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AN INFORMAL REPORT

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Status of Work Related to
The King Reactor
as of March 1972

los alamos
scientific laboratory

of the University of California

LOS ALAMOS, NEW MEXICO 87544

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Compiled by

L. D. P. King

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STATUS OF WORK RELATED TO THE KING
REACTOR PROPOSAL AS OF MARCH 1972

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as of March 1972

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STATUS OF WORK RELATED TO THE KING REACTOR PROPOSAL

AS OF MARCH 1972

Abstract

The purpose of this report is to summarize past and present work aimed at developing and constructing an advanced high intensity fission neutron facility using the Kinetic Intense Neutron Generator or KING Reactor concept.

Some of the promising features of this concept are first briefly described and referenced to existing reports. Then the results of a preliminary study and cost estimate for constructing a 25 MW KING Reactor at Los Alamos are presented. This work represents the efforts of a small task group assembled in November and December 1971. Finally the status of work and experiments related to the understanding and design of such a reactor system which are now in progress are summarized.

1. INTRODUCTION AND SUMMARY

The proposal for the construction of a 25 MW Kinetic Intense Neutron Generator or KING Reactor at Los Alamos¹ continues to look very promising. Such a reactor would provide the Laboratory with a versatile steady state neutron and gamma ray facility second to none. The status was recently discussed.²⁶

A small task force has spent some time during November and December to come up with a preliminary design and cost estimate for the construction of such a reactor. In spite of the much more costly AEC nuclear construction code requirements, a 25 MW KING Reactor could be constructed at the present UHTREX facility at a comparable cost to such existing high flux neutron research facilities as are at Brookhaven and Oak Ridge.

All existing steady-state high intensity neutron facilities have been flux limited by thermal stress problems in the fuel elements. Any appreciable increase in heat extraction from conventional fuel element type cores is not anticipated since thermal stresses and vibrational forces are approaching the strength of available fuel element materials.

The KING reactor concept circumvents this flux limitation by making use of the large total heat capacity inherent in a rapidly moving liquid fuel. The use of an aqueous uranyl sulfate fuel with its high uranium solubility² coupled with rapid flow through the reactor core has a number of important advantages over existing fuel element type research reactors.

1) No internal heat exchanger is required even at large total power outputs.

2) A complete optimization of the core geometry is possible for the first time to produce the highest fluxes most efficiently.

3) Very high pseudo-specific powers are possible in the rapidly moving fuel mode of operation.

4) Low fuel operating temperatures are possible even for large total power outputs in the small core. This assures chemical stability in the fuel and eliminates or minimizes corrosion problems.

5) Frequent core replacement due to uranium burnup encountered in existing high flux reactors does not occur in the KING Reactor due to the large

total fuel inventory in the fuel system external to the core region.

6) The liquid form of the fuel permits continuous or batch-wise fuel additions or reprocessing without reactor shutdown.

7) Previous work with aqueous fuels has shown the inherent safety of such systems due to the large prompt negative temperature coefficient. A recent study of the Kinglet experiments has confirmed these results.³

8) Conventional types of water-to-water heat exchanger design are adaptable to the main fuel heat exchanger since it is external to the core in the KING Reactor.

Calculations⁴ have shown that this reactor concept is the first design that permits a complete optimization for neutron flux. This is due to the elimination of any internal fuel structure and the high solubility of uranium in the aqueous uranyl sulfate fuel. The core geometry can then be adjusted to give complete neutron thermalization in a central flux trap while maintaining a small total core size necessary for high neutron fluxes. Thermal neutron fluxes of $2-3 \times 10^{14}$ /MW are indicated.

A high performance index or ratio of thermal flux to reactor power is important in the design of a super high flux reactor since total power output and uranium burnup can become important cost limiting items.

A gamma irradiation facility separate from the KING reactor core, which uses a portion of the circulating fuel, looks very attractive because of the high intensities available and the large gamma-to-neutron ratios achievable.

At a 25 MW reactor power level and fuel inventory of 1500 liters, for example, the mixed fission product gamma activity is calculated to be 2×10^6 curies/liter. The associated low energy delayed neutrons can be made vanishingly small with only a small loss in the gamma activity by neutron shielding and increasing the delay time following fuel core exit to arrival in the separate gamma facility.

The reactor containment vessel and piping systems are simplified by low pressure and

temperature requirements. For operation at 25 MW below approximately 32°C, no pressurization is required in the gas chamber to avoid boiling. For short-time experimental operation to determine higher power data at higher temperatures, a maximum vessel operating pressure of 500 psi is anticipated to reduce radiolytic gas voids and suppress boiling.

The KING Reactor system will require installation of a primary-secondary heat exchange system, different from and larger than the UHTREX systems, which will dissipate heat from fuel-to-water-to-air. The secondary heat dump to atmosphere is proposed to be with open cycle cooling towers.

A bypass-type fuel cleanup system is planned to remove fission fragments and other fuel sediments on in-cycle basis. Spent fuel processing will not be part of this project. A fuel reservoir will be provided as an emergency sump for the normal complement held in piping, heat exchangers and reactor sump. The piping volume now estimated to be about 1500 liters will be minimized to keep fuel inventories as low as practicable.

Present design criteria for a 25 MW KING reactor at Los Alamos call for no release of radioactive material to the environment. The fuel system will have complete double containment. A secondary containment shell in the present UHTREX building and the new adjacent experimental building will prevent escape of any radioactive gases to the atmosphere.

Solid fission products will be removed on a small batch basis from the fuel for packaging and storage according to established regulations. While not to be considered as waste stream, at some time it may be necessary to convert the whole irradiated fuel charge to a solid form for shipping and/or storing purposes. In this instance, the end product would be a dry uranium compound containing a great deal of activity. This material would be packaged and stored or shipped as prescribed by the applicable regulations.

Gaseous fission products after extraction from the fuel solution will be diluted to less than 1% of its original concentration by a stream of helium. The helium purge stream then sweeps the gas mixture through a demister, a radiolytic gas recombiner, a steam condenser and heat exchanger. Excess oxygen

is maintained in this stream by a controlled feed just upstream of the recombiner. The helium is recirculated while the recombined radiolytic gas or water is returned to the fuel circulating system. A part of the purge stream flows intermittently through a sidestream cleanup loop for the removal of gaseous fission products, molecular sieves in a liquid nitrogen cooled cold trap adsorb the noble gases, oxygen and other volatiles that contaminate the helium.

After a period of on-line use, the cold trap is isolated and its coolant shut off. As the bed warms up, the adsorbed gases flow to a fission gas compressor which loads the desorbed gases, principally oxygen, into high pressure gas bottles. The bottles are then removed to another facility for recovery of the fission gases, mainly ⁸⁵Kr and ¹³⁵Xe.

Radiation for workers will be kept at approved levels. Ventilation of the secondary containment to prevent build up of activation products will be handled by closed circulating loops with suitable filters.

Within the rooms air moves from the regions least likely to be contaminated to regions where contamination is more likely and then returns to the filters before recirculation.

2. REACTOR SITE

A small task force has spent some time during December 1971 to come up with a preliminary design and cost estimate for a 25 MW KING Reactor at Los Alamos.

This study proposes to utilize the former UHTREX Reactor Facility at TA-52 which is available and not presently in use. This building with its auxiliary equipment was designed for a 3 MW experimental reactor and therefore offers numerous features ideally suited for a KING Reactor. This is particularly true in regard to the secondary containment shell and the means for handling radioactive fuel and gases. The building shielding was not designed, however, for 25 MW reactor operation or to provide a neutron beam research capability.

Since it is desired that a 25 MW KING reactor provide the most versatile and advanced experimental

beam and irradiation facilities, it has been found advantageous for a number of reasons to locate the reactor with its biological shield in a new experimental building located in close proximity to the east shield wall of the UHTREX building.

1) The reactor core geometry and beam facilities can be integrated as a unit to provide the best possible research facility rather than trying to fit the reactor core into a modified shield wall and room.

2) No costly shielding modifications are required in the existing UHTREX building when the higher power KING reactor is placed in its own new biological shield.

3) The UHTREX reactor core would not have to be removed since sufficient room exists for all the KING reactor auxiliaries if the UHTREX auxiliaries only are removed.

Existing utility systems for building ventilation, filtration and site water supply and storage will be upgraded to meet current criteria and new demands, and a new 13.4 kV power line will be extended to the site.

Figure 1 is a general plot plan for the proposed 25 MW KING reactor facility. The relation of the proposed new reactor and experimental facility building in relation to the existing UHTREX building is shown. All reactor auxiliaries which contain radioactive materials will be handled in the secondary containment portion of the UHTREX building.

3. BUILDINGS AND FACILITIES

Figure 2 is a floor plan of the proposed new KING reactor and experimental facilities building which is placed adjacent to the east wall of the existing UHTREX building. Figure 3 is a longitudinal section through the reactor room and Fig. 4 a cross section of the reactor center line through the reactor room showing its relation to the existing UHTREX building.

The new building addition will have a gross area of approximately 5800 sq ft and will be 50 ft high. The experimental space will house experiments utilizing neutron beams from the reactor. Space in the building above the reactor will be

usable for flux trap insertions for maximum irradiations and production of transuranium elements, and for additional beam ports. Space is available to the east for long neutron flight paths and for future expansion of the experimental area.

Figures 5 through 9 show some of the details of the existing UHTREX facility which will be available for the KING Reactor when the UHTREX core and its auxiliary equipment are removed. The proposed reactor additions are now shown against the east wall.

Figure 5 is a plan view of the ground floor immediately over where the KING reactor primary and secondary heat exchanger and gas handling equipment will be placed within the secondary containment shell. The removable shield blocks which permit easy access to KING Reactor auxiliary equipment, below, are shown.

Figure 6 is a floor plan of the operating level one floor below ground level. The former UHTREX reactor and recuperator room shown will be used to contain such radioactive elements of the KING reactor as the external fuel loop, fuel pumps and primary and secondary heat exchangers. The diluent gas handling and clean up equipment will make use of the existing gas clean up room. The location of the control room is also shown.

Figure 7 is a plan view of the basement level. Figures 8 and 9 are East-West and North-South sections through the existing facility. Both of these figures show the secondary containment shell region which is ideally suited to house all of the auxiliary equipment for the KING Reactor which contains radioactive material or is subject to nuclear radiations.

The advantageous arrangement of the 30 ton crane for handling the large shield blocks and any heavy equipment components is shown. After removal of the shield blocks the special remotely controlled Minotaur manipulator shown can be used for repair or replacement of radioactive reactor auxiliary equipment anywhere within the secondary containment shell.

The subbasement level region which will be used for fuel discharge is also indicated in Figs. 8 and 9. Fuel storage as well as emergency

Fig. 1. KING reactor - plot plan.

Fig. 2. KING reactor - floor plan

Fig. 5. UHTREX building - ground floor plan.

Fig. 6. UHTREX building - operating level.

Fig. 7. UHTREX building - basement level floor plan.

Fig. 8. UHTREX building-east/west section.

Fig. 9

Fig. 9. UHTREX building - north-south section

dump tanks will be placed in this region. These are discussed in Section 7.6 below.

A preliminary overall cost estimate for the KING Reactor buildings as well as the overall facility is given in Appendix A.

4. REACTOR DESIGN

The basic features of the KING reactor concept have been summarized previously.¹ A number of features of any high flux research reactors must, by necessity and experience be similar. This is particularly true in the choice of reflector material and the general layout of the experimental irradiation and beam facilities within the biological shield.

The KING reactor is no exception to this rule so that the design of the experimental beam and port facilities near the core and within the biological shield can make use of much previous design experience. Although the flux optimization possible in the KING reactor reduces the size of the core over those in existing high flux reactors such as the HFBR at Brookhaven and the HFIR at Oak Ridge beam and port facilities can remain much the same.

The core itself is substantially different in the KING reactor, however, due to the use of a liquid fuel and the novel mode of high velocity fuel replacement within the core.

Figure 10 illustrates the core model arrived at from initial optimization studies.⁴ This model has been used in later calculations for a preliminary design study for a 25 MW reactor. The small efficient core of the KING reactor with its central flux trap region can achieve high central fluxes as well as large external leakage fluxes.

The unique mode of operation using a rapidly circulating aqueous uranyl sulfate fuel also permits the incorporation of a powerful gamma ray facility essentially free from neutrons and well separated from the reactor core region.

The reactor physics of the KING reactor are discussed in Appendix B. The effects of void formation in the core are discussed in Appendix C.

Calculations related to the neutron and gamma activity in the circulating fuel solution are given in Appendix D.

5. EXPERIMENTAL FACILITIES

A 25 MW KING reactor has the potential of being the most versatile high flux research facility. This is due to its ability to provide high intensity thermal and fission spectrum neutrons as well as a powerful independent fission gamma ray source.

Figure 11 shows a section through the proposed 25 MW KING reactor. The design of the experimental beam and irradiation facilities in the biological shield will make use of the experience gained in the design and operation of the Los Alamos Omega West Reactor, the Oak Ridge HFIR and the Brookhaven HFBR Reactor.

Numerous radial and tangential irradiation and beam ports will penetrate the thick heavy biological shield and beryllium reflector. Only a single typical port is shown in Fig. 11.

It is planned to incorporate a graphite thermal column in the KING reactor to satisfy the need for a large accessible volume with well thermalized neutrons and a minimum of gamma ray background. The need for such a facility is well documented at Los Alamos and the design of the new column will be based on experience gained from the Water Boiler and Omega West Reactor facilities. The thermal column region is not visible in the figure.

Since the fuel circulates from the reactor core to an external heat exchanger a portion of the fuel can be diverted through a special gamma ray facility completely separate and independent of the core proper.

Calculations (see Appendix D) indicate that for a 25 MW steady state reactor power level and a 2000 liter total fuel volume an essentially neutron free intensity of 2×10^5 curies/liter is available for the design of a gamma ray facility.

A five liter fuel volume (15 in. of 5 in. diam. pipe) for example could therefore provide a very intense neutron free gamma source of about 10^6 curies or 4×10^{16} MeV/sec completely separate from the other reactor core irradiation capabilities mentioned above.

Fig. 10. KING reactor neutronic model.

6. SHIELDING CONSIDERATIONS

The overall shielding for the KING reactor consists of two principal parts: (1) the old UHTREX building designed to house a 3 MW reactor with its auxiliaries in the former reactor-recuperator room and (2) the new addition containing the reactor and experimental facilities.

The UHTREX building shielding has been described.⁵ Since the KING reactor will be placed in a new experimental facility building direct radiations from the reactor are substantially reduced within the old building. Part of this stems from the 5 ft thick east exterior wall of the old building and some from additional shielding supplied by the new building wall and biological shield of the reactor itself (see Fig. 12). Distances between the new reactor location versus the UHTREX core location are greatly increased to all parts of the building in particular to the control room area which is at the west end of the old building.

These two factors more than make up for the factor of 8 increase in power between the old and new reactors.

Shielding for the new building addition is provided by the biological reactor shield surrounding the core on the sides and a 24 ft water depth above the core (see Fig. 11). The overall core shield consists of a 2 ft thickness of water surrounding the beryllium reflector, followed by the 1-in. thick steel tank wall and 6 ft of heavy concrete.

Results of calculations on neutron and gamma activity, fission product inventories and radiation heating in a KING reactor are given in Appendix D.

7. REACTOR SYSTEMS

7.1 General Description

The KING Reactor is a high intensity flux, research reactor which uses a liquid fuel flowing at high velocity through the core to generate 25 MW in a small critical volume. The components that comprise the total reactor system can be divided into a number of flow loops. These loops are listed below and are shown in the block diagram in Fig. 12.

- Fuel flow system
- Fuel storage system

- Radiolytic gas condenser cooling system
- Reactor hardware and pool cooling system
- Fuel storage cooling system
- Fuel storage emergency cooling system
- Cooling tower system
- Diluent gas system
- Fuel processing system.

The fuel loop contains a solution of $\text{UO}_2\text{SO}_4 + \text{H}_2\text{O}$ which is pumped through the reactor core in a critical configuration. Flow velocity through the core is maintained fairly high to limit the temperature rise in the fuel solution. As it passes through the core, fission product and radiolytic gases are formed in the fuel solution due to the fission process and the dissociation of the H_2O by fission fragment bombardment. The mixture of gas and fuel solution flows from the reactor to one or two liquid-gas separators where the gases are separated from the fuel solution and piped to the diluent gas system for processing the fission gases and recombining the radiolytic gas. The degassed fuel solution then passes through the fuel to water heat exchanger where the fuel temperature is reduced before it enters the fuel pump for another cycle.

An intermediate water-to-water heat exchanger is used between the fuel heat exchanger and the cooling tower water. This secondary cooling loop isolates the water from the cooling tower water in case of a fuel leak into the water side of the fuel heat exchanger.

Cooling the reactor hardware and the water in the pool surrounding the reactor is provided by the reactor hardware cooling loop. Water that is circulated through the hardware and pool passes through a water-to-water heat exchanger where the thermal energy is transferred to the cooling tower water.

The radiolytic gases are recombined in the diluent gas system. This produces steam which is converted to water in the steam condenser. Cooling for the steam condenser is provided by the radiolytic gas condenser loop. Cooling water from the condenser passes through a water-to-water heat exchanger transferring the condenser energy to the cooling tower water. The water that is generated in recombining the gases is pumped back into the fuel loop.

Fig. 12. Cooling water - single-line diagram.

A fuel storage system is required for storage of the fuel when the fuel loop is emptied or when dumping the fuel is necessary in case of an emergency. Cooling for the fuel storage is provided by the fuel storage cooling loop which is another water-to-water heat exchanger connected to the cooling tower system.

An emergency cooling system is tied into the storage cooling loop and consists of a water storage tank which can provide cooling water by gravity flow. A hold-up storage tank is used in conjunction with the emergency cooling to collect and store the emergency cooling water after it passes through the fuel storage system.

7.2. Fuel Loop Details

The nominal design power for the reactor has been selected at 25 MW. Of this amount, it has been estimated from preliminary calculations that 22.2 MW of energy is deposited in the fuel solution, 1.5 MW in the reactor hardware and pool water, and 1.3 MW in generating radiolytic gases. The fuel flow rate was based on removing 22.2 MW of power by allowing the fuel temperature to rise 54°F during its transit through the core. This ΔT was selected because the specific power generated in the fuel is well below the value where precipitation in the fuel can occur. It also keeps the maximum fuel temperature low which helps minimize the corrosion problem. Calculations for this study were based on a core which had a water moderating island surrounded by an annular fuel region. The fuel annulus has a 10.64 cm i.d. and 16.84 cm o.d. Some design points for the fuel loop are shown below.

Power generated in fuel	22.2 MW
Fuel temperature inlet to core	122°F
Fuel temperature exit	176°F
Core ΔT	54°F
Fuel flow rate	2942 gpm-526 lb/sec
Fuel velocity in core	45 ft/sec
Gas volume in fuel solution exit	≈8%
Core exit pressure	≈125 psi

Values of specific heat and density used were $c_p = 0.75 \text{ Btu/lb-}^\circ\text{F}$ $\rho = 0.046 \text{ lb/in.}^3$ Pressure level at the core exit was estimated at 125 psi. This

pressure is necessary to compress the gas bubbles and void pressure drop problems. After the fuel solution leaves the core it enters two 5 in. diam pipes to continue the flow to the separators. The reason for splitting the flow into two pipes is to reduce the fluid velocity and therefore the overall pressure drop in the fuel loop. This results in reducing the pumping power requirements for the fuel pump. A single pipe of larger diameter cannot be used because it can go critical.

An accumulator is located between the reactor and gas separator to adjust the operating pressure of the fuel loop system and to provide a volume for the liquid fuel to expand into as it is heated.

From the accumulator, the fuel solution enters one or two gas separators where the radiolytic gases are separated from the solution and piped to the diluent gas system for processing. At present the gas separator has not been designed, but it may be similar to the type used by Oak Ridge in the HRE-2 liquid reactor. This separator is an in-line pipe design which consists of a section of pipe with stationary vanes located at the inlet and exit ends of the pipe. The purpose of the vanes is to impart a rotation velocity in the fluid as it enters to develop a centrifuge action on the fluid. Gases are collected in the void formed along the center line axis and are piped off to the diluent gas system. Vanes at the exit are used to straighten the flow and to recover some of the rotational energy.

Separators for the KING reactor should operate at about 100 psia to reduce the gas volume and consequently the size of equipment in the diluent gas system and the pumping power requirements.

After leaving the gas separator, the fuel passes through the fuel heat exchanger. This heat exchanger will probably be a tube and shell type design with fuel flowing through the tubes and cooling water on the shell side. Hopefully, the design can be such that the tubes can be removed from the shell for cleaning and repair if necessary. Fuel enters the heat exchanger at 176°F and leaves at 122°F while cooling water enters at 100°F and leaves at 125°F.

On leaving the heat exchanger, the fuel enters the pumping system where the pressure is increased

and again starts the cycle through the fuel loop. The pumping system contains a main and secondary pump with associated valves and hardware for controlling the flow between the two.

The main pump is normally used during the regular operations. It is capable of pumping about 3000 gpm with a ΔP rise across the pump of 150 psia, which is necessary to overcome the impedance of the loop. The loop impedance is only a rough estimate at this time -- a more accurate value will depend on the details of the piping, valves, heat exchangers and gas separator in the final design. The horsepower for the pump will probably be about 500 hp. Cost estimates for this pump have been made based on a can-type pump; however, present feelings are that due to the high fuel radiation level this type of pump may not be suitable. A better candidate for this application may be a limited leakage pump which will require some development effort in the seal area to adapt it for the KING reactor conditions.

The secondary pump is a smaller pump and is used during shutdown or emergency conditions. It will be normally used to circulate the fuel when the reactor is shut down, such as for a 5-day a week operation, etc. Circulating the fuel is necessary to remove the energy generated in the fuel due to fission fragment decay. Flow rate for this type of operation has been selected at 25% the normal flow. At this value, the flow rate is generous and at the same time greatly reduces the loop flow impedance and pump horsepower requirements.

The secondary pump also acts as an emergency back-up pump in case of a main pump or electrical power failure. Power to the pump is provided by the main and emergency power distribution system -- if the main electrical system fails the emergency power comes on automatically. The general philosophy of a two-pump system is used on all the other loops in the reactor system.

7.3. Fuel Cooling Loop

The final dump for the energy generated in the reactor systems is by evaporating water to the atmosphere. This is accomplished by the cooling tower system. The cooling tower water cannot be used in direct contact with components containing

radioactive fluids or materials because of the problem of leaks and induced activation in the cooling tower water. The general philosophy adopted is to use an intermediate cooling loop between the cooling tower system and the radioactive components to minimize the possibility of discharging radioactive material to the atmosphere. The fuel cooling loop is one of the intermediate cooling systems and consists of a tube and shell type heat exchanger, a pumping system, and a deionizer. Water from the primary fuel heat exchanger is pumped at 6033 gpm and 125°F to the tube side of the intermediate fuel water-to-water heat exchanger and leaves at 100°F. Cooling tower water on the shell side enters at 85°F and exits at 110°F--its flow rate is 6033 gpm. The ΔT rise for both sides is the same. Water from the tube side is piped to the secondary loop (pump system), which like the fuel loop, contains two pumps and necessary hardware controlling the flow rate and flow path. Under normal operations, all the water passes through the main pump. After leaving the pump the water flows to the fuel heat exchanger and passes through the deionizer unit to the return leg of the loop. The purpose of the deionizer is to continually process a small amount of the main flow so that the quality of the loop water is maintained. The resistivity and pH of the water are controlled to minimize corrosion and radiolytic gas formation. The impedance in the loop has been estimated and a minimum 25 psi ΔP across the pump is necessary. The flow rate and ΔP require a motor with a 200 hp capability. The secondary (cooling) loop does not have the same design restraints as the fuel loop, and therefore the pipe and valves can be much larger to reduce the fluid velocity and the flow impedance.

The secondary pump serves the same functions as previously described for the fuel loop--mainly for shutdown and emergency operating conditions. Flow rate through the pump is 25% that of the main pump.

7.4. Reactor Hardware and Pool Cooling Loop

The reactor hardware cooling loop is similar to the fuel cooling loop. It has a heat exchanger between the cooling tower system and the reactor, a pump system, and deionizer unit. Flow through the loop is 409 gpm for both the tube and shell side.

Water on the tube side enters at 125°F and leaves at 100°F while the shell side enters at 85°F and leaves at 110°F. The pump system and deionizer serve the same function as in the fuel cooling loop. Power requirement for the main pump is estimated at 25 hp. Flow from the pump is split into two lines just prior to entering the reactor pressure vessel. Most of the water flows through the reactor pressure vessel to cool the hardware in the pressure vessel -- mainly the beryllium reflector and irradiation capsules located in the center moderating island. After passing through the reactor, the flow is piped out of the system back into the return leg of the loop. The water that was diverted to the second line is dumped into the reactor pool for circulation before being removed. Since the pool is operated at atmospheric pressure, a submerged pump is necessary in the pool to pump the water back into the return leg. Estimates for this flow rate have not been made at this time. Power removal for the total loop is 1.5 MW.

7.5. Radiolytic Gas Condenser Cooling System

Gases which were removed by the gas-separator for processing in the diluent gas system are mostly $H_2 + O_2$. These gases are recombined forming steam which is condensed to water in the steam condenser. Cooling water for this condenser is provided by the condenser cooling loop which again contains similar components to other loops described. Inlet and exit water temperatures for the tube and shell side of the heat exchanger are the same as the reactor hardware and fuel cooling loop. Flow rate for the 1.3 MW of power generated is 355 gpm with a pressure drop in the system estimated at 30 psi. A 15 hp motor for the main pump should be sufficient.

7.6. Fuel Storage System

A storage capacity is needed when it is necessary to remove the liquid fuel from the fuel loop. Removal of the fuel from the loop will be required for repair work or component replacement or during certain emergency conditions.

The design utilizes two storage units, both capable of holding the fuel loop volume. The design of the storage unit as envisioned today is made from

a number of 5 in. diam pipes which are stacked parallel to each other and are spaced apart. Both ends of these pipes are welded into header pipes. The pipe size and spacing is selected to prevent the fuel from achieving a critical configuration.

Cooling of the fuel is necessary because of the decay heat generated from fission fragment decay. This can be accomplished by wrapping the pipe with tubing and flowing water through the tubing. The storage units are located at a lower elevation than the fuel loop system so that they can be filled by gravity flow as well as pumped flow. A gas pressurization system is tied into the system to provide overpressure on the fuel to refill the fuel loop. The storage unit capacity is estimated to be between 1000 and 1500 liters per unit. Actual volume required depends on the final design volume of the fuel loop system.

7.7. Fuel Storage Cooling System

Cooling for the fuel storage system is supplied by the usual heat exchanger, deionizer, and pumping system loop. Usually the system will not be operating unless there is fuel in the storage unit. Cooling flow rate is expected to be variable, with a large amount flowing initially followed by a reduction to perhaps 20% and then zero flow after the decay rate is low enough to cool to the atmosphere.

Calculations indicate that after 25 MW of operation for 35 days, the power generation in the fuel will be 800 kW at 100 seconds after shutdown, 450 kW for 1000 seconds, and 75 kW at the end of a day.

Increasing the operating time prior to shutdown only slightly increases the power generation-- it does however hold up the heating rate at the longer elapsed times. A ballpark estimate for the flow rate is between 150 to 200 gpm for the first day and 30 gpm for the rest of the time before flow shutdown.

7.8. Fuel Storage Emergency Cooling

An emergency fuel cooling system is maintained on the line in the event of complete power failure to both the main and standby cooling pumps. This system consists of an emergency storage water tank

which is gravity pumped through the system. Water from the tank is piped into the system by opening a three-way valve downstream of the main and standby pump systems. A similar three-way valve is located just upstream of the fuel storage heat exchanger to divert the water into a holding tank. The holding tank is used to contain the water which may have some slightly induced activation and also act as a secondary containment vessel in the event of a leak occurring into the cooling pipe during emergency operations.

7.9. Cooling Tower System

All the cooling water for the reactor system is provided by a standard manufactured open spray cooling tower. The tower is capable of supplying 30 MW of cooling with a design inlet water temperature of 110°F and exit temperature of 85°F. The pumping system is composed of two main pumps, each capable of pumping 4100 gpm and one standby pump with a 2050 gpm capacity. The main pumps require a 200 hp motor for each pump and 25 hp motor for the standby pump. No deionizer unit is used in the loop. Some treatment of the water will be necessary to reduce the algae growth.

7.10. Diluent Gas System

7.10.1 General Description

A preliminary flow sheet for the diluent gas system is shown in Fig. 13. From the liquid-gas separator in the fuel solution circulating loop, a mixture of gases, formed by the radiolytic dissociation of water and in the fission process, flows into the diluent gas system. The major components of the mixture are hydrogen and oxygen, the latter is slightly less than stoichiometric balance with the hydrogen. Other components are the fission product noble gases and halogens.

Immediately on entry to the gas system, the gas mixture is diluted to less than 1% of its original concentration by a stream of helium circulating at the rate of 3300 lb/h within the purge loop of the gas system. The helium stream then sweeps the gas mixture through a demister where water droplets and mist are coalesced and returned to the fuel solution loop. From the demister, the

helium flows to the recombiner and there, on the surface of platinum-coated pellets of alumina, the hydrogen and oxygen react to form steam. Just upstream from the recombiner, oxygen feeds into the stream in quantities controlled to maintain an excess of oxygen in the circulating helium. In the recombiner, the heat of reaction raises the helium stream temperature above 700°F and superheats the steam. To return the helium to the temperature of the fuel solution and to condense the steam, the next unit in the purge loop is the condenser, a heat exchanger cooled with water from an independent cooling circuit. The water condensed out of the helium drains into a system which collects and recharges water into the fuel solution circulating system. The helium flows out of the condenser into the purge gas blower and then returns to the demister for another cycle.

Although the purge loop removes the radiolytic gases from the helium stream, the volatile fission products that are not entrained in the demister or plated out on cool surfaces continue to circulate in the helium. To remove these radioactive contaminants, a part of the purge helium flows, intermittently and at a maximum rate of 35 lb/h, through a sidestream cleanup loop. The sidestream leaves the purge loop on the discharge side of the blower and flows through a water cooled condenser and centrifugal separator where water vapor is removed prior to the entry of the helium into one of the dryers. There, the water content of the gas is reduced to very low levels by adsorption on molecular sieves. The now dry helium then flows through heat exchangers where its temperature is reduced in preparation for entry into a liquid nitrogen-cooled cold trap. In the cold trap, molecular sieves adsorb the noble gases, oxygen, and any other volatiles that contaminate the helium. Beyond the cold trap, the now clean helium warms up in a recuperative heat exchanger and then enters a diaphragm compressor. Its pressure restored or elevated, the helium enters a storage tank from which it can return to the purge loop or recycle through the cleanup loop. The tank can serve also as a storage place for helium when the gas systems are shutdown and emptied and as a source for helium to be used in the pressure balancing of the fuel solution circulating system.

FURGE LOOP

Item	Name	Description
1	Demister	18" dia x 4' long tank with horizontal, 4" thick s. s. Yorkman demister element.
2	Recombiner	24" dia x 20' long shell with 18 annular tubes, 3" dia outside, 1" dia inside. Annulus packed with 1/8" dia alumina spheres, coated with 0.5% Pt.
3	Oxygen Supply	Cylinder bank with automatic feed to maintain measured O ₂ concentration in purge gas.
4	Condenser	24" dia x 20' long shell and tube heat exchanger.
5	Purge Gas Blower	Single-stage, high speed, centrifugal compressor to develop 5 psi, approximately 40 HP.
5a	Aux Blower (not shown)	Single-stage centrifugal compressor in parallel with 5 approx. 1 HP.
-	Piping	Main loop is 4", Sch. 40, seamless pipe
-	Valves	300 psi, bellows-sealed, air-operated, plug

GENERAL NOTES

1. Gross dimensions of large items are approximate, for space allocation use only.
2. All items in contact with wet helium are made of 316 s. s.

He CLEANUP AND STORAGE SYSTEM

Item	Name	Description
6	Water Separator	Double-pipe cooler and centrifugal separator.
7	Dryers (2)	1' dia x 4' long tank filled with 1/8" dia spheres, 3A molecular sieve internal electric resistance heaters for bed regeneration, 2 KW.
8	Dryer Vent Cooler	Double-pipe condenser for water removal during regeneration.
9	Vacuum Pump	Kimray EE-15
10	Recuperator	He to He heat exchanger
11	Precooler	He to He heat exchanger
12	Cold trap	1' dia x 5' double wall tank filled with 13 X molecular sieve.
13	He ₂ Supply	Standard He ₂ dewar and feed controls.
14	Fission Gas Compr.	2-Stage, diaphragm compressor, 5 psig to 1000 psig, 1 SCFM, 10 HP.
15	Fission Gas Storage	Cylinder rack, standard H bottles.
16	He Compressor	1-Stage diaphragm compressor, 100 psig to 500 psig, 10 HP.
17	He Storage	5' dia x 20' storage tank (UNIKEM HP receiver)
-	Piping	1/2" Sch 40, seamless pipe
-	Valves	300 PSI, bellows-sealed, air operated, plug (Meln 431270K)

Fig. 13. Diluent system - 25 MW KING reactor.

7.10.2 Purge Loop Details

Demister

The demister is a vertical tank, 18 in. diam x 4 ft long, through which the helium flows from bottom to top. Located across the tank near its top is a 4 in. thick pad of corrugated mesh knitted from small diameter stainless steel wire. Impaction of water particles on the wires causes the water to coalesce into large drops which fall into the bottom of the tank and drain back into the fuel solution. The tank and the mesh are made of Type 316 stainless steel, as are all other gas system components that come in contact with the acidic water vapor carried in the flowing helium.

Recombiner

Catalysis, by platinum deposited on 1/8-in. diam alumina spheres, effects the recombination of hydrogen and oxygen at the 176°F entrance temperature of the recombiner. The catalyst is contained in tubes hung from a plate in the top of the recombiner vessel. An annulus, formed between an outer, perforated, 3 in. diam tube and an inner perforated 1 in. diam tube, holds the spheres in each tube and exposes the maximum area of catalyst to the gas, which passes through the catalyst from the outside of each tube to its center and then upward and out of the vessel through the top. Maximum size of the vessel, by simple scale up from an existing recombiner, is 2 ft diam x 20 ft long. Further design studies will reduce the size of the recombiner and define its exact operating conditions.

The heat released by the recombination of the water causes the temperature of the flowing helium to rise above 700°F. This heat release dictates the high flow rate of the purge helium. At lower flow rates the helium temperature would rise to levels incompatible with the normal materials of construction for process equipment.

At the entrance of the recombiner, a controlled flow of oxygen enters the purge stream. The oxygen flow is regulated automatically by a control loop which senses oxygen concentration in the flowing helium. The excess oxygen performs two functions. It ensures complete removal of hydrogen in the recombiner and it maintains an oxidizing

atmosphere in the gas systems to prevent reduction of the oxide film which protects the stainless steel against corrosion.

Condenser

A shell and tube heat exchanger, with helium in the tubes and demineralized water in the shell, cools the helium stream to 176°F and condenses the recombined water. The load on the condenser is approximately 1.3 MW. Water drained from the condenser returns to the fuel solution via a collection system. The water flow out of the condenser is monitored as an indication of the recombiner performance.

Purge Gas Blower

Pressure drop through the purge loop is restored in a single stage, high speed, centrifugal blower. At a differential pressure across the blower of 5 psi, the blower power consumption is about 40 hp. Flow through the purge loop is controlled by regulation of the blower speed. Pressure buffered shaft seals prevent helium leakage from the blower.

A standby blower, of the same type as the on-line blower but with only 25% of its capacity, operates on power from the site emergency generators to ensure purging of the gases evolved from the fuel solution during reactor shutdowns.

Piping

The purge gas loop is made of 4 in. stainless steel pipe joined by radiographically inspected welds. Valves are of the forged body, bellows sealed, remotely operated type developed for systems which handle radioactively contaminated gases. Transducers are doubly contained and seal welded into the piping. The whole system is helium leak tested.

7.10.3 Cleanup and Storage System Details

Water Separator

A simple, water cooled heat exchanger coupled to a centrifugal separator, unit reduces the load on the dryer beds which follow. The heat exchanger is cooled with a flow of a few gal/min of water from the cooling circuit which produces the lowest temperature.

Dryers

To prevent the blockage of the heat exchanger and cold trap which are upstream in the cleanup loop, one of two dryers reduces the water content in the helium stream to a few parts per million. The desiccant in each 1 ft diam x 4 ft long bed is 1/8 in. spheres of molecular sieve. At an entering helium air temperature of 100°F, each dryer has enough adsorptive capacity for 100 h on line.

When a dryer becomes saturated, it is valved off, and the flow passes through the second dryer. The saturated desiccant is regenerated by heating it with internal electrical resistance heaters to 600°F while a vacuum is maintained in the dryer vessel. Water leaves the desiccant as a vapor and is condensed in a cooler in the line to the vacuum pump.

The desiccant, a molecular sieve with a small pore diameter, is not a good adsorbent for the noble gases at room temperatures, but some fission products may be entrained in the beds. Therefore, the beds are purged before regeneration and the exhaust gas from the vacuum pump is handled as a radioactive waste stream.

Cold Trap

The cold trap and its associated heat exchangers are designed to maximize the heat removal capabilities of liquid nitrogen. At cryogenic temperatures, a 4 in. diam x 4 ft bed of Type 13x sieves has the capacity to quantitatively remove all of the noble gases and the oxygen from 6 recycles of the purge loop helium. The cold trap is used only intermittently to reduce contaminant levels in the purge helium and is most efficient if the reactor is shut down and the oxygen concentration in the purge gas is reduced to a minimum.

After a period of on-line use, the cold trap is isolated and its coolant is shut off. As

the bed warms up, the adsorbed gases build up pressure and flow to the fission gas compressor, which loads the desorbed gases, principally oxygen, into high pressure gas bottles. The bottles are then removed to another facility for recovery of the fission gases, mainly ^{85}Kr and ^{135}Xe .

Helium Compressor

Motive power for helium flow through the cleanup system is supplied by a single-stage diaphragm compressor, which receives helium at 100 psia and discharges it at a maximum pressure of 500 psia. The high discharge pressure is used when the purge loop is emptied. Ordinarily, the compressor simply makes up the pressure loss through the cleanup loop. Maximum flow through the loop is fixed by the capacity of this compressor.

Helium Storage

On startup of the gas system, helium flows into the loops from a 50 ft³ storage tank which is charged from gas cylinders. On shutdown, the helium, after purification in the cleanup loop, is returned to the storage tank by the helium compressor. Helium for use in regulating pressures in the fuel solution circulating loop is supplied from the storage tank.

Piping

The piping in the helium cleanup and storage system is 1/2-in., sch 40, stainless steel seamless pipe. It is built to the same standards as the purge loop.

7.11 Fuel Processing

7.11.1 Introduction and Summary

The feasibility of processing irradiated fuel solution from the proposed 25 MW KING Reactor in the UHTREX facility was examined. The fuel was assumed to be 2000 liters containing about 200 g ^{235}U /liter as UO_2SO_4 in 0.5 M excess H_2SO_4 . It was concluded that the existing high-level hot cells (with modifications to the ventilation) and portions of the secondary containment would be adequate for the absolutely essential processing requirements, i.e., (a) removing fission and corrosion product solids from the fuel solution on a side-stream basis and (b) converting spent fuel solution to solid form suitable for shipment. It was also concluded that

the facilities would be adequate to include side-stream partial cleanup of soluble impurities from irradiated fuel solution on a limited basis (approximately 20 liters or 1% per day).

7.11.2 Facilities

Equipment which would require direct viewing and manipulator access would be placed in the existing hot cells. These are identified as the transfer lock (Room 212) and the transfer cell (Room 213) in UHTREX drawings. Only the latter can be considered a full-time hot cell. The transfer lock, as now constituted, may have limited use as a hot cell because of the door in its front wall. This door provides the only reasonable access to the cells and when open, precludes there being any activity in that cell. Any permanent modification to the transfer lock shielding is not feasible. At best certain small, simple pieces of equipment (filters, hydroclones, containers for storage of solids, etc.) might be installed in this cell, if they could be completely walled in by moveable shielding (lead bricks or plates). Having only one full-time hot cell would determine the complexity and through-put of any chemical processing.

Certain other equipment such as continuous evaporators, distillate tanks, and perhaps storage for one or two reactor fuel charges could be located in the fuel changing room (Room 217) and in the UHTREX reactor drive area which is directly below. The interior of these areas cannot be viewed directly and in high radiation fields physical access is possible only to the upper room by means of the Minotaur; therefore any equipment installed there would have to be very reliable and highly instrumented.

7.11.3 Solids Removal

Figure 14 shows a solids removal step.

It would serve to separate insoluble fission and corrosion products from the circulating fuel. Removal of these solids would reduce erosion and corrosion of reactor components and piping and remove some nuclear poisons. Fission products which might be removed to varying degrees in this step are those with a limited solubility in the fuel solution. These are the rare earths, the alkaline earths, zirconium, rubidium, palladium, and rhenium. ⁶

SOLIDS REMOVAL (INSOLUBLE FISSION AND CORROSION PRODUCTS)

ONLY ONE HYDROCLONE-FILTER UNIT WOULD BE USED AT A TIME. A TYPICAL FLOW MIGHT BE 200 LITERS/DAY THROUGH A HYDROCLONE. ABOUT 180 LITERS (DEPLETED IN SOLIDS) WOULD BE RETURNED TO THE FUEL LOOP, WHILE 20 LITERS WOULD BE FILTERED BEFORE BEING RETURNED.

Fig. 14. Solids-removal diagram.

ADU = Ammonium diuranate

Fig. 15. Converting spent-fuel solutions to solid form.

A small side stream would be processed through a hydroclone (hydraulic cyclone separator) to concentrate solids into a small volume (typically $\leq 10\%$ of the starting volume). The concentrated solids would be removed from the smaller volume by filtration. The overflow and filtered underflow streams now depleted in solids would be returned to the reactor. The solids would be packaged and stored.

The solubility of the rare earth sulfates decreases about one-half for each 25°C rise in temperature. Operating the side stream at an elevated temperature could be very useful in increasing the effectiveness of this step.

ORNL has reported a great deal of work on developing hydroclones to remove solids from reactor fuel solutions.^{7,8,9} However from the reported results that were located, it was concluded that this approach may not have been an unqualified success.^{10,11,12,13,14} It is felt, however, that their results were adversely affected by the large amounts of solids that were deposited in various parts of the reactor. Also the operation of the reactor, and therefore the hydroclone plant, was terminated before all their difficulties were understood and remedied.

Development of the solids removal step for the KING Reactor would require additional work.

7.11.4 Converting Spent Fuel Solution to Solid Form

Figure 15 shows a flow-sheet for a

process that could be used to convert spent fuel solution into solid form suitable for shipment for storage or processing off-site. The process involves precipitating uranium and soluble, non-volatile fission products and daughters, as well as soluble corrosion products, from basic solution in the form of ammonium diuranate and impurity hydroxides. The precipitate would be filtered and dried. The filtrate would be evaporated to yield a discardable distillate and a concentrate that would be recycled. The transfer cell (Room 213) would provide more than adequate space for the precipitation, filtration, and drying steps for 20 liters of spent fuel solution per day. Evaporators would be best located in the fuel charging room (Room 217), where they would be accessible to the Minotaur upon removal of ceiling shield blocks.

7.11.5 Processing Irradiated Fuel Solution

The activity of the fuel solution in the KING Reactor would essentially reach equilibrium in little more than one day of operation.¹⁵ The fission product inventory of the fuel solution would be reduced somewhat by loss of volatile fission products and by precipitation and removal of insoluble fission products, but chemical processing on a side-stream basis at the rate of 1% of the fuel volume per day would have virtually no effect on the activity. On the other hand, such chemical processing to remove soluble fission products and daughters, as well as soluble corrosion products, might prolong the useful lifetime of the fuel as follows. Consider for example, the case of 80% of fission products and daughters remaining in soluble form (20 g per day generated), constant generation of soluble corrosion products at the rate of 20 g per day, and about 50% removal of these in side-stream chemical processing of 20 liters of fuel solution per day. If the fuel were run for about 200 days and then such processing started, removal of impurities would balance their generation.

Figure 16 shows chemical processing which might provide partial cleanup of the uranium for return to the reactor. This proposed scheme consists of the following steps:

(a) Neutralize with NH_4OH up to a 20 liter batch of fuel solution. Add sufficient $(\text{NH}_4)_2\text{CO}_3$ to effect complete solution of the uranium, at the same time precipitating most of the chemical impurities as insoluble hydroxides or carbonates. Filter off the precipitate, which is packaged and discarded. Approximately 50% of the impurities might be precipitated in this step.

(b) Boil the filtrate to precipitate ammonium diuranate (ADU). Filter the ADU and calcine to UO_3 . This partially purified compound would then be dissolved in H_2SO_4 to yield recycled fuel solution.

(c) Evaporate the ADU filtrate to yield a discardable distillate and a concentrated residue. The latter would be recycled or converted to a solid and discarded.

The above is based on a Turret Reactor fuel recycle process which was developed at LASL to handle a uranyl nitrate solution containing less than 50 g U per liter.¹⁶ The applicability of the

Fig. 16. Processing irradiated fuel solution.

process to the KING Reactor fuel solution would have to be determined by chemical experiments on simulated fuel solutions containing the most important impurities in appropriate concentrations.

Equipment for the above processing would be piped so that spent fuel solution could be converted to solid form as shown in Fig. 15.

8. INSTRUMENTATION

8.1. Nuclear Instrumentation

The preliminary design of the KING reactor core region indicates that the nuclear instrumentation requirements are relatively straightforward.

Density changes from temperature and void formation create a large negative coefficient of reactivity that makes this reactor system essentially self stabilizing and thus eliminates or reduces the requirements for numerous mechanically driven regulating rods. A single annular shim will probably be used for shutdown and startup of the nuclear processes. This shim will isolate the nuclear chain reaction from the effects of fuel circulation.

Present plans indicate that the reactor nuclear instrumentation will consist of the following.

1. Neutron count rate channels (2) using either BF_3 neutron pulse counters or fission counters.

2. Linear power channels (3) using conventional instruments.

3. Log power channels (2) using conventional instruments.

4. Safety channels (4) using independent neutron flux monitoring channels.

Items 1, 2, and 3 are shown schematically in Fig. 17.

8.2. Process Instrumentation

The following assumptions, as to the philosophical approach to meeting the instrumentation needs for a 25 MW KING Reactor are explicitly stated below even though they may be obvious.

1. The instruments in the safety and control systems will be of high quality with sufficient redundancy in channels to ensure a high degree of reliability.

2. In most cases costs are given for new instrumentation packages. Where existing instrumentation may be utilized it is shown separately as a potential credit. This approach disregards the existing instrumentation. A brief review of the UHTREX instrumentation indicated that the costs in manpower to adopt UHTREX instrumentation to KING might be as high as replacement costs, at least for some subsystems. Also, many of the power supplies, recorders, and other instruments have already been removed from the UHTREX instrumentation systems.

3. The instrumentation cost estimate shown in Appendix A is tied to the present concept of the KING Reactor facility.

4. No attempt was made to design an instrument and control package; rather a list of requirements was made with an estimate of the instruments to meet the requirement.

Safety Systems

A list of possible safety systems is given in Table 1. The safety systems will be independent of the control instrumentation and the process computer. It is assumed that three independent channels of information are to be used for each system. In general, loss of flow in any loop would be detected by measuring the differential pressure across a fixed impedance, such as heat exchanger. Loss of fluid in a loop would result in a drop in absolute pressure. Rupture of the fuel loop could be detected by a rate of demand in the fuel pressure controller or a change in accumulator liquid level as well as by radiation detectors in the secondary containment or the cooling water systems. A hot spot detection system, consisting of thermocouples welded to fuel piping and components is indicated to be necessary from the difficulties experienced by HRT from corrosion enhanced by heating from precipitation of fuel. While the type of construction of the fuel loop is not known at this time, a flange leakage system similar to that used with HRE and HRT will be assumed to be required.

In addition, information from the following radiation monitoring systems should be available in the control room:

1. A health physics system.
2. An exhaust stack monitoring system.

Fig. 17. Nuclear instrumentation.

TABLE I
KING SAFETY SYSTEMS

<u>Safety System</u>	<u>Sensor/Comments</u>
neutron flux	ion chambers
period	calculated
fuel temperature high	RDT or thermocouple
loss of fuel flow	differential pressure across heat exchangers
loss of fuel	urge tank liquid level, fuel pressure
fuel pressure high	pressure transducer
loss of water coolant (all water systems)	pressure transducer
loss of water flow (all water systems)	differential pressure across heat exchangers
water contamination	scintillation crystals in all water loops
fuel storage temperature (and pressure)	thermocouple, pressure transducer
"hot spot" detection in fuel piping and components	surface mount thermocouples
flange leakage system	pressure switch
criticality outside reactor	neutron sensitive instruments in fuel processing cell, other locations as required
fuel leakage	radiation monitors

Also direct measuring differential pressure between fuel and water coolant loop.

3. A fuel leakage monitoring system.

4. Secondary containment area and ventilation monitoring system.

Control Systems

A list of possible control requirements is given in Table II. It is assumed that those variables required for closed-loop control will have three independent information channels. The assumption is made that a process computer will be available for control purposes and that the computer will be sized to handle the number of control functions listed in Table II.

Plant Component Monitoring

In addition to the instrumentation required for operational safety and control,

instrumentation should be provided to give information on the physical condition of certain system components and to indicate needed maintenance or incipient failure. Instrumentation in this category includes pressure drop measurements across filters, heat exchangers, vibration measurements near pump bearings, pump motor power requirements, etc.

Process Computer

The purpose of providing an on-line computer is to aid in the operation of the reactor, the fuel loop and coolant loops and the gas cleanup system consistent with the philosophy of utilizing the operator to supervise the operation rather than include him as an integral part of it. The computer is intended to monitor a large number of variables

TABLE II
KING PROCESS CONTROL REQUIREMENTS

<u>controlled variable</u>	<u>manipulated variable</u>	<u>comments</u>
flux	rod position	coarse control, startup and shutdown
flux	fuel flow rate	fine control with closed loop control; control valve in fuel loop. This controller may be rather complex since other variables affect the output.
fuel pressure	make up pressure valve position	closed-loop control
fuel temperature	fuel cooling water flow rate	closed-loop control
steam condenser helium-exit temperature	shell-side water flow rate	closed-loop control
fuel storage temperatures	water coolant flow rate	closed-loop control
reactor hardware coolant temperature	coolant flow rate	closed-loop control
secondary water temperature	cooling tower spray, air flow	manual control
secondary water distribution	valve positions	manual or automatic control
surge tank and demister fuel levels	pump or valves	closed-loop control

to ensure that they are within safe operating limits, to provide capability for control of reactor flux, power, fuel and coolant flow rates, fuel temperature and pressure, to provide logs of data and calculated variables in engineering units, and to monitor operational activities and changes in the status of system components. The computer should have the capability to initiate shutdown or other appropriate action upon implication from one of the safety systems, but at all times the operator would have manual controls, meters, recorders and other instruments to maintain the reactor at steady state or shut it down.

If the UHTREX on-line computer is still available when the KING components are being selected it could be utilized. The 8K core memory could restrict some control functions but otherwise it would be adequate. Due to the substantial reductions in costs of computer hardware, however, the

savings would not be greater than about a third of the estimated computer cost package, and could be much less due to software costs. In the (apparently unlikely) event that one or more of the original UHTREX process computer programmers would be available to adapt the UHTREX software the savings in cost that could be realized by using the UHTREX computer would be substantial. In the more likely case, that the programmers familiar with the UHTREX computer would not be available, the software development costs for a new KING computer, with vendor supplied support software and a larger core memory, may be much less than those for the existing computer.

The existing UHTREX computer uses an SDS 920 CPU with a 8192 word core, and word length of 24 bits. A replacement computer would probably have a smaller 16 bit word and a much larger core size, perhaps 24 K in central memory. The present

state of the art would indicate the use of a data disk with 10^5 - 10^6 word capacity instead of the 32 K drum used in the existing computer.

Use of UHTREX Instrumentation

The UHTREX building contains a sizable system of installed cables between the secondary containment and the control room. It is assumed that these cables will be utilized and that for most of the data channels the only wiring requirement is between the instrument sensor and a function box in the secondary containment and between a terminal strip in the control room and the appropriate scan, display or alarm device.

If chromel-alumel (Type K) thermocouples are selected for KING they would be compatible with the existing Type K cold junctions and wiring with a sizable savings in cost over alternate thermocouple types. It may also be possible to utilize the existing transducer signal conditioning after replacement of power supplies. This could result in a savings of up to \$200- 300 per pressure channel. Otherwise, for this preliminary cost estimate it seems reasonable to disregard the existing instrumentation.

An estimate of the overall sensor requirements shown in Table III is broken down into fuel and cooling water subsystems, shown in Fig. 18. Estimates of redundant channels required in the safety and control systems are also included. The information in Table IV is included as an aid in estimating control room annunciator panel, meter recorder and computer scan requirements.

The overall instrumentation cost estimate summary shown in Appendix A is based on the system drawing shown in Fig. 12 of Section 7.1 and the diluent gas system shown in Fig. 13 of Section 7.10. The computer cost is estimated to include a hardware package of \$120-150K. Using a multiplication factor of 2.7 times hardware costs to get total computer costs gives a range of \$325-400K. This is a "rule of thumb" estimate that has been used in several articles on process computer costs. Alternately, the software development of UHTREX required seven manyears,¹⁷ which, at \$50K per manyear is \$350K for software development. Assuming

that a similar effort is required for KING gives \$100-500K for the total cost figure of a process computer.

9. PRELIMINARY THOUGHTS ON THE HEALTH MONITORING SYSTEMS FOR THE KING REACTOR

9.1 Area Monitoring System

Gamma radiation levels will be measured by 20 area monitors located throughout the facility. Any person working in an area being monitored may read at a glance the current level in mR/h, may see that the monitor is functioning properly, and will be warned by audible and visible alarm if the high level setpoint is exceeded. The same information will be indicated on a central panel in the control room.

Table V lists the 20 channels of area monitoring, the monitor locations, and their ranges. Figures 19, 20, and 21 show health monitoring equipment locations on plan views of each level of the facility.

Four of the 20 area monitoring detectors will be located inside the secondary containment, to serve as health monitors during shutdown periods and as operational monitors during reactor operation. Three detectors will be located outside the building: in the heat dump area, at the stack, and at the wall of the secondary containment. The high level alarms of four of the detector channels will be connected by a "three out of four" coincidence circuit to the emergency alarm system. These four detectors will be widely separated, in peripheral locations outside the secondary containment. An alarm condition on three of them would indicate a major release of fission products to the containment. All four of these channels will be equipped with built-in auxiliary power supplies to ensure continued operation despite loss of all facility power.

The detectors in the area monitoring system may be Nuclear Measurements Corporation Model GA-2T, all-transistorized, gamma-scintillation detectors, using NaI crystals (low range) and activated plastic crystals (high range). Scintillation detectors of this type show good stability and sensitivity and are not susceptible to jamming in high radiation fields. Each detector will have two levels of alarm: (1) a low level, fail-safe alarm

TABLE III
BASIC PROCESS INSTRUMENTATION - KING 25 MW

plant sub-system	Temperature		Pressure		Differential Pressure		fluid ⁽¹⁾ radiation sensors	pump ⁽²⁾	other requirements
	basic no. reqd	with redundancy	basic no. reqd	with redundancy	basic no. reqd	with redundancy			
fuel sub-systems									
main fuel loop	6 ⁺	12-16 ⁺⁽³⁾	9 ⁺	~ 20 ⁺	4	10	--	3	uranium concentration pH
fuel storage	2	4-6	3	6	2	4	--		
radiolytic gas and bypass	-	15-20	-	15-20	-	~ 6		3,4	oxygen, entrained fuel in gas ⁽⁵⁾
coolant sub-systems									water pH and/or electrical resistivity all sub-systems ⁽⁶⁾
fuel coolant water	5	10	2	4	5 ⁽⁴⁾	10	2-4	2	
steam condenser coolant	4	6-8	2	4	3	5-7	2	2	
reactor hardware coolant	4	6-8	2	4	3	5-7	2	2	
pool ⁽⁶⁾	-	~10	(1-2)	(2)	(2)	(2)	1-2	1,2	
fuel storage ⁽⁷⁾	3	~ 6	3	~ 6	1	2 ⁺	2	(1)	
secondary water and cooling tower	5-8	12-16	2	4	7	~14	2 ⁺	2	
"hot-spot" detection ⁽⁸⁾	-	~50	-	-	-	-	-	-	
flange leakage detection ⁽⁸⁾	-	-	-	~ 30	-	-	-	-	
totals		~150		~100		~60	~14	18	

Notes -

- (1) Scintillation crystals or ion chambers to measure activity in fluid.
- (2) Each pump will be directly instrumented for speed (rpm), power drawn, bearing vibration.
- (3) Both t/c and RTD's required, high quality sensors with probable high cost to install.
- (4) Includes pressure differential between fuel and water as a direct measurement.
- (5) If required in gas exit from denister.
- (6) It is assumed that a heat exchanger, pump, etc. will be added to cool the pool water.
- (7) It is assumed that a closed cooling loop will be provided.
- (8) The numbers given here are only guesses.

Fig. 18. KING reactor fuel and cooling water systems.

TABLE IV
ALARM, DISPLAY, COMPUTER SCAN ESTIMATES

1. Alarm	Flux level	
visual panels	fuel temperature, pressure, loss of flow	
audio alarm	primary water coolant temperature, loss of flow	
	other coolant water loop temperatures (4),	
	loss of flow (4)	
	radiation contamination in water (4)	
	hot spot or flange leakage indication	
	fuel spill, fuel processing criticality,	
	experimental area accident	
2. Display		
meters -	fuel temperature (2), pump RPM and/or flow rate,	
main console	fuel pressure (power?), coolant pressures (55),	
	coolant temperatures (~ 10)	
	pump RPM or flow rate (5)	
	flux level (3), control rod position (2), period,	
	control valve positions.	
Graphic Panel -	valve positions, pump power, M.S. bad heaters, etc.	
Recorders -	eight channel strip chart,	
Radiolytic Gas		
special		
and controls		
3. Computer Scan	Analog Channels	Digital Channels
Temperature	~ 150	
Pressure transducers	~ 10	
switches		30 ⁺
Differential pressure		
transducers	~ 60	
Pump speed	~ 20	
current	~ 20	
voltage	~ 20	
Position	10	~ 200
Nuclear flux	10	
Coolant radiation		
contamination	- 15	
Liquid level	10	
Vibration, stress (?)		
Other	_____	_____
Est. requirements	~ 400	~ 400

that will indicate failure of the unit, and (2) a high level radiation alarm. The level of both alarms will be set with an electronic control circuit that will include a light source. The light source will allow a check of detector and alarm circuitry without a radioactive check source.

Power will be supplied from the diesel generator-backed power supply. Each channel will emit a 0- to 50-mV dc signal to either the on-line computer for periodic logging of levels or a paper recorder.

TABLE V

AREA MONITORING SYSTEM

<u>Channel Identification</u>	<u>Area Monitor Location (Room No.)</u>	<u>Range (mR/h)</u>
H-4001	Ventilation equipment room (101)	0.1 - 10 ²
H-4002	Electrical equipment room (103)	0.1 - 10 ²
H-4003	Exhaust filter room (105)	0.1 - 10 ²
H-4004	Exhaust filter room (105)	0.1 - 10 ²
H-4005	Heat dump area (outdoors)	0.1 - 10 ²
H-4006	Stack survey (outdoors)	10 - 10 ⁴
H-4007	Inside secondary containment (106)	10 - 10 ⁶
H-4008	Inside secondary containment (106)	10 - 10 ⁶
H-4009	Control room (202)	0.1 - 10 ²
H-4010	Cell operating area (211)	0.1 - 10 ²
H-4011	Cell operating area (211)	0.1 - 10 ²
H-4012	Fuel transfer lock (212)	10 ² - 10 ⁷
H-4013	Fuel transfer cell (213)	10 ² - 10 ⁷
H-4014	Remote maintenance corridor (215)	0.1 - 10 ²
H-4015	Electrical equipment corridor (301)	0.1 - 10 ²
H-4016	Mechanical equipment room (305)	0.1 - 10 ²
H-4017	Maintenance corridor (306)	0.1 - 10 ²
H-4018	Gas cleanup piping room (307)	10 - 10 ⁴
H-4019	Subbasement (401)	10 - 10 ⁴
H-4020	East wall (outdoors)	10 ² - 10 ⁷

Ground level health monitor locations

**Legend: A, gamma area; P, air particle; S, stack
D, portal (door); H, hand and foot**

Fig. 19. Health-monitor locations - ground level.

Operations level health monitor locations

Fig. 20. Health-monitor locations - operating level.

Basement levels health monitor locations

Fig. 21. Health-monitor locations - basement level.

9.2. Air Particle Monitoring System

Three fixed-filter, air particle monitors will continuously sample for radioactive particles in the air of the cell operating area, the exhaust filter room, and the mechanical equipment room. The monitors will be Eberline Instrument Corp. Model AIM 3BL which use β - γ end window GM detectors. The air particle monitors will draw a 10-ft³/min sample and will indicate a rise ≥ 40 c/min above background in an atmosphere of 10^{-9} $\mu\text{Ci}/\text{cm}^3$, with a collection time of 1 min. The monitors will be large stand-mounted units, with shielding around the detector for background rejection. Each air particle monitor will be equipped with a recorder, a local meter, and local failure and high level alarms. Alarm signals and a 0- to 50-mV dc signal, for computer logging, will be transmitted to the control room.

A fourth particle monitor will be intended for intermittent use in the control room. The unit, an Eberline Instrument Corp. Model AIM3, will use a scintillation detector, and draws a 2-ft³/min sample. Alarm and computer logging signals will be identical to those of the other three air particle monitors.

9.3. Stack Monitoring System

In the stack monitoring system, a sidestream, isokinetic sampling loop will deliver stack effluent to a Tracerlab MAP-1B/MGP-1A combination air particulate and radiogas monitor, which uses all solid-state circuitry, except for the photomultiplier tubes in the detectors. The system is shown in Fig. 22.

An isokinetic sampling head, to be located inside the stack, will draw a 10-ft³/min sample at a point 44 ft above the base of the stack. The sample will move through a stainless steel pipe, with a minimum number of bends, to the stack monitor, to be located in a small room near the stack base. Just inside the room will be a tap where grab samples can be taken.

First in the line of flow will be the air particulate monitor in which a shielded, plastic phosphor, beta-scintillation detector will scan a moving filter tape. The detector and an integral, solid-state preamplifier will be contained in an

aluminum housing sealed with an O-ring. A log rate meter, to be mounted in a nearby chassis, will indicate the detector output over a 10^{-} to 10^6 cpm range and will have single-threshold discrimination. Outputs from the meter will be high level alarm signals to local and control room warning devices and two data signals: a 0- to 10-mV output to a nearby strip chart recorder and a 0- to 50-mV output to the on-line computer for logging purposes.

From the air particle monitor, the sample stream will move to a charcoal cartridge where iodine will be adsorbed. The stack monitoring system cannot detect gaseous radioiodine at very low levels in the sample stream. Therefore the cartridge will be a removable iodine concentrator that can be examined periodically in a more sensitive multi-channel analyzer to be located elsewhere in the facility.

Past the charcoal cartridge, the sample stream will enter the radiogas monitor. There a Tl-activated NaI, scintillation crystal, photomultiplier detector will extend through a wall recess into a shielded gas sampler. The log rate meter to be connected to the detector will have single channel analyzer capability derived from spectrometer input circuitry and adjustable gain, window, and threshold. There will be local and control room high level alarms. A 0- to 50-mV dc analog signal to the on-line computer will be recorded periodically as a data log entry. The monitor output will be also recorded on a strip chart recorder mounted at the stack monitor. In coordination with the continuous radiogas monitor analysis, grab samples will be taken throughout the reactor lifetime for gamma counting of gross activity and for the determination of the contributions of specific isotopes to the gross activity. The grab samples will be run on an on-site multi-channel analyzer.

Beyond the radiogas analyzer, a sealed, sliding-vane, positive displacement pump will return the sample to the stack. The pump and an associated control valve will produce a maximum flow of 10 ft³/min, regulated to $\pm 5\%$. Flow, to be measured by an in-line meter, will be indicated on the local control panel and monitored by a high-low flow alarm.

Fig. 22. Stack-monitoring system.

Detection Capabilities

Maximum permissible concentrations (MPC's) for many isotopes in unrestricted areas are so low that the count rates produced are not detectable, above background, by most commercially-available stack monitors. Therefore, the design of the stack monitor takes advantage of an atmospheric dilution factor which may be applied to MPC's to arrive at average tolerable levels in the stack. These "stack MPC's" generally fall within the range of detection of the stack monitor. This technique which was used for UHTREX is probably not acceptable for KING. More sensitive detectors may be required.

Early in the operation of the reactor, the stack monitor will be used in the gross counting mode with maximum sensitivity and low alarm setpoint. When the isotopes or groups of isotopes normally released to the stack are identified by gamma-spectrum analysis of grab samples, optimum discriminator and spectrometer window settings will be determined, and the monitors will be set for minimum background and more effective detection of isotopes of particular interest.

The count rate signals from the air particle channel and radiogas channel will be recorded on a two-pen recorder built into the stack monitor. One detector of the area monitoring system will be mounted on the stack to transmit a gross indication in the event that a major accident drives the stack monitor off-scale. The range of the detector will be 0.01 to 10 R/h.

10. PRELIMINARY THOUGHTS ON HEALTH PHYSICS FOR THE KING REACTOR

10.1 Los Alamos Scientific Laboratory Radiation Safety Policies

The duties and responsibilities of individuals and groups who work with radioactive materials are defined by Laboratory policies described below.

Individual - Individuals at the Los Alamos Scientific Laboratory who have any contact with radioactive materials or radiation are responsible for:

1. Keeping their own exposures to radiation and also those of others as low as possible and, specifically, below the maximum permissible levels. Exceptions are exposures received in

medical examinations and treatments and, with prior approval by the Health Division, cases where urgent situations require exposures in excess of the recommended levels.

2. Wearing the prescribed monitoring equipment (pocket dosimeters, film badges) in radiation areas. Requesting that neutron films be read when exposure to 0.1 rem is suspected.

3. Surveying hands, shoes, and body for radioactivity and removing loose contamination to the tolerance levels, in accordance with the recommended decontamination procedures, before leaving a contaminated area.

4. Wearing appropriate protective clothing whenever clothing contamination is possible, and not wearing such clothing outside the hot laboratory areas. Using gloves, hoods, and respiratory protection (respirators or supplied-air masks, as recommended by the Health Division) when necessary. Using proper techniques and facilities in operations involving radioactive materials.

5. Observing the recommended procedures in regard to eating and smoking in contaminated areas.

6. Reporting injuries and ingestion or inhalation accidents promptly to the Health Division and carrying out the recommended corrective measures. Cooperating in any and all attempts to evaluate exposures, particularly by promptly returning requested urine specimens.

7. Carrying out recommendations of the Health Division in roping off hazardous areas, posting warning signs, and otherwise controlling special hazards for which they are responsible. Contacting the Health Division (Group H-1) about recommended procedures for high level radiation areas or when conditions are altered significantly.

8. Cleaning up contamination for which they are responsible. Arranging for a survey by Group H-1 and carrying out the decontamination recommended for the area when changing locations or ending an operation.

9. Properly storing and labeling radioactive materials for which they are responsible.

10. Properly packing, in sealed containers whenever possible, and labeling with origin and date, all contaminated waste materials.

Supervisors - Supervisors are responsible for ensuring that the above individual responsibilities are discharged by those under their control, and are further responsible for:

1. Instructing those employees for whom they are responsible in the use of safe techniques and in the application of approved radiation safety practices.

2. Providing such staff as is required for packaging contaminated waste and for decontaminating anything for which they are responsible.

3. Furnishing information to the Health Division concerning individuals and activities in their areas, particularly additions to or deletions from their personnel rosters.

4. Contacting the Health Division whenever they anticipate major changes in operational procedures, new techniques, alterations in physical plant, or new operations that might lead to personnel exposures.

5. Preparing written Standard Operating Procedures for all operations involving hazardous amounts of radioactive materials.

Monitoring Group, H-1 - This group is responsible for:

1. Furnishing consulting services on all aspects of radiation protection.

2. General surveillance of all radiation activities, including assistance to individuals and supervisors in discharging their responsibilities.

3. Distribution and processing of personnel monitoring equipment, including keeping of personnel exposure records, notifying individuals and their supervisors of exposures greater than the permissible levels for their situation, and recommending appropriate restrictions.

4. Supervision and coordination of the waste disposal program, including keeping of waste storage records.

5. A continuous program of environmental hazard evaluation and hazard elimination.

6. Storage, leak testing, issue, and disposal of radioactive sources; supervising the shipping of radioactive material. Further, the inspection of requests for sources to assure standardization of capsules for general utility.

7. Scheduling urinalysis measurements when required.

8. Decontaminating large pieces of equipment (e.g., motor vehicles) and assisting personnel decontamination in cases of contaminated wounds or persistent personal contamination.

10.2. Facility Layout

Control Areas - A control area is considered to be any area that could become contaminated. The control areas will include on the ground level: the exhaust filter room; on the operating level: the hot toilet, the cell operating area, the two cells, the hall and adjacent remote maintenance corridor, the loading dock, and the east stairwell; on the basement level: the mechanical equipment room and the maintenance corridor; and all four levels of the secondary containment structure. All accessible areas are to be routinely monitored and kept clean.

The only access between cold and control areas will be the door to the change room near the Health Physics office. Anyone entering the control area will be required to wear protective clothing consisting of at least a smock and shoe covers.

Change Room - Personnel must pass through a locker room to enter the control areas. Here the person entering makes a complete change of clothing or merely puts on a smock and shoe covers, depending upon conditions or the work to be done in the control area.

A portal monitor, located between the change room and the hot toilet, detects any contamination before it is carried into the change room. Another portal monitor is located at the main entrance of the facility.

Decontamination Provisions - Adjoining the change room is a hot toilet which includes showers. Decontamination of personnel is possible here. Tools or other materials which cannot be decontaminated in place are sent to a LASL facility which specializes in decontamination.

10.3. Personnel Monitoring

Film Badges - Film badges will be issued on a monthly basis to all personnel routinely located at the facility. Badges will be issued to visitors, when Health Physics personnel consider it necessary, before visitors are allowed to enter the control area.

Badges are constructed of plastic and have an open window and two filter areas. The areas under these two filters are used to estimate the thermal neutron exposures as well as the gamma exposure; the open window area is used to estimate beta and soft gamma exposure. Each badge contains three films (low range gamma, high range gamma, and fast neutron) and a silver phosphate glass rod for measuring high level exposure. In certain areas where the probability of physical contamination is high, an unsealed, removable plastic cover is placed on the film badge to prevent its contamination.

Dosimeters - Self-reading pocket dosimeters may be issued to personnel working in a radiation field where it might be possible to receive as much as 100 mR within a week. Pocket dosimeters will be read at least daily, more often if the radiation levels warrant it.

Reporting - A copy of each Group's monthly exposure report is sent to the Group Leader's and Division Leader's offices at the end of each month. Also, quarterly cumulative reports are sent to the Group's office. An annual exposure is calculated for each individual and is added to the previous accumulated exposure total recorded in his medical folder.

Form letters are sent to an individual's supervisor when the individual's monthly exposure record indicates that, should he continue to be exposed at the same rate for the remainder of the year, he would receive more than 5 rem, the present LASL administrative yearly acceptable exposure.

Should an individual too closely approach a cumulative exposure of 3 rem in any one calendar quarter, his area Health Physicist is notified. An informal investigation, usually by phone call to the man and his supervisor, is then initiated by the Health Physicist.

Should an individual actually receive an exposure greater than 3 rem in any one calendar quarter, a formal investigation is conducted by the area Health Physicist. Within 72 h of the time that H-1 learns of the exposure, the Health Physicist must submit a report of his findings and recommendations to the H-1 Group Office. The H-1 Group Leader, in turn, must submit the report to the AEC, Division of Operational Safety, in Washington. A copy of the report is sent to the Group Office or supervisor of the individual involved.

10.4. Protective Equipment

Face Masks - Before the KING Reactor reaches initial criticality, all operating personnel will be fitted for full-face respirators by the Industrial Hygiene Group, H-5. The respirators are equipped with combination canisters that will protect against organic vapors, acid gases, ammonia, and hydrocyanic acid gases; radioiodine, and radioactive aerosols; and other toxic particulates. At the same time these respirators are fitted, the personnel are instructed in the use of the respirators and tested in a chamber for proper respirator fit. All respirators are inspected at least once a year. If they are used, they are sent to a group which decontaminates or renews the respirators before they are returned.

Air Packs - Self-contained breathing apparatus is placed in cabinets at emergency stations throughout the facility. Periodic personnel training and apparatus maintenance is provided by Group H-5.

Protective Clothing - Protective clothing, consisting of smocks, coveralls, shoe covers, head covers, and surgical gloves is available in the change room at all times.

10.5. Monitoring Equipment

Portable Survey Equipment - The Laboratory has a large supply of survey instruments of all types readily available at all times. The instruments kept at the KING facility are listed in Table VI.

These instruments are routinely sent out every 6 months for maintenance and recalibration unless repair is necessary before this time. Batteries are replaced by the health monitor when necessary,

TABLE VI
PORTABLE SURVEY INSTRUMENTS

Type	No.	Range	Scales
Geiger-Müller	3	0-20 mR/h	3
Ion chamber	3	0.1 mR/h - 10^3 R/h	3
Alpha	2	0 - 10^5 cpm	3
Neutron	1	0 - 5×10^5 events/min	4
Neutron	1	0 - 250 m rem/h	4

and proper operation is routinely checked with either an internal or external source.

Laboratory Equipment - The facility will be equipped with a gas-flow proportional counter plus a scaler and associated electronics. This unit can be used on either the alpha or the beta plateaus for counting swipes, air samples, etc.

A multi-channel analyzer equipped with a NaI crystal will be available to aid in identifying and measuring gamma-emitting isotopes in samples of any kind.

Portal Monitors - Two portal monitors will be installed in the building; one at the main entrance to the building, and the other in the door between the hot toilet and the change room. Anyone passing from a control to a cold area must pass through the portal monitor. The sensitive unit located at the main entrance can be set to detect small amounts of beta-gamma contamination.

Hand and Foot Monitor - A hand and foot monitor will be located inside the hot toilet room. The monitor will detect any hand or foot contamination and will have a frisker for the rest of the body. The alarm levels on this unit will be set very low in order to keep any contamination from getting into the change room.

Floor Monitors - A beta-gamma floor monitor will be normally used daily in the control areas and only weekly in the cold areas unless a significant amount of contamination is found in the control area.

An alpha floor monitor will be used during periods when fresh fuel is loaded or at any other time when there is a possibility of alpha contamination.

10.6. Routine Surveys

Smear Surveys - Surface contamination will be surveyed by using an appropriate instrument. If a contaminated area is found, swipes will be taken to aid in localizing contamination and removing it.

Material to be moved from the site will be surveyed by instrument and smear. The material will then be tagged with activity information by the monitor who made the survey. A laboratory counter will be available at the site for alpha and beta counting of any smear tests that are made.

Air Sampling - In addition to the four air particle monitors described above, the KING facility will have a vacuum system with 20 outlets to which air sampling heads may be attached. Each outlet will have its own valve which may be used to adjust the entering air flow or to shut it off completely. The normal procedure will be to use a 2-1/8-in. HV-70 filter paper on as many units as is found desirable. The units will sample the air at 2 cfm during the time the area is occupied, and the filters will be counted the next day with automatic counting equipment for both alpha and beta activity.

Daily reports will be issued to the Health Physicist for the facility. If any air concentrations are found to be over the LASL action levels (1×10^{-10} $\mu\text{Ci}/\text{cm}^3$ for ^{235}U and 3×10^{-9} $\mu\text{Ci}/\text{cm}^3$ for fission products), an investigation will be made to discover the reason for the high air counts, and a recommendation will follow on how to prevent a recurrence.

10.7. Bioassay

At LASL, urine samples are collected routinely from personnel working with materials considered hazardous. In the event of an incident, accident, or emergency, further urine sampling is carried on to furnish detailed information about actual exposure. If the need exists for rapid estimation of exposure, analytical procedures may be abridged, shortening the time required for analysis, with a consequent loss of sensitivity and specificity.

Two whole-body counters are available at all times to determine the amount of, and to identify, any gamma-emitting nuclides that may have been taken internally.

10.8. Medical Examinations

A complete physical examination is made of every person before he is hired. At three-year intervals thereafter everyone is reexamined. For those who work with radioactive materials, blood samples are taken every 6 months, particularly for a white cell count, and other routine blood tests are also performed.

11. SAFETY ANALYSIS

The inherent self-controlling features of solution type reactors are well known. Experiments on the Los Alamos Water Boiler reactor¹⁸ and the Kinetic Water Boiler Experiments KEWB¹⁹ first demonstrated the remarkable stability which is provided by the large negative temperature coefficient and gas bubble formation. These experiments showed that automatic shutdown occurs under all abnormal operating conditions by these inherent physical properties of aqueous uranyl fuel solutions.

A safety analysis report has recently been prepared prior to the operation of the Kinglet experiments.³ These experiments are designed to test out all nuclear properties of the rapidly moving liquid fuel core of the KING Reactor concept. Status of these experiments is discussed briefly in the next section.

This safety analysis report confirms that the inherent safety and stability of the aqueous liquid fuel reactors is retained in the KING reactor mode of operation.

The Kinglet experiments are expected to supply all the necessary information to check the theoretical analyses which have been performed on the KING reactor.

12. STATUS OF RELATED RESEARCH AND DEVELOPMENT

A new feature in reactor operation is encountered in the KING reactor concept due to the high velocity at which the liquid fuel moves through the small active core region.

It seemed desirable therefore to initiate two types of experiments to proof test the concept by answering the following questions.

1. What are the nuclear stability characteristics in a dynamic liquid core which is controlled by reflector neutrons rather than the normal delay neutrons and which at the same time produces appreciable voids from radiolytic gas production?

2. Are there erosion effects which can influence the very low corrosion expected from the low operating fuel temperatures?

Experiments to answer these questions are well under way and are briefly described below.

Additional experiments are planned, as soon as funds become available, to assist in the design of the fuel and gas handling equipment for a 25 MW research reactor. Work is required to determine which of several available liquid gas separators is best suited to remove the radiolytic gas from the fuel solution during operation. The seal design of existing potential fuel pumps should be investigated to assure zero leakage of radiolytic gas. Sealed rotor type pumps are not expected to be suitable due to the high radiation field which will exist in the neighborhood of the fuel.

12.1 The Kinglet Experiments

The unusual features of core operation required in the KING Reactor concept made it evident that credence in such a reactor could only be achieved by an experimental demonstration.

No one is willing to commit funds for a full scale reactor project when no experimental proof of principle exists particularly when the core design requires the fuel to move at high velocity and the delay fission neutrons normally considered to be so important for reactor control play no role.

On the other hand to proof test the nuclear characteristics of a high flux reactor without a full scale project does not seem very feasible at first sight.

Fortunately the small core size required by the KING reactor combined with a novel mode of operation has permitted the construction of the Kinglet assembly which will proof test the concept under realistic full scale conditions.

This device has been described some time ago.¹ Static nuclear experiments have been performed on

Kinglet as well as dynamic hydraulic tests using water. The results of this work have been documented in the Kinglet Safety Analysis report.³

Dynamic critical experiments which will serve as the proof test for the KING reactor concept have been delayed since September, 1971. More stringent AEC regulations requiring a full scale safety analysis report and startup approval have delayed the startup of the dynamic test. Approval is not expected before March.

12.2 Status of Corrosion Testing for the KING Reactor

Introduction

The KING Reactor will use an aqueous fuel of approximately 0.9 M uranyl sulfate (200 g ²³⁵U per liter) in 0.5 M sulfuric acid. The virgin fuel solution will also be 0.05 M in ferric sulfate and 0.01 M in cupric sulfate. Corrosion studies have been reported by ORNL on uranyl sulfate fuel solutions for the Homogeneous Reactor Test (HRT) at high temperature and pressure (275-300°C and up to 1000 psi).²⁰ Although the KING Reactor will operate at much lower temperatures and pressures than used in the HRT, the KING Reactor fuel has considerably higher uranium and acid concentrations. Therefore it was deemed advisable to investigate the corrosion properties of the KING reactor fuel.

In the KING reactor the fuel will be pumped through a core reaction zone at a fairly high velocity of the order of 50-100 ft/sec. Thus it was necessary to consider not only the corrosion of materials of construction by the fuel but also erosion by the rapidly flowing fuel. It was decided to study the corrosion and erosion effects by constructing several test loops in which simulated fuel solution would be pumped past specimens of various metals and alloys at the desired velocities and at the temperatures to be encountered in the reactor.

12.2.1. Fuel Solutions for Corrosion Testing

Corrosion studies will be made with both a virgin fuel solution and with a fuel simulated to have undergone fission in the reactor for 1 year corresponding to a burnup of about 2.3%.

The virgin fuel has been prepared using depleted uranium and has the following composition: U-217 g/liter (0.914 M); Fe-5.63 g/liter (0.1008 M); Cu-0.66 g/liter (0.0104 M); SO₄-153.5 g/liter (1.60 M). Determinations of density and viscosity made on an almost identical solution gave the following results:

Temperature °C	Viscosity cp	Density g/cm ³
30	1.80	1.316
50	1.19	1.304
80	0.74	1.284

The simulated fuel will be doped with various elements corresponding to fission products and daughters. A computer program has been developed for this fuel and reactor operation by P. Iwanchuk, C-4, from equations derived by B. Morrison²¹ for the calculation of the concentration of fission products as a function of reactor running time. These calculations were done assuming a core volume of 10 liters, a total fuel solution volume of 2000 liters, power of 25 MW, and a maximum operating temperature of 80°C. Under these assumptions the average thermal neutron flux in the fuel in the core would be 2.6×10^{14} neutrons/cm²/sec and the fission rate would be 7.75×10^{13} fissions/sec/cm³.

Those fission products which would be present in the fuel solution after 1 year of irradiation in concentrations less than 50 mg/liter were arbitrarily omitted from the first simulated fission product ("fissium") mixture. The elements Br, Kr, I and Xe were eliminated from consideration because they would be expelled from the solution as gases. Since it was desired to study the corrosion initially with a solution rather than a slurry, Ba and Sr were eliminated since the solubility products of the sulfate of these elements would be exceeded in less than 12 h running time. Therefore the following elements in the amounts shown will be used to prepare a "fissium" mixture for addition to the fuel solution:

<u>Element</u>	<u>Concentration g/liter</u>
Rb	0.062
Y	0.094
Zr	0.593
Mo	0.440
Tc	0.112
Ru	0.241
Te	0.065
Cs	0.330
La	0.090*
Ce	0.475
Pr	0.151
Nd	0.378
Total	3.051

*Actually 0.181 g La/liter will be formed but the smaller amount will be used to prevent precipitation of the sulfate.

12.2.2 Materials

A literature survey has indicated that tantalum, zirconium, titanium, Carpenter No. 20 Cb-3 stainless steel and type 316 stainless steel would probably be the best materials of construction for the reactor from the standpoint of resistance to corrosion. Tantalum has been found to be completely resistant to 75% H_2SO_4 solutions at temperatures up to at least 190°C.²² Zirconium has been reported to be completely resistant at 100°C in H_2SO_4 of up to 75% concentration.²³ The reported corrosion data on titanium indicate that it should be tested under actual service conditions. Resistance of Ti to H_2SO_4 has been found to be good below about 5% acid concentration at room temperature; higher concentrations of acid produce very heavy attack.^{24,25} However, tests by a large chemical concern with process equipment involving H_2SO_4 up to 22% at high temperatures and pressures showed Ti unaffected whereas the best of alloy steels was chewed up in a few hours.²⁴ Schlain²⁵ reports that type 316 stainless steel corrodes at a rate of 0.04-0.97 mils per year (mpy) in air-aerated 5% H_2SO_4 at 35°C, and that Carpenter 20 stainless steel corrodes at a rate of 0.08 mpy (air) and 0.91 mpy (N_2) in 10% H_2SO_4 at 35°C. Whereas titanium is reported²⁵ as corroding at rates of 190 mpy and 811 mpy in 5% air-aerated H_2SO_4 at 60°C and 100°C,

respectively, Carpenter 20 steel is attacked at a rate of only 0.68 mpy at 60°C in 10% H_2SO_4 in air.

Based upon the results of the literature survey, 9 metals have been selected for corrosion testing: types 304, 316, 321, 347 and Carpenter 20 Cb-3 stainless steels, tantalum, titanium, zirconium and zircaloy-2. Although the nuclear properties of tantalum prohibit its use in appreciable quantities in the core region of a thermal reactor, thin films could well be used to improve the corrosion resistance of zirconium if found desirable.

12.2.3. Corrosion Test Loops

The materials are to be tested as tubes 0.437 in. i.d. by 0.437 in. long. The specimens have been polished by machining, degreased with trichloroethylene and will be separated from each other and from the specimen holder with Teflon as shown in the photograph Fig. 23, and drawing Fig. 24.

The design of the original corrosion test loop is shown in a line drawing in Fig. 25. When flow tests were made with water in this test loop, it was found that the flowmeter should be placed at the top of the right leg to eliminate two elbows, in order to decrease the flow impedance enough to obtain desired flow rates in the test section. A second loop was constructed with this modification. The relief valve line was then removed as unnecessary because of the inherently limited output pressure of the pump. The second loop as now in use is shown in Fig. 26. Tests with the second loop have shown that the by-pass is not necessary for controlling flow rates in the test section. The first loop is being modified to incorporate all of the above changes.

Both of the test loops, exclusive of the pumps, have been constructed entirely of type 316 stainless steel. The specimen holder is also of 316 stainless steel. The piping is 2 in. tubing and is flared down to 0.5 in. internally at the entrance and exit of the specimen holder. The specimen holder can hold specimens, of various shapes such as tubes, plates, or pins to allow testing under different flow conditions.

Fig.23 . Specimen holder with corrosion test specimens.

Fig. 24 . Specimen holder showing corrosion test specimens in place with Teflon separators.

The pumps are 5 hp Model GB Chempumps with 6.5 in. impellers, constructed of Carpenter 20 stainless steel, and are rated at 40 gpm water at 150 ft head. A typical cross section is shown in Fig.27 . The pumps are fitted with heat exchangers. These pumps were supposed to have 15 mil thick tantalum liners on the stator and rotor in order to avoid any possibility of damage to

Fig. 26 . Original test loop. Modification 1.

these parts by corrosion. However, the manufacturer was unable to fabricate tantalum liners and supplied the pumps with Carpenter 20 stainless steel liners.

The 6.5 in. impeller in the pump will rotate at 3450 rpm. Thus the impeller tip will have a velocity of 98 ft/sec and will be subject to the most severe conditions of corrosion and erosion in system. The impeller, bearings and the rest of the pump will be inspected after each run. The system was originally designed and has been constructed to operate at or just slightly above atmospheric pressure. It is possible that the KING Reactor may be operated at core inlet pressures up to 160 psi. The test loops can be modified with not too much difficulty to operate under the higher pressure conditions.

12.2.4 Corrosion Testing

The second loop was first tested using water. Velocities of about 75 ft/sec were obtained with the water at 80°C. This temperature was easily maintained for 8 h. The system was then drained and loaded with 25 liters of the virgin fuel solution containing depleted uranium. A solution temperature of 80°C was reached within 1/2 h of start-up and the velocity in the test section was about 70 ft/sec. On December 30, the system had been operated with fuel at 80°C and about 70 ft/sec for 3 days of a planned initial 12-day run.

Fig. 25. Original corrosion test loop design.

Fig. 27. Typical cross-section of Model GB Chempump.

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APPENDIX A - PRELIMINARY COST ESTIMATE

A preliminary cost estimate has been made for the construction of a 25 MW KING reactor at the existing UHTREX site. The estimate includes removal costs of existing UHTREX reactor equipment not needed for the new reactor system.

The cost estimate has been made to meet the requirements of the 1971 Edition of the ASME Boiler and Pressure Vessel Code Section III for Nuclear Power Plant components. This new code has increased the cost of nuclear components by a factor of about 3 to 5. All fuel and fission gas handling components for the KING reactor have been assumed to fall under this new code.

1. Total construction costs	\$ 10,700,000
2. Engineering design and inspection @ 20% of construction costs	2,140,000
3. Standard equipment including Laboratory furniture	15,000
4. Removal costs less salvage including removal of existing gas cleanup system	<u>325,000</u>
Subtotal	\$ 13,180,000
5. Escalation @ 28% of above costs	<u>3,690,000</u>
	16,870,000
6. Contingency @ 25% of above costs, including escalation	<u>4,130,000</u>
Total Project Cost Estimate	\$ 21,000,000

1. Construction Costs

A. Improvements to land including grading, drainage diversion and paving \$ 20,000

B. Building

1) Experimental building addition. Approximately 5800 sq ft @ \$65.00	380,000
2) Modifications to the existing building to accommodate new construction	
a. Installation of containment and ventilation system	\$50,000
b. Hot cell ventilation modifications	30,000
c. Two motor control centers	20,000
d. Emergency power supply	50,000
e. Uninterruptable power supply	80,000
f. Service Extensions to new building	<u>20,000</u>
	250,000

C. Special facilities

1) Regular and heavy concrete reactor core shielding	70,000
2) Stainless steel water tank reactor cover and experimental parts	160,000
3) Graphite thermal column	290,000
4) Heat exchangers	500,000
5) Pumps and valves	1,000,000
6) Reactor core components	
a. Be reflector	440,000
b. Be machining	75,000
c. Hardware	250,000
d. Pressure vessel	100,000
e. Remote handling equipment	150,000
f. Testing, assembly and installation	<u>185,000</u>
	1,200,000

7) Auxiliary reactor components

a. 100 ft of 5-in. Carpenter 20cb-3 with necessary fillings	20,000
b. Expansion tank with water make up	35,000
c. Four liquid gas separators	25,000
d. Fuel feed and extractor system	50,000
e. Two disconnect devices	140,000
f. Two fuel storage and dump tanks 1200 ft of 5 in. & 2000 ft of 3/4 in.	180,000
g. Fuel processing gas purge loop	<u>175,000</u>
	625,000

8) Gas recombiner and purge system

a. Demister	10,000
b. Recombiner	190,000
c. Condenser	190,000
d. Main blower	30,000
e. Piping	7,000
f. Auxiliary blower	20,000
g. Valves	73,000
h. Installation	<u>30,000</u>
	550,000

[illegible]

14) Cooling tower loop		
a. Cooling tower, 30 MW	56,000	
b. Pumps, 2 main, 1-standby	17,000	
c. Pipe, 360 ft-16 in. sch. 40 carbon steel	7,200	
d. Pipe, 60 ft-6 in. sch. 40 carbon steel	300	
e. Water treatment system	2,900	
f. Valves, 2-12 in. pump control carbon steel	6,000	
g. Valves, 1-8 in. pump control, carbon steel	2,000	
h. Valves, 1-12 in. temp control, carbon steel	5,000	
i. Valves, 2-16 in. float control, carbon steel	12,000	
j. Valves, 8-6 in. gate, carbon steel	500	
k. Valves, 5-16 in. gate, carbon steel	<u>12,000</u>	
		120,000
15) Instrumentation		
a. Fuel system, temp, pressure, pump speed	130,000	
b. Cooling water, temp, pressure, radiation	110,000	
c. Misc. process, level, valve position, vibration	46,000	
d. Remote maintenance machine Minotaur	30,000	
e. Monitoring, sub- systems, TV, sound, fuel processing, safety	21,000	
f. Health area mon- itoring and meteorological	90,000	
g. Computer-control, monitoring, data logging	500,000	
h. Control room and graphic panel	70,000	
i. Nuclear instrumentation	250,000	
j. Fuel processing	<u>15,000</u>	
		1,262,000
16) Computer, experiment and data logging		500,000

17) Wiring and conduits		
a. Penetration seals	6,000	
b. Conduit and radia- tion resistant wiring, 2000 ft-1 in. conduit w/4 #10	20,000	
200 ft-5 in. conduit w/7 #500 MCM	10,000	
c. Conduit and standard insulation 1500 ft-1 in. conduit w/4 #10	5,000	
1000 ft-1-1/4-in. conduit w/4 #6	4,000	
1000 ft-5 in. conduit w/7 #500 MCM	<u>15,000</u>	
		60,000
18) Installation except where noted		2,500,000
D. Utilities		
1) Electrical		
a. 600 ft-5 in. conduit w/15 kV cable	7,000	
b. 3000 ft-13.4 kV aerial	18,000	
c. 1-15 kV, 500 MVA air circuit breaker	15,000	
d. 3-15 kV, fusible load break switches	10,000	
e. 1-1000 KVA unit substation	30,000	
f. 1-auto-transfer scheme - battery	3,000	
g. 1-480 V, 3Ø switch gear	<u>12,000</u>	
		95,000
2) Gas and water		
a. 10,000 gal water tank	80,000	
b. 700 ft-8 in. water line	6,300	
c. 240 ft-1-1/4 in. gas line	1,200	
d. Relocate fire hydrant and piping	2,000	
e. Misc. testing, cutting, check out	<u>10,500</u>	
		100,000
Total Construction Cost		\$ 10,700,000

APPENDIX B
REACTOR PHYSICS OF THE KING REACTOR

1. Background

In the KING reactor an aqueous uranyl sulfate solution (H/U atomic ratio $\sim 100:1$) moves rapidly through a Zr tube in which criticality cannot be obtained except in the presence of a reflector. Thus the reflector defines the "core region." The geometric configuration of a KING reactor is indicated schematically in Fig. 1.

The basic design conditions for a KING reactor are as follows: (1) The H/U ratio in the core region is sufficiently large so that a KING reactor should operate as a thermal reactor. (2) To achieve a maximum neutron flux the reactor core must be small so that reflector moderation is necessary. (3) During core passage at full power the fuel solution temperature rises and voids are formed due to radiolytic gas formation. Fuel density, and local fission density therefore depend on the distance from the core entrance, and on the power level of the reactor. (4) The fluence experienced by the fuel solution during core passage must not lead to fuel decomposition or boiling.¹ This requirement, as well as the uranium solubility limit, imposes a constraint on the uranium concentration. (5) For realistic flow speeds the fuel residence time in the core is of the order of 0.1 sec or less. Therefore delayed neutron precursors are swept out of the core before decaying and the dynamic response of the KING reactor will vary with flow velocity, and will be based on a substantially reduced delayed neutron fraction. The response will depend, however, also on the effect of the neutrons slowed down in the reflector; these neutrons will constitute an extra "delayed neutron group" affecting the dynamics of the KING reactor.

For the conceptual design of a KING-type reactor the following reactor physics aspects require investigation:

- (1) Overall flux optimization to achieve maximum flux in the flux trap.
- (2) Effect of slowing down and thermalization in different media (fuel solution, central flux trap containing water, exterior Be reflector).
- (3) Local changes in flux, energy disposition and temperature rise caused by the density changes in the fuel solution during core passage.

(4) Representation of the effect of density changes in standard multigroup neutronics calculations.

(5) Neutron spectra in the flux trap, fuel solution, and Be reflector and their effect on few group constants necessary for detailed design calculations.

(6) Reactor dynamics effects of the fuel motion, density changes, and time delays caused by reflector moderation.

(7) Effects of heat transfer requirements on neutronic performance of a KING reactor.

The results obtained so far from preliminary reactor analysis calculations are summarized in the following sections, together with comments on further desirable work in reactor analysis for the KING Reactor.

2. Survey and Optimization Calculations

(a) Choice of Cross Sections

Preliminary survey calculations for the KING reactor, and for the KINGLET experiment, have been performed using the Hansen-Roach² 16 group cross section set. The use of these cross sections for a survey calculation appears well justified in view of older results of Stratton³ who has shown that criticality calculations using the Hansen-Roach cross section set describe very well the results of the critical experiments⁴ of Dixon, Callihan and others at ORNL. This point appears particularly pertinent because the ORNL experiments include aqueous solutions of ²³⁵U in the 50-300 H/U atomic ratio range, that is in the range of uranium concentrations relevant for the KING reactor design.

The Hansen-Roach cross section set does not provide explicitly for upscattering, and a free-gas model for scattering by hydrogen is assumed. It is therefore desirable to complement the survey calculations described here by more detailed calculations taking into account neutron thermalization by bound hydrogen, as discussed in the next section.

For the preliminary survey, the effects of anisotropic scattering and of the thermal and epithermal group structure were investigated by criticality calculations for a KING-type reactor without a central flux trap (KINGLET geometry).

Four cross section sets were considered:

- (1) Hansen-Roach, 16-group-two tables.

- (2) Hansen-Roach, 16-group transport approximation.
- (3) Hansen-Roach-Sapir, 24-group transport approximation.
- (4) ENDFB based (LaBauve), 27-group transport approximation.

Figure 2 summarizes some of the results of these calculations, showing the fission density as a function of radius for the four cross section sets. In a KING-type reactor, reflector moderation (in the central flux trap as well as in the Be reflector) is very important. A sharp increase in thermal flux, and therefore in fission density, is to be expected at core-reflector interfaces. It is apparent from Fig. 2 that the two-table 16 group cross section set gives a sharper increase of the fission density near the interface than the transport approximation calculations. The two-table calculation will therefore predict the largest, and thus most unfavorable, ratio of peak-to-average flux and peak-to-average energy deposition in the core. Results from a two-table calculation should therefore permit a more realistic estimate of the effects of void formation on the behavior of a KING-type reactor. The two-table 16-group Hansen-Roach cross section set was therefore chosen for the survey calculations reported here.

(b) Parameter Study

A first parameter survey dealt with the effects of changes in flux trap radius a , in thickness of the cylindrical shell containing the fuel solution, and in fuel concentration. The effect of replacing the Be reflector by a nickel or a graphite reflector was also considered. These calculations dealt with an idealized system, that is a system without cooling channels for the control vane and the Be reflector. The effects of the cooling requirements on the KING reactor performance are reported separately in Section 5.

Figure 3 shows the fission density obtained from a one-dimensional calculation for a critical system with concentration $c = 220$ g/liter of ^{235}U .

For lower concentrations $c = 170, 140, 110$, and 86 g/liter of ^{235}U , the spikes in the fission density at the interfaces between fuel and flux trap and between fuel and Be reflector are reduced, but they are still very noticeable, as evident from Table I. Figures 4-1 to 4-16 show representative flux distributions for a KING system. The flux

peaking in the flux trap and the Be reflector occurs also in the lowest energy group for the lower fuel concentrations, but the flux levels are reduced. Concentrations beyond 220 g/liter of ^{235}U have not been investigated, consistent with the overall design assumption of a conservatively defined KING system.*

Results of a preliminary optimization study have been reported separately.⁵ The performance index I for the reactor, defined by Feinberg⁶ as the ratio of maximum thermal flux in the flux trap to the reactor power reaches a well-defined maximum for an island radius of about 4.25 cm. The magnitude of the performance index increases with fuel concentration, but the rate of increase appears to decrease for $c \sim 220$ g/liter. Representative results, and a comparison with existing or projected high flux reactors are given in Table II.

Calculations have also been performed with flux traps of larger radii (e.g., 5.25 cm) containing irradiation samples. Indications are that the samples reduce the flux by about 40%, in good agreement with the flux reduction of about 50%, reported for thin samples in the HFIR.⁷ Interesting transplutonium production rates could be obtained with a KING reactor operating at 100 MW. With no other absorbers present it appears that a flux of 2×10^{16} may possibly be reached. In such a situation, a KING reactor could produce in a two-day run about ten times the yield of ^{257}Fm expected from a similar sample in a complete HFIR irradiation cycle.⁸

3. Thermalization Study

A preliminary 27 group cross section set was obtained from ENDF-B data files, using processing codes MC² and ETQG operating at LASL.⁹ Insufficient support was available to ascertain the intrinsic consistency of the procedure and of the resulting cross sections. A review of the thermalization data, reported separately¹⁰ led to the conclusion

* In such a system the fuel solution experiences less fluence during a core passage than that encountered without decomposition by similar fuel in the KEWB experiment.¹

that a simple calculation using the Nelkin model should be adequate for the preliminary design study of the KING reactor.

The computer program LANEL¹¹ has been developed and checked which provides for point cross sections in the low energy region (e.g., below 1 eV) for neutron scattering by bound hydrogen. This program uses the parameters describing energy transfer introduced by Nelkin, but can be adapted to other, more recent data. Provisions are included to treat anisotropic scattering in the multi-table format (up to four tables are permitted).

The link required to make these cross sections useful for DTF-IV or TWOTRAN multi-group calculations has not yet been implemented. For this purpose, it is required to obtain first fine group cross sections for the region below 1 eV and then to mesh them with a chosen cross section set for the energy region above 1 eV.

As a first step to investigate the effects of upscattering in the thermal region, a 25 group transport approximation (one-table) DTF IV calculation was performed; the preliminary results of this calculation are reported in Section 9.

4. Effects of Void Formation in a KING Reactor

The fuel density in a representative geometric volume element in the KING reactor core varies during core passage as a function of energy deposition, that is of the fission density. An exact accounting of fuel density variations and of their effects on reactivity and on flux distributions will therefore require iterative two-dimensional calculations, because the fission density in turn can only be determined if the fuel density is prescribed.

For the simple model of laminar flow one obtains (See Appendix C) --

$$-\frac{\partial \rho(r,z)}{\partial z} = \frac{\epsilon}{\epsilon_v} \rho_0 F(r,z) \frac{1}{v(r,z)} \quad (1)$$

where $\rho(r,z)$ is the fuel density, $F(r,z)$ the fission density, $v(r,z)$ the axial flow velocity, ρ_0 the fuel density at core entrance, z the height above the core entrance, ϵ the energy in joules released from fission, and ϵ_v the energy deposition (in joules/cm³) required to create 1 cm³ of void in the fuel solution. For laminar flow, the average fuel density $\bar{\rho}(z)$ can be related to the average normalized fission density

$F_0(z)$, the averages being taken over the cross sectional areas of the fuel channel, with the result that

$$\bar{\rho}(z)/\rho_0 = \exp \left\{ - (P/\epsilon_v) \int_0^z F_0(z') dz' \right\} \quad (2)$$

where P is the average power density, τ the average core residence time, H the effective core height and

$$F_0(z) = \frac{\int_0^z F(z') dz'}{\int_0^H F(z') dz'}. \quad (3)$$

Eq. (3) is useful for an iterative approach. One-dimensional calculations have been performed indicating that the local variations in $\bar{\rho}(z)$ are not introducing major changes in the neutronic calculations. Table III summarizes the result of DTF-IV reactivity calculations for a KINGLET type system: $\Delta k^{(1)}$ is the calculated change in k from a full density solution obtained from a first iterate for $\bar{\rho}(z)$, while k_L is the value of k calculated assuming that $\bar{\rho}(z)$ varies linearly with z ; the same average void fraction $[\rho_0 + \rho_L(H)]/2$ is chosen in both calculations. It is apparent that the difference $k_L - k^{(1)}$ is quite small; this result is also reflected in Fig. 5, where the fission density in a full density KINGLET assembly is compared to the fission density calculated for a 20% average void fraction using a first iterate fuel density. Assuming a linear fuel density variation, the fission density obtained is practically identical to the one found using the first iterate $\bar{\rho}(z)$.

For lesser core heights (~ 55 cm) the deviations in fission density between full density fuel and fuel with a 20% void fraction will be enhanced, but the error introduced by using a simple approximation for $\rho(z)$ should not be increased materially.

The effect of outward leakage in the energy groups above 1 eV and of the inward leakage in the lowest groups is accounted for here by introducing appropriately normalized fictitious (positive and negative) absorption cross sections into the 16 group DTF-IV calculations, using results obtained from previous criticality calculations assuming a full density fuel solution.

5. Cooling Requirements and KING Reactor Performance

Heat transfer limitations on the performance of the KING reactor may arise from the necessity of cooling the Zr core tube and the Be reflector. In these survey calculations, the following configurations were considered:

(1) A 0.952 cm water gap outside the Zr core tube, providing outside cooling for the core tube and the control cylinder; 10% voids (filled with water) in the inner 25 cm thickness of Be in the reflector, and 5% voids (filled with water) in the outer 25 cm thickness of the Be reflector.

(2) A 0.254 cm (0.1 in.) water gap providing cooling for the Zr core tube and the control cylinder, adjoining a 0.63 cm Al follower of the control cylinder, surrounded by a 0.254 cm water gap; 5% of water-filled voids throughout the Be reflector.

(3) Configurations as above with a nickel or graphite reflector, instead of the Be reflector.

Cases (1) and (3) led to a substantial reduction in the performance index. Some of the results obtained for case (2) are summarized in Table IV.

Even for case (2) a noticeable reduction in neutronic performance occurs. Fission density and group neutron flux are shown in Figs. 6 and 7. While the fission density is not changed very much the effect of the 0.2 in. of water between core and Be reflector causes a reduction in thermal flux, as shown in Fig. 7. The reduction in performance index is almost 20%, so that

$$I \approx 18 \times 10^{10} \text{ neut/cm}^2/\text{kW}$$

and the performance of a KING reactor would be less than four times as good as that of HFIR.

Thus it appears essential to perform a coupled heat transfer-neutronics study of the KING reactor, with particular emphasis on optimizing the heat transfer from the point of view of the neutronic effects of the coolant. Reduction in the amount of water near the core could be obtained by increasing coolant flow rates, with ensuing advantages for the KING reactor performance. This problem goes, however, well beyond the reactor physics and heat transfer aspects of the KING reactor design: if economic and engineering considerations require a

rapid turnaround of the fuel solution,* to minimize fuel inventory, then a near equilibrium distribution of fission products could build up; this would entail a considerable increase** of heat deposition in the Zr core tube and the reflector, compared with the values assumed by Malenfant in his investigation of "Radiation Heating in a KING Reactor." See Appendix D.

An alternative, but unattractive, possibility would be to cool the Zr core tube, the control cylinder, and the Be reflector with heavy water. Even if the purity requirements were relaxed ($D_2O/H_2O \sim 8:2$ molecular ratio or so) the ensuing engineering requirements could well lead to unfavorable consequences, that is complications in design and increased costs.

6. Preliminary Two-dimensional Reactor Statics Calculation for a KING Reactor

For a more detailed study of the KING reactor design two-dimensional flux plots and fission density distributions are essential. Results of a sample TWOTRAN 16 group calculation are shown in Figs. 8-1 to 8-16 for the same geometry (including Zr and Be cooling) as used for Figs. 6 and 7. Symmetry with regard to the midplane is assumed, as well as full density fuel. End effects are clearly noticeable, both in the flux trap and the reflector. As expectable, the flux decreases sharply with radius for the higher energy groups. Flux peaking in the flux trap and the Be reflector

* For example for an inventory of 1000 liters, a 10-liter core volume and a 0.1 sec residence time, the fuel would in the average have a rest period of only 10 sec, insufficient for fission product decay and perhaps even for "healing" of possible fuel solution radiation damage effects.

** Malenfant assumes that all delayed gammas and betas are swept out of the core region so that they will not be available to contribute to the power in the system. Since delayed gammas and beta kinetic energy contribute each about 7 MeV/fission, as compared to the 8 MeV/fission of prompt gamma energy, the contribution to reflector heating from the delayed gammas may be quite noticeable; the beta kinetic energy is here irrelevant since this contribution is likely to heat primarily the fuel solution.

are evident for the thermal (16th) group. The decrease in flux between core and reflector in that group is due in part to the absorption in the coolant, in part due to the outward leakage in the region of the aluminum follower of the control cylinder. A detailed representation of the effect of the fuel solution outside the core has not been attempted in this calculation, the primary objective being to check on the validity of the one-dimensional calculations. The same k value was found in the two-dimensional calculation as in the one-dimensional survey calculation. The flux plots shown in Figs. 8-1 to 8-16 may also be of interest for the preliminary design of beam tubes penetrating the Be reflector. For design cost estimates it may be of interest to note that this calculation required about 8 minutes of CDC 7600 time.

7. Fast Flux Enhancement in a KING Reactor

The KING reactor, based on the use of an aqueous fuel solution surrounding a central water island serving as a thermal flux trap, is intended to provide a maximum thermal flux for research and testing purposes. It appears, however, that the basic KING reactor configuration could be utilized in order to create an enhanced fast flux in a central irradiation facility as well.

To investigate this possibility, a modified KING reactor geometry has been considered, as shown in Fig. 9. In this version, the central water island is replaced by an empty hole, surrounded by two water-cooled cylindrical fuel elements. A larger cross sectional area of the core tube, as compared to the KING reactor parameter study, is required here in order to overcome the decrease in reactivity caused by the lack of reflector moderation in the central region. The cylindrical fuel elements are assumed to correspond in composition (^{235}U loading aluminum as structural material) to those used in the HFIR core.

Results of a sample calculation are summarized in Table V. It must be noted that heat transfer considerations may limit the attainable fast flux. The illustrative results presented in Table V lead to an average power density of about 16 MW/liter in the converter plate region while the average power density over the whole HFIR core is only 1.9 MW/liter, reaching 4.3 MW/liter at "hot spots."

Aside from the problem of heat transfer from fuel plate to coolant, it is apparent that about 4-5 MW would have to be extracted from the two fuel plate cylinders, a number barely reached at 50 ft/sec flow velocity and 70°C temperature rise in the coolant channels, and not allowing for cooling of the inner surface of the Zr core tube. Thus a converter plate arrangement would either have to provide for better cooling, that is larger water gaps and probably reduced flux or a nonconservative design permitting boiling of the coolant in the three inner channels surrounding the converter plates, or increased coolant flow velocities.

The fission density obtained in a KING-type reactor with fast flux enhancement is shown in Fig. 10. The neutron flux, based on a 16-group calculation for $k = 1.09$ (to account for realistic temperature and void effects) is shown in Figs. 11-1 to 11-16. It is apparent from Figs. 11-1 to 11-5 that the geometry of the fast flux region is far from optimal. It would be advantageous to have a larger inner radius for the flux trap region, and possibly some heavy element scattering medium, such as ^{238}U or Pb, inside the irradiation area. No optimization study for a "fast flux trap" has been attempted since it would require a preliminary detailed heat transfer analysis.

The fast flux enhancement is, however, very strong: Fig. 12 shows a sample histogram, comparing group fluxes at the center of the fast flux enhancement version with those attained in a KING Reactor using a water flux trap.

8. Reactor Dynamics of the KING Reactor

The dynamics of the KING reactor are influenced by

- (1) formation of voids and their fluctuations
- (2) temperature rise of fuel during core passage
- (3) delayed neutron fraction
- (4) delay times of reflector moderated neutrons.

In view of the short core residence times of about 0.1 sec and the assumed recycle time between 10 and 20 sec of the fuel solution, the effective neutron fraction will be reduced. Its value will depend both on the residence time and the recycle time. For the anticipated residence and cycle times the difference between "prompt critical" and "delayed

critical" will obviously be less than in ^{235}U fueled with fixed fuel elements. It is therefore of some interest to obtain an estimate of the effect of the delay in return due to slowing down and diffusion of the neutrons moderated in the Be reflector. A sample calculation for the KINGLET geometry has been performed using Monte Carlo techniques. The results indicate a distribution of delay times which could be approximated by one or two fictitious delay neutron groups. The fraction of delayed neutrons and their delay times appear significant, and of interest for future reactor kinetics studies of the KING reactor; by themselves they would facilitate control of the KING system. On the other hand, the prompt negative temperature coefficient measured for a KINGLET geometry appears adequate to assure the safety of the KING reactor, as discussed elsewhere by Wimett et al.¹² Qualitatively it appears that even if a sudden pump failure were to occur the only observable effect would be a moderate rise in fuel temperature probably without substantial fuel boiling.

9. Comparison of KING Reactor Parameters Obtained from 16 and 25 Group Calculations

In the parameter study, the Hansen Roach 16-group cross section set has been used which has the following limitations:

- (1) Coarse group structure in the low energy region (only three groups below 1 eV).
- (2) No up-scattering
- (3) Hydrogen is treated in the free atom approximation.

As a first step towards more detailed calculations, the criticality calculation as reported in Section 5 for a KING reactor with reflector and Zr tube cooling has been repeated using the cross section set reported by J. Sapir.¹³ This cross section set has been used in the analyses of the Phoebe-2 and Pewee reactors at Los Alamos. The median energy of neutrons causing fission in these reactors range from 0.4 eV in Pewee-1 to 70 eV in Phoebe-2; thus the spectrum in these reactors is considerably harder than the KING reactor spectra. Results obtained for the KING reactor should therefore be considered as tentative at best.

This cross section set was used because it has been tested against critical experiments, and because it has the following advantages:

- (1) The group structure in the low-energy region is much finer than in the 16-group set (12 groups below 1.125 eV).
- (2) Upscattering is allowed (up to four groups upward) in the low-energy region.

On the other hand hydrogen is still treated in the free gas approximation.

The 25-group calculations permit to compare the group-average fluxes in the low-energy region with a Maxwellian flux. Figure 13 shows the values of

$$\frac{\phi}{\phi_{\max}} = \frac{\frac{1}{(E_j - E_{j+1})} \int_{E_{j+1}}^{E_j} \phi(E) dE}{\frac{1}{.005} \int_{.025}^{.020} \phi(E) dE}, \quad (E_{j+1} < E_j)$$

the energy group 23, extending from 0.025 to 0.020 being the group where the group flux reaches its maximum value both in the flux trap and the fuel region. It is apparent that the flux in the fuel is noticeably harder than in the water island, and that this effect becomes quite pronounced in the epithermal region. Comparing the average values for the flux trap region with the Maxwellian distribution for $kT = 0.025$ eV, one notices a slight shift of the flux in the trap to higher energies, compared to the Maxwellian flux. For the flux in the fuel region this shift (as indicated by the location of the intersection of the curve for the Maxwellian flux with the bar indicating the calculated group value for $\phi/\phi_{\max}$ relative to the midpoint of the bar) is quite pronounced in most energy groups. For comparison, a Maxwellian with $kT = 0.028$ eV is also indicated in Fig. 13; this curve agrees better with the flux in the fuel solution in the region between 0.04 to 0.08 eV, and as expected does not represent the flux in the region above 0.1 eV, where the $1/E$ tail begins to dominate.

The pronounced flux peaking obtained from the 25-group calculation, and the good agreement between both calculations in the higher energy range is also apparent from the comparison shown in Figs. 14a and b.

The fission density distribution obtained from the 25-group calculation leads to $k \approx 0.99$. Using

the transport approximation leads to less sharp peaks at flux trap and reflector interfaces, as discussed before. Consequently the critical dimensions are increased compared to the 16-group value. Figures 15-1 to 15-25 show the group fluxes obtained. Figures 15-19 to 15-25 indicate the depressing effect of the cooling layer and of the Al control sleeve follower on the flux.

Table VI shows a comparison of the parameters obtained by a 16-group criticality calculation with a 25-group calculation leading to $k = 0.99$. The flux hardening illustrated in Fig. 13 is reflected in the reduction of fissions in the low-energy groups and in the increased core volume resulting from the 25-group calculation. This flux hardening is also reflected in the reduced performance index for the total region below 0.09 eV, and in the lower flux.

The last consideration also illustrates the need to be extremely careful in comparing results obtained by different calculational methods for different systems, e.g., HFIR, SM2, and the KING reactor. If one were to consider only the energy region between 0.025 eV and 0.020 eV as the "true" thermal region, the flux peaking shown in Fig. 13 were to lead to a "thermal flux in the region below 0.035 eV" of 4.7×10^{16} neutrons/cm² sec and an excessively high (190×10^{10}) performance index I . Clearly the value of I is meaningful only if calculations using identical reference data are used, and if it is related to the anticipated irradiation uses, i.e., of the integral of the product of flux in presence of absorber and absorber cross section.

Notwithstanding these considerations it appears that in view of the results of the 25-group calculation one may state that the 16-group results are to be considered as conservative estimates of the eventual KING reactor performance as a thermal neutron irradiation facility.

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TABLE B-I

CRITICAL KING REACTOR PARAMETERS AS FUNCTION OF FUEL LOADING (g/l) AND FLUX TRAP RADIUS Q(cm)
(for 40 cm height of full density fuel)

	$\Omega = 3.75$ cm			$\Omega = 4.25$ cm			$\Omega = 4.6$ cm			$\Omega = 5$ cm				
Fuel Loading in (g/l)	140	175	220	140	175	220	140	175	220	86	110	140	175	220
Critical Mass (in kg U^{235})	.74	.81	.89	.79	.93	.96	.84	.91	1.01	.78	.83	.90	.98	1.08
Critical Core Volume (in liters)	5.3	4.6	4.1	5.7	4.9	4.4	6.0	5.2	4.6	9.1	7.5	6.4	5.6	4.9
Core Thickness (in cm)	3.58	3.22	2.92	3.55	3.19	2.88	3.55	3.19	2.87	4.70	4.07	3.58	3.21	2.89
Apparent Specific Power (in MW/l)	4.8	5.4	6.2	4.4	5.1	5.8	4.2	4.8	5.4	2.8	3.3	3.9	4.5	5.1
Apparent Maximum Local Specific Power (in MW/l)	8.1	9.5	11.1	7.6	9.0	10.5	7.3	8.55	10.1	4.5	5.6	6.8	8.1	9.5
Maximum to Average Temperature Rise 1.0eV	1.7	1.75	1.8	1.72	1.77	1.83	1.73	1.79	1.85	1.63	1.69	1.74	1.80	1.86
$10^{-15} \times \int_0^{\infty} \phi(E)dE$ in Flux Trap *	5.2	5.4	5.5	5.3	5.5	5.6	5.3	5.4	5.6	4.4	4.8	5.1	5.3	5.5
% fissions due to neutrons below .1 eV	81	79	76	81.5	79	77	82	79	77	86	84	82	80	77

* Local maximum value at
25 MW Power Level

TABLE B-II

REPRESENTATIVE PARAMETERS FOR HIGH-FLUX REACTORS

	KING Reactor			
	HFIR (7)	SM-2 (6)	175 g ^{235}U /liter	220 g ^{235}U /liter
Performance index I, n/cm ² sec/kW	5.5×10^{10}	6×10^{10}	21.9×10^{10}	22.5×10^{10}
Reactor power rating (MW)	100	50 - 100	25	25
Maximum unperturbed thermal flux in flux trap, n/cm ² sec	5.5×10^{15}	3.0×10^{15a}	5.5×10^{15}	5.6×10^{15}
Representative fast flux in Be reflector	4.0×10^{15b}	10^{15b}	2.4×10^{14}	2.6×10^{14}
Core volume, liters	50.8	44	4.9	4.35
Core loading, kg ^{235}U	9.4	11	0.93	0.96
Length of operating cycle, days	14		Continuous Operation	
Coolant flow rate, ft/sec	42		50 ^c	50 ^c
Flux trap radius, cm	6.35	7	4.25	4.25
Height of active core, cm	50	40 - 50	40 ^d	40 ^d
Thickness of reflector, cm	30.5		50	50
Apparent specific power, MW/liter (av.)	1.9		5.0 ^e	5.75 ^a
hot spot	4.3		9.0	10.5

^aImprovement to 5 to 8×10^{15} anticipated (Ref. 6).

^bIn core.

^cFuel-solution flow rate.

^dHeight for full-density fuel, corresponding to core height of 50 to 55 cm under operating conditions.

^eFor full-density fuel.

TABLE B-III
ILLUSTRATION OF VOID EFFECTS ON REACTIVITY IN KING-TYPE REACTOR

$\bar{\rho}(H)$	$\Delta k^{(1)}$	$\delta k = k_L - k^{(1)}$	$\delta k/k^{(1)}$
0.6	0.0209	0.00173	0.002
0.7	0.0139	0.000764	0.0008
0.8	0.0084	0.000274	0.0003

H = 71 cm

TABLE B-IV
EFFECT OF HEAT TRANSFER REQUIREMENTS ON KING REACTOR PERFORMANCE
Fuel loading 220 g/liter and 40 cm full density fuel height

Flux Trap radius C (in cm)	4.6	4.25	4.25
	No Reflector or Zr	Cooling	With Cooling*
Critical Mass (in kg ²³⁵ U)	1.01	.96	1.12
Core Volume (in liters)	4.6	4.35	5.1
Core Thickness (in cm)	2.875	2.88	3.27
Apparent Specific Power (in MW/l)	5.44	5.75	4.98
Apparent Local Maximum Specific Power (in MW/l)	10.	10.5	8.5
Maximum to Average Temperature Rise	1.85	1.83	1.70
$\int \phi(E) dE$ thermal	in Flux Trap**	5.6×10^{15}	4.5×10^{15}
	in Fuel**	1.4×10^{15}	1.2×10^{15}
	in Be Reflector**	1.2×10^{15}	1.8×10^{15}
$\int \phi(E) dE$ 1.4 MeV 0.9 MeV	in Flux Trap**	3.6×10^{14}	1.8×10^{14}
	in Fuel**	4.7×10^{14}	4.2×10^{14}
	in Be Reflector**	2.5×10^{14}	5.1×10^{13}
% fissions below .1 eV	79	77	78
Performance Index I (neutrons/cm ² sec/kw)	22.4×10^{10}	22.6×10^{10}	18×10^{10}

* 5% by volume of Be replaced by light water

Zr cylinders surrounded by two light water annuli of 0.1" thickness and

Al control cylinder follower

** Local Maximum Values

TABLE B-VI

Comparison of 16 Group and 25 Group Calculations for a KING Reactor

		16 Group ^{1/}	25 Group ^{2/}
Flux trap radius (in cm)		4.25	4.25
Critical mass (in kg)		1.12	~ 1.36
Critical volume (in liters)		5.1	~ 5.6
Core thickness (in cm)		3.27	~ 3.52
Maximum to average temperature rise		1.70	~ 1.62
$\int_0^{0.09} \phi(E) dE$	Flux trap ^{5/}	4.5×10^{15} ^{3/}	2.7×10^{15} ^{4/}
	Core ^{5/}	1.2×10^{15} ^{3/}	0.77×10^{15} ^{4/}
	Be reflector ^{5/}	1.8×10^{15} ^{3/}	1.11×10^{15} ^{4/}
1.4 MeV $\int \phi(E) dE$ 0.9 MeV	Flux trap ^{5/}	1.8×10^{14}	1.72×10^{14}
	Core ^{5/}	4.2×10^{14}	3.94×10^{14}
	Be reflector ^{5/}	5.1×10^{13}	6.56×10^{13}
% fission in thermal region		70% ^{2/}	72% ^{4/}
Performance Index		18×10^{10}	11×10^{10}
based on		Group 16	All groups below 0.09 eV

1. Two-table (anisotropic) scattering
2. One table transport approximation
3. for group 16 (thermal)
4. for groups 20-25 ($E < 0.09$ eV)
5. local maximum values

TABLE B-V

Estimated Flux Levels in FFTF, in PBF, and in a King Type Reactor with Fast Flux Enhancement

Reactor	Power Level	Calculated Flux Spectrum $\int \phi(E) dE$			
		$10 \text{ MeV} \leq E \leq 0.82 \text{ MeV}$	$0.82 \text{ MeV} \leq E \leq 5.5 \text{ keV}$	$5.5 \text{ keV} \leq E \leq 0.5 \text{ eV}$	$E < 0.5 \text{ eV}$
PBF*	20 MW	1.4×10^{13}	3.1×10^{13}	2.9×10^{13}	1.8×10^{13}
FFTF*	400 MW	estimated total flux 7×10^{15}			
KING***	25 MW	$10 \text{ MeV} \leq E \leq 0.1 \text{ MeV}$	$0.1 \text{ MeV} \leq E \leq 3 \text{ keV}$	$3 \text{ keV} \leq E \leq 0.4 \text{ eV}$	$E < 0.4 \text{ eV}$
		1.6×10^{15}	0.44×10^{15}	0.77×10^{15}	0.26×10^{15}

* Power Burst Facility Test Space $8\frac{1}{2}$ " ID ;

** Fast Flux Test Facility

*** King Type Reactor Test Space estimated as $3\frac{1}{2}$ " ID.

Values near center of test space : 40 cm full density fuel height.

Fig. B-1. Schematic of horizontal section of the KING reactor.

Fig. B-3. Fission density as a function of radius.

Fig. B-2. Effects of choice of cross section set.

Fig. B-4. Neutron Flux vs radius for 16 energy groups.

Fig. B-4 (cont.)

Fig. B-4 (cont.)

Fig. B-5. Fission density vs core elevation.

Fig. B-6. Fission density as a function of radius.

Fig. B-7. Neutron flux vs radius.

Fig. B-8. TWOTRAN 16-group calculation flux plots.

IDENTIFICATION

A A A A
B B B B
C C C C
D D D D
E E E E
F F F F
G G G G
H H H H
I I I I
J J J J
K K K K
L L L L
M M M M
N N N N
O O O O
P P P P
Q Q Q Q
R R R R
S S S S
T T T T

CONTOUR VALUE

1.927E-09
1.690E-05
3.379E-05
5.069E-05
6.759E-05
8.448E-05
1.014E-04
1.183E-04
1.352E-04
1.521E-04
1.690E-04
1.869E-04
2.028E-04
2.197E-04
2.366E-04
2.534E-04
2.703E-04
2.872E-04
3.041E-04
3.210E-04

MT-2-95 IN 2-D (FELIX ADLER)
GROUP NUMBER 2

IDENTIFICATION

CONTOUR VALUE

A	7.085E-09
B	3.284E-05
C	6.568E-05
D	9.851E-05
E	1.313E-04
F	1.642E-04
G	1.970E-04
H	2.299E-04
I	2.627E-04
J	2.955E-04
K	3.284E-04
L	3.612E-04
M	3.940E-04
N	4.269E-04
O	4.597E-04
P	4.925E-04
Q	5.254E-04
R	5.582E-04
S	5.910E-04
T	6.239E-04

MT-2-95 IN 2-D (FELIX ADLER)
GROUP NUMBER 3

IDENTIFICATION

CONTOUR VALUE

A	3.079E-09
B	1.580E-05
C	3.159E-05
D	4.739E-05
E	6.319E-05
F	7.898E-05
G	9.478E-05
H	1.106E-04
I	1.264E-04
J	1.422E-04
K	1.580E-04
L	1.738E-04
M	1.896E-04
N	2.053E-04
O	2.211E-04
P	2.369E-04
Q	2.527E-04
R	2.685E-04
S	2.843E-04
T	3.001E-04

Fig. B-8 (cont.)

HT-2-06 IN 2-D (FELIX ADLER)
GROUP NUMBER 4

IDENTIFICATION

A - A - A - A
B - B - B - B
C - C - C - C
D - D - D - D
E - E - E - E
F - F - F - F
G - G - G - G
H - H - H - H
I - I - I - I
J - J - J - J
K - K - K - K
L - L - L - L
M - M - M - M
N - N - N - N
O - O - O - O
P - P - P - P
Q - Q - Q - Q
R - R - R - R
S - S - S - S
T - T - T - T

CONTOUR VALUE

6.814E-09
2.268E-05
4.616E-05
6.773E-05
9.030E-05
1.129E-04
1.356E-04
1.580E-04
1.806E-04
2.032E-04
2.257E-04
2.483E-04
2.709E-04
2.935E-04
3.160E-04
3.386E-04
3.612E-04
3.838E-04
4.063E-04
4.289E-04

HT-2-06 IN 2-D (FELIX ADLER)
GROUP NUMBER 5

IDENTIFICATION

A - A - A - A
B - B - B - B
C - C - C - C
D - D - D - D
E - E - E - E
F - F - F - F
G - G - G - G
H - H - H - H
I - I - I - I
J - J - J - J
K - K - K - K
L - L - L - L
M - M - M - M
N - N - N - N
O - O - O - O
P - P - P - P
Q - Q - Q - Q
R - R - R - R
S - S - S - S
T - T - T - T

CONTOUR VALUE

1.709E-08
2.212E-05
4.421E-05
6.631E-05
8.841E-05
1.105E-04
1.326E-04
1.547E-04
1.768E-04
1.989E-04
2.210E-04
2.431E-04
2.652E-04
2.873E-04
3.094E-04
3.316E-04
3.536E-04
3.757E-04
3.978E-04
4.199E-04

Fig. B-8 (cont.)

HT-2-06 IN 2-D (FELIX ADLER)
GROUP NUMBER 6

IDENTIFICATION

A - A - A - A
B - B - B - B
C - C - C - C
D - D - D - D
E - E - E - E
F - F - F - F
G - G - G - G
H - H - H - H
I - I - I - I
J - J - J - J
K - K - K - K
L - L - L - L
M - M - M - M
N - N - N - N
O - O - O - O
P - P - P - P
Q - Q - Q - Q
R - R - R - R
S - S - S - S
T - T - T - T

CONTOUR VALUE

7.943E-09
1.543E-05
3.085E-05
4.628E-05
6.170E-05
7.712E-05
9.254E-05
1.080E-04
1.234E-04
1.388E-04
1.542E-04
1.697E-04
1.851E-04
2.005E-04
2.159E-04
2.313E-04
2.468E-04
2.622E-04
2.776E-04
2.930E-04

HT-2-06 IN 2-D (FELIX ADLER)
GROUP NUMBER 7

IDENTIFICATION

A - A - A - A
B - B - B - B
C - C - C - C
D - D - D - D
E - E - E - E
F - F - F - F
G - G - G - G
H - H - H - H
I - I - I - I
J - J - J - J
K - K - K - K
L - L - L - L
M - M - M - M
N - N - N - N
O - O - O - O
P - P - P - P
Q - Q - Q - Q
R - R - R - R
S - S - S - S
T - T - T - T

CONTOUR VALUE

7.234E-09
1.179E-05
2.357E-05
3.636E-05
4.713E-05
5.891E-05
7.070E-05
8.248E-05
9.426E-05
1.060E-04
1.178E-04
1.296E-04
1.414E-04
1.632E-04
1.649E-04
1.767E-04
1.885E-04
2.003E-04
2.121E-04
2.239E-04

Fig. B-8 (cont.)

MT-2-DE IN 2-D (FELIX ADLER)
GROUP NUMBER 8

IDENTIFICATION

A A A A
B B B B
C C C C
D D D D
E E E E
F F F F
G G G G
H H H H
I I I I
J J J J
K K K K
L L L L
M M M M
N N N N
O O O O
P P P P
Q Q Q Q
R R R R
S S S S
T T T T

CONTOUR VALUE

7.495E-09
1.050E-05
2.099E-05
3.148E-05
4.197E-05
5.247E-05
6.296E-05
7.345E-05
8.394E-05
9.443E-05
1.049E-04
1.154E-04
1.259E-04
1.364E-04
1.469E-04
1.574E-04
1.679E-04
1.784E-04
1.889E-04
1.994E-04

MT-2-DE IN 2-D (FELIX ADLER)
GROUP NUMBER 9

IDENTIFICATION

A A A A
B B B B
C C C C
D D D D
E E E E
F F F F
G G G G
H H H H
I I I I
J J J J
K K K K
L L L L
M M M M
N N N N
O O O O
P P P P
Q Q Q Q
R R R R
S S S S
T T T T

CONTOUR VALUE

8.144E-09
1.006E-05
2.011E-05
3.016E-05
4.021E-05
5.027E-05
6.032E-05
7.037E-05
8.042E-05
9.047E-05
1.005E-04
1.106E-04
1.206E-04
1.307E-04
1.407E-04
1.508E-04
1.608E-04
1.709E-04
1.809E-04
1.910E-04

Fig. B-8 (cont.)

MT-2-95 IN 2-D (FELIX ADLER)
GROUP NUMBER 10

IDENTIFICATION

A A A A
B B B B
C C C C
D D D D
E E E E
F F F F
G G G G
H H H H
I I I I
J J J J
K K K K
L L L L
M M M M
N N N N
O O O O
P P P P
Q Q Q Q
R R R R
S S S S
T T T T

CONTOUR VALUE

6.043E-09
6.767E-06
1.353E-05
2.029E-05
2.705E-05
3.381E-05
4.057E-05
4.733E-05
5.409E-05
6.085E-05
6.761E-05
7.437E-05
8.113E-05
8.789E-05
9.465E-05
1.014E-04
1.082E-04
1.149E-04
1.217E-04
1.285E-04

MT-2-95 IN 2-D (FELIX ADLER)
GROUP NUMBER 11

IDENTIFICATION

A A A A
B B B B
C C C C
D D D D
E E E E
F F F F
G G G G
H H H H
I I I I
J J J J
K K K K
L L L L
M M M M
N N N N
O O O O
P P P P
Q Q Q Q
R R R R
S S S S
T T T T

CONTOUR VALUE

5.761E-09
6.087E-06
1.217E-05
1.825E-05
2.433E-05
3.041E-05
3.649E-05
4.257E-05
4.865E-05
5.473E-05
6.082E-05
6.690E-05
7.298E-05
7.906E-05
8.514E-05
9.122E-05
9.730E-05
1.034E-04
1.096E-04
1.156E-04

Fig. B-8 (cont.)

HT-2-06 IN 2-D (FELIX ADLER)
GROUP NUMBER 12

IDENTIFICATION

A A A A
B B B B
C C C C
D D D D
E E E E
F F F F
G G G G
H H H H
I I I I
J J J J
K K K K
L L L L
M M M M
N N N N
O O O O
P P P P
Q Q Q Q
R R R R
S S S S
T T T T

CONTOUR VALUE

6.542E-09
6.457E-06
1.291E-05
1.936E-05
2.581E-05
3.226E-05
3.871E-05
4.516E-05
5.161E-05
5.806E-05
6.451E-05
7.096E-05
7.742E-05
8.387E-05
9.032E-05
9.677E-05
1.032E-04
1.097E-04
1.161E-04
1.226E-04

HT-2-06 IN 2-D (FELIX ADLER)
GROUP NUMBER 13

IDENTIFICATION

A A A A
B B B B
C C C C
D D D D
E E E E
F F F F
G G G G
H H H H
I I I I
J J J J
K K K K
L L L L
M M M M
N N N N
O O O O
P P P P
Q Q Q Q
R R R R
S S S S
T T T T

CONTOUR VALUE

6.197E-09
5.828E-06
1.165E-05
1.747E-05
2.329E-05
2.912E-05
3.494E-05
4.076E-05
4.658E-05
5.240E-05
5.822E-05
6.405E-05
6.987E-05
7.569E-05
8.151E-05
8.733E-05
9.316E-05
9.898E-05
1.048E-04
1.106E-04

Fig. B-8 (cont.)

HT-2-96 IN 2-D (FELIX ADLER)
GROUP NUMBER 14

IDENTIFICATION

A - A - A - A
B - B - B - B
C - C - C - C
D - D - D - D
E - E - E - E
F - F - F - F
G - G - G - G
H - H - H - H
I - I - I - I
J - J - J - J
K - K - K - K
L - L - L - L
M - M - M - M
N - N - N - N
O - O - O - O
P - P - P - P
Q - Q - Q - Q
R - R - R - R
S - S - S - S
T - T - T - T

CONTOUR VALUE

5.322E-09
4.854E-06
9.702E-06
1.455E-05
1.940E-05
2.425E-05
2.910E-05
3.394E-05
3.879E-05
4.364E-05
4.849E-05
5.334E-05
5.819E-05
6.304E-05
6.788E-05
7.273E-05
7.758E-05
8.243E-05
8.728E-05
9.213E-05

HT-2-96 IN 2-D (FELIX ADLER)
GROUP NUMBER 15

IDENTIFICATION

A - A - A - A
B - B - B - B
C - C - C - C
D - D - D - D
E - E - E - E
F - F - F - F
G - G - G - G
H - H - H - H
I - I - I - I
J - J - J - J
K - K - K - K
L - L - L - L
M - M - M - M
N - N - N - N
O - O - O - O
P - P - P - P
Q - Q - Q - Q
R - R - R - R
S - S - S - S
T - T - T - T

CONTOUR VALUE

9.855E-09
1.507E-05
3.012E-05
4.618E-05
6.023E-05
7.529E-05
9.034E-05
1.064E-04
1.205E-04
1.355E-04
1.506E-04
1.666E-04
1.807E-04
1.957E-04
2.108E-04
2.268E-04
2.409E-04
2.560E-04
2.710E-04
2.861E-04

Fig. B-8 (cont.)

Fig. B-8 (cont.)

Fig. B-9. Schematic representation of KING reactor with fast-flux enhancement.

Fig. B-10. Fission density vs radius from a 16-group, two-table calculation for the KING-type reactor with fast-flux enhancement.

Fig. B-11. Neutron flux vs radius for 16 energy groups.

Fig. B-11 (cont.)

Fig. B-11 (cont.)

Fig. B-11. (cont.)

Fig. B-11 (cont.)

Fig. B-12. Group flux comparison with and without fast-flux enhancement.

Fig. B-13. Comparison of group average fluxes in low-energy region to Maxwellian flux.

B-14A. Comparison of 16-group and 25-group calculations in higher energy range.

Fig. 14B. Comparison of 16-group and 25-group calculations in higher energy range.

Fig. B-15. (cont.)

Fig. B-15. (cont.)

Fig. B-15. (cont.)

Fig. B-15. (cont.)

Fig. B-15. (cont.)

APPENDIX C
EFFECTS OF VOID FORMATION IN A KING REACTOR

Abstract

A simple model is presented for the calculation of fuel density changes due to void formation in a KING-type reactor, assuming steady state operating conditions and essentially unperturbed laminar flow of the fuel solution through the reactor core. Representative numerical results have been obtained for the axial changes of fuel density, neutron flux and power distribution arising from the presence of voids in the fuel solution.

1. Changes in Fuel Solution Density During Core Passage

The simple model considered here is based on the assumptions of steady state reactor operation and essentially unperturbed laminar steady state flow of the fuel solution through the reactor core. Turbulent mixing will reduce the radial density changes of the fuel, but will not significantly affect the axial density changes. Figure 1 illustrates schematically the KING reactor geometry and the assumed flow conditions. Even for this simplified situation the neutron balance equation for the KING reactor becomes nonlinear: at each elevation z , the fission density $F(r, z)$, and thus the energy deposition in a given volume of fuel solution, is proportional to the local fuel solution density $\rho(r, z)$, which in turn depends on the energy deposition that has occurred in the fuel volume during its core passage to the elevation z .

To describe the effects of void formation a representative geometric control volume ΔV will be considered, assuming that ΔV moves with the flow velocity $v(r, z)$ through the core in the axial (z) direction. Let $\Delta V_v(r, z)$ denote the void volume in ΔV and $\Delta V_{sol}(r, z)$ the volume of full density fuel in ΔV at the position r, z in the core. Then, denoting the void fraction by $F(r, z)$

$$\Delta V_{sol} = \Delta V - \Delta V_v = \Delta V [1 - F(r, z)] \quad (1)$$

and the fuel solution density $\rho(r, z)$ is

$$\rho(r, z) = \rho_0 \frac{\Delta V_{sol}}{\Delta V} = \rho_0 [1 - F(r, z)] \quad (2)$$

ρ_0 being the fuel solution density in the absence of voids. The energy deposition $\Delta E(r, z)$ in ΔV during the time $\Delta t = \Delta z/v(r, z)$ required for moving ΔV from

elevation z to the elevation $z + \Delta z$ in the core is given by

$$\Delta E(r, z) = \epsilon \Delta t \Delta V F(r, z) \quad (3)$$

where

$$F(r, z) = \sum_{g=1}^G \Sigma_g^f(r, z) \phi_g(r, z), \quad (4)$$

$\Sigma_g^f(r, z)$ being the macroscopic fission cross section for energy group g , ϕ_g , the corresponding flux, and ϵ the conversion factor denoting the energy in joules released per fission.

Since

$$\Delta t = \Delta z/v(r, z) \quad (5)$$

one has

$$\Delta E(r, z) = \epsilon \frac{\Delta z}{v(r, z)} \Delta V F(r, z) \quad (6)$$

The total energy deposition in ΔV , up to the moment when ΔV reaches the elevation z , is thus given by

$$E(r, z) = \Delta V \epsilon \int_0^z F(r, z') \frac{1}{v(r, z')} dz' \quad (7)$$

This amount of energy available for void formation must now be compared to the energy required to create the void volume ΔV_v . The KEWB experiments¹ indicate that the energy deposition required to create 1 cm³ of void in an aqueous uranium sulfate solution is

$$\epsilon_v = \text{joules/cm}^3 \quad (8)$$

Thus one obtains from (1), (2), and (7)

$$E(r, z) = \epsilon_v V_v(r, z) = \epsilon_v \left(1 - \frac{\rho(r, z)}{\rho_0}\right) \Delta V \quad (9)$$

Equation (9) is an integral equation for $\rho(r, z)$, differentiating with reference to z one obtains from (7) and (9) for a fixed value of r :

$$-\frac{\partial \rho(r, z)}{\partial z} = \frac{\epsilon}{\epsilon_v} \rho_0 F(r, z) \frac{1}{v(r, z)} \quad (10)$$

Since the fission density $F(r, z)$ can only be determined from the neutron balance equation if $\rho(r, z)$ is prescribed, it is apparent from Eq. (10) that in principle a system of coupled nonlinear neutronic-hydraulic equations must be solved to determine the neutron flux distribution. An iterative approach is therefore necessary to determine the neutronic behavior of a KING reactor at non-zero power levels.

2. Equation for the Axial Bulk Flow Effects

To simplify the hydraulic equation (10) it will be assumed that one may represent $v(r, z)$, $\rho(r, z)$, and $\theta_{th}(r, z)$ as products of functions:

$$v(r, z) = \bar{v}(z) D_1(r) \quad (11)$$

$$\rho(r, z) = \bar{\rho}(z) D_2(r) \quad (12)$$

$$F(r, z) = \bar{F}(z) \phi(r) \quad (13)$$

It will be useful to normalize $\phi(r)$ so that

$$\left[2\pi \int_{R_0}^R \phi(r) r dr / \pi(R^2 - R_0^2) \right] = 1 \quad (14)$$

so that $\bar{F}(z)$ is the fission density averaged over the cross sectional area of the core tube at z .

These assumptions appear to be acceptable because for a KING reactor the aspect ratio $H/(R - R_0)$ is large (~ 10) so that the v , ρ , and θ_{th} will be slowly varying functions of z .

Using the continuity equation for the total flow through the cross sectional area of the core tube one finds

$$2\pi \int_{R_0}^R v(r, z) \rho(r, z) r dr = \pi(R^2 - R_0^2) v_B \rho_0 \quad (15)$$

where v_B is the axial bulk flow velocity of the fuel solution at the entry to the core region. With the approximations (11) and (12) one obtains

$$\bar{v}(z) \bar{\rho}(z) \int_{R_0}^R D_1(r) D_2(r) r dr = \frac{1}{2} (R^2 - R_0^2) v_B \rho_0 \quad (16)$$

With the assumptions (11) to (13) one may express Eq. (10) in the form

$$-\bar{v}(z) D_1(r) D_2(r) \frac{d\bar{\rho}(z)}{dz} = \frac{\epsilon}{v} \rho_0 \bar{F}(z) \phi(r) \quad (17)$$

Integrating over the cross sectional area of the core tube and using Eq. (15) one obtains

$$-\frac{1}{\bar{\rho}(z)} \left[\frac{1}{2} (R^2 - R_0^2) v_B \rho_0 \right] \frac{d\bar{\rho}(z)}{dz} = \frac{\epsilon}{v} \rho_0 \bar{F}(z) \int_{R_0}^R \phi(r) r dr \quad (18)$$

Therefore, using the normalization of Eq. (14) one obtains

$$-\frac{1}{\bar{\rho}(z)} \frac{d\bar{\rho}(z)}{dz} = \frac{\epsilon}{v} \frac{1}{v_B} \bar{F}(z) \quad (19)$$

and, since $\bar{\rho}(z) = \rho_0$,

$$\frac{\bar{\rho}(z)}{\rho_0} = e^{-\frac{\epsilon}{v} \frac{1}{v_B} \int_0^z \bar{F}(z') dz'} \quad (20)$$

where one may recall that dimensionally the exponent has the form

$$\frac{(\text{joule/fissions})}{(\text{joule/cm}^3)} \left(\frac{1}{\text{cm/sec}} \right) \left(\frac{\text{fissions}}{\text{cm}^3/\text{sec}} \right)$$

It is useful to express $\bar{\rho}(z)$ in terms of the average in-core residence time τ of the fuel solution, defined by

$$\tau = H/v_B \quad (21)$$

and of the average power P (watts/cm³) defined by

$$P \cdot H \cdot A = \epsilon A \int_0^H F(z') dz' \quad (22)$$

where A is the cross sectional area of the fuel region. Introducing the normalized fission density

$$F_0(z) = \int_0^z F(z') dz' / \int_0^H F(z') dz' \quad (23)$$

one obtains

$$\frac{\bar{\rho}(z)}{\rho_0} = e^{-\frac{P \tau}{\epsilon v} F_0(z)} \quad (24)$$

In the iterative procedure of determining $F_0(z)$, one will in the j^{th} step have

$$\frac{\rho^{(j)}(z)}{\rho_0} = e^{-\frac{P \tau}{\epsilon v} F_0^{(j-1)}(z)} \quad (25)$$

For $j=1$ one may assume a simple sinusoidal axial fission density distribution so that

$$F_0^{(0)}(z) = \frac{\int_0^z A \sin \frac{\pi}{H} z' dz'}{\int_0^H A \sin \frac{\pi}{H} z' dz'} = \sin^2 \frac{\pi}{2H} z \quad (26)$$

and

$$\frac{\rho^{(1)}(z)}{\rho_0} = e^{-\frac{P \tau}{\epsilon v} \sin^2 \frac{\pi}{2H} z} \quad (27)$$

3. Numerical Results

The aims of the calculations reported here were to obtain an estimate for the order of magnitude of the void effects, and, second, to ascertain whether the results, for example for k , are sensitive to the specific form of $\bar{\rho}(z)$. With these goals in mind

parallel criticality calculations were made, using first $\rho^{(1)}(z)$ as given by Eq. (27), and then using an approximate linear form

$$\bar{\rho}_L(z) = \rho_0(1 - \alpha \frac{z}{H}) \quad (28)$$

In this case α was adjusted so that

$$\bar{\rho}_L(H) = \rho_0(1 - \alpha) = \bar{\rho}^{(1)}(H) = e^{-\frac{P \tau}{\epsilon v}} \quad (29)$$

Representative density distributions are shown in Fig. 2.

In Fig. 3a two fission densities are shown that were obtained using $\bar{\rho}_L(z)$ with $\alpha = 0$ (uniform density) and $\alpha = 0.2$ (10% average void). For a long core tube, $H = 71$ cm) the effect of the voids appears small but noticeable in the shift of position of maximum fission density to a point $z_m < H/2$ that is clearly noticeable.

Figures 3b and 3c show the corresponding flux distributions in groups 1 and 15 of a 16 group calculation. The void effect again appears as small but definite shift of the flux maxima. The relative smallness of the effect can be understood as due to two causes: (1) KING type reactors are strongly reflector moderated, and the slow neutrons returning from the central water island or the Be reflector tend to average out the effects due to the non-uniform fuel density distribution; (2) the core length of the test case considered (71 cm) is rather long and somewhat larger effects can be expected in a KING reactor with an active core height of 55 cm.

Values for k obtained by the first approximation ($k^{(1)}$) are compared in Table I with the values (k_L) obtained by the linear approximation of Eq. (28). One-dimensional DTF-IV calculations were performed to calculate $k^{(1)}$ and k_L ; the radial leakage effects in the different groups were simulated by introducing additional (positive or negative) capture terms in the neutron balance equation. These results indicate that, at least to calculate k , the linear approximation appears adequate for the case $(R-R_0)/H \sim 0.1$, which applies to the KING-type reactors.

Fig. C-1. Schematic of KING reactor and assumed flow conditions.

Fig. C-2. Representative density distributions.

Fig. C-3A. Fission density vs core elevation.

Fig. C-3B. Flux distribution vs core elevation, group 1 ($E > 3$ MeV).

Fig. C-3C. Flux distribution vs core elevation, group 15 ($0.1 \text{ eV} \leq 0.4 \text{ eV}$).

APPENDIX D

1. Fission Basis of Power and Radiation Heating in a KING Reactor

It has long been recognized that the relationship between fission rate and power in a nuclear system is a function of specific reactor, operating history, and the definition of the power producing region. For the KING Reactor, the situation is complicated by the rapid circulation of the fuel solution. Following a suggestion by W. L. Kirk (N-DO) that the power be defined on the basis of the rate of energy deposition in the fuel, it has been determined that 3.71×10^{10} fissions/sec will be needed to develop one watt. The derivation of the normalization and a complete breakdown of radiation heating for a model of the KING Reactor are presented.

The model employed for the calculations is indicated on Fig. 1 and the material composition by region is given in Table 1. With the exception of the fuel annulus which was run with a thickness of 3 cm (rather than 3.1 cm), the model is that described by H. J. Newman in an N-3 memo dated July 14, 1971. Energy depositions from prompt gamma rays from fission and neutron kinetic energy were calculated with DTF (transport theory). Neutron capture rates, determined from the neutron transport calculations, were then used to generate sources to calculate the energy depositions from capture gamma rays. A summary of the results is given in Table 2 (the number of significant figures carried is not to be considered indicative of the uncertainty). Table 3 provides the same results in units of power normalized to 25 MW deposition in the fueled region.

Two observations are made on the information in Table 2.

1) Delay gammas and betas were considered to be swept from the active core region so they would not be available to contribute to the power in the system. There are no prompt betas.

2) The 0.1765 MeV/fission of neutron kinetic energy labeled "other" went into several reactions--principally: Be (n,2n), Zr inelastic scattering, and high energy particle reactions. Some of this energy ultimately appears in the system as neutron kinetic, gamma ray, or particle kinetic energy.

The cross sections employed did not permit a partitioning of this energy to the various sources and regions so the contribution is not included.

Details of the radial distribution of the axially averaged energy depositions are given by material in Tables 4-6. Peak values may be estimated at about 1.5 times the average.

Estimated uncertainty in the heating rates is 15%.

2. Gamma Ray Dose Rate in the Vicinity of the KING Reactor Fuel

The gamma ray dose rate in the vicinity of a pipe filled with KING Reactor fuel at near equilibrium concentration of fission products has been calculated. A source of 1.5×10^8 curies of mixed fission products was assumed to be distributed in 2000 liters of solution. The fuel was considered to be contained in a 5 in. i.d., 1/2 in. wall, steel pipe. Gamma ray dose rate as a function of distance from the pipe center is shown in the table below.

Distance from Pipe Center, cm	Gamma Dose Rate rads/sec	Gamma Heating Rate watts/g (Fe)
7.62	3.0×10^3	0.027
10.0	2.2×10^3	0.020
15.0	1.4×10^3	0.013
25.0	8.3×10^2	0.0076
50.0	4.0×10^2	0.0037
100.0	1.7×10^2	0.0016
200.0	6.1×10^1	0.00056

The calculation is expected to be conservative (i.e., high) as all fission products were assumed to be contained in the fuel and no on-line cleanup was considered. Gamma ray heating rates presented above may also be of interest as they may be sufficiently high to require consideration.

3. Fission Product Inventory in the KING Reactor

Gross fission product concentrations in the fuel of a KING Reactor are indicated in Tables 7-9. The values should be most useful for estimating thermal cooling requirements.

The results were derived from "Studies of ²³⁵U Post-Fission Gamma and Beta Radiation and of Post-Shutdown Radiation Release from Fission Operations at Constant Power," by Peterson, Graves and Seale (internal N-2 document). Complete

and instantaneous mixing of the fuel is tacitly assumed in Tables 7-9. Conversion of units to curies was made assuming a single 1-MeV photon or electron per disintegration.

4. Neutron and Gamma Activity of the KING Reactor Fuel

A gamma irradiation facility using the KING reactor fuel looks attractive because of the high gamma intensities available and large gamma-to-neutron ratios which can be achieved. Data are presented to substantiate the claim.

Neutron activity of the KING fuel was calculated using standard delayed neutron half-lives, decay constants, and yields (Keepin, "Physics of Nuclear Kinetics," Addison-Wesley, p. 90). A 2000 liter total fuel volume was assumed along with a core volume of 7.07 liters and a fuel flow rate of 208 liters/sec (3300 gpm). Because a significant portion of the total neutron activity results from delayed-groups with half-lives short compared to the 9.61 sec fuel cycle time, no mixing of the fuel was assumed.

Gamma ray activity was calculated with the same considerations but complete and instantaneous mixing was assumed. The latter is a good assumption for the gamma activity for decay times of-the-order-of or longer than the cycle time.

Calculated results as a function of time after the last pass of a fuel sample through the core are shown in Fig. 2. A power level of 25 MW was assumed with a fission-rate-to-power conversion of 3.7×10^{10} fissions/sec-watt.

In addition to the dramatic increase in the gamma-to-neutron ratio with time, the low energy of the delayed neutrons makes their effectiveness in producing radiation damage small. After about 100 sec, the only effective delay group has a half-life of 55.72 sec and an energy of 250 keV (Keepin, op. cit., p. 95). As such, the potential exists for an intense gamma ray irradiation facility with vanishingly small neutron backgrounds.

Table 1
KING Material Description

Region cm	Composition	
	atoms/barn-cm	g/cm ³
Island 0.005.0	H - 0.06686 O - 0.03343	H - 0.1119 O - 0.8881
Tube 5.0-5.32	Zr - 0.0432	Zr - 6.543
Fuel 5.32-8.32	H - 0.049902 O - 0.027475 S - 0.0004205 U - 0.0004205	H - 0.0835 O - 0.730 S - 0.0224 U - 0.1642
Tube	Zr - 0.04324	Zr - 6.543
Void 8.955-9.910	-	-
Reflector ⁽¹⁾ 9.910-59.910	H - 0.00334 O - 0.00167 Be - 0.1174	H - 0.00559 O - 0.0444 Be - 1.757

(1) Reflector contains 5% H₂O.

FIG. 1 KING GEOMETRY

TABLE I

EFFECT OF VOIDS ON REACTIVITY IN KING-TYPE REACTOR

$\bar{\rho}(H)$	$\Delta k^{(1)}$	$\delta k = k_L - k^{(1)}$	$\delta k/k^{(1)}$
0.6	0.0209	0.00173	0.002
0.7	0.0139	0.000764	0.0008
0.8	0.0084	0.000274	0.0003

H = 71 cm

Table 2
Energy Balance in a King Reactor
MeV/fission

Source	Energy Deposition In							Other
	Island	Zr Tube	Fuel	Zr Tube	Reflector	Leakage	Total	
Fission Product Kinetic Energy	-	-	165.0000	-	-	-	165.0000	-
Neutron Kinetic Energy	.7574	.0038	1.6582	.0117	1.6953	.5672	4.87	.1765
Prompt Gamma	.4000	.494	1.338	1.233	2.923	.812	7.2	-
Delay Gamma	-	-	-	-	-	-	-	-
Delay Beta	-	-	-	-	-	-	-	-
U Capture Gamma	.0786	.0912	.2402	.2313	.5769	.1329	1.3512	-
Zr Capture Gamma	.0074	.0096	.0152	.0304	.0799	.0138	.1563	-
H Capture Gamma	.0938	.0540	.0823	.0874	.6595	.0695	1.0466	-
Be Capture Gamma	.0075	.0089	.0165	.0385	1.0307	.4382	1.5404	-
Total	1.3447	.6615	168.3504	1.6323	6.9653	2.0336	181.1645	-

Table 3
Energy Balance in a KING Reactor
MW for 25 MW in the Fuel Region

Source	Energy Deposition In							Other
	Island	Zr Tube	Fuel	Zr Tube	Reflector	Leakage	Total	
Fission Product Kinetic Energy	-	-	24.5025	-	-	-	24.5025	-
Neutron Kinetic Energy	.1125	.0006	.2462	.0017	.2518	.0842	.7232	.0262
Prompt Gamma	.0594	.0734	.1987	.1831	.4341	.1206	1.0692	-
U Capture Gamma	.0117	.0135	.0357	.0343	.0857	.0197	.2007	-
Zr Capture Gamma	.0011	.0014	.0023	.0045	.0119	.0020	.0232	-
H Capture Gamma	.0139	.0080	.0122	.0130	.0979	.0103	.1554	-
Be Capture Gamma	.0011	.0013	.0025	.0057	.1531	.0651	.2287	-
Total	.1997	.0982	25.0000	.2423	1.0345	.3019	26.9028	-

Table 4

Gamma Radiation Heating-KING
Axial Average Values for 25 MW Total
Deposition in the Fuel Region - watts/g

		All Sources Included							watts/g	
Radius-cm	Be	C*	H	O	S	U	Zr	Total		
0.0 - .5		17.32	33.83	17.49				19.32		
- 1.0		17.36	33.88	17.52				19.35		
- 1.5		17.42	33.99	17.57				19.41		
- 2.0		17.49	34.15	17.66				19.51		
- 2.5		17.61	34.37	17.76				19.62		
- 3.0		17.74	34.63	17.89				19.76		
- 3.5		17.90	34.98	18.07				19.96		
- 4.0		18.14	35.44	18.30				20.22		
- 4.5		18.48	36.10	18.64				20.59		
- 5.0		19.02	37.16	19.62				21.58		
- 5.16		19.70					26.33	26.33		
- 5.32		20.75					25.97	25.97		
- 5.82		22.55	44.09	22.71	22.31	89.56		35.46		
- 6.32		24.04	47.03	24.22	23.78	92.58		37.34		
- 6.82		24.40	47.74	24.59	24.14	92.81		37.71		
- 7.32		23.93	46.81	24.11	23.66	90.73		36.93		
- 7.82		22.67	44.34	22.84	22.42	86.46		35.07		
- 8.32		20.55	40.18	20.70	20.33	79.96		32.05		
- 8.638		16.95					20.81	20.81		
- 8.955		13.06					18.01	18.01		
- 9.433		11.25					-	-		
- 9.910		10.68					-	-		

*This is the heating that would be experienced by an infinitesimal sample of carbon at the specified locations.

Table 5

Neutron Radiation Heating - KING

Axial Average Values for 25 MW Total

Deposition in the Fuel Region - watts/g

Radius-cm	Be	C*	H	O	S	U	Zr	watts/g Total
0.0		2.39	158.22	1.47				19.01
- .5		2.41	159.56	1.49				19.18
- 1.0		2.46	163.16	1.52				19.61
- 1.5		2.54	168.70	1.57				20.27
- 2.0		2.66	176.33	1.64				21.19
- 2.5		2.81	186.44	1.74				22.41
- 3.0		3.02	199.74	1.87				24.01
- 3.5		3.29	217.68	2.04				26.17
- 4.0		3.69	243.35	2.28				29.26
- 4.5		4.32	284.19	2.68				34.18
- 5.0		4.85					.146	.15
- 5.16		5.12					.154	.15
- 5.82		5.60	366.39	3.48	.98	.025		33.16
- 6.09		6.09	397.38	3.78	1.06	.027		35.97
- 6.82		6.26	408.28	3.89	1.09	.028		36.96
- 7.32		6.19	404.41	3.84	1.08	.027		36.60
- 7.82		5.89	386.41	3.65	1.03	.026		34.95
- 8.32		5.33	351.61	3.29	.94	.024		31.78
- 8.638		4.72					.147	.15
- 8.955		4.26					.134	.13
- 9.433		3.95					-	-
- 9.910		3.81					-	-
9.910	5.45	3.48	237.78	2.11				6.09
11.93	3.20	1.99	140.57	1.18				3.58
13.44	2.25	1.38	98.89	.81				2.51
15.96	1.29	.79	57.19	.46				1.44
18.49	.76	.46	33.89	.267				.86
22.0	.359	.216	15.95	.124				.40
27.0	.135	.081	5.99	.046				.151
32.0	.052	.031	2.30	.018				.058
42.0	.008	.0048	.355	.0069				.009
51.9	.0013	.0008	.056	.0004				.0014

*This is the heating that would be experienced by an infinitesimal sample of carbon at the specified locations.

Table 6
Total Radiation Heating - KING
(Normalized to 25 MW in the Fuel)

Increment cm	Average Heating watts/cm ³ y	n	y+n	kW in the for 55 cm y	Radial Height n	Increment y+n
0.0-0.5	19.2	19.0	38.2	.83	.82	1.65
- 1.0	19.4	19.1	38.5	2.51	2.48	4.99
- 1.5	19.4	19.6	39.0	4.19	4.23	8.42
- 2.0	19.5	20.3	39.8	5.90	6.13	12.03
- 2.5	19.6	21.2	40.8	7.63	8.24	15.87
- 3.0	19.8	22.4	42.2	9.39	10.65	20.04
- 3.5	20.0	24.0	44.0	11.21	13.48	24.69
- 4.0	20.2	26.2	46.4	13.10	16.96	30.06
- 4.5	20.6	29.2	49.8	15.12	21.49	36.61
- 5.0	21.2	34.2	55.4	17.39	28.05	45.44
- 5.16	172.5	1.0	173.5	48.44	.27	48.71
- 5.32	170.1	1.0	171.1	49.28	.29	49.57
- 5.82	35.4	33.2	68.6	34.06	31.92	65.98
- 6.32	37.3	36.0	73.3	39.08	37.73	76.81
- 6.82	37.3	37.0	74.3	42.33	41.96	84.29
- 7.32	36.8	36.6	73.4	45.01	44.71	89.72
- 7.82	35.0	35.0	70.0	45.77	45.73	91.50
- 8.32	32.0	31.8	63.8	44.59	44.32	88.91
- 8.64	136.3	1.0	137.3	126.83	.89	127.72
- 8.96	118.0	.9	118.9	113.88	.85	114.73
- 9.43	-	-	-	-	-	-
- 9.91	-	-	-	-	-	-
-10.41	15.9	11.0	26.9	28.13	19.47	47.60
-10.92	14.3	9.6	23.9	26.58	17.78	44.36
-11.42	12.9	8.3	21.2	25.12	16.23	41.35
-11.93	11.8	6.8	18.6	23.92	13.93	37.85
-12.43	10.7	6.5	17.2	22.75	13.72	36.47
-12.94	9.85	5.73	15.58	21.78	12.68	34.46
-13.44	9.05	5.08	14.13	20.82	11.69	32.51
-13.95	8.38	4.54	12.92	20.01	10.84	30.85
-14.45	7.75	4.04	11.79	19.19	10.01	29.20
-14.96	7.22	3.62	10.84	18.51	9.29	27.80
-15.46	6.71	3.24	9.95	17.80	8.59	26.39
-15.96	6.28	2.91	9.19	17.21	7.98	25.19
-16.47	5.87	2.61	8.48	16.59	7.38	23.97
-16.97	5.51	2.35	7.86	16.07	6.85	22.92
-17.48	5.17	2.11	7.28	15.53	6.34	21.87
-17.98	4.88	1.90	6.78	15.07	5.88	20.95

Table 6 (Continued)

Radial Increment cm	Average Heating watts/cm ³			kW in the Radial Increment for 55 cm Height		
	γ	n	$\gamma+n$	γ	n	$\gamma+n$
0.0-18.49	4.59	1.71	6.30	14.58	5.43	20.02
-18.99	4.34	1.55	5.89	14.17	5.05	19.22
-19.50	4.09	1.39	5.48	13.73	4.67	18.40
-20.00	3.88	1.26	5.14	13.35	4.33	17.68
-21.00	3.58	1.08	4.66	25.31	7.65	32.96
-22.00	3.22	.89	4.11	23.90	6.56	30.46
-23.0	2.92	.72	3.64	22.64	5.62	28.26
-24.0	2.65	.60	3.25	21.43	4.82	26.25
-25.0	2.403	.488	2.891	20.29	4.12	24.41
-26.0	2.184	.401	2.585	19.19	3.52	22.71
-27.0	1.990	.330	2.319	18.17	3.01	21.18
-28.0	1.812	.272	2.084	17.17	2.58	19.75
-29.0	1.654	.224	1.878	16.24	2.20	18.44
-30.0	1.508	.185	1.693	15.33	1.88	17.21
-31.0	1.381	.152	1.533	14.51	1.60	16.11
-32.0	1.263	.126	1.389	13.71	1.37	15.08
-33.0	1.156	.104	1.260	12.94	1.17	14.11
-34.0	1.058	.087	1.145	12.21	1.00	13.21
-35.0	.970	.072	1.042	11.53	.85	12.38
-36.0	.889	.059	.948	10.87	.725	11.59
-37.0	.814	.049	.863	10.24	.618	10.86
-38.0	.746	.041	.787	9.63	.526	10.16
-39.0	.684	.034	.718	9.07	.449	9.52
-40.0	.627	.028	.655	8.53	.382	8.91
-41.0	.575	.023	.598	8.02	.326	8.35
-42.0	.526	.019	.545	7.52	.277	7.80
-42.9	.482	.016	.498	7.06	.236	7.30
-43.9	.442	.013	.455	6.62	.201	6.82
-44.9	.404	.011	.415	6.18	.170	6.37
-45.9	.370	.0093	.379	5.79	.145	5.93
-46.9	.338	.0077	.346	5.41	.124	5.53
-47.9	.308	.0064	.315	5.04	.105	5.15
-48.9	.281	.0053	.286	4.69	.089	4.78
-49.9	.256	.0045	.260	4.36	.076	4.44
-50.9	.232	.0037	.236	4.04	.064	4.10
-51.9	.211	.0031	.214	3.74	.055	3.80
-52.9	.191	.0025	.193	3.45	.046	3.50
-53.9	.172	.0021	.174	3.17	.039	3.21
-54.9	.155	.0018	.157	2.91	.033	2.94
-55.9	.139	.0014	.140	2.65	.027	2.68
-56.9	.123	.0011	.125	2.40	.022	2.42
-57.9	.109	.0009	.110	2.15	.018	2.17
-58.9	.095	.0007	.096	1.91	.014	1.92
-59.91	.081	.0005	.082	1.65	.010	1.66

Table 7

Beta and Gamma Sources in King Fuel Following
 3×10^6 seconds (34.7 days) at 25 MW.

Time after shutdown seconds		Beta kw	Gamma kw	Beta curies	Gamma curies
1.0		962	868	1.62/8	1.46/8
2.15		871	823	1.47/8	1.39/8
4.64		764	763	1.29/8	1.29/8
10.0		656	690	1.11/8	1.16/8
21.5		553	609	9.33/7	1.03/8
46.4		463	525	7.81/7	8.86/7
* 1.0/2	1.7 min	384	444	6.48/7	7.49/7
2.15/2	3.6 min	315	368	5.31/7	6.21/7
4.64/2	7.7 min	256	298	4.32/7	5.03/7
1.0/3	16.7 min	206	236	3.48/7	3.98/7
2.15/3	35.8 min	161	184	2.72/7	3.10/7
4.64/3	1.3 hours	123	142	2.08/7	2.40/7
1.0/4	2.8 hours	91.4	108	1.54/7	1.82/7
2.15/4	5.97 hours	66.2	81.6	1.12/7	1.38/7
4.64/4	12.9 hours	46.5	60.4	7.85/6	1.02/7
1.0/5	1.16 days	31.3	44.6	5.28/6	7.52/6
2.15/5	2.5 days	21.3	33.1	3.59/6	5.58/6
4.64/5	5.4 days	14.6	24.0	2.46/6	4.05/6
1.0/6	11.6 days	9.74	15.9	1.64/6	2.68/6
2.15/6	24.9 days	6.03	8.59	1.02/6	1.45/6
4.64/6	53.7 days	3.28	3.77	5.53/5	6.36/5

* Read 1.0/2 as 1.0×10^2

Table 8

Beta and Gamma Sources in the King Fuel Following
 1×10^7 seconds (115.7 days) at 25 MW

Time after shutdown seconds		Beta kw	Gamma kw	Beta curies	Gamma curies
1.0		970	878	1.64/8	1.48/8
2.15		879	832	1.48/8	1.40/8
4.64		772	773	1.30/8	1.30/8
10.0		664	700	1.12/8	1.18/8
21.5		561	619	9.46/7	1.04/8
46.4		471	535	7.95/7	9.03/7
* 1.0/2	1.7 min	392	454	6.61/7	7.66/7
2.15/2	3.6 min	323	377	5.45/7	6.36/7
4.64/2	7.7 min	280	308	4.45/7	5.20/7
1.0/3	16.7 min	214	246	3.61/7	4.15/7
2.15/3	35.8 min	169	194	2.85/7	3.27/7
4.64/3	1.3 hours	131	151	2.21/7	2.55/7
1.0/4	2.8 hours	99.4	118	1.68/7	1.99/7
2.15/4	5.97 hours	74.1	91.1	1.25/7	1.54/7
4.64/4	12.9 hours	54.3	69.9	9.16/6	1.18/7
1.0/5	1.16 days	39.1	53.9	6.60/6	9.09/6
2.15/5	2.5 days	28.9	42.2	4.88/6	7.12/6
4.64/5	5.4 days	21.9	32.5	3.69/6	5.48/6
1.0/6	11.6 days	16.3	23.4	2.75/6	3.95/6
2.15/6	24.9 days	11.5	14.5	1.94/6	2.45/6
4.64/6	53.7 days	7.24	7.75	1.22/6	1.31/6

* Read 1.0/2 as 1.0×10^2

Table 9

Beta and Gamma Sources in the King Fuel Following
 3×10^7 seconds (347 days) at 25 MW

Time after shutdown seconds		Beta kw	Gamma kw	Beta curies	Gamma curies
1.0		976	882	1.65/8	1.49/8
2.15		885	837	1.49/8	1.41/8
4.64		778	777	1.31/8	1.31/8
10.0		670	704	1.13/8	1.19/8
21.5		567	624	9.57/7	1.05/8
46.4		477	540	8.05/7	9.11/7
* 1.0/2	1.7 min	398	458	6.71/7	7.73/7
2.15/2	3.6 min	329	382	5.55/7	6.44/7
4.64/2	7.7 min	270	312	4.56/7	5.26/7
1.0/3	16.7 min	220	251	3.71/7	4.23/7
2.15/3	35.8 min	175	199	2.95/7	3.36/7
4.64/3	1.3 hours	137	156	2.31/7	2.63/7
1.0/4	2.8 hours	105	122	1.77/7	2.06/7
2.15/4	5.97 hours	80.2	95.9	1.35/7	1.62/7
4.64/4	12.9 hours	60.4	74.7	1.02/7	1.26/7
1.0/5	1.16 days	45.1	58.7	7.61/6	9.90/6
2.15/5	2.5 days	34.8	46.8	5.87/6	7.90/6
4.64/5	5.4 days	27.7	37.0	4.67/6	6.24/6
1.0/6	11.6 days	21.9	27.7	3.69/6	4.67/6
2.15/6	24.9 days	16.7	18.2	2.82/6	3.07/6
4.64/6	53.7 days	11.7	10.6	1.97/6	1.79/6

* Read 1.0/2 as 1.0×10^2

Fig. D-2. Gamma and neutron activity in the KING reactor fuel.

APPENDIX E KING REACTOR HEAT EXCHANGER

Summary

A parametric study of optimum heat exchanger designs was made for tube diameters of 3/8, 1/2, and 5/8 in., and tube spacings (o.d. to o.d.) from zero to 0.5 in. Studies were made for fuel on the tube side and on the shell side, for single counter-flow tube pass and for double tube pass. The effect of variations in the tube wall thickness and fouling resistance was also looked at.

Discussion

The designs studied were determined on the basis of minimum cost for given tube diameter, spacing, flows, temperatures, and number of passes. The cost factors used in this determination were:

- a) Fuel solution cost = \$2200/liter, fuel (220 g U/liter fuel concentration)
- b) Pumping power cost = \$1600/kW (\$200/kW for pumps, \$1400/kW for 20 years continuous operation)
- c) HE hardware cost = \$300/liter, tube side (This is extremely approximate, but is not a large factor in the overall cost).

Some of these factors may not be important in the initial proposal cost, but seemingly should be taken into account in the heat exchanger design. Other factors which affect the design selection such as fabricability, servicing, reliability, etc., are obviously not factored into these studies, except in the selection of tube diameters.

A limitation which was considered is the critical tube spacing from neutronic considerations. Figure 1 shows the minimum allowable tube spacing from data of D. R. Smith, N-2.

Calculation Methods

The equations used in these heat exchanger calculations are outlined below. Many simplifications have been made for these preliminary calculations. In particular, mean fluid and tube wall properties were used throughout, and the heat transfer coefficients and pressure drop equations were reduced to:

$$h = K_H G^{0.8} / D_H^{0.2}$$

where

$$K_H = 0.023 k \text{ Pr}^{0.4} / \mu^{0.8} = \text{constant}$$

$$\Delta p = K_P G^{1.8} \ell / D_H^{1.2}$$

where

$$K_P = (f \text{Re}^{0.2}) \mu^{0.2} / 2\rho = \text{constant}$$

$$(f \text{Re}^{0.2} \approx 0.19 \text{ for smooth tubes})$$

The properties (including K's) of fuel solution and water are shown in Table 1.

Symbols

- A_S = tube o.d. surface area
- A_S^* = dimensionless $A_S : A_S / (R \dot{m}_1 \bar{C}_{pl})$, a function of fluid inlet, outlet temperatures.
- d_i = tube i.d.
- d_o = tube o.d.
- D_H = hydraulic diameter
- f = friction factor ($4 \tau_w / 1/2 \rho v^2$)
- G = fluid mass flux ($\dot{m} / A_F$)
- h = convective HTC
- k = thermal conductivity (fluid)
- k_w = thermal conductivity (tube wall material)
- L = total tube length - all passes
- $\dot{m}$ = fluid mass flow rate
- N = number of tube passes
- n = number of tubes
- Δp = fluid pressure drop
- P = pump power (input) $\div \dot{m}_1$
- P_r = Prandtl number
- R = total heat transfer resistance
- r_c = fouling resistance
- s = spacing between tube o.d.'s
- T = fluid temperature
- v = fluid volume in HE $\div \dot{m}_1$

μ = viscosity

η = efficiency

ρ = density

σ = tube spacing function = $\frac{2\sqrt{3}}{\pi} (1 + \frac{S}{d_o})^2 - 1$

Subscripts

1 - tube side

2 - shell side

F - fuel

w - tube wall

Dimensionless surface area, A_s^*

$$A_s^* \equiv \frac{A_s}{R \dot{m}_1 \bar{C}_{p1}}$$

let $T_{1,in}$ = tube side inlet temperature.

$T_{1,out}$ = tube side outlet temperature.

$T_{2,in}$ = shell side inlet temperature.

$T_{2,out}$ = shell side outlet temperature.

(1) Single pass counterflow:

$$A_s^* = \frac{\phi}{\phi - 1} \ln \left(\frac{T_{1,in} - T_{2,out}}{T_{1,out} - T_{2,in}} \right)$$

(2) Double tube pass (recommended for $N \geq 2$)

$$A_s^* = \frac{\phi}{\sqrt{\phi^2 + 1}} \ln \left(\frac{1 + \sqrt{1 + \phi^2}}{1 - \sqrt{1 + \phi^2}} \right)$$

$$\text{where } \phi = \frac{\dot{m}_2 \bar{C}_{p2}}{\dot{m}_1 \bar{C}_{p1}} = \frac{T_{1,in} - T_{1,out}}{T_{2,out} - T_{2,in}}$$

$$r = \frac{T_{1,out} - T_{2,in} + T_{1,in} - T_{2,out}}{T_{2,out} - T_{2,in}}$$

Specific Fluid Volume:

V = fluid volume ÷ tube side flow ($\dot{m}_1$)

$$V_1 = \frac{n}{\dot{m}_1} L \frac{\pi}{4} d_i^2 = L/G_1$$

$$V_2 = \frac{n}{\dot{m}_1} L \frac{\pi}{4} d_o^2 \sigma = V_1 * \left(\frac{d_o}{d_i} \right)^2 \sigma$$

Tube o.d. Surface Area:

$$A_s / \dot{m}_1 = \frac{n}{\dot{m}_1} L \pi d_o = A_s^* \bar{C}_{p1} R$$

$$\text{then } V_1 = B_1 R \quad \text{where } B_1 \equiv \frac{A_s^* \bar{C}_{p1} d_i^2}{4 d_o}$$

Heat Transfer Resistance:

$$R = \frac{d_o}{d_i} \frac{1}{h_1} + \frac{1}{h_2} + r_w$$

$$\text{where } r_w = \frac{d_o}{2k_w} \ln \frac{d_o}{d_i} + r_c$$

$$\text{assume } h_1 = K_{H1} G_1^{0.8} / d_i^{0.2},$$

$$\text{and } h_2 = K_{H2} G_2^{0.8} / D_{H2}^{0.2} \quad \text{where } D_{H2} = d_o \sigma.$$

R can be rewritten as:

$$R = B_2 G_1^{-0.8} + r_w$$

$$\text{where } B_2 = \left(\frac{d_o}{d_i} \right) d_i^{0.2} \left(\frac{1}{K_{H1}} + \frac{1}{K_{H2}} H^{0.8} \sigma \right)$$

$$\text{and } H \equiv N \left(\frac{\dot{m}_1}{\dot{m}_2} \right) \frac{d_o}{d_i}$$

$$\text{then } V_1 = B_1 (B_2 G_1^{-0.8} + r_w)$$

Total Pumping Power:

$$(P \equiv \text{power} \div \dot{m}_1)$$

$$P = \frac{\Delta P_1}{\rho_1 \eta_1} + \frac{\dot{m}_2}{\dot{m}_1} \frac{\Delta P_2}{\rho_2 \eta_2}$$

$$\text{assume } \Delta P_1 = K_{p1} G_1^{1.8} L / d_i^{1.2},$$

$$\text{and } \Delta P_2 = K_{p2} G_2^{1.8} L / N / D_{H2}^{1.2}$$

(i.e., $f \propto Re^{-0.2}$)

Then P can be expressed as:

$$P = B_1 B_3 (B_2 G_1^2 + r_w G_1^{2.8})$$

where

$$B_3 = d_i^{-1.2} \left[\frac{K_{p1}}{\rho_1 \eta_1} + \frac{K_{p2}}{\rho_2 \eta_2} \left(\frac{d_i}{d_o} \right)^2 H^{-2.8} \sigma^{-3} \right]$$

Minimum Cost Criterion

The "total" cost (divided by $\dot{m}_1$) is assumed to be made of the cost of the fuel solution in the heat exchanger, the hardware cost and the pumping cost (including pumps) over some period of time.

$$C = \beta_v V_f + \beta_{HE} V_1 + \beta_p P$$

β_v = fuel solution cost/liter, fuel

β_{HE} = hardware cost/liter, tube side fluid (this is a function of d_o/d_i , s/d_o , approximately)

β_p = pumping cost/kW

For given flow ratio, tube size and spacing, and fluid temperatures, then, the "optimum" design can be found by:

$$\frac{\partial C}{\partial G_1} = 0$$

Define a parameter $E = V_f/V_1$ such that $E = 1$ for fuel on the tube side, and

$$E = V_2/V_1 = \left(\frac{d_o}{d_i}\right)^2 \quad \sigma \text{ for fuel on the shell side. Then}$$

side. Then

$$\beta_v V_f + \beta_{HE} V_1 = (\beta_v E + \beta_{HE}) V_1 \equiv \alpha_v V_1$$

The resulting implicit relation for G_1 is:

$$G_1^{2.8} (B_2 + 1.4 r_w G_1^{0.8}) = \frac{0.4 \alpha_v B_2}{\beta_p B_3}$$

The typical variation of cost factor, tube length, and number tubes with G_1 around the optimum point is shown in Fig. 2.

Parametric Studies

Most calculations were made using a tube wall thickness of 0.045 in. (0.1143 cm) and a "fouling factor" of 300 in.²-°R-sec/Btu (102 m²-K/MW).

Other imposed parameters were:

Fuel: Flow = 526 lbm/sec (239 kg/sec)

$T_{in} = 80^\circ\text{C}$ (353.2 K)

$T_{out} = 50^\circ\text{C}$ (323.2 K)

Water: Flow = 852 lbm/sec (387 kg/sec)

$T_{in} = 37.8^\circ\text{C}$ (311.0 K)

$T_{out} = 51.7^\circ\text{C}$ (324.9 K)

Results of the studies are shown in Figs. 3-7 and Table 2.

The design chosen to be submitted for a cost estimate is summarized below. Due to length limitation this design is a "doubled-up" type with a single shell and a baffle plate down the middle, but is still a single pass counterflow type flow scheme.

Tubes: 0.5 in. (1.27 cm) o.d., 0.045 in. (0.1143 cm) wall thickness

610 tubes, total length 55+ ft (17 m)

0.75 in. (1.905 cm) center-to-center spacing in triangular array. Carpenter 20C6-3-Cb-3

Shell: Inside length = 28 ft (8.5 m), 28 in. (71 cm) i.d., 304L stainless

Fuel on tube side, single pass.

A preliminary sketch of this heat exchanger is shown in Fig. 8.

Other characteristics of this design are:

Fuel pump power = 141 HP (105 kW)

Fuel side pressure = 40 psi (2.76 bars)

Water pump power = 90 HP (67 kW)

Water pressure drop = 12 psi (0.83 bar)

Heat transfer area = 4050+ ft² (376 m²)

Fuel velocity = 12 ft/sec (3.7 m/sec)

Water velocity = 11 ft/sec (3.4 m/sec)

Overall heat transfer resistance = 95- in.²-sec-°R/Btu (320 $\frac{\text{m}^2\text{-K}}{\text{MW}}$).

TABLE 1

Assumed Properties of a $\text{UO}_2\text{SO}_4/\text{H}_2\text{O}$ Solution

(a) with 220 g U/liter

T (°C)	ρ (lbm/in. ³)	$\mu \times 10^5$ (lbm/in.-sec)	$k^* \times 10^5$ $\frac{\text{Btu/sec}}{\text{in.-}^\circ\text{R}}$	C_p (Btu/lbm-°R)	$K_H \times 10^3$	$K_P \times 10^3$
30	0.0462	8.34	0.820	0.75	.779	.814
45	0.0461	6.22	0.848		.894	.767
50	0.0460	5.71	0.855		.930	.758
60	0.0458	4.82	0.870		1.005	.736
65	0.0457	4.43	0.876		1.044	.725
75	0.0455	3.75	0.888		1.125	.705
80	0.0454	3.50	0.893		1.160	.696
90	0.0452	3.04	0.902		1.235	.680
100	0.0449	2.67	0.910		1.308	.667
125	0.0442	2.03	0.916		1.465	.641

*Values for water.

(b) Water

T (°C)	$k \times 10^5$ $\frac{\text{Btu/sec}}{\text{in.-}^\circ\text{R}}$	$\mu \times 10^5$ $\frac{\text{lbm}}{\text{in.-sec}}$	P_r	ρ $\frac{\text{lbm}}{\text{in.}^3}$	$K_H \times 10^3$	$K_P \times 10^3$
0	.738	10.04	13.6	.03612	.762	.1044
20	.799	5.64	7.02	.03605	.1004	.933
40	.840	3.66	4.34	.03584	.1231	.860
60	.870	2.64	3.02	.03552	.1432	.813
80	.894	1.986	2.22	.03510	.1634	.777
100	.910	1.579	1.74	.03462	.1813	.752
120	.917	1.308	1.446	.03406	.1972	.738
140	.914	1.111	1.241	.03345	.2107	.727

TABLE 2

EFFECT OF VARIATIONS IN TUBE WALL THICKNESS
AND FOULING FACTOR

FUEL ON TUBE SIDE, SINGLE PASS

a) Tube thickness (Fouling Factor = $300 \text{ in.}^2\text{-sec-}^\circ\text{R/Btu}$)

d_o (in.)	Wall Thickness (in.)	N	V_f (liters)	L (ft)	P (HP)	C (M\$)
3/8	.025	791	749	58.0	305	2.16
"	.045	1055	720	54.4	315	2.28
1/2	.025	392	912	74.5	310	2.50
"	.045	484	916	72.9	314	2.66
"	.065	606	895	69.9	321	2.80

b) Fouling Factor (W.T. = .045 in.)

d_o (in.)	r_c ($\text{in.}^2\text{-sec-}^\circ\text{R/Btu}$)	N	V_f (liters)	L (ft)	P (HP)	C (M\$)
3/8	0	955	523	43.7	307	1.75
"	300	1055	720	54.4	315	2.28
"	1000	1219	1154	75.5	334	3.45

Fig. E-1. Minimum allowable tube spacing.

TYPICAL VARIATION WITH G_1 FOR
 $d_i = 0.41$, $d_o = 0.50$, $s = 0.25$, $r_c = 300$, $\dot{m}_1/\dot{m}_2 = 0.617$
 $T_1 = 80 \rightarrow 50^\circ\text{C}$, $T_2 = 37.8 \rightarrow 51.7^\circ\text{C}$, $N = 1$
 (FUEL ON TUBE SIDE)
 $a_v = 2500$ \$/LITER
 $\beta_p = 1600$ \$/kw

Fig. E-2. Typical variation of cost factor, tube length, and number of tubes with optimum fluid mass flux on tube side.

Fig. E-3. Parametric studies of heat-exchanger design.

Fig. E-5. Parametric studies of heat-exchanger design.

Fig. E-6. Parametric studies of heat-exchanger design.

Fig. E-6. Parametric studies of heat-exchanger design.

Fig. E-7. Parametric studies of heat-exchanger design.

Fig. E-8. Preliminary design of KING reactor heat exchanger.