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AEC RESEARCH AND
DEVELOPMENT REPORT

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FOR FUTURE APPLICATION
CONVERSION DEVICES M-3679
(63rd Ed.)

TECHNICAL INFORMATION IN SUPPORT OF
THERMIONIC REACTOR EXPERIMENT FACILITY
COST ESTIMATE

MASTER

FEBRUARY 1970

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Nuclear Systems Programs
Space Systems
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NUCLEAR THERMIONIC POWER OPERATION
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I. INTRODUCTION AND SUMMARY

A. INTRODUCTION

This document records the technical description and accompanying backup data used to generate the Thermionic Reactor Experiment cost estimate which was submitted to the U.S. Atomic Energy Commission in November 1969. This work was performed under U.S. Atomic Energy Commission Contract AT(04-3)-771, Construction, Planning and Design (39-04-66-10). The work was carried out by the Nuclear Thermionic Power Operation of the General Electric Company.

The principal focus of activity was to estimate the Plant Acquisition construction cost for the Thermionic Reactor Experiment Facility. The thermionic reactor concept is in an evolutionary phase wherein the exact nature of the thermionic fuel element and the reference design of the reactor are not well established. In fact, the Thermionic Reactor Experiment (Trex) itself is intended to serve as a tool in further defining both the TFE and useful reactor configurations. Because of this evolving nature, it is important to be as flexible as possible in the approach to the design, construction and operation of Trex. At the same time, to define a realistic cost estimate specific hardware must be chosen. In this study, these specific choices for hardware were made for costing purposes, but the overlying philosophy of the approach to Trex continues to be one of flexibility.

A number of guidelines were specified for the Thermionic Reactor Experiment Facility cost study. These ground rules included: reactor power level at least 100 kWe, U^{235} fueled thermionic fuel elements, a fast spectrum and a reflector which mocks up a 4π shieldable flight reference design. Additional design objectives were imposed: the Trex must be doable, that is readily fabricated making the maximum use of technology and components; the total system must be safe to operate in the manner planned and the operation straight-forward and efficient.

The Trex concept presented is designed to meet these objectives. The features enumerated below are examples of how this has been done. The liquid metal coolant system employs components developed and in use such as

pumps, flowmeters and heaters. The instrumentation and control is, in the main, standard off-the shelf equipment or of a design where practicability has been shown. The helium gas cooling of the fission product trap and cesium reservoir in the thermionic fuel element handle is a practical way to meet the heat transfer requirement without the problems of high stress concentrations produced in the vessel internal structure by the use of low temperature liquid metal. The reflector control cup may be maintained in place or replaced with reasonable ease. The thermionic fuel element electrical connection configuration can be made reliable and easy-to-operate while maintaining containment integrity and inert cover gas purity. The fuel handling machine provides an efficient, safe method for fuel element exchange by employing an inert gas atmosphere control and multiple storage positions.

Safety considerations were an integral part of the design process. For example, the auxiliary cooling system is an accepted, workable way to accommodate core afterheat if the prime system fails. The reflector control cup assembly which fits over the core end of the reactor vessel may be scrambled or lowered for a substantial shutdown margin (\$18) during refueling operations. The reactor vessel has a double wall to a point well above the core to assure coolant coverage of the core. The control philosophy of the TREX essentially separates the several major plant systems so they may be operated individually, thus reactor operations are simplified and made safer. Safety is enhanced by limiting the lowering of the fuel handling machine to a position adjacent to the reactor (rather than over it) and rolling it into position over the reactor.

This report is organized so the concept is described beginning with the thermionic fuel element and in turn discusses the core structure, reactor vessel and dome, reflector cup and lift and reactor design. The TREX control philosophy, power conditioning equipment and instrumentation and control are then discussed. Following these are sections describing coolants and a facility description and proposed modification, safeguards and schedule.

B. SUMMARY

The following paragraphs summarize the principal features of the Thermionic Reactor Experiment, - reactor, equipment, facility.

1. Reactor and Vessel

a. Internal Configuration

The reactor core consists of 163 - 12 cell thermionic fuel elements (TFE) with an outer row of 48 beryllium/oxide internal reflector rods. The use of high worth driver elements in the center of the core, longer cells, TFEs with a larger number of cells, or other methods may be required to adjust core reactivity to meet operational requirements. The core arrangement is enclosed by a shroud which is mounted from the top inside the reactor vessel.

b. Core and Shroud Structure

The initial core structure consists of a bundle of circular cross section process tubes which are restrained at the top and bottom by grid plates and extend through the reactor vessel top plate. The TFE and its handle are inserted into and held in position in the core by these tubes. The process tube cluster is spread slightly at the top end of the shroud to provide space for the electrical and mechanical connection at the top end of the process tube and TFE. The use of process tubes improves TFE coolant flow characteristics and minimizes sheath circumferential temperature differences. Flow orifices are located on the bottom end of the process tubes and may be readily changed since the process tubes are removable. Positive core support and spacing is provided by the process tubes and establishes a reliable means for installation and removal of the core elements. The process tubes have a wall thickness of approximately 40 mils and are

tapered at the core end so they are close fitting yet provide the small spread required at the top. The bottom of the core process tube has a wall thickness of 20 mils with a resultant web thickness of 40 mils. Spacers in the process tube wall provide a uniform coolant annulus of 40 mils between the TFE and tube. Grids at the top and bottom of the shroud space support the process tubes. The shroud provides a radial restriction and support in the core region. The shroud assembly, which is mounted to the top flange of the reactor vessel as a unit, may be unbolted and the entire core assembly removed.

c. Thermionic Fuel Element

The thermionic fuel element planned to be used for the Thermionic Reactor Experiment (TREX) is similar to that proposed for previous 300 and 100 kWe powerplants for space application. It is also similar to the design used in the 3-cell TFEs tested in the General Electric Test Reactor and makes maximum use of developed technology. These TFEs are 1.01 inches in diameter with an outer sheath material of Ceramvar. In the TREX concept, there are twelve (12) thermionic cells inside the sheath which are oriented such that a pair of TFEs may be connected in series. Beryllium oxide has been added to the lower end of the TFE. This provides a bottom reflector where the TFE-reflector distance change is minimized as a function of thermal expansion. The top end of the TFE is reduced in diameter to facilitate coolant cross flow. A cylindrical tube (handle) is attached to the top of the TFE and extends to the top of the shroud and reactor vessel. Inside the handle is located the cesium reservoir and fission product trap.

Fission products are vented from each emitter into the interelectrode space and ported from the TFE through a tube to a 3 foot long charcoal sorbent trap located above the cesium reservoir.

TFE electrical power is transmitted from the point of connection at the top end of the TFE through a molybdenum conductor, a mineral insulated cable and subsequently above the fission trap to polyvinyl chloride insulated cable to the connector at the top end of the TFE handle. In the region above the fission product trap, the electrical conductor is spiraled to reduce the radiation streaming from the fission product trap. In addition to the electrical power cable there are also thermocouple, direct potential and cesium reservoir heater power leads carried through to the connector at the top end of the TFE handle.

The TFE connector provides a reliable, easy to make and break electrical and mechanical connection for the TFE. The connector arm to which the TFE connections are made slides into a slot in the TFE handle. Actuation of the connector may be done reliably without remote handling tools or interruption of cover gas containment. Further, the TFE or process tube and TFE may be removed with a simple 60° rotation of the fuel element transfer machine grapple inside the detent.

d. Non-Thermionic Fuel Element

It is planned that the first TREX core would consist of non-thermionic fuel elements (NTFEs) which match the TFE in reactivity. The use of NTFEs which are not nuclearly equivalent to TFEs could impose undesirable safety or operational restrictions. These NTFEs would be replaced in a planned sequence with TFEs until a full core was loaded. This stepped exchange is planned to take advantage of the better known characteristics of the NTFE core and to provide an operational fast reactor in which the first TFEs may be placed for evaluation.

The non-thermionic fuel element is uranium dioxide inside a stainless steel sheath of a diameter identical to that for the TFE. The enrichment of the uranium dioxide will be approximately 46%

to match the reactivity of the thermionic fuel element. Thermocouples located inside the NTFE to measure the temperature of the sheath along its length in the core region will be routed through the handle to a connector similar to that of the TFE. Thermionic fuel elements, non-thermionic fuel elements, and beryllium oxide internal reflector rods all would be interchangeable in the core structure.

e. Internal Reflector Rods

The beryllium oxide internal reflector rods consist of beryllium oxide contained inside a stainless steel sheath. The external configuration of the beryllium oxide rods would be like the TFEs except for the electrical connections at the top end.

f. Reactor Vessel

The reactor vessel shown in Figure 1 is approximately 13 feet long, 23-1/2 inches in diameter at the top end and has a 20 inch inside diameter in the core region. The reactor vessel has a double wall in the core region to increase the probability of coolant remaining above the core in the event of single vessel leaks. The vessel wall thickness has been increased beyond the ASME code requirements in the region above the core to withstand the pressure transients associated with certain reactor excursions. Nozzles are provided for inlet and outlet coolant, overflow level control and the helium cooling gas. The reactor vessel is mounted in the test cell from the top. The electrical connector mechanism, which provides the connection point for power and instrumentation from the TFE handles, is housed in a helium leak tight dome at the top of the reactor vessel. Fixed at the top of the electrical connection dome housing is a 30 inch gate valve to provide access for fuel element exchange.

The NaK coolant enters the reactor vessel through three (3) 3 inch nozzles located just above the core, goes down through a downcomer

Figure 1 Reactor Vessel

region between the shroud and the reactor vessel into a plenum beneath the core. From the plenum, coolant flows through the 40 mil coolant annulus between the element and process tube. The NaK exits the reactor vessel after having passed through the necked-down cross flow region at the top end of the thermionic fuel elements. The selection of up-flow through the reactor supports convective cooling forces. An opposite core flow direction opposes this convection cooling, and at the low flow velocities would be expected to produce regional stagnation of the coolant in the core with attendant large stresses induced in the core structure and fuel elements. Reliable liquid metal level control in the reactor vessel is maintained with a pump from the drain tank and an overflow line.

The conduction and nuclear heating of the fission product trap and cesium reservoir located in the TFE handle require these two components to be cooled. Helium gas cooling has been incorporated rather than introduction of a cold liquid metal coolant at the top of the vessel since the latter would require unreasonably large flow velocities or introduce the regional stagnation problem noted previously. A cooled helium gas flow is injected between the two grid plates at the top end of the shroud structure, enters the annulus between the TFE handle and the process tube through holes in the process tube and flows in this annulus past the fission product trap and the cesium reservoir and exits below the cesium reservoir. An outlet helium nozzle is placed in the reactor vessel at this location. The exit helium is circulated in an external loop through a vapor trap, cooled, then returned to the top end of the process tube. Calculations indicate the trap can be cooled to 100°C and the cesium reservoir to 275°C without electric heat. Control of the cesium reservoir temperature is maintained by the use of thermocouples to sense the temperature and an electrical heater to adjust the temperature as needed.

2. Reflector Cup

Located outside the reactor vessel is the external reflector cup assembly which provides the reactivity control of TREX. (Figure 1). The calculated reflector worth is approximately 5% in the configuration shown. Calculations also indicate that if the internal beryllium oxide rods were replaced with TFEs, the control element total worth would increase to over 7.5%. The 6 inch poison backed drums are positioned by bearings at the top and bottom ends and driven with shafts through a shield plug by actuators located beneath the shield plug. At the outer diameter of the reflector cup assembly is located a band of steel approximately 2 inches thick to maintain reactor vessel integrity in the event of certain reactor excursions. The actuators are mounted on the lower side of the shielding block to assure limited activation of these components. The shield also provides personnel protection for actuator maintenance when the reactor is shut down.

The reflector cup assembly may be raised and lowered as a unit and scrambled. When in the lowered position, the shutdown margin for the reactor is approximately \$18, assuring substantial secondary shutdown capability. Procedurally, the cup would be lowered to preclude an incident associated with fuel handling. In addition, the reflector cup assembly may be completely removed from the reactor vessel and taken out of the test pit for maintenance, and component or system replacement. Ease of maintenance is demonstrable. However, additional design effort will be needed to establish proper clearances, heat removal and bearing and actuator configurations.

3. Coolant and Gas Systems

The proposed liquid metal coolant system concept has two coolant loops with an immediate heat exchanger to separate the coolant material which circulates through the reactor and becomes radioactive and the coolant which is routed external to the containment for heat removal. Design concepts are based on the use of NaK 44 and stainless steel for piping

and components exposed to the liquid metal. In addition to the primary and secondary loops, there is an auxiliary coolant loop which assures core after-cooling. Cleanup and monitoring systems are provided as a part of each of the two main coolant loops and may also be connected to the auxiliary coolant system. A helium gas system is provided to cool the helium gas for the fission product trap and the cesium reservoir. It also provides a control of the helium cover gas in the reactor vessel, the helium cover gas in the surge, vent and drain tanks and controls the liquid metal vapor in the instrumentation dome.

The primary loop is connected with three (3) 3 inch inlet and outlet lines to the reactor vessel. The low ΔP tube and shell intermediate heat exchanger, a 600 gpm electromagnetic pump, a flow meter, an electric heater and a vent tank are the primary loop components. These components are located above the core to preclude siphoning of the coolant from the core in the event of a pipe or component rupture. The 100 kWe heater is sized to enable liquid metal heat up in the absence of nuclear heat in the core. Similar equipment, in addition to the NaK-to-air heat exchanger, is in the secondary loop. The elevation of the intermediate heat exchanger is such that about 5% of the full power heat energy may be removed by convection only in the primary and secondary loops. Thus, if pump and blower power is lost, and the piping remains intact, after-heat removal is provided by this and the auxiliary system.

The auxiliary cooling system provides backup reactor after-heat-removal in the event of a primary loop pipe breakage and liquid metal fire. The auxiliary loop components are located so that heat removal is achieved solely by convection cooling and no electrical power is required. The auxiliary cooling system also includes an intermediate heat exchanger between the primary and secondary loop and a NaK to air cooler in the secondary loop.

Separate cleanup and purity monitoring service systems are provided for primary and secondary loops. In the radioactive primary service system are located two cold traps for reducing impurities to around 5 parts/million and a hot trap to reduce the impurity level to less than 1 part/million. These two cleanup systems would be used alternately. The hot trap may not be required for the proposed stainless steel, Ceramvar system. Also in the service system is an oxygen meter, a sample station for removal of liquid metal for chemical analysis and a plugging meter for liquid metal purity monitoring. The service system for the secondary coolant loop is identical to that for the primary loop except for the deletion of a hot trap which is considered to be unnecessary for that system. Drain tanks used for supply and storage are located at a low point in each system.

All of the piping and components in the liquid metal coolant systems, except the reactor vessel, would be mineral insulated. The reactor vessel would have metallic insulation to facilitate in-service inspection of the reactor vessel, if required.

The helium cooling of the fission product trap and cesium reservoir has been described above. The helium coolant system cools and recirculates the gas. Helium was selected because of its heat transfer properties and because of common containers, it is also the cover gas in the reactor vessel and surge, vent and drain tanks.

Helium gas in the electrical connection dome is planned to be held at a higher pressure than the reactor vessel to limit the NaK vapor in this space. The connection dome helium gas loop incorporates a charcoal trap for the removal of fission products to reduce contamination levels in that zone.

4. Fuel Handling Equipment

Fuel handling equipment for the TREX is designed for the movement of thermionic fuel elements, non-thermionic fuel elements and beryllium oxide rods to and from the reactor core as well as for certain

operations in a Radioactive Materials Laboratory (RML). The fuel handling machine is a shielded, multipositioned, storage and transfer machine which may be located over the reactor, storage pit or RML for the insertion or removal of core elements.

The concept for the fuel handling machine is one in which an uncontaminating cover gas atmosphere is maintained while the fuel element is withdrawn from the reactor by connecting a grapple in the fuel handling machine to the top of the fuel element. The element then is drawn vertically into a cavity, one of eight in a revolving barrel inside a shielded case, which is the central part of the fuel handling machine. When the fuel element is inside the fuel handling machine, the grapple is disconnected and the revolver rotated to a cavity in which a new fuel element is located. The grapple is attached to the top end of the new element and it is inserted into the position from which the original fuel element was withdrawn. The fuel handling machine may then be moved to a new position and the process repeated until the transfer operation is complete. The number of positions in the revolver was arbitrarily selected and subsequent study of the operational procedures and efficiency may indicate a different number is desirable. The fuel handling machine is designed so the volume in which the fuel element is housed will have an inert helium environment. Thus a valve is required at the bottom of the fuel handling machine body. For the transfer operation the fuel handling machine is placed over the top of the large valve in the electrical connection dome and the space between evacuated and purged with helium. The two valves are then opened for the transfer operation. Seals between the fuel handling machine and the reactor vessel are needed to insure no leakage during the transfer operation.

The fuel handling machine concept would have the capability for moving on rails in the X and Y position and as well as tipping to accommodate the slope associated with the spread process tube arrangement. Positioning and indication instrumentation would be

provided so that precise positioning would be possible. Further, it is contemplated that viewing windows would be placed on the sides of the electrical connection dome for visual or television access to the region of the electrical and mechanical connectors at the point above the end of the TFE handle.

Once the element is inside the fuel handling machine, it would be transported either to a storage position adjacent to the reactor vessel or would be removed from the region in which the transfer operation was performed. From there it would be moved to either an alternate storage area or to a location in which the fuel elements would be discharged for transfer to an RML.

5. Instrumentation and Control

Figure 2 shows a block diagram of the TREX control system concept. This design concept is one which separates nuclear from hydraulic control by preheating the core and establishing flow rates before reactor startup. Further, by virtue of the preheating and the constant emitter temperature control characteristic of the power conditioning modules, nuclear and electrical control interactions are minimized. Thus, the reactor control assumes a role quite similar to conventional reactor control systems, thereby simplifying the task of the reactor operator and enhancing safety of operation.

a. Nuclear

The basic nuclear control mode is manual, meaning that the reactor operator manually operates the moveable control segments of the external reflector, thereby controlling the reactivity of the reactor and, in turn, the reactor power level. The indicating equipment from which he derives the information to perform these changes is connected to nuclear sensors located

FIGURE 2. TREX CONTROL SYSTEM

in a region outside the reactor vessel adjacent to the core. This nuclear instrumentation consists of two channels of source range, three channels of intermediate range and three channels of wide range instrumentation in addition to channels specifically provided for automatic power control. Control ranges are such that there is at least two decades of overlap between adjacent ranges and the power level range extends 8 decades from above full power level down to a point a few decades above source range. This nuclear instrumentation is standard state-of-the-art equipment.

The second nuclear control mode is envisioned for TREX because of the weak feedback mechanisms in the reactor which in turn may cause drift once the steady full power has been reached. To overcome such a condition, an automatic power level mode has been incorporated into the nuclear control system. In particular, a control drum, positioned so it has only a small reactivity worth would be controlled from a separate nuclear sensor from those used for indication and safety to hold the reactor power to a set power point selected by the reactor operator.

A third control mode is envisioned for eventual space power application and is shown by dashed lines in Figure 2. In this mode, the plant, is put on fully automatic control and an error sensing the dc bus voltage is fed back and employed to control the reflector control drum and thereby reactor power level. This mode may be experimentally verified in the TREX.

b. Hydraulic

The heat removal system consists of a primary loop which transports the heat to an immediate heat exchanger and a secondary loop which transports the heat to a heat dump. Electromagnetic pumps are incorporated in both the primary and secondary loops. In addition, electrical heaters are incorporated in the loops to facilitate service and maintenance on the heat removal system

and to permit the NaK system to be electrically heated to approximately 780°K prior to reactor startup. The heat dump contains a blower for providing the air flow and a set of louvers which are used to regulate the air flow. This regulation of air flow with the adjustment of louvers is done automatically with control dependent upon the temperature of the secondary coolant loop. The primary and secondary pumps and heaters and heat dump blower in the heat removal system will be controlled by manual adjustment of regulated power supplies. Prior to reactor startup, full flow will be established in both the primary and secondary pumps since sufficient power will be delivered to the heaters to raise the loop temperatures to slightly in excess of 780°K . The NaK temperature at the heat dump exit will then be controlled to 780°K by placing the louvers on automatic control as shown schematically. Prior to reactor startup the heat dump louvers will be slightly opened and as the reactor subsequently is increased in power, the louvers will be automatically opened to accommodate the increased heat load.

c. Electrical

The electrical control system is shown schematically on the right hand side of Figure 2. Thermionic fuel elements are connected electrically in pairs, one positive and one negative with respect to ground. The pair of TFE current leads are fed to a power conditioning module. These leads are the ones which pass through the dome electrical connector and were previously discussed. The power conditioning modules in TREX serve three functions. The first is to control the electric current extracted from each TFE pair and indirectly, the TFE emitter temperature. The TFE pair current is controlled to a demand signal which is linearly related to the thermal power level of the reactor. The power conditioning modules for TREX for each pair of thermionic fuel elements will be such that the current control set point may be adjusted from either a local site or perhaps the control room.

The second function of the power conditioning module is to perform a reliability function in the event that an open circuit failure is encountered in either of the TFE pair. An electronic switch across the input terminals to the power conditioning modules can be closed in this event to convert the open circuit to a short circuit and permit the power from an unfailed adjacent element to continue to be used.

The third function of the power conditioning module is to step up the dc voltage to a value which is more consistent with the demands of the power user. In the power conditioning module will be located equipment for individual TFE diagnostic measurements. Power conditioning equipment will be located in the test area adjacent to the test cell or in it to minimize low voltage cable losses. Output of the power conditioning modules is connected in parallel to the dc bus which contains the electrical load. In addition, the bus will contain a ballast load and a ballast load control circuit. The configuration for TREX is one which allows the adjustment of the load to accommodate the requirements of the production of power in the thermionic fuel elements in the reactor. This is accomplished by varying the ballast load through an electronic switch so that the dc bus voltage remains constant.

An attractive feature of the electrical control scheme which utilizes TFE current control, thus constant emitter temperature, is realized in the electrical control scheme for the cesium reservoir temperature. Since the optimum cesium reservoir temperature in the thermionic fuel element is strongly dictated by the operating emitter temperature and weakly influenced by other variables, it is possible to fix the cesium reservoir temperature if the emitter temperature is fixed. Thus, the decision to operate at a fixed emitter temperature permits the cesium reservoir temperature to be operated at a fixed temperature also. This is accomplished through a simple set point control of the cesium reservoir. The TFE cesium reservoir temperatures would be established prior

to reactor startup and would normally not be changed until after shutdown. However, the control units will be located in the control room and should the need become necessary, such a change could be made with ease.

A plant safety system is incorporated as a part of the instrumentation for TREX. Three actions have been tentatively selected for plant safety responses to trouble signals from one or more of a variety of components or sensors. The first is a reactivity insertion block which prevents either the reactor operator or the automatic power level control system from inserting reactivity. The second safety action is a rundown which would involve withdrawing reactivity from the reactor at the maximum rate which is approximately 4¢/second. The third safety action is the scram. Scram is instituted by fast rotation to the least reactive position of the poison backed drums and by rapid lowering of the reflector control cup. It is envisioned that the scram could be accompanied by a programmed flow reduction in both the primary and secondary coolant systems to reduce thermal shock on the reactor core. Physical and electrical arrangement of all the nuclear and safety equipment is planned to meet the requirements of IEEE 279. This incorporates the use of coincident circuitry as well as redundant channels to reduce the likelihood of inadvertent scrams and to assure safety action when appropriate.

In addition to the instrumentation and control functions described above, there will be sensing instrumentation installed so the condition of any given test device or auxiliary system component will be indicated in the control room promptly for observation and action, if necessary. Further, certain of these signals will be connected to the safety system which will institute safety action automatically. Thus, there are flow meters in each loop of the coolant system, level and pressure sensors, thermocouples and voltage and current measurement points in a wide variety of

locations in the TREX system. There are position indicators and controls for the control rods, measurement and recording of voltage and power output levels of each of the thermionic fuel elements, recording of all of the signals which are developed from the nuclear channels and a variety of other diagnostic and measurement equipment.

6. Facilities

a. Fast Reactor Core Test Facility (Los Alamos)

The Thermionic Reactor Experiment study was made based on the use of the Fast Reactor Core Test Facility (FRCTF) at Los Alamos. This facility is located at the Ten site of the Los Alamos Scientific Laboratory, a location approximately 1 mile from the city of Los Alamos, New Mexico.

The FRCTF has two test pits which are 22 feet by 22 feet square and 37-1/2 feet deep. The test pits are housed in a 47 feet by 100 feet bay serviced by a 50 ton crane. Adjacent to the test pits is the secondary sodium room. The control room is located adjacent to the test bay and a shielded radioactive sodium room is located adjacent to the test pits below floor level and has common containment with the test pits. On the opposite side of the two test cells are upper and lower level remote disassembly cells. A hatch in the floor of the test building provides access through which material may be moved from the test building to the upper disassembly cell. Separate ventilation equipment provides cooling and a controlled nitrogen atmosphere for the disassembly cells, the reactor test pits and the radioactive sodium room.

Electrical power is supplied to the Fast Reactor Core Test Facility by a ring loop network. Two transformers supply a total of 2000 kVA to the facility. In addition, there is a 75 kVA emergency

diesel generator for selected emergency systems in the facility, as well as a 25 kVA internal combustion motor generator. Hold-up tanks for radioactive waste are located in the facility. A disposal facility exists elsewhere on the LASL site.

b. Proposed Modifications

The proposed modification to one of the two test pits in the Fast Reactor Core Test Facility for TREX is shown in Figure 3. At about one-third of the depth of the test pit is located a shielding floor. The reactor core is located in the right front quadrant of the test pit and is shown below the refueling machine. The reactor is mounted from the top, at the main shielding floor. Below the shield floor and around the reactor is additional shielding. Access to the reflector cup is provided in the shield.

A storage pit into which core elements may be placed is located behind and in line with the reactor vessel and refueling machine. A rail system between the reactor and the storage pit is provided for the fuel handling machine.

In the third quadrant of the test pit is located the primary loop of the liquid metal cooling system. The primary to secondary heat exchanger is located at an elevation above the core, as are all of the components in the primary coolant system to provide no-pump convective cooling and to prevent siphoning of core coolant. A shield is required between the radioactive primary loop and the operation gallery.

An access way is provided in the fourth quadrant of the pit. The access way is planned to be used principally for the entrance and exit of personnel to the space below the reactor for maintenance of the actuators on the reflector control cup. A removable stairway will be used. The reflector control cup may be installed and removed from its position below the reactor core through the access way.

FIGURE 3. FAST REACTOR CORE TEST FACILITY TEST CELL

The space above the refueling gallery is enclosed with a room whose walls and ceiling are concrete lined with steel plate. This room provides containment and nitrogen atmosphere control during the time the reactor is in operation. Further, in the event that breach of the liquid metal inert cover gas containment occurs during the fueling operations, this space may be evacuated and nitrogen gas valved into the room to control the potential liquid metal fire. A hatch through which the refueling machine may be moved is provided in the roof. This hatch is located over the storage well to preclude damage to the reactor should the refueling machine be dropped. Personnel access to the refueling gallery is through a double personnel lock from the test building floor. Provision is made for either nitrogen or air to be valved to the space underneath the reflector cup and in the operating gallery so personnel may enter these spaces without special breathing apparatus.

The nitrogen cooling system for the test pit must have ducts added to direct cooling nitrogen to components of the TREX. Further, safeguards considerations indicate a potential requirement for relocation of the cooling ducts located outside the building in such a fashion as to reduce the likelihood of their rupture should an earthquake occur. At least two alternatives exist for accomplishing this.

Structures may have to be provided for certain items of equipment that may be required. Among these are the effluent filter pit and the radioactive gas filtration system. A stack must be constructed which is connected to these two systems.

A. THERMIONIC FUEL ELEMENT DESIGN

1. General Description

The TFE and handle assembly for the thermionic reactor experiment consists of a tube approximately 12 feet in length which contains the thermionic fuel element, its associated hardware and instrumentation and leadout system necessary to bring the electrical power from the reactor core region up to the dome area at the top of the reactor. The essential features of this assembly are the TFE (enclosed axially by reflector material), a cross flow plenum for coolant flow, a cesium reservoir, a fission product trap and the electrical leadout system. These features are depicted schematically in Figure 4. A more detailed definition of the TREX-configured TFE is delineated in Figure 5.

In describing the features of the TFE and handle assembly, the present concept of the TFE will be used, i. e., the slip fit, wet flashlight approach. The concepts and features external to the TFE, however, apply equally well to alternate TFE configurations.

2. Reactor Core Section

The electricity generating portion of the TFE, which also forms the active core region of the reactor, consists of the assembly of a number of emitter/collector pairs as shown in Figure 6. Each emitter consists of a tungsten can of a wall thickness of approximately 30 mils containing a nuclear fuel such as UO_2 . The fissioning process in this fuel heats the emitter to temperatures high enough for electron emission to occur. During this fission process, fission products are generated which must be accommodated somewhere within the TFE and handle. The proposed accommodation system will be discussed later in more detail.

The electrons emitted from the emitter drift across the interelectrode region of approximately 10 mils spacing and are then collected on the adjacent collector, thus, building up an electrical potential of about 1/2 volt between the emitter/collector pair. To allow the electrons to drift across the interelectrode region without building up a space charge, a supply of cesium ions is needed. These are provided by a cesium supply system which furnishes the proper cesium pressure.

FIGURE 4. THERMIONIC FUEL ELEMENT AND HANDLE

Figure 5. Thermionic Fuel Element and Handle (Detail Drawing)

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FIGURE 6. THERMIONIC FUEL ELEMENT (TFE) SCHEMATIC

The electrons leaving the emitter remove a considerable amount of heat and deposit it at the collector. This process, known as emission cooling, counts for approximately 1/2 to 2/3 of the total heat removed from the emitter. This heat, deposited at the collector, then flows through the sheath insulator and sheath tube shown in Figure 6 and from there into NaK coolant flowing around the TFE.

The electrical current flowing in a TFE is usually quite large (in the order of 150 - 200 amperes). For this reason, only welded or brazed joints are used in the electric current path. In order to increase the total voltage produced by a TFE, several (approximately 12 emitter/collector pairs) are connected electrically in series. This series connection is accomplished by forming the TFE from emitter/collector subassemblies as shown in Figure 6. Thus, the collector from one emitter/collector pair is connected to the emitter of the next adjacent emitter/collector pair. At each end of the TFE, a flexible lead is provided in order to accommodate differential thermal expansion as shown in Figure 7. At one end of the TFE one flexible lead is grounded to the sheath tube, and hence to the reactor proper. The opposite polarity lead, at the other end of the TFE, is brought out through a ceramic/metal seal to a leadout system at the top of the TFE. In order to form a pair of TFEs, one of which produces a positive voltage and one which produces a negative voltage with respect to ground, every other TFE has an alternate orientation. That is, half of the TFEs have a bottom collector grounded and the other half have a bottom emitter grounded. This concept of TFE pairs will be discussed later in more detail.

Figure 8 shows the vicinity surrounding the active TFE portion in greater detail. A neutron reflecting material at the top and bottom of the TFE is composed of beryllium oxide clad in a suitable metal container. Above the top reflector is a necked-down portion of the TFE which provides a plenum region for cross flow of the liquid metal coolant. Above this plenum region is located the ceramic/metal insulator seal mentioned previously. At the top of the TFE envelope is an expansion bellows which is provided to accommodate the differential expansion between the various metals used in fabricating the TFE.

FIGURE 7. REACTOR CORE REGION

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FIGURE 8. TFE AND HANDLE

3. TFE Handle

The successful operation of a TFE imposes several requirements upon the TFE handle. First, in order to neutralize space charge in the inter-cell region of the TFE, a cesium supply system is required. Second, the fission process in the UO_2 fuel generates fission products. If these products were allowed to build up in sufficient amounts within the TFE, the thermionic process would be impaired. Thus, a fission product accommodation system is required. Third, an electrical leadout system is needed which reliably prevents interruption of the high currents and hence, a reduction in the electron cooling which would significantly overheat the emitters. Due to the high currents involved, the leadout system has a low resistance in order to minimize the voltage drop in this portion of the over-all system.

a. Cesium Supply System

In addition to neutralizing the space charge within the interelectrode region of a TFE, cesium vapor is also required to adjust the emitter work function to increase the electron emission and also to adjust the collector work function to increase the total voltage output from each thermionic cell. In order to accomplish each of these three functions, the cesium vapor pressure must be carefully controlled. For typical operating conditions this optimum cesium vapor pressure can be obtained by maintaining a liquid cesium reservoir at a temperature in the range of 300 - 350°C. Since this temperature range is well below the coolant temperature, it is necessary to locate this liquid reservoir in the TFE handle above the coolant level. In the present design, the reservoir is located 12 inches above the liquid coolant level. A cesium reservoir tubulation then connects the TFE envelope with the cesium reservoir as shown in Figure 9.

In order to insure that the temperature of the cesium reservoir actually determines the cesium pressure within the TFE, it is essential that there is no other region within the TFE envelope which is cooler than the cesium reservoir. For this reason in the present design, it is planned to braze the cesium reservoir tubulation directly to the main power lead from the converter as shown

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FIGURE 9. CESIUM SUPPLY SYSTEM

in Figure 9. Thermal analyses have shown that this power lead has a relatively uniformly decreasing temperature. Hence, no cold spots shall occur between the TFE and the cesium reservoir using this method.

To provide for adequate temperature control, it is planned to cool the cesium reservoir to below the required operating temperature by use of helium surrounding the TFE handle. The cesium reservoir would then be raised to the required temperature using a heater surrounding the reservoir. Two thermocouples located in the cesium reservoir would sense the temperature and supply a signal to a controller which would automatically maintain the proper temperature by adjusting the electric heater power.

Two additional requirements of the liquid cesium reservoir supply system are:

- An adequate supply of liquid cesium must be maintained in the cesium reservoir in order to make up for cesium losses due to chemical reactions within a TFE, with off-gases and with fission products. It must also make up for losses through the fission product port, which will be described later.
- Since the cesium reservoir is located above the TFE, it is necessary to design a reentrant reservoir to prevent the liquid cesium from flowing down through the cesium reservoir tubulation and into the TFE. A "standpipe-type" design is used for this purpose.

b. Fission Product Accommodation System

The fission products generated by the fissioning of the nuclear fuel are vented from each emitter into the thermionic converter volume. The gaseous products, primarily xenon and krypton, can degrade the performance of the TFE both thermally and thermionically if allowed

to build up to significant pressures (above approximately 5 torr). To remove these gases from the TFE, a port similar to those used on General Electric Test Reactor in-pile tests, is planned. The gases would then be routed through fission gas tubulation to a sorbent pump where they are stored.

The port, which is a capillary tube maintains the proper flow rate of fission gases from the envelope with a pressure of approximately 5 torr in the converter envelope (provided that a hard vacuum is maintained on the downstream side of the port). This size port also restricts the cesium loss to a value of less than 1/2 grain of cesium per year. The only restriction on the operating temperature of the port is that it be maintained high enough so that the cesium vapor will not condense and plug the port. In the present design, the port operates near the coolant outlet temperature. The port, fission product tubulation, and sorbent pump are shown in Figure 10.

The present design of the sorbent pump utilizes fine grain charcoal granules to sorb the fission gases. For the sorbent pump to operate properly, it is necessary to maintain its operating temperature at less than 100°C. For this purpose the sorbent pump is located in the TFE handle well above the cesium reservoir, and out of the high gamma heating region. This location is about 2 feet above the reactor coolant level.

For typical TFE operating conditions, the amount of fission gases which must be accommodated, (assuming approximately 80 percent release) is 200 cc at 100°C for a lifetime of 10,000 hours. This amount of fission gas can be accommodated by a sorbent pump having a charcoal filled volume of 225 cc. The sorbent pump pressure at the end of the 10,000 hour life would then be 35 torr and hence, the pressure of fission gas inside the converter volume would be around 40 torr. This pressure still allows adequate performance of the thermionic converter. To provide 225 cc of volume for the sorbent pump, assuming that 0.6 of the TFE cross sectional area is available, requires that a sorbent pump of approximately

FIGURE 10. FISSION PRODUCT ACCOMMODATION

3 feet in length be used. The additional cross sectional area in the handle is needed for the electrical leadout system and structural materials.

Since the fission products stored in the sorbent pump are radioactive, there is a need for a significant amount of shielding between the sorbent pump and the seal plate at the top of the TFE handle. This shielding will be accomplished using the electrical leadout system described later.

Double containment of the fission gases is provided for in the region above the coolant cross flow plenum. This is desirable since this region contains the ceramic/metal seals and bellows assemblies which could possibly cause leakage of the fission gases from the TFE envelope. This double containment is provided by the outer tubing of the TFE handle and by the seal plate at the top of the TFE handle.

c. Electrical Power Leadout System

The objectives in designing the electrical power leadout system are to minimize the total electrical resistance in the current circuit and to maintain high reliability against failures due to open circuit. To accomplish this, the size of the current leads is made as large as possible within the restrictions of the size of the TFE and handle and all-welded or brazed construction is used

Within the TFE there are three main sources of resistance. There is the resistance that is associated with each of the emitter/collector subassemblies within the TFE. This value, which varies depending on the particular design of the emitter/collector subassemblies, is typically in the order of 565 microhm per cell at operation conditions. Second, there is the resistance associated with each of the flexible leads joining emitter/collector assemblies to the leadout system. These two flexible leads have a resistance of approximately 200 microhms each. Third, there is a resistance associated with the leadout which

removes the electrical current from the top flexible lead, past the reflector and plenum region, through the ceramic/metal seal and bellows region and to the connector outside of the TFE. The material chosen for this leadout is molybdenum due to the high temperature requirement within the TFE. A typical resistance associated with this lead is approximately 1400 microhms. Thus, within the TFE, there is a resistance of 1800 microhms plus the resistance associated with each of the emitter/collector assemblies

The size of the leadout system outside of the TFE varies considerably depending on the location within the TFE handle as shown in Figure 11. The nature of the conductor and the conductor size in each of the various regions in the TFE handle is shown in Table 1. The total resistance associated with this leadout system in the TFE handle is 1480 microhms. The insulation used in each of the regions above the TFE is either ceramic or a mineral insulation with the exception of the final large conductor above the fission product trap. The insulation in this upper region can be of an organic nature such as polyvinyl chloride, since the radiation effects in this region are at acceptable levels for organic materials.

An additional feature of the leadout system is that the region just above the TFE proper is at a potential other than ground. For this reason, a second insulator seal is employed at a point past the point at which the flexible leads have been joined to the TFE. This second insulator seal allows the cesium reservoir and sorbent pump to be at ground potential. Another feature is that the large conductor at the top of the TFE handle is spiraled so that this large mass of copper supplies the shielding necessary to suitably attenuate the radiation activity from the fission products in the sorbent pump.

In addition to the large current carrying lead in the TFE handle, there are seven other leads. Two of these are used to supply electric power to the cesium reservoir heater. Four are for the two thermocouples used for redundant measurement of the temperature of the cesium reservoir. The seventh is a non-current carrying lead used to sense the voltage potential directly at the point where the current leaves the TFE envelope.

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FIGURE 11. ELECTRICAL POWER LEAD-OUT SYSTEM

TABLE 1

Region	7	6	5	4	3	2	1

Length, (Ft.)	1/3	3-1/2	3-1/2	1	1-1/2	1/2
Avg. Temp. (°C)	100	100	100	250	400	500
Area (Cir.Mic)	66,000	300,000	100,000	106,000	66,000	69,000
Resistivity Ω -CM Ft.	12	12	12	17	23	28
Res. ($\mu\Omega$)	60	140	400	160	520	200

Total Copper Lead Resistance = $148 \mu\Omega$

Total Cu + Mo Resistance = $3300 \mu\Omega$

(Includes $200 \mu\Omega$ Expansion Each End, plus $1400 \mu\Omega$ for Mo)

In addition to this standard amount of instrumentation supplied with each TFE assembly, provision will be made for additional instrumentation to approximately 5 percent of the TFEs. Some possible examples of this instrumentation are emitter thermocouples, collector thermocouples, temperature measurement of the fission product trap, neutron and gamma flux chambers, heat flux measuring devices and sheath thermocouples.

B. NON-THERMIONIC FUEL ELEMENTS, INTERNAL REFLECTOR RODS AND DRIVER FUEL ELEMENTS

1. Non-Thermionic Fuel Elements

a. Introduction

The nuclear checkout of the thermionic reactor experiment is planned with a reactor core assembled from non-thermionic fuel elements. The non-thermionic fuel elements will then be replaced by thermionic fuel elements (TFEs) until the entire core consists of TFEs. The design goal for the selection of the non-thermionic fuel is to approximate, as closely as possible, the nuclear and hydraulic characteristics of the TFEs. This will assure minimum perturbations of the operating characteristics of the reactor as non-thermionic fuel elements are replaced with TFEs.

The reactor core is designed so that the liquid metal coolant inlet temperature is 527°C with a maximum temperature of 633°C and a design outlet temperature of 627°C with a 733°C maximum bulk temperature. The heat transfer fluid will be Na-44 wt%K. The TFEs will be fueled with UO_2 with a Ceramvar outer sheath. The nominal composition of Ceramvar is as follows: 27 wt% Ni, 24 wt% Co. with Fe constituting the remainder.

The external geometry of the non-thermionic fuel elements will be the same as the TFEs to duplicate the hydraulic characteristics of the TFEs. Since the volume of fuel in the non-thermionic elements is greater than that in the TFEs, the enrichment will be adjusted to match the nuclear characteristics of the TFEs. The enrichment factor will also take into consideration the presence of other materials in the TFEs, i. e., tungsten (W), niobium (Nb) and alumina (Al_2O_3).

b. Selection of Materials and Design Considerations

The proposed first choice for the fuel-clad for the non-thermionic fuel is UO_2 clad with one of the Type AISI 300 series stainless steels. The choice is predicated on the vast operating experience

and knowledge on the UO_2 -stainless steel system in terms of irradiation experience, mechanical properties and compatibility in an alkali metal cooled fast spectrum environment. The second major consideration is the ability to achieve equivalence between the non-thermionic fuel and the TFEs most simply and reliably within the framework of existing technology. The UO_2 -AISI 300 series stainless steel system will be discussed in terms of the nuclear and materials requirements imposed upon it.

c. General Description

The basic design of the non-thermionic fuel element is shown in Figure 12. The fuel elements are approximately 1 inch in diameter with an active fuel length of 16 inches. Depleted UO_2 pellets 0.5 inch long are in contact with both ends of the enriched fuel to act as thermal barriers for the end BeO reflector material. The fuel pellets are dished to accommodate the difference in thermal expansion between the edge and center of the fuel. The fuel assembly is spring loaded to assure good thermal contact throughout the fuel column. The volume occupied by the spring also acts as a plenum region to accommodate fission gases. The fuel element is completely sealed above the plenum volume and the total active fuel element is 33.5 inches long. A stainless steel extension tube, shielding and fuel handling top assembly complete the fuel element.

d. Requirements of Driver Fuel Element

The TREX core will first consist of all non-thermionic fuel elements which will be replaced by TFEs a few at a time and the reactor physics, response, and thermionic performance evaluated. Eventually the entire core will be TFEs. The lifetime of a particular non-thermionic fuel element may vary from a few hundred hours to 10,000 hours. To match the thermal output of TFEs of approximately 9 kW/ft, will require a total burnup for a non-thermionic fuel element for 10,000 hours of approximately 0.5 at.%(atomic percent) or 4,000 MWd/T, a very modest burnup for UO_2 . Burnups of up to 120,000 MWd/T in a thermal flux have been accomplished on mixed oxide fuels (80 w/o UO_2 - 20 PuO_2) and up to 75,000 MWd/T in a fast flux in EBR II⁽¹⁾ under conditions similar to those required for the non-thermionic fuel elements.

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FIGURE 12. NON-THERMIONIC FUEL ELEMENT

e. UO₂ Fuel Behavior(1.) Solid Fission Products

With a liquid metal coolant inlet temperature of 527°C the surface of the fuel will be at 627°C with the centerline temperature at approximately 1627°C. At the outlet coolant temperature of 627°C the centerline temperature of the fuel will be 1727°C.

The effect of fission products, both gaseous and solid in a large part determine the lifetime of a fuel element. Both types of fission products contribute to fuel swelling and for any particular set of environmental conditions their contribution must be assessed.

Approximately 1.7 atoms per atom fissioned are generally regarded as solid fission products while 0.3 atoms per atom fissioned are the inert gasses xenon and krypton. The specific volume of the solid fission products can alter the final volume of the fuel due to the difference between the initial and final total atomic volumes of the fissioned atoms. The stable solid fission products produced are Zr, Ce, Nd, La, and X along with the volatile elements Rb, Cs and I.

Anselin⁽²⁾ has calculated the swelling expected from solid fission products based on analytically determined concentration of fission products. His analysis includes the summation of both positive and negative volume changes and includes the stable state of the fission products in the UO₂ matrix i.e., oxidation states more or less stable than UO₂. An exact number for the volume change due to solid fission products is also a function of the irradiation conditions. These include temperature, fuel pin design, fuel-cladding gap, density of fuel, and initial stoichiometry

A value for the maximum solid fission product swelling rate is 0.32 percent $\Delta v/v$ per at. % burnup if complete utilization of the vacancies created is assumed and 1.32 percent $\Delta v/v$ per at. % burnup if none of the vacancies are consumed. Using an average of these two above figures gives a swelling rate expected from solid fission products to be under 0.5 mil on the radius of the non-thermionic fuel in 10,000 hours at power levels of 9 kW/ft. The extent of fuel swelling as a result of solid fission products is therefore very minimal for this application.

(2.) Gaseous Fission Products

The contribution of the gaseous fission products Xe and Kr is more difficult to assess. Both gases are extremely insoluble in UO_2 and nucleate out in gas bubbles. If the temperature in the fuel is high enough the gases will migrate up the temperature gradient to the center of the fuel and be released to the plenum region.

As a result of the fissioning process, the gas production rate of Kr and Xe is 1.3×10^{-6} STD $\frac{\text{cc}}{\text{watt-hour}}$. The internal pressure caused by this gas release will result in a hoop stress on the stainless steel clad with subsequent deformation by creep. The limiting extent of creep has been calculated. Assuming a 50 cc plenum volume in the fuel element and 100% gas release, a 150 psi pressure within the fuel element at operating temperature is generated and this pressure causes a hoop stress of 2350 psi. The elastic strain resulting at this stress is 1.28×10^{-4} in/in and the contribution due to creep is an additional 1×10^{-4} in/in after 10,000 hours. This results in a total strain of 2.28×10^{-4} in/in which corresponds to a change in the diameter of the clad of 0.2×10^{-3} in. This amount of deformation is insignificant.

This calculation is conservative in that it assumes 100% gas release and the total pressure at the end of 10,000 hours was assumed to be acting on the system initially. All the fission gases are not expected to be released from the fuel in an actual operating fuel element. The amount of fission gases released is strongly dependent on the center temperature of the fuel. Gas release data of Nichols,⁽³⁾ is shown as a function of time and centerline temperature in Figure 13. The expected release for 10,000 hours at a centerline temperature of 1750°C is expected to be approximately 15%.

The swelling behavior of the mixed oxide 80% UO_2 - 20% PuO_2 in a 316 stainless steel clad is shown as a function of total burnup in Figure 14 and summarizes the irradiation experiences in EBR-II.⁽⁴⁾ The clad temperature for these irradiations was approximately 555°C. These data also show the swelling behavior of the cladding material. The major contribution to the total swelling appears to be the swelling of the clad for irradiations up to $\sim 40 \times 10^3$ MWd/T.

For thermionic applications, the maximum anticipated required burnups for 10,000 hours operation will be 4000 MWd/T. This corresponds to a swelling rate of less than 0.05% on the diameter. For the 1" diameter non-thermionic fuel rod this corresponds to less than 0.5×10^{-3} in.

f. UO_2 - Clad Compatibility

The compatibility of the fuel and clad must also be assessed. The extent of interaction between UO_2 and stainless steel is dependent on the initial stoichiometry in the fuel. Lauritzen⁽⁵⁾ has conducted studies on the compatibility of the mixed oxide (80% UO_2 - 20% PuO_2) with potential cladding material for fast breeder reactor applications. Capsules containing urania-plutonia of three different oxygen to metal ratios were maintained in intimate contact with the clad at an interfacial temperature of 760°C.

FIGURE 13. PREDICTED GAS-RELEASE BY "BUBBLE DIFFUSION" MODEL VERSUS EXPERIMENTAL RESULTS, FOR UO_2 FUEL RODS. (Ref. 3)

FIGURE 14. FUEL PIN SWELLING VERSUS BURNUP

The stoichiometries of the fuel were 1.97, 2.00, and 2.04 O/M and the five cladding alloys were types AISI 304, 316, 321, 347 stainless steel and Incoloy-800. The test duration was 1000 hours. The results of the metallographic examination are reproduced in Figure 15. These results indicate excellent compatibility, i.e. very little interaction between the fuel and clad of stoichiometric and hypostoichiometric fuels with all the austenitic steels tested. The capsules containing hyperstoichiometric fuel sustained minor attack to the cladding. Because of the chemical similarities between urania and mixed oxides the extent of interaction between urania and stainless steel is expected to be essentially the same as the mixed oxide. By controlling the initial stoichiometry of the fuel to an O/M of 2.00 or less, oxidation of the stainless claddings can be minimized.

Recent findings indicate that at burnup of 40,000 MWd/Te and over, the volatile fission products Cs and Rb may begin to concentrate in quantities sufficient to cause intergranular attack of the clad. No evidence of attack from this cause has been seen at the much lower burnups of 4000 MWd/Te required for the thermionic reactor experiment. This leaves an adequate margin of safety and permits the non-thermionic fuel elements to comfortably meet the 10,000 hour design goal.

g. Irradiation Damage to Clad

Exposure of stainless steels to a fast flux environment causes swelling due to void formation and irradiation induced mechanical property changes. Carlander,⁽⁶⁾ et al have measured the tensile properties of AISI 304 stainless steel along with immersion density measurements to determine irradiation induced swelling. The material was irradiated to a peak exposure of 4.8×10^{22} n/cm² at temperatures ranging from 371 to 463°C in the EBR-II reactor.

FIGURE 15. INTERACTION OF AUSTENITIC STAINLESS STEEL WITH MIXED OXIDE FUEL IN THE PRESENCE OF SODIUM. COUPLES ANNEALED FOR 1000 HOURS AT 1400°F. ALL PHOTOMICROGRAPHS: AS-POLISHED, 500X (Ref. 5)

Both the mechanical property changes and void formation are a function of the fluence and irradiation temperature. The void formation is strongly temperature dependent with the formation beginning at temperatures greater than 350°C, attaining a maximum at 500°C, and falling back to zero at some higher temperature. The void volume fractions were found to rise exponentially with fluence. There is the possibility of a nucleation time required for void formation and the voids do not appear to form at fluences less than 10^{22} n/cm² (6.7). For the thermionic reactor experiment, the clad is expected to receive a fluence of 1×10^{21} n/cm² for 10,000 hours of operation. This is an order of magnitude lower than that required for the initiation of void formation.

The irradiation damage in a fast reactor environment, where the neutron energy exceeds 1 MeV in the temperature range of 400°C to 600°C is characterized by a homogeneous distribution of large dislocation loops and nearly empty voids. The effect of the fast neutron damage on the mechanical properties of 304 SS is dramatic. The yield strength rises rapidly with fluences greater than 1×10^{22} n/cm² and does not appear to saturate with increasing neutron exposure up to 4.8×10^{22} n/cm². The tensile strength also increases over this fluence range but not nearly as rapidly as the yield strength as shown in Figure 16a.

Figure 16b shows the effect of test temperature on 304 SS irradiated to fluences near 2×10^{22} n/cm² at temperatures of approximately 463°C. The yield strength of the unirradiated material is essentially independent of temperature, whereas the yield strength of the irradiated material decreases with increasing test temperature until 750°C at which temperature the irradiated yield strength is approximately the same as the unirradiated material. The decrease in irradiated yield strength with increasing test temperature is due to an annealing out of the irradiation induced defects. A similar decrease in the tensile strength with increasing test temperature is shown in Figure 16c.

FIGURE 16a

The True Yield Strength and Fracture Strength (in thousands of psi) of Type-304 Stainless Steel at a Test Temperature of 450°C vs Neutron Exposure At Irradiation Temperatures ~ 450°C.

Effects of Test Temperature on the True Fracture Stress of Type-304 Stainless Steel Irradiated at 463°C

FIGURE 16b

The Effect of Test Temperature on the True Yield Stress of Type-304 Stainless Steel Irradiated to Fluences near $2 \times 10^{22} n/cm^2$ at Temperatures ~ 463°C.

FIGURE 16c

FIGURE 16

For TREX applications, the clad is expected to see a fluence of 1×10^{21} n/cm² for 10,000 hours of operation. On the basis of these data an increase in yield strength in the temperature range of 550 to 650°C can be expected to be less than 10% and the increase in tensile strength less than 15%.

The effect of test temperature and dose level on the true strain for AISI 304 is shown in Figure 17a. The true strain is a marked function of the fluence level and at 2×10^{22} n/cm² the ductility is markedly reduced. At 1×10^{21} n/cm² the ductility can be expected to be decreased by 20%. Figure 17b shows the same trend in reduced ductility with increasing fluence at one particular test temperature, namely 450°C.

The major conclusions on the radiation induced mechanical property changes are that the lower irradiation temperatures result in higher strength and lower ductilities for a particular neutron exposure. The effect of increased neutron exposure at any given temperature, results in sharply increased yield strengths and corresponding decreased ductilities. At the neutron exposure levels for the non-thermionic fuel for TREX for 10,000 hours, the change in the tensile and yield strengths are expected to increase less than 15% with an accompanying loss in ductility of less than 20% of the unirradiated values.

h. Liquid Metal Coolant - Clad Compatibility

With the outside of the clad in contact with a Na-44 wt. %K liquid metal coolant, the interactions and corrosion between the clad and coolant must be considered. There is a great amount of information available on the behavior of the austenitic stainless steels in a sodium environment in the temperature range applicable to TREX. Typical kinetic corrosion curves for the stainless steels and Incoloy-800 in 700°C sodium containing less than 10 ppm oxygen are shown in Figure 18.⁽⁸⁾ The corrosion process is characterized by weight losses with an initial decreasing rate law followed by long-term linear kinetics.

FIGURE 17a (Ref. 6)

EFFECT OF TEST TEMPERATURE ON THE TRUE STRAIN
OF TYPE-304 STAINLESS STEEL IRRADIATED AT 463°C

FIGURE 17b (Ref. 6)

THE EFFECT OF NEUTRON EXPOSURE AND IRRADIATION
TEMPERATURE ON THE TRUE UNIFORM STRAIN OF
TYPE-304 STAINLESS STEEL DURING TESTING AT 450°C

FIGURE 17

FIGURE 18 CORROSION RATES OF MILL ANNEALED STAINLESS STEELS IN 1300°F SODIUM
 (Ref. 8)

The steady state corrosion rate for the AISI 300 series stainless steels under these conditions varied between 0.4 to 0.6 mil/yr. Little or no difference in the corrosion rates are expected by replacing sodium with NaK-44 as supported by the work of Devan⁽⁹⁾ who found the corrosion rates of Inconel in NaK-44 to be very similar to those in sodium.

The compositional changes of the surface layers in Type 316 stainless steel after 2000 hours exposure to 700°C sodium is shown in Figure 15. Chromium and nickel are depleted at the surface to a depth of approximately 0.2 - 0.4 mils with the resulting enrichment of iron. This depletion in chromium and nickel reduces the stability of the austenitic phase and causes a transformation to a ferritic layer at the surface. The effect of this thin ferritic layer on the mechanical properties is negligible⁽¹⁰⁾. The compositional changes at the surface occur fairly rapidly and result in the initial changing rate law. Once the ferritic layer is established, the linear kinetics take over and the ferritic layer does not appear to grow with time.

(1.) Effect of Oxygen on Corrosion Rates

The single most important parameter influencing the behavior of the Type 300 stainless steels in sodium or NaK is the concentration of oxygen in the liquid metal. The effect of oxygen concentration and temperature on the corrosion rate of the various stainless steels is shown in Figure 20a and the velocity dependence in Figure 20b.⁽¹¹⁾ The maximum clad temperature for the driver fuel will be 633°C with the NaK flow in the fuel channels of 5 - 7 ft/sec. On the basis of these data, the maximum corrosion rate expected is approximately 0.3 - 0.4 mil/yr at the 10 ppm oxygen level. This is a reasonable oxygen level. Maintenance of a 10 ppm oxygen level in large reactor systems has been demonstrated⁽¹²⁾. As a general rule, corrosion rates of 1 mil/yr or less are tolerable in large reactor systems.

1 mil

FIGURE 19. MODIFIED COMPOSITIONS OF GRAIN BOUNDARIES AND SURFACE LAYERS IN TYPE 316 STAINLESS STEEL AFTER 2000 HOURS EXPOSURE TO 1300°F SODIUM 4,5 (Ref. 8)

NP 2401

FIGURE 20a (Ref. 11)

FIGURE 20b (Ref. 11)

Effect of Velocity on the Rate of Metal Loss
from Various Stainless Steels (bore dia constant)

FIGURE 20. EFFECT OF TEMPERATURE AND O_2 LEVEL ON RATE OF METAL LOSS
OF VARIOUS STEELS IN FLOWING SODIUM

Further reductions in the corrosion rates may be achieved by lowering the oxygen level to 5 ppm, or lower, by hot trapping of the liquid metal. At this oxygen level the corrosion rate decreases to 0.1 mil/yr.

(2.) Mass Transport

In liquid metal systems the phenomenon of mass transport has been observed. Of particular concern is the carbon transport between the same material at different temperatures in the liquid metal system or between dissimilar metals because of the possible effects on the mechanical properties of the cladding or unstabilized structural materials. In monometallic stainless steel systems the carbon transport results in decarburization in the hot temperature regions and carburization in the low temperature regions. The effect of the carbon on mechanical properties is to decrease the yield and creep strengths and increase ductility in the hot regions whereas the mechanical properties are influenced in an opposite way in the low temperature regions. However, extent of carbon transport is not well defined nor is information available on the movement of carbon between stainless steel and Ceramvar.

i. Summary

The use of UO_2 fuel in a stainless steel clad for the non-thermionic fuel for TREX is summarized below. The fuel will be required to see a burnup of approximately 0.5 at.% for a maximum anticipated lifetime of 10,000 hours. With such a low burnup, the buildup of both solid and gaseous fission products is small and swelling, as a result of these fission products, will be negligible. The compatibility of UO_2 with the AISI 300 series stainless steel is good. Interactions between the clad and the volatile fission products, Cs, Rb, and I have not been observed at these low level burnups.

At a fluence of 1×10^{21} n/cm², the maximum fluence the clad will see, swelling due to void formation in stainless steel is not observed. The start of swelling is seen at a fluence level of a decade higher.

The change in the mechanical properties as a result of irradiation damage is expected to be small. The yield and tensile strengths are expected to increase less than 15% and the loss in ductility less than 20% of the unirradiated values.

The corrosion behavior of stainless steel in Na-K 44 in the temperature range of 550 - 650°C at oxygen levels of 10 ppm is very good. Corrosion rate of less than 0.5 mil/yr are expected. The extent of carbon transport and its effect on mechanical properties must be further evaluated.

The results of this preliminary investigation indicate that the non-thermionic fuel elements will meet the 10,000 hour design goal with adequate margin.

2. Internal BeO Reflector Rods

The reactor core is designed with a row of BeO reflector rods surrounding the non-thermionic fuel or TFEs. The external geometry of the BeO reflector rods is the same as the fuel elements to facilitate the handling of the rods. The temperature of reflector rods will be maintained the same as the rest of the core by controlling the coolant flow through the reflector process tubes. The reflector rods, being at the edge of the core, will see a fluence of a 3×10^{20} n/cm² for 10,000 hours exposure.

The reflector rods will consist of pressed BeO pellets in a stainless steel clad. The irradiation and corrosion behavior of the clad has been assessed in the non-thermionic fuel discussion. The radiation effects in BeO will be discussed.

Irradiation of BeO results in dimensional and volumetric changes by three mechanisms: 1) expansion due to the accumulation of interstitials and vacancies as both point defects and clusters; 2) swelling caused by the accumulation of He and oxygen gases formed or released in transmutation reactions; and 3) microcracking, i. e., grain boundary separation.

Expansion is the predominant growth mechanism, in the absence of microcracking, at dose levels up to approximately 10^{21} n/cm². The accumulation of He, H₂³ (tritium), and O₂ gases in irradiated material also contributes to the bulk volume expansion under non-microcracking conditions. He and H₂³ are formed in the (n, 2n) and (n, α) reactions with Be and the destruction of Be results in an excess of oxygen which is assumed to enter the gaseous state. Most of the gases formed are retained in the bulk (70 - 90%) in non-microcracked material in the form of inter and intragranular bubbles.

The swelling of BeO is a function of the total fluence, temperature of irradiation and grain size. The resistance to microcracking is extended through the use of fine grained material. The irradiation temperature is an important parameter in determining the swelling behavior. Higher irradiation temperatures lower the swelling for any given dose level.

Radiation induced volume changes are shown as a function of fluence and irradiation temperature in Figure 21 and is based on randomly oriented specimens with a grain size of 5 to 20 microns in diameter. The volume expansion, expected for BeO, based on 10,000 hours exposure accumulating a fluence of 3×10^{20} n/cm², is less than 0.5 volume percent in the temperature range of 500 - 600°C. This amount of swelling can be accommodated through the proper dimensional design of the reflector rods.

The results of this study lead to a high level of confidence in the ability of the reflector rod to exceed a 10,000 hour design life.

FIGURE 21. VOLUME EXPANSION IN RANDOMLY ORIENTED SPECIMENS
OF 5 - TO 20 - MICRON GRAIN SIZE

3. High Worth Driver Fuel Elements

a. Introduction

If required, as one of the possible reactivity adjustments, high worth driver fuel elements can be used in conjunction with the thermionic fuel elements. These driver fuel elements would be compatible with the process tube design as described previously.

b. Design Considerations

Most of the parameters previously discussed for non-thermionic fuel elements are applicable to high worth drivers. A special consideration is the thermal design of the driver fuel elements due to the highly enriched UO_2 .

In calculations of thermal performance of stainless steel clad UO_2 fuel elements of the SEFOR type, the liquid metal film drop and the clad temperature drop may be neglected since they amount to only 10°K or 20°K . The temperature drop in the fuel is best calculated from experimental values of $\int k \, dT$. The most recent data for UO_2 available has been obtained from Reference 13.

The heat transfer relationship used is given by Equation (1).

$$q = \frac{\frac{2}{r_o} \int_{T_o}^{T_i} k(T) \, dT}{1 - \frac{2r_i^2}{r_o^2 - r_i^2} \log(r_o/r_i)} \quad (1)$$

In this equation q is the fuel surface heat flux, T , the fuel temperature and r a radial position in the fuel. The subscripts o and i refer to the outside and inside surfaces respectively. In the calculations performed r_i has been taken to be zero for a solid pellet. The denominator of Equation (1) was, therefore, unity in the calculations described here. Data for a fuel density of 95% were used.

In the treatment of heat transfer across the gap between the fuel and the clad, gas conduction and thermal radiation were included. The following relationships were used:

$$q = q_R + q_c \quad (2)$$

$$q_R = \sigma \left[\epsilon_{UO_2} (1 - \epsilon_{ss}) T_o^4 - \epsilon_{ss} T_{ss}^4 \right] \quad (3)$$

$$q_c = \frac{k_{mix}}{\Delta x} (T_o - T_{ss}) \quad (4)$$

$$k_{mix} = k_o \left[1 - \frac{\alpha}{2} (T_{UO_2} + T_{ss} - 2000) \right] \quad (5)$$

$$k_o = \frac{1}{2} \left[\frac{1}{k_{He}} \left(\frac{1}{P} \right) + \left(\frac{1-P}{P} \right) \left(\frac{F_{xe}}{k_{xe}} + \frac{F_{Kr}}{k_{Kr}} \right) \right] \\ + \frac{1}{2} \left[\left(\frac{1}{P} \right) k_{He} + \left(\frac{1-P}{P} \right) (F_{xe} k_{xe} + F_{Kr} k_{Kr}) \right] \quad (6)$$

Several assumptions are implicit in these relationships. It was assumed that the fuel element is charged with helium at one atmosphere and has a hot fuel/clad radial gap of Δx . The calculations are done at the end of life when the combined fission gas plus helium pressure has built up to P atmospheres. The temperature dependence of the gasses has been assumed to be linear and the same for each gas over the temperature range studied. The radiation calculation assumes that the emissivity of the UO_2 is sufficiently high that reflection from it may be neglected.

The values of the constants used in the calculations are given in Table 2. The calculations using the data in Table 2 assume the maximum temperature occurs at the center of the core, a good approximation. Hot spot factors have not been included but these may appropriately be included as an uncertainty allowance in the heat flux rating of the fuel element. Central fuel temperatures at all times prior to end of life will be lower than those corresponding to these calculations.

Two factors not accounted for here will reduce central fuel temperatures at end of life to a significant degree. First, the fuel sinters to higher densities and forms a small central cavity. This increases the values of $\int k dT$ and results in $r_i > 0$. Second, fuel swelling will reduce the hot fuel/clad gap and increase the gas conduction in the gap.

The results of the calculations are shown in Figure 22. The results clearly show the incentive for incorporating sufficient fission gas plenum volume to reduce end of life pressures to the 2 atmosphere level. In addition, end of life hot gap values of 4 mils or less are important.

TABLE 2

$r_o = 1.1 \text{ cm}$	(Radius of Fuel Pellet)
$\epsilon_{\text{UO}_2} = 0.7$	(Emissivity of UO_2)
$\epsilon_{\text{ss}} = 0.3$	(Emissivity of Stainless Steel)
$T_{\text{ss}} = 850^\circ\text{K}$	(Stainless Steel Clad Temperature)
$\Delta x = 0.01 \text{ cm}, 0.02 \text{ cm}$	(Radial Increment)
$k_o = 3.3 \times 10^{-4} \text{ w/cm}^\circ\text{K} \text{ (p = 10 atm)}$	(Combined Ke/Xe Conductivity)
$10.6 \times 10^{-4} \text{ w/cm}^\circ\text{K} \text{ (p = 2 atm)}$	
(Reference 14)	
$F_{\text{Xe}} = 0.854$	(Mole Fraction of Xenon)
$F_{\text{Kr}} = 0.146$	(Mole Fraction of Krypton)
$\alpha = 0.0006$	(Temperature Coefficient of Thermal Conductivity)

FIGURE 22. HEAT TRANSFER IN NON-THERMIONIC FUEL

It should also be noted that increased heat transfer capability could easily be provided by reduction of the fuel radius to lower values (e. g. one centimeter). This reduction will increase the allowable heat flux in an approximately linear proportion while reducing the actual delivered heat flux in a specific reactor design in proportion to the square of the radius. This powerful variation indicates that it will be a relatively simple design problem to size a high worth driver to be consistent with flux requirements while maintaining centerline fuel temperatures considerably below the melting temperature.

C. CORE INTERNALS

1. Introduction

The core internals include all of those components inside the reactor vessel except for the TFEs, NTFEs and reflector rods. The design objectives laid out for the core internals are listed as follows:

- a. Prototypic of the space thermionic reactor where appropriate.
- b. Provision for replacing TFEs within the core.
- c. Replaceable core internals.
- d. Utilize state-of-art components wherever possible.

In some instances, these design objectives resulted from the ground rules placed on the TREX experiment; others were self-imposed. In the course of the TREX study, a conceptual design was developed which is presented in this section. This design is one way of responding to the design objectives.

The core internals will be discussed in the following order; process tubes, tube bundle and core shroud. Figure 23 shows the general arrangement of the core internals. Figure 24 delineates a vertical cross-section of the assembled core.

2. Process Tubes

The core structure is composed of a triangular pitch array of process tubes. The process tubes provide flow channels into which the fuel elements are inserted. The bottom end of the process tube is orificed to enable each process tube to be individually adjusted for flow conditions. This is an important design feature for it allows the flow in an individual channel to be optimized for a TFE, NTFE, internal reflector rod or even a high worth driver fuel element if the latter is used.

This optimization of flow could be important in maximizing thermohydraulic stability of the fuel element within the process tube. This stability phenomenon is discussed in section II.F.3. Another feature which maximizes this stability is the location of the supports between the fuel element and its individual process tube. These supports will be located to maintain concentricity within the tube, minimize load inducing structural restraints and maximize thermohydraulic stability.

FIGURE 23. THERMIONIC REACTOR CORE AND VESSEL

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Figure 24. Thermionic Reactor Core and Vessel Showing Cross-Section of Assembled Core

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The overall length of the process tube is approximately 13 feet.

The fuel handling machine mates with the upper end of the process tube. The NaK coolant flow through the core enters the bottom of the process tubes and exits through holes which are provided in the process tube wall above the active core. The NaK then flows radially from inner regions of the core toward the outer plenum where the coolant exits from the reactor.

3. Core Tube Bundle

The core structure of the TREX reactor is composed of a bundle of process tubes. Each of these tubes is removable. Triangular pitch arrangement of the process tubes is shown in Figure 24.

A nominal clearance of 0.003 inch is provided between process tubes.

Therefore, each process tube sees a channel which is defined by the six surrounding process tubes. The process tubes will tend to restrain each other. The bundle of process tubes is contained within a core shroud which has a hexagonal shape as shown in Figure 25. The inside dimension of the hexagonally shaped shroud is slightly larger than the nominal distance across the appropriate number of process tubes. As shown in Figure 24, the flow baffle surrounding the hexagonal shroud forms the inner wall of the downcomer region, which is the coolant inlet annulus. Therefore, the shroud operates at the inlet coolant temperature. The process tubes themselves will operate at a temperature representative of the core average coolant temperature, which is higher than the core inlet. With proper selection of materials, it is anticipated that this favorable thermal expansion characteristic can be used to advantage as a mechanism for core clamping.

It is possible that core clamping through the use of differential expansion is not an adequate means. Mechanical core clamping arrangements have been conceived which utilize the sector shaped areas between the hexagonal shroud and the flow baffle for a means to exert a force on the flats of the shroud.

It is anticipated that the process tubes will possess sufficient flexibility to allow removal and reinsertion with the 3 mil nominal clearance between process tubes. However, process tube removal and reinsertion must be demonstrated through a realistic environmental test.

Center-to-center distance between process channels within the core region is nominally 1.133 inches. The process tubes are for all practical purposes touching each other and a larger center-to-center pitch (1.375 inches) at the top of the shroud was provided. This eased space demands for electrical and mechanical connection to the top end of the TFE and process tube and provided a ligament in the top grid. Skewed cores, process tubes which are cantilevered from the core region and conical shaped cores were considered for accomplishing the desired pitch at the top of the shroud while maintaining a tight pitch in the core region. A conical shaped core was selected, however, the final configuration will depend on further design studies. In particular, if the electrical and mechanical connection to the TFE and process tube can be accomplished in a smaller space and a vessel top plate arrangement be designed to provide an adequate reactor vessel closure, it would be highly desirable to have a straight core with its reactivity gains and advantages of simplified refueling machine design. In the concept selected for this cost study, the tube pitch changes from 1.133 inches at the bottom of the active core to 1.144 inches at the centerline of the core to 1.155 inches at the top of the active core. The angle from process tube to process tube is approximately six minutes. This angle accumulates with each row of process tubes from core centerline. However, the centerline of the outer row of process tubes is less than one degree from core centerline. Both cylindrical process tubes and process tubes with hexagonal outer surface and a cylindrical inner surface were examined. Again, on the basis of mechanical simplicity, the cylindrical process tube was selected. The lower end of the process tube incorporates either a taper or stepped diameters so that the process tubes may be nested diameter to diameter within the active core region.

4. Core Shroud

The core shroud consists of a cylindrical tube with varying diameters and grid plates at each end. The core shroud fits within the reactor vessel. A design study of the core shroud is shown in Figure 25. Along with the inner wall of the reactor vessel the core shroud defines the inlet and outlet plenum in the downcomer region. The inlet plenum is separated from the outlet plenum

Figure 25. TREX Core Shroud Design Study

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by a Belleville washer arrangement which limits leakage between the higher pressure inlet plenum and the lower pressure outlet plenum. It has been judged unnecessary and possibly even undesirable to attempt to completely seal the two plena from each other. As shown in Figure 25, spacers are incorporated on the outer diameter of the core portion of the core shroud. These spacers maintain concentricity between the cylindrical flow baffle of the core shroud and the inner wall of the reactor vessel. The upper spacers shown in Section EE of Figure 25 perform a secondary role of orificing flow into the downcomer region. By placing a modest flow impedance between the low velocity plenum region and the somewhat higher velocity downcomer region, the downcomer flow will be well distributed.

The lower grid plate is welded into the lower end of the shroud. This is shown in Figure 24. This grid performs several functions. It provides a positive location for the lower end of the process tubes. A second function is to accommodate differential thermal expansion between the process tubes and the shroud. The upper end of the process tubes are fixed into the upper grid plate of the shroud. Therefore, there will be differential movement of the process tube nozzle in the lower grid plate hole. Clearance between the process tube nozzle and the bottom grid plate holes will be sized to allow some bypass flow through the interstices between the process tubes in the core.

The need for a cooling system to maintain desired temperatures in the upper end of the TFE handle was discussed in II. A. 3. This capability is provided by a helium cooling gas system. The portion of the helium system within the reactor vessel will be discussed here. The balance of the helium system will be discussed in II. J. 2. Helium cooling gas flows into a plenum between the shroud and the reactor vessel. Helium flows inward into a second plenum between the two grid plates at the upper end of the core shroud. The helium flows into holes provided in the upper end of the process tube. The gas then flows down the annulus between the TFE handle and the inside diameter of the process tubes. At the level just above the NaK coolant overflow line, a second set of

holes are provided in the process tube to permit the helium gas to escape and flow over to the helium outlet. The clearance between the diameter of the holes in the lower of the two upper grid plates and the corresponding diameter on the process tube is maintained small enough to limit leakage of helium flow down the interstices between the process tubes. The extreme upper grid plate has step diameters with the larger diameter at the upper end. Both of these diameters are larger than the diameter of the holes in the lower of the two upper grid plates. Once unseated from the three, step-diameters the process tube has very generous clearances for the remaining withdrawal. In the upper end of the shroud it may be necessary to incorporate a shield grid which will reduce neutron streaming and gamma streaming through the interstices of the process tubes.

Prior to installation in the reactor vessel, the process tubes will be loaded into the shroud. Except for instances where the entire core must be unloaded at once, all process tube replacements will be made on a one-for-one basis so that during process tube insertion, the process tube will always see a defined channel into which it is inserted. From Section AA in Figure 24 it can be seen that thimbles can be contained in the core at the corners of the hexagonal shape core shroud. Thimbles can be used for instrumentation access to the bottom plenum region or other locations along the core. At two locations, hexagonal shaped baffles are provided to restrain the upper end of the process tubes as shown in Figure 24.

A variety of 300 series stainless steels may be used for fabricating the core shroud. These materials have been shown to possess excellent corrosion resistance in the expected coolant environment and commonly have been fabricated into geometries typical of the core shroud. Welding requirements and material specifications will be specified to meet Section III of the ASME Boiler Code.

5. Summary

The core internal structure is composed of nested process tubes mechanically restrained by a shroud structure. The process tubes are inclined relative to each other so that a 1.375 inch pitch is maintained at the top of the shroud with a 1.144 inch pitch at the centerline of the active core. The process tubes are removable from the reactor vessel. A helium cooling system is provided to cool the upper end of the TFE handle. The core shroud is part of the flow baffle separating the down-flow from the up-flow core regions and the inlet and outlet plenum. The shroud and process tubes have a simple cylindrical geometry which is easily fabricated.

D. REACTOR VESSEL AND ELECTRICAL CONNECTION DOME

1. Introduction

The reactor vessel provides the receptacle in which the thermionic reactor core is placed for test. It contains the NaK coolant and an inert cover gas for the coolant. The top flange of the reactor vessel extends a few inches above the test pit operating floor. The electrical connection dome is functionally part of the reactor vessel and is its top closure. The dome contains the mechanisms for connecting power leads and instrumentation leads to the TFE handle. The dome has a large valve on top for fuel handling machine access to core element handles. The design objectives for the reactor vessel and electrical connector dome are as follows:

- a. Ease of fabrication and installation using state-of-the art components.
- b. Simple, reliable electrical and mechanical connections.
- c. Conformance to Section III of the ASME Boiler Code and other applicable codes.
- d. Containment of certain nuclear excursions.
- e. Protection against core draining.
- f. Helium leak tight.

In the next section design considerations for the reactor vessel are discussed. The specific geometry presented is a reasonable way of meeting the design objectives. Section II. D. 3 will describe the electrical connector dome and how it functions. Figure 23 shows the general arrangement of the reactor vessel and electrical connection dome.

2. Reactor Vessel

The TREX reactor vessel will be designed to meet the requirements of Section III of the ASME Boiler Code for nuclear vessels. The design conditions have been established as follows:

- a. Pressure - 200 psig
- b. Temperature - 1300°C
- c. Life - 10 Years
- d. Thermal cycles - 400
- e. Earthquake load - horizontal - 0.5 g

The nominal operating conditions for the first TREX reactor vessel are as follows:

- a. Pressure - 10 psig
- b. Coolant temperature - 1160°F
- c. Interior coolant - Na-44K
- d. Exterior coolant - N₂ with maximum 4% by vol. O₂
Dew point + 12°F

The TREX reactor vessel is a simple cylindrical vessel approximately 13 feet long. In the active core region the reactor vessel is approximately 21 inches outside diameter. The portion above the core region has an o.d. of approximately 24 inches. The double wall vessel in the core region assures vessel coolant containment. Each of the double walls is approximately 1/4 inches thick. The space between the two vessels will be monitored for leaks in either wall. The wall thickness of the reactor vessel will be determined by handling considerations, fabrication or by need for containment of certain nuclear excursions. Boiler code requirements are likely not the limiting case for wall thickness, except in the case of nozzle reinforcement.

There are three (3) 3 inch schedule 40, coolant inlet nozzles spaced 120° apart. Approximately 7 inches higher on centers are three (3) 3 inch schedule 40, coolant outlet nozzles, also spaced 120° apart but staggered 30° from the inlet nozzles. These penetrations can be seen in Figure 23. Other penetrations of the reactor vessel include a 3 inch overflow nozzle, a 2-1/2 inch helium outlet nozzle, a 2-1/2 inch helium inlet nozzle and a smaller penetration for the coolant level indicator. A short distance above the active core region, a flange is welded to the inside of the reactor vessel which mates with a flange on the core shroud. A Belleville spring provides the flow barrier between the inlet plenum and the outlet plenum. Thermal baffles may be provided to reduce thermal stresses in the nozzles and in the internal flange area.

The reactor vessel will be fabricated of a 300 series stainless steel. There are a variety of such steels which are fully qualified from the

standpoint of fabricability and of compatibility with the liquid metal coolant environment. Ordinary fabrication methods will be used for the reactor vessel.

3. Dome and Electrical Connections

a. Introduction

The electrical connection dome is mounted to the upper flange of the reactor vessel. In the dome space electrical connections may be made to all of the potential 211 TFE units. Figure 26 shows a general layout of the volume. Connection is made by multiple contact bars which are hinged so they may be moved aside to provide access for refueling. Each connector bar is equipped to make connection to 8 TFEs. Although many of the connector bars actually service fewer units, only two sizes will be used to provide interchangeability. Two sizes rather than one are used since the close spacing requires a staggering of the vessel penetrations.

The connector bars must be at two positions relative to the vessel wall because of the hexagonal pattern. (See Figure 27) For core elements other than TFEs, the connector bars are used only in a hold-down function.

The top access to the dome volume is through a thirty inch diameter gate valve. This access is provided for element exchange operations employing the fuel handling machine. The dome volume contains helium at a slightly positive pressure.

b. TFE and Process Tube Handles

The TFE handle assembly is designed to serve three functions. First, it forms one side of an electrical connector through which electrical power and instrumentation signals from the TFE are carried to the Control Room. The concept chosen for

FIGURE 26.

FIGURE 27. ARRANGEMENT OF CONNECTOR BARS (PLAN VIEW) 1/4 SCALE

this design is shown in Figures 28 and 29. Low current signals are transmitted through seven spring connectors imbedded in plastic for insulation.

The TFE power output is carried through a large area contactor which is spring loaded for positive, reliable contact. These terminals are on the inside of a slotted plastic cylinder which forms the top of the TFE handle.

The second function of the TFE handle assembly is to positively locate the top of the TFE in the process tube. This is accomplished by the slotted hexagonal sleeve which nests into the process tube handle. (See Figure 30).

The third function is to provide a locating and lifting feature for the grapple of the fuel handling machine. This is necessary for the individual removal of TFEs or TFEs and process tubes from the core.

c. Electrical Connector Bars

Power and instrumentation leads are carried through the electrical connection at the top of the TFEs to the electrical connector bar. This bar may be an epoxy casting with copper contactors and conductors cast in place. The epoxy blade has a steel structural support. The connector bar is designed to engage a maximum of eight TFEs by mating the slot previously defined in the top of the TFE (See Figure 28). The high current contact is at the bottom of the connector bar acting against the spring loaded copper contact in the TFE handle. The low current contacts are on the sides of the connector bar at two elevations.

The connector bar is actuated by a toggle action linkage illustrated in Figure 31. It is manually operated by a push-pull shaft. The shaft may have either a sliding seal plus a static seal in the clamped position or a bellows seal.

FIGURE 28. TFE HANDLE COMPONENTS

Figure 29. TFE Handle Assembly

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Figure 30. Assembly of TFE into Process Tube

FIGURE 31. Connector Bars in Electrical Connector Dome

d. Refueling and Maintenance Valve

The refueling and maintenance valve is in the top of the dome volume and provides access for the fuel handling machine to the top of the core. It is a thirty-inch diameter gate valve.

e. Containment Dome

The dome itself is approximately thirty inches by fifty inches by twenty inches high and is fabricated of 0.75 inch thick stainless steel. There is approximately a fifteen inch clearance between the top reactor vessel flange and the bottom of the gate valve.

E. REFLECTOR CONTROL CUP AND LIFT ASSEMBLY

1. Introduction

The reflector control cup and lift assembly includes the reflector control cup, a shield, the control drum actuators and the lift mechanism. The design objectives laid out for these components are:

- a. The reactor control must mockup a 4π shieldable flight reference design.
- b. Prototypic
- c. Functional design using state-of-art components
- d. Capability to raise and lower the reflector control cup

The conceptual design described herein is the product of a trade-off study of a number of design approaches to the design objectives. The subsystems will be described in order; reflector control cup, shield, control drum actuators and lift assembly. Figure 32 shows the general arrangement of this reflector control cup and lift assembly. A more detailed definition of the assembly is delineated in Figure 33.

2. Reflector Control Cup

The reflector control cup consists of the control drums, the reflector segments, the strain limiter cylinder, bearings, and other components.

a. Control Drums

The control drums are 6 inches in diameter, $27\frac{5}{8}$ inches long and are contained in a can of high nickel alloy or austenitic stainless steel. The nuclear poison portion of the control drums constitutes approximately 2 inches in the diameter in a chord geometry. The balance of the 6 inch drum diameter (4 inches) is composed of beryllium oxide. This is illustrated in Figure 34. Clearances will be provided between the internal components of the control drum and the outer cylindrical can to ease fabrication and accommodate thermal expansion differences between the drum and can. From the reflector and poison materials, the space between the drum and can will be filled with a high conductivity

FIGURE 32. CUP AND SHIELD ASSEMBLY

Figure 33. Detailed View of Cup and Shield Assembly

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FIGURE 34. SECTOR OF REFLECTOR/CONTROL CUP

material such as liquid metal or a gas to enhance heat transfer . Energy is passively rejected from the container can to surrounding components. Twelve control drums are located in a pattern surrounding the reactor vessel on equal centers, as shown in Figure 33.

b. Reflector Segments

A segment of beryllium reflector material is located between each control drum. In the transverse plane, this reflector segment has an hour glass shape as shown in Figure 34 and forms an approximately cylindrical channel around the control drums. Both the reflector segments and the control drums extend the length of the active core and the upper and lower axial reflector in the axial plane and is illustrated in Figure 33. In the axial section through a reflector segment on Figure 33, cuts were made on approximately 3 inch centers beginning on the end of the reflector segment facing the reactor vessel and through approximately 80 percent of the length of the segment. The purpose of the cuts is to minimize thermal distortion by accommodating differential expansion between the face of the segment adjacent the reactor vessel and in the face of the segment away at the exterior surface of the reflector.

The beryllium metal reflector segment may require coating (beryllide) or canning to protect it from corrosion in the nitrogen environment (~4 volume percent oxygen). A spacing structure is provided (as shown in Figure 34) on the side of the reflector segment facing the reactor vessel. This surface may also be coated or require another surface material to minimize friction and damage when sliding along the reactor vessel. Clearances between the reflector segments and the control drums, and between the reflector segments and the reactor vessel will be ample to assure freedom of motion.

c. Strain Limiter

A 1-1/2 inch thick steel cylinder is mounted at the periphery of the reflector segments. The purpose of this cylinder is to limit straining of the reactor vessel in the event of certain nuclear excursions. In the event of such an excursion, the reactor vessel wall load would be transmitted through the reflector segments to the strain limiter.

The reflector segments are mechanically fastened to the strain limiter in the concept shown. A braze-type bond would be provided between the reflector segments and the strain limiter to enhance heat transfer from the reflector segments. The reject heat is conducted to the outside surface of the limiter and into cooling fins (see Figure 33).

The strain limiter material should closely match the thermal expansion coefficient of beryllium metal. 310 stainless steel may meet this requirement. The lower end of the strain limiter has been reduced in thickness to accommodate differential expansion between the limiter cylinder and the shield plug to which it is fastened (see Figure 33).

d. Bearings

Self-aligning bearings are provided at each end of the control drum. The demand placed on bearings by a control system of this type is different from those of other bearing applications. In this instance, the drums will stay in one static position for long periods of time, after which small motions are necessary. In this type of application, the initial friction of the bearing is of significant importance; whereas, in the usual high speed bearing applications, friction is important. The bearing design tentatively selected has been developed by Atomics International for reactor control systems of the type used in TREX. In this configuration,

the control drum bearing shafts rotate within the cylindrical bore of a ball bushing. The ball bushing is contained within a spherical socket. The ball can also rotate within this socket to serve as a secondary rotational bearing if necessary. The socket is composed of two caps as shown in Figure 33. The bearing design accommodates differential expansion between the control drums and the strain limiter (see Figure 33). The self-aligning feature allows angular displacement of the control drum bearing shafts, resulting from thermal bowing of the control drums.

The bearing material for the ball will be selected on the basis of further evaluation. Candidates include graphite and cermet materials. This type of bearing has been operated on endurance runs as long as 13,500 hours for total cumulative exposures of 225,500 hours under operating conditions at least as demanding as those anticipated for TREX.

e. Assembly

Components described in the four preceding subsections are fitted together to form the reflector control cup assembly. The control drums and bearings are assembled in a unit consisting of the reflector segments and the strain limiter cylinder. Heat shields are then attached and the strain limiting cylinder fastened to the shield plug.

The shaft through the shield is connected to the lower end of the control drum shaft by means of a bellows-type, flexible joint. Bellows flexible joints are suited to this application because of high accommodation of both axial and angular misalignments, while maintaining a high degree of torsional rigidity. Thus, little play will be introduced into the control drive linkage.

A duct is then fitted around the strain limiting cylinder as shown in Figure 33. This duct directs the flow of cooling nitrogen upward past the cooling fins brazed on the strain limiting cylinder. A portion of the nitrogen flow is ducted inward and, by means of the heat shields, is directed past the lower bearings. The flow then passes the control drums and the reflector segments. The upper heat shield directs the nitrogen around and over the upper bearings. By maintaining bearing temperatures at 500°F or below, a variety of bearing materials may be used.

3. Shield

As noted in Section 2d. and 2e, the reflector control cup is fastened to a shield plug. The shield plug has two functions. First, it reduces to a low level the neutron activation of the actuator position sensors and position switch components below the shield plug. Second, the shield plug acts as a gamma ray shield to allow manned access into the region below the shield plug during reactor shutdown.

Final dimensions, geometry and composition will be determined by detailed design. The shield plug is approximately 30 inches thick with the upper fifteen inches cylindrical in shape and the lower fifteen inches square. When the reflector cup is in the operating position (up) the shield plug fits within a space in the permanent concrete shielding below the reactor. Shafts are provided for each control drum as part of the control actuator linkage. Bearings are provided for these shafts at the top and the bottom of the shield plug. Provisions will be made in the shield plug for the instrumentation as needed.

4. Control Actuator Assembly

The control actuator assembly includes the actuator, scram mechanism, position sensor, and limit switches. It is desired that the TREX actuators be prototypic if reliable units are not prohibitive in cost. A

tentative selection was made of the control actuator developed by Atomics International for the SNAP reactors. Specifically, the mechanisms proposed here are essentially ambient temperature models of those used on the S8DR reactor. Conventionally insulated copper has been substituted for the bi-metal alumina insulated conductor in the motor, and standard material substituted for high temperature materials in the actuator and limit switches. These changes are expected to reduce the cost by about a factor of 2.

Tentative design specifications for the TREX actuator are as follows:

a. Actuator

1) Design Specifications

1.	Control Drum Weight	115#
2.	Control Drum Motion	700 steps over 180° travel
3.	Control Drum Speed	4 steps/second
4.	Actuator Step	One complete magnetic revolution of actuator (four current pulses, 1.8° rotation per pulse)
5.	Position Accuracy	± 0.2° at control drum
6.	Scram Time	180° in 500 msec., maximum
7.	Switch Indication	a) All in or all out b) Scram spring wound
8.	Operating Conditions	1. Temperature - 300°F max. 2. Atmosphere a. Gas-nitrogen b. Oxygen-4% by volume c. Dew Point- +12°F max. d. Pressure- 2.5 psig max. 3. Lifetime a. Time at Temp. - 30,000 hr. b. No. of steps- 1,000,000 c. No. of scrams, 10,000

The actuator, scram mechanism, switches and position indicator which meet the above specifications are described in the following sections:

a) Actuator

The actuator selected is a 4-pole, variable-reluctance stepping motor. An actuator step is a complete magnetic revolution of the actuator. Magnetic rotation consists of four current pulses each producing 1.8° rotation of the stepper motor rotor. The speed specification requires the actuator to operate at four magnetic revolutions per second. This corresponds to a stepper motor speed of 4 rpm. A 4 rpm actuator speed with a 1,500 oz.-inch inertial load and 6 A phase current will produce 105 oz.-inches of dynamic output torque. This output torque compares to 120 oz.-inches at stall with a 6 A phase current. This is sufficient torque to wind the scram spring, which is the maximum torque requirement demanded of the actuator. The output pinion on the stepper motor drives a spur gear which is, in turn, connected through the scram spring and clutch arrangement to the control drum. The gear reduction is selected to produce approximately 0.25° rotation of the control drum per actuator step.

Actuators of this type have been subjected to individual endurance tests as long as 16,000 hours with total exposure of 143,133 hours on a total of nineteen units. A reduction in actuator cost can be achieved by replacing the high-temperature conductors and insulators with materials equally effective, but of lower temperature capability.

b) Scram Mechanism

The scram mechanism is coupled in series with the actuator motor and the control drum. With the control drum full in (least reactive position), the scram spring is wound with the actuator motor mechanism. After winding the scram spring a limit switch signals that the procedure has been completed. A clutch coil is energized, which connects the scram spring housing directly to the control drum shaft. The wound and cocked scram spring is at that point locked in the control-drum drive. If the clutch coil is deenergized for any reason, the cocked scram spring immediately drives the drum into the least reactive position. A scram snubber will be provided to decelerate the drum at the end of its travel. This mechanism has been used successfully in the ground test of SNAP reactors.

c) Control Drum Position Sensor and Limit Switches

The control drum position sensor is a rotary, differential transformer which was developed for the SNAP reactors. As with the actuators, conductors and insulators used will be selected to take advantage of the lower operating temperature of the TREX application.

Limit switches are provided for full in, full out position control drums indication. A third switch is provided to signal when the scram spring is wound. In view of the modest operating requirements of TREX, a variety of switches are available which would be satisfactory for the application.

5. Shield Plug Lift Mechanism

An objective of the TREX control system is to allow removal of the reflector control cup from the core region to provide a secondary scram mechanism and an additional shutdown margin during refueling procedures. A second objective is to provide for maintenance or replacement of control system components or for the complete removal of the reflector control cup for major maintenance or replacement.

The lift mechanism consists of four, 2-1/2 inch diameter, ball bearing lead screws. The lower ends of the lead screws are mounted in self-aligning bearings embedded in the floor of the cell. Two lead screws are located on opposite sides of the shield plug. The upper ends of the four lead screws are mounted into a heavy steel frame. The shield plug has a heavy structural support mounted on the bottom side in each of the four corners. This support structure in turn rests on pick ups which are fastened to ball nuts on the lead screws.

The shield plug can be lowered by rotation of the lead screws until the weight of the shield plug and reflector cup assembly rests on a cart. At this point, the ball nuts no longer support the shield plug and may be lowered out of the way of the cart. The cart containing the shield and reflector cup then may be driven remotely on rails fastened to the floor of the reactor cell to a position directly below the access hatch in the operating floor.

When the reflector cup, shield assembly is returned to the location below the reactor, the ball bearing lead screws are actuated which pick up the reflector cup, shield assembly and raises it to a position just below the reactor vessel. Sensors will be provided to indicate the alignment between the reactor vessel and the reflector cup. The steel framing around the top of the ball bearing lead screws and in turn the reflector control cup assembly may be moved in an X and/or Y direction for alignment with the reactor vessel. The reflector control cup then may be raised to the normal maintenance position, a point at which the top of the reflector cup is level with the bottom of the active core region. Alignment between the reactor vessel and the reflector cup may be checked again, and if necessary, adjusted. The reflector control cup then may be raised into the operating position.

Maintenance of the control system components is achieved by lowering the reflector control cup to the point where the top of the reflector cup is level with the bottom of the active core. The space below the reactor then is purged with air to allow a personnel access.

6. Summary

The TREX reflector control arrangement consists of twelve drums surrounding the reactor core outside the reactor vessel. The control drums rotate within channels formed by segments of beryllium reflector material connected to a thick cylinder. In the event of a reactor excursion, straining of the vessel is limited by a fixed cylinder surrounding the reflector segments. The reflector cup is mounted on a shield plug which may be raised for operation and lowered for scram, maintenance or safety during reactor refueling operations. The reflector control cup may be replaced if required. Provision is made for maintaining alignment between the reactor vessel and the reflector cup as the reflector cup is raised to the operating position. Control drum actuators are similar to those developed specifically for space reactor use. These actuators and associated scram mechanisms, position indicators, and limit switches are state-of-the-art components and will require only confirmatory testing.

F. REACTOR DESIGN

1. Introduction

In this chapter the preliminary design studies which have been carried out to establish the TREX reactor design features are presented. The discussion of these activities is presented in three parts. In the first portion of the discussion a description of the TREX reactor will be presented. The discussion will include an outline of the reasons for various design concept selections in the TREX reactor. In the second section of this chapter the thermal-hydraulic design characteristics of the core are presented. Particular emphasis has been given to the thermal-hydraulic stability of the fuel element and core structure. In the final section of the chapter, nuclear design considerations for the TREX reactor are presented.

2. Reactor Description

Five of the design objectives of the TREX study have had a strong impact on the definition of the TREX reference design. These are presented in Table 3 .

Requirement for a full thermionic core requires that a prototypic TFE concept be incorporated as the TREX reference thermionic fuel element. That objective, incorporated with the specification of U-235 fuels with unmoderated cores and a 100 kWe electrical rating for the all TFE core, implies a design which is criticality limited. The achievement of reasonably high fuel volume fractions in the TFEs is mandatory.

The objective of replaceable fuel elements requires variations in the design of the top of the reactor from prototypic concepts in which the TFE leadouts are welded into the reactor vessel head. Small deviations from prototypic concepts in the nature of the core structure have also resulted from this objective.

The requirement for a 4π shieldable reference design has posed significant challenge in the definition of external reflector designs. Temperature levels, heat removal and control swing are strongly influenced by the

TABLE 3

DESIGN OBJECTIVES

FULL THERMIONIC CORE, TFE CONCEPT

U_{235} FUEL, FAST CORE

4π SHIELDABLE FLIGHT REFERENCE DESIGN

100 kWe RATING FOR 1st ALL TFE CORE

FUEL ELEMENTS REPLACEABLE

requirement for reasonable flight 4π shield weights.

In Figure 35, a schematic of a radial section of the TREX reactor core is shown. The external reflector indicates an incorporation of neutron adsorber in the space between control drum overhangs for purpose of shielding. The 4 inch external reflector using 6 inch control drums with the poison region overhung is believed to be a reasonable approach for both TREX and flight reference design systems.

The region internal to the reactor vessel consists of an outermost annulus surrounding the core region through which the coolant inlet flow passage is located. The fuel element positions are surrounded by a hexagonal type core shroud which also serves the purpose of restraining the process tube positions. The 163 fuel element positions may contain either thermionic fuel elements or nuclearly equivalent, partially enriched, stainless clad UO_2 non-thermionic fuel elements. The outermost ring of rod positions is occupied by internal reflector elements which are stainless steel clad beryllium oxide. The diameter of the internal reflector rods is nominally the same as the fuel elements.

In Figure 24 a vertical cross section of the reactor vessel and its internals are shown. Coolant is introduced above the reactor and passes down the annulus between the vessel wall and the core shroud to a lower plenum. There it turns and proceeds upward to the core lower grid plate, into the individual process tubes and then up through the reactor core. It flows to a plenum area above the axial reflector region where it is allowed to flow outward radially into an outlet plenum and then out from the reactor vessel. The active core length is 16 inches with a 5 inch lower axial reflector and a 6 inch upper axial reflector. Flow into each process tube position is controlled by an orifice at its inlet end. These orifices are sized individually for each position as required to deliver only the coolant flow necessary. The flow requirements for the non-thermionic fuel elements, the thermionic fuel elements and the beryllium oxide internal reflector rods vary markedly and efficient utilization of the heat transfer system flow capabilities does require some degree of flow control.

FIGURE 35. REACTOR ASSEMBLY SCHEMATIC RADIAL SECTION

In Table 4 selected parameters of the 1" TFE reactor concept are presented. It is interesting to note that the fuel inventory in each thermionic fuel element is approximately 1 kg while the inventory in each non-thermionic fuel element is 1.63 kg or 0.456 enriched UO_2 . The total fully enriched UO_2 inventory for a thermionic fuel element core with thirteen central driver fuel elements is 147 kg.

In Table 5 the important thermal-hydraulic design parameters of the 1" TFE core design are indicated. For a 100 kWe electric output to the user, a thermal power of 1500 kWt is required. The coolant inlet temperature of 527°C has been assumed with a design coolant ΔT of 100°C for 1500 kWe thermal operation. The coolant flow rate is 290 gallons/minute with an average core flow velocity of 4.6 ft/second. The downcomer flow velocity is a little over 7 ft/second and the overall reactor ΔP is about 3.5 psi. The maximum film drop in the core is about 10°C . Maximum clad temperature will, therefore, occur at the coolant exit end of the core.

The initial reactor design analyses were performed with the TFE 620 cell design, and the k_{eff} was found to be less than the required design value (configuration 1 and 2). This reactivity deficit tentatively was made up by the substitution of 13 fully enriched driver elements in the center of the core (configuration 4).

	<u>Configuration</u>	<u>k_{eff}</u>
	Design value	1,024
1.	151 1" TFE core	0.981
2.	163 1" TFE Core	0.992
3.	163 1" TFE Core, TREX Configuration	0.986
4.	150 1" TFE, 13 central driver	1.036
5.	187 1" TFE, 14 cells each	1.024
6.	187 1" TFE, longer cells	1.033
7.	163 1" TFE, open lattice	1.038
8.	163 1.16" TFE	1.033
9.	163 1.24" TFE, 50 mil emitter wall	1.024

TABLE 4

1" TFE REACTOR

NUMBER OF ROD POSITIONS	211
NUMBER OF TFEs	150
NUMBER OF DRIVER ELEMENTS	13
INTERNAL REFLECTORS	
RADIAL	
MATERIAL	BeO/ss
DIAMETER	1 INCH
AXIAL	
MATERIAL	BeO/ss
THICKNESS, TOP	6 INCHES
BOTTOM	5 INCHES
EXTERNAL REFLECTOR	
MATERIAL	Be—BeO/ss
THICKNESS	4 INCHES
FUEL ELEMENTS	
THERMIONIC	
FUEL/CLAD	.93UO ₂ /W
INVENTORY/ELEMENT	.95 kg
NON—THERMIONIC	
FUEL/CLAD	.456UO ₂ /ss
INVENTORY/ELEMENT	1.63 kg
DRIVER	
FUEL/CLAD	.93UO ₂ /ss
INVENTORY	1.63 kg
TOTAL INVENTORY (.93UO ₂)	164 kg

TABLE 5
THERMAL-HYDRAULIC DESIGN

THERMAL POWER, KW	1500
COOLANT INLET TEMPERATURE, °C	525
DESIGN COOLANT ΔT , °C	100
COOLANT FLOW RATE, GPM	290
CORE FLOW VELOCITY, FPS	4.6
DOWNCOMER FLOW VELOCITY, FPS	7.2
MAX. CHANNEL ΔP , PSI	2.5
REACTOR ΔP , PSI	3.5
MAXIMUM FILM DROP, °C	10

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This interim core design was utilized to determine the configuration of the reactor vessel and control cup so the conceptual design and cost estimation of these TREX components could proceed. Drawings, design calculations, and cost for these and other system components were prepared using this 163-1" TFE, 48-1" reflector rod configuration. Then additional calculational effort was devoted to the development of cores which had all TFEs. The results of these analyses are configurations 5 through 9.

All these latter configurations provide the required design k_{eff} . Configurations 5 and 6 define a longer core with 187 TFEs. Calculations performed for a 14 cell TFE for configuration 1 indicated the axial center-to-minimum fission ratio to be 1.5, a value likely to be higher than acceptable. Further, the larger number of TFEs required (187 vs 163) cases an additional derating of core performance for 100 kWe output.

Criticality calculations indicated that with the process tubes removed (configuration 7), the TFEs must be spaced about 40 mils from each other to achieve the desired k_{eff} . The coolant flow past the TFE in an open lattice design such as this is not uniform and preliminary calculations of circumferential temperature differences indicate stress levels of sufficient magnitude to induce TFE damage when using Ceramvar as a sheath material. Sheaths of greater thickness and of materials with larger thermal conductivity (such as niobium) could potentially solve this difficulty at the cost of increased TFE diameter and a tight requirement for coolant purity. An inadequate amount of analysis has been performed to assess the heat transfer, TFE stability (bowing), hydraulics and mechanical requirements of the open lattice configuration.

Increasing the TFE diameter to 1.16 inches while maintaining all thicknesses external to the fuel fixed and leaving axial lengths and dimensions unchanged meets the design value for k_{eff} . However, there is potential for TFE design change (such as thickened emitter walls) which necessitate a larger TFE diameter, i. e., a 1.24" TFE produces the desired k_{eff} with a 50 mil emitter wall. The core and TFE designs are strongly interrelated and must be progressively integrated in order to optimize both. Results to date clearly indicate that broad range analytical and experimental effort is required before a final reference configuration may be established, an effort beyond the scope of this study. However, while cost estimates are based on a core design using 1" TFEs, an increase in TFE diameter to accommodate a 1.24" TFE does not have a significant influence on cost. (2-3% of TREX total construction cost).

Seven of the design features of the TREX core design are listed in Table 6. These features have been chosen for incorporation into the design in a self-consistent fashion. Particular selections made do not necessarily represent the best overall reactor design. They do, nevertheless, represent a solution to the various requirements imposed upon the thermionic reactor. The process tube structure has been chosen from results of the known sensitivity of small fast spectrum cores to fuel element motion. The process tube structure restricts fuel element motion in a non-cumulative fashion to a relatively small displacement away from the process tube axis. The process tubes themselves may be mechanically restrained from motion by direct imposition of clamping forces on the process tubes. This avoids directly imposing clamping forces on the TFE. Furthermore, hot spots are not caused as would be the case if fuel element surfaces were in contact with one another or in contact with structures to restrain them. In addition, the process tubes permit individual orificing of the flow in each channel to serve the particular heat removal needs of that channel. As a result of the experimental nature of this reactor, widely differing types of elements may be inserted into process tubes. If this is done, flow may be conserved, if so desired, by changing the process tubes and the orifice provisions in the tube. The incorporation of a core shroud is required by the tube pass flow scheme within the reactor vessel. The core shroud also serves the purpose of mechanically restraining and clamping the process tube structure. If the reactor were designed with a one pass flow regime within the vessel external coolant return piping would be required which could create a significant shield weight penalty.

NaK 44 has been selected as the coolant for the design concept TREX. The choice could also have been NaK 78 which has a lower melting point. The use of NaK rather than sodium as the coolant for the TREX heat transfer system simplifies the design and operation of the heat transfer system by eliminating the need for trace heating. The use of NaK rather than sodium in the flight reference design is also desirable in that orbitable startup of the reactor will require preheating and melting of the coolant to the melting point or maintaining coolant temperature levels above the melting point until reactor power is achieved. This process is greatly simplified for space reactors with lower melting point coolants. The effect of the use of NaK 78 compared to NaK 44 in the TREX system is -0.5%k and a modest specification change for the pumps.

A double walled reactor vessel has been incorporated as a result of TREX safety considerations. Serious nuclear accidents cannot be postulated in TREX unless coolant is lost from the reactor core. The double wall

TABLE 6
REACTOR CHARACTERISTICS

PROCESS TUBE STRUCTURE

CORE SHROUD

NaK COOLANT

DOUBLE WALLED VESSEL

INTERNAL AXIAL REFLECTORS

INTERNAL RADIAL REFLECTOR

EXTERNAL REFLECTOR AND CONTROL ELEMENTS

vessel minimizes the possibility of a vessel rupture leading to loss of core coolant coverage.

The internal axial reflectors are incorporated into TREX and into the flight system design in order to minimize the thermal load on the external reflector, and also to minimize the fast neutron fluence on the external reflector. This design feature has had a significant cost in terms of the worth of the external reflector control system. Axial reflectors are internal to the reactor vessel both in the TREX and in the flight reference design because of the need for close proximity of these reflectors to the active core region. Cooling the axial reflectors is also greatly simplified since the reactor coolant may be used for this purpose.

The external reflector has been incorporated as a result of the desire to use reflector control for the small fast reactor concept, and the desire to eliminate the need for reactor vessel penetrations for control element actuation. Several design approaches to external reflectors and external reflector control elements are possible. For the TREX reference design, the use of control drums with poison backs have been selected as a result of the design objective requiring that reflector design be compatible with 4π shielded flight system. The 4 inch external reflector thickness is the maximum reflector thickness which is allowed when fission rate peaking at the outer edge of the core is considered. Poison back control drums incorporated with the 4 inch reflector thickness would result in significant reactivity penalties compared to a 100% unpoisoned 4 inch thick reflector and would not have the control swing required for the reactor. For this reason, a 6 inch drum design has been selected in which the poison backing overhands the 4 inch reflector thickness. This then permits 4 inches of solid reflector material all the way around when the control poison is fully out. It is possible that this thick neutron poison zone can also serve some purpose in shielding low energy neutrons for 4π shielded designs.

3. Thermal-Hydraulic Stability

The annular thickness of coolant between the fuel element surface and the process tube wall has been tentatively selected at 40 mils. On a 1 inch

diameter fuel element this represents a relatively thin coolant annulus. The flow velocities for heat removal, however, are modest, less than 5 ft/second for the average channel. Nevertheless, for such a small coolant annulus, even small distortions in the fuel element geometry, such as would be caused by cocking, bowing or ovality, could cause significant variations in the mean 40 mil annular thickness. This could cause local hot spots. For this reason, some consideration has been given to fuel elements bowing and the resulting hydraulic heat transfer interactions.

In order to demonstrate the need for concern over this problem, consider a sheath temperature distribution which superimposes a cosine component azimuthally on the overall axial sheath surface temperature distribution as shown in Equation (7). In the case of a constant amplitude of the cosine component, the deflections for a simple tube are shown by Equations (8) and (9). In Equation (8) the radius of curvature of the rod resulting from the cosine circumferential temperature variation is shown. In Equation (9), the deflection at the free end for one end rigidly fixed is shown as indicated in Figure 36. The terminology used in the analysis is defined as follows:

- R = radius of the fuel element
- e = thermal expansion coefficient
- β = amplitude of the temperature variation
- L = fuel element length
- ρ = radius of curvature of the fuel element bow
- Δ = simple free end deflection

In the case of a simple cosinusoidal, azimuthal temperature variation, the deflection in the fuel element is the arc of a circle and for a simple single layer tube the distortion is stress free. In the case of composite tube walls such as Ceramvar-alumina pressure bonded assemblies, some stresses result in the bowing process as a result of the thermal expansion mismatch of Ceramvar and alumina.

In Table 7, examples of fuel element bowing are shown. In both cases a temperature amplitude of 10°K and a tube radius of 1 inch has been selected.

Sheath temperature distribution

$$T_w(Z, \theta) = \alpha(Z) + \beta(Z) \cos \theta \quad (7)$$

Deflections for constant β

$$\rho = \frac{R}{e \beta} \quad (8)$$

$$\delta = \frac{e \beta L^2}{2 R} \quad (9)$$

FIGURE 36. FUEL ELEMENT BOWING

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TABLE 7
EXAMPLES OF BOWING

Material	Length (inches)	Radius (inches)	β $^{\circ}\text{K}$	δ Mils	Stress psi
Kovar	16	0.5	10	18	-----
Kovar/ Al_2O_3 .020/.020	16	0.5	10	21	500

In the case of a simple Kovar tube a deflection of 18 mils is calculated stress free. In the case of a 20 mil Kovar 20 mil alumina composite, a deflection of 21 mils is calculated as a consequence of the larger thermal expansion coefficient of alumina. A stress level of 500 psi is calculated as a consequence of the thermal expansion coefficient mismatches between Kovar and alumina.

Deflections of this magnitude lead to a concern over hydraulic feedbacks to the sheath temperature distributions. That is, if the tube bows and deflects and restricts flow on one side and increases flow on the other side, it is possible to envision situations in which the change in hydraulics enhances the circumferential temperature variation in the fuel element sheath. This could lead to a runaway situation in which the tube bows until it contacts the process tube wall. This phenomena has been examined in further detail and is discussed herein. The equations which describe the flow of coolant, the heat transfer and the sheath temperature distributions in the fuel element have been derived. The model used to develop the mathematical description is shown in Figure 37. The process tube has been assumed to be circular in cross section as is the TFE. The TFE is assumed to have a bowed configuration of arbitrary nature in the process tube channel with restraints at each end as indicated. The eccentricity of the fuel element axis from the process tube axis is assumed to occur within a plane. The general flow element for the coolant channel is also indicated with the major flow component being the axial u component and the circumferential flow denoted by v being small compared to u . Equation (10) represents the film drop for heat transfer from the fuel elements surface into the coolant and the Equation (11) represents the transfer coefficient correlation utilized. It is assumed in calculating heat transfer coefficient that only the local coolant hydraulic diameter was involved. In Equation (12) the heating of the coolant is represented and the effect of circumferential flow is incorporated. Equation (13) is a simple continuity of flow requirement. In Equation (14) an approximation has been made assuming that the axial flow velocity can be calculated around the circumference of the tube by assuming that the pressure does not vary around the circumference of the tube. This is considered a good approximation for the small eccentricities encountered. Equation (15) is the simple geometric expression for the local coolant annulus thickness in terms of the local eccentricity of the fuel element in the channel. Equation (16) indicates the azimuthal variation of the surface heat flux

FIGURE 37. HYDRAULIC STABILITY

TFE HEAT TRANSFER

$$q(Z, \theta) = h(Z, \theta) \left[T_w(Z, \theta) - T(Z, \theta) \right] \quad (10)$$

HEAT TRANSFER COEFFICIENT

$$h(Z, \theta) = \frac{k}{2t(Z, \theta)} \left[5.4 + .025 N_{Pe}^{.73} \right] \quad (11)$$

COOLANT HEATING

$$\begin{aligned} q(Z, \theta) = \rho C_P \frac{\partial}{\partial Z} \left[t(Z, \theta) U(Z, \theta) T(Z, \theta) \right] \\ + \frac{\rho C_P}{R} \frac{\partial}{\partial \theta} \left[t(Z, \theta) V(Z, \theta) T(Z, \theta) \right] \end{aligned} \quad (12)$$

CONTINUITY OF FLOW

$$\frac{\partial}{\partial Z} \left[t(Z, \theta) U(Z, \theta) \right] = -\frac{1}{R} \frac{\partial}{\partial \theta} \left[t(Z, \theta) V(Z, \theta) \right] \quad (13)$$

AXIAL FLOW VELOCITY

$$U(Z, \theta) = a_1 N_{Re}^{.1} t^{.5}(Z, \theta) = a t^{2/3}(Z, \theta) \quad (14)$$

$$\approx \frac{\dot{w}}{2\pi \rho R d} \left(1 - 2/3 \frac{\epsilon(Z)}{d} \cos \theta \right)$$

FIGURE 38. MATHEMATICAL MODEL

LOCAL COOLANT DIMENSION

$$t(Z, \theta) = d - \epsilon(Z) \cos \theta + R \left(1 - \sqrt{1 - \left[\frac{\epsilon(Z)}{R} \right]^2 \sin^2 \theta} \right) \quad (15)$$

SURFACE HEAT FLUX

$$q(Z, \theta) = q_Z(Z) + q_\theta(Z) \cos \theta \quad (16)$$

SHEATH TEMPERATURE DISTRIBUTION

$$T_w(Z, \theta) = \alpha(Z) + \beta(Z) \cos \theta$$

$$2\pi \alpha(Z) \equiv \int_0^{2\pi} T_w(Z, \theta) d\theta \quad (17)$$

$$\pi \beta(Z) \equiv \int_0^{2\pi} T_w(Z, \theta) \cos \theta d\theta$$

FUEL ELEMENT BOWING

$$\frac{d^2 \epsilon(Z)}{dZ^2} + \frac{\epsilon \beta(Z)}{R} + \frac{1}{R_o} = 0 \quad (18)$$

FIGURE 39. MATHEMATICAL MODEL

which is described with a simple linear cosine term.

In order to calculate the bowing of the fuel element, only the first Fourier cosine harmonic has been considered since this is the major contributor to such deflections. The manner of calculating this component is shown in Equation (17). Equation (18) represents the fuel element bowing equation. It may be seen that only circumferential sheath temperature variation amplitude must be known in order to solve for the eccentricity as a function of Z . Boundary conditions must, however, be applied to Equation (18) which are related to the nature of the mechanical support and alignment of the fuel element within the process tube.

This set of equations is highly non-linear and solution of them is a difficult numerical problem. Of primary importance in these equations, however, is the relative stability of the fuel element in the process tube channel. For example, consider initially a fuel element perfectly centered within the process tube channel. In the event of some perturbation of this situation, a local hot spot in the fuel element wall will occur. This will cause some bowing and distortion of the TFE geometry and a resulting disturbance of the coolant flow profiles. In the case of a stable system, the deflection resulting from the perturbation will be self-limiting. It is quite possible, however, to encounter unstable situations in which the perturbation will grow without limit until the fuel element interferes with the process tube wall. The stability of these equations may be studied in a relatively simple fashion by linearizing the various relationships derived. In order to do this, three assumptions must be made. First all functions are expanded in Fourier cosine series to the first term and all higher order terms are neglected. Second, it is assumed that $\frac{\epsilon}{d}$ is much less than 1 so that all second order or higher terms in ϵ are discarded. This is a good approximation for stability analysis since in the case of stable systems, such displacements will, indeed, be small. The third assumption is that $Q(z)$ and $Q(\theta)$ are independent of Z . This last assumption may be somewhat restrictive but nevertheless will give a good idea of the stability bounds upon the fuel elements and channels of this design.

The differential equation which results from these assumptions is given by Equation (19). Equation (19) has been derived simply by substituting for $\beta(Z)$ in Equation (18). $\beta(Z)$ has been determined from Equations (10) - (17) by using the linearizing assumptions indicated. In the process of

DIFFERENTIAL EQUATION

$$\begin{aligned}
 \frac{d^3 \epsilon(Z)}{dZ^3} + \frac{e q_Z}{kR} \frac{\left[2.7 - \frac{1}{370} \left(\frac{2L q_Z}{k \Delta} \right)^{.73} \right]}{\left[2.7 + .0125 \left(\frac{2L q_Z}{k \Delta} \right)^{.73} \right]^2} \frac{d \epsilon(Z)}{dZ} \\
 + \left(\frac{10}{3} \right) \left(\frac{e q_Z}{kR} \right) \left(\frac{k \Delta}{2 L q_Z} \right) \frac{\epsilon(Z)}{d} \\
 + 2 \left(\frac{e}{kR} \right) \left(\frac{k \Delta}{2 L q_Z} \right) q_\theta = 0
 \end{aligned} \tag{19}$$

CHARACTERISTIC EQUATION

$$\mu^3 + q\mu + r = 0 \tag{20}$$

SOLUTION

$$\begin{aligned}
 \epsilon(Z) = & \left\{ A \sin \left[\left(\frac{\mu_1 - \mu_2}{2i} \right) Z \right] + B \cos \left[\left(\frac{\mu_1 - \mu_2}{2i} \right) Z \right] \right\} \\
 & + \exp \left(\frac{\mu_3 Z}{2} \right) \\
 & + C \exp(-\mu_3 Z) - \frac{3}{5} d \frac{q_\theta}{q_Z}
 \end{aligned} \tag{21}$$

FIGURE 40. LINEARIZED EQUATION

substitution, however, it may be seen that a third order derivative in eccentricity has been introduced. This equation under the assumptions made is a very simple equation. It is third order and linear with constant coefficients. The coefficients have an interesting form which consists of groupings of dimensionless parameters. This equation is solved by obtaining solutions to the characteristic equation which is given by Equation (20). The general solution to the equation is given in Equation (21). μ_1 , μ_2 and μ_3 are the three roots of the characteristic equation. These roots depend only upon the coefficients of the linearized differential equation and therefore, the dimensionless parameters of the system. The constants A, B and C in the solution are constants which must be determined by application of the boundary conditions. Only two of the constants may be derived from the boundary conditions since the third must be obtained from the substitution into Equation (18) which resulted in the third derivative in Equation (19). Therefore, only two of the constants are truly available for application to boundary conditions.

Three types of boundary conditions may be considered. First, the case of a simple rigid end restraint may be applied. In this case the eccentricity and its derivative are specified at one end of the fuel element. The studies done for rigid end constraints have specified these parameters at the flow exit end. The case of a simple end restraint may also be studied in which the eccentricity is specified at the inlet and exit end of the heat transfer channel. Finally, the third type of restraint which may be considered is the case in which the eccentricity may be specified at the inlet and the derivative of the eccentricity may be specified at the outlet. This is the type of boundary condition which would apply to the TREX core in which the inlet end is restrained and the exit end is attached to a 10 foot long lead tube.

Results of calculations using these equations are shown in Figures 41-45. In Figures 41, 42 and 43 the results for Kovar sheaths have been shown. The critical value of heat flux above which the heat transfer hydraulic relationship is unstable is plotted versus the flow velocity in feet/second. The dotted lines on this plot represent lines of constant core coolant ΔT . The solid constant line drawn at 30 w/sq cm represents the mean operating point for thermionic and non-thermionic fuel elements. The three boundary

FIGURE 41. CRITICAL HEAT FLUX FOR 30 MIL COOLANT ANNULUS

FIGURE 42. CRITICAL HEAT FLUX FOR 40 MIL COOLANT ANNULUS

FIGURE 43. CRITICAL HEAT FLUX FOR 50 MIL COOLANT ANNULUS

FIGURE 44. CRITICAL HEAT FLUX FOR 40 MIL COOLANT ANNULUS AND STAINLESS STEEL SHEATH

FIGURE 45. CRITICAL HEAT FLUX FOR 50 MIL COOLANT ANNULUS AND STAINLESS STEEL SHEATH

conditions discussed have been applied to the differential equation and plotted in each of the figures. It may be seen that boundary condition #1 provides the greatest margin in the case of the simple end restraint. Since the stability line lies above the 100°C core ΔT line, we may infer that the TFE is thermohydraulically stable. The margin at the 30 w/sq cm level is fairly small for the case of the 30 mil coolant annulus and if the flow velocity were dropped from 6-1/4 to 5-1/4 feet/second, the fuel element would bow in the channel until it touched in the center of the core. A much greater margin is achieved in the case of a 40 mil coolant annulus which is the reference design. In this case, the margin is from 4.8 ft/second down to about 3 ft/second or 60% of the design rates flow. For the case of a 50 mil coolant annulus, even larger design margins are available.

The case of stainless steel sheath tubes which will be used for the non-thermionic fuel element has also been studied. (Figures 44 and 45). It may be seen that the stability margin is much less here due to the larger coefficient expansion of stainless steel. For the 40 mil coolant annulus, the core is thermohydraulically unstable for a 100° core ΔT . Stability is just barely achieved if a 50 mil coolant annulus is provided. Clearly, more attention must be given to the heat transfer of the non-thermionic and high worth driver fuel elements in order to avoid excessive bowing and formation of hot spots in the fuel element surfaces. Further attention must be given to the method of aligning and supporting the stainless clad fuel elements within the process tube channels. However, there is sufficient design flexibility in mechanical support, flow and annulus thickness to optimize the parameters to obtain a stable design.

4. Nuclear Design

A series of preliminary criticality calculations have been carried out for the TREX core. These calculations have used a one-dimensional multi-group transport program, ANISN. Uniform ground rules with regard to criticality objectives were utilized. These are shown in Table 8. An operating excess reactivity level of 2.4% was specified. In addition, a total of 4% excess reactivity was incorporated for computational bias in

TABLE 8
REACTIVITY REQUIREMENTS*

TEMPERATURE	.019
BURNUP	.002
CONTROL	.003
TOTAL	.024

CALCULATIONAL DESIGN VALUE FOR $k_{\text{eff}} = 1.024$

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*NO STUCK ROD PROVISION

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the ANISN calculations using the existing cross section data thus the design k_{eff} is 1.024 clean, cold. With regard to the computational allowances, previous studies have indicated that heterogeneity, for example, surrounding of UO_2 by tungsten and Ceramvar, is about a 1% computational error in the homogenized calculations normally performed. Similarly, streaming through low density areas in the core also represents a 1% error with respect to the homogeneous calculation. In addition to these errors, comparison of the ANISN cross section data with the thermionic reactor critical experiment indicates a potential 2% error in this computational package. These computational biases are all positive and operate simultaneously. Therefore, they have been subtracted from the calculated values for k_{eff} .

A series of preliminary parametric calculations have been carried out to determine the TREX core. For this purpose two reactor concepts were initially defined. The details of reactor configuration #1 are outlined in Table 9. This concept contains 199 process tube positions within the reactor vessel. Of these, 151 positions are allocated for fuel elements and the remaining 48 for internal reflector rods. Internal and external reflector materials were chosen to be beryllium oxide with an external reflector thickness of 5 inches, all in the interest of maximizing k_{eff} . The axial reflector material chosen for the parametric study was beryllium with thicknesses of 6 inches and 5 inches above and below the core. A coolant annulus of 40 mils and a process tube wall of 20 mils were also chosen. The TFE reference design employed in these studies incorporated current technology cell design. The calculated k_{eff} for this configuration did not achieve the desired value of 1.024 when fully loaded, and in fact, is 4.3% short. In addition, the values of radial peak to minimum fission ratios are somewhat higher than desired. The center peak to minimum radial ratio of 1.31 is acceptable. However, the edge value of 1.41 is thought to be somewhat excessive. The edge fission spike extends approximately across the outer ring of thermionic fuel elements. The maximum tolerable value for this ratio has not been established. It is suspected that values as high as 2 are unacceptable. Arbitrarily an objective value of 1.25 has been established. The 1.41 for the parameters of configuration #1 exceeds this desired value. The values of the axial peak to minimum ratio are acceptable for this design. A goal for the center peak to minimum value of 1.4 or less is desired axially. This

TABLE 9
REACTOR CONFIGURATION No. 1

NUMBER OF ROD POSITIONS	199
NUMBER OF TFEs	151
INTERNAL REFLECTOR MTL.	BeO
EXTERNAL REFLECTOR MTL.	BeO
EXTERNAL REFLECTOR THICKNESS	5 INCHES
AXIAL REFLECTOR MTL.	Be
AXIAL REFLECTOR THICKNESS	
UPPER	6 INCHES
LOWER	5 INCHES
COOLANT ANNULUS THICKNESS	0.040 INCHES
CORE TUBE WALL THICKNESS	0.020 INCHES
DESIGN FEATURES	
CIRCULAR SHROUD	
SS FILLERS IN CUSP REGIONS	
620 EMITTER DESIGN	
K_{eff}	0.981
P/M RADIAL	
CENTER	1.31
EDGE	1.41
P/M AXIAL	TOP/BOTTOM
CENTER	1.38/1.41
EDGE	1.14/1.38

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value has been chosen on the basis of previous thermionic fuel element performance studies. The maximum permissible value for the edge peak to minimum ratio has not been established at this time. This is believed to be a less serious effect, however, than a transverse gradient across the fuel element. As a result, the value of 1.38 for this design is believed acceptable. A schematic cross section of Configuration #1 is shown in Figure 46.

Parametric studies of design variations for Configuration #1 have been carried out. In Table 10 four of these calculations are shown. To these cases a non-circular shroud of a general hexagonal shape was incorporated thereby permitting more downcomer flow area. This, in turn, allowed decreasing the reactor vessel diameter slightly bringing the external reflector closer to the core. This design change causes a K effective increase of 0.09% and had minimal effects on the peak to minimum fission ratios as indicated in Table 10. It is seen from Case #2 in comparison with the reference case that incorporation of beryllium filler material in the circular shroud region increases the reactivity worth by 1.3%. In making this gain, however, the value of the radial edge to fission ratio has risen to 1.47. In the reference case the triangular cusp shaped region between process tubes was assumed to be primarily filled with stainless steel material. Removal of this stainless material causes a 1.2% loss in reactivity. Case #4 was a preliminary investigation of the potential influence of gamma shielding material outside of the external reflector. The case studied specifically was 3 inches of nickel added outside of the external reflector region. This resulted in a surprising increase in reactivity of 2.1% and had a very strong impact on the radial edge to minimum fission rate shield increasing it to 1.68. Table 11 shows further design variations in the radial reflector region which have been investigated. In Case #1 the substitution of natural UO_2 for the beryllium oxide of the internal reflector rods is investigated. This resulted in nearly 4-1/2% loss in reactivity but did decrease the value of radial edge to minimum fission ratio to 1.12. In order to regain some of the reactivity lost in Case #1, Case #2 was the same calculation but with 10% enrichment in the UO_2 . This use of enrichment still resulted

CONFIGURATION NO. 1

FIGURE 46.

TABLE 10
DESIGN VARIATIONS FOR
CONFIGURATION NO. 1

CASE	K	RC/M	RE/M	TC/M	TE/M	BC/M	BE/M
REFERENCE	0.981	1.31	1.41	1.38	1.14	1.41	1.38
1. NONCIRCULAR SHROUD	0.990	1.30	1.44	1.38	1.14	1.41	1.38
2. Be FILLER IN SHROUD	0.993	1.28	1.47	1.38	1.14	1.41	1.38
3. REMOVAL OF SS CUSP MATERIAL	0.969	1.28	1.41	1.37	1.13	1.44	1.38
4. CASE 1 WITH 3-INCHES OF Ni EXTERNAL TO REFLECTOR	1.030	1.26	1.68	1.38	1.14	1.41	1.38

TABLE 11
DESIGN VARIATIONS FOR
CONFIGURATION NO. 1

CASE	K	RC/M	RE/M	TC/M	TE/M	BC/M	BE/M
REFERENCE	0.981	1.31	1.41	1.38	1.14	1.41	1.38
1. NATURAL UO ₂ INTERNAL REFLECTOR	0.937	1.54	1.12	1.38	1.14	1.41	1.38
2. CASE 1 WITH 10% ENRICHMENT	0.955	1.53	1.03	1.38	1.14	1.41	1.38
3. CASE 2 WITH Be FILLER IN SHROUD	0.969	1.47	1.03	1.38	1.14	1.41	1.38
4. CASE 3 WITH 6-INCH BeO REFLECTORS	0.973	1.43	1.05	1.38	1.14	1.41	1.38

in a 2.6% loss in reactivity from the reference case while dropping the radial edge to minimum fission ratio value further to 1.03. From these studies it does not appear that substitution of UO_2 for beryllium oxide in the internal reflector rods would be a worthwhile tradeoff particularly in view of the power which would be generated when sufficient enrichment is incorporated into these elements to achieve acceptable reactivity levels. The third Case of Table 11 incorporated beryllium filler in the shroud of Case #2 and resulted, as before, in a significant increase in reactivity. Case #4 involved increasing the external reflectors thickness of Case #3 to 6 inches. This resulted only in a 0.4% increase in reactivity.

In Table 12 further reflector parametric studies are presented. A 4 inch beryllium oxide reflector was calculated to investigate the effect of thinner reflectors on the radial edge to minimum fission ratio. The 4 inch reflector very nearly achieved the desired goal of 1.25 with the loss of only 0.8% in reactivity. Further studies were then done to determine the influence of gamma shielding materials behind the 4 inch beryllium reflector. Cases 2, 3 and 4 investigated using 1, 3 and 5 inches of nickel. The reactivity gains in these cases were quite significant increasing the k_{eff} from 0.973 to 0.992 for 5 inches of nickel. The radial edge to minimum fission ratio value suffered, increasing to 1.54 for 5 inches of nickel.

In Table 13, parametric variations of the axial reflectors are presented. The change from the reference design of beryllium axial reflectors to beryllium oxide reflectors is presented. An increase in k_{eff} of 0.8% results from the use of beryllium oxide. Cases 2-5 investigate thicker axial reflector segments. It is seen that increases in top reflector thickness do not offer significant increases in reactivity and the improvements resulting from increasing the bottom reflectors thickness are also minimal.

In Table 14 variations in the core design are investigated. The first case represents the calculation for one central fully enriched driver fuel element. It is seen that the increase in reactivity is 0.3%. Case 2 is for 7 central driver elements representing a total gain of about 2.7%. Incorporation of central driver fuel elements, therefore, can offer significant increases in reactivity but at the cost of lower overall efficiency. In Case 3 and 4

TABLE 12
DESIGN VARIATIONS FOR
CONFIGURATION NO. 1

CASE		K	RC/M	RE/M	TC/M	TE/M	BC/M	BE/M	CESR-2123
	REFERENCE	0.981	1.31	1.41	1.38	1.14	1.41	1.38	
1.	FOUR INCH BeO REFLECTOR	0.973	1.34	1.28	1.38	1.14	1.41	1.38	
2.	CASE 1 PLUS 1-INCH Ni	0.984	1.31	1.44	1.38	1.14	1.41	1.38	
3.	CASE 1 PLUS 3-INCHES Ni	0.990	1.28	1.53	1.38	1.14	1.41	1.38	
4.	CASE 1 PLUS 5-INCHES Ni	0.992	1.28	1.54	1.38	1.14	1.41	1.38	

TABLE 13
DESIGN VARIATIONS FOR
CONFIGURATION NO. 1

CASE	K	RC/M	RE/M	TC/M	TE/M	BC/M	BE/M
REFERENCE	0.981	1.31	1.41	1.38	1.14	1.41	1.38
1. BeO AXIAL REFLECTORS	0.989	1.31	1.41	1.37	1.07	1.44	1.17
2. CASE 1 WITH 7-INCH TOP REFLECTOR	0.990	1.31	1.41	1.37	1.07	1.44	1.17
3. CASE 1 WITH 8-INCH TOP REFLECTOR	0.990	1.31	1.41	1.37	1.08	1.44	1.17
4. CASE 1 WITH 6-INCH BOTTOM REFLECTOR	0.991	1.31	1.41	1.38	1.07	1.40	1.25
5. CASE 1 WITH 7-INCH BOTTOM REFLECTOR	0.992	1.31	1.41	1.37	1.07	1.36	1.32

TABLE 14
DESIGN VARIATIONS FOR
CONFIGURATION NO. 1

CASE	K	RC/M	RE/M	TC/M	TE/M	BC/M	BE/M
REFERENCE	0.981	1.31	1.41	1.38	1.14	1.41	1.38
1. ONE CENTRAL DRIVER FUEL ELEMENT	0.984	1.33	1.40	1.38	1.14	1.42	1.37
2. SEVEN CENTRAL DRIVER FUEL ELEMENT	1.008	1.36	1.41	1.38	1.14	1.42	1.37
3. THIRTEEN CELL TFE	1.001	1.31	1.43	1.45	1.13	1.49	1.37
4. FOURTEEN CELL TFE	1.016	1.29	1.45	1.49	1.13	1.53	1.38
5. TWELVE CELL TFE 1.5 INCH LONG CELL	1.022	1.30	1.43	1.48	1.13	1.53	1.37
6. 199 TFEs	1.016	1.52	1.24	1.37	1.15	1.40	1.41

increases in the length of the reference thermionic fuel element were investigated by means of adding additional cells. 13 cells per TFE, as opposed to 12 cells per TFE, results in a gain in reactivity of 2%. 14 cells represents a gain of 3.5%. It should be noted, however, that in the case of the 14 cell TFE, the center-to-minimum axial fission ratios have increased to the level of 1.5 which represents a significant increase above the desired 1.4. The value of 1.5 may be acceptable, however, and may be considered. In Case 5 a 12 cell TFE with longer cells was investigated, 1.5 inches instead of 1-1/3 inches. This resulted in a reactivity gain of about 4.1%. Once again, the center-to-minimum axial fission ratios were increased to a level of about 1.5.

In the 6th case of Table 14, the replacement of the internal reflector rods by TFEs was investigated. This permitted the loading of 199 TFEs. The gain in k_{eff} here was 3.5% but the value of center-to-minimum radial fission ratios increased to 1.5. This may be an acceptable value, since current levels may be adjusted radially, however, it probably represents the upper limit of permissible variation. The radial edge to minimum is greatly improved down to the desired level of 1.25.

A number of reactivity coefficient calculations were carried out for Configuration #1. Results of these calculations are shown in Table 15. It may be seen that the Doppler reactivities are quite small in comparison with the various expansion coefficients. It may also be seen from Table 15 that the total worth of the external reflector is 12.7%. This value is adequate for providing reactor control by means of elements in the external reflector. It is necessary, however, to achieve about one-half of the total external reflector worth in the control system for a comfortable margin in control system worth.

In Table 16, information is presented relative to the second configuration defined for the preliminary TREX physics studies. This configuration incorporates 211 process tube positions which are filled with 163 fuel element positions surrounded by 48 reflector rod positions. Alternatively, the 211 process tubes may be filled with 211 fuel elements if desired.

TABLE 15
REACTIVITY STUDIES
CONFIGURATION NO. 1

	Δk
DOPPLER REACTIVITY	
U ₂₃₅ (300°K to 2100°K)	+.0016
U ₂₃₈ (300°K to 2100°K)	-.0002
W/Re/Me (300°K to 2100°K)	-.0024
Nb (300°K to 900°K)	-.0002
EXPANSION AND DENSITY	
REACTOR COLD TO OPERATING	-.0179
CORE + DOWNCOMER COLD TO OPERATING	-.0133
CORE COLD TO OPERATING	-.0059
TOTAL LOSS OF NaK FROM CORE	-.0023
TOTAL EXTERNAL REFLECTOR REMOVAL	-.1274

TABLE 16
CONFIGURATION NO. 2

NUMBER OF ROD POSITIONS	211
NUMBER OF TFEs	163
INTERNAL REFLECTOR MTL.	BeO
EXTERNAL REFLECTOR MTL.	BeO
EXTERNAL REFLECTOR THICKNESS	5 INCH
AXIAL REFLECTOR MTL.	Be
AXIAL REFLECTOR THICKNESS	
UPPER	6 INCH
LOWER	5 INCH
COOLANT ANNULUS THICKNESS	.040 INCH
CORE TUBE WALL THICKNESS	.020 INCH
DESIGN FEATURES	
CIRCULAR SHROUD	
SS FILLERS IN CUSP REGIONS	
620 EMITTER DESIGN	
K _{eff}	0.992
P/M RADIAL	
CENTER	1.36
EDGE	1.42
P/M AXIAL	TOP/BOTTOM
CENTER	1.38/1.41
EDGE	1.14/1.38

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The calculations for an all TFE loading of this concept indicate a k effective of 0.992 which is 1.1% higher than that for configuration #1. The radial center-to-minimum fission profile ratio is seen to be slightly greater than for Configuration #1 with other fission ratio values being essentially the same. A schematic cross section for Configuration #2 is shown in Figure 47. The parametric studies carried out for Configuration #2 were much more limited than for Configuration #1 with only two cases calculated. These are shown in Table 17. It may be seen from Table 17 that incorporating beryllium filler in the shroud zone is worth somewhat more for Configuration #2 than for Configuration #1. Similarly, filling all process tube positions with TFEs gains slightly greater reactivity than was the case for Configuration #1.

The results of these calculations for the two configurations defined were utilized in coming to the definition of the reactor configuration utilized for the TREX cost study. Additional calculational results for this design have been carried out and are presented in Table 18. This configuration is seen to differ slightly from Configuration #2 in that the calculated k_{eff} is 0.986 which is 0.6% less than Configuration #2. It may also be noted that a single wall vessel increases the k_{eff} by less than half a percent.

A calculation was performed to determine the enrichment of a stainless clad UO_2 non-thermionic fuel element. Case #2 indicates that the required enrichment is 45.6%. The k gained by incorporation of 13 UO_2 high worth drivers is shown in Case 3 to be 5.1%; if UO_2PUO_2 drivers are included the gain is 7.1%. Similarly, the gain for the non-thermionic fuel element core with 13 central high worth drivers is 5%, indicating that the nuclear characteristics of the non-thermionic fuel element core are overall very similar to the thermionic fuel element core. In the fifth and final case of Table 18, the case of filling all 211 process tubes with non-thermionic fuel elements, the central 13 elements being high worth fully enriched UO_2 drivers, is shown and has a gain of 7% over the all-TFE core in this case.

CONFIGURATION No. 2

FIGURE 47.

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TABLE 17
DESIGN VARIATIONS FOR
CONFIGURATION NO. 2

CASE		K	RC/M	RE/M	TC/M	TE/M	BC/M	BE/M
	REFERENCE	0.992	1.36	1.42	1.38	1.14	1.41	1.38
1.	Be FILLER IN SHROUD ZONE	1.013	1.30	1.54	1.37	1.15	1.40	1.41
2.	211 TFE's	1.021	1.57	1.25	1.37	1.15	1.40	1.41

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TABLE 18
DESIGN CALCULATIONS

	K	RC/M	RE/M	TC/M	TE/M	BC/M	BE/M
1. All TFE Core (for single wall vessel K = 1.015)	.986	1.38	1.29	1.38	1.14	1.41	1.38
2. All NonThermionic Core Enrichment = 45.6%	.986	1.37	1.24	1.40	1.06	1.34	1.28
3. TFE with $^{13}\text{UO}_2$ Drivers with $^{13}\text{UO}_2 - \text{PuO}_2$ Drivers	1.037 1.057	1.36	1.31	1.36	1.12	1.39	1.34
4. Non-TFE Core with Thirteen UO_2 Drivers	1.036	1.42	1.26	1.37	1.06	1.31	1.29
5. Non-TFE Core, no BeO Internal Reflectors, with Thirteen UO_2 Drivers	1.056	1.69	1.26	1.37	1.06	1.30	1.29

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Other ways for increasing the reactivity have been examined. In these, influences on fuel volume fraction has been employed for comparison purposes. The reference design thermionic fuel element which is identical to TFE-620 has a permissible UO_2 volume fraction of .321 for the core volume. The required fuel volume fraction to achieve the design k effective is .349. Several ways have been identified for resolving this discrepancy between the required fuel volume fraction and the maximum permissible fuel volume fraction. The most straightforward way of accomplishing an increase in reactivity is to increase the number of cells in the reactor by increasing the total number of fuel elements from 163 to 187 and by increasing the number of cells/fuel element from 12 to 14. A summary of the decrease in required volume fraction to meet design k effective is shown in the top portion of Table 19. It should be noted in these reactivity allowances that some provision has been made for fuel pellet coring. This allowance is 1% which is approximately .003 in volume fraction. This 1% coring allowance should be considered the overall average value. In the actual detail core design the coring would be restricted to the central zone of the core and would be in amounts considerably in excess of 1 volume percent of the fuel in that region. The effect of coring the fuel pellets on central fuel temperatures is quite pronounced. In this case, it is necessary as a consequence of the significantly larger peak-to-minimum ratios both axially and radially which result from the increase in core size. It may be seen from a summary of Table 19 that these design changes achieve the required criticality goal.

A second way of achieving the required design k_{eff} is outlined in Table 20. This involves increasing the number of TFEs from 163 to 187 while increasing their length from 16 inches to 18 inches. The number of cells/thermionic fuel element is fixed at 12 cells. The increased fuel element length is taken up entirely in increasing the length of the fuel portion of the cells. In the top portion of Table 20 the reductions in required fuel volume fraction due to increases in core size are noted. It should be observed here that a 1% allowance for coring has been made. In the lower part of Table 20, the increase in the available fuel volume fraction due to the change in cell lengths has been noted. In comparing

TABLE 19
INCREASE NUMBER OF CELLS
 CONFIGURATION 5

REFERENCE REQUIRED VF		.349	
NUMBER OF TFEs	187		
ΔVF		-.14	
NUMBER OF CELLS/TFE	14		
ΔVF		-.017	
CORING ALLOWANCE FOR FUEL CENTRAL TEMPERATURE, RE- DUCTION, % VOLUME	1		EFFECT OF CHANGES ON REQUIRED VF
ΔVF		+.003	
RESULTANT REQUIRED VF		.321	
REFERENCE AVAILABLE VF		.321	
EXCESS VF		.000	EFFECT OF CHANGES ON AVAILABLE VF
k_{eff}		1.024	

TABLE 20
INCREASED NUMBER OF TFEs WITH LONGER CELLS
 CONFIGURATION 6

REFERENCE REQUIRED VF		.349	
NUMBER OF TFEs	187		
Δ VF		-.014	
LENGTH OF CORE, INCHES	18		
Δ VF		-.017	EFFECT OF CHANGES ON REQUIRED VF
CORING ALLOWANCE FOR FUEL CENTRAL TEMPERATURE RE- DUCTION, % VOLUME	1		
Δ VF		+.003	
RESULTANT REQUIRED VF		.321	
REFERENCE AVAILABLE VF		.321	
LENGTH OF CORE	18		
Δ VF		+.006	EFFECT OF CHANGE AVAILABLE VF
NEW AVAILABLE VF		.327	
EXCESS VF		+.006	
k_{eff}		1.033	

TABLE 21
INCREASED TFE DIAMETER
 CONFIGURATION 8

REQUIRED VF		.349	
NEW TFE DIAMETER	1.16		
Δ VF		-.025	EFFECT OF CHANGES ON REQUIRED VF
CORING ALLOWANCE FOR FUEL CENTRAL TEMPERATURE RE- DUCTION, % VOLUME	6		
Δ VF		+.021	
RESULTANT REQUIRED VF		.345	
AVAILABLE VF		.321	
NEW TFE DIAMETER			
Δ VF		+.030	EFFECT OF CHANGES ON AVAILABLE VF
NEW AVAILABLE VF		.351	
EXCESS VF		+.006	
k_{eff}		1.033	

the available maximum fuel volume to the required fuel volume fraction, it may be noted that an excess volume fraction of .006 exists. This excess is worth about 1% in reactivity.

A third way to achieve the required reactivity goals is summarized in Table 21. This method involves increasing the available fuel volume fraction by increasing the TFE diameter from 1.01 inches to 1.16 inches while maintaining all thicknesses external to the fuel fixed and leaving axial lengths and dimensions unchanged. In the top portion of Table 21 the reductions in fuel volume fraction corresponding to the increased core diameter are given. It should be noted here that a 6 volume percent allowance has been made for coring. This greater allowance is made necessary by the increase in fuel diameter and by the somewhat larger radial peak-to-minimum fission distribution for the larger diameter core. Once again, the 6% is a core average number. In actuality this coring process would be confined to the central regions of the core with peripheral cells being uncoring.

The 6 volume percent coring will, in this case, have a modest effect on the power generated in the central cells. Once again, however, major effect of coring in reduction of central fuel volume fraction is only from .349 to .345 as a result of the large penalty associated with fuel pellet coring. In the lower half of Table 21, however, the increased volume fraction available from the larger diameter TFE is given. This improvement is more than adequate to reach the desired reactivity goals. The excess volume fraction of fuel available is, once again, .006 which corresponds to approximately 1% in reactivity.

It should also be noted that for this case, UC-ZrC fuel could be substituted for the UO_2 fuel and fuel pellet coring for the purpose of reducing central fuel temperature or flattening the fission profile in the core center would not be required. This advantage could be used to reduce the amount by which the TFE diameter would have to be increased.

A fourth method for enhancing the criticality of the reference thermionic core would be the removal of the process tube core structure. In Table

22 the reactivity effects of this approach is shown. The calculations shown here are simplified in that only the effect of increasing the fuel volume fraction on reactivity is shown. Removal of the stainless steel of the process tubes also has a reactivity effect which has not been considered. The values shown represent the reactivity effect of core compaction possible by removal of the process tubes and a reduction of the TFE spacing from 133 to 40 mils. The single effect of this reduction in pitch corresponding process tube removal was inadequate to achieve criticality in the system. The additional compaction and reactivity increase by reducing TFE pitch by an additional 40 mils was sufficient to produce a reactivity margin.

It is believed that the core volume fraction associated with the process tubes represents a near minimum amount for any internal structure which is mechanically independent of the fuel elements themselves. A significant reduction in this structure, therefore, is unlikely. It is further believed that the coolant volume fraction which have been allowed in the present design are also nearly at minimum values consistent with a conservative design.

The structure has been incorporated in order to utilize a conservative design approach and is prototypic of recent space powerplant reference designs. Benefits which are provided by the incorporation of internal core structure include improved heat transfer and hydraulic characteristics and elimination of thermal/mechanical feedback mechanisms which may cause unstable behavior in the reactor. It is possible that the structural requirements of the reactor core could be met without the incorporation of independent structural members. In this case, an open lattice core could be considered in which fuel elements are held together by an external structural member and are clamped together. The fuel elements could be designed to bear upon one another at special points along the length of the coolant channels. The reactivity benefits which may be realized from elimination of internal core structure are quite significant, and could be extremely valuable in resolving criticality difficulties brought about by design requirements in the thermionic fuel element. It would, for example, make possible the design of a TFE core using the present design at the 100 kWe level without requiring

TABLE 22
REMOVE PROCESS TUBES
 CONFIGURATION 7

REQUIRED VF		.349	
REMOVE PROCESS TUBES AND REDUCE TFE PITCH, INCHES	1.050		EFFECT OF CHANGES ON REQUIRED VF
Δ VF*		+.014	
RESULTANT REQUIRED VF		.363	
AVAILABLE VF		.321	
REMOVE PROCESS TUBES AND REDUCE TFE PITCH, INCHES	1.050		EFFECT OF CHANGES ON AVAILABLE VF
Δ VF		+.052	
RESULTANT AVAILABLE VF		.373	
EXCESS VF		.010	
k_{eff}		1.038	

*This estimate represents a large extrapolation of currently available calculations.

the utilization of driver fuel elements or derating thermionic performance levels to low values.

Solution of the criticality problem by elimination of internal core structure and minimization of coolant volume fraction in an open lattice core introduces new and potentially severe design and developmental problems. The transient stability of the open lattice core would require extensive investigation to establish, with a high level of confidence, that the core could not expand or contract in response to temperature or power in a manner which could cause damaging transients. The influence of non-uniform flow around the periphery of the fuel element and the potential existence of hot spots would require careful investigation to determine with confidence that this would not be detrimental to long fuel element life. A further disadvantage that has direct bearing upon the objectives of the TREX program is that in such a core, the removal and insertion of fuel elements would be more difficult.

The approaches discussed for the preceding five tables have involved design changes in the TFE or core intended to respond only to criticality requirements. It has been assumed that no need existed to respond to design problems in the thermionic fuel element or other aspects of core design. Changes made in Tables 19 to 22, therefore, did not involve increasing material thicknesses or incorporating fins or other devices into the fuel to improve heat transfer or mechanical stability. In the event of a requirement for such changes in the thermionic fuel element, the criticality problem is almost certain to be made more severe by the nature of such changes.

As a typical example of the type of change one might anticipate, consider an increase in emitter thickness to 50 mils to respond to the emitter distortion. Let us also assume, however, that all other material thicknesses and dimensions remain unchanged except for diameters. That is, the length of the converter cells, the thickness of emitter end caps or other material thicknesses in the cell will not be increased.

The increase in emitter thickness to 50 mils represents a significant criticality penalty. Once again, a variety of means for compensating this penalty may be determined. In Table 23 one method for compensation has been presented which involves only an increase in the fuel element diameter. The core configuration is left unchanged with 163 fuel elements surrounded by a single row of internal reflector rods. The fuel elements consist of 12 cells each having an overall length of 16 inches. In the top portion of Table 23 the reduction in required fuel volume fraction for criticality is indicated. It is seen that the increase in TFE diameter results in an increase in core diameter which reduces the required volume fraction by over 10%. This gain, however, is almost entirely lost by the need for a fairly large coring allowance associated with fuel central temperature reduction for the larger diameter fuel elements. In addition, the thickening of the emitter wall results in a slight loss of reactivity due to increased tungsten fraction. The sum of these two losses reduces the gain produced by increased core diameter in required fuel volume fraction from over 10% to less than 2%.

In the lower half of Table 23 the increased availability of fuel volume fraction due to the increased TFE diameter is shown. This is shown in two components. The increased volume fraction resulting from an increase in TFE diameter at fixed emitter wall thickness and a loss due to an increase in wall thickness from 30 to 50 mils. The sum of these two effects permits an increase in available fuel volume fraction of approximately 7%. The resultant new available volume fraction is exactly equal to the resultant required volume fraction which means that criticality can be achieved in the system with all appropriate design provisions being made. There is, in this case, no excess available volume fraction and if greater design margin was desired, some further design modification would be required which would be permissive of greater reactivity. This could be accomplished for example, by the use of UC-ZrC fuel which would reduce or eliminate the requirement for coring the fuel pellets as a result of the higher thermal conductivity of the carbide fuel material.

TABLE 23

INCREASED EMITTER THICKNESS (50 MIL LATERAL WALL)
CONFIGURATION 9

REFERENCE REQUIRED VF		.349	
INCREASE TFE DIAMETER, INCHES	1.240		
Δ VF		-.038	
CORING ALLOWANCE FOR CENTRAL FUEL TEMPERATURE REDUCTION, % VOLUME	7		EFFECT OF CHANGES ON REQUIRED VF
Δ VF		+.024	
THICKEN CLAD, INCHES	.050		
Δ VF		+.008	
RESULTANT REQUIRED VF		.343	
REFERENCE AVAILABLE		.321	
INCREASE TFE DIAMETER, INCHES	1.240		
Δ VF		+.049	
THICKEN CLAD, INCHES	.050		EFFECT OF CHANGES ON REQUIRED VF
Δ VF		-.027	
NEW AVAILABLE VF		.343	
EXCESS VF		.000	
k_{eff}		1.024	

A comprehensive evaluation of all of the different design techniques which may be used to compensate in the criticality and thermal design sense for TFE or other core design modifications are too extensive for discussion here. A conclusion which may be made based upon the preliminary investigations which have been carried out to date are that margin for design modifications for solution of developmental problems does exist. This margin, however, is limited and such modifications must be carefully limited. It is essential in order to achieve good reactor designs that diluents in the fuel and thicknesses of cladding regions must be maintained at absolute minimum values.

In Table 24 reactivity study results for the non-thermionic core design are shown. The Doppler coefficients are quite small as noted before in Table 15. They are somewhat larger, however, than the results obtained for the thermionic fuel element core of Configuration #1. This is a consequence of the incorporation of large amounts of U-238 in the fuel material. The expansion and density coefficients are larger than was the case for Configuration #1 with the cold to operating reactivity swing of 2.8%. This is primarily a consequence of the larger expansion coefficient of the stainless steel clad non-thermionic fuel elements.

In Table 25 the results of reactivity studies for the control system are shown. In this table upper and lower bound calculations of the reactivity lost when the control poison is turned fully in from fully out are shown. These calculations have been done for two ways of representing the two-dimensional geometry of the control elements in a one-dimensional calculation. It may be seen that the range of calculated worth is from 5.5% to 6.9%. This result is about what was anticipated for the control system worth, and is probably an adequate amount.

Control worth studies for the control system worth with high worth UO_2 central drivers have also been carried out. These are shown in Table 26. It may be seen for the TFE and non-TFE core with UO_2 central drivers that the control system worth is between 5% and 6%. In the case of a non-thermionic fuel element core and 13 central drivers with no

TABLE 24
REACTIVITY STUDIES
 (NONTHERMIONIC CORE)

	Δk
DOPPLER REACTIVITY	
U ₂₃₅ (300°K to 2100°K)	+ .0013
U ₂₃₈ (300°K to 2100°K)	- .0031
EXPANSION AND DENSITY	
REACTOR COLD TO OPERATING	- .0289
CORE & DOWNCOMER COLD TO OPERATING	- .0227
CORE COLD TO OPERATING	- .0141
TOTAL LOSS OF NaK FROM CORE	- .002.0

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TABLE 25
CONTROL WORTH STUDIES

ALL TFE CORE

ZrB₂ IN DRUMS TURNED FULLY IN

	Δk
1. Homogenized ZrB ₂ - Be Region Inside	-.069
Homogenized ZrB ₂ - BeO Region Outside	
2. Separate ZrB ₂ , Be Regions Inside	-.055
Separate ZrB ₂ , BeO Regions Outside	

TABLE 26
CONTROL WORTH STUDIES

	Δk
TFE CORE WITH THIRTEEN UO_2 DRIVERS	-.061
NON-TFE CORE WITH THIRTEEN UO_2 DRIVERS	-.062
NON-TFE CORE WITH THIRTEEN UO_2 DRIVERS, NO BeO INTERNAL REFLECTORS	-.077

FOR ALL CASES:

1. ZrB_2 in drums turned fully in.
2. Homogenized ZrB_2 -- Be region inside.
3. Homogenized ZrB_2 -- BeO region outside.

beryllium oxide reflector rods, that is a 211 fuel element core, the control system worth increases to a maximum value of 7.7%.

The results of these studies are quite encouraging with regard to control system worth. The indications are that some design margin in the worth of control elements does exist. It should be noted, however, that these calculations have been done with one-dimensional calculations and the calculated margins are not extremely large. Some caution will be required with regard to more accurate estimates of the control system worth and establishing the required value of the control system worth in future studies, before the adequacy of the conceptual design presented here is fully established.

The calculated neutron spectrum for Configuration #2 is shown in Figure 48. The core absorption and leakage are approximately 70% and 30% respectively.

FIGURE 48. TREX CONFIGURATION 2 SPECTRUM

G. TREX CONTROL

1. Introduction

The purpose of this chapter is to discuss an overall control philosophy for the thermionic reactor experiment. The topics to be discussed include nuclear, hydraulic and electrical control, startup-shutdown considerations and the plant safety system. The present discussion will draw heavily upon descriptions of the several plant components presented in other chapters of this report. In addition, the present discussion is concerned primarily with control philosophy. The methods for implementing this philosophy will be discussed in subsequent sections.

2. Overall Plant

A schematic representation of the overall TREX power plant is shown in Figure 49. It consists of the reactor and the associated nuclear control instrumentation; the heat removal system consisting of a primary loop, intermediate heat exchanger, secondary loop and heat dump; and the electrical system consisting of the TFE current leads, cesium reservoirs, power conditioning modules and the dc bus containing both the customer's load and a ballast load. With regard to the nuclear portion of the plant, the primary variable to be controlled is reactivity, as determined by the positions of the moveable reflector control drums in the external reflector. In the heat removal system the primary variables to be controlled are the power supplies for the primary and secondary pumps and electrical heaters, the power supply for the heat dump blower and the settings on the moveable heat dump louvers. In the electrical area, the primary variables to be controlled are the electrical currents extracted from each thermionic fuel element, the cesium reservoir temperature for each thermionic fuel element and the dc bus voltage. Subsequent sections of this chapter will deal with the methods which have been selected for controlling each of these major power plant variables.

3. Nuclear Control

The selected nuclear control modes are presented in block diagram form in the schematic representation of the plant in Figure 49. The basic

Figure 49. TREX Control System

nuclear control mode is manual, meaning that the reactor operator manually operates the moveable external reflector control segments to control reactivity and, in turn, the reactor power level. A second nuclear control mode, however, will also be employed for the TREX reactor. Due to the weak feedback mechanisms in this reactor, concern exists that the power level may tend to drift, once a steady-state full power level has been reached. To overcome such drift, an automatic level control mode has been incorporated into the nuclear control system. This mode is also represented in the block diagram of Figure 49. In this automatic level control mode a shim control drum having only a small remaining reactivity worth will be automatically controlled to hold the reactor power level to a set point selected by the reactor operator. This control mode will be discussed further later in this section.

A third nuclear control mode is envisioned for an eventual space power plant and is represented by the dashed blocks in Figure 49. In this control mode the plant is put on fully automatic control and an error signal, sensing the dc bus voltage, is fed back and employed to control the reflector control drums and thereby, the reactor power level. This control mode will not be employed during the early stages of TREX but may be subsequently incorporated.

That completes the block diagram description of the reactivity control system. Let us now turn to the details of how these various control modes are implemented in practice.

Figure 50 shows a representation of the reactor core, vessel and external reflector showing the twelve control drums which are employed for reactivity control. This figure has been discussed previously and will not be discussed further here.

To gain an overall understanding of the nuclear control issues, it is worthwhile to tabulate some of the characteristic reactivity parameters for the TREX reactor. This has been done in Table 27. These

FIGURE 50. REACTOR ASSEMBLY SCHEMATIC RADIAL SECTION

TABLE 27
REACTIVITY PARAMETERS

	<u>NTFE</u>	<u>TFE</u>
External Reflector Worth (\$)	19.6	
Control Drum Worth (\$)	8.0	
Total NaK Loss (\$)	-0.35	
Burnup (\$)	-0.30	
Control Margin (\$)	0.20	
Cold to Hot (\$)		
a. NaK Preheat (780°K)	-3.50	-2.38
b. Startup	-0.63	-0.41
Power Coefficient (β /% Δ P)	-0.50	-0.22

parameters are shown both for a core consisting of all non thermionic fuel elements and for a core consisting of all thermionic fuel elements. The first five reactivity parameters are the same for both cores. Note that the total worth of the external reflector is \$19.6. Hence, removal of this reflector put the plant in a very safe shutdown condition for refueling and other maintenance operations. The worth of the moveable control drums is \$8.0. (More recent calculations have indicated a 20% higher control drum worth which has not been incorporated into this report. This is the reactivity that is available for control of the plant during the startup, normal operation, shutdown and scram. The loss of NaK in the plant is worth -35¢. Hence, dry cores are safer than wet cores and the discussions to follow will therefore be restricted to wet cores. Burnup during a 10,000 hour run only amounts to 30¢, so there is no need for large amounts of reactivity in the control system to be allocated to burnup. Approximately 20¢ remaining in the moveable control system at full power at the end of life is adequate control margin to permit startup in a reasonable period and effective control during operation. The cold to hot reactivity swings in Table 27 have been broken into two parts. The first is the loss in reactivity in heating the reactor from room temperature to an isothermal 780°K. The second is the loss of reactivity in heating the reactor from an isothermal 780°K to the full power temperature distribution. For reasons which will be discussed subsequently, it is planned to electrically preheat both the primary and secondary coolant systems to approximately 780°K prior to reactor startup. The 780°K represents the coldest temperature in the secondary loop during normal full power operation and occurs at the heat dump exit. Note (Table 27) that the NTFE and TFE cores lose \$3.50 and \$2.38 respectively during electrical preheat. Once preheated, it takes only an additional 63¢ to start up the NTFE core and 41¢ to startup the TFE core.

The last entry in Table 27 is the power coefficient at full power for the driver and TFE cores. The power coefficients are expressed in cents of reactivity feedback per percent increase in reactor thermal power. For the NTFE core the power coefficient is - 50¢/% ΔP and for the TFE core the corresponding number is -22¢/% ΔP . Note that both of these numbers are small. In the TFE core, for example, the addition of a 1¢ spurious reactivity signal would lead to approximately a 5% increase in reactor thermal power level. One of the major issues in coming to terms with an overall control philosophy for the thermionic reactor experiment surrounds the method of dealing with the weak reactivity feedback mechanisms and correspondingly low power coefficient.

Based on the reactivity parameters in Table 27, control system reactivities for the NTFE and TFE cores are compared in Table 28 for different settings of external reflector control drums. For meaningful comparison, it is assumed that both cores are loaded to the point where the excess reactivity, with the drum reactivity full in a full power at the beginning of life, is approximately 50¢. This 50¢ includes the 30¢ required for burnup and the 20¢ control margin of Table 27. With the cores so loaded, reactivity immediately following a scram in both the NTFE and TFE cores is -\$7.50. This initial scram reactivity is adequate to bring the thermal power level down to approximately 15% on the prompt jump immediately following the scram. The power level, of course, would decay very rapidly thereafter.

The shutdown reactivities with the drum reactivity full out at room temperature are -\$3.37 and -\$4.71 for the NTFE and TFE cores respectively. These are the reactivities that would prevail immediately after raising the reflector cup following a refueling operation. Somewhat greater shutdown margins would be desirable since the values shown in Table 28 are only several times the reactivity worth of the most reactive elements which are handled during a refueling operation. On the other hand, the ability to remove the reflector cup during refueling operations provides inherent safety and the nuclear instrumentation would be operable during the time when the reflector cup was being raised. Excess reactivity due to the overloading of the core will be detected and further raising of the cup will be prevented. Note that the thermionic reactor experiment like most reactors is prompt critical if the drum reactivity is fully inserted at room temperature. Precautions to prevent this event from taking place will, of course, be incorporated into the plant safety system. The shutdown reactivities with the reflector cup removed for the NTFE and TFE cores are -\$15.00 and -\$16.3, respectively.

The preceding discussion has dealt with the several reactivity worths in the thermionic reactor experiment but has not addressed the incremental scale upon which this reactivity must be handled by the control system or the question of how fast the control system must be able to insert and withdraw reactivity. This topic will be taken up next. An important consideration in the selection of incremental reactivity step

TABLE 28
CONTROL SYSTEM REACTIVITIES

DRUM REACTIVITY FULL IN

REACTOR TEMPERATURE CONDITIONS	REACTIVITY (\$)	
	NTFE Core	TFE Core
Room Temperature	4.63	3.29
Coolant Preheat Temperature	1.13	0.91
Full Power Temperature	0.50	0.50

DRUM REACTIVITY FULL OUT

REACTOR TEMPERATURE CONDITIONS	REACTIVITY (\$)	
	NTFE Core	TFE Core
Room Temperature	-3.37	-4.71
Coolant Preheat Temperature	-6.87	-7.09
Full Power Temperature (SCRAM)	-7.50	-7.50

SHUTDOWN REACTIVITY WITH REFLECTOR CUP REMOVED (\$)

NTFE Core	-15.0
TFE Core	-16.3

sizes and reactivity and insertion and withdrawal rates is that of reactivity noise.

The main source of thermal reactivity noise in TREX is associated with fluctuations in the NaK return temperature from the heat dump. These could be caused, for example, by changes in the ambient temperature in the vicinity of the heat dump. It is readily shown that the fractional changes in reactor thermal power level ($\Delta P_T/P_T$) due to a change ΔT_S in the NaK temperature T_S at the heat dump exit is given by the relations:

$$\frac{\Delta P_T}{P_T} = - \frac{2 \Delta T_S}{T_O - T_S} \quad (22)$$

where T_O is the coolant temperature at the core exit. This relation indicates that power level variations due to thermal reactivity noise can be held within acceptable bounds provided the temperature of the NaK at the heat dump exit varies by an amount which is small compared to the overall ΔT across the plant. In TREX this equates to controlling the temperature of the NaK at the heat dump exit to within a few degrees and it will be shown in the hydraulic control section that provisions have been incorporated to insure that this is accomplished.

Mechanical reactivity noise is a most serious concern in the TREX reactor. Mechanical reactivity noise can be introduced through small motions of a fuel element in the reactor, of the core within the vessel or of the external reflector. This type of noise can have either d-c components or can be oscillatory with a characteristic frequency spectrum. In a reactor with weak feedback mechanisms, mechanical reactivity noise can lead to significant variations in the reactor thermal power level. To gain a feel for the mechanical reactivity noise problem, consider a simple case in which a step increase in reactivity $\Delta \rho$ is introduced through the motion of some mechanical component. The response of the reactor power level versus time is shown schematically in Figure 51. Initially the power level increases due to the prompt jump with the percentage change in power level being equal to the number of cents of activity in the spurious signal. Following the initial prompt jump, the power level

OPEN LOOP RESPONSE

NOTES:

K = static power coefficient of reactivity

T = effective delayed neutron lifetime ($\sim 13 \text{ sec.}$)

FIGURE 51. MECHANICAL REACTIVITY NOISE

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increases approximately linearly with time at a rate proportional to the spurious signal. After a long period of time, the incremental power level saturates at a value equal to the reactivity signal divided by the overall power coefficient of the reactor. Note that the prompt jump, the initial rate of power increase and the final incremental change in power level are all proportional to the magnitude of the spurious noise signal.

The equations shown in Figure 51 permit numerical estimates of how the frequency of control action is influenced by the magnitude of the spurious reactivity signal. For example, assume that $\pm 3\%$ variation in reactor thermal power level will be tolerated. Then for a spurious reactivity input less than approximately $2/3\%$, no control action is required since the feedback mechanism will be adequate to regulate the thermal power level of the reactor within the $\pm 3\%$ band. For a spurious reactivity input greater than approximately $2/3\%$, a control action will be required. For a 1% spurious step of reactivity, for example, approximately 26 seconds will elapse before reactor power level will rise above the permissible 3% band. A control action would be required at this point.

Two conclusions can be immediately drawn from the preceding results. First, the TREX reactor will self-regulate against spurious reactivity signals of the order of tenth of cent. Hence, the smallest incremental reactivity step required by the control system is of the order of a tenth of a cent. The second conclusion that can be drawn is that there is considerable incentive to reduce the sources of mechanical reactivity noise in TREX. This implies that the mechanical design of the core and internal and external reflectors should be such as to prohibit the motions of components which could lead to reactivity inputs of the order of a cent. This can be accomplished through the use of tight core structure and careful control of the mechanical linkages between the reactor and the external reflector.

Control drum specifications which have been selected for TREX are shown in Table 29. The worth per drum is 66% which is obtained by dividing the total \$8 drum worth by 12 drums. Based on the preceding noise consideration, the actuators have been geared to permit a total of 700 reactivity steps between the full out and full in positions. This equates to a minimum reactivity step of approximately a 0.10% which

TABLE 29
CONTROL DRUM SPECIFICATIONS

WORTH/DRUM (β)	66
MAXIMUM NUMBER OF STEPS	700
MINIMUM REACTIVITY STEPS (β)	0.094
STEPPING RATE (SEC^{-1})	4
INSERTION-WITHDRAWAL RATES (β/SEC)	0.37
FULL INSERTION OR WITHDRAWAL TIME (SEC)	176
SCRAM TIME – FULL WITHDRAWAL (SEC)	0.5

is adequate in light of the preceding noise considerations. A four step/second stepping rate on the drums has been selected to provide a reactivity insertion/withdrawal rate per drum of approximately $37\text{¢}/\text{second}$. In addition, the stepping rate of 4 steps/second implies a total full insertion or withdrawal time for a drum of 176 seconds. The scram time for 90% withdrawal of drum reactivity has been selected at $1/2$ second.

The control drum specifications of Table 29 are for a single control drum. The manner in which the 12 drums are operated to achieve coarse and fine control is indicated in Table 30. The coarse control is achieved by operating eleven of the twelve drums in a ganged control mode. When this is done, the reactivity step is approximately 1.03¢ and the insertion/withdrawal rate is approximately $4\text{¢}/\text{second}$. The control mode for the eleven ganged drums is manual, i. e., the drums are operated directly by the reactor operator. Course manual control will be used to start up the reactor, achieve a given power level and shut down the reactor.

As previously mentioned, concern exists over the tendency of the thermionic reactor power level to drift due to the weak reactivity feedback mechanisms. For this reason, a fine or shim control mode has also been provided. This is accomplished by using the twelfth drum as an individual shim control drum. The reactivity step is approximately $0.37\text{¢}/\text{second}$ for the shim control drum. The shim control drum can be operated in either a manual mode by the reactor operator, or be placed in an automatic level control mode as indicated in the block diagram of Figure 49. In the automatic control mode, the shim control drum is used to regulate the power level of the reactor to a set point prescribed by the reactor operator.

Major interlocks envisioned for the reactivity control system are also indicated in Table 30. There are three interlocks. The first requires that single drum control actions be restricted to the preselected fine control drum. This insures that the eleven ganged drums are always operated as a group and are never permitted to get out of phase. This has two advantages. First, it insures that thermal power distributions

TABLE 30
CONTROL DRUM OPERATION

COARSE CONTROL

Number of Drums	11 (ganged)
Reactivity Step (β)	1.03
Insertion-Withdrawal Rate (β /sec)	4.13
Control Mode	manual

FINE CONTROL

Number of Drums	1
Reactivity Step (β)	0.094
Insertion-Withdrawal Rate (β /sec)	0.37
Control Mode	manual/ automatic

INTERLOCKS

- Single drum control actions are restricted to the pre-selected fine control drum.
- Remaining worth on fine control drums must be less than 10 β for automatic level control.
- Manual control supersedes automatic.

throughout the reactor will not be distorted as a result of non-symmetrical control drum patterns. Second, it facilitates keeping a running index of the total inserted reactivity by counting the total number of pulses which have been applied to the eleven drum bank. The shim control drum will be selected prior to reactor startup and can not be changed until shutdown. At that time, a different drum may be selected as the fine control drum to avoid premature failure of one of the actuators due to the frequent control actions which may be required on the shim control drum. The second interlock requires that the remaining worth on the fine control drum must be less than 10¢ before automatic power level control is permitted. This interlock is incorporated for safety reasons and prevents the handling of a large amounts of reactivity by the automatic power level control system. In practice, the fine control drum will be inserted prior to reactor startup to the point where the remaining worth on the drum is approximately 5¢. The reactor can then be started up in the manual control mode and switched into the automatic level control mode once the desired power level had been achieved. The plant will revert to manual control if the remaining worth on the shim control drum ever exceeds 10¢. The third interlock provides that manual control supercede automatic control. This permits the reactor operator to take over manual operation of the reactor at any time.

Figure 52 shows the TREX nuclear controls as viewed by the reactor operator. Control actions are accomplished through a select and operate switch. The select switch has 13 settings. The twelve control drums can be selected individually, or gang operation of drums 1 through 11 can be selected. Drum 12 is considered to be the fine or shim control drum. The operate switch permits reactivity to be manually inserted or withdrawn on the selected control drum(s). Prior to reactor startup, drums 1-11 can be individually inserted and withdrawn for checkout. Selection of any of drums 1-11 will automatically place all other drums on a reactivity rundown sequence, and the selected drum cannot be inserted until all other drums are in the least reactive position. Once drums 1-11 have been checked out, the reactor operator may initiate the reactor startup sequence by selecting the fine control drum 12 and inserting it to within approximately 5¢ of its limit. He then selects gang operation of drums 1-11 and manually starts up the reactor and

FIGURE 52. CONTROL DRUM OPERATION

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achieves the preselected power level. Once this is established, he can, if desired, switch to automatic level control. At any time the reactor operator is permitted to make manual control actions involving either the shim control drum or the gang of drums 1-11. If at any time the reactor operator selects drums 1-11 during operation of the reactor, the reactor will be put on a reactivity rundown sequence.

4. Hydraulic Control

The schematic representation of the TREX heat removal system is shown in Figure 53. It consists of a primary loop which transports the heat to an immediate heat exchanger and a secondary loop which transports the heat to a heat dump. Electromagnetic pumps are incorporated in both the primary and secondary loops. In addition, electrical heaters are incorporated in both loops to facilitate service and maintenance on the heat removal system and to permit the NaK system to be electrically preheated to approximately 780°K prior to reactor startup. The heat dump contains a blower for providing air flow and a set of louvers which may be used to regulate the air flow.

The primary and secondary pumps and heaters and the heat dump blower in the TREX heat removal system will be operated manually from regulated power supplies. Prior to reactor startup, full flow will be established in both the primary and secondary pumps and sufficient power will be delivered to the heaters to raise the loop temperatures to slightly in excess of 780°K with the heat dump louvers closed. The NaK temperature at the heat dump exit will then be controlled to 780°K by placing the louvers in automatic control mode as shown schematically in Figure 53. Prior to reactor startup, the heat dump louvers will be only slightly open. As the reactor is started up, the louvers will automatically open to accommodate the increased heat load while maintaining the NaK return temperature at 780°K .

5. TREX Electrical Control

TREX electrical system is shown schematically in Figure 54. Thermionic fuel elements are connected electrically in pairs, one positive and one negative with respect to ground. The pair of TFE current leads are fed to a power conditioning module. The power conditioning module in TREX serve three important functions. The first is to control the electrical current extracted from each TFE pair. The TFE pair current, I_1 , is controlled to a demand signal, I_1^* , as shown schematically in Figure 54. The demand signal I_1^* is linearly related to the measured thermal power level of the reactor as will be discussed subsequently. The second function of the power conditioning modules is to perform a reliability function in the event that an open circuit failure is encountered in either TFE or a TFE pair. An electronic switch located at the input terminals of the power conditioning modules can be closed in this instance to convert the open circuit to a short circuit and permit the power from the unfailed element to continue to be processed. The third function of the power conditioning module is to step up the dc voltage level to a value which is consistent with the demands of the power user.

The output terminals of the power conditioning modules are connected in parallel on the d-c bus which contains the electrical load. In addition, the dc bus will contain a ballast load and associated control circuit as shown schematically in Figure 54. To understand the role of the ballast load in TREX, it is helpful to first discuss methods for regulating the dc bus voltage on an eventual thermionic power plant flight system.

Two methods for regulating the dc bus voltage are envisioned for the eventual thermionic power plant. The first involves the use of a ballast load on the dc bus. The ballast load may be varied by means of electronic switching to maintain the dc bus voltage constant. This type of control provides rapid and precise regulation of the bus voltage over a range of power levels dictated by the capacity of the ballast load. A second method for regulating the bus voltage on the eventual thermionic power plant consists of controlling the reactor thermal power

FIGURE 54. TREX ELECTRICAL CONTROL

level in response to an error signal from the dc bus. This is shown schematically by the dashed control feedback in Figure 54. In this control scheme, if the bus voltage were to fall below the desired value, reactivity would be inserted raising the thermal power level of the reactor. This, in turn, would raise the electrical current extracted from each of the TFE pairs and would increase the bus voltage. The process would continue until the error signal derived from the dc bus was reduced to zero. This second method for regulating the dc bus voltage in a thermionic power plant is attractive in that it reduces or even eliminates the need for a ballast load. On the other hand, this method of bus voltage regulation is much slower than that which can be accomplished with a ballast load, and requires frequent movement of the control drums on the reactor. It is felt that a combination of the two control schemes will probably be employed on the eventual power plant. Variations in ballast load will be employed for fast regulation of the bus voltage in response to small and perhaps frequent changes in the user electrical load. Gross changes in electrical load will be accommodated on a slower basis by changing the power level of the reactor. The capacity of the ballast load and the relatively frequency of ballast load control actions versus reactivity control actions will be dictated in large measure by the mission power profile and the need for fast load change capability by the user.

In TREX, a user load will not be present. For this reason a 100 kWe capacity ballast load will be employed on the plant and provisions will be incorporated to regulate the dc bus voltage using the ballast load as shown schematically in Figure 54. During the latter phases of the experimental program, a feedback from the ballast load to the reactivity control system may be incorporated.

Figure 55 shows the electrical control law which will be employed to control the TFE pair electrical currents. This control law is based upon the fact that, for a fixed emitter temperature, the electron cooling per ampere in a thermionic converter is nearly constant over a broad range of current densities and is only weakly influenced by variations in electrode work functions, cesium reservoir temperature

$$I_i^* = \text{MAX} \begin{cases} I_o \\ A_i + B_i P_T \end{cases}$$

FIGURE 55. TFE CURRENT CONTROL

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and interelectrode spacing. The solid curve in Figure 55 calls for controlling the TFE pair electrical current at open circuit until a threshold reactor thermal power level is reached. At this point the TFE emitters will have achieved operating temperature and will be rejecting heat to the collectors through radiation, cesium conduction and structural losses only. Once this threshold reactor thermal power level is reached, additional increases in reactor thermal power will be accompanied by corresponding increases in the TFE pair electrical current to balance the electron cooling heat transfer against the increased power generation. In this manner the emitter temperatures of thermionic fuel elements are held approximately constant over a range of electrical loads from effectively zero to 100%.

Due to sheath insulator arcing concerns, it may be undesirable to operate the thermionic fuel elements at open circuit as the emitter approaches full operating temperature. This can be prevented in two ways. One would be to provide a small current demand on the fuel elements low reactor power levels as indicated by the dashed curve in Figure 55. Another way is to incorporate an internal shunt into the thermionic fuel element to reduce its open circuit voltage to permissible levels.

The TFE electrical current control scheme depicted in Figure 54 has a number of attractive features which are discussed in the following paragraphs.

1. The electrical control is modular. As each TFE pair is added to the reactor, a power conditioning module is incorporated and the output is put on the existing dc bus. Since there is no feedback from the electrical to the nuclear/hydraulic control systems, nuclear and hydraulic control of the plant are unchanged. These features facilitate building up from an all-driver core to an all-TFE core in a stepwise fashion.
2. Operation of TREX from the reactor operator's viewpoint is similar to that of any reactor. Specifically, the reactor operator

is primarily concerned with reactivity, thermal power and hydraulic variables and is not routinely concerned with TFE electrical currents or cesium reservoir temperatures. The TFE electrical current are controlled with response to the thermal power level of the reactor, but do not influence the thermal power in a feedback sense. The reactor operator may, of course, be required to respond to alarm signals originating in the electrical systems of the plant.

3. Flexibility exists in the choice of TFE operating conditions. Specifically, the constants A_i and B_i may be varied for different TFE pairs to reflect non-uniformities in the power distribution in the reactor and/or the desire to operate different TFEs at different emitter temperatures.
4. Reduced thermal cycling of the emitters occurs as the thermal power level of the reactor or electrical load are changed. During a scram the electrical current from the TFE is immediately reduced to zero permitting the emitters to radiation cool to the collectors. This leads to a far less severe thermal shock than will occur if the current continues to be extracted from the TFE pair even after the reactor thermal power has been reduced to zero.
5. Control of the cesium reservoir temperatures is straightforward. The optimum cesium reservoir temperature in a thermionic fuel element is strongly dictated by the operating emitter temperature and only weakly influenced by other variables. Hence, the decision to operate at fixed emitter temperature permits the TFE cesium reservoir temperatures to also be fixed. This is accomplished through a simple set point control of the cesium reservoir temperature as shown schematically in Figure 54. The TFE cesium reservoir temperatures will be established prior to reactor startup and will normally not be changed until after reactor shutdown.
6. Since the electron cooling per ampere in a thermionic converter is only weakly sensitive to variations in electrode work functions,

interelectrode spacing and cesium reservoir temperature, the control law depicted in Figure 55 permits the thermionic fuel elements to be operated at the desired emitter temperature even if variations in these quantities are occurring. It is significant to note that the output voltage for a thermionic converter is strongly influenced by changes in the electrode work functions, interelectrode spacing and cesium reservoir temperature. Hence, electrical control schemes which employ output voltage as a control variable are extremely sensitive to variations in these quantities.

7. The present electrical control scheme is permissive of cell and TFE failures. In the event of a cell short, for example, the selected electrical current control law will preserve the desired current and temperatures in the remaining cells of the thermionic fuel element. In the event of a seal failure leading to TFE open circuit, a protective switch in the power module will short out the open circuited element. The current in the unfailed element will not be changed, and therefore, the temperatures in that element will not be changed. In the event of a cracked sheath insulator and an electrical arc to ground, the output of the cells between the short and ground will of course be lost. The remaining cells in the fuel element which has suffered the arc and all of the cells of the sister element, however, will continue to operate in exactly the same fashion as if the arc had not occurred. These attractive features are not present in electrical control schemes which are based on sensing and control of the TFE output voltage.

8. In the event of cell degradations and failures, the power level of the reactor is raised to make up the voltage deficiency on the dc bus. Hence, the effects of failure are shared by all the remaining cells in the reactor; not by a few cells in the vicinity of those which have failed.

9. A faster reactivity feedback from core structure and coolant is accomplished in the present electrical control scheme. Specifically, if the thermal power level of the reactor is raised by an

incremental amount, this same amount of energy is immediately transferred into the collector by means of the electron cooling process. The long thermal time constant associated with the fuel and emitter is thus short circuited.

10. The present electrical control scheme precludes the possibility of thermionic burnout. Figure 56a shows plots of emitter temperature versus thermal power for a constant voltage electrical control law and for the present electrical control law. In the case of constant voltage control, the curves exhibit a burnout character. The dashed curve corresponding to an output voltage of 0.9 volts/cell is representative and a typical full power operating point on this curve is shown. With the constant voltage control law, burnout can be accomplished in any of several different ways. One is to increase the thermal power of the reactor while continuing to request the same voltage. Another is to reduce the operating cesium reservoir temperature in one or more thermionic fuel elements. This has the effect of shifting the family of curves to the left in Figure 56a and will eventually lead to the operating point being above the critical heat flux. These two methods of thermionic burnout can, in principle, be prevented by maintaining control over the reactor thermal power level and over the operating cesium reservoir temperatures. A third method for achieving thermionic burnout in the constant voltage control scheme is more difficult to avoid, however. This occurs when some of the cells in a series string either degrade or fail, thereby, causing the other cells to be operated at higher output voltage. Note (Figure 56a) that the critical heat flux for burnout is less at the higher output voltages. Hence, failure of a few cells in a series string can quickly lead to a situation in which the remaining cells are operated past the burnout limit and a runaway situation develops. The probability of this event taking place is greater in reactors having few cells per series string than in reactors having many cells per series string.

The emitter temperature versus thermal power relation for the present control law, Figure 56b does not show a thermionic burnout behavior. In particular, the emitter temperature rises to the full operating temperature at approximately 1/3 design rated power, and is constant for thermal powers above this level.

$T_C = 1000^\circ\text{K}$
 $T_R = 610^\circ\text{K}$

FIGURE 56. INFLUENCE OF ELECTRICAL CONTROL LAW ON THERMIONIC BURNOUT

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Enumerated below are features of the TFE control scheme which require additional investigation.

1. The electrical control described herein which employs control of the TFE current assumes the cells to be stable thermally. If thermal changes do occur such as that associated with emissivity changes, or because of reduced thermal impedance across a spacer pin, the current control does not take it into account. Calculations indicate the worst such case to cause 100°C change in the emitter temperature.
  2. A constant gain (voltage control) power regulation module can be made lighter than a pulse-width modulated or a phase-angle controlled (current control) power regulation module. Estimates of the difference amount to 1 - 1.5 lb/kW on the overall system performance, and occur due to larger filtering requirements for the presently envisioned current control scheme. However, due to the loss of system reliability when using the voltage control scheme, the additional power plant component redundancy necessary to provide a given end-of-life power requirement is likely to cause a system weight penalty which is far greater than that paid above.
  3. A reduction in modularity, by combining more TFEs per regulation unit or by using low voltage rotating conversion equipment (acyclic motors) may yield a direct weight savings over the recommended approach. However, the restoration of system reliability is once again likely to be the more important weight factor.
6. Plant Startup and Shutdown

The plant startup and shutdown sequence is shown in Figure 57. During the first phase of plant startup the coolant system is electrically preheated to 780°K and full flow is established in the primary and secondary EM pumps. These operations are accomplished manually. The final step in preparing the coolant system for reactor startup is to place the heat dump louvers in the automatic control mode so that the NaK temperature at the heat dump exit is automatically controlled to approximately 780°K .

The second phase of plant startup involves the cesium reservoirs. With the reactor preheated to 780°K , the cesium reservoirs can be brought to full operating temperature without distilling cesium into the fuel element internals. When this is accomplished, the cesium reservoirs are placed in an automatic set point control mode to maintain the desired operating temperatures.

The third phase of reactor startup involves supplying switching power to the power conditioning modules and establishing the TFE current demand.

With the coolant system, cesium reservoirs and power conditioning system prepared for plant startup, as described above, the thermionic reactor can be started. The first step in the reactor startup sequence is to insert the fine control drum to within approximately 5¢ of its full in position. The reactor is then manually brought to power on a slow period using the coarse control drums. Once the desired operating power level is achieved, the reactor can be placed in the automatic power level control mode if desired.

The plant shutdown sequence is the inverse of the startup sequence described above.

- **COOLANT SYSTEM**

PREHEAT TO 780°K

ESTABLISH FULL FLOW

HEAT DUMP LOUVERS IN AUTO CONTROL

- **Cs RESERVOIRS**

HEAT TO OPERATING TEMPERATURE

ESTABLISH AUTO CONTROL MODE

- **POWER CONDITIONING**

ESTABLISH TFE CURRENT DEMAND

- **REACTOR STARTUP**

INSERT FINE CONTROL DRUM TO DESIRED POSITION

START REACTOR USING COURSE (FINE) DRUMS

ACTIVATE AUTO LEVEL CONTROL (OPTIONAL)

FIGURE 57. PLANT STARTUP AND SHUTDOWN

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7. Plant Safety System

Three actions have tentatively been selected for the plant safety system in response to trouble signals from one or more of the various plant components. The first is a reactivity insertion block which prevents either the reactor operator or the automatic power level control system from inserting reactivity.

The second safety system action is a rundown which would involve withdrawing reactivity from the reactor at the maximum rate which is approximately 4¢/second. The rundown action can be either sealed or unsealed and a decision of which is the best option has not been made at the present time. If the rundown action is unsealed, it could be interrupted by the reactor operator and the plant could be restored to full power operation. In the event that the rundown action were sealed, it would go to completion and the reactor operator could then restart the reactor if desired. In this sense, the rundown would be similar to a slow scram. The incentive for making the rundown a sealed action is that it avoids placing the operator in a situation where fast decisions are required. On the other hand, making the rundown a sealed action has the disadvantage in that it could lead to a several hour shutdown of the plant for relative minor event

The third safety system action is the scram which is, of course, a sealed action. It is presently envisioned that the scram would be accompanied by a programmed flow reduction in both the primary and secondary coolant systems to avoid excessive thermal shock on the reactor core. By reducing the primary and secondary flow rates, the rate at which the ΔT across the reactor core collapses can be greatly reduced.

Tables 31 and 32 contain a tabulation of various trouble signals which may be detected in several plant subsystems and indicates a tentative selection of a safety system response to each trouble signal. These trouble signals were considered in the selection of the three safety system responses described above. No attempt has been made at the present time to prescribe the sensing logic or trip points for the several trouble signals.

TABLE 31

TREX SAFETY SYSTEMS RESPONSES

System	Trouble Signal	Action		
		Block	Rundown	SCRAM
Nuclear Instrumentation	Malfunction (SRM)	X		
	Low Flux (SRM)	X		
	Short Period (SRM)	X		
	Malfunction (IRM)			X
	Short Period (IRM)	X		
	Short Period (IRM)			X
	Malfunction (WRM)			X
	Low Flux (WRM)			X
	High Flux (WRM)	X		
	High Flux (WRM)			X
Electrical Power	Loss of Power to Reflector Drive Motor			X
	Loss of Power to Primary Pump			X
	Loss of Power to Secondary Pump			X
	Loss of Power to Heat Dump Blower			X
	Loss of Power to Building			X
Primary Coolant	Low Level in Expansion Tank			X
	High Level in Expansion Tank			X
	Low Temperature at Core Outlet			X
	High Temperature at Core Outlet			X
	Low Primary Flow			X
	High Temperature in Primary Pump Winding			X
	High Activity			X
	High Reactor Dome Pressure			X

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TABLE 32

TREX SAFETY SYSTEM RESPONSES

System	Trouble Signal	Actions		
		Block	Rundown	SCRAM
Secondary Coolant	Low Level in Expansion Tank			X
	High Level in Expansion Tank			X
	High Temperature at Heat Dump Exit			X
	Low Temperature at Heat Dump Exit			X
	Low Secondary Flow			X
	High Temperature in Secondary Pump Windings			X
	High Activity			X
Heat Dump	Low Flow			X
TFEs	Low Current			X*
	Low Cesium Reservoir			X*
	Hot Channel			X
Driver Fuel	Hot Channel			X
Power Conditioning	High Temperature		X	
Reactor Vessel and Cavity	High Reflector Temperature High Vessel Temperature NaK Between Double Wall			X
Radiation	High Activity in Building			X
Other	High Seismic Activity			X

* No action unless TFE is to be saved.

NP 2371

H. POWER CONDITIONING EQUIPMENT

1. Introduction

This section describes the Power Conditioning Equipment concept for the 100 kW(e) Thermionic Reactor Experiment and presents the conceptual designs of the electronic assemblies which are contained in the equipment. Clarification of the ground rules, on which the design is based, is established by specifying the design goals and the pertinent performance characteristics of the reactor experiment. This is hopefully achieved by comparing the design objectives of the ground based unit and the space application model. The introductory presentation is followed by an explanation of the subsystem block diagram and by a detailed description of each assembly. Control functions of the power conditioning equipment are also discussed in II.G.5.

2. General Description

The power conditioning equipment circuitry for the Thermionic Reactor Experiment differs from the space design in several ways. In the space unit, the output voltage is regulated by readjusting the reactor thermal power as the load demand changes. In the reactor experiment design, the output voltage control is performed by varying the load bank resistance. This allows study of thermionic performance at a constant thermal power input fixed by the engineer. After operational experience is gained it will not be difficult to remove the load bank controller and feedback the load voltage to adjust reactor thermal power to demonstrate reactor control concepts. Another difference is in the design objectives of the two subsystems. The main goal of the reactor experiment design effort is to produce a reliable unit at a very low cost. In the space design, where such constraints as the radiation effects, the thermal variations, the conversion efficiency, the allocated volume, and the subsystem weight must be considered, the design goals are obviously different. The objectives in space are to build a unit which is completely reliable, very efficient, small, light, and which adequately compensates for the thermal and radiation effects.

Figure 58 illustrates the planned arrangement of the Power Conditioning Equipment for the Reactor Experiment. The illustration shows two electronic assemblies, which are connected in parallel to a common dc bus. They are isolated from each other except at the thermal power input terminal and at the bus. To accomplish the high reliability objective, a circuit breaker protection scheme is planned at these locations.

The complete subsystem, with a 100 kw capacity, consists of approximately 75 modules. Each module is comprised of two Thermionic Fuel Elements (TFE's), a dc converter, a control unit, a signal generator, a low voltage power supply (L.V.P.S.), and a fault detection and isolation circuit.

The TFEs are series connected in pairs with the center connection grounded. Their outputs are wired to the dc converter, where the low voltage dc output of the TFEs is transformed to a higher voltage for transmission and use. The converter is driven by two pulse trains, whose characteristics are determined by an error voltage, derived by comparing the actual current, I , and the desired current, I^* . The actual current signal is obtained from the current sensor in the TFE current loop, while the desired current signal is derived from the signal generator. There, it is produced according to the constant emitter temperature law as expressed by equation $I^* = -A + BP_T$. A and B in this expression are constants which are set at some predetermined level for each TFE pair. See II.G.5.

In addition to providing the actual current, I , to the summing network, the current monitor acts as a controller of the fault detection and isolation equipment. For example, if an open circuit occurs anywhere in the TFE current loop, the current sensor produces a zero level fault signal. This command initiates the closure of the appropriate contactors to short out both TFE power sources. The action either isolates the faulty TFE and prevents the output voltage of the sister element from going into the open circuit region, or it avoids the open circuit condition for both elements, when the failure is in the load.

FIGURE 58. ELECTRICAL CONTROL AND POWER PROCESSING SUBSYSTEM

If the fault occurs in one TFE, that element remains shorted while the other TFE unit is inserted into the power processing loop. However, if the fault is in the load, both power sources stay short circuited. For this condition, the whole assembly is considered to be inoperative.

An important reliability improvement, which has been incorporated in the reactor experiment design, is the low voltage power generation technique. Instead of building one low voltage power source for all the electronic assemblies, a decision was made to provide a separate power source for each unit. The concept is compatible with the modular design approach chosen for this equipment.

All the low voltage power supplies are diode coupled to a single auxiliary power source. The function of this source is to provide the hotel load power during the thermionic power plant start up, and also to serve as a redundant power source when the plant is operating in the quiescent state.

The output voltage regulation is performed by adjusting the load bank resistance. This is accomplished by a voltage sensing circuit which directs a stepping switch. For each step in the advance direction some resistance is added to the load bank. Each retarded step removes an equal fraction of the load bank resistance.

3. Electrical Design Description

Figure 59 represents the conceptual implementation of the equipment. Some circuitry, however, is omitted from the schematic. These circuits are the Low Voltage Power Supply, the Load Bank and the Fault Detection and Isolation Assembly. They will be discussed separately.

The schematic is divided into six blocks. Each section is given an appropriate title to emphasize the primary purpose of the enclosed circuitry. A pictorial illustration of some wave forms is also given to facilitate the functional description.

The power source section is an area which contains a pair of TFE units and various components that are closely associated with the fuel elements. Each TFE has an instrumentation shunt a set of diagnostic terminals, a single pole switch,

FIGURE 59. ELECTRICAL CONTROL AND POWER PROCESSING CIRCUIT FOR THE
100KW THERMIONIC POWERPLANT

and an augmenting power supply. This supply is used when the TFE voltage degrades to some predetermined minimum value. Normally, however, the single pole switch is closed, thus the augmenting power source is out of the circuit. The instrumentation shunts are used as monitors of the TFE current. Their outputs are continuously sampled by the data system and/or are displayed for visual inspection.

The diagnostic terminals are provided for the purpose of performing the I-V characteristics studies. These tests are initiated by energizing K1, K2, K3 and K4. The sequence of contactor operations is described in the following discussion.

One requirement of the power conditioning equipment is that it continue to process power from one TFE when the other unit has either failed or when its I-V characteristics are being examined. Also, any TFE must be short circuited when a failure occurs within the unit to prevent damage to its sister element. In addition, both TFEs must be shorted when the failure occurs in the load.

To achieve these objectives, a means of shorting the faulty unit, or both units, or isolating the TFE which is being tested, a provision for completing the current path for the element whose power is being processed is incorporated in the design. These functions are provided by contactors K2 and K3. For example, if the top TFE fails, closure of contactor K3A provides a short circuit to the faulty unit and a current path for the bottom TFE. However, since the potential across C1, at this instant, is equal to the output voltage of the top TFE, the capacitor must be removed from the circuit to avoid the voltage transient from appearing across the bottom TFE. This is accomplished by de-energizing K4A before K3A is closed. For the diagnostic measurement case, the sequence of events is somewhat different. Here, K4 is again energized first, but K2 must be opened before the test is initiated.

The input filter consists of two capacitors, a pair of normally closed contacts, and the bleeder resistors. As mentioned in the previous paragraph, the contacts K4A and K4B are opened only when the TFE

failure occurs or when the I-V characteristic are being examined. For the power source failure condition, one capacitor is permanently disconnected, while for the diagnostic measurement case the capacitor is removed before the initiation of the test cycle, discharged during the test, and again inserted into the power processing loop when the test is terminated.

The dc converter is composed of the main power transformer, the full wave rectifier bridge, the output filter, the power switching transistors, and the base drive circuitry. The transformer has a low voltage center tapped winding, current sensing coils on the primary side and a high voltage winding on the secondary side. In addition, the secondary has all the required coils for the development of the internal power sources. This arrangement of the secondary is not shown in Figure 59. It will be illustrated when the Low Voltage Power Supply is discussed.

The TFE current is switched through the primary by seven power transistors connected in parallel and driven by a constant current source. The source is a modified Darlington amplifier which is directly driven by commands from the pulse width modulator.

The signal generator section is composed of the desired current, I^* , generator, the summing amplifier network and several buffer amplifiers for isolation purposes. The actual current, I , signal is derived from the current sensor. The desired current, I^* , signal is produced by the current generator. These signals are of opposite polarity, so the summing amplifier output represents the difference, I^*-I .

The pulse width modulator consists of two auxiliary integrated circuit shaping networks, the current limit circuit and several transistor current amplifier stages. The function of the circuit is to produce pulse trains whose duty cycle is proportional to the magnitude and polarity of the input error voltage. Since the error voltage is a direct measurement of the difference between the desired and the actual currents, the duty cycle of the pulse train changes to force the error signal to a zero level. Thus, the current drawn from the TFE is regulated as required by $I = -A + BP_T$.

a.) Low Voltage Power Supply Description

Figure 60 describes the Low Voltage Power Supply Assembly. The schematic shows two low voltage power sources which are diode coupled to the auxiliary source. In the complete power plant there are actually 75 of these units, one of each TFE pair. The output current and voltage requirements of the components, shown as blocks in the illustration, are specified and the sub-system operation is described.

During power plant startup, the auxiliary supply provides the base drive current to the converter power transistors and power to other electronic components of the power regulation equipment. When the output voltage level of any particular regulator exceeds the auxiliary voltage magnitude to which it is coupled, the auxiliary source current is cut off. The power is then provided by the TFEs. This transfer of power sources continues until all the internal power is supplied by the thermionic power plant. At this time the auxiliary power source becomes a redundant supply.

b.) Load Bank Description

The load bank is designed to self-regulate the plant output voltage at 100 volts $\pm 10\%$. This is accomplished by a voltage sensing circuit, as illustrated in Figure 61, which directs the load stepping switch. In the illustration, the stepping switch is represented by contacts K5, K6, etc. For each step in the advanced direction, a contactor adds some resistance to the dc bus and for each retarded step the same resistance is removed. Thus the bussed output voltage is held constant at 110 VDC $\pm 10\%$ by load regulation.

The above discussion pertains to the voltage regulating portion of the load bank, which is designed to dissipate approximately 20KW. The rest of the power plant output is dissipated in the resistor bank which is controlled by an operator. He may change the resistance of this load bank by either closing or opening contacts K1, K2, K3, K4, etc. This option is required since the plant may be operated at various power output levels. In addition, during buildup of the TFE core, the load can keep pace with the number of TFE pairs in the reactor.

FIGURE 60. LOW VOLTAGE POWER SUPPLY ASSEMBLY

FIGURE 61. LOAD BANK FOR THE 100kW POWERPLANT

c.) Fault Detection and Isolation Assembly Description

The schematic of the Contactor Driver and Logic Circuit, as illustrated in Figure 62, presents the conceptual design of the required equipment. The circuit is part of the Fault Detection and Isolation Assembly. Specifically, it represents the isolation portion.

The contactor operation is performed sequentially to avoid the open circuit condition for the TFE. In the normal mode, given in the truth table, the contactors K1 and K3 are opened while K2 and K4 are closed. When the fault occurs, either in the TFE current loop or in the load circuit, the fault command is present, so K4 opens to isolate the filter capacitor and K3 closes to short circuit the TFE. If the failure is in one of the power sources, the operator can remove the faulty unit from the power processing loop by applying the inhibit command to the contactor driver and logic assembly. If the failure occurs in the load, all the TFE units are short circuited. This condition produces a command to scram the reactor.

The contactor operation for the I-V characteristic measurements is also specified in the truth table. The operational sequence is designated by the circled numbers for the "on" and "off" modes. In both cases the contactors are energized in an order which prevents the TFE from seeing an open circuit.

Figure 63 presents the fault detection part of the Fault Detection and Isolation Assembly. The equipment consists of 75 signal processing circuits which monitor the current flowing in each TFE pair and the load.

If the load current drops down to zero, all the outputs assure a "O" state. This condition represents a catastrophic failure; therefore the reactor is scrammed. However, if the failure occurs in the primary loop of one power conditioner, only that unit is disabled either temporarily or permanently, while the others continue to process the TFE power.

CONDITION	A	B	C	D	E	F	G	H	K1	K2	K3	K4
NORMAL	1	1	0	1	0	0	1	1	0	C	0	C
FAULT COMMAND PRESENT	0	1	0	1→0	0→1	0	1	1	0	C	C (2)	0 (1)
DIAGNOSTIC SW "ON"	1	0	1	1→0	0→1	0→1	LATCH K4 1→0	LATCH K1 1→0	C (3)	0 (4)	C (2)	0 (1)
DIAGNOSTIC SW "OFF"	1	1	0	0→1	1→0	1→0	0→1	0→1	0 (2)	C (1)	0 (4)	C (3)

FIGURE 62. SOLENOID DRIVER AND LOGIC CIRCUIT

FIGURE 63. FAULT DETECTION LOGIC

4. Packaging of the Electronic Assemblies

The power conditioning equipment is packaged in two major assembly areas; the power processing equipment area and the control circuitry area. These locations are discussed in section II.M. of this report.

The mechanical configuration of the power processing module is illustrated in Figure 64. The module contains the electrical components in the dc converter, the filter, the contactor and the power source sections of the electrical control and power processing circuit.

This equipment is packaged in standard relay racks, as shown in Figure 65. Each rack accommodates six power processing modules which are cooled by the air drawn at the bottom front and exhausted through the top. The air is drawn by the blower equipment which is located in the bottom compartment.

The permanent connection for each power processing module is made at the rear, behind the hinged doors. Therefore, the front panel requires only the augmenting power supply receptacles and the single pole switches.

The layout of the control circuitry module is illustrated in Figure 66 and the packaging of the units is shown in Figure 67. The equipment is cooled in the same manner as the power processing hardware. There are 18 modules per rack. Each module has a front panel which contains the inhibit and diagnostic switches, the "A" and "B" adjustments, the low voltage power supply outputs, the indicator lights and the current error display.

Packaging of the 100 kW Load Bank is comprised of two major assemblies; the weather-protected resistor load dump and the control and contactor module for indoor installation. The load dump requires approximately 180 cubic feet of space if forced air-cooled or about 480 cubic feet if natural convection-cooled. It is located in a shed outside the main test facility. The control and contactor module, whose arrangement is presented in Figure 68 is installed in a relay rack housed in the test facility control room. The mechanical design of this equipment is shown in Figure 69.

1. H.V. OUTPUT FILTER INDUCTOR
2. RECTIFIERS
3. H.V. OUTPUT FILTER CAPACITOR
4. TRANSISTORS
5. TRANSFORMER
6. BASE DRIVE CIRCUITRY
7. INPUT FILTER CAPACITORS
8. CONTACTORS
9. ALUMINUM PANEL
10. CURRENT SENSOR
11. INSTRUMENTATION SHUNTS

FIGURE 64. POWER PROCESSING MODULE

FIGURE 65. POWER PROCESSING RACKS

FIGURE 66. CONTROL CIRCUITRY MODULE

FIGURE 67. CONTROL CIRCUITRY EQUIPMENT RACKS

- 1. LEVEL DETECTORS
- 2. SOLENOID DRIVERS
- 3. CONTACTORS

FIGURE 68. CONTROL AND CONTACTOR MODULE

FIGURE 69. LOAD BANK CONTROL AND CONTACTOR RACKS

I. INSTRUMENTATION AND CONTROLS

1. Design Philosophy

The TREX instrumentation and controls must serve two general purposes. They must provide the indication and control necessary for the safe operation of the reactor and associated systems (Figure 70), and it must provide a permanent record of reactor and systems performance and life history for evaluation and analysis. Where safety considerations permit, one instrumentation channel may serve more than one purpose, but in general, past experience indicates that the reactor and plant operations should be both physically and electrically separate from the data recording and evaluation operations. A design objective is for the instrumentation to be state-of-the-art.

A prime requirement of the instrumentation system is that it shall allow the reactor to continue to operate even if the automatic digital data system should fail. A normal startup of the reactor would probably not be made without having the digital data system available to record the experiment parameters, but the operator should not be completely dependent upon the digital data system for indication or alarm functions.

The key parameters of each system will have individual signal conditioning and high and low analog alarm limits as required. The analog signal will be used to power meters on secondary panels, high and low trip circuits, continuous or multipoint strip chart recorders, and the readout to the automatic digital data system. The high and low trip circuits will supply signals to an annunciator system with a "First out" capability and both audible and visual indication. When a key parameter alarm is received, the audible alarm sounds, the annunciator panel light associated with the alarmed system begins flashing and a light next to the alarmed parameter indication on its graphic panel also flashes to indicate the particular trouble spot.

FIGURE 70. TREX ELECTRICAL CONTROL

Should the alarm be of a precipitous nature, only the annunciator and the graphic panel lights associated with the first alarm to occur will flash. Secondary alarms will result in a steady light to indicate their tripped condition. Pressing an acknowledge button on the reactor console removes the audible alarm and causes the flashing lights to become steady. Subsequent alarms will again initiate the audible alarm and "First out" capability. An alarmed point returning to normal will extinguish the associated lights.

High and low alarm conditions are also used by the safety system to either prevent reactivity increase or to scram the reactor as required. If multiple sensors are used for any parameter, such as reactor inlet plenum temperature, the safety logic will provide for 2 out of 3 or better logic level as required before initiating safety action. The temperature indicated on the graphic panel will be manually selected from the multiple signals available.

All parameters, including the key parameters above, will be continuously monitored by the automatic digital data system for out of limit conditions. Any one point being out of limits will result in a printout of the time, channel identification, and reading, on an alarm output typewriter. For those parameters with multiple sensors, any one reading being out of limits causes an alarm printout and is an "early warning" to look for trouble or detector malfunction. If the digital data system computer is appropriately programmed, it can also provide a safety logic which could be a non-control backup for the analog system. The digital data system provides the primary record of all experiment data and will be programmed to give a readout of the experiment data at preset time intervals or on demand and to print out data from all inputs on demand according to any one of several programs stored in the computer.

Several continuous and multipoint strip chart recorders will be available for trend type recording of selected parameters. Portable oscillographic or magnetic tape systems will be used for higher speed recording during special tests, such as transient testing, as required.

Figure 71 shows one proposed layout of the control room which provides separation and at the same time allows close communication between the reactor operator and data system operator.

The proposed startup procedure of the reactor calls for the reactor coolant inlet to be at operating temperature and flow before the reactor is brought on line, and the safety logic is so designed. At this point, the coolant loops are under automatic control and the reactor operator need only concern himself with nuclear instrumentation and actuator controls. Safety actions of "Interlock" or "Scram" originating from nuclear signals result in transients to the coolant loop automatic control, but require no manual action by the operator. For these reasons, the reactor control console is presently envisioned to contain only those indicators, controls, recorders, switches, and lights connected with reactor operation, namely, the nuclear instrumentation, actuator operation and indication, and automatic level control.

FIGURE 71. TREX CONTROL ROOM

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2. Nuclear Instrumentation

The nuclear instrumentation is made up of ten neutron flux measurement channels and associated recorders, indicators, and switching. Two source range monitors (SRM), three intermediate range monitors (IRM), and three wide range monitors (WRM) provide the overlapping indications and level trips necessary for reactor startup, operation, shutdown, and safety action. A separate WRM will be used to provide a signal for automatic level control of the reactor, once the desired operating level has been reached manually. A power range monitoring (PRM) channel will be used to supply a power level signal for automatic control of the TFE output currents, thereby providing automatic control of TFE emitter temperatures. The two channels used for automatic control are completely independent of the channels used for safety action.

Chassis for the nuclear instrumentation will be housed in panels visible to the reactor operator as shown in the Main Control Room layout. (Figure 71). These panels will also contain the level trip circuits, power supplies, and the SRM and IRM recorders. Remote indication of level and period for these channels along with range selection, level indication and recording of the WRMs will be provided on the reactor console.

A block diagram of typical nuclear channels to be used for reactor safety and control is shown in Figure 72 with the expected operating range of each channel shown in Figure 73.

FIGURE 72. TREX NUCLEAR CHANNELS

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FIGURE 73. OPERATING RANGES TREX NUCLEAR CHANNELS

a.) Neutron Detectors

All neutron detectors are mounted in wells at the circumference and slightly above the reactor core. The detectors are not moveable once they are positioned in the wells. Each detector is cooled by dry nitrogen to maintain the power and signal cables below 300°F. The temperature of each detector is monitored by two chromel-alumel thermocouples and displayed, monitored, and alarmed in the Main Control Room.

b.) Detector Power Supply

All reactor nuclear control channels are powered from a battery system supplying 100 ampere hours of + and -26.5 volts dc. The batteries are continuously charged from the facility emergency supply bus, but could sustain the nuclear instrumentation for 8 hours of continuous operation without other sources of power. Indicating recorders for the nuclear instrumentation are supplied from the facility emergency bus which is powered by a facility contained motor generator during loss of site electrical power.

c.) Source Range Monitoring (SRM) Signal Processing

The two source range monitoring channels operate from chambers located in wells around the reactor. Preamplifiers for the channels are located near the electrical connection dome. The amplified signal is fed to the source range monitor chassis located in the nuclear panels of the control room.

Each SRM is designed to count pulses over the range from 0.1 to 10^6 counts/sec. (seven decades) and to display this count in logarithmic form. The SRMs are also designed to indicate the reactor period based on the count rate level change and the assumption that the count rate is linear with neutron flux. Both count rate level and period are displayed for both channels on the front panel of the SRM chassis and on remote meters on

the reactor console. The count rate levels are recorded by one pen on each of 2 two-pen recorders, with the second pen of each recorder being reserved for recording of an IRM signal. With this arrangement, loss of one recorder will not cause loss of all recording for one range of neutron monitoring.

A six-decade scaler and timer is connected to each SRM channel to provide accurate pulse counting during low level tests and criticality measurements. Also, pulse outputs from the first 3 decades of each scaler may be switch selected from the reactor console to be routed to an audio amplifier with speakers in the control room and reactor cell to give audible indication of the count rate.

Three separate trip outputs are provided by each SRM. The first is a malfunction trip which activates upon loss of power, loss of high voltage output, or the switch not in the operate position. This trip is connected to the safety system.

The second trip is a fast period trip set at 15 seconds to warn the operator that he is approaching a 10-second period for which there is a trip in the IRM channels that initiates a safety system "SCRAM", and to provide an "Interlock" which prohibits further movement of reflector control drums toward a more reactive position. The third trip is a low count rate level trip which also provides an "Interlock" if either SRM count rate is below 2 counts/second, indicating a possible channel malfunction.

In order to protect the detectors, the high voltage to the detector is turned off automatically by a level trip in an IRM channel at some point above full scale on the SRM. The high voltage will be automatically turned on again as the reactor is shut down. Automatic bypass of the <2 counts/second trip is provided by switch positions on the WRM channels.

d.) Intermediate Range Monitoring (IRM) Signal Processing

The chambers for the three IRM channels are also located in wells around the reactor. Each IRM operates over a 7 decade range of input current from 10^{-11} amperes to approximately 10^{-3} amperes. The flux level is displayed on a log scale and the period on a hyperbolic scale of - 100, ∞ , 10 seconds. All level and period indications appear on both the reactor console and on the front panel of their respective chassis. A selection arrangement will be provided to allow selection of the IRM level or period to be recorded on the second pen of the previously mentioned (SRM) 2-pen recorders or the third (IRM) 2-pen recorder. Procedural requirements will dictate that no recorder shall obtain its two signals from the same IRM channel, nor shall both signals be of the same function.

Each IRM channel has four active trip outputs. The same trips from each of the three IRM channels are connected in a 2-out-of-3 logic to provide any safety action, thereby helping to eliminate spurious actions and allowing maintenance or calibration of one channel at a time during continuous reactor operation. One trip is adjusted to turn off the SRM high voltage as previously mentioned. The second trip, set at 15 seconds, provides an "Interlock" and an audible alarm to warn the operator he is approaching the third trip level at 10 seconds which will "Scram" the reactor. The fourth trip is the malfunction, which will also "Scram" the reactor if 2 out of the 3 IRM channels should fail. A separate level output is provided on each IRM for monitoring and recording by the data system.

e.) Wide Range Monitoring (WRM) Signal Processing

WRM signals are provided by gamma compensated ion chambers located in wells around the reactor. The WRM chassis are located in the nuclear panels in the Main Control Room with the range switch, remote meter, and recorder for each channel on the reactor console. The WRM operates over 18 linear ranges

from 1×10^{-12} to 3.16×10^{-3} amperes input current. Each decade is covered by two ranges, one being the approximate 10 multiple of the next so that the range is approximately even from one range to another and the trips set at "percent of full-scale" values appear the same on each range.

Three of the four WRM channels have inputs to the safety system and are wired for a 2-out-of-3 coincidence logic. The fourth WRM channel is used for automatic level control of the reactor. It is not connected to the safety system and will be discussed later. Each of the three WRMs connected to the safety system have a malfunction trip output and four level trip outputs. The level trips are set at 5, 10, 110, and 120 percent of full scale. The malfunction, 5 percent, and 120 percent trips all provide a "scram" action. The 10 percent trip provides only an audible alarm to warn the operator to switch range. The 110 percent trip provides both an audible alarm and the "Interlock" function to prevent control drum motion to a more reactive position.

Two 2-pen recorders on the reactor console record all four WRMs and a separate output is provided for monitoring and recording by the data system. Auxiliary contact decks on the range switches provide range information to the automatic data system and to the safety system for bypassing of certain safety functions.

f.) Nuclear Secondary Panels

The nuclear secondary panel contains the chassis, power supplies, recorders and the trip circuits associated with the nuclear instrumentation. It also contains the actuator drive circuitry, position indication power supply and the plant safety system. The equipment and installation shall be designed to meet the requirements of the "Proposed IEEE Criteria for Nuclear Power Plant Protection Systems", IEEE No. 279, dated August, 1968.

3. Automatic Level Control

The TREX reactor is expected to have such a small negative temperature coefficient that it will not be self-leveling and will therefore require the reactor operator to be continually adjusting the fine control drum in order to hold the reactor power level.

To alleviate this condition, a system will be designed to allow automatic level control once the desired operating level has been attained with manual control. The fourth WRM channel previously mentioned will be used to provide a measurement of the existing reactor power level.

With the fine control drum actuator being a bi-directionally stepping motor, two plans are presently envisioned to provide automatic control, depending on the "fineness" of control desired. The first, and coarsest, scheme would use the trip levels of the fourth WRM set at ± 5 and ± 10 percent of the desired operating point. Upon drifting to the ± 5 percent level, the trip level would actuate the proper relay logic to give the fine control actuator one (or a set number) of steps in the proper direction. The logic may also include time delay between steps if desired or necessary. Drifting to the ± 10 percent level would result in an alarm and return to manual control. No safety functions would be bypassed or hampered in any way. If closer control is required than can be obtained with the built-in adjustable trips, the WRM level will be compared to an adjustable reference voltage to derive an error signal which can be amplified and used to provide the signal to the above-described logic. Again, no safety functions would be bypassed or hampered.

4. Automatic TFE Current Control

Control of the TFE emitter temperatures is accomplished by varying the TFE pair output current in a manner proportional to reactor power. A power range monitor channel (PRM), reading 0 to 150 percent in one linear range, will be used to supply the signal to a power amplifier which in turn will furnish the control signal to all of the TFE power conditioning modules for control of each TFE pair output current.

The PRM input signal is from an uncompensated ion chamber located in a well adjacent to the reactor. The PRM chassis will be mounted in the TFE experiment panel.

The emitter temperature of all TFE pairs in the reactor is dependent on the control signal from this channel of instrumentation. A highly reliable, or redundant, system must be developed and will be connected into the reactor safety system to "Scram" the reactor in case of a malfunction or loss of control.

5. Actuator Control and Position Indication

The actuator control and position indication functions are both contained on the reactor control console. For normal reactor operations, it is desirable to have all control elements equally inserted to prevent undesirable flux distributions; therefore, an actuator sequencing logic will be incorporated. For extended operations, the fine control element will be placed on automatic control to maintain the reactor at the desired power level. With these features in mind, the actuator control logic will provide the following:

- . A selector switch on the reactor control console shall allow operation of one actuator at a time over its full range of travel for checkout of motion, position indication, and limit switch actions.
- . Selection of a particular control actuator shall apply "run down" power to all other control actuators and shall not permit operation of the selected actuator until all other control actuators are at their least reactive limit.
- . The selector switch position for operation of the fine control actuator shall be between the "Sequencer" position and the positions for selection of coarse control rods 1 through 12.
- . The fine control actuator shall have alarms at approximately 20% and 80% of full travel to warn the operator of such condition during automatic operation. Trip points at approximately 10% and 90% of full travel shall cause a "run down" if exceeded during automatic operation. It shall also be required that the fine control actuator be inserted to >10% of full travel before sequenced operation of the coarse control actuators can be

started. Future expansion of the logic can easily be accomplished to allow the 20% and 80% alarm limits to be used to apply insert or withdraw signals to the sequencer should such fully automatic action be desired.

A scram signal shall immediately release all control elements to be spring driven to their least reactive position and shall also automatically cause a simultaneous "run down" of all actuators.

Operation of any actuator to a more reactive position, by sequencer or by manual selection, is only possible if the safety circuit "Interlock" bus is clear.

Operation of the actuators, manually selected or sequenced, is accomplished with an "Insert-Neutral-Withdraw" lever switch which is spring loaded to the Neutral position from the "Insert" position.

After sequencer operation has begun, it shall be possible to stop insertion or withdrawal and select the fine control actuator for manual or automatic operation without causing a "rundown" of the coarse control actuators.

Automatic level control of the reactor will be accomplished through use of a ten-turn set point potentiometer which provides a setting corresponding to the desired level indication of the selected Power Range nuclear channel; an error meter to indicate the amount of error between the set point and the actual indication; and lighted push buttons for "Manual" or "Automatic" operation. Automatic operation shall be permissible with the actuator selector switch in the fine control position. Selection of the sequencer shall cause control to revert to "Manual". Selection of any coarse control actuator will cause a "run down" of all actuators. The error signal will be supplied to 4 adjustable trip circuits to provide automatic operation of the fine control actuator and safety action. One set of high-low limits will be adjusted to cause an insert or withdraw step of the fine control actuator. The second, wider set of high-low limits provide an alarm and cause the

control to revert to manual. The error signal must be between these limits before the control can be switched to automatic or control will immediately revert to manual.

Control element position is sensed through use of a 400 cps differential transformer. The output of the transformer is rectified to give a dc output of 0 to 10 volts corresponding to a control element rotation of 0 to 180°. The 0 to 10 volt dc output will power individual position indicating meters on the reactor control console and also provide the signal for readout of control element position by the digital data system. Limit lights shall also be provided next to the position indicator for each actuator. In addition to the position indicating meters, there will be two bi-directional pulse counters, one to count the number of insert pulses to the fine control actuator, and one to provide a count of the total number of insert pulses to the course control actuators. The total number of pulses to the coarse control actuators is an indication of the total reactivity inserted by the 11 course control actuators. Once a critical position (total count) has been established for the coarse control actuators, a coincidence circuit will be set at a predetermined count above the critical count. If the coarse control actuators are inserted to the point where total count coincidence occurs, the "Interlock" bus will be opened, preventing further reactivity insertion. Adjustment of the coincidence count shall be under controlled access.

6. Reactor Safety System

The reactor safety system has two primary functions. First, to permit motion of the control elements to a more reactive position only if the required nuclear and plant systems are within their established limits and operational procedures are followed. Secondly, to scram all control elements when an unsafe condition is approached or critical plant systems fail. Tables 33 and 34 is a proposed list of parameters for safety system inputs, and the presently envisioned high or low limits required to cause safety action. The table indicates whether the action is an "Interlock" to prevent reactivity increase by actuator motion or a "Scram" to immediately release all control elements to be spring returned to their least reactive position.

TABLE 33

TREX SAFETY SYSTEM ACTIONS

SYSTEM	SIGNAL	LOGIC	ACTIONS			
			Alarm Only	Interlock	Rundown	Scram
NUCLEAR	SRM MALFUNCTION	1/2		X		
	SRM LOW COUNT RATE	1/2		X		
	SRM SHORT PERIOD	1/2		X		
	IRM MALFUNCTION	2/3				X
	IRM SHORT PERIOD	2/3		X		
	IRM SHORT PERIOD	2/3				X
	WRM MALFUNCTION	2/3				X
	WRM LOW LEVEL	2/3				X
	WRM LOW LEVEL	1/3	X			
	WRM HIGH LEVEL	2/3		X		
	WRM HIGH LEVEL	2/3				X
ELECTRIC POWER	LOSS OF POWER TO: REFLECTOR DRIVE PRIMARY COOLANT PUMP	1/1				X
	SECONDARY COOLANT PUMP	1/1				X
	AIR BLAST COOLER FANS	1/1				X
	BUILDING	1/1				X
	CORE OUTLET TEMPERATURE HIGH	2/6				X
PRIMARY COOLANT	LOW	2/6			X	
	LOW PRIMARY FLOW	1/1				X
	LOW VENT TANK LEVEL	1/1				X
SECONDARY COOLANT	LOW VENT TANK LEVEL	1/1				X
	LOW FLOW	1/1				X
	HEAT DUMP EXIT TEMP: HIGH	1/2				X
	LOW	1/2			X	
TFE	LOW CESIUM TEMPERATURE	1/200			X	
	LOW OUTPUT CURRENT	1/200			X	
	LOW OUTPUT VOLTAGE	1/200	X			

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TABLE 34
TREX SAFETY SYSTEM ACTIONS

System	Signal	Logic	Actions			
			Alarm Only	Interlock	Rundown	Scram
Reactor	High Reflector Temperature	2/20			X	
Vessel	High Vessel Temperature	2/12			X	
	NaK Between Double Wall	1/2				X
Power Conditioning	High Temperature	1/100			X	

A third function of the safety system is to provide a "run down" or simultaneous operation of all actuators to their least reactive position. This function serves as a backup to the scram of the control elements and as an operational feature to prevent operation of more than one actuator at a time during reactor systems checkout. If desired, the "run down" function or a fourth function of sequential reactivity withdrawal can be easily included in the safety circuit to be activated by selected signals.

The reactor safety system is powered from the same + and - 26.5 volt battery that supplies power to the nuclear instrumentation. It is therefore functional at all times when nuclear instrumentation is available.

The complete reactor safety system from detectors or transducers through final control elements shall be designed and constructed to meet the latest requirements of the "Proposed IEEE Criteria for Nuclear Power Plant Protection Systems", IEEE No. 279, dated August 1968.

7. TFE Voltage, Current, and Cesium Temperature

The TFE voltage and current signals will be brought to the TFE Experiment Panel through fused, twisted and shielded pairs of wires from the reactor and TFE Power Conditioning Modules. Both voltage and current will be permanently wired into the digital data system for printout, monitoring, and alarm. Each current and voltage signal shall also be wired to a patch panel to allow selection of three voltages and three currents for trend type recording by three dual-pen continuous strip chart recorders.

Each TFE has two chromel-alumel thermocouples in the cesium reservoir. All of these thermocouples shall be brought through 150°F reference junctions and signal conditioning to the digital data system for readout. Each cesium reservoir has its own automatic temperature controller and provisions will be made for manual selection of the thermocouple used for control. High and low limit alarms are an integral part of each controller and will be used to provide alarm signals.

The TFE Experiment Panels contain the 150 individual cesium reservoir temperature controllers, three dual-pen voltage and current strip chart recorders, the 24 point strip chart recorder, the associated patch panel and the TFE automatic current circuitry.

8. Power Conditioning Modules

Figure 74 is a schematic of the circuitry between the TFE pairs and the power conditioning modules. The circuitry of Figure 74 is required to allow the performance of I-V diagnostics on the operating TFEs. Unlimited personnel access to the power conditioning modules is required during reactor operation. Yet the modules must be as close as possible to the reactor to prevent excessive voltage drop in the low voltage-high current leads. The current measuring shunt is at the power conditioning module. The TFE direct potential voltage leads are also available at the module.

Three switches are shown in Figure 74. One for the insertion of a dc power supply to establish the dc operating point at the desired current level for a degraded TFE. The second removes the TFE from the power conditioning module and connects it to the I-V diagnostic equipment. The third provides a means of shorting a TFE in the event of an arc or provides a by-pass path for the companion TFE during I-V diagnostics or an open circuit event.

The only instrumentation at the power conditioning modules is the TFE current shunt. Jacks are available on each module for monitoring of each TFE voltage and current by means of plug-in voltmeters and for use in I-V diagnostics. The I-V diagnostic instrumentation consists of several portable units. Each contains an ac sweep transformer, a variable resistive load and sweep triggering and recording equipment. I-V sweep recording may be done with an oscilloscope and camera or with digital sample and hold equipment with plotter or data system readout.

FIGURE 74. POWER REGULATOR SCHEMATIC

9. Coolant and Gas System Instrumentation

a.) Primary Coolant Loop

The loop external to the reactor consists of a vent tank, the primary to secondary heat exchanger, the primary EM pump, an EM flowmeter and an electric heater for startup and control of reactor inlet temperature. Primary loop flow is controlled through manual control of the primary EM pump voltage from the graphic panel in the control room. Power to the electric heater controlling the reactor coolant inlet temperature is controlled from a recorder-controller located on the main graphic panel.

Instrumentation for the primary coolant loop will measure the coolant flow rate, NaK level in the vent tank, and coolant temperature at various points in the loop (primarily at inlet and outlet of the reactor and primary to secondary heat exchanger). Primary loop parameters and alarm conditions will be displayed on a graphic panel visible to the reactor operator.

b.) Secondary Coolant Loop

The secondary coolant loop contains one automatic control associated with the reactor coolant systems. The NaK temperature at the exit of the NaK to air heat exchanger is sensed and used to control the air blast cooler louvers which control the air flow through the heat exchanger. With both primary and secondary NaK loops at full flow and the NaK heater in the primary loop at full power, the system is maintained at the desired startup temperature by having the louvers almost fully closed. As the reactor is brought to power, the louvers open proportionately and control the loop temperature.

Instrumentation for the secondary loop will measure the expansion tank level and pressure, the NaK flow rate, the temperature at the inlet and outlet of the primary to secondary and secondary to air heat exchangers and the inlet and outlet pressure and temperature of the air for the NaK to air heat exchanger. Secondary loop parameters and alarm conditions will be displayed on a graphic panel visible to the reactor operator.

c.) Auxiliary Cooling Loop (Primary)

The auxiliary cooling loop provides emergency after-heat cooling of the shutdown reactor in case of a main coolant loop failure, but is also on line and functioning during normal reactor operations. Instrumentation for the auxiliary cooling loop will measure the vent tank level and pressure, the NaK flow rate, the temperature at the inlet and outlet of the reactor and the temperature of the EM pump windings and cell. Auxiliary cooling loop parameters and alarm conditions will be displayed on a graphic panel visible to the reactor operator.

d.) Auxiliary Cooling Loop (Secondary)

The secondary auxiliary cooling loop operates on convective flow of NaK from the secondary side of the intermediate heat exchanger to the auxiliary air cooler, which is also convection cooled. The instrumentation for the secondary of the auxiliary cooling loop measures the level and pressure of the expansion tank, the outlet temperatures of the intermediate heat exchanger and auxiliary air cooler and the NaK flow rate. The temperature of the NaK at the exit of the auxiliary air cooler is used to automatically control the dampers and electric heater in the air cooler stack. The loop parameters and alarm conditions will be displayed on a graphic panel visible to the reactor operator.

e.) Primary Loop Service

The primary loop service system provides for both initial and continuous cleanup of the primary coolant loop NaK by both hot and cold traps. It also provides for monitoring of the NaK purity by a plugging indicator and an oxygen monitor and samples of NaK for laboratory analysis. The hot and cold traps and plugging detector require automatic temperature controls. A primary service system graphic panel, displaying important parameters and alarm conditions will be visible by the reactor operator.

f.) Secondary Loop Service

The secondary loop service system is identical to the primary loop service system except that no hot trap is provided for NaK purification. The graphic panel for the secondary loop service system will be visible to the reactor operator.

10. Coolant and Service Loop Measurements

NaK level measurements are required in the reactor vessel, the primary and secondary vent and expansion tanks of both the main and auxiliary coolant loops and the primary and secondary drain tanks. Continuous level measurements are provided over fixed ranges by top mounted electrical resistance type detectors known as "J" probes, operated from very stable dc power supplies. Discrete level probes, known as "I" probes, will be used to indicate that the level has exceeded the range of the "J" probe where necessary.

Level indication for the "J" probe is by a reverse acting vertical scale meter which measures the voltage drop across the active section of the probe. The voltage drop will also be used as a signal to voltage level trips for alarm and safety actions. Redundant alarm and safety action is provided by the discrete level "I" probes which give a step change in voltage when the tip is immersed in NaK, shorting out the skin section of the probe.

All temperature measurements will be by stainless steel sheathed chromel-alumel thermocouples welded and strapped to the outside of the insulated piping or vessel or brazed into the cesium reservoir of the TFE. The stainless steel sheath will extend out of the high temperature zone to a point where it is convenient and safe to change to unsheathed thermocouple extension wire. All thermocouples will have 150°F controlled reference junction temperature. All thermocouples will also have individual signal conditioning to provide a 0 to 5 volt output.

Pressure measurements are required for two different fluids; liquid NaK and helium. Pressure measurements for NaK will be made with a eutectic backed stainless steel diaphragm, capillary tube and helium leak tight bourdon tube. The measurement bridge shall have a low

impedance output of at least 0 to 5 volts dc.

NaK flow will be measured using electromagnetic flowmeters: a rectangular cross section stainless steel tube mounted between the poles of a permanent magnet. The flowing conductor, NaK, passing through the magnetic field generates a voltage at right angles to both the magnetic field and moving conductor proportional to the magnetic field strength and NaK flow rate. Multiple contacts will be attached to the tube to measure the voltage generated. The flowmeter temperature will be measured to provide temperature corrections to the flowmeter calibration. Signal conditioning will be used to provide a minimum of 0 to 5 volt dc signal output.

Helium flow measurements will be made by rotometer tubes with differential transformer electrical output to provide maximum integrity of the NaK cover gas systems. Both analog and digital outputs will be available for flow indication. Other gas flow measurements will be made by rotometer tubes with both visual and differential transformer output.

11. Radiation Monitoring

Four types of radiation monitoring are used in various areas of the facility: a stack monitor for measuring the activity of gases being released to the atmosphere from the helium cover gas and nitrogen ventilation systems; continuous air monitors for monitoring airborne particulate in critical areas; current type ionization chambers for monitoring gamma radiation in high radiation level areas; and GM counters for general area coverage. Indication and alarm for all monitoring channels is contained in the radiation monitoring panels of the Main Control Room. Two multipoint recorders provide a recording of all radiation levels.

12. Data System

The automatic data system will serve two major purposes; provide a permanent record of all pertinent experiment and plant measurements in the form of a typewritten record and computer compatible punched cards; and provide monitoring and alarm functions for the above experiment and plant measurements and other measurements which may not

necessarily be useful in reactor and plant performance analysis, but will aid in overall plant operation. To provide flexibility in programming of the nearly 2000 expected inputs and the output functions of recording, punching, monitoring and alarming, a small digital computer of the "Executive" type would be utilized. Such a computer will also permit outputs in engineering units, minor program computations, logic to be applied to alarm functions, and easy access to changes in programming and alarm limits.

The computer and data system, while monitoring and alarming both critical and non-critical parameters, will perform no direct reactor or plant safety functions, it is not connected to the reactor safety system. Many of the safety system functions require a logic, such as 2-out-of-3, before providing a safety action. Monitoring of all of these inputs by the data system will provide an alarm to warn the reactor operator of an out-of-limit condition on each input, thereby allowing corrective action to be taken, if possible, before a safety system action is demanded. Once plant operating experience has been attained, proper programming would also allow the computer and data system to provide the reactor operator with advisory information.

Under normal operation, the data system continuously monitors inputs for out-of-limit conditions according to one program in the computer. Upon reaching an out-of-limit condition, the system will provide a print-out of the time, data channel, and level on the alarm typewriter visible to both the reactor and data system operators. Priority interrupts may be used in the monitoring program to allow critical functions to be monitored at more frequent intervals than non-critical functions, or to allow computation and updating of visual displays for operator information.

At preset time intervals, or upon operator demand, a data recording program will be initiated which will result in a typewritten and punched card output of current reactor, plant, and experiment variables and computed data according to a second program stored in the computer. Values which are out of limits during this data printout will be indicated on the typewritten record in a conspicuous manner. The punched card output will be used for later computations and analysis by a larger general purpose computer capable of much more complex programs than those available to the small computer which controls the data system.

Upon demand, the computer and data system shall be capable of providing a typewritten record of all input variables used for data recording and monitoring to allow data system trouble shooting and cross calibration of input channels.

The monitoring and alarm program for the computer and data system shall allow scanning of inputs at a rate not less than 100 points per second. Such a scanning speed generally places limitations on the data system with regard to low level signals such as thermocouple outputs. Data system input point selection and the analog to digital conversion for low level signals are both involved. To alleviate this problem, all low level signals will have individual signal conditioning resulting in an output level of at least 0 to 5 volts. Such individual signal conditioning will also allow multiple use of the output for indicating meters, level alarms, recorders, controllers, and data system use.

A simplified block diagram of the proposed TREX data system which would allow the above operations is shown in Figure 75. The computer and operators console are the heart of the system with the computer controlling the input and output switching, monitoring logic, program sequencing, and computations according to the programs stored in its memory. The operators console allows the operator to give the computer specific commands

and to make minor changes in the stored programs. Major program changes, or complete new programs, are entered into the computer through either a punched card or a punched tape reader, allowing pre-checking of the program and a short change-over time. Provision is shown on Figure 75 for inclusion of digital inputs to the system. Such inputs may come from specific instruments with digital outputs, but will be more commonly used for manual input of data which is not readily converted to electrical signals by other means: Test No.; Core configuration; TFE numbers; etc. As previously mentioned, all analog signals will be conditioned to high levels for high speed scanning by the multiplexer for input to the Analog and Digital Converter and thence to the computer for scaling, computation, comparison and output to the selected output devices.

13. Electrical Power

The existing primary electrical and facility electrical systems of the FRCTF would be used. The electrical substation has two 1000 kva transformers which take their output from the Los Alamos ring bus. Primary power for the Los Alamos ring bus is supplied by two 115 kV transmission lines from a ring bus in Albuquerque. The Los Alamos 13.2 kV ring bus furnishes power to two 20,000 kVa transformers with the systems interconnected to not interrupt power to either 20,000 kVA transformer. Should both 115 kV or both 13.2kV lines fail, two emergency generators supply 20,000 kVA from the Los Alamos steam turbine generators.

Should all primary power fail, two emergency generators are available in the facility. An emergency power bus is supplied by a 20 kW generator which is normally driven by a synchronous electric motor. With failure of the primary power, generator drive is transferred to an internal combustion engine which is always running during reactor operation. The emergency bus supplies power to the data system and selected instrumentation loads. A 75 kW emergency motor-generator unit is

FIGURE 75. TREX Data System

available for other less critical building loads such as ventilation systems, small motors, and limited lighting. Upon loss of primary power, the internal combustion engine is automatically started, and when up to speed, the emergency loads are transferred to it.

The nuclear instrumentation and safety systems are powered by an unregulated source of dc power typical of a lead acid storage battery. Plus and minus 26.5 volts is selected because this is the normal terminal voltage of a 12-cell battery on floating charge. The plus and minus 26.5 volts is dropped to plus and minus 20 volts by a preregulator for operation of trip circuits and is further lowered to plus and minus 15 volts by a voltage regulator and is used as the basic regulated supply to all nuclear instrumentation. This system is standard to all GE nuclear power reactor systems. The batteries are on constant charge during normal operations and allow several hours of uninterrupted nuclear channel operation when all other sources of power fail.

J. COOLANT AND GAS SYSTEMS

1. NaK Coolant and Services Systems

a. Purpose

The purpose of the NaK coolant system is to transfer heat from the reactor in a safe and reliable manner. Consistent with this objective, a general guiding principle in the design is that the TREX coolant systems should have such high reliability that the probability of gross fuel melting leading to release of fission products outside the coolant system is negligible.

b. Functional Requirements

The NaK coolant system consists of two independent heat transfer circuits designed so that either circuit can remove decay heat following full power operation even in the event that there is a malfunction in the other loop (such malfunctions include the possibility of a pipe break) and/or a NaK fire. In the event of loss of electrical power each coolant circuit is arranged so as to promote natural circulation and provide removal of decay heat.

Each coolant circuit consists of two loops in series (Fig. 76); a primary loop which contains the radioactive NaK and a secondary loop which contains the nonradioactive NaK. (Shown in Figure 77 .) This arrangement minimizes the risk of release of radioactivity to the environs since the entire primary NaK system is contained within leak-tight, steel-lined concrete cells. A nitrogen atmosphere with less than 5% oxygen surrounds the primary coolant system so as to minimize the possibility of a NaK fire in event of a primary coolant loop leak.

Continuous NaK makeup is provided to the reactor from the primary drain tank with automatic high level overflow from the reactor to the drain tank. Both overflow and makeup capacities are sized to provide level control within specified limits even under potential accident conditions involving pipe or heat exchanger tube breaks. To minimize thermal shocks to the system due to loss of pumping power, the two sets of windings of the EM pump in the main primary coolant loop are driven by two independent motor generator sets with flywheels.

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FIGURE 76. TREX NaK COOLANT SCHEMATIC

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Figure 77. Coolant System
Flow Diagram

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The entire primary and auxiliary coolant systems are elevated above the reactor vessel main coolant nozzles to prevent siphoning in event of a leak in the auxiliary primary loop. All important parameters of the NaK systems are monitored and those vital to reactor safety are connected to the automatic scram system. The coolant system is designed to withstand temperature, pressure and radiation conditions corresponding to a future 3 MWt operation.

The NaK service system is designed to perform the following tasks: (See Figure 78).

- . Safely and reliably charge, store, fill and drain NaK in the two (main and auxiliary), primary and secondary coolant loops.
- . Initially purify and maintain NaK purity so that the plugging temperature in each NaK loop is 50°F below the lowest measured temperature in that particular loop. For most of the normal reactor operating temperatures this results in plugging temperatures of less than 400°F.

c. NaK Coolant System Operation

The NaK coolant system consists of two independent heat transfer circuits, a main and an auxiliary coolant system.

The main coolant system is capable of removing up to 3 MW of heat while the auxiliary system is designed for a heat removal capacity of 0.073 MW. The auxiliary system serves as an operating standby facility to remove reactor decay heat in the event the main system is out of service. In addition to the main and auxiliary coolant system, there is a pump-around loop which maintains the reactor NaK level and provides NaK to the reactor vessel in the event of an emergency. A flow diagram for the coolant system is shown in Figure 77 and the physical arrangement of the piping and equipment is shown on the Reactor building layout Figure 79.

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FIGURE 78. TREX NaK SERVICE SCHEMATIC

Figure 79 , Primary Loop Piping Arrangement to Coolant

All exterior NaK piping and equipment are traced with tubular electric resistance heaters underneath the insulation. Heater circuits are controlled manually from temperatures read from thermocouples mounted along the piping and equipment. All components are insulated to minimize heat loss during normal operation and to permit the systems to be preheated to approximately 700°F for startup. All NaK lines are sloped a minimum of 1/8-inch per foot for ease in venting and draining.

1. Main NaK Coolant System

The main NaK coolant system consists of a radioactive primary loop which transfers up to 3 MW of thermal energy from the reactor core to the main intermediate heat exchanger (IHX) and a non-radioactive secondary loop which transfers the heat from the main IHX through the main air blast cooler (ABC) to the atmosphere.

a.) Main Primary System

The main primary coolant loop is located in a concrete-shielded nitrogen atmosphere in the east pit of the test building. The nitrogen atmosphere, containing less than 5% oxygen, is maintained in the cell at pressures slightly below the surrounding outside atmosphere to prevent out-leakage.

The main primary NaK coolant enters the reactor vessel through three 3" inlet nozzles at (3 MW design conditions) 980°F and a flow rate of 580 gpm. The NaK at 1160°F leaves the reactor vessel through three 3" outlet nozzles and enters the upper shell of the main IHX, where 3 MW of energy is transferred to the secondary NaK. The 980°F NaK leaves the main IHX and flows through a 4-inch line to the main primary pump, which returns the NaK through a heater to the reactor. A temporary strainer, is located in the reactor inlet line to prevent construction debris in the system from entering the reactor. This strainer is removed from the system after full flow NaK circulation

but prior to reactor operation. The main IHX is located above the reactor core center to provide a minimum of 15 gpm emergency flow by natural convection. The main primary pump is located below the main IHX to assure a positive head on the pump suction.

The NaK temperature into and out of the reactor vessel is measured by resistance-type elements to obtain a plant heat balance. Thermocouples are used to determine the temperature into and out of the main IHX. These temperatures are recorded on a multi-point recorder in the control room and are used to monitor the performance of the IHX. The reactor outlet coolant temperature is also recorded on a high temperature scale recorder (up to 1500°F) to provide a record of high temperatures in the event of a system malfunction or an accident.

The NaK pressure differential through the reactor vessel is measured across the main IHX inlet, the main primary pump inlet and at the reactor vessel inlet. These pressure measurements are used to monitor the system pressure losses and the main primary pump performance. The main primary NaK flow is measured by means of a magnetic flowmeter located on the pump discharge.

The main intermediate heat exchanger (IHX) is a shell and a tube unit with the primary NaK on the shell side and secondary NaK on the tube side. The primary flow is single pass, counter flow. NaK enters at the top and leaves at the bottom of the exchanger shell. The secondary NaK enters and leaves the exchanger through 4-inch nozzles located at the top and bottom heads of the unit. The secondary NaK flows through a tube coil within the exchanger. The exchanger is mounted vertically to prevent shell side stratification and promote natural circulation. Flow through the unit is such that the hot secondary NaK leaves at the upper nozzles to prevent stratification and promote natural circulation.

A vent tank is connected to the upper exchanger shell to provide space for the collection and venting of gas. The tank is equipped with a 2-inch diameter vent nozzle at the top and a level probe in the side. The level probe sounds an alarm in the control room when the dome is about half full of gas and the NaK surface falls below the probe location.

The main primary pump is a helical induction, electromagnetic, nitrogen cooled pump which is designed to deliver 580 gpm of 980^oF NaK with a 30 psi developed head. In addition, the pump is designed to pump 100 gpm at 1300^oF to meet the isothermal test requirements.

Six chromel-alumel thermocouples are located in the discharge cooling gas stream, two are located on the pump duct, and two are provided on the transition pieces to monitor temperatures during operation. Of the three thermocouples on each winding, one is recorded, the second is used to trip the pump off and alarm if the temperature exceeds 350^oF and the third is a spare. Of the two thermocouples on the pump duct, one is used to shut off the nitrogen coolant on low duct temperature (below 350^oF) and shut off the main primary pump on high temperature set point (<1300^oF) and the other is used as a spare. One of the thermocouples in the cooling nitrogen discharge is used to monitor gas outlet temperature; the other is used as a spare.

The main primary pump has power supplied to each of its two stator windings from separate synchronous motor-generator sets. This redundancy is necessary in order to maintain coolant flow in the event of loss of the electrical power supply to one winding due to a component failure. In the event of loss of plant power, each motor-generator set is equipped with a flywheel which will allow a gradual reduction in flow from 580 gpm to 100 gpm in about two minutes.

The main primary loop has two 1-inch sodium service system connections. Both connections are in the 4-inch pump suction

line (the system low point) and are used to drain the system and to remove and return NaK from the loop for purification system.

b.) Main Secondary System

The main secondary NaK loop is located principally outside the shielded area in an air atmosphere in the non-radioactive NaK room. The main air blast cooler (ABC) is located outside the building on the heat dump equipment pad.

The main secondary NaK flows from the main secondary pump in the non-radioactive sodium room through a 4-inch line to the tube side of the main IHX in the reactor pit. The NaK rate is about 570 gpm at 880°F entering the IHX and the NaK is heated to 1060°F due to the transfer of 3 MW energy in the heat exchanger. The NaK flows from the IHX in a 4-inch line, up through the IHX equipment rack to the operating floor and then through the non-radioactive NaK room to the air blast heat exchanger (ABC). The ABC reduces the temperature to 800°F. The NaK then flows from the ABC through the main secondary pump to a 4-inch magnetic flowmeter. The tube bundle of the main ABC is elevated above the IHX to provide a minimum of 15 gpm emergency flow by natural convection. A small stream of NaK, about 11 gpm, bypasses the ABC and flows through a 1-inch line to the main secondary NaK expansion tank to provide a positive means of disengaging any gas from the system. The main NaK expansion tank is the high point in the main secondary system. The expansion tank provides gas space for expansion and the pressure of the He cover gas controls the secondary NaK system pressure. The secondary NaK pump is located about 15 ft. below the ABC tube bundle to provide positive head at the pump suction.

Instrumentation for temperature, pressure and flow are provided for process control of the system, for monitoring equipment performance and for safety trips. The secondary NaK temperatures to and from the main IHX and to the main

ABC are measured by thermocouples and recorded on a multi-point recorder in the control room to provide performance data for the heat exchangers. The NaK temperature at the ABC outlet is recorded in the control room and also controls the ABC blower speed to hold the outlet temperature at a given setting. The high temperature alarm is set at 950°F.

The pressure in the main expansion tank is monitored on the He supply to the vessel. The He supply line is also equipped with a pressure safety trip to scram the reactor in the event the expansion tank cover gas drops below 25 psig. The cover gas pressure is maintained above 25 psig to assure a positive pressure on the main secondary pump and to assure that the secondary NaK pressure will be greater than that of the primary NaK at the IHX so that any leakage will be into the primary system.

The main secondary NaK flow is measured by means of the magnetic flowmeter in the 4-inch pump outlet line. The flowmeter reading is indicated and recorded in the control room. A low flow alarm with variable setting is used during steady state operation to indicate loss of flow. A signal from the magnetic flowmeter also controls the main secondary pump to maintain the NaK flow constant at a given setting. In addition to the flow indicator, recorder, alarm and controller, the magnetic flowmeter has three independent flow signals which are part of the safety system to scram the reactor in the event of low flow rate (less than 20% of the set point flow).

The main air blast cooler is a forced-air, finned-tube, heat exchanger designed to transfer up to 3 MW of energy from the secondary NaK to the ambient air. The NaK nozzles are located so that the hot secondary NaK enters the top of the bundle to prevent stratification and promote natural circulation.

The NaK outlet temperature from the main air-blast cooler is controlled at a given setting by varying the blower speed through a variable speed coupling. The speed control can regulate the unit from 10 to 100 percent of design duty. The blower speed is indicated in the control room. The blower is also equipped with remote, manually operated, radial inlet vanes for control in the lower range.

The main air-blast cooler tube bundle may be isolated, to maintain NaK temperature during down-time, with pneumatic, remote operated, insulated doors located above and below the tube bundle. Each set of doors, one above and one below the bundle, may be operated individually and is capable of being opened or closed within 30 seconds. The tube bundle housing is insulated and heaters are mounted below the heat exchanger tube bundle to preheat the unit during startup or to maintain temperatures during stand-by operations. A smoke detector is located on the air discharge of the unit to monitor NaK leaks.

The main secondary pump is a helical induction electromagnetic, air cooled, pump designed to deliver 570 gpm with a 30 psi developed head. This pump is identical to the main primary pump, although the normal operating flow rate and the temperature are not as high as required in the primary system.

Six chromel-alumel thermocouples are located in the winding end turns (three on each winding), two are located in the discharge cooling gas stream, two are located on the pump duct, and two are provided on the transition pieces to monitor temperature during operation. Of the three thermocouples on each winding, one is recorded, the second is used to trip the pump off, scram the reactor and alarm if the temperature exceeds 350°F and the third is a spare. Of the two thermocouples on the pump duct, one is used to shut off the cooling air blower on low temperature (below 350°F) and shut-off the main primary pump on high temperature (above 1050°F) and the other is used as a spare.

The power supply and control system for the main secondary pump consists of an auto-transformer-induction voltage regulator combination. The pump voltage is remotely set from the control room and the flow can be held constant on the magnetic flowmeter.

A smoke detector is located in the pump coolant gas discharge line to monitor for NaK leaks.

The main secondary NaK expansion tank is a 62 gallon vessel which contains the thermally expanded NaK volume from the main secondary loop. The tank is the main secondary loop high point and any entrained gas in the NaK is collected in the vapor space. The secondary expansion tank is equipped with a level J-probe which indicates and records the NaK level in the control room and trips high and low level alarms.

The main secondary loop has two 1-inch sodium service connections. Both secondary connections are in the 4-inch pump suction line (the system low point) and are used to drain the system and to remove NaK from the loop for purification.

The operating pressure in the main secondary system is determined by the pressure of the main secondary expansion tank which must be above 25 psig to assure adequate suction pressure on the main secondary pump and to assure that the secondary loop pressure is always maintained above that of the primary at the main IHX. A safety trip on the argon inlet line to the expansion tank, set at 25 psig, will scram the reactor in the event of low cover gas pressure.

2. Auxiliary NaK Coolant System

The auxiliary NaK coolant system consists of a radioactive primary loop which transfers up to 256, 000 Btu/hr of thermal energy from the reactor core to the auxiliary intermediate heat exchanger and a non-radioactive secondary loop which transfers the heat from the IHX through the auxiliary air blast cooler to the atmosphere.

The auxiliary coolant system is capable of removing the above heat generation using only natural circulation of NaK in the auxiliary system and air through the auxiliary air-blast cooler with a reactor outlet temperature of 1160°F or less.

a.) Auxiliary Primary System

The primary loop is located within a concrete shielded, nitrogen atmosphere cell (radioactive NaK room) in the test building. The nitrogen atmosphere, containing less than five percent oxygen, is maintained in the cell at pressures slightly below the surrounding outside atmosphere to prevent out-leakage of contaminants. The auxiliary NaK coolant enters the reactor vessel through a 2 inch nozzle at the 3 MW design temperature of 980°F and a flow rate of about 15 gpm. The auxiliary inlet and outlet nozzles are dipped about 2 feet below the main NaK nozzles to assure system operation in the event of a rupture in the main primary piping. There is a check valve in the auxiliary inlet line external to the reactor vessel to prevent backflow of NaK in the event the auxiliary primary pump fails. The NaK leaves the reactor vessel through a 2 inch nozzle and a 2 inch line and enters the upper shell of the auxiliary intermediate heat exchanger where energy is transferred to the secondary NaK. The 980°F NaK leaves the auxiliary intermediate heat exchanger and flows through the auxiliary primary pump and flow meter, which returns the NaK to the reactor through a 2 inch line. The auxiliary loop piping and equipment, external to the reactor vessel, are located above the reactor main coolant nozzles to prevent siphoning action which could lower the NaK below the main coolant nozzles in the event of a pipe break in the auxiliary coolant system. In the event that main system coolant flow is lost the flow rate in the auxiliary primary loop will increase because of the reduction in back pressure (reactor core pressure drop) on the auxiliary primary pump.

Instrumentation for temperature, pressure and flow are provided for process control of the system, for monitoring equipment performance and for safety trips. The NaK

temperature into and out of the reactor vessel is measured by resistance-type temperature elements to obtain a plant heat balance. These temperatures are recorded and operate high temperature alarms.

The NaK pressure is measured at the auxiliary vent tank. The auxiliary primary NaK flow is measured by means of a magnetic flowmeter, located in the 2-inch pump outlet line. The magnetic flowmeter reading is indicated and recorded in the control room. A low flow alarm set at 10.0 gpm indicates loss of flow in the system. The flowmeter is provided with a spare connecting lead.

The auxiliary intermediate exchanger is a shell and tube unit with the primary NaK on the shell side and secondary NaK on the tube side. The primary flow is single pass, counter flow; NaK enters at the top and leaves at the bottom through 2 inch nozzles. The secondary NaK enters and leaves the exchanger through 2 inch nozzles located at the top and bottom heads of the unit. The secondary NaK flows through a single-pass tube within the exchanger. The exchanger is mounted vertically to prevent shell side stratification and provide drainage. Flow through the unit is such that the hot primary NaK enters and the hot secondary NaK leaves at the upper nozzles to prevent stratification and promote natural circulation.

A 12 inch diameter dome is connected to the upper exchanger shell to provide space for the collection and venting of gas. The dome is equipped with a 2-inch diameter vent nozzle at the top and a level probe in the side. The level probe sounds an alarm in the control room when the dome is about half full of gas.

The auxiliary primary pump is an electromagnetic pump designed to deliver up to 15 gpm of NaK at temperatures up to 1050°F for reactor shutdown cooling. This pump is in the auxiliary system primary loop to provide continuous flow in this loop against the reactor vessel ΔT and in the direction

of natural convective flow. This assures immediate convective cooling upon main loop pump failure or electrical power loss. Two chromel-alumel thermocouples are located in the winding end turns. One is recorded in the control room and is used to shut off the pump and alarm if the temperature exceeds 350°F.

b.) Auxiliary Secondary System

The auxiliary secondary NaK loop is located principally inside the non-radioactive NaK room in an air atmosphere. The auxiliary air cooler is located outside the building. The auxiliary secondary NaK flows through a 2-inch line to the tube side of the auxiliary intermediate heat exchanger in the radioactive NaK room. The NaK rate is about 15 gpm at 880°F entering the IHX and the NaK is heated to temperatures up to 1060°F due to the transfer of 256,000 Btu/hr of energy in the heat exchanger. The NaK flows from the IHX in a 2 inch line which passes through the operations building and outdoors to the auxiliary air cooler where the temperature is reduced to 880°F. The NaK then flows from the air cooler through a 2 inch magnetic flowmeter to the auxiliary IHX. The auxiliary secondary NaK expansion tank provides a positive means of disengaging any gas from the system. The auxiliary expansion tank is the high point in the secondary system. The expansion tank provides helium gas space for NaK expansion and the pressure of the He cover gas controls the secondary NaK system pressure.

Instrumentation for temperature, pressure and flow are provided for process control of the system, for monitoring equipment performance and for safety trips. The secondary NaK temperatures to and from the auxiliary air cooler are measured by thermocouples and recorded on a multi-point recorder in the control room to provide performance data for the heat exchangers. The NaK temperatures at the air cooler outlet controls the air cooler air dampers to hold the outlet temperature at a given setting. The high temperature alarm is set at 750°F.

The pressure in the auxiliary expansion tank is monitored at the He supply to the vessel. The secondary NaK expansion tank cover gas is maintained above 25 psig to assure that the secondary NaK pressure will be greater than that of the primary NaK at the IHX so that any leakage will be into the primary system.

The auxiliary secondary NaK flow is measured by means of the magnetic flowmeter. The flowmeter reading is indicated and recorded in the control room. A low flow alarm, set at 10.0 gpm, indicates loss of flow.

The auxiliary air-blast cooler is a natural circulation finned tube heat exchanger designed to transfer up to 256,000 Btu/hr of energy from the secondary NaK to the ambient air. The NaK nozzles are located so that the hot secondary NaK enters the top of the bundles to prevent stratification and promote natural circulation.

The NaK outlet temperature from the auxiliary air cooler is controlled at a given setting by varying the air-damper at the inlet to the tube bundle. The dampers can regulate the unit from 10 to 100 percent of design duty. The cooling air is supplied by natural convection.

The auxiliary air cooler tube bundle may be isolated to maintain NaK temperature during downtime using pneumatic, remote operated, insulated doors located above and below the tube bundle. The tube bundle housing is insulated and heaters are mounted below the heat exchanger tube bundle to preheat the unit during startup or to maintain temperatures during standby operations. A smoke detector is located on the air discharge of the unit to monitor for NaK leaks.

The auxiliary secondary NaK expansion tank is a 12 gallon vessel which contains the thermally expanded NaK volume. The expansion tank is located at the auxiliary secondary loop high point and any entrained gas in the NaK is collected in the vapor space. The secondary expansion tank is equipped with a

level J-probe which indicates and records the NaK level in the control room and trips high and low level alarms. In addition to the level detectors, the expansion tank is equipped with a 1-inch sample connection and a 1-inch helium line. The sample connection is closed with a blind Marman flange. The He line is used for He cover gas feed and bleed to automatically maintain system pressure and is equipped with a vapor trap to prevent carry over of NaK into the argon system.

The auxiliary secondary loop has two inch NaK service connections. Both of these connections are in the 2 inch NaK line at the outlet of the auxiliary air-blast cooler. The operating pressure in the auxiliary secondary system is determined by the pressure of the auxiliary secondary expansion tank which must be above 25 psig to assure that the secondary loop pressure is always maintained above that of the primary at the auxiliary IHX.

3. Pump Around Loop

The pump around loop continuously circulates NaK between the reactor vessel and primary drain tank to maintain the reactor NaK level. (Figure 80)

a.) NaK Flow

NaK is pumped from the primary drain tank through a 1 inch line by the pump-around pump, to the reactor vessel at a rate of 5 gpm. During reactor operation the NaK overflows through a 3 inch line back to the drain tank.

The size of the reactor overflow line, 3 inches, is indicated by an accident involving a ruptured tube in one of the intermediate heat exchangers rather than any requirement concerned with NaK level control in the reactor. The 3 inch overflow line has an isolation valve for filling the system with NaK.

b.) Instrumentation

The NaK flow is measured by a one inch magnetic flow-meter located in the inlet line of the pump-around pump.

56 PM

FIGURE 80. TREX NaK OVERFLOW SCHEMATIC

this flow is recorded in the main control room. A low flow alarm is set to sound at flows below 2.0 gpm.

The NaK temperature at the inlet to the reactor vessel is recorded in the control room. The temperature in the loop and drain tank will approach that of the reactor vessel outlet and will differ slightly only due to the heat losses from the drain tank and other components in the loop.

c.) Pump-Around Pump

The pump-around pump is an electromagnetic pump designed to deliver 5 gpm of 980°F NaK at 30 psi developed head. The frictional pressure losses in the one inch line from the drain tank to the reactor vessel are small (about 1 psi) because of the low flow (5 gpm) in the 1 inch piping. The principal head requirement is static.

The pump is equipped with 2 thermocouples on the winding. One is used as a spare and one is used to, (1) indicate the temperature, and (2) alarm in the control room and (3) shut off the pump at winding temperatures above 350°F. (See Specification 8-Appendix A)

4. Service Systems

The primary and secondary service systems are located in the radioactive sodium room and the secondary sodium rooms respectively. Major components of these systems are described below.

a. Cold Traps

The function of the cold trap is to remove NaK oxides from the NaK and thereby prevent possible formation of plugs of NaK in piping, fuel channels and orifices. Each of the four cold traps in the service systems is designed for a nominal NaK flow rate of 5 gpm and has a retention capacity of 200 lb. NaK oxide. Both primary and secondary cold trap designs are identical except for minor variations in length for the NaK inlet and outlet nozzles and gas coolant inlet and outlet nozzles.

The cold trap design utilizes an integral vertically oriented economizer, together with a finned vessel which serves as a cooler and oxide accumulator section. The vessel consists of

a 39-inch length of 24-inch schedule 20, type 304 stainless steel pipe with elliptical dished type upper and lower vessel heads. Inserted into the top dished head is a 42 inch length of 8-inch schedule 40 type 304 stainless steel pipe, which is the external shell for the economizer.

The cold trap cooler consists of 215 fins, each 37-inches long x .050-inches thick x 1-inch high, welded to the external surface of the 24-inch diameter vessel. Each fin is segmented into approximately 4-inch sections along its length.

Two traps are provided in each of the primary and secondary service system loops so that one can be in operation while the other is allowed to remain idle to permit delay of radio-active NaK prior to removal.

b. Hot Trap

The hot trap is a component of the primary loop service system to reduce the oxygen in the liquid metal to ≤ 1 ppm from that produced by cold trap operation (5-10 ppm). Flow through the trap is 5 gpm. The hot trap removes oxygen from the liquid metal coolant by reaction with zirconium, is operated at 1400°F to enhance this reaction and was designed to provide 1 in² of zirconium per 5 grams of NaK in the primary loop. The trap consists of two stainless steel frames in which spaced zirconium plates are mounted. These are enclosed in an 18-inch diameter Sch. 10 pipe approximately 6-ft long. There is 250 lbs of 0.015 inch zirconium sheet in the planned design. This amount of zirconium will accommodate a reduction of 10 ppm oxide to ≤ 1 ppm (introduced by re-fueling operation) two times per month for 10,000 hours continuous operation. The zirconium will contain 2% oxygen by weight at the end of this period. The hot trap operates with a heater and economizer.

c. Plugging Indicators

The plugging indicators are designed to measure plugging temperatures down to 40°F using flow rates of 1/2 to 1 gpm. Primary and secondary plugging indicator designs are identical.

The plugging indicator assembly is comprised of three sections; electromagnetic flowmeter section, economizer section, and fin-cooling section. NaK enters the plugging indicator through the flowmeter tube, flows through the economizer annulus, reverses direction through the fixed-orifice plate and returns through the inner tube out to the service system piping.

Nitrogen or air coolant flows counter to the annular sodium flow. Cooling of the oxide laden NaK continues until the NaK temperature drops to below the oxide saturation temperature, at which time oxide precipitation commences to restrict NaK flow through the orifice plate. The thermocouple in the NaK stream indicates stream temperature at which flow restriction through the orifice starts.

d. Oxygen Meter

The oxygen meter utilizes a galvanic cell and a voltage measuring and balancing device. The galvanic cell consists of a thoria-yttria ceramic electrolyte tube with a copper-cuprous oxide reference electrode inside, at the bottom of the ceramic tube. The electrode is sealed into a housing with a bottom tee through which the liquid metal flows. (Satisfactory operation has been demonstrated within a flow range of 0.1 gpm to 3 gpm). The electrode assembly in contact with flowing liquid metal forms a galvanic cell which develops a voltage that is inversely proportional to the logarithm of the oxygen activity (concentration) in the liquid metal. Measurement of oxygen concentration from 1.0 to 100 ppm has been demonstrated in sodium and potassium.

2. Helium Cover Gas and Cooling Systems

a. Purpose

Helium gas is used to provide an inert atmosphere in the plant vessels and piping which contain NaK. It is also used as the heat transfer medium in a cooling loop which is used to maintain the desired temperature in the TFE fission product trap and cesium reservoir. The electrical connector dome located above the reactor vessel and the fuel storage tank also have a Helium (He) atmosphere. (See Figure 82).

BLANK

Figure 81. Helium System

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FIGURE 82. TREX HELIUM COOLING SYSTEM

b. Functional Requirements

1. General

Helium systems include the He distribution system and the TFE cooling loop. The He distribution system provides a source of inert gas blanketing for all NaK surfaces and provides adequate pressure and capacity to fill the NaK systems. The TFE cooling loop provides the capability for continuous cooling of the TFE cesium reservoir and fission product trap. (See Figure 82).

2. Helium Distribution

The helium distribution system is comprised of the necessary equipment and facilities to receive, distribute and exhaust He cover gas for the various NaK system operations, such as transfers, system billing, etc. The system is comprised of a supply station for bottles or a tank truck, a pressure regulating system, exhaust system, and system regulating devices.

3. TFE Cooling Loop

The TFE cooling loop is comprised of the necessary blowers, coolers, and demisting units and associated piping, valves and controls to assure reliable continuous cooling of the TFE fission product traps and cesium reservoirs. To assure continuous operation the system is 100% redundant and is designed to allow a "defrosting like" back flush of one set of coolers while continuing operation with the other. This capability is provided to assure continuous operation even though one of the cooling systems is fouled with condensed NaK.

c. System Operation

1. Helium Storage

A helium storage facility consisting of banks of portable helium cylinders has a storage capacity of 3,000 scf. This capacity provides storage for about three months normal operation based on an average use rate of 600 scf per month. Table 35, Plant Helium Use Rate, tabulates the estimated helium usage per month for each item that will require helium on a periodic basis. Pressure and flow measurements are taken from the helium supply headers, to provide a check on helium consumption.

TABLE 35

PLANT HELIUM USE RATE

ITEM	SCF/MONTH
Refueling	100
Fuel Storage Tank	100
Reactor Cover Gas Pressure Control	200
Primary Drain Tank	50
Secondary Drain Tank	50
Expansion Tanks	100
Cubic Feet of Helium used per month	600

2. Helium Supply

Helium flows from the storage area through a pressure reducing valve which lowers the He pressure to 40 psig. It then flows to the various stations where it is used. At each station the He pressure is again reduced to the desired pressure by a pressure control regulator. (See Figure 81). Excess He will be vented to air from secondary system components and to the gaseous radwaste system from primary equipment.

3. Helium TFE Cooling Loop

A recirculating loop is shown on Figure 81 for cooling the TFE fission product traps and cesium reservoirs. The TFE clad at the fission product trap must be kept below 70°C (158°F) in order for the trap to operate properly. The cesium reservoirs must be cooled to approximately 200°C (392°F) to maintain the proper cesium gas vapor pressure.

The helium cooling loop provides the needed cooling by forcing cold (50°F) helium down a 40 mill annular gap between the process tube and the TFE. The required cooling per tube is approximately:

	<u>watts/tube</u>
1. Cesium reservoir heater plus conduction	50.0
2. Electrical head losses	24.0
3. Neutron & Gamma Heating	24.0
4. Convective heating from the NaK surface	2.0
Total	100.0

Thus, the total cooling load for 211 TFE's is approximately 21,000 watts. In order to remove the cesium reservoir design heat load, the helium must exit from the reservoir area at a temperature no greater than 366°F . Convective heating from the NaK surface increases this temperature to 375°F before it exits from the reactor vessel. To keep the fission product trap below 70°C (158°F) an inlet helium temperature of 50°F is required due to the cooling load which the helium sees ahead of the fission product trap. To supply the required cooling 188 lb/hr of helium must be circulated with a temperature rise of 325°F .

Each of the two He cooling loops are designed to provide 100% of the design cooling load. The loops each contain a cooling vapor trap, a freon cooler and a helium circulator.

The cooling vapor traps are designed to accomplish two functions at once. They reject 13 kw to the nitrogen atmosphere in the pit area and remove NaK vapor from the circulating He.

The cooling vapor trap reduces the temperature of the circulating He from 375°F to 175°F. Condensed NaK vapor is drained back to the primary drain tank.

The helium freon coolers are designed to remove an additional 8 kw of heat from the circulating He. In doing so the freon coolers lower the temperature of the helium from 175°F to 50°F. In order to reduce the possibility of a freon to helium leak, a double wall heat exchanger is incorporated. Stagnant helium is used as an intermediate fluid in the annular gap between the tube walls. A freon detector is used to monitor the stagnant helium in the gap for freon leaks.

The helium is circulated by one of two full capacity blowers operating in parallel. Each blower takes suction first through a cooling vapor trap and then through a freon expansion coil. Each circulator is designed to circulate 188 lb/hr (285 cfm) through a 0.5 psi head.

The helium TFE cooling system is 100% redundant in that either of two parallel cooling loops is sufficient. Because the freon expansion coils may become fouled with condensed NaK vapor, the freon system is designed to lower the operating expansion coil pressure (saturation temperature) of the freon as required to maintain a 50°F helium outlet temperature. By lowering the saturation temperature from 45°F to 20°F (its design limit) the log mean temperature in the freon cooler can be increased by 100%. This capability allows the freon cooler to operate satisfactorily even when severely fouled.

Periodically the freon coolers can be "defrosted" with a back flushing operation which circulates hot helium backwards through a deactivated cooling loop before passing it into the operating loop. This operation heats the deactivated freon cooler to 375°F, melting any condensed NaK vapor. The NaK then drains to the cooling-vapor trap where it is caught and returned to the primary drain tank.

4. Electrical Connector Dome Helium Supply

The TFE electrical connector dome is slightly pressurized with clean filtered helium to eliminate the possibility of NaK vapor laden He leaking up through the TFE grid plate into the electrical connector area. Pressurizing the connector dome assures a drain flow of clean He through all leaks in the TFE grid plate.

Pressurized helium is supplied to the electrical connector dome by a diaphragm pump which takes suction from the TFE cooling loop circulator outlet. The pump is designed to circulate 9 lb/hr at a head of one psi to the electrical connector dome. To assure complete removal of NaK vapor and other particulates from the He which is supplied to the electrical connector dome, a high efficiency charcoal filter is placed between the supply pump and the connector dome.

3. Cell (Nitrogen) Overpressure Relief System

a. Purpose

This system has a two-fold purpose. First, it maintains a negative static pressure in the spaces served by ventilation system No. 2. This provides that any cell leakage will be into these spaces rather than out of them. The second function of this system is to relieve any overpressure occurring in the cells.

b. Functional Requirements

(1.) Maintenance of Cell Static Pressure

All spaces served by ventilation system No. 2 are to be maintained at a negative static pressure. This area has the greatest likelihood of containing radioactive airborne contaminants. The negative pressure criterion serves to reduce the possibility of release of such contaminants to the surrounding areas.

The negative pressure is maintained by the use of the overpressure vent system blowers. These blowers draw from the cell atmosphere and exhaust to the stack.

(2) Relieving Cell Overpressure

Various situations may arise whereby the spaces served by ventilation system No. 2 become pressurized to some extent. The overpressure vent system will be used to reduce any overpressure before any cell damage is incurred. The cell atmosphere will be vented through a series of filters to remove at least some of the possible airborne contaminants.

c. System Operation

A schematic flow diagram of the overpressure vent system is shown in Figure 83. The major components are indicated.

(1) Maintenance of Cell Static Pressure

Under normal operating conditions, outside leakage into the cell area served by ventilation system No. 2 will cause the cell pressure to approach atmospheric conditions. When the differential pressure reaches some minimum value, the static pressure control blowers will be actuated. The pressure relief valves in the cell are actuated simultaneously with the blowers to allow intake from the cells. The blowers draw the cell atmosphere through a series of filters and then exhaust to the stack. The filters are provided to remove any radioactive airborne particles.

(2) Relieving Cell Overpressure

A high pressure condition will cause the pressure relief valves in the cell to open automatically. The high pressure in the cell space will force the gases through the system. Flow is routed through one of two banks of filters. From there it bypasses the blowers and is exhausted through the stack.

NOTE:
 TI — Test Injection Line
 SX — Sampling Line
 RM — Radiation Monitor

FIGURE 83. OVERPRESSURE VENT SYSTEM

Only one relief valve and one bank of filters is required for any foreseeable situation. Two relief valves are employed to provide redundancy and reliability. The flow is directed through one series of filters until its pressure drop becomes excessive due to clogging of the filters. When this occurs this bank is closed off and the parallel one accepts the flow.

In the event that the flow through the filters is not adequate to relieve the cell overpressure, the filter by-pass line can be used. This line allows a high flow rate and should reduce cell overpressure very quickly.

d. Component Description

(1) Pressure Relief Valves

Two 10" diameter gate valves are provided. They are actuated either on receiving a high pressure signal or on signal of minimum differential pressure conditions.

(2) Filters

Two filter banks are provided. Each of them consists of the following:

- . Prefilter or roughing filter--this unit removes the relatively large airborne particles in the gas flow.
- . Absolute filters--two are provided for each bank. These are HEPA type filters and have a very high efficiency for removing even small particles. They will remove 99.97% of all particles down to sizes as small as 0.3 microns.
- . Charcoal filters--these units are provided primarily for the removal of any iodine or iodides which may be present in the gas flow. These filters are at least 99% efficient for this purpose. An absolute filter is placed downstream of the charcoal unit to catch any charcoal granules which may be released.

Under clean conditions each filter bank can accommodate at gas flow rate of 500 CFM with a pressure drop of less than 6 inches H_2O . Each of the filters is of the replaceable, throw-away type. If they become clogged they can be replaced. Test lines are provided to test the efficiency of the filter units.

(3) Blowers

Two full capacity static pressure control blowers are provided. They each have a rating of 500 CFM at a static pressure of 15 inches H_2O . The flow through the blowers is controlled by dampers.

4. Gaseous Radioactive Waste System

a. Purpose

The purpose of the Gaseous Radwaste System is to collect, store, and filter any radioactive waste gases generated by the operation of the thermionic reactor experiment. The system also receives and processes excess helium resulting from normal operational requirements of the reactor system.

b. Functional Requirements

This gas system essentially fulfills three functional requirements. First, it collects and processes reactor and primary system helium which may possibly contain radioactive fission gases. Second, the system collects the helium discharged from both the primary and secondary coolant systems which results from normal operational coolant level changes. The third function of the gaseous radwaste system is to collect and process the fission gases released when fuel is disassembled after having been irradiated.

(1) Fission Product Removal from Process Helium

Fission gases may possibly be released into the reactor cover gas and thence to the remainder of the primary system cover gas spaces due to a fuel failure of some type. Such a release can also occur in the irradiated fuel storage pits.

Once this is detected, the contaminated spaces will be purged with clean helium. The purging process will be sufficient to reduce the gas activity to acceptable levels. The purged gases will be collected by the gaseous radwaste system. Here the gas will be stored in decay tanks until the short-lived fission product activity has been reduced significantly. Decay times of at least 100 days are anticipated. Following this holdup time the major gas activity should be due to decay of Kr-85. Its activity will remain essentially constant.

Once the short-lived activity has decreased significantly, the collected gases are passed through activated charcoal filters. These filters should collect even the Kr-85. The effluent gas should be clean helium. It can either be stored in the clean helium storage tank or vented through the stack to the atmosphere. If the clean helium is stored, it can be reused according to operational needs.

(2) Helium Collection Due to Normal Reactor Operations

During normal reactor operation it is estimated that about 600 scf of helium will be used per month. Unless a fuel failure of some type has occurred, the helium exhausted will be essentially clean and free from fission gases or other radioactive material. This exhaust helium will be collected by the gaseous radwaste system. Here it will be stored for reuse or vented to the atmosphere.

(3) Collection of Fission Gases from Disassembled Irradiated Fuel

The length of a complete TFE is on the order of 12 feet. This length is prohibitive in terms of shipping the TFEs intact. To reduce the shipping and handling difficulties the irradiated elements will be cut into shorter lengths.

Before the TFE is disassembled completely, the fission gases will be drawn off. The fission gases collected from this process will be transferred to the gaseous radwaste system. Here they will be held up in decay tanks until the short-lived activity has decreased significantly.

c. System Operation and Description

A schematic flow sheet of the gaseous radwaste system is shown in Figure 84. Only the major components are included in this diagram.

(1) Gas Collection

Gases from all sources, whether clean or contaminated are collected in a common low pressure draw tank. This tank is maintained at a pressure lower than that of any of the collection points. The low pressure is maintained by one of two compressors. These compressors are actuated by a high pressure signal from the draw tank. In the interests of reliability two units are provided in parallel.

The low pressure draw tank has a volume of 250 ft³. The two compressors are rated at 20 CFM at a total developed head of 130 psig.

(2) Gas Storage in Decay Tanks

Three high pressure decay tanks are provided to handle the exhaust from the compressors. All radioactive gases will be held up in these tanks until the short-lived activity has decreased to an insignificant level. The normal operating pressure is 125 psig or lower, depending on the volume of gas being handled. Once a tank is at this peak pressure it is valved off for the required decay period. The other two are then used to handle any additional storage needs in the interim period.

Following an appropriate holdup period, the full tank can be exhausted through the charcoal filter. This can be done

NOTE: Refer to Appendix A for
Component Specifications

FIGURE 84. GASEOUS RAD WASTE CLEANUP SYSTEM

directly, or the gas from the decay tank can be fed into the intake of the compressor header. The compressors will then be used to increase the gas pressure to feed it through the filters and thence to the cleaned helium storage tank.

(3) Filtration

a) General Concept

All contaminated helium in the gaseous radwaste system will be passed through one of the two charcoal filters provided. The primary purpose of these filters is to remove Kr-85 from the helium. They will also remove any minute traces of iodines which may be present.

Two charcoal filters are provided in parallel. The exhaust from a decay tank is passed through one filter until the Kr-85 activity in the effluent helium at the filter outlet reaches a significant level. When this occurs, this filter is valved off and the gas flow is directed through the alternate charcoal filter. The cleaned helium issuing from the filters is either stored in the cleaned helium storage tank or can be released to the atmosphere through the stack. The use of the compressors allowed the clean helium storage tank pressure to be as high as 125 psig.

Provision is made for regenerating the charcoal filter. This is accomplished by pulling a vacuum on the filter through the use of the vacuum pump provided. Heaters are also provided in the filters to aid in removing the contaminants trapped during the filtration process. The contaminated gases drawn off by the vacuum operation should have a relatively high concentration of radioactive species. The volume involved is quite small, and these contaminated gases can be transferred to high pressure vessels for permanent storage.

b) Component Description

Each of the filters contain 11 cubic feet of activated coconut shell charcoal (~110 lb.). They are provided with electrical resistance heaters with a capacity of 2100 watts. This rating should be sufficient to increase the charcoal temperature by 400°F in one hour's time.

The vacuum pump used to regenerate the charcoal filter is capable of a flow rate of 2 CFM at an inlet pressure of 0.001 Torr.

The clean helium storage tank is identical to the helium decay tanks. It has a volume of 100 ft³ and has a normal maximum pressure of 125 psig.

K. FUEL HANDLING EQUIPMENT

1. Fuel Handling Machine

a. Functional Description

The function of the refueling operation is primarily to remove and replace fuel elements (either TFEs or non-thermionic fuel elements (NTFE) from the core of the reactor. These elements are inserted and removed from fixed process tubes extending a length of about 144 inches from a top grid plate to the bottom of the lower axial reflector. A secondary function of the refueling operation is to remove and replace the individual process tubes in the reactor core. These process tubes are removable either with or without the contained fuel elements. The heat generation of a single fuel element at the time of refueling is sufficiently small (< 100 watts each) to make it unnecessary to provide forced cooling in the refueling machine.

In a typical operation, the refueling machine is brought into the room above the reactor along with a load of six new fuel elements (TFEs or NTFEs fuel). After indexing over a particular fuel location, the refueling machine is bolted and sealed to the top surface of a 30 inch gate valve and the intermediate space purged with helium. The refueling operation proceeds after opening the gate valve and a valve at the bottom of the refueling machine.

The following is a summary of the normal refueling operation:
(See Figure 85.)

- (1) Load refueling machine with six new fuel (TFEs or NTFEs) elements while establishing or maintaining a helium atmosphere within the machine. A supply bottle of helium is carried along with the refueling machine in order to provide slight positive pressure and prevent in-leakage of air.
- (2) Install refueling machine onto rails of the indexing mechanism - engage two locking pins below rails to prevent tipping of the carriage.

Figure 85. Refueling Sequence

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- (3) Move refueling machine horizontally to index over fuel element to be replaced. Travel is by electric motor with manual fine adjustment. Two visual scales are used for precise location in X and Y dimensions.
- (4) Clamp refueling machine to top plate of gate valve (manually). A spring loaded refueling machine carriage permits this clamping by exceeding the preload of the springs. Slightly spherical surfaces on the refueling machine bottom and on the top surface of the gate valve assure accurate alignment with the slight angularity of the fuel elements (max. angle = 0.79°).
- (5) Purge the space between the refueling machine and the gate valve with helium.
- (6) Open 30" gate valve and the valve at the bottom of the refueling machine.
- (7) Extend the guide tube from the refueling machine through the two valves to the top of the fuel element handle. This guide tube has a hexagonal inner surface to fit the fuel element handle (or process tube handle). It is actuated by a rack driven from a pinion in the body of the refueling machine. At its lower end, the guide tube has two small spring loaded "door latch" type of fingers which prevent the removal of the process tube when the fuel is raised. (These fingers are retracted by rotating the grapple in the direction for grappling the process tube.)
- (8) Rotate the barrel of the refueling machine to align an empty position with the guide tube.
- (9) Extend the fuel grapple its entire stroke length to contact the fuel element handle.
- (10) Rotate the grapple 60 degrees to engage the fuel element handle. Rotation is accomplished by turning the upper end of the central spline within the grapple shaft.

- (11) Raise the grapple, raising the fuel into position in the barrel of the refueling machine.
- (12) Engage stop to prevent dropping of the fuel element (this is not shown in the drawing).
- (13) Rotate the grapple to the unlocked position and raise to above the top of the drum.
- (14) Rotate the drum of the refueling machine to align a new fuel element with the guide tube.
- (15) Lower and engage grapple (by rotation).
- (16) Release stop (see step 12).
- (17) Lower fuel element into core.
- (18) Rotate grapple to release, and raise grapple to top of its stroke.
- (19) Rotate barrel of the refueling machine to position which allows retraction of the guide tube (one position in the barrel is provided for this purpose only).
- (20) Retract guide tube.
- (21) Close gate valve and valve at bottom of refueling machine.
- (22) Unclamp refueling machine and relocate a new position.
- (23) Repeat steps (4) to (22) as required to replace up to six fuel elements.
- (24) Relocate refueling machine to position under access hole in ceiling (or proceed to step (29)).
- (25) Release two locking pins in refueling machine [see step (2)] to permit raising of refueling machine.

- (26) Raise refueling machine with crane through hole in ceiling and place in location over hot cell.
- (27) By steps similar to (4) to (22), deposit fuel into cell.
- (28) If desired, introduce new fuel elements into refueling machine.
- (29) [Alternate to steps (24) to (28)]. Locate refueling machine over in-cell storage pit, and deposit fuel into pit by procedure similar to steps (4) to (22). A helium atmosphere is maintained.

The times to perform three selected fuel handling sequences have been estimated. The sequences are:

1. Loading six newly delivered TFEs in the Fuel Storage Well.
2. Transferring six TFEs from Fuel Storage Well to TREX core.
3. Removing a TFE from TREX core to Disassembly Cell.

The first sequence involves the transfer of TFEs from a sealed shipping container into the Fuel Handling Machine, the movement of the Fuel Handling Machine into the Reactor Cell and finally the deposition of the six TFEs into the Fuel Storage Well awaiting their use in the Reactor Core. The total time involved for this transfer is approximately four hours.

The second sequence is the transfer of six TFEs from the Fuel Storage Well to the Reactor Core, the substitution of TFEs for NTFEs and the transfer of NTFEs into the Fuel Storage Well. The time involved for this transfer is approximately 4-1/2 hours. There is a possible need for two Fuel Storage Wells; one for new TFEs and the other for irradiated TFEs and NTFEs.

The third sequence is the removal of a TFE from the Reactor Core to the Disassembly Cell for examination. As currently envisioned, this involves lifting the Fuel Handling Machine through the hatch of the Reactor Cell into the Disassembly Cell. The approximate time for this operation is 3-1/2 hours.

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FIGURE 86. ASSEMBLY OF TFE INTO PROCESS TUBE

FIGURE 87. FUEL HANDLING MACHINE CAPACITY

The amount of reactor down time is different than certain of the fuel handling sequences in that portions of the sequences may be performed during reactor operation. In any case, a detailed time and motion study is necessary concurrent with preliminary design of the facility and fuel handling machine.

The refueling machine is capable of also removing and replacing process tubes. These process tubes have handles which extend to the same level as the tops of the fuel element handles, with the two units nesting together as in Figure 86. The process tube is engaged by a 60° rotation of the grapple in a direction opposite to that required to engage the fuel element. This rotation causes a cam surface on the grapple shaft to depress the two process tube hold-down fingers and prevent interference until the process tube is raised. The process tube can be raised and replaced either by itself or with the fuel element using the same grappling method. When the process tube is raised from the core position, friction or binding between process tubes might tend to cause an adjacent tube to be lifted. This is prevented by the overlap of the guide tube of the refueling machine.

b. Detailed Description

(1) Purpose

- (a) To remove and replace fuel elements (TFEs or driver--same configuration)
- (b) To remove process tubes--with or without fuel elements
- (c) To install process tubes.

(2) Capacity: 6 fuel elements in one loading (see Figure 87)

(3) Storage drum: 8 position storage drum in refueling machine has space for 6 plus one extra for removing first unit and one position for retraction of guide tube. Holes hexagonal to prevent rotation of fuel handles

- (4) Total Length of process tube: 144 inches
- (5) Total length of fuel element: 142 inches
- (6) Max. diameter of process tube (excluding handle): 1-1/4 inches
- (7) Size of handle: 1.345 inches across flats of hexagon (See Figure 88)
- (8) Distance from refueling machine to tope of process tube: 35 inches
- (9) Total stroke of refueling grapple: 189 inches
- (10) Method of lifting: chain and sprocket
- (11) Grapple actuation: rotation of rack +60° to engage fuel element , and -60° to engage process tube--actuated by manual rotation of upper end of two telescoping square splines.
- (12) Guide tube: extends from refueling machine to top of fuel element and process tubes, retracts into refueling machine (through gate valve) when not in use. Actuated by rack and pinion -(inside hexagonal to prevent rotation of fuel element or process tube) 41 inch stroke.
- (13) Process tube hold-down: two fingers in guide tube prevent lifting of process tube when engaging fuel element--cam on grapple retracts fingers when rotated to engage process tube.
- (14) Shielding: 6 inches of lead (~6000 lbs) surrounding lower end of 7 inches diameter storage drum
- (15) Indexing: refueling machine must scan a 24 inches dia. circle, with indexing accurate to 0.060 inch. It may also be free to tilt up to 0.79° from the vertical.

FIGURE 88. TFE HANDLE ASSEMBLY

- (16) Valve: a rotary valve (~2-1/2" dia. opening, 3" thick) at the lower end of the refueling machine provides a gas seal as well as shielding.
- (17) Seal to gate valve: The 58" dia. lower surface of the refueling machine has a face seal near the outer periphery, sealing against the flange of the mating 30" gate valve, and allowing adjustment over a 24 inch circle.
- (18) Method of angular alignment: The mating surface between the refueling machine and the gate valve is provided with a 2100 inch spherical radius.
- (19) Gas system: The refueling machine operates with an internal atmosphere of helium. A cylinder of helium is mounted on the refueling machine to maintain a slight positive pressure to prevent in-leakage of air.
- (20) Seals: All penetrations are provided with seals.
- (21) External gas system: Helium is provided for purging the space between the refueling machine and the gate valve (not part of refueling machine)
- (22) Actuation: All actuation is manual except for the grapple lift and the two linear indexing motions. The motor drives are provided with manual override for fine adjustment.
- (23) Material: The inner surfaces of the refueling machine are lined with 304 stainless steel. The rest of the machine is carbon steel (except for the lead shielding).
- (24) Weight:
lead	6000 lbs.
carbon steel	2000
204 S.S.	2000
Total	10,000 lbs.

2. Fuel Handling Equipment

The equipment planned for the remote handling operations on the thermionic fuel element and other thermionic reactor experiment core components is described in the following section. The cost of equipping and operating a radioactive materials laboratory (RML) coupled with the existence of such facilities already equipped and operational dictate that post operative examinations of radioactive components (primarily fuel elements) will be done at a place other than the test facility. Therefore, equipment is described only for operations needed to prepare the components for shipment to an RML.

a. Hot Cell Cleaning Station

The core element (TFE, NTFE or reflector rod) is stored for transfer inside the fuel handling machine in a clean helium environment upon removal from the reactor core. The core element may then be moved to a storage pit, or to an opening in the upper hot cell hatch. It is likely that liquid will adhere to the exterior of the core element. Therefore, the element must be cleaned before RML operations are performed. Cleaning is achieved by inserting the element (single element capacity is currently envisaged) into a cylindrical container. This chamber is equipped so that the moisture content may be changed from a low or zero value to a point where spray flushing of the element is occurring. This operation may be manual or automatic. Reaction of the liquid metal with water and flushing of the core element exterior should provide adequate cleaning. A drain for potentially radioactive material from the exterior of the core element must be provided. In addition, it must be possible to have an initial environment of clean helium so reaction with the liquid metal on the elements does not occur inside the fuel handling machine during element transfer to the cleaning station. The core element is reinserted in the fuel handling machine after cleaning and transferred to the storage or to the RML for additional operations.

b. Gas Sample Machine

The only core element which requires a gas sample to be taken is the thermionic fuel element. Samples of the gaseous constituents

of the TFE interelectrode volume are needed for post operative analysis. Since the sectioning operation required for shipment prohibits subsequent sampling, this operation is performed first. This operation is performed by moving the cleaned TFE in the Fuel Handling Machine to a second opening in the test building. At this location, the TFE is inserted into the gas sample fixture. This fixture provides mechanical support for the TFE and the controlled environment envelope piercing operating in which the gas sample is taken. The apparatus planned is similar to that currently in successful use in the Vallecitos Nuclear Center RML.

c. Cutoff Machine

The TFE is moved vertically from the gas sample machine to a location where it is cut into lengths suitable for shipment in available casks. Other core elements may be moved directly from the cleaning chamber to the cutoff machine. Elements may also be moved directly from the Fuel Handling Machine to a transfer cask, if on site Los Alamos Scientific Laboratory burial is permitted and no post operational examinations are required.

Cutting of the core elements is planned to be accomplished with an abrasive saw. The fixture in which this operation is performed provides mechanical support for the core element during the cutoff operation. A similar unit has been used successfully at the Vallecitos Nuclear Center RML.

d. Other RML Equipment

After cutting the elements to length, the sections are moved from the cutoff machine to the transport cask with manual manipulators. The cask is moved from the location in which elements are loaded through an anteroom, and placed in a position below the test building crane hook with a transfer cart. In addition to at least one pair of manual manipulators, shielding windows must be installed in all the upper floor remote disassembly openings. Installation of windows in the lower hot cell will not be required. The openings provided for windows in the lower cell may require leak-tight covers be installed.

L. PRELIMINARY SHIELDING CALCULATIONS FOR TREX

Calculations were made to provide preliminary design basis for the several shields needed for the proposed TREX configuration. These are described in the following sections.

1. Coolant Activity

The amount of induced activity in the coolant is dependent on the specifics of the reactor design and on the primary loop inventory. The following relation shows what quantities are needed to compute the activity:

$$A = \frac{C}{V_T} \left[N_{Na} \sum_i V_i (\overline{\phi \sigma}_{Na})_i + N_{K41} \sum_i V_i (\overline{\phi \sigma}_{K41})_i \right] \quad (23)$$

where C = conversion constant (3.7×10^{10} dps/curie)
 V_T = total primary loop volume
 N_b = concentration of species b atoms/cc
 V_i = volume of region i in primary loop
 $(\overline{\phi \sigma}_a)_i$ = average flux - n γ cross section produce in region i of primary loop

In deriving this relation, it was assumed that only the Na 23 (n γ) and K41 (n γ) contribute significant amounts to the coolant activity. There is a possible ~5% error due to ignoring (n ρ) and (n α) reactions. It was also assumed that the half-lives of the radioactive species are long compared with a loop circuit time and that the reactor has run at constant power for several half-lives. As can be seen, the crux of obtaining a good estimate for the coolant activity rests with obtaining accurate values of $\overline{\phi \sigma}$. It is well-known that the flux spectrum varies quite radically throughout a reactor, and is especially sensitive to the details of the reflector design. It is, therefore, anticipated that as the design evolves, there may be significant (the order of a factor of 2) changes in the estimate provided at this time.

Both one and two-dimensional transport theory calculations of a preliminary reactor design containing 151 non-thermionic fuel elements

and 90 BeO rods in the core are available. These calculations reflect the current thoughts on the axial reflector design, so mockup axial calculations were run assuming a 6-inch Mo shield above the upper axial reflector. In order to obtain estimates, the data available from the above calculations were related by working with averages of both flux and cross sections by defining:

$$\bar{\sigma}_i = \frac{\int dV_i \int_0^{\infty} \phi(E) \sigma(E) dE}{\int dV_i \int_0^{\infty} \phi(E) dE} \quad (24)$$

$$\bar{\phi}_i = \frac{\int dV_i \int_0^{\infty} \phi(E) dE}{\int dV_i} \quad (25)$$

The average flux in each region of the reactor was ratioed to the average core flux, thus, only requiring an absolute number for the core flux. This latter value was obtained from the following expression:

$$\bar{\phi}_{\text{core}} = \frac{1.2 \times 10^{13}}{\bar{\sigma}_{f \text{ core}}} \left(\frac{P}{M_{25}} \right) \quad (26)$$

where $\bar{\sigma}_{f \text{ core}}$ is the core averaged one-group fission cross section of U-235

P is the reactor thermal power, kW

M_{25} is the U-235 inventory, kg

Table 36 shows data derived for the coolant activity estimate. Using a U-235 inventory of 132 kg, the following relations result:

$$A_{\text{Na}} = \frac{3}{V_T} \text{ mC/cc/kW} \quad (27)$$

$$A_{\text{K41}} = \frac{0.3}{V_T} \text{ mC/cc/kW} \quad (28)$$

where V_T is the total loop volume in liters.

TABLE 36. Summary of Data Used for Coolant Activation

Region	Coolant Volume, l	$\bar{\phi}/\bar{\sigma}_{\text{core}}$	$\bar{\sigma}_{\text{Na}}, b$	$\bar{\sigma}_{\text{K41}}, b$	$\bar{\sigma}_f$
Core	11.0	1.0	0.0013	0.0057	1.58
BeO Region	7.1	0.62	0.0065	0.019	-
Downcomer	14.6	0.45	0.0090	0.024	-
Lower Plenum	15.3	0.074	0.012	0.043	-
Lower Axial Refl.	1.5	0.40	0.011	0.029	-
Upper Axial Refl.	4.6	0.41	0.010	0.038	-
Upper Plenum*	8.3	0.074	0.0085	0.031	-

* Only a 6-inch height was used in the calculations as the $n\gamma$ reaction rates are negligible above this level.

The total loop volume is roughly 1000 liters and the reactor thermal power is 1500 kW. The equilibrium coolant activity in the loop would then be 1.3 mC/cc; about 3 curies/kW of reactor power.

2. Fission Product Inventory

The fission product inventory in the reactor depends on the power-time history of the reactor and to some extent on the fissionable isotope and neutron spectrum in the reactor. Data for fission product yields were obtained from Meek and Rider⁽¹⁵⁾ for the U-235 with fission spectrum neutrons. The fission product activities were computed with the aid of the RIBD-1 code⁽¹⁶⁾ which follows the history of 450 individual isotopes for an arbitrary power-time history. Two histories were investigated: continuous operation for one (1) year and 67% load factor for one (1) year (~ 2 months up, ~ 1 month down in a repeating cycle). The fission product activity saturates quite rapidly, so that the results for the two cases were not significantly different at shutdown, and only minor differences were observed for long periods after shutdown. Figure 89 provides a summary of the results per MW of reactor operation for the first power-time profile

Figure 89. Fission Product Activity/MW of Reactor Operation

(worst case). Volatile solids are defined as ^{33}As , ^{34}Se , ^{37}Rb , ^{42}Mo , ^{51}Sb , ^{52}Te and ^{55}Cs . A.R.F.P. stands for all the rest of the fission products.

3. Neutron Leakage

The neutron leakage spectrum from the reactor will be strongly dependent on the details of the radial and axial reflector configurations. However, the total neutron leakage is not as sensitive, as may be seen from the following relation:

$$n = v (3.1 \times 10^{13}) P L_f \text{ neut/sec} \quad (29)$$

where P is the reactor power in kW

L_f is the fractional leakage from the reactor

For a large variety of fast reactor designs studied in the past, L_f is in the range of 25-40%. A two-dimensional calculation has estimated the value of L_f for a preliminary reactor design using 151 non-thermionic fuel elements and 90 BeO rods with a 3-inch external Be reflector as 30% and the average spectral distribution of this leakage is given in Table 37. It should be remembered that the spectral distribution is very sensitive to the details of the reflector design; and in particular, the high energy leakage may change markedly as the design evolves.

4. Gamma Ray Leakage

There are several sources of gamma rays in this reactor. These include prompt fission, fission products (delayed), neutron capture and inelastic scattering. By far the largest contribution to the gamma ray sources are prompt fission γ s ($\sim 7.8 \text{ Me/fission}$) and fission products ($\sim 6.5 \text{ Me /fission}$).⁽¹⁷⁾ Capture in U-235 and in tungsten product $\sim 6.4 \text{ Me /capture}$ and these two materials account for almost 95% of the absorptions in the reactor. For the purposes of this estimate, it was assumed that all captures produce 6.4 Me distributed

TABLE 37. Estimated Neutron Leakage Spectrum

Energy Range, ev	Fraction of Neutrons in Energy Band, %	Leakage n/sec for 1000 kW
$2.5 - 10.5 \times 10^6$	1.2	2.8×10^{14}
$1.4 - 2.5 \times 10^6$	5.7	1.3×10^{15}
$0.8 - 1.4 \times 10^6$	4.6	1.1×10^{15}
$4 - 8 \times 10^5$	5.4	1.3×10^{15}
$2 - 4 \times 10^5$	6.9	1.6×10^{15}
$1 - 2 \times 10^5$	5.8	1.3×10^{15}
$4.65 - 10 \times 10^{14}$	6.0	1.4×10^{15}
$1 - 4.65 \times 10^{14}$	12.6	2.9×10^{15}
$1 - 10 \times 10^3$	16.0	3.7×10^{15}
$1 - 10 \times 10^2$	13.8	3.2×10^{15}
0.215 - 100	17.5	4.0×10^{15}
0.001 - 0.215	4.5	1.0×10^{15}

TABLE 38. Leakage γ s from the Reactor After
100 Hours of Full Power Operation

Energy Group, Mev	$\bar{E}$, Mev	Effective Core Self Absorption	Leakage γ γ /sec / fission	Leakage γ γ /sec @ 1 MW
0 - 0.4	0.3	0.93	0.51	1.5×10^{16}
0.4 - 0.9	0.7	0.78	1.3	4×10^{16}
0.9 - 1.35	1.0	0.74	0.62	2×10^{16}
1.35 - 1.8	1.5	0.67	0.43	1.3×10^{16}
1.8 - 2.2	2.0	0.62	0.29	9×10^{15}
2.2 - 2.6	2.5	0.64	0.18	5.5×10^{15}
2.6 - 3.0	2.8	0.63	0.11	3.4×10^{15}
3.0 - 4.0	3.5	0.63	0.14	4.3×10^{15}
4.0 - 5.0	4.5	0.65	0.049	1.5×10^{15}
5.0 - 6.0	5.5	0.68	0.016	5×10^{15}
6.0 - 7.0	6.5	0.71	0.003	9×10^{13}
7.0 - 7.5	7.2	0.71	0.0006	2×10^{13}

in the thermal capture γ ray spectrum tungsten.⁽¹⁸⁾ The capture strength is estimated at 1.6 Me /fission. Since this is the primary source of high energy gamma rays, the estimates provided at this time for the high energy gamma leakage are subject to significant changes as the reactor design evolves. For this estimate, inelastic scattering was set at ~ 0.1 Me /fission, distributed with the inelastic γ spectrum of tungsten.⁽¹⁸⁾ The latter estimate is rather arbitrary and needs to be borne out by more definitive computations. In computing the leakage γ s from the reactor, it was recognized that the average reactor density of near 7 gm/cc would allow only the gammas generated in the outer layer of the core to escape. Using Rockwell's⁽¹⁹⁾ formula for a sphere:

$$\gamma/\text{sec at surface} = SB (\mu Z) \left[\frac{3}{2\mu R} \left(1 - \frac{1}{2\mu R} + \frac{e^{-2\mu R}}{2\mu R} \right) \right] \quad (30)$$

where S is the total source

B is the buildup factor

R is the core radius

Z is the effective thickness of this source

μ is the attenuation coefficient

it is possible to compute a core self-absorption. Data for gamma ray buildup and attenuation were taken from Kapo,⁽²⁰⁾ using tungsten as the model. No attenuation by the reactor reflectors was assumed. Table 38 shows the results, including the effective core adsorption factor (e.g., 93% of the lowest energy gamma rays never escape from the core). The γ leakage for 1000 kW operation is also shown.

5. Fission Product Trap Activity

Fission gases and some volatile solids migrate from the reactor core to the charcoal fission product trap. The activity of the trap depends very strongly on the effective holdup time of these fission products in the fuel/emitter. Calculations were performed assuming that the following products (plus subsequent daughters) migrate to the top: Br,

Kr, Rb, Te, I, Xe, Cs. Table 34 shows the total available activity in these products plus daughters for these cores:

TABLE 39. Fission Trap Activities per
MW of Reactor Power

Gamma Energy Group, Mev	Case I [*] curies	Case II ^{**} curies	Case III ^{***} curies
.1 - .4	604,000	207,000	165,000
.4 - .9	388,000	103,000	20,500
.9 - 1.35	317,000	47,100	30,600
1.35 - 2	183,000	63,700	14,200
2 - 3	77,400	6,400	730
3 - 4	16,700	65	4
4 - 5	17,600	40	3

* No holdup time, no shutdown time

** No holdup time, 5 hours shutdown

*** 10 hours holdup, 5 hours shutdown

6. Estimated Shutdown Radiation Levels at the Refueling Floor Level

For the proposed layout of the reactor experiment, contributions to the radiation levels at the refueling floor include:

- core γ s
- fission trap γ s
- NaK activity
- Actuation γ s

The computations concluded that the dose levels to all areas except directly above the core were less than 1 mv/hr. Later studies may show that the concrete thickness can, in fact, be reduced. Directly above the core, there is a significant amount of streaming possible, and the streaming contributions dominate. The basic formula used in calculating the attenuated and streaming components were

$$\phi = BS \frac{e^{-\Sigma_R X}}{4 \pi R^2} \quad (\text{neutrons and gammas}) \quad (31)$$

$$\phi = \phi_o \frac{r^2}{2 L^2} e^{-\Sigma_R X} \quad (\text{neutron and streaming}) \quad (32)$$

$$\phi = 5 \phi_o \frac{r^2}{2 L^2} e^{-\Sigma_R X} \quad (\text{gamma streaming}) \quad (33)$$

where $\Sigma_R X$ represents attenuation through a homogeneous medium. The removal cross section approach⁽²¹⁾ was used for neutrons, and the buildup-attenuation approach was used for gammas.

Using the data presented on the first 5 sections and assuming a ten hour holdup time for fission product gases, the five hour shutdown dose rate directly above the core was estimated to be of the order of 10-100 R/hr. This level indicated the need to cut down the amount of streaming by reducing the TFE handle clearance in the process tube bundle at the top, and by using a thick grid plate between the process tubes which has small clearances, and perhaps a stepped arrangement to eliminate direct streaming.

7. Shutdown Radiation Levels to Actuators

Using the same procedures described in Section 6, a thickness of 24 inches of magnetite concrete followed by 6 inches of iron appears to be adequate to reduce the shutdown and neutron activation dose rate to the actuator to permissible levels for maintenance. When detailed designs are performed, care must be taken to minimize the streaming paths for radiation, which were not taken into account in the computations.

8. Evaluation of Conceptual Design Shielding Wall Thicknesses
Delineated in Figure 90 .

a. Reactor Shield (Blast Shield)

The blast shield is shielded adequately to limit neutron activation so that personnel access into the spaces external to this shield is permissible after reactor shutdown. However, the opening in this shield for the NaK inlet-outlet coolant pipes will expose personnel standing on the platform below the IHX to an activation gamma dose rate of about 1 R/h. To overcome this minor deficiency, provision will be made to install gamma shielding blocks (or bricks) around the inlet-outlet NaK coolant pipes following their installation.

b. Fuel Storage Tank Shield

The height of this shield will be lengthened so that there is no opening between the shield and the refueling cell floor. This will be done since it is expected that the TREX fuel assemblies in this storage tank will emit a dose rate of about 1 R/h due to neutron activation of parts of the assemblies located about 3 feet above the active fuel region.

Two weeks after reactor shutdown, a dose rate of 5 R/h was calculated 10 feet from the outer surface of the 2.0-foot thickness fuel storage tank shield wall. The assumption was made that 10% of the core fuel, having undergone irradiation for 1 MWt and 0.67 capacity factor, was located in the spent fuel storage tank. This radiation level may be too high and can be reduced to

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Figure 90. Test Cell Arrangement

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about 10 mR/h by doubling the shield wall thickness to 4.0 feet - or a mass equivalent of 425 gm/cm^2 .

c. Shield Wall Around the IHX Equipment

Calculations show that the magnetite concrete shield thickness is adequate to attenuate the primary coolant activity dose rate to about 1 mR/h, assuming the IHX as the source. Hence, the IHX equipment shield walls are adequate everywhere except above the control rod drive shield plug.

Geometric attenuation and limited access requirements will probably make the thickness adequate here also. If it is not, a solution would seem to be some gamma shielding blocks (or bricks) stacked above the control rod drive shield plug.

d. Refueling Machine Shield Wall and Roof

The proposed thickness is adequate to permit personnel access outside of the shield walls during 1 MWt operation. Later detailed analyses may permit the roof thickness to be reduced assuming that required access to the roof area will be limited.

M. FACILITY

1. Introduction

The Thermionic Reactor Experiment study was made based on the assumption that the Fast Reactor Core Test Facility (FRCTF) at Los Alamos would be available for use for the experiment. This facility is located at the Ten site of the Los Alamos Scientific Laboratory, a location approximately 1 mile from the city of Los Alamos, New Mexico. The FRCTF had been originally planned to house the molten plutonium test reactor.

The facility arrangement is one in which the test reactor is installed in either of two pits which are in the floor of the main test building. The control room and other ancillary equipment space is provided in the facility. There are in addition two remote disassembly cells adjacent to the test pits located below the level of the main floor.

2. Site and Facility Layout

Figure 91 is a plot plan of the Ten site. The FRCTF is located on the northeast corner of the complex. The FRCTF is labeled "Thermionic Element Test Facility". The other buildings in this complex are in use by LASL. A floor plan of the FRCTF is shown in Figure 92 at the 100 foot elevation. The two test pits are 22' x 22' square and 37-1/2' deep. Adjacent to the west test pit is space for a gas cleanup cell. South of the test pits is the non-radioactive sodium room, the location proposed for the TREX secondary coolant loop. The test pits are housed in a 47' x 100' bay serviced by a 50 ton crane. The control room is located adjacent to the test bay and has an area of approximately 1500 square feet. Mechanical equipment for normal building functions such as heating, ventilating and emergency power are located in a room north of the test bay. There are also offices provided for a limited number of the operations personnel. At the 100 foot level a pad south of the facility has been constructed on which the secondary loop liquid metal-to-air heat exchanger may be located.

Figure 93 is the 80.5 foot elevation. In the center of the figure are the east and west reactor pits with the radioactive sodium room located

FIGURE 91. PLOT PLAN—TEN SITE

FIGURE 92. FACILITY FLOOR PLAN—100-FT. ELEVATION

FLOOR PLAN-ELEVATION 80'-6"

FIGURE 93. FACILITY FLOOR PLAN-80.5-FT. ELEVATION

to the south of them. On the right end of the radioactive sodium room is a pit for the primary coolant loop drain tank. Two storage wells are located between the two test pits. Adjacent to the two test cells is the upper level remote disassembly cell. There are three shielding view windows and provision for three pairs of the manual manipulators. A hatch in the floor of the main test building provides access through which material may be moved from the test building into the upper disassembly cell. In the floor of the upper disassembly cell are removable slabs which allow access from the upper to the lower disassembly cells. In the upper disassembly cell there is shielded equipment transfer space in which remote cell equipment maintenance may be performed and through which material transfers may be made from the upper disassembly cell. A hatch between the equipment transfer room and the main area of the test building provides a means for material transfer from the upper disassembly cell to the test building. The electrical equipment room is immediately below the control room and provides cable connection space to the control and data equipment located in the control room.

Figure 94 shows a layout of the facility at the 62.5 foot elevation. The east and west reactor pits are in the center of the figure. The thickness of the shield walls is shown. The lower disassembly cell is located north of the west test cell. Again, as in the upper cell, there are three viewing windows and provision for the location of three pairs manual manipulators. The ventilation equipment room No. 1 is located to the right of the lower disassembly cell. This ventilation equipment provides cooling and a controlled nitrogen atmosphere for both the upper and lower disassembly cells. The sample transfer space is on the west end of the lower disassembly cell. Samples may be moved from the lower disassembly cell, through a shielded door, to a 12 ton freight elevator for movement from this floor to the building main floor (100' level).

Ventilation equipment room #2 is located west of the test pits. This system provides heat removal and a controlled nitrogen gas environment for the east and west reactor pits and the radioactive sodium room. The equipment has a 1.6×10^6 Btu/hr heat removal capability and employs nitrogen-to-water heat exchangers for cooling. The coolant

FLOOR PLAN-ELEVATION 62'-6"

FIGURE 94. FACILITY FLOOR PLAN-62.5-FT. ELEVATION

ducting (4' diameter) is located outside the building as shown in Figure 95.

Figure 96 is a vertical section of the FRCTF through the upper and lower disassembly cells showing the vertical relationship of these spaces. The viewing window and manipulator penetration locations can be seen for both disassembly cells. The hatches through which material may be transferred from the test bay floor through to the lower disassembly cell is illustrated. Also shown is the equipment transfer and transfer cell arrangement. The hook clearance for the 50 ton crane is 45 feet.

Figure 97 is a section through the east pit showing its relation to the radioactive sodium room and the depth and location of the radioactive sodium dump tank pit. No test pit shielding cover was constructed. The hatch over the radioactive sodium room provides vertical access for equipment placement.

Electrical power is supplied to the Fast Reactor Core Test Facility by a ring loop network. Two transformers supply a total of 2000 kVA to the facility. In addition there is a 75 kVA emergency diesel generator for selected emergency systems in the facility, as well as a 25 kVA internal combustion motor generator.

Holdup tanks for radioactive waste are located in the facility. A disposal facility exists elsewhere on the LASL site.

3. Proposed Modification

Listed below are the proposed facility modifications to provide capability for the performance of the Thermionic Reactor Experiment.

a. Test Pits

An artists concept of the proposed modification to one of the two test pits in the Fast Reactor Core Test Facility is shown in

II-305

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FIGURE 95. VENTILATION SYSTEM NO. 2 DUCT LAYOUT

FIGURE 96. FACILITY VERTICAL SECTION

FIGURE 97. FACILITY VERTICAL SECTION

Figure 98. This section is through the east side of the east reactor test pit and is similar in view to that shown in Figure 97. About 2/3 of the distance from the test pit floor to the 100 foot level is located a shielding floor provided to shield personnel in the space above the reactor from the radiation emanating from the thermionic reactor core below. The floor shield thickness is 4-1/2 feet.

For convenience of discussion, the space in the test pit may be divided into four quadrants in which functions or equipment have been located. The reactor core is located in the right quadrant of the test pit and is shown below the refueling machine. The reactor is mounted from the top, at the main shielding floor. Above the reactor is the instrument dome and the fuel handling machine which have been described previously in this report. Below the shield floor and around the reactor is located a blast and shock wave shield. Outside the shock shield is a concrete biological shield for personnel protection during actuator maintenance. There is an opening in the cylindrical shaped outer concrete shield through which the reflector cup may be moved on the rails shown. Shielding thickness are based on the use of magnetite concrete. More detailed information is shown on Figure 90.

A storage pit into which core elements may be placed is located in the second pit quadrant. There is a rail system between the reactor and the storage pit on which the fuel handling machine may be rolled. The storage pit has a valve at the top end which serves the same function as the valve in the top of the reactor vessel dome. The storage pit has a controlled helium environment.

In the third quadrant of the test pit is located the primary loop of the liquid metal cooling system. (See Figure 79.) As shown in Figure 98, the elevation of the primary to secondary heat exchanger is above the core, as are all of the components in the primary coolant system. These vertical locations were selected to assure that liquid metal

FIGURE 98.

coolant could not be siphoned from the reactor core should there be a break in the primary coolant system piping. Further, with the vertical location of the components as shown, and with the secondary loop NaK-to-air heat exchanger at a still higher elevation, in the absence of electrical power, approximately 5% of the heat energy generated by the reactor may be removed by convection cooling. Between the operating gallery in which the reactor core and the storage pit are located and the space in which the primary coolant loop is located is a wall which must be constructed to provide biological shielding for personnel in the operating gallery. This shield wall is needed since the primary loop liquid metal will become activated during reactor operation. There will be penetrations provided near the top of the primary loop hatch routed from the primary-secondary heat exchanger to the secondary loop to be located in the secondary sodium room.

An access way is provided in the fourth quadrant of the pit. The access way is planned to be used principally for the entrance and exit of personnel to the space below the reactor for maintenance of the actuators on the reflector control cup. It is planned that there would be a stairway which could be placed in and be removable from this access way. In addition to access to the area beneath the reactor, there is also a shielding door at an intermediate level through which maintenance personnel may enter the space in which the primary coolant loop is located. The reflector control cup also may be installed and removed from its position below the reactor core through the access way. Normally this operation would be performed infrequently but space is provided so this may be accomplished should the need arise.

Hatches for equipment handling by the crane are planned for the primary coolant loop and the region adjacent to the reactor. The space above the refueling gallery is enclosed with a room whose walls and ceiling are approximately 1 foot of concrete. The room must be steel lined for safety and to enable a controlled nitrogen

environment. This room height must be great enough to allow the horizontal movement of the refueling machine between the reactor and the storage pit. A hatch through which the refueling machine may be moved is provided in the roof. This hatch is located over the storage well to preclude damage to the reactor should the refueling machine drop. The room above the operating gallery also provides containment and nitrogen atmosphere control during the time the reactor is in operation. Further, in the event that breach of the liquid metal inert cover gas containment occurs during the fueling operations, this space may be evacuated and nitrogen gas valved into the room to control the potential liquid metal fire. Personnel access to the refueling gallery is through a double personnel lock from the test building floor. A stairway is planned between the personnel lock and the operating floor of the refueling gallery. Provision is planned for either nitrogen or air to be valved to the space underneath the reflector cup so that when maintenance is required personnel may enter this space without need for breathing apparatus. The same provision is made for the space in which the primary coolant loop is located. Personnel would not be able to enter the space in which the primary loop was located until the liquid metal had been drained to the drain tank and the radioactivity within acceptable limits. Isolation valves are located in the test pit ventilation ducts so the spaces may be vented or isolated as appropriate.

The construction work required in the pit is indicated below. The steel pit liner would be removed to the degree necessary to expose the concrete behind it. Concrete would be removed to point where structural steel is exposed and the structural members tied in for the pit walls and floors to be added. At this point, forms would be placed and concrete poured to complete the structure of the shielding floor, the shielding around the reactor and between the reactor and primary coolant loop space, and the shielding between the refueling gallery and the primary coolant loop. After the concrete has been placed the steel liner would be replaced as necessary to complete the containment. The liner in both pits and the liner in

the radioactive sodium room would be leak checked and leaks repaired to bring the leak rate to approximately 2 to 3 percent per day. In the leak test which was performed at the time the facility was completed, the leak rate was found to be approximately 30% per day.

b. Building Cooling Systems

Ventilation systems for the environmental control of the test pits and the radioactive sodium room and the two remote disassembly cells is already installed. Ventilation system #1 provides cooling and control of the oxygen content in the nitrogen cover gas in the two disassembly cells. Ventilation system #2 provides oxygen control and heat removal for the components located in the two test pits and the radioactive sodium room. Duct work must be added so that cool nitrogen may be directed to the appropriate components in the test pit (about 20 kW is required for the reflector control cup and shield).

Safeguards considerations indicate a potential requirement for relocation of the cooling ducts located outside the building in such a fashion as to reduce the likelihood of their rupture should an earthquake occur. The present arrangement is shown in Figure 95. Two alternatives are available for duct relocation. One would require drilling a hole in the wall between the primary sodium room and the ventilation #2 equipment space (6' 6" thick). Such a relocation would also necessitate moving the ducting and perhaps the coolers.

A second alternative is to relocate the coolers in the secondary sodium space and route the ducts through the floor to the radioactive sodium room below. Either alternative is expected to be expensive because of the size of the equipment and ducting. In either of these two arrangements, valves would be installed in the ducts to assure environment and containment control.

c. Reactivation

A number of items of FRCTF equipment would require checkout and reactivation. Both ventilation systems 1 and 2 are in this category. Further, restoration of certain spaces would be needed. For example, the hatchway between the radioactive sodium room and ventilation system #2 equipment room would have to be re-installed and/or replaced. There is a temporary shielding wall in the secondary sodium room which must be removed. The paint on the primary sodium room would have to be removed to assure that proper leak checking and repair could be achieved. A number of temporary partitions would have to be removed. The suitable operation of the radioactive waste systems would have to be assured.

d. Equipment Installation

In addition to the building modification and restoration described above, housing of certain items of equipment may be required. Among these is the effluent filter pit which has been described previously. Connection of this filter pit to the duct systems and to the containment volume would be necessary. A building to house the equipment and the filter beds may be needed. There would also be a need to house the radioactive gas filtration system. A stack must be constructed which is connected to these two systems.

e. Office Space

The office space which is provided in the FRCTF is adequate for only a portion of the personnel which are expected to be assigned to the Thermionic Reactor Experiment. Thus, additional office space will be needed. If office space is not available elsewhere in the complex, office trailers would be installed adjacent to the facility for the housing of these personnel. Approximately 1500 square feet of area is needed. Electrical power for heating and lights is required.

3. Summary

The facility capability of the Fast Reactor Core Test Facility at Los Alamos is one in which the thermionic reactor experiment could be

accommodated with a reasonable quantity of modification. The modifications which have been considered are conceptual in nature and may change as a result of additional design. It does appear, however, that all of the functions that are needed for the Thermionic Reactor Experiment can be performed in this facility. Further, the modifications which are expected to be required consist more of additions to the facility that already exist than major alterations of that which has been constructed.

N. SAFEGUARDS

1. Safety Considerations of a 3 MW_t Experiment

a. Introduction and Summary

This study represents a preliminary review of the ability of the Los Alamos Fast Reactor Core Test Facility (FRCTF) to safely contain the Thermionic Reactor Experiment (TREX). The FRCTF was originally designed to accept fast reactor experiments with sodium or NaK coolant and with molten plutonium fuel. The maximum accident envisaged for these early fast reactors was the possible loss of coolant and fuel from the reactor or external piping system. No energy release from a reactor excursion was anticipated from these experiments. Therefore, the facility was designed as a concrete structure with steel lined vaults which are filled with a nitrogen atmosphere to minimize the effects of a radioactive sodium spill. The vaults were designed for a modest leakage and internal pressure (24%/day leakage at 2.5 psi) and the ventilation system was designed to maintain low oxygen concentration (2 to 4%) and high heat removal capability ($>1 \times 10^6$ Btu/hr). Separate vaults were provided for the reactor and coolant systems. An isometric pictorial of the facility as it was originally designed and constructed is shown in Figure 99.

The TREX differs from earlier reactors anticipated for the facility in that its heat removal equipment is smaller, and the fast reactor utilizes fully enriched uranium-oxide fuel. Because the TREX heat removal equipment is small (3 MW_t) the nitrogen cooling system provided for the FRCTF is oversized, since it was designed for reactors with a thermal output of 15 MW. On the other hand, since the TREX is a fully enriched uranium-oxide fueled reactor it has a very small Doppler coefficient and energy release accidents can be postulated for it. Although these hypothetical energy release accidents have a very low probability of occurrence provision for their accommodation must be included in the reactor-facility complex. Because of this energy release possibility, and because of more stringent safeguards enforcement since the design of the FRCTF, modifications

FIGURE 99. FAST REACTOR CORE TEST FACILITY

to the facility are necessary. These modifications are discussed in more detail earlier. The principal modifications from a safety point of view include:

1. Reduced vault leakage capability.
2. Possible inclusion of a blast and heat absorbing shield around the reactor.
3. Re-routing of existing nitrogen coolant lines so that all circulating coolant piping is contained within the building.
4. Addition of isolation valves to the nitrogen ventilation and cooling system.
5. Addition of nitrogen system over-pressure vent piping, and a radio-active smoke clean-up system.

The safety review of the TREX involved a very limited study of the reactor and facility. The purpose of the review was to determine whether it is technically and economically feasible to contain TREX within a modified FRCTF. Sufficient analyses have been completed to allow a tentative conclusion concerning the technical feasibility of housing the TREX within FRCTF. This conclusion is that the facility can be modified to contain a series of extremely improbable hypothetical reactor energy release accidents and not exceed the criterion for radio-activity release as defined in 10 CFR 100. The facility can also readily accommodate a series of radioactive NaK coolant loss accidents without exceeding the radioactivity release criterion of 10 CFR 100. (Reference 22)

From the point-of-view of the reactor-facility complex, the most stringent safety requirements are imposed by the hypothetical energy release accidents. Two classes of reactor energy release accidents were considered. Both classes involved assumptions concerning the deliberate invalidation of inherent and engineered safeguards in an extremely improbable manner. In the first class of accidents reactivity insertion ramps in the reactor of up to 5\$/second were postulated. From an analysis of the resulting energy release it appears that it may be possible to

design the reactor vessel to withstand this energy release without failure, although severe strain to the vessel would ensue and the core would be destroyed. The second class of reactor energy release accident involved the postulation of reactivity insertion rates of 10\$/second or larger. The resulting energy release could be large enough to violate the integrity of the reactor vessel with the consequent release of shock, blast and thermal energy. For this released energy it may be necessary to include a blast and heat absorbing structure around the reactor. It appears possible to design such a structure for energy release resulting from reactivity ramps greater than 10\$/second but less than 100\$/second. The largest ramp hypothesized in a semi-mechanistic model of very low probability was 10\$/second.

In this very limited study it was not possible to perform transient heat transfer analyses of the structure under energy release conditions or to explore all possible reactivity addition mechanisms. Neither was it possible to exhaustively study the vessel integrity as a function of energy release. Therefore, numerous uncertainties still exist which could influence the cost of the facility, and possibly the need for certain development tests.

The principal uncertainties which remain are as follows:

1. Assurance that the reactor vessel integrity will not be violated for energy release accidents arising from reactivity insertion rates of up to 5\$/second.
2. Assurance that the upper limit of reactivity insertion rate is 10\$/second or less for all events of sufficiently high probability to warrant consideration.
3. Determination of the upper limit of shock energy and thermal energy vented from a failed reactor vessel in response to the reactivity insertion rate defined in (2).
4. Determination of the energy release-heat absorbing capability for a blast-heat absorbing structure around the reactor responsive to the requirements of (3).

b. Existing Facility Description and Proposed Changes or Additions

A description of the Fast Reactor Core Test Facility (FRCTF) at Los Alamos is contained in other documents (Ref. 23) and will not be repeated in detail here. Basically the facility consists of a building and equipment complex designed to accommodate a fast reactor. The building includes hot cell facilities, reactor equipment vaults, a radioactive coolant equipment vault, equipment handling facilities, and building ventilation equipment.

The radioactive vaults consists of:

- 1) an east reactor pit which is a 22' x 22' x 27' deep room constructed of reinforced concrete and lined with steel plate,
  - 2) a west reactor pit which is similar to the east pit,
  - 3) a radioactive coolant equipment room which is approximately 60' long x 28' wide and x 33' tall.
- The radioactive coolant equipment room is also constructed of reinforced concrete and is lined with steel plate.

The radioactive equipment vaults are interconnected and one ventilation system services them all. This ventilation system is a nitrogen coolant system which maintains the atmosphere of these rooms at a maximum of 2%-4% oxygen under conditions of air-leakage of 27% per day. The vaults are currently constructed and tested to maintain a maximum out leakage of 24% per day at an internal test pressure of 2.5 psi. It is anticipated that this leak rate can be improved to 2.5% per day at 2.5 psi by plugging up various leaks. This anticipated improvement is based upon experience with the Los Alamos UHTREX containment building where a leak of about 2% per day at 2 psig was achieved and has been maintained. The FRCTF design is, from the point of view of containment superior to that of the UHTREX.

Although the existing facility is designed and tested for 2.5 psig pressure, certain of the components can withstand substantially greater pressure than this 2.5 psig. The weakest component in the system is believed to be the blower casing for the nitrogen ventilation system. This unit is designed for a maximum pressure of approximately 4.0 psig. Most of the walls in the reactor vaults will withstand pressures of up to 11 psig or greater.

In the original design of the FRCTF it was not believed that the facilities then envisaged for installation in the building complex could experience an incident which would cause the building or ventilation equipment to exceed the design pressure or other limits which might cause this equipment to fail. For this reason the ventilation system has no means for isolating the equipment or piping from the low leakage vaults. A large piping manifold (4 ft in diameter) is external to the building and penetrates the wall of the radioactive coolant room in several places. Some of this piping leaves the radioactive coolant room through an external wall and is buried below grade in the surrounding dirt prior to re-entering the building in the ventilation equipment room.

It was originally anticipated that a fire in any one of the radioactive sodium vaults might cause a slight increase in pressure which would necessitate venting of the excess pressure through a filter system and thence up a stack. Currently neither the venting equipment nor the stack is built.

The proposed major changes to the facility are as follows:

- 1) inclusion of a refueling floor in the reactor pit and an upper reactor pit cover. This cover has equipment access plugs for the refueling machine, primary coolant system and the reactor control cup.
- 2) sealing off of the west reactor pit from the ventilation area of the radioactive coolant room and the east reactor pit.
- 3) location of the reactor flange and dome on the refueling floor with the reactor suspended beneath this floor in a shielding structure which may have blast absorbing capability.

- 4) location of the main radioactive NaK equipment in the reactor pit and the primary purification and service systems in the radioactive coolant room.
- 5) space allowance for a shielded fuel handling machine between the refueling floor and the reactor pit cover.
- 6) location of reactor control drive maintenance facilities under the reactor in the reactor pit.
- 7) provision for maintaining a nitrogen atmosphere during normal operation in the east reactor pit and radioactive coolant room in all locations.
- 8) provision for flushing the nitrogen filled locations in the reactor pit and radioactive coolant room with air during refueling or maintenance operations.
- 9) relocation of the 24 inch ventilation (over-pressure relief) pipe which now extends from the radioactive sodium room to a potential external filter location, so that this pipe extends inside of the building and to a new filter system location. The filter system will filter vented gases from the over-pressure relief system and will be located in or near the non-radioactive sodium room.
- 10) inclusion of shutoff valves in ducting to isolate the radioactive coolant room from the external ventilation system equipment.
- 11) inclusion of vent piping system to relieve over-pressure conditions in the nitrogen vaults which will exhaust through a filter system prior to discharge through a stack.
- 12) man access to the reactor control cup mechanisms below the reactor and provision for removing the control cup from the reactor pit in a shielded cask.

A preliminary equipment arrangement drawing with many of these modifications is shown in Figure 99. The refueling floor was provided for supporting the reactor vessel within the confinement of the reactor pit. Refueling operation of the reactor is carried out from the reactor floor within air atmosphere. During operation of the refueling machine containment is affected by the cover of the reactor pit which includes a sealed

located within the Radioactive Coolant Room. A preliminary piping and instrument diagram of this radioactive NaK equipment and piping is shown in Figure 77. The primary system loops are designed for normal operation of 3 MW_t and can operate at a reduced thermal output. Other changes to the facility are also proposed, but they are not pertinent to the major safety issues so they are not discussed here.

c. Overall Safety Philosophy

(1) During Normal Operation of the Reactor

During normal operation of the reactor the reactor pit and the radioactive sodium room are sealed from the outside atmosphere and the ventilation system maintains the oxygen content of these areas below 2%. Cooling is provided to maintain the ambient temperature at slightly above the supply cooling water temperature. (The cooling system is considerably oversized for the TREX equipment). The refueling hatch plug in the cover of the reactor pit is in place and sealed to this roof.

The facility is designed to accommodate three classes of accident. The first class involves the possible failure of a component or pipe in the primary system which results in the loss of coolant from that system. The second class involves hypothetical reactivity addition accidents, which although very improbable in occurrence might result in substantial core damage; and the third class involves a "design basis accident" which is a series of hypothetical events of extremely low probability in which a very large reactivity addition is assumed. The basic philosophy covering each of these classes will be briefly discussed.

Concerning first the primary system failure accident where a pipe or component fails, it is assumed that all of the NaK within the primary system piping and equipment external to the reactor is lost from the system. The reactor is designed so that in this instance the coolant over the core cannot drain from the reactor. A double vessel is provided

refueling machine hatch. An operator can leave the refueling floor through a ladder or stairway and two bulkhead type doors which lead to the main floor level. Interlocks prevent the opening of one door unless the other is closed. Interlocks also prevent refueling operations from being carried out unless the hatch plug is in place and sealed.

It is intended that all ventilation piping which penetrates the wall of the radioactive sodium room be provided with quick shutoff valves to permit isolation of this room in the event that a piping failure occurs. The possibility of ventilation piping failure is considered extremely remote, even if Los Alamos were to suffer an earthquake. Even though Los Alamos is rated as earthquake zone 2, there is evidence to suggest that there has not been a severe earthquake in thousands of years (as evidenced by unstable rock formations known as "tent rocks" located throughout the area). Furthermore, it is believed that detailed analysis of the large external coolant piping will show that failure is unlikely even with an earthquake. Despite this belief, quick shutoff valves are included for isolation purposes.

Equipment for venting excess ventilation system pressure in the radioactive coolant room and the reactor pit is proposed in order to alleviate possible problems related to the design basis accident (to be discussed later). This equipment includes pressure relief valves, interconnecting piping, absolute filters for removing radioactive NaK oxide, a charcoal trap and a discharge stack. The filters and charcoal trap are located on one of the upper floors of the reactor building.

The radioactive NaK equipment which is included both within the reactor pit and the Radioactive Coolant Room includes an intermediate heat exchanger, a heater, piping and primary system pump for removal of reactor heat during normal operation. Also included is an auxiliary heat removal loop for removing decay heat from the reactor during emergency conditions. A dump tank is provided for dumping the NaK from the primary system and as a sink for the reactor overflow level control. Cleanup equipment for maintaining the purity of the primary NaK is also

in the core region of the reactor so that a leak in one vessel will merely drain into the second vessel without uncovering the core, and the primary piping enters and leaves the vessel above the core region. An auxiliary coolant loop is provided to remove the decay heat from the reactor by natural convection through the auxiliary intermediate heat exchanger to a secondary NaK system and thence to an external air heat dump exchanger. No electrical power is necessary to operate this auxiliary cooling loop.

Coolant which escapes from the defective component or pipe spills into steel catch pans, which are located beneath the equipment and which limit the surface area of the spilled NaK. Spray shields around the components and piping prevent a leak from dispersing NaK beyond the boundaries of the catch pan under the equipment or into the pit or room steel liners. As will be shown later, if the ventilation system for the room is operative under the conditions just described, no pressure rise in the room occurs and no radioactive materials leak from the room. If the ventilation equipment is assumed to be inoperative, a modest pressure rise occurs (less than 1 psi) and a very small leakage from the building can occur. This leakage is insufficient to cause difficulty in terms of the limits established in 10CFR100 (Reference 22) for radioactivity release.

The second class of accidents involving modest reactivity addition accidents (from 1 to 5\$ per second) can result in rather large energy releases within the core. It should be emphasized that no consistent mechanisms can be envisaged for inserting reactivity additions of this magnitude. Nevertheless they were assumed to occur and the resultant energy release was computed. This energy release will manifest itself as shock energy, blast or pressure energy, and as heat. It appears that it will be possible to design the vessel

to accommodate all of these manifestations of energy release without destruction of the vessel, although the upper limit of reactivity insertion which can be tolerated may be somewhat less than 5\$/second. Therefore, for this class of accident although the reactor core is all or partially destroyed, vessel integrity is maintained and coolant continuity is sustained. Decay heat from the fuel remnants can be removed by means of the heat removal system. Fission products are not released to the pit atmosphere and there is no increase in temperature or pressure of the vault atmosphere.

The third class of accident is an extremely improbable series of hypothetical events which are designated as the design basis accident (DBA). No credible mechanisms can be postulated which will lead to this accident. It consists primarily of determining the largest reactivity component which can be inserted by a pseudo mechanistic approach and assuming that this event occurs even though the mechanisms of obtaining the insertion violate numerous engineered safeguards and natural laws.

Insertion of the large reactivity ramp rate in the DBA results in a sufficiently large energy release that the integrity of the reactor vessel may be violated. From the limited analyses performed for this safety review it appears that it may be possible to design the reactor vessel so that it can accept the largest reactivity ramp hypothesized without violation of its integrity. In this event, the DBA would be similar in effect to the second class of accident previously discussed. However considerably more work will be required before it would be possible to depend solely upon the vessel to contain the DBA. For that reason, it has been assumed that additional measures may be required to contain the DBA. These measures would include a shock and blast shield which surrounds the reactor vessel. Any shock and blast energy which escapes from the ruptured vessel is absorbed in the shock and blast shield which

surrounds the vessel. Heat energy is also dumped into the atmosphere within the blast shield and this causes a rapid rise in pressure of the gas, which slowly escapes through small passages around piping and components. Heat sinks within the blast structure absorb most of this heat energy, but some heat and gas escapes to the vault atmosphere and causes a rise in pressure. This rise in pressure is within the design pressure limit of the vault. Following the initial pressure surge, it is assumed that the primary coolant system also fails and spills its contents into the vault. The vaults and ventilation systems are designed to accept the consequences of all of these events.

The design basis accident is not anticipated to result in any damage to the nitrogen ventilation and cooling system equipment and therefore this accident results in a release of radioactive products which is considerably below the limits established by 10CFR100. (Reference 22) Even with the cooling system inoperative, the radioactive release limits are not exceeded.

To summarize the safety philosophy for the three classes of accident postulated during normal operation of the reactor, the reactor and building complex are designed so that under all of the mechanistic and non-mechanistic models postulated for unlikely accident conditions, the integrity of the FRCTF will not be impaired and radioactivity release is acceptably low.

It should be noted that maintenance of the FRCTF integrity is dependent upon maintaining the integrity of the steel liner which covers the concrete in the vaults. Two circumstances could result in stressing this liner. One circumstance involves the release of hot NaK onto the relatively cold steel plate causing it to warp severely and possibly fail. To prevent such rapid thermal shocks

the drip pans and splash shields are provided so that hot NaK from a failed component will not have direct access to the steel liner. Transfer of heat through the nitrogen atmosphere of the vault will of course occur, but this will be over an extended time period and should not result in the failure of the liner although some local yielding could occur. The second event which might cause some local stressing involves possible earthquake loading. Here, however, the massive concrete structure to which the liner is attached should readily absorb any earthquake loads with a minimum of stress being transferred to the liner. It should be possible therefore to demonstrate that liner integrity will be maintained under all postulated conditions.

To cover the possibility that an earthquake might impose high stresses on the nitrogen ventilation system piping, the existing piping will be re-routed into the building and designed for expected earthquake loads. Additionally isolation valves will be included at the point of juncture of this piping with the radioactive NaK room wall. Vent system piping which leaves the building will be designed to minimize earthquake stresses and will include an isolation valve.

(2) During Refueling Operations

A major criterion imposed upon the refueling sequence is that refueling operations will never be carried out unless the reflector control mechanisms are in their fully retracted positions below the reactor core. This assures that the reactor is 18 \$ subcritical and cannot undergo any reactivity accidents through any credible sequence of events. Thus the only situation assumed to have sufficient probability of releasing fission products to be considered results from a hypothetical NaK fire.

During refueling of the reactor the atmosphere in the refueling room will be refilled with air. Personnel will

operate equipment and the refueling machine within this space. Prior to installation of the refueling machine it is necessary to remove a cover flange over the top of the valve which allows access to the TFE handles over the reactor. Subsequent to removal of this flange the refueling machine will be placed on the valve and sealed, and the atmosphere between the refueling machine and the valve seat will be purged and filled with an inert gas. A valve is also located in the refueling machine. Upon assurance of the seal integrity both the valve in the refueling machine and in the reactor will be opened and fuel can be raised into the refueling machine. During this period it is possible that the seal between the refueling machine and the reactor valve could fail. The normal procedure under this circumstance would be to continue to supply the reactor and refueling machine with inert gas and to shut the refueling machine and reactor dome isolation valves. However on the assumption that this may not be done it is conceivable that air could get into the reactor and a small NaK fire ensue. This would result in the release of radioactive NaK smoke to the room over the refueling floor. The operator could escape through the double door personnel access to the operating floor. The refueling room could then be flooded with nitrogen and the reactor vessel flooded with helium or argon to replace any helium lost from the broken seal, thus extinguishing the fire in the NaK. The escaping fission products under these circumstances are expected to be less than for the accidents described during normal operation of the reactor. In no instance will the refueling machine be used on the reactor unless the hatch plug is in place and closed.

It should be noted that removal of the refueling machine by means of the building crane to the operating floor can only be affected through a hatch which is located approximately eight feet horizontally distant from the reactor. For removal by the crane it is transported to the hatch on a rail system. Thus if the machine falls it cannot fall upon the reactor head.

In summary the safety philosophy during refueling operations is to assure that the reactor is always in a condition such that a reactivity addition accident cannot occur and to assure that a containment barrier always exists over the reactor and refueling machine which cannot be violated under any foreseeable circumstance and is adequate to cope with a hypothetical fire situation.

(3) During Maintenance Operations

It is anticipated that some maintenance operations may be necessary on portions of the primary system coolant equipment or upon reactor control mechanisms. Prior to carrying out any such maintenance a basic safety requirement is that the control mechanisms be moved to the retracted "maintenance" position below the reactor which makes the reactor 18 \$ subcritical. For coolant system maintenance the reactor primary coolant will be drained to the coolant dump tank. Valves isolating this tank from the primary system will be closed during such maintenance. Sufficient fuel will be transferred from the reactor prior to the maintenance so that the decay heat can be radiated from the reactor vessel without assistance of the auxiliary cooling system should it become inoperative. The excess fuel will be stored in the storage pit external to the reactor pit. Activated rods in the reactor pit storage are shielded so they will not have to be removed. During maintenance the pit will be flooded with air and personnel can enter to work. All maintenance carried out in this manner will require a carefully documented plan prior to approval of the repair program.

d. NaK Fires

A consideration in the adequacy of the FRCTF is the manner in which the facility responds to a loss of coolant accident. In the event that coolant is lost from the primary system by the failure of a component or of the piping, heat is transferred from the spilled coolant into the vaults. If the accident is of the type which assumes that the initiating cause is a primary system component failure, then the sources of energy for transferring heat to the vault are the stored energy in the coolant and the reaction energy of the NaK and the oxygen in the cell atmosphere. If the accident is of the type where the initiating cause is a reactivity addition followed by energy release in the reactor and failure of the vessel, then two additional sources of energy are liberated. These additional sources are the energy from the reactivity addition accident which is not absorbed as shock energy and the decay heat of the fuel.

To determine the magnitude of the pressure and temperature-time history in the vault atmosphere after one of these accidents, use was made of the Los Alamos program "SPILL". This program is described in Reference 24. Basically the program is a series of material and energy balance expressions which take into account the geometry of the vaults, the oxygen concentration in the cell atmosphere, the heat removal capability of the nitrogen cooling system, the leak rate from the vault, the volume and area of the NaK spilled, the heat capacity of equipment in the vault and the controlled maximum pressure of the vault atmosphere. In addition to determining the pressure and temperature history of the vault the program also computes the fractional leakage from the vault as a function of time.

The burning rate of SPILL is a function of oxygen concentration of the cell and is based upon test data obtained in the burning of sodium fires at Atomics International. A similar program at Atomics International entitled "SOFIRE" has confirmed the validity of this burning rate, $0.4 \text{ lb of Na/h-ft}^2\text{-}\% \text{ O}_2$ (Reference 25). The SPILL program makes two simplifying assumptions which result in an overestimate of the pressure in the cell. The first assumption is that all of the energy of combustion is liberated

into the cell atmosphere. Tests at Atomics International have indicated that a substantial amount of the combustion energy is radiated and conducted to the pool of liquid metal from which the gaseous products are being vaporized to support combustion. This radiated and conducted energy fraction is as high as 75% of the total energy. The second simplifying assumption is that the only heat sink for removal of energy from the nitrogen atmosphere of the vaults is the steel liner which covers the concrete and the steel drip pans into which the NaK is spilled. No allowance for the transfer of heat from the steel liner to the concrete is made. This results in a reasonable approximation of the cell pressure rise in the early periods of the accident (up to one hour), but after that period of time this assumption results in a gross overestimate of the pressure of the cell. Indeed since the leakage rate of gas from the cell is so small the gradual transfer of heat from a steel container (which has NaK in it) back into the cell atmosphere can result in an increase in the vault pressure after an initial period in which the pressure decreases. A more correct representation which included the transfer of heat from the steel to the concrete would undoubtedly show a substantial decrease in pressure in the cell as a function of time after the initial pressure rise.

To determine the consequences of various accident conditions on the atmosphere in the vaults several cases were selected for input into the program SPILL. These cases are shown in Table 40. In all of the cases it is assumed that the cell leak rate has been improved so that it is proportional to cell pressure and leaks 2.5% per day at 2.5 psig. The volume of NaK spilled in each case represents an upper limit of the amount of NaK available for loss under the assumed accident conditions. Similarly the area of the NaK spill represents an upper limit of the available area in the spill pan. The initial oxygen concentration in most cases is assumed to be 2% for the nitrogen cleanup system.

TABLE 40

INPUT DATA FOR NaK FIRES

Case No.	1	2	3	4	5	6
Flowrate thru cooler, WV, ft ³ h x 10 ⁻⁶	0	0	3.18	3.18	0	3.18
Leak rate, LP %/day @ 2.5 psig	2.5	2.5	2.5	2.5	2.5	2.5
Vol. of NaK spilled, VNAK, ft ³	46.0	46.0	46.0	46.0	46.0	46.0
Area of NaK spilled, AI, ft ²	100.0	100.0	100.0	300.0	100.0	100.0
Initial O ₂ Concent., COI, mole fract.	0.20	0.20	0.20	0.20	0.20	0.20
Vent valve control press, DPC, psi	100.0	0.72	100.0	0.72	100.0	0.72
Thermal capacity of spill pan, XTC, Btu/°F	584	584	584	584	2336	584
Type of NaK spilled; 1=NaK 44	1	1	1	1	1	1
Initial Temp. of NaK, Y(3), °F	1200	1200	1200	1200	1200	1200
Initial heat dump, Cs, 10 ³ Btu	0	0	0	0	0	0

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All cases:

Free vault volume - 57,900 ft³

Mass of steel in vault - 163,000 lb.

In the first case it is assumed that the nitrogen cooling system is inoperative and there is no vent system (or the vent system is inoperative) for venting excess pressure from the cell through the radioactive scrubber and filtering system. Under these conditions any leakage of radioactive products would be directly from the vaults at ground level. From the results of the computer run it is seen that a pressure of 0.77 psig occurs after about one hour. At the end of one hour 0.03 moles have leaked from the vault out of a total of 104 moles. This represents a leakage of approximately 0.029% within the one hour period. The computer program indicates that a maximum pressure of 1.58 psig was reached after 30 hours. For the reasons discussed previously concerning the conservative nature of the assumptions used in SPILL, it is believed that this pressure is over conservative and in fact the pressure would be less than 0.83 psig after 20 hours due to the transfer of heat from the cell liners to the concrete. Other analyses at Atomics International have shown that this pressure decay is on the order of a factor of 2 within a 20 hour period. This would result in a cell pressure of approximately 0.4 psig after 20 hours. The total leakage at the end of the 20 hour period for the more correct reduced pressure would be something less than 0.6 moles instead of the 1 mole indicated in the computer printout. The total fraction leaked at the end of one day, therefore, would be less than 1/2% of the total volume of the vaults. This fractional release rate would also be decaying with time.

Case #2 shown in Table 40 is similar to case #1 except it is assumed that a vent valve opens when the pressure reaches 0.72 psig and closes when the pressure drops below this value. Under these conditions the pressure has increased to a maximum of 0.72 psig within 0.3 hours at which point the vent valve opens and vents the excess pressure and closes again after the pressure is reduced. At the end of a one hour period the vault pressure

is approximately 0.5 psig and a total of 0.02 volume percent has leaked from the vault and 1.7 volume percent has been vented through the radioactive vent system.

The third case is similar to the first two except that the nitrogen cooling system is assumed to be operating at its original design capacity. This cooling system is oversized for the requirements of the TREX. Therefore, under these conditions the cell pressure did not rise at all and there was zero leakage from the cell.

In the fourth case it was assumed that the area for oxidation of NaK was increased arbitrarily by a factor of 3 even though the initial area assumptions are considered conservative. Again in this case as in case #3 the cooling system imposed a zero pressure rise in the cell atmosphere and zero leakage from the cell resulted.

In the fifth case it was assumed that the cooler was inoperative as in case #1 but the thermal capacity of the spill pins which collect the spilled NaK was arbitrarily increased by a factor of 3. This resulted in a slight decrease in the pressure rise of the cell compared with case #1.

In the sixth case it was assumed that the nitrogen cooling system was operating at its original design capacity and that air was present initially in the cell when the spill occurred. The nitrogen cleanup system to minimize oxygen was also operating at design capacity. Under these conditions the maximum pressure in the cell occurred at 0.05 hours and was 0.51 psig. Because of the capability of the nitrogen cooling system the pressure rapidly decreased below this value and became zero at 1/2 hour. The total leakage from the cell was approximately 0.006 volume percent.

e. Hypothetical Reactor Accidents

As noted previously two classes of reactivity addition accidents were considered. Both classes are hypothetical in nature and both necessarily violate engineered safeguards designed to prevent their occurrence and in some cases violate natural laws (such as the requirement that material of a given density and mass must occupy a fixed volume). The first class of accident is typified by a modest reactivity addition (from 1 to 5\$ per second). The second class of accident involves reactivity additions which are much larger and are between 10 and 100\$ per second.

Although no mechanistic model which is completely consistent can be devised to obtain reactivity addition rates as large as 5\$ per second, numerous hypothetical events can be conceived which might add modest amounts of reactivity in a short period of time. It was therefore assumed that although such events have a very low probability of occurrence they are possible and the reactor-facility complex should be designed to accept them. Using the modified Bethe-Tait type of analysis described in GEAP-4809 (Reference 26), the energy release for reactivity addition rates of from 1 to 5\$ per second were calculated. For a reactivity addition rate of 5\$ per second the total energy was computed to be about 10 MW-sec and this corresponds to approximately 5 pounds of TNT. Using methods developed by the Naval Ordnance Laboratory the shock and blast effects were determined. These effects are transmitted to the reactor and vessel structures and result in the generation of internal missiles and of strain to the reactor vessel. The work energy from this magnitude of energy release is totally absorbed within the structure. It appears that the vessel can be designed to withstand this absorption without failure.

From the above it is seen that for the class of accidents which accepts the possibility of a moderate reactivity addition, no energy or fission products are released to the vault atmosphere and no radioactivity hazard exists.

For the second class of accident very large reactivity addition rates are assumed to occur (greater than 10\$ per second but less than 100\$ per second). The mechanics for inserting these large reactivity rates are extremely improbable. Indeed no rationally consistent method can be determined which will result in this type of incident. Substantially more effort will be required to demonstrate that 10\$/sec is the upper limit of all accidents which have a sufficiently high probability to warrant consideration.

Using the same methods as discussed for the moderate reactivity additions a total energy release for 10\$ per second was computed to be about 680 MW-sec. The available work from this energy release is between 15 and 85 MW-sec and corresponds to a TNT equivalent of 8 to 40 pounds. For reactivity addition rates approaching 100\$ per second the total energy release approaches 1,550 MW-sec and the equivalent TNT is between 40 and 200 pounds. All of these values are sufficiently large that they can cause failure of the reactor vessel. Subsequent to the failure of the vessel, shock energy, blast or pressure energy and heat are released from the vessel. Surrounding the reactor vessel is shielding material and a shock absorber. This shock absorber may be constructed in any number of ways such as the laminated structure used on the EBWR (Reference 27). The shock energy released from this magnitude of an excursion can be absorbed in a properly designed shield. In addition to the shock energy (in the form of pressure waves traveling at sonic velocity), the atmosphere contained within the confines of the shock shield will be raised within a short period of time to high pressures (perhaps on the order of 200 psia), and very high temperatures will exist from vaporized NaK, fuel, and steel. For a total energy release of around 680 MW-sec the equivalent heat energy of perhaps 200 MW-sec or approximately 200,000 Btu might be available in a fairly mobile form as vaporized or liquid materials. For an energy release of 1,500 MW-sec perhaps 700 MW-sec or approximately 700,000 Btu might be available with a high degree of mobility. If this heat energy were to be deposited in the reactor vault atmosphere it would cause an immediate rise in pressure which would exceed the capability of the facility to withstand that pressure.

Therefore, the blast shield must be designed to contain the heat energy for a long enough period of time to absorb significant portions of it in heat sinks. These heat sinks might be in the form of large steel sections with a geometry arranged to facilitate heat transfer. For example, a cylindrical steel slab which is 2' thick and surrounds the reactor vessel and experiences a temperature rise of 500°F could absorb 1,000,000 Btu. It is believed that the blast structure can be designed to withstand a relatively high pressure and not vent large fractions of the heat energy for a long enough period of time so that most of the energy can be absorbed in heat sinks. The determination of the upper limit of heat absorbing capability of the structure must await detailed transient heat transfer calculations and possibly a development-test program. It appears however that the limit for the structure will be found for an energy release corresponding to reactivity addition rates which are greater than 10\$ per second and less than 100\$ per second. If, the upper limit of reactivity addition for the DBA is later demonstrated to be 10\$/second or less, then the design of the blast and heat absorbing structures will be considerably simplified.

For the design basis accident, which will be within the reactivity addition limits noted above, it is assumed that some energy will escape from the blast shield. To stay within the limits of the design pressure of the vaults and ventilation system equipment (2.5 psig) this heat energy must be less than 90,000 Btu. To determine the effect of dumping some fraction of this heat into the vaults a series of SPILL cases were prepared in which 46,000 Btu were arbitrarily dumped into the vault atmosphere within a very short period of time. It was assumed that this was heat energy which had escaped from the blast shield subsequent to rupture of the reactor vessel. It was also assumed that simultaneous with the failure of the vessel the primary piping system failed and leaked the entire content of the NaK to one of the spill pans where it transferred its heat to the cell atmosphere and reacted with the 2% oxygen within the cell. Four cases were prepared for consideration and they are shown in Table 41 as cases 7, 8, 9 and 10. In case #7 it was assumed that the ventilation and

TABLE 41

INPUT DATA FOR HYPOTHETICAL REACTOR ACCIDENTS

<u>Case No.</u>	7	8	9	10
Flowrate thru cooler, WV, ft ³ /h x 10 ⁻⁶	0	3.18	0	1.0
Leak rate, LP, %/day @ 2.5 psig	2.5	2.5	2.5	2.5
Vol. of NaK spilled, VNAK, ft ³	43.0	43.0	43.0	43.0
Area of NaK spilled, AI, ft ²	100.0	100.0	100.0	100.0
Initial O ₂ Concent., COI, mole fract.	0.02	0.02	0.02	0.02
Vent valve control press, DPC, psi	0.72	0.72	100	0.72
Thermal capacity of spill pan, XTC, Btu/°F	584	584	584	584
Type of NaK spilled; 1=Nak 44	1	1	1	1
Initial temp. of NaK, Y(3), °F	1100	1100	1100	1100
Initial heat dump, * Cs, 10 ³ Btu	46.3	46.3	46.3	46.3

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*Inserted into vault atmosphere within less than 10 seconds.

All cases:

Free vault volume — 57,900 ft³

Mass of steel in vault — 163,000 lb.

cooling system were inoperative but that an over pressure vent system was functionally operative. From the results it can be seen that the maximum pressure of 1.27 psig is obtained almost immediately. This results in the opening of the vent valve and the release 2.91 moles or 2.8 volume percent through the radioactive scrubber system. The pressure within the first hour drops to 0.43 psig and the leakage through the first hour is approximately 0.02%. Although the computer program shows the pressure continuing to rise with time, for the reasons discussed previously, this is believed to be due to the over conservative heat transfer model in SPILL and, in fact, the pressure would decrease after approximately the first hour.

For case #8, the same conditions as case #7 were assumed except that the ventilation cooling system was assumed to be operating. In this instance the maximum pressure obtained was 0.83 psig at 0.014 hours which resulted in a leakage of 0.0003 moles after which the pressure decreased to zero because of the heat being removed by the cooling system.

In case #9 both the cooling system and the vent valves were assumed to be inoperative. Under these conditions the maximum pressure was 1.64 psig at approximately 0.014 hours and this decreased to 0.78 psig after one hour. The total quantity leaked in one hour was approximately 0.03 volume percent. Subsequent to the one hour period the pressure is seen to rise again. This pressure rise is believed to be erroneous.

For case #10 similar conditions to case #8 were assumed except that the cooling capacity of the nitrogen system was reduced by two-thirds. Under these circumstances the maximum pressure obtained was 1.13 psig at 0.01 hours after which the pressure dropped to zero. The total fraction leaked was 0.001% and the total quantity vented was approximately 1%.

From the above data it is seen that for the assumed conditions the vaults can readily accept the design basis accident and maintain a very low leakage fraction even if the ventilation cooling system and vent valve are assumed to be inoperative. For those cases in which the cooling system is operative the leakage is reduced essentially to zero immediately following the initial pressure pulse. Providing that the blast shield and heat absorbing sinks are properly designed therefore, it is concluded that the FRCTF can safely contain a design basis accident of the magnitude discussed above.

f. Radioactivity Release from NaK Fires and Hypothetical Accident Conditions

Computations were made of the radioactivity released from the various NaK fires and hypothetical reactivity addition accidents previously discussed. In each of the cases calculations were made of the thyroid dose, the lung dose and the bone dose using the Sutton diffusion equations and other techniques described in Reference 28 .

Table 42 shows the results of the computations. Each of the cases shown corresponds to the same case numbers previously discussed under NaK fires and hypothetical reactor accidents. All of the doses are cumulative doses at a distance of 3/4 of a mile from the FRCTF. For the first six cases the fractional fission product inventory assumed to be available for leakage from the building is 1% of the total core inventory. For cases #7 through #10 the available fission product inventory is assumed to be 2%. The efficiency for removal with the scrubber of iodine and NaK is assumed to be 99.99% for iodine and 99.88% for NaK. These efficiencies were demonstrated in a test of a sodium-fire wet scrubber system at Atomics International (Reference 29). In addition to the wet scrubber or filter there is the charcoal filter which is downstream from the scrubber and which also has a high efficiency for the removal of iodine.

TABLE 42

RADIOACTIVITY RELEASE RESULTS
(ACCUMULATED DOSE IN REM AT 3/4 MILE FROM FRCTF)

<u>Case No.*</u>	1	2	3	4	5	6	7	8	9	10	11
Thyroid Dose	0.6	0.4	0	0	0.28	0.023	9.8	0.058	17.5	0.2	165
Lung Dose	0.9	0.64	0	0	0.47	0.032	0.7	Negl.	1.2	Negl.	11.5
Bone Dose:	Approximately 1/6 of Lung Dose.										

*Cases correspond to the same case numbers identified in Tables 40 and 41.

From an examination of Table 42 it is seen that all of the first ten cases are substantially below the 25 rem whole body radioactive dose or 300 rem thyroid dose specified as a limit in 10CFR100. (Ref. 22) Thus, all of the classes of accidents previously considered, including the DBA, fall within the prescribed limits of 10CFR100. (Reference 22)

It is recognized that circumstances could occur such that the scrubber and filtering system would work at reduced efficiency. Moreover, it is conceivable that a higher fraction of fission products could be released to the building and vented through the vent system than was assumed in previous cases. To obtain an upper limit on this value case #11 was computed. In this case 100% of the available iodine and 5% of the other fission products in the reactor core were assumed to be available for release from the vault atmosphere. It was further assumed that instead of the very low total vent fractions computed for the design basis accident and other NaK fire accidents, 30% of the building volume was vented through the scrubber and filtering system which operated at only 90% efficiency. The remaining fission products were discharged directly from the stack to the atmosphere. Under these conditions the total dose to the thyroid $3/4$ of a mile from the facility is 165 rem and the total lung dose from gross fission products is 11.5 rem. Because of the extremely conservative assumptions used in this calculation the probability of occurrence is believed to be sufficiently low as to not be considered a part of the design basis accident. Even under these circumstances however it is seen that the expected dose is within the 300 rem limit for thyroid and the 25 rem total body dose established in 10CFR100. (Reference 22)

With regard to the radioactivity release values, it should be noted that the computations were based upon a person remaining in the passing radioactive cloud at the $3/4$ mile distance for the entire period of its passage. This meets the requirement for a "Low Population Zone" described in 10CFR100. (Ref. 22) Actually, the population within this radius consists of technical personnel on the laboratory site (approximately 400 persons) and a residential trailer court of approximately 250 persons. In an emergency, evacuation procedures could be carried out so that the 2 hour

exposure limits of an "Exclusion Area" could apply. This would allow increased release fractions to apply.

2. Reactivity Addition Models

a) Introduction

A brief discussion of a number of specific features and characteristics of the Thermionic Reactor Experiment (TREX) which can influence the rate and extent of unintended reactivity insertion is presented in order to provide a basis for more detailed discussions. The core internals consist of a core structure of 1.130 inch diameter stainless steel process tubes held within a core shroud structure. A pattern of 211 tubes is arranged on a triangular pitch of 1.133 inches. The reference loading has 163 fuel element positions encircled by a ring of 48 locations containing stainless steel clad BeO reflector rods (See Figure 100). Axial reflector material is provided by means of BeO segments attached to the top and bottom of the fuel element rods.

This assemblage is contained within a double walled reactor vessel. Inlet and outlet coolant pipes are provided in the side of the vessel wall well above the level of the core. The core shroud and flow baffles above the core region provide for the coolant flow to be directed downward from the inlet flow plenum between the core shroud and the reactor vessel. The flow is turned at the bottom of the reactor vessel and is then directed upward through the core into the exit plenum region.

FIGURE 100. REACTOR ASSEMBLY SCHEMATIC RADIAL SECTION

The reactor coolant system has been designed in such a fashion that cooling is always provided for the reactor core. In the event of pump failure, natural convection can remove the core after heat. A rupture of the primary coolant system can prevent natural convection cooling, but such rupture will never drain the reactor vessel below the coolant inlet plenum. Loss of coolant by rupture of the pressure vessel is only a remote possibility as a result of the double walled vessel construction. Cooling is provided in two ways in the event of a primary coolant system rupture. A small natural convection emergency cooling system is incorporated. In the event of failure of the emergency cooling loop also, the reactor vessel can reject the after-heat to the cell atmosphere if the reflector cup is removed and cell temperatures are maintained near design levels.

These cooling provisions are a fundamental and important aspect of the TREX safety since core disassembly with subsequent rapid reassembly into a prompt critical configuration cannot take place in the presence of adequate cooling. Core meltdowns represent the only means by which high reactivity insertion rates may be achieved in the TREX so that the probability of such an event is reduced to very low levels by this approach to coolant system design.

In addition to the coolant system reliability which minimizes the probability of a reactivity addition accident, instrumentation is provided which is designed to detect early malfunctions of the reactor and prevent propagation of local disturbances. This instrumentation includes nuclear sensors which scram the reactor on short period or high power, primary coolant system temperature and flow sensors which scram the reactor under adverse conditions. Table 43, 44, 45 show possible safety system response to various malfunctions.

In the discussion of reactivity insertion mechanisms which follows, a discussion of mechanisms for reactivity insertion which do not involve loss of coolant or core rearrangement will be discussed.

TABLE 43 TREX SAFETY SYSTEM RESPONSES

System	Trouble Signal	Trip Set	Logic	Actions		
				1	2	3
Nuclear Instrumentation	Malfunction (SRM)		1/2	X		
	Low Flux (SRM)	10% F.S.	1/2	X		
	Short Period (SRM)	15 sec.	1/2	X		
	Malfunction (IRM)		2/3			X
	Short Period (IRM)	15 sec.	2/3	X		
	Short Period (IRM)	10 sec.	2/3			X
	Malfunction (WRM)					X
	Low Flux (WRM)	5% F.S.	2/3			
	High Flux (WRM)	105% P	2/3	X		
	High FLUX (WRM)	110% P	2/3			X
Electrical Power	Loss of Power to Ref. Drive Motor					X
	Loss of Power to Primary Pump					X
	Loss of Power to Heat Dump Blower					X
	Loss of Power to Building					X
Primary Coolant	Low Level in Exp. Tank					X
	High Level in Exp. Tank					X
	Low Temp. at Core Outlet	770°K				X
	High Temp. at Core Outlet	910°K				X
	Low Primary Flow	80% S	1/1			X
	High Temp. in Primary Pump Winding					X
	High Activity					X
	High Reactor Dome Pressure					X

*Actions

1. Reactivity Insertion Block
2. Rundown
3. Scram

Notes

F.S. — Full Scale
S — Set Point
P — Full Power

TABLE 44 TREX SAFETY SYSTEM RESPONSES

System	Trouble Signal	Trip Set	Logic	Actions		
				1	2	3
Secondary Coolant	Low Level in Exp. Tank	790°K	1/1			X
	High Level in Exp. Tank					X
	High Temp at Heat Dump Exit					X
	Low Temp. at Heat Dump Exit	770°K				X
	Low Secondary Flow	80% S				X
	High Temp. in Sec. Pump Windings					X
	High Activity					X
Heat Dump	Low Flow	80%	1/1			X
TFEs	Low Current	80%S		Scram if TFE is to be saved. Otherwise no action.		
	Low Cesium Reservoir Hot Channel	S-20°K				
Driver Fuel	Hot Channel					X
Power Condition	High Temperature	350°K			X	
Reactor Vessel and Cavity	High Reflector Temp. High Vessel Temp. NaK between Double Wall					X

*Actions

1. Reactivity Insertion Block
2. Rundown
3. Scram

Notes

F.S. — Full Scale
S — Set Point
P — Full Power

Notes

- F.S. - Full Scale
S — Set Point
P — Full Power

The discussion will then turn to examining the means by which core rearrangements can take place and the consequences of such events.

b) Reactivity Insertion Mechanisms with Reactor Integrity

Several possible methods for unintended reactivity insertion exists for small fast reactors of the TREX type. These will be discussed individually in this section.

(1) Core Compaction

Compaction of the TREX core is greatly limited by the incorporation of the process tube structure in the reactor core. Some degree of compaction could take place, however, since a 3 mil clearance is provided between process tubes and a 40 mil coolant annulus is provided between the fuel element and the process tube inner surface. Compaction of the 3 mil tolerance between process tubes is a cumulative compaction effect. Calculations show, however, that only 3¢ in reactivity will result from the full compaction. The 40 mil coolant annulus only allows motion of the fuel element locally about a fixed reference point so that cumulative motion inward is not possible. In addition, the fuel element is restrained in the process tube so that total motion is restricted to much less than the potential 80 mils. Calculations have not been performed for this effect at this time, but the reactivity effect is believed to be only a few cents.

Axial compaction of the core is also strongly restrained. In the case of the TFEs, some compaction can take place either by emitters falling onto the collector and disk (about 20 mils non-cumulative motion) or by fuel slumping within the emitter. Fuel slumping cannot occur to any significant extent as a consequence of the segmentation of the fuel into tungsten cans and the very high loading fractions of the emitters.

In the NTFEs and DFEs, the potential for extensive fuel slumping conceptually exists, but the specific design approach involves fully loading these elements with fuel so that significant slumping is physically impossible on the basis of lack of available volume for slumping to occur. Calculations have not been

done at this time for such axial compaction effects, but for the TREX fuel elements there appears to be very little reactivity available from such effects, almost certainly much less than one dollar.

(2) Reflector Compaction

Motion of internal reflector material is restricted in much the same manner as fuel motion so that large reactivity effects from the motion of such material are not available. The external radial reflector is designed with clearances between the reflector cup and the reactor vessel and with clearances between the drums and the channels in which they rotate. Calculations indicate, however, that about 1/3 inch of inward motion by the external reflector material would be required to produce a \$1 reactivity insertion. Within the range of design clearances, a motion of this magnitude is impossible unless extremely large forces are applied which exceed the strength of the structures involved. Practical reactivity insertions from this source are probably limited to the range of 10¢ to 30¢.

(3) Cold Coolant Insertion

The reference design reactor loaded with NTFEs has a cold to operating reactivity loss of over \$3. For this reason, if cold NaK is suddenly introduced into the core without corrective action from the control system, a prompt critical insertion of reactivity is conceptually possible. Such an event is an extremely remote possibility, however, since no supply of cold NaK exists any where in the primary loop. The reactor coolant overflow system is not a real source of cold NaK as a consequence of the low flow rates and the high normal operating temperature of the primary overflow system.

Even in the event of introducing cold NaK in large quantities, however, the insertion of the available reactivity would require cool down of the fuel elements and the core structure. This cool down process would take significant time so that rapid reactivity insertion rates ($> \$1/\text{sec}$) would not be possible.

A full assessment of the actual reactivity effects of introducing cold NaK will require a transient simulation. Detailed studies will be required to determine actual reactivity insertion rates and reactor responses. It is quite likely that no physically possible event of this type could produce a prompt critical situation for the TREX design.

(4) Coolant Voiding Events

Reactivity effects associated with coolant voiding are all negative for the TREX core so that voiding incident cannot cause a reactor excursion.

(5) Control System Failure

If a complete failure of the reactor control system is postulated so that the full reactivity available in the control system is inserted at the maximum rate capability of the drive mechanisms with the scram, setback and reactivity insertion block systems, all inoperative, several types of excursion are possible. If this type of event were to happen during cold criticality experiments, much more than prompt criticality is available. The maximum rate of reactivity insertion is on the order of $5\text{¢}/\text{second}$, however, so that the resulting excursion should have a relatively small explosive energy release.

If this same type of accident is considered for a reactor operating at design conditions and prompt critical reactivity is available, a similarly limited excursion will take place with less total energy release involved as a result of the higher power level at which the event is initiated. It is quite possible that prompt critical reactivity is not available at the normal reactor operating point. Nevertheless, at the minimum power level, a large fraction of β will be available so that very high reactor power levels could be achieved even in the absence of prompt critical reactivity. High power excursions would lead to boiling of the coolant, which in turn would lead to core meltdown depending on the amount of reactivity available.

Failure of the control system in the manner assumed would lead to serious core damage for each of the three situations considered. The prompt critical excursion would, in fact, result in low intensity explosions. These excursions would not have enough energy to rupture the reactor vessel and would be contained. The fuel would be in a molten and vaporized state, however, and the reactor internals would have suffered a nearly complete loss of integrity.

Upon completion of the prompt critical event, the core material would still be within the reactor vessel and would presumably be available for reassembly in the lower portions of the reactor vessel. The core material would either fall to the bottom as fragmented pieces of random size and shape, or as molten material. The violence of the excursion would, however, widely disperse the UO_2 fuel, particularly into the regions above the core, so that reassembly into a critical configuration is unlikely. Critical configurations are considered unlikely on the basis that very compact and well-reflected arrangements of the UO_2 and other core materials are required if criticality is to be achieved.

The slower excursions associated with sub-prompt critical insertions would not result in dispersion of the fuel material, however, and it is an event of this nature which potentially represents the greatest hazard. If the reactor power level reached a sufficiently high level as a consequence of a reactivity insertion extensive NaK boiling could occur. At atmospheric pressure level in the reactor vessel, boiling could be achieved at reactor power levels on the order of 100% above the design power level. This condition could be achieved with reactivity insertions on the order of 50¢. Boiling in the reactor could lead to vapor locking of the primary coolant pump with subsequent loss of flow. The

core could then boil dry with subsequent melt-down of the reactor core. It should again be emphasized that in order for an excursion with subsequent boiling and meltdown to occur, a series of improbable events must happen. These events include:

1. The control mechanism fails and continuously inserts reactivity.
2. Operating personnel take no corrective action despite alarms from nuclear instrumentation, temperature instrumentation and possibly high fission gas activity.
3. The automatic setback and scram circuits all fail to operate.
4. The resulting total reactivity addition is less than prompt critical which would cause disassembly of the core and terminate the excursion, but is large enough to cause coolant boiling.
5. The coolant boils from the channels and meltdown of the fuel occurs.

c) Reactivity Insertion Mechanisms for Core Disassembly and Reassembly

Two types of core disassembly should be considered. First, explosive disassembly resulting from prompt critical insertions should be considered. The reactivity insertion mechanisms available while the reactor is intact are limited to slow insertion rates. For example, the maximum rate of reactivity insertion from the control system is 5¢ per second. The resulting excursion would not rupture the reactor vessel and the core materials would be contained within the vessel. Those portions of the core material directed horizontally toward the vessel wall or downward toward the bottom of the vessel could remain within the reflected zone and contribute toward reassembly of a critical geometry. A portion of the core material would be directed upward into the helium cooled zone above the coolant surface. The molten and vaporized UO_2 could condense and freeze in this zone and to a high degree be prevented from returning to the reflected zone. This would result in reducing the amount of fuel available for criticality and, therefore, prevent a second excursion.

The second type of core disassembly could result from loss of coolant with subsequent meltdown of the core. The chain of events necessary for loss of coolant from boiling induced by control system failure and reactivity increase has already been discussed. The coolant system integrity and instrumentation system has also been reviewed. Additional factors which reduce the probability of loss of coolant include specific instrumentation which is sensitive to coolant disturbances. This instrumentation includes thermocouples in each NTFE and DFE, nuclear instrumentation (which would detect short period or high power levels), and fission gas detectors in the helium coolant stream.

Despite the instrumentation and other safeguard features designed to minimize the probability of a coolant loss and fuel meltdown accident, consideration should be given to the consequences of this type of accident.

The meltdown process should be considered both for the TFE core and the NTFE core. Meltdown of TFE assemblies is inhibited by the incorporation of refractory materials, particularly the tungsten clad which has a very high melting point. This is illustrated schematically in Figure 6 which shows the structural features of the TFE. In order for meltdown and reassembly to occur, it is necessary for the Ceramvar outer sheath to melt (which would occur if high power were maintained without coolant), for the alumina liner to break up or melt and for the niobium collector to break up or melt. It is possible that these materials could fail in this manner if the power level were sustained for the proper period of time.

The TFE materials surrounding the tungsten clad will melt first in the central TFEs. Since niobium and alumina have higher melting points than stainless steel or Ceramvar, it is likely that much of this material will run out of the core zone onto the grid plate with the TFEs maintained in a vertical position. Calculations show that this loss of material from the core results in a loss of reactivity of about \$1.5 for the most reactive possible

configuration. This effect combined with loss of coolant brings the total reactivity loss to about \$1.8 not including reactor expansion losses. Since the entire loss of cooling incident discussed here was initiated by less than \$1, the power level characteristic of that which prevails during the loss of stainless steel and Ceramvar cannot be maintained. The process of melting alumina and niobium, therefore, must proceed at a slower rate. In order for the fuel to fall down upon the core grid plate in a compact configuration, it is not necessary to melt all of the niobium, but the joint between adjacent emitter collector pairs must be broken in order to allow these to fall down in an efficiently packed configuration. This process must occur over a relatively long period of time with the central portion of the fuel falling out first. As this proceeds, further reactivity must be lost causing the disassembly process to proceed at ever decreasing rates.

Achieving high packing fractions ($> 50\%$) from randomly falling cylinders is quite unlikely. Nevertheless, assuming packing fractions of 60% to 70% can be achieved excess criticality in the resulting configuration of \$8 and \$14 respectively result. Achieving these levels of excess reactivity also depends strongly upon achieving exactly the proper geometry in the reassembled material. For example, if the axial reflector material were partially mixed into the core material (an almost certain state of affairs) then the total reactivity of the assembly would be less.

The important quantity in this entire process is the rate at which reactivity is being added just as the assembly reaches prompt critical. From a consideration of the foregoing steps required to reassemble the core, it is seen that an extremely unlikely series of events are required to occur. By the nature of these events, the process must occur over a period of time characterized by the rate of meltdown of the core material. This process cannot proceed in a simultaneous fashion as a consequence of radial and axial fission distributions and the higher central temperatures of a non-cooled core. This process must, therefore, occur over a period of length ranging from

several seconds to several minutes depending upon the power level at which the final meltdown process occurs. Since reactivity is lost as the coolant is removed and reflector and core materials begin to drop down to the core grid plate, the latter portion of the process must occur at low power levels. For this reason, it is likely that minutes rather than seconds characterize the time required. Reactivity insertion rates of above \$1/second, therefore, will not occur and \$10/second would be a conservative design basis.

Sufficient calculations have not been performed at this time to adequately determine the reactivity levels and insertion rates. In particular, the role of the TFE's and the behavior of NTFE's have not been fully examined in formal calculations. Further and more conservative mechanistic evaluations of reactivity insertion mechanisms must be made. The calculations which have been carried out at this time have all been associated with the 100 kWe reactor. The specific design features for the 300 kWe reactor have not been developed at this time. It is unlikely, however, that the changes in the design associated with the larger reactor will significantly affect the estimates of the excursion phenomena discussed here. Based upon the results obtained at this time, it is believed that the present design basis selected for the reactor containment analysis is adequate.

3. Energy Release Analysis

a. Bethe-Tait Analysis

Estimates have been made of the magnitude of possible energy releases resulting from a nuclear excursion in the TREX core. A modified Bethe-Tait method was used for the analysis. The calculational technique and the major assumptions are described in GEAP-4809.⁽²⁶⁾ For calculational convenience the equivalent spherical core geometry was assumed rather than the cylindrical geometry. This approach is believed to yield more conservative results than would have been obtained had the cylindrical model been employed. Table 46 gives the reactor core characteristics used in the analysis. The major goal of this effort was to define the range of energy releases hypothetically possible from the TREX core as a result of various reactivity insertion rates. For these purposes a non-mechanistic approach was employed. The reactivity insertion rates chosen for study were intended to envelop what may be physically possible in the TREX core. The values used ranged from 1\$/sec to 100\$/sec. The upper limit of 100\$/sec is believed to be very nearly improbable for this reactor. No physical mechanism can be conceived which would yield such a high insertion rate.

Because of the limited time available the rate at which reactivity was inserted into the core was the only parameter which was varied. Had it been possible, the study of several additional parameters would have been undertaken. Those felt to be important are neutron lifetime and the disassembly energy threshold. At present the appropriate values for both of these parameters are somewhat uncertain. The neutron lifetime used in the calculations was 5.5×10^{-7} sec. This value is relatively short. Past studies of this type have generally shown that short neutron lifetimes result in larger energy releases than is the case for longer lifetimes. Since the value used in the present Bethe-Tait calculations was on the low side of what is actually expected for the TREX core, the results are believed to be conservative in this regard.

TABLE 46
TREX CORE CHARACTERISTICS

Composition (Volume fractions):

UO₂	0.321
Steel	0.152
Additional Structure	0.277
Coolant	0.250
Core Height	32.1 cm
Core Diameter	33.8 cm
Radius of Equivalent Sphere	22.5 cm
Reflector Savings	
Radial	13.6 cm
Axial	11.7 cm
Neutron lifetime	5.5×10^{-7} sec.
Normal Operating Power	1 MWt

The disassembly energy threshold is the energy density in the core at which disassembly due to fuel vapor pressures is assumed to begin. If a high energy density threshold is assumed, and thus a high vapor pressure, more reactivity can be inserted into the core than if a low threshold energy is used. This is because the core is assumed to remain intact farther into the excursion. This additional reactivity insertion generally results in a higher energy release. For the TREN accident analysis the disassembly threshold was held at 1340 joules/gm. This is a typical value used in such analyses. Some considerable effort would likely be required to demonstrate that this disassembly energy is or is not the most appropriate for the TREN core. Additional studies varying this parameter would be useful in reducing the uncertainties that presently exist. They would serve to more adequately define the influence of the disassembly energy threshold on the energy release resulting from an excursion. As such, these studies would provide a more complete envelop of the energy release to be expected for various accident situations.

The Bethe-Tait analysis of the TREN core assumed that the only negative reactivity feedback operative was that due to the core disassembly itself. All thermal expansion feedbacks were assumed to be negligible. For the thermionic fuel elements this is a fairly accurate representation. Each fuel pin is composed of several small fuel sub-elements which are individually suspended within the sheath tube. The thermal expansion of one element is not felt by any of the other elements within the tube. Any thermal expansion reactivity effects, therefore, would have to be the result of thermal expansion of the sheath tube itself. However, the sheath tube is fairly well insulated from the fuel. During normal operation a significant portion of the heat generated in the fuel is transferred via the electron cooling process. This mode can lose its effectiveness rapidly if the fuel operation deviates significantly from the design conditions. Thus during an excursion the bulk of the heat transfer within the TFE's would be carried on by normal conduction and radiation processes. These modes in the TFE design are such that the effective gap resistance between the fuel and clad is quite high. The response of the sheath tube to fuel temperature increases will be very sluggish. Thus little, if any, thermal expansion feedback effects can be expected from the thermionic fuel elements.

The TREX core may employ driver fuel elements. These elements will resemble more typical power reactor fuel pins. The thermal expansion of this fuel during an excursion will likely result in some negative reactivity feedback which would serve to limit the severity of the energy release to some extent. The magnitude of this effect is uncertain at this time, however, and the calculations assumed that such feedbacks were negligible. Conversely, the TREX core has a slightly positive Doppler coefficient. A value of $A_{Dop} = +0.0008$ was used in the analysis. The use of the positive Doppler with the assumption that the expansion feedbacks are negligible should tend to maximize the calculated energy release resulting from a given reactivity insertion rate. Therefore, the estimates of the magnitude of the energy releases should be fairly conservative.

The results of the Bethe-Tait analysis are shown in Figure 101. The calculated energy releases are shown as a function of reactivity insertion rate. Both the total energy and the available work energy are presented. As is to be expected the energy release increases reactivity insertion rate. However, the slopes of the curves decrease with the higher insertion rates.

Calculations of available work versus ramp rate are based on the assumption that the fuel vaporized in the excursion expands isentropically down to a pressure of one atmosphere. This ideal process maximizes the available work possible from such an expansion. Thus these estimates are undoubtedly conservative. Most likely a significant fraction of this energy would be converted into heating. Consequently, it would not be available to do work on the reactor vessel or vessel internals.

Calculations have been made by Jankus⁽²⁹⁾ which indicate that the available work may be overestimated by as much as a factor of 10 when the assumption of isentropic expansion of the fuel vapor is employed. The KIWI experiments also demonstrated that the calculation of available work is very conservative. Based on this evidence it appears that the actual available work resulting from a nuclear burst may be less than the ideal value by a factor of 5 to 10.

The available work curve of Figure 101 reflects the above uncertainty. The upper bounds of the cross-hatched area represent the calculated, ideal expansion values. The lower bound reduces the above values by a factor of 5. Based on the available evidence, a reduction in the calculated available work by a factor of 5 appears reasonable as a design criterion.

The fact that the TREX core has negligible negative reactivity feedbacks except the core disassembly process itself means that power pulsing will not occur during the excursion. More typical reactors with large negative Doppler coefficients have been shown to display this pulsing characteristic. With the present core, however, the power increases smoothly up to the point of disassembly. A typical trace of the instantaneous core power versus time is shown in Figure 102. The time scale is the time from the onset of the reactivity insertion. The power trace in Figure 102 results from a reactivity insertion rate of 10\$/sec. The power traces for other reactivity ramps are quite similar in shape.

The results shown in Figures 101 and 102 were generated assuming a value of +0.0008 for the Doppler coefficient. An additional case was run wherein the Doppler coefficient was assumed to be zero. A comparison of the resulting energy releases for these two values of Doppler are tabulated below. The releases are based on a reactivity insertion rate of 10\$/sec.

<u>A_{Dop}</u>	<u>Total Energy Release, MW -Sec</u>	<u>Available Work, MW -Sec</u>
0.0	598	68
+0.0008	678	84

The differences tabulated above are not spectacular. They do show that the positive Doppler coefficient does contribute toward obtaining a higher energy release.

The computer output resulting from the Bethe-Tait analysis of the TREX core is presented herein. The starting point for these calculations is not the onset of the reactivity insertion. Rather it is a point in the excursion just prior to the core achieving disassembly energy densities. The core conditions from time zero (onset of reactivity insertion) up to this point are calculated by a transient analysis code. The transient analysis code used was entitled

FIGURE 101. TREX CORE ENERGY RELEASE VERSUS REACTIVITY INSERTION RATE

FIGURE 102. TREX CORE POWER TRACE DURING EXCURSION

FOX E-II. This code is a FORE-II ⁽³⁰⁾ type computational model in that it predicts reactor response to reactivity and flow perturbations assuming no change in the core geometry. The code predicted the reactor behavior prior to the point in an excursion where core disassembly could occur. It provided the information required to proceed with the Bethe-Tait analysis.

The following tables contain the important output from the computerized Bethe-Tait analysis. The tabulated output given as a function of time are:

- a) peak instantaneous power density
- b) peak energy density
- c) K effective
- d) total energy release
- e) available work

Note that (a) and (b) are given as densities. They are also peak values. To obtain the total for the whole core the values tabulated must be multiplied by the core mass of UO_2 (1.544×10^5 gm) and the ratio of average to peak power (0.717).

Each table was generated for a specific reactivity insertion rate. Two values of Doppler coefficient were also used. The following list describes the values used as input for each case.

<u>Table #</u>	<u>A_{Dop}</u>	<u>Reactivity Insertion Rate, \$/sec</u>
47	+0.0008	1.0
48	0.0	10.0
49	+0.0008	10.0
50	+0.0008	100.0

TABLE 47
BETHE-TAIT ANALYSES OF TREX CORE

Reactivity Insertion Rate = 1\$/sec.
A_{Dop} = +0.0008

Time, sec.	Peak Instantaneous Power Density, joules/gm—sec.	Peak Energy Density joules/gm	K _{effective}	Total Energy Release,joules	Available Work,joules
0.	6.29022E 04	1.11070E 03	1.01699E 00	3.03701E 08	0.
1.52588E-05	6.33128E 04	1.11166E 03	1.01699E 00	3.03965E 08	0.
2.44141E-04	6.99381E 04	1.12689E 03	1.01700E 00	3.08130E 08	0.
1.95312E-03	1.67314E 05	1.31386E 03	1.01709E 00	3.59251E 08	0.
2.13623E-03	1.86637E 05	1.34622E 03	1.01710E 00	3.68100E 08	4.18262E 01
2.18201E-03	1.91919E 05	1.35468E 03	1.01710E 00	3.70469E 08	4.36285E 02
2.27356E-03	2.03082E 05	1.37296E 03	1.01711E 00	3.75412E 08	3.52249E 03
2.42615E-03	2.23605E 05	1.40548E 03	1.01712E 00	3.84304E 08	2.15078E 04
2.67029E-03	2.62051E 05	1.46462E 03	1.01713E 00	4.00474E 08	1.19055E 05
2.97546E-03	3.21961E 05	1.55335E 03	1.01715E 00	4.24735E 08	4.97036E 05
3.26064E-03	3.99898E 05	1.66296E 03	1.01717E 00	4.54709E 08	1.46782E 06
3.58582E-03	4.93740E 05	1.79923E 03	1.01710E 00	4.91968E 08	3.57599E 08
3.89099E-03	5.33345E 05	1.95948E 03	1.01658E 00	5.35785E 08	7.38991E 06
4.19617E-03	3.23427E 05	2.09818E 03	1.01479E 00	5.73711E 08	1.18189E 07
4.37927E-03	1.23221E 05	2.13840E 03	1.01262E 00	5.84708E 03	1.32883E 07
4.53186E-03	2.95966E 04	2.14895E 03	1.01003E 00	5.87592E 08	1.36871E 07

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TABLE 48

BETHE-TAIT ANALYSES OF TREX CORE

Reactivity Insertion Rate - 10\$/sec

 $A_{Dop} = 0.0$

Time, Sec.	Peak Instantaneous Power Density, joules/gm-sec	Peak Energy Density joules/gm	$K_{effective}$	Total Energy Release, joules	Available Work, joules
0.	4.08724E 05	1.10477E 03	1.01796E 00	3.02080E 08	0.
1.52588E-05	4.22234E 05	1.11111E 03	1.01797E 00	3.03613E 08	0.
2.44141E-04	6.93326E 05	1.23606E 03	1.01800E 00	3.37979E 08	0.
3.66211E-04	9.08932E 05	1.33323E 03	1.01802E 00	3.64547E 08	0.
3.77655E-04	9.32509E 05	1.34376E 03	1.01803E 00	3.67428E 08	1.00166E 01
3.85284E-04	9.48586E 05	1.35094E 03	1.01803E 00	3.69390E 08	1.92732E 02
4.04358E-04	9.90075E 05	1.36942E 03	1.01803E 00	3.74445E 08	2.61487E 03
4.34875E-04	1.06050E 06	1.40070E 03	1.01803E 00	3.82997E 08	1.75959E 04
4.73022E-04	1.15605E 06	1.44295E 03	1.01804E 00	3.94549E 08	7.15598E 04
5.18799E-04	1.28283E 06	1.49872E 03	1.01805E 00	4.09798E 08	2.26794E 05
5.95093E-04	1.52778E 06	1.60565E 03	1.01806E 00	4.39036E 08	8.84215E 05
6.71387E-04	1.82234E 06	1.73310E 03	1.01807E 00	4.73885E 08	2.42324E 06
7.47681E-04	2.17497E 06	1.88520E 03	1.01806E 00	5.15474E 08	5.44361E 06
9.00269E-04	3.02618E 06	2.28062E 03	1.01781E 00	6.23596E 08	1.91087E 07
1.05286E-03	3.40406E 06	2.78856E 03	1.01619E 00	7.62484E 08	4.30991E 07
1.10626E-03	3.02421E 05	2.96193E 03	1.10477E 00	8.09889E 08	5.14101E 07
1.11675E-03	2.90304E 06	2.99303E 03	1.01441E 00	8.18393E 08	5.29406E 07
1.23882E-03	1.05623E 06	3.23580E 03	1.00959E 00	8.84772E 08	6.52549E 07
1.33938E-03	2.13574E 05	3.28770E 03	1.00395E 00	8.98965E 08	6.79673E 07
1.37615E-03	6.51383E 04	3.29357E 03	1.00015E 00	9.00568E 08	6.82752E 07

TABLE 49

BETHE-TAIT ANALYSES OF TREX CORE

Reactivity Insertion Rate = 10\$/sec

 $A_{Dop} = +0.0008$

Time, sec.	Peak Instantaneous Power Density, joules/gm-sec	Peak Energy Density joules/gm	$K_{effective}$	Total Energy Release, joules	Available Work, joules
0.	4.08724E 05	1.10477E 03	1.01796E 00	3.02080E 08	0.
1.52588E-05	4.22250E 05	1.11111E 03	1.01797E 00	3.03813E 08	0.
1.83105E-04	6.09911E 05	1.19664E 03	1.01803E 00	3.27201E 08	0.
3.66211E-04	9.30376E 05	1.33533E 03	1.01810E 00	3.65123E 08	0.
3.70026E-04	9.38835E 05	1.33890E 03	1.01810E 00	3.66098E 08	0.
3.79562E-04	9.60365E 05	1.34795E 03	1.01810E 00	3.68574E 08	8.18849E 01
3.94821E-04	9.95989E 05	1.36288E 05	1.01811E 00	3.72655E 08	1.35118E 03
4.17709E-04	1.05228E 06	1.38631E 03	1.01812E 00	3.79062E 08	8.61430E 03
4.55856E-04	1.15433E 06	1.42836E 03	1.01814E 00	3.90561E 08	4.76503E 04
4.94003E-04	1.27682E 06	1.47452E 03	1.01816E 00	4.03183E 08	1.45987E 05
5.32150E-04	1.39426E 06	1.52526E 03	1.01816E 00	4.17054E 08	3.42040E 05
5.70297E-04	1.53547E 06	1.58159E 03	1.01826E 00	4.32320E 08	6.85813E 05
6.23703E-04	1.76191E 06	1.66897E 03	1.01823E 00	4.56352E 08	1.53912E 06
6.99997E-04	2.15603E 06	1.81785E 03	1.01828E 00	4.97059E 08	3.94509E 06
7.76291E-04	2.65269E 06	2.08057E 03	1.018308 00	5.47022E 08	8.59615E 06
8.52585E-04	3.26118E 06	2.22551E 03	1.01825E 00	6.08526E 08	1.67412E 07
9.89914E-04	4.32574E 06	2.75270E 03	1.01731E 00	7.52678E 08	4.14296E 07
1.04713E-03	4.32533E 06	3.00275E 03	1.01611E 00	8.21048E 08	5.34268E 07
1.15395E-03	2.77819E 06	3.39813E 03	1.01237E 00	9.29160E 08	7.38245E 07
1.23024E-03	1.07198E 06	3.54396E 03	1.00654E 00	9.69034E 08	8.17357E 07
1.29890E-03	1.82068E 05	3.58161E 03	9.97366E 01	9.79329E 08	8.35121E 07
1.32179E-03	7.65010E 04	3.58443E 03	9.93386E 01	9.80100E 08	8.39683E 07

TABLE 50

BETHE-TAIT ANALYSES OF TREX CORE

Reactivity Insertion Rate = 100\$/sec.

 $A_{Dop} = +0.0008$

Time, sec.	Peak Instantaneous Power Density, joules/gm-sec	Peak Energy Density joules/gm	$K_{effective}$	Total Energy Release, joules	Available Work, joules
0.	1.86531E 06	1.14910E 03	1.02093E 00	3.14201E 08	0.
1.52588E-05	2.09016E 06	1.17924E 03	1.02097E 00	3.22443E 08	0.
6.86646E-05	3.13755E 06	1.31681E 03	1.02109E 00	3.60058E 08	0.
7.62939E-05	3.32842E 06	1.34147E 03	1.02111E 00	3.66800E 08	4.31482E-01
8.10623E-05	3.45403E 06	1.35763E 03	1.02113E 00	3.71221E 08	6.81990E 02
8.96454E-05	3.69318E 06	1.38829E 03	1.02115E 00	3.79605E 08	9.62135E 03
9.91821E-05	3.98001E 06	1.41486E 03	1.02117E 00	3.89604E 08	4.27964E 04
1.16348E-04	4.55871E 06	1.49803E 03	1.02122E 00	4.09611E 08	2.24198E 05
1.35422E-04	5.31036E 06	1.59195E 03	1.02128E 00	4.35292E 08	7.69899E 05
1.54495E-04	6.19836E 06	1.70147E 03	1.02134E 00	4.65238E 08	1.95877E 06
1.73569E-04	7.25050E 06	1.82944E 03	1.02141E 00	5.00229E 08	4.18487E 06
2.07901E-04	9.67490E 06	2.11778E 03	1.02155E 00	5.79071E 08	1.25250E 07
2.46048E-04	1.34685E 07	2.55485E 03	1.02169E 00	6.98579E 08	3.25705E 07
2.72751E-04	1.70512E 07	2.96042E 03	1.02172E 00	8.09474E 08	5.13356E 07
2.74658E-04	1.73409E 07	2.99321E 03	1.02171E 00	8.18442E 08	5.29495E 07
3.128058E-04	2.41512E 07	3.77855E 03	1.02144E 00	1.03318E 09	9.48913E 07
3.50952E-04	3.11723E 07	4.84308E 03	1.01887E 00	1.32426E 09	1.63691E 08
3.89099E-04	2.69789E 07	6.01660E 03	1.00851E 00	1.64513E 09	2.77995E 08
4.19617E-04	1.07496E 07	6.60300E 03	9.89352E-01	1.80547E 09	3.66568E 08
4.38698E-04	3.12878E 06	6.72693E 03	9.70939E-01	1.83936E 09	3.89276E 08
4.56949E-04	7.01686E 05	6.75273E 03	9.52383E-01	1.84642E 09	3.94204E 08
4.63486E-04	2.09667E 05	6.75668E 03	9.39039E-01	1.84750E 09	3.94966E 08
4.71115E-04	6.74678E 04	6.75765E 03	9.27394E-01	1.84776E 09	3.95153E 08

b. Blast Effects on Reactor Vessel

The plots in Figure 101 show the magnitude of the energy releases resulting from various reactivity insertion rates. The total energy generated is much higher than the available work for nearly all cases considered. This is because the bulk of this energy appears as stored energy in the fuel. Only a relatively small fraction of the energy release is available to do work on the vessel or other reactor structure. Thus it is the available work portion of the total energy which is of primary concern in defining possible hazards to the reactor vessel and its integrity.

The available work energy generated in a nuclear excursion will be dissipated in one of three possible forms. First, a shock wave resulting from the blast energy can be created. The energy in this shock wave will be partially absorbed by deformation of the reactor vessel and internal structure. This shock wave can also be a source of missile generation. Second, a quasistatic pressure will result from the blast. This pressure is of relatively short duration, but can be considered as static in establishing its effect on the reactor vessel. The third and possibly largest fraction of the available work energy generated during the excursion will be exhibited as heat energy which will serve to directly increase the temperature of the reactor vessel and its internals.

(1) Shock Wave Effects

The shock wave generated by the nuclear excursion can be assumed to be similar to that generated by a TNT type blast. The vessel damage resulting from such a shock wave can be predicted from the results of extensive testing programs which have been carried out with the use of high energy explosives. In the present study the results published in NOLTR 63-140 ⁽³¹⁾ were used to estimate vessel deformation

resulting from the blast. The working relationship used is as follows:

$$W = \left[\frac{1.407 \sigma_t \epsilon (3.41 + 0.117 R_i/t) (R_o^2 - R_i^2)^{1.85}}{10^5 \rho^{-0.85} (1.47 + 0.0373 R_i/t)^{0.15} R_i^{0.15}} \right]^{0.811} \quad (34)$$

where W = equivalent blast charge, # TNT

ϵ = maximum vessel strain, in/in

R_i = initial internal radius of vessel, ft.

R_o = initial external radius of vessel, ft.

t = initial vessel wall thickness, ft.

ρ = density of wall material, #/ft³

σ_t = true wall stress, #/in²

The true wall stress is given by:

$$\sigma_t = \sigma_y + \left[\frac{\sigma_u (1 + \epsilon_u) - \sigma_y}{\epsilon_u} \right] \epsilon \quad (35)$$

and σ_y = conventional yield stress of vessel material, #/in²

σ_u = conventional ultimate stress of vessel material, #/in²

ϵ_u = conventional ultimate strain of vessel material, in/in.

The above relationships define the magnitude of the charge which will cause a maximum strain ϵ in the vessel. All of Wise's experiments were conducted in cylindrical vessels. His tests showed that, for vessels of uniform thickness, the limits of the blast damage extended only about two radii axially along the vessel wall. At points more distant than this from the blast center only a negligible amount of strain resulted from the blast shock wave.

The present calculations were made assuming that the equivalent TREX vessel wall thickness in the core area was 2.5 inches. The vessel wall itself is about one-half inch thick, but the additional material can be added at the outer diameter of the radial reflector. This will effectively add hoop strength to the vessel wall. All of the vessel material was assumed to be at the reactor vessel radius rather than at the outer radius of the reflector. This should make the estimates of the vessel damage conservative since less wall mass was assumed in the calculations than will actually be present.

The radial reflector is a fairly massive structure. This structure was not assumed to offer any tensile restraint to straining of the reactor vessel due to the action of the blast shock wave. However, the reflector mass will act as an inertial restraint and will be effective in reducing vessel strain. In order to account for its influence in the calculations of vessel damage, its mass was added to that of the vessel wall. This was done by increasing the wall density an appropriate amount in Eq. (34).

The results of these calculations are shown in Figure 103. Maximum vessel deformation is shown as a function of the blast energy release. The units of energy release are in terms of pounds of TNT.

The vessel material is assumed to be capable of an ultimate strain on the order of 40%. However, it is likely that the vessel will be fabricated with welds or other discontinuities. A more practical limit to design is a strain of about 15%. Beyond this there would be considerable uncertainty about the vessel integrity. Figure 103 shows that an energy release equivalent to about 13# TNT will produce this type of strain in the reactor vessel. Blasts greater than this can be assumed to cause failure of the vessel. Note that the energy release of concern in causing this type of deformation is the available work energy, not the total energy generated during an excursion. The 13# TNT limit is equivalent to about 26 MW-sec. of available work.

Under the assumption that the fuel vapor expansion is isentropic, a reactivity insertion rate of slightly greater than 2.00 \$/sec will result in available work near 26 MW/sec. If this expansion is a non-ideal, non-isentropic process, then a reactivity insertion rate of nearly 20 \$/sec can likely be tolerated without violating the vessel integrity.

The internal components of the reactor vessel may possibly act as missiles under the action of the blast shock wave. If these potential missiles possess enough energy, they may possibly penetrate the vessel walls. The vessel should be designed

FIGURE 103. TREX VESSEL DEFORMATION RESULTING FROM
TNT LIKE ENERGY RELEASE

such that it is capable of withstanding the effects of any missiles for certain ranges of nuclear excursions. The philosophy was adopted here that missile penetration should be avoided for any excursion which results in less than 26 MW-sec of available work. If this energy level is exceeded the vessel will likely fail from excessive strain due to the action of the blast shock wave. Therefore, there is little to be gained in providing protection against missile penetration for energy releases greater than about 26 MW-sec.

Several of the vessel interval components were analyzed to define their capability as missiles. The energy imparted to a given component depends on the blast energy and the distance to the blast center. Cole ⁽³²⁾ found that the specific impulse delivered to an object from the blast shock wave could be defined as follows:

$$I = KW^{1/3} \left(\frac{W^{1/3}}{R} \right)^B \quad (36)$$

where I = specific impulse, #-sec/in²

W = weight of charge, #

R = distance from center of blast to object, ft.

K, B = constants

For most explosives B is near unity. Making this simplification, Eq. (36) becomes:

$$I = K \frac{W^{2/3}}{R} \quad (37)$$

This impulsive energy will be converted into kinetic energy.

The velocity imparted to the object is:

$$V = \frac{IAg}{M} \quad (38)$$

where V = velocity, in/sec

I = specific impulse, #-sec/in²

A = area receiving impulse, in²

M = mass of object, #

g = gravitational constant, in/sec²

The resulting kinetic energy is given by:

$$KE = \frac{1}{2} \frac{MV^2}{g} = \frac{1}{2} \frac{I^2 A^2 g}{M} \quad (39)$$

If a missile strikes the reactor vessel, its energy must be absorbed by shearing of the vessel walls.

The energy absorption capabilities of the wall can be roughly approximated as follows:

$$E = \frac{\sigma_{\text{Shear}} \times \pi D t}{\text{shear force}} \times \frac{t}{\text{Action length}} \quad (40)$$

Where E = energy absorption, ft-#
 σ_{shear} = shear stress of vessel walls, #/in²
 D = diameter of area being sheared off, ft.
 t = wall thickness, in.

This simple relationship was found to give conservative results when compared to the experimental findings described in SRIA-36⁽³³⁾.

To avoid penetration by a given missile, the energy absorption capabilities of the vessel wall described in Eq. (40) must be greater than the kinetic energy of the missile as described by Eq. (39).

One component which may act as a missile is the core lower grid plate. A shock wave striking this plate may cause it to shoot downward and damage or rupture the bottom head of the reactor vessel. An analysis was made to define the required thickness of the bottom head to prevent penetration by the lower grid plate. The results are shown in Figure 104. The minimum required thickness is shown as a function of the available work blast energy equivalent. As noted previously the reactor vessel will likely fail if the blast equivalent exceeds 13# of TNT. To prevent missile failure of the vessel bottom head for accidents releasing less than this energy its thickness should be at least 0.9 inches.

FIGURE 104. REQUIRED VESSEL BOTTOM THICKNESS TO WITHSTAND MISSILES GENERATED BY BLAST EFFECTS

The missile penetration resistance of the vessel lower head is believed to be considerably greater than is implied by the present analysis. This is because any object striking this portion of the vessel will first have to travel through some length of coolant. It would lose considerable energy in so doing. Similarly, the grid plate is held in place by a core shroud cylinder. Severing this structure to free the grid plate will also require considerable energy. Neither of these energy absorbing mechanisms were considered in the present analysis. The results shown in Figure 104, therefore, are very conservative.

A segment of a thermionic fuel element may also act as a missile. Prior to or during a large energy release the process tubes and TFE sheath tubes will likely lose their integrity. It is possible that an intact fuel segment could then be exposed to the effects of the blast shock wave. Such a missile was assumed to be generated in the energy release. For calculational purposes the segment was assumed to have a length to diameter ratio of about 2.3. The maximum vessel penetrating capability of such an object would be realized if its longitudinal area, rather than its cross-sectional area, received the impulse from the shock wave. It could penetrate most easily if it struck the vessel wall in a head-on manner. In this way the area of the vessel wall receiving the force of the missile would be the cross-sectional area of the missile. These conditions maximize the missile penetrating capability.

The analysis performed assuming the above conditions showed that this type of missile may pose the greatest hazard to the reactor vessel. For a blast energy release of about 13# TNT, the calculations indicated that the vessel wall thickness should be slightly greater than one inch in order to absorb all of the missile energy. In the region of the core the vessel itself is only about one-half inch thick. Thus if such a missile were unimpeded the present calculations indicate that it could penetrate the wall. It is unlikely that this could be the case. Several items would serve to limit the missile energy. First, coolant will likely be present. Second, portions of the

peripheral core structure such as the BeO rods and their process tubes would be intact at this point. Finally the core shroud adds effective wall thickness to the vessel. All of these items would absorb energy from this hypothetical missile. Further, the vessel walls themselves are backed up by the radial reflector. Its structure would serve to enhance the energy absorption capabilities of the vessel wall. All of these factors can be considered to add effective thickness to the vessel walls. Their effects are difficult to quantitize.

The present analysis assumed that this hypothetical missile received its energy and imparted it to the vessel wall in a very idealistic manner. The possibility of this being achieved in a reactor accident is quite remote. This factor, together with the energy absorption capabilities of the structure and coolant in the path of the missile, make it highly unlikely that the vessel integrity will be violated by missiles of this type resulting from blast energies of less than 13 MW-sec.

The reactor upper grid plate may also be susceptible to damage resulting from the blast shock wave. The plate itself is removed from the core area and would not receive a shock wave directly. However, process tubes and the fuel element structure extends downward from this region into the core zone. A shock wave generated in the core could act as a hammer blow to the lower end of this long structure. The energy so imparted could then be delivered to the upper grid plate before the process tubes and fuel elements had time to deform. This assumes that some mechanism is provided to prevent the process tubes from merely jumping out of positions in the upper grid plate without transferring their kinetic energy to the plate.

This type of hazard to the upper grid plate was analyzed assuming that all of the vertical structure acted in unison. If this were the case the grid plate would have to absorb the energy delivered to this structure by the shock wave. Since the grid plate has many penetrations, its limiting energy absorption capability is defined by the webbing between penetrations. The present analysis was based on a configuration

which gave only about 7% of the affected diameter as being webbing. That is, of the total diameter of the upper grid plate only about 7% of it was assumed to be effective in absorbing energy. The remainder is taken up by penetrations.

The calculations of this possible hazard indicated that the upper grid plate thickness need only be about 1.25 inches to absorb the impulsive energy delivered to it in this manner. This applies to an energy release equivalent to 13# TNT.

Water hammer effects resulting from the movement of reactor coolant should not pose a problem to the reactor grid plate. The TREX vessel configuration is such that such a slug could not move upward without incurring significant frictional resistance from the process tubes and other internal structure. Its energy would be dissipated in frictional heating and distorting the vessel internals. Thus it is unlikely that water hammer hazards pose as significant a threat to the vessel integrity as do the other destructive mechanisms associated with a blast type energy release.

(2) Quasi-Static Blast Pressure

The reactor vessel will be pressurized as a result of the fuel vaporized during a nuclear excursion. The vaporization of coolant can also contribute to the vessel pressurization. Because the fuel vapors are at a much higher temperature than the balance of the reactor materials, they should be cooled fairly quickly. This cooling will serve to reduce the pressure. However, it can reasonably be assumed that the fuel vapor is cooled slowly enough such that the internal pressure can be considered static in terms of its effects on the vessel.

The quasi-static pressure resulting from a nuclear burst can be estimated from the following relationship:

$$P = \frac{E (K-1)}{V} \quad (41)$$

where p =   vessel pressure
            E =   energy release portion which does not go
                                 into shock wave or heating.
            K =   ratio of specific heats for the fuel vapor, C_p/C_v
            V =   vessel volume available for vapor expansion

The energy assumed in the calculation is that portion of the available work which is not manifested in either the shock wave or rapid heating of the reactor coolant or structure. It can be conservatively estimated that about half of the available work will contribute to pressurization of the vessel. Further, approximately half of the vessel volume is available for vapor expansion.

The pressure as calculated by Eq. (41) is quite sensitive to the value of K . The value of K is somewhat uncertain at this time because little or no published experimental evidence is available on C_v and C_p values for UO_2 vapor. Reference (26) estimated a value of C_v determined by adding together the degrees of freedom of the UO_2 molecule. This estimate, plus the relationship between C_p , C_v , and R , leads to a value of K of 1.152. This value was used in the calculation of vessel pressurization.

Using the above assumptions and conditions it is estimated that a quasi-static pressure of about 600 psi will result from an accident in which 26 MW-sec of available work energy is released. As noted previously, an energy release of this size could be initiated by reactivity insertion rates as low as 2.20 \$/sec or as high as 20 \$/sec. The exact value depends on how nearly the vapor expansion process approximates an ideal expansion. An internal pressure of 600 psi will cause a vessel hoop stress of about 16,000 psi. Assuming the vessel walls are at coolant inlet temperatures, this stress is slightly below the yield stress of the wall material. Thus the quasi-static internal pressure generated by an accident of this magnitude will not cause failure of the vessel.

The present calculation assumed that half of the available work energy goes into the generation of a quasi-static pressure. Even if all of the available work went into pressurizing the vessel the resulting pressure would be about 1200 psi. This would cause wall stresses in the neighborhood of 32,000 psi. The vessel would yield to some extent under these conditions. However, it is unlikely that this pressure would lead to gross failure of the vessel.

More serious accidents will cause greater pressurization of the vessel. However such accidents will likely cause vessel failure due to the action of the blast shock wave. Thus for all cases considered the shock wave effects pose greater hazards to the integrity of the reactor vessel than do the effects of vessel pressurization.

The foregoing discussions relate primarily to the available work portion of the energy generated during a nuclear burst. For accidents considered wherein the reactor vessel remains intact the total energy generation must also be considered. The bulk of this energy is deposited in the fuel during the excursion. Consequently, the fuel temperatures are several thousand degrees higher than those of the bulk of the reactor materials and coolant following the accident. Given sufficient time the reactor contents will achieve thermal equilibrium. In the process coolant will be vaporized with the result that the reactor vessel will be pressurized to some extent. During the time when the coolant vapor is being generated, portions of it will also be condensing on the cooler walls of the upper vessel structure. The pressures achieved during this transient process are a function of the relative rates of vapor generation and condensation, the heat removal from the vessel via coolant flow, and the heat transfer to other heat sinks within the vessel. To correctly evaluate this pressurization an analysis must be made of these transient heat transfer processes.

Some estimates of vessel pressurization were made based on several simplifying assumptions. These sought only to

establish the post-accident near-equilibrium conditions. For the purposes of these calculations the total energy release resulting from a 5\$/sec reactivity insertion was used. This corresponds to about 530 MW-sec of energy. Assuming that no heat is transferred away from the vessel, the calculated final temperature of the reactor vessel and all internals was found to be somewhat less than 1850°F. At this temperature the vapor pressure of NaK is on the order of 90 psi. This pressure is negligible in terms of the vessel capabilities, even at these elevated temperatures. Thus it is expected that the coolant vapor pressure resulting from a 5\$/sec reactivity insertion accident will not significantly stress the vessel once equilibrium has been achieved. The transient conditions prior to equilibrium require more study before any potential pressurization hazard to the vessel can be defined.

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