applied to the poison rod, as a scram assist force. A pressure of 40-50 psig will provide a 2g assist to the scram acceleration resulting in a total scram time, for a 36-inch stroke, of less than 0.35 seconds. Unassisted scram time, with a rod weight of 125 lbs., is about 0.7 seconds. The maximum rod velocity with a 2g assist is about 25 feet per second.

The entire safety rod and drive system is removed using the transfer cask. It is also possible to manually detach and remove the drive system without removing the safety rod from the reactor. The drive system is a small packaged unit, about 4-1/2 feet high, containing two Slo-Syn stepping motors which drive a common ball screw and nut through appropriate gear trains. One motor will raise or lower the poison rod at a rate of 2.5 in/min while the other drives the rod at 120 in/min. The drive is capable of applying a load of about 3000 lbs. The motors are powered and controlled separately and have electro-magnetic clutches and brakes to insure fail-safe operation and prevent inadvertent fast withdrawal. A rod position indicating ball screw has been provided which permits the limit switches to be located in an accessible area. The poison rod can be raised or lowered in incremental steps of .001 inch with this drive system.

A comparison of control mechanisms for the alternate and "Reference" designs indicates that the degree of mechanism complexity and fabrication cost is approximately equivalent. The reliability of both systems, as a result of using existing components, should be satisfactory. However, additional features provided on the alternate design would increase the reliability of the system and warrant further study.

VIII. LOOPS

Closed loop locations in the vertical core reactor concept are shown in Figure III-A. This arrangement provides two 6-inch and two 8-inch diameter loops located along the edge of the core region and three 1-1/2 inch diameter loops in the core center. Other combinations of closed loops could be substituted in this concept, but it should be observed that the placement of closed loops in the vertical core reactor is subject to two restrictions. One is that the loops should not prevent the offset handling mechanisms from reaching driver fuel positions. The other restriction is the limited space available for handling equipment at the operating floor level. It was found necessary to tilt the outer loops slightly ($3\text{-}1/2^\circ$) to provide adequate handling space as shown in Figure VI-A.

The loop tubes are about 40-feet long and are supported by the reactor shield plug near the upper end of the loop. Coolant supply lines penetrate the tubes in an accessible position above the reactor shield plug. A loop tube is composed of two concentric double-walled tubes which provide a thermal barrier between the inlet and outlet loop coolant and between the outlet loop coolant and the reactor coolant. These tubes are connected at the upper end only, and expansion between the double walls of the tubes is accommodated by a bellows connection between the walls. Welded construction is used for all connecting joints that are exposed to either loop or reactor coolant. Materials of construction are assumed to be equivalent to those indicated in the interim reference concept report, BNWL-CC-400.

Test specimens are removed from the outer loops, immersed in their own loop coolant, in a transfer pot as described in Section IX of this report. The upper section of the loops is ob-round allowing the specimen to remain immersed in loop coolant while being placed in the pot. The pot is capable of

removing 100 Kw of decay heat, although the capability of decaying a specimen within a test loop has been provided and would considerably lower the heat removal requirements. The transfer pot technique is not used for the removal of center loop test specimens since the space available prevented the use of ob-round tubing. As a result, it is necessary to remove a center loop to remove the specimen for examination. A simplified removal procedure, shown in Figure IX-E, uses a system of valves to provide natural circulation of the loop coolant during removal. Shielding and thermal insulation are placed around each of the center and outer loops such that work can be performed on any one loop without draining or disturbing neighboring loops.

The closed loop test facilities of this reactor concept compare favorably with those provided by the "Reference" design for the initial operating phase. The reduction of possible gravitational effects upon test specimens and handling equipment, due to the use of smaller loop tilt angles, is an advantage of the vertical core concept. Loop construction and handling techniques are approximately equivalent. Direct instrument connections between experiments and the reactor exterior for all driver core subassemblies is not possible in the vertical core concept. However, a means of indirectly instrumenting these positions has been provided by instrument fingers located immediately above each subassembly. Open loop testing could also be accomplished in a closed loop position of the vertical core reactor with complete specimen instrumentation. Other test loops provided by both concepts include a rabbit facility and an oscillator rod.

IX. MECHANICAL HANDLING SYSTEM

The mechanical system described in Figures IX-A through IX-E is a concept of the handling and transfer equipment which services the vertical core, fixed shield plug reactor. The service functions include handling driver fuel, test fuel specimens, shield plugs, loop tubes and control rods both within the reactor and above-floor within the reactor building. The objective of this design study was to determine the equipment size, complexity, fabricability and estimated development effort required to produce a functional and reliable system.

This reactor concept consists of a vertical core of free thermal bowing subassemblies, with in-reactor handling mechanisms operated through a fixed shield plug and closed test loops extending down through the plug and core as shown in Figure VI-A. Handling operations within the reactor include the transfer of fuel from the core to decay storage positions and the transfer of fuel and other subassemblies to the transfer pot of the above reactor handling cask when it is lowered through an exit port in the reactor shield plug. The above reactor handling cask is used to remove fuel, closed loop test specimens, loop tubes and control rods from the reactor, while a separate handling cask is used to remove the shield plugs.

The fuel handling cask, Figure IX-A, is essentially a shielded, gas-tight tubular vessel 48-feet high, with a 5-1/2 foot outer diameter, weighing approximately 150 tons. The cask is supported and positioned by an overhead polar bridge crane as shown in Figure IX-A. Sealing of the transfer cask to the reactor shield plug during transfer operations is accomplished with the use of tubular, gas tight, extensions which encircle each of the items to be transferred. These extensions allow the loop coolant supply lines to remain in place above the reactor shield plug during handling operations.

The mechanisms within the fuel handling cask include a gripper and transfer pot mounted on separate carriages. The transfer pot is used for fuel handling while both the gripper and transfer pot are required for removing test specimens from the outer loops. Only the gripper is required to remove the center loops, outer loop tubes and control rods. Handling adapters are manually attached to these items prior to their removal using the gripper. A variable reach feature allows the gripper to remove any of the control rods or center loops while the cask is sealed to the center sealing adaptor.

According to the design criteria the fuel handling equipment must be capable of handling decay heat levels as high as 100 Mw. To satisfy this requirement, fuel and test specimens are removed from the reactor immersed in sodium in the transfer pot. Decay heat is removed from the sodium by NaK coolant circulating through the space between the inner and outer walls of the pot. A pump is mounted on the pot carriage, within the transfer cask, to provide a NaK flow of about 4 cfm to a finned heat exchanger that is also located on the carriage. Heat is transferred from the heat exchanger by the combined effects of radiation and forced argon circulation within the cask. A radiating surface of 80 square feet and an argon (100° F inlet) flow rate of 35 ft/sec is more than adequate to remove 100 Kw of heat with NaK temperature of 600° F and 500° F at the inlet and outlet of the heat exchanger. The argon circulation system is a closed, shielded system requiring an argon to air heat exchanger.

The transfer pot and gripper carriages are positively driven in both directions to provide positive carriage position feedback to the operator. The carriages can be serviced by removing a wall section near the top of the cask or by disengaging the carriage from the cask and lowering it

into a maintenance pit with the equipment hoist. The fuel transfer cask has been presented as a single station subassembly transfer machine. However, the design could be developed to accommodate a number of subassemblies without significantly affecting its reliability or cost.

Three variable-reach offset handling mechanisms in a fixed shield plug, and three exit ports are used in the handling and transfer of driver fuel, as shown in Figure VI-A. These mechanisms are mounted on the reactor shield plug and positioned such that all core, transfer pot and decay storage positions can be reached without disturbing the closed loops or control rod packages. Enough detailed design has been completed on the mechanism, as shown in Figures IX-B and IX-C, to establish the size of drive package required and the linkage configuration. A series of concentric tubes, each actuated by a separate screw, are used to adjust the gripper offset, actuate the gripper latch and elevate the gripper. Bellows seals are used to prevent sodium deposition on the ball screws. Significant features of the mechanism include the ability to change the gripper offset without affecting the gripper elevation and the capability of applying forces up to 10,000 lbs. for removal of stuck subassemblies. Also, the mechanism can be lifted vertically out of the reactor for maintenance or the drive package can be detached without removing the mechanism from the reactor. Maintenance penetrations have been provided in the reactor shield plug adjacent to each of the mechanisms.

Figure IX-D is a schematic illustration of the method used to remove test specimens from the outer loops. These loops have been tilted $3-1/2^{\circ}$ from a vertical axis to provide access for the transfer cask. Figure IX-E shows the removal technique used for center loop test specimens, which involves use of the loop tube itself as the transfer pot. A valving system has been provided which allows natural circulation of the loop coolant during transfer. Shielding has been provided around the loops and coolant supply lines to permit

cutting and welding operations when removing or installing a loop.

The handling system concept presented in this report represents only a preliminary design effort. It is possible however, to make an evaluation of the system as to its reliability and feasibility. The offset handling mechanism and drive system utilize proven concepts and developmental effort required to design and fabricate these items could successfully be accomplished within a scheduled time period. Although the transfer cask is physically large, it contains relatively simple mechanisms that can be maintained and adjusted.

The fuel handling system for both this concept and for the "Reference" design as described in BNWL-CC-400 requires a fuel transfer cask and also a plug transfer cask. The overall cost of these casks and complexity of the mechanisms within the casks are similar for both systems and neither would be expected to produce design problems affecting a scheduled development program. The precision polar gantry with its universal tilting table and sensitive homing devices, as required for the "Reference" design, would be a large cost item as a result of its multiple angular positioning requirement, whereas the alternate concept requires only a two position device.

The need for variable reach offset handling mechanisms in the alternate concept represents additional items of development and fabrication cost. However, the ability of this concept to provide decay storage facilities within the reactor and the possibility of using a multiple station fuel transfer cask are significant advantages. Both of these capabilities would substantially contribute towards satisfying the general criteria of developing a reactor with a high availability factor.

FIGURE IX-A

DATE	BY	CHARGE	DATE	BY	DATE	BY
11/1/68	BAH/MLR	REVIEW VERTICAL CASK CONCEPT				
11/1/68	BAH/MLR	REVISED FUEL LOOP AND TEST SPECIMEN HANDLING CASK				
11/1/68	BAH/MLR	ATOMIC POWER DEVELOPMENT ASSOCIATES, INC.				
11/1/68	BAH/MLR	6XN-6278

FIGURE IX-B

DATE	BY	CHKD	DATE	APPR	DATE
CHANGE					
TITLE: FIVE VERTICAL CORE CONCEPT FUEL HANDLING MECHANISM					
ATOMIC POWER DEVELOPMENT ASSOCIATES, INC.					
DRAWN: 4-07					
6XN-6276

FIGURE IX-C

LTN	NO.	DATE	BY	CHKD	DATE	APP'D	DATE
CHANGE							
TITLE: FRT VERTICAL CORE REACTOR							
SUBJECT: FUEL HANDLING MECHANISM DRIVE SYSTEM							
ATOMIC POWER DEVELOPMENT ASSOCIATES, INC.							
DRAWN BY: J. E. AL							
CHECKED BY: J. E. AL							
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X. SUMMARY COMPARISON OF ALTERNATE AND REFERENCE DESIGN CONCEPTS

The vertical core reactor in this alternate concept is appreciably smaller than the reference concept reactor in both plan and elevation dimensions, although both concepts have the same fuel region length in both driver fuel subassemblies and loop test specimens. The alternate concept reactor is also fueled with a smaller critical mass and operates at a lower power level for the same flux levels. The overall result is a reduction in size of plant equipment and reactor building, and a reduction in capital and operating costs.

The alternate concept employs proven features and components to a much greater extent than the reference concept, leading to reduced development time and cost, a higher level of confidence that the design and construction can be completed within a predetermined budget and schedule, and expectation of higher on-line availability of the facility.

A noteworthy feature of the alternate concept is the provision of in-reactor decay storage for the driver fuel, which should increase the reliability of operation of the fuel transport equipment. It is also noteworthy that the operation of transferring the driver fuel out of the alternate concept reactor does not involve the use of any gripper mechanism which must pass into and out of the primary coolant sodium during the process of transfer. The use of a pantograph-type variable-reach arm on the 3 offset handling mechanisms which operate in the reactor does not have direct precedent in existing in-reactor fuel handling equipment in sodium-cooled reactors. However, its use in such application is already planned in both British and German designs for sodium-cooled fast power reactors. Furthermore, in this alternate concept the pantograph and gripper mechanism remain in the primary sodium

coolant once the machine is installed in the reactor, and, therefore, the mechanism is not thereafter subjected to a change in environment.

The alternate concept test facilities are relatively limited in number and location in the reactor but they are adequate for effective use of the reactor for testing purposes and are judged to be the maximum number of testing facilities that can be accommodated in a test reactor without degrading its on-line availability. Moreover, the alternate concept does have the advantage that the loops in the high flux test hole in the center of the core can be varied from a cluster of 3 small diameter loops, as shown in this report, to other loop arrangements, including installation of an 8-inch diameter loop. In addition, the maximum off-vertical angle of $3\text{-}1/2^\circ$ of the peripheral loops minimizes concern about gravity effects on test specimens which might be considered objectionable.

The relatively smaller degree of flexibility in number and location of loops and a lower level of accessibility for internal maintenance operations are the major concessions that have been made in the alternate concept to achieve the advantages outlined above.

XI. RECOMMENDATIONS

The vertical core alternate concept proposes a compact reactor and plant that will involve a minimum amount of innovation in design and that will incur minimum development, capital, and operating costs. The reactor will provide test facilities which are judged to be sufficient to maintain an adequate level of output of test results for the fast power reactor development program. The alternate concept should, therefore, receive careful consideration for FFTF.

This alternate concept is the product of a brief study which was limited to little more than one man-year of total effort. It is, therefore, far below the level of design development of the reference concept. Furthermore, the relatively modest test facilities of this alternate concept, combined with the smaller size of the reactor, the shorter length test loops, the smaller size handling equipment, and the lower reactor power level, leave little similarity to the reference concept in the size of the reactor building, the size and arrangement of the external test facilities, and the overall size of the plant. Therefore, it is recommended that this concept receive additional design study at a high enough level of effort to include layouts of the reactor building and equipment, and external test facilities in sufficient detail to provide a basis for a cost estimate that can be compared to that for the reference concept.