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**CONFERENCE ON SPECIAL TOPICS ON NUCLEAR
EDUCATION AND RESEARCH HELD IN
GATLINBURG, TENNESSEE, AUGUST 28-30, 1967**

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CONTENTS*

Nuclear Systems for Space and Propulsion	
J. D. Lafleur	1
Astronuclear Engineering Education at the University of Virginia	
J. L. Meem	30
Fusion and Plasmas	
J. L. Dunlap	36
Plasma Physics Education at Pennsylvania State University	
E. H. Klevans	58
Nuclear Safety Research and Development	
W. B. Cottrell	70
Reactor Dynamics - A Foundation for Nuclear Safety	
D. L. Hetrick	95
Direct Energy Conversion Programs at NASA	
Roland Breitwieser	136
Direct Energy Conversion Studies with Emphasis on Those Involving Nuclear Radiation	
G. H. Miley	171
Radiation Effects in Solids	
D. K. Holmes	208
Radiation Effects and Nuclear Materials Program at North Carolina State University	
T. S. Elleman	252
Projections	
J. A. Lane	276

*Papers presented but not included in this report are as follows: "Nuclear Applications to Space Power at the University of Arizona" by M. V. Davis; "Direct Energy Conversion Research - University of California at Berkeley" by L. M. Grossman; "Reactor Safety in Nuclear Engineering Education" by F. J. Jankowski; "Radiation Effects Program at Iowa State University" by Glenn Murphy; and "Plasma Physics Education in the Department of Nuclear Engineering at the University of Michigan" by R. K. Osborn.

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FOREWORD

The objectives of the conference on "Special Topics in Nuclear Education and Research" were to report the activities of selected universities in nuclear fields not directly related to power reactors; to present the history and future of these activities; and to assist other universities to develop and diversify their nuclear interests.

The Education Division of the American Nuclear Society served as the program committee, under the chairmanship of Professor G. H. Miley of the University of Illinois. Our warm appreciation goes to the committee for the excellence of the presentations and discussions. A subcommittee of the ANS Education Division, under the chairmanship of Professor D. L. Hetrick of the University of Arizona, served as a technical review group for the papers included in the proceedings. Their diligent efforts contributed greatly to the excellent quality of the papers presented.

The selection of papers for this publication was made in consultation with Professor Miley and was based mainly on considerations of a good balance in both technical content and in contributions by participants from universities and other organizations.

It is a pleasure gratefully to acknowledge the untiring efforts of Dr. Lewis Nelson, the conference co-chairman, and of members of my own staff, including Drs. J. E. Mott and W. M. Jackson; Bernice Gulley, and Lana Kirk, in making the conference and these proceedings successful.

W. W. Grigorieff
Assistant to the Director for Special Projects
Oak Ridge Associated Universities

NUCLEAR SYSTEMS FOR SPACE POWER AND PROPULSION

J. D. Lafleur
U. S. Atomic Energy Commission

The nuclear systems under development in the U. S. are divided into two main programs: the Systems for Nuclear Auxiliary Power (SNAP) program and the Nuclear Engine for Rocket Vehicle Application (NERVA) program. These systems are primarily the responsibility of the National Aeronautics and Space Administration and the Department of Defense. Although there are other potential users, their uses are further off in the future than for these agencies.

The two main applications for nuclear systems--electric power and propulsion--are illustrated in Fig. 1. In both cases the nuclear component is used as a heat source, for there is no magic quality of the nuclear reactor or isotope to be taken advantage of other than the generation of heat.

In the case of electric power, a reactor, or mass of radioisotopes, generates heat which by radiative heat transfer or by some kind of a plumbing system, is transferred to a static or a turbogenerator system that generates electricity. Not shown in this sketch is the large thermal radiator, which is required to dispose of the excessive heat produced whenever a thermodynamic power generation cycle operates.

In the space propulsion cycle liquid hydrogen is the most effective propellant. The propellant is pumped through a reactor in a system that is described as the NERVA rocket. The expanded liquid hydrogen then flows out the rear through the nozzle to create thrust. The advantage is that with a nuclear propulsion system you can use molecular hydrogen. Since hydrogen is the lightest molecule available, the hydrogen for a given temperature can achieve a high specific impulse.

I. Space Power Systems

Figure 2 shows about where various power competitors fit in space. On the vertical axis the power level is shown in kilowatts, and on the horizontal axis the lifetime is shown in hours. Generally speaking, the isotope, solar and reactor systems are used for the longer lifetimes of up to 10,000 hours. The isotope and solar systems are used where the power requirements are low (up to a few kilowatts), whereas reactors are used when the power requirements are above a few kilowatts.

The dynamic chemical systems, the fuel cells and primary batteries, have a shorter lifetime chiefly because of their higher weight as a function of total energy. Generally, if to be used for several weeks, the batteries (the

Fig. 1--Nuclear energy for space power and propulsion.

Fig. 2--Space power sources for various lifetimes and power levels.

primary batteries) and fuel cells become quite heavy. The chemical dynamic system is for a few hours' use in a high-power application of some type, such as a radar-jamming device.

Reactors are quite heavy at low power because they require a certain critical mass plus some heat-transfer machinery and a shield. Once we get that minimum weight assembled, the use of a reactor can result in a considerable amount of additional power for about the same weight or a small increase in weight, and thus they become more attractive for the higher power applications.

Costwise, the solar and the isotope systems are quite expensive. The plutonium-238 fuel systems that we are now using, and which are now considered to be qualified for space, cost several thousand dollars per electrical watt, with the 5 per cent conversion efficiency that we have today (thermoelectrics). The cost of solar systems rises in direct proportion to the power level, but they are not quite as expensive as isotopes with the current 5 per cent power isotope-thermoelectric conversion efficiency. A small reactor system might cost a million dollars, and for a higher-power system that million-dollar investment can give more power; hence, the cost per kilowatt comes down. The nuclear system is competing with the solar system (generally the photovoltaic system today) for the long lifetime missions.

If solar cells are used in places where the spacecraft goes into the shade, such as behind the earth or behind the moon or in trajectories far from the sun, battery energy storage will be required. A cyclical recharge for secondary batteries can be used; and the cost, weight and other penalties associated with the batteries must be considered.

Weight, size, cost and availability of the isotopes are important. Plutonium-238, which is the currently qualified isotope, is in fairly short supply. Although we do not now see any urgent shortage during the 1960's, we are using essentially all that we can make. One way to overcome any shortage is to develop and learn to use cheaper and more readily available isotopes, such as strontium-90. Strontium-90 is used in some of the terrestrial SNAP systems for navigation buoys, weather stations and other remote earth-and-sea applications.

The size and area of a power system are important. The solar system has to have a certain total area to intercept enough energy from the sun. The photovoltaic device operates at about 10 or 11 per cent efficiency, and total area is a direct function of the power level.

With the thermodynamic power systems which characterize the use of the isotopes and reactors, system size depends largely on the area of the thermal radiator needed to dispose of excess cycle heat. If we are willing to sacrifice a little efficiency, we can have a higher-temperature radiator, which will reduce the size of the power system. Area becomes important also with higher-power systems because of the problem of packaging radiators with large surface areas in available spacecraft or launch vehicles.

There are other special mission factors such as distance from the sun or the time during which a craft has to be within the shadow of the moon or the earth. A prime example is the surface of the moon, which has a night-

time period of about 350 hours. For that reason nuclear power will be especially useful on the surface of the moon.

These relevant factors lead us to a group of development objectives, which ties together this development program of power systems and power conversion devices. The objectives are shown in Table I.

TABLE I

Space Nuclear Power Systems Technology Objectives

1. Increase Heat Source Temperature
2. Develop More Efficient Power Conversion Concepts
3. Demonstrate and Improve Reliability
4. Reduce System Costs
5. Improve Methods for Assuring Safety

First, because the thermodynamic power device has to dispose of the excess waste heat in space by thermal radiation, the radiator must be at a high temperature relative to space, yet must be at the low temperature point of the heat cycle. Thus, the heat source will have to be at high temperatures to attain a respectable Carnot efficiency. Therefore, it is one of the main objectives in the several reactor and isotopic fuels programs to develop higher temperature capability.

The second development objective is to get higher efficiency power conversion devices. For the lower-power systems, the inventory of isotopes in the heat source which is the expensive, scarce item of the power system is reduced when efficiency is increased.

In the case of the higher-power devices which will have reactor heat sources, the size of the reactor determines the size and weight of the shield. This reactor-shield assembly plus the radiator account for about two-thirds of the weight of some of the advanced power systems. Again efficiency, which could limit the size of the reactor and the size of the radiator, is an important factor.

Methods of analyzing for safety have to be improved. A large part of the total analysis and experimental effort still goes into the various considerations of safety in all of the missions for which power systems are being developed today. As more is learned, less new research and development is required; but the competitors with which we have to compare have no safety problem. Because these devices are launched into space, and eventually many come down--and looking at it in the worst possible way, they can come down anywhere--we have to constantly check and improve our methods for insuring safety. Reliability is also vital in order to get useful systems, and cost is always important.

Now let's look at the various SNAP power systems being developed, or that have already been launched, and also look at the advanced technology

program to see how these fit into the picture. First, up to now, six SNAP devices have been launched. Five were radioisotope devices, and one was a small reactor.

A. Radioisotope Power Devices

The first two radioisotope devices were SNAP 3A units producing about three watts each. SNAP 3 was the unit President Eisenhower showed before national television on his desk in 1959. The first unit launched on the DOD Transit experimental navigation satellite has been in orbit for six years (since June 29, 1961), and as of today has the record for uninterrupted generation of power in space. The second SNAP 3 device operated for about eight or nine months, and then a failure quieted the whole satellite. As far as we know, it is generating voltage with an open circuit right now.

Three SNAP 9A units were launched in 1963 and 1964. They were 25-watt units, also using plutonium-238 with thermoelectrics. Two of the units are still working in orbit, and the third failed to orbit when the rocket guidance system failed. That third one was important because it came down over the Pacific Ocean and burned up due to re-entry heating as designed. The small amount of plutonium-238 that was released is being counted around the world today by the AEC fallout monitoring program. It is being dispersed safely, exactly as designed.

Generally speaking, we are going away from the burn-up re-entry devices and are moving in the direction of devices designed to re-enter intact. In that way we achieve safety without having to disperse and distribute isotopes into the atmosphere.

Figure 3 shows the SNAP 19 unit, which has two 25-watt generators in tandem for a total of 50 watts. It will be mounted as shown on the sensing ring of the Nimbus-B Satellite, a NASA meteorological experimental system. Two of the Nimbus satellites have been launched. The next one to be launched will be this one with the first nuclear unit aboard. Those large solar paddles generate an effective average power of about 200 watts initially, but only about 80 watts is left to power the experiments after accounting for the housekeeping load. So the 50 watts end-of-life minimum power that the new SNAP 19 units will generate becomes a substantial part of the total power capability of the system.

Figure 4 shows a cutaway of SNAP 19, illustrating the simplicity of the low-power isotope thermoelectric devices. A capsule containing plutonium dioxide fuel is mounted in the center of the device. It radiates heat to a bank of lead telluride thermocouples. The extra heat is conducted to the fins which in turn radiate it to space.

The power generator elements can be wired in series and parallel to give the desired optimum combination of voltage and current and to assure against open-circuit failures. The solar cells have this advantage, too. Because there are so many of them, it is easy to wire them in such a way that the electrical circuit is simple and reliable.

Fig. 3--Radioisotope electric generator on Nimbus-B satellite (SNAP-19).

SNAP 19 RADIOISOTOPE ELECTRIC GENERATOR

DESIGN POWER OUTPUT 25 WATTS
DESIGN WEIGHT 30 LBS

NASA RN67-1273
12-22-66

Fig. 4--Cutaway diagram of SNAP-19 radioisotope electric generator.

Radiator area is not important in the low-power systems, but it becomes important in the high-power systems such as SNAP 27 shown in Fig. 5. This is a 70-watt unit that will be launched on the Apollo mission that makes the first landing on the moon. Figure 6 shows an artist's concept of the lunar excursion module (LEM) which has landed on the moon. The astronaut is about to remove the experiment package that he will leave on the moon, called the Apollo Lunar Surface Experiments Package (ALSEP). The generator and the various pieces of the experiment station that will be left on the moon are shown in their storage racks.

The LEM carries the plutonium-238 capsule for the SNAP 27 in a graphite container mounted on the side of the vehicle, as shown. This is done in order to simplify the thermal design of this system since the generator is not fueled until required. In case of an abort early in the mission in which the device comes back inadvertently into the earth's atmosphere, this graphite container will protect the fuel so it will not disperse; it will re-enter intact.

Figure 7 shows an artist's view of the astronaut returning after emplacing the generator and several experiment packages. The experiment package is being developed by Bendix and the SNAP 27 generator by General Electric. There is a unit now in ground testing in Valley Forge, Pennsylvania; and one flight unit will be delivered to Bendix at Cape Kennedy later this year.

Figure 8 shows one of several advanced technology programs that we have under way which we call the "Lightweight" thermoelectric panel. The small pellets of lead telluride in the center are thin elements. Used in a system, this optimized panel should result in a low-weight thermoelectric generator. Again the thermoelectrics give an efficiency of only about 5 per cent, which is not good when using scarce, expensive fuels. The conversion system (the generator, in contrast to the heat source) weighs about one-half of the 1 pound per watt of the current isotope systems. This new concept will cut that down to where the conversion system weighs about 20 per cent of the total weight of the system or possibly a little less. With attendant improvement in heat-source technology, we hope to reduce the weight of the isotope devices from 1 pound per watt to about one-third or one-half pound per watt.

Figure 9 is a phantom view of SNAP 29, a several hundred watt unit, showing the generator and radiator assemblies. In this case to meet the requirements of DOD short-lived missions, we are going to use in space for the first time polonium-210, which like plutonium-238 is an alpha emitting isotope. Polonium-210 has a high-power density but only a short half-life of about 138 days. The system which will be ground tested in the late 1960's (probably the end of 1969) will be good for about three to five months in several types of earth orbital missions for DOD or NASA.

The isotope Brayton cycle is shown in Fig. 10. To get higher efficiency, this concept uses more complex power conversion equipment. This is the cycle which NASA hopes will produce about 5-1/2 kW electrical using a 25 kW thermal plutonium-238 radioisotope heat source. It heats a gas working fluid, probably argon or some optimized mixture of inert gases, which drives a turbine-compressor alternator by a recuperative Brayton cycle. The excess heat is ultimately disposed in a liquid-filled radiator loop. NASA has requested the AEC to furnish a heat source, and we are developing the 25 kW thermal plutonium-238 heat source.

Fig. 5--SNAP-27 radioisotope electric generator, showing the large radiator area required for high power systems.

Fig. 6--Artist's concept of the lunar excursion module (LEM), which has landed on the moon, showing astronaut about to remove the experiment package.

Fig. 7--Apollo lunar surface experiment package (ALSEP) with SNAP-27 radioisotope electric generator.

Fig. 8--"Lightweight" thermoelectric converter panel being developed under one of the advanced technology programs.

Fig. 9--Phantom view of the SNAP-29 radioisotope electric generator, showing the generator and radiator assemblies.

Fig. 10--Radioisotope--gas turbine electric power system (Brayton cycle).

Twenty-five thermal kilowatts is equivalent to a large part of the plutonium-238 that will have been produced up to the 1970's, when delivery is needed. The advantage of this cycle is that, although the weights are not remarkably better than current systems, the efficiency of the Brayton cycle will be somewhere between 20 and 30 per cent instead of the 5 per cent that we now have available. So for a given electrical power, the radioisotope requirement should be only one-fourth to one-fifth of that required with present thermoelectric systems.

B. Reactor Power Devices

Turning to the use of reactors for the higher power devices, Fig. 11 shows the SNAP 10A system and heat cycle. The bare reactor is 8 inches in diameter and 16 inches with the reflectors, which serve as the reactor control. To start the reactor the reflectors are rotated. The reactor heats NaK to 1050° F and pumps it with a static pump. This is an electromagnetic pump which generates its own power using the heat contained in the liquid. No external power is required, and there are no moving parts. The liquid flows from the pump into the supporting legs, where the power is generated by thermoelectric conversion and after the temperature drops about 150° F, the NaK returns to the reactor for more heat.

The thermoelectric conversion system for SNAP 10A which is shown in Fig. 12 represents another improvement in thermoelectrics. The liquid metal from the reactor flows through the tubes, transferring heat to the thermoelectric units. Each unit has its own radiator plate for getting rid of excess heat.

A SNAP 10A unit was launched in April, 1965, and operated in space for 43 days, generating a little over 500 watts of electrical power. It was designed to operate for a year. The satellite voltage control system failed after 43 days, and the reactor shut itself down as it was designed to do.

An exact copy was operated on the ground for 10,000 hours in California. The system was designed to be passively self-controlling, i. e., after the initial start up and stabilization of performance, the whole control system was turned off; and the reactor operated on its negative temperature coefficient for the full one-year design life, dropping in average NaK temperature by about 20 to 30 degrees during that time and generating a minimum of 500 watts in the process. However, the control system could be turned back on and more reactivity inserted by means of the control drums at any desired time. After operating one year on the ground in the test system, we did insert some reactivity just to prove we could get it back up to the 1050° F initial condition. It worked, and we operated it for another two months and then shut it down. The system was taken apart, and it was apparently as good as when it started.

This SNAP 10A system was a zirconium hydride fuel matrix with uranium-235 fuel. The big brother of the SNAP 10A is the SNAP 8. Whereas the SNAP 10A reactor in that application operated at about 40 kW thermal, SNAP 8 can put out about 600 kW.

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Fig. 11--SNAP-10A reactor power system and heat conversion cycle.

Fig. 12--Advanced direct-radiating thermoelectric concept for the SNAP-10A system.

Figure 13 shows the next reactor of the SNAP 8 program. We ran an experimental reactor at full power for 10,000 hours. When we took the system apart, we found cracked fuel elements. We believe we now know why they cracked and how to prevent them from cracking, and we are now assembling the core which will be operated with complete space-configured reactor paraphernalia. The reactor location is in a pit about five stories down. An expanded view of the reactor and experimental shield is also shown. The piping around the pit takes the heat from the reactor into a heat dissipation cycle which will probably be an air blower heat exchanger.

The SNAP 8 system power will be about 10-30 kW with the electric systems or 30-100 kW with the NASA mercury turbine system. We hope to run the second SNAP 8 reactor at power starting next spring.

For the high power levels (hundreds to thousands of kilowatts) there are two main approaches to reactor power systems. The first is the one that was called, until recently, SNAP 50. This system is designed to have a bigger and better reactor and a higher temperature power conversion system than SNAP 8. The cycle is illustrated in Fig. 14. A fast reactor, powered with uranium nitride fuel, operates at a temperature of 2200° F. Heat is transferred by molten lithium to the heat exchanger (boiler) where steam at 2150° F is produced to drive the turbine generator. The cycle has a separate liquid-loop radiator. There is no immediate need for this system, so we are now only in a research phase learning to use the materials required for the high temperatures that will be needed.

Systems for the power levels that will be useful in the various types of missions under consideration are also being studied. These include the advanced auxiliary power, i.e., station-keeping electrical power requirements and the advanced electrical propulsion requirements of the 1980's.

The other advanced high power concept is the in-core thermionic reactor in which the emitter is heated by reactor fuel in the center of the diode. The emitter is a metal surface which boils off electrons, resulting in a flow to a colder collector plate. The electron flow is the electrical power. Thermionics offers the advantage of no moving parts for reliability, but has the disadvantage of requiring high temperatures. Whereas we have been operating it at 1000 to 1300 degrees in our hydride reactors, and the Rankine system will require 2000 degrees, the emitter of a thermionic device for good performance should operate above 2600° F, preferably about 3000° F.

We are now in the stage of building fueled diodes, learning to use them and learning to build them reliably. The next step (which we hope will be in the early seventies) will be to put these diodes together in one of several possible configurations for operation as the core of a reactor. Then, in concept, we would have a pot out of which would come electric power. The cycle will require a radiator into which liquid metal will flow to be cooled; but it will be able to operate at quite high temperatures; and this is one of the advantages of the thermionic system.

A summary showing the growth of isotopic space power technology is given in Table II.

Fig. 13--SNAP-8 ground test facility. Cutaway of system with expanded view of reactor and experimental shield.

Fig. 14--Fast reactor--advanced Rankine electric power system.

TABLE II
Isotopic Space Power Technology Growth

	<u>Power/Unit Electric Watts</u>	<u>Efficiency Per Cent</u>	<u>Specific Power Pounds/Watt</u>	<u>Available</u>
SNAP 9A, SNAP 19	3 - 30	4 - 5	1.0	Today
SNAP 27	50	4 - 5	0.6	1968 (Apollo)
SNAP 29	500	4 - 5	1.0	1970 (Ground Test)
Isotope - Improved Thermoelectric	--	7 - 9	0.3 - 1.0	1970 (Ground Test Capability)
Isotope - Organic or Mercury Rankine	2,000 - 10,000	10 - 20	0.3 - 0.5	Early 70's (Ground Test Capability)
Isotope - Brayton	5,500	20+	0.15 - 0.25	Early 70's (Ground Test)

The present systems, SNAP 9A and SNAP 19, are operating at about 4 to 5 per cent efficiency; and they weigh about 1 pound per watt. The best systems under development for isotopes will, hopefully, operate at higher than 20 per cent efficiency and will weigh between one-fourth and one-half pound per watt. The target date for these is the early seventies.

The growth of reactor space power technology is presented in Table III. The SNAP 10A and SNAP 8 units of the current systems can produce power up to about 50 kilowatts and will weigh about 300 to 1000 pounds per kilowatt. The advanced liquid-metal and the in-core thermionic reactors will put out megawatts and will weigh as little as 20 to 100 pounds per kilowatt. The weight of reactor systems is always highly dependent upon the geometry of the system. In space one tries to use shadow shielding for the reactor. There are no atmosphere or protruding objects as on earth to scatter neutrons back into the payloads, so we build a shadow cone which hides the reactor from the personnel or devices to be protected.

If the spacecraft is long and narrow, a light shield results because the frustrum of a cone which contains the shield is a small volume. All of the low-weight conceptual systems (20 pounds per kilowatt) are based upon a long, narrow, optimized geometry.

TABLE III

Reactor Space Power Technology Growth

	Fuel Surface Temperature °F	Power Electrical kW	Weight Objective Pounds/kW
SNAP 10A - 8 Gas-Cooled Reactor w/Brayton	1300 2200 - 2700	Up to 50 100 - 200	300 - 1000 100 - 200
Liquid Metal Reactor w/Potassium Turbo- generator	> 2000	100 - 10,000	20 - 100
In-Core Thermionic Reactor	2800 - 3400	100 - 10,000	20 - 100

III. Space Propulsion Systems

Turning now to space propulsion systems especially for the advanced high-energy missions of the future, one of the most important factors to be considered in the rating of a propulsion system is the specific impulse. Specific impulse is a number expressed in seconds which tells the thrust in pounds that can be generated by a given propulsion system for a given flow rate in pounds per second of propellant.

Since everything used in propellant systems after the system is in space has to be launched initially from the earth, the propellant requirements for some of the complex advanced missions of the future are quite large. Take a manned vehicle with large mass, that has to go all the way to Mars, for instance. It must leave the earth from a parking orbit, correct its position and its direction along the way, and slow down at Mars to do useful things there. Then it must inject itself back into an earthbound trajectory, make course corrections along the way, and finally slow itself down for recovery at the earth. The entire operation requires a large amount of contained propulsion capability.

As for conventional chemical systems, the best of them mixes oxygen and liquid hydrogen in the rocket to give a specific impulse of about 440 seconds, i. e., 440 pounds of thrust per pound per second of propellant flow. The nuclear rocket, because it can use H_2 , a light-weight molecule, can get much more effective use of the propellant for a given temperature. Specific impulse feasibility on the order of twice the 440 value is being demonstrated for the nuclear rocket system that is now being tested.

This is the advantage of the nuclear rocket. A typical manned interplanetary mission of the future that is in the process of being planned (the manned Mars mission) would require with chemical propulsion about 6,000,000 pounds of propellant to be launched into earth orbit. Using a nuclear rocket about one-third that much would be required. When you consider that a Saturn V, which is our largest launch vehicle, costs about one hundred to two

hundred million dollars per launch and puts up about a quarter of a million pounds, this saving amounts to tens of Saturn V's or billions of dollars per interplanetary mission.

The evolving NERVA engine is shown in Fig. 15. The reactor is about the size of an office desk--4 or 5 feet in diameter. The one that we are now starting to develop will have a thrust of 200,000 to 250,000 pounds and a power of 5,000 MW--roughly the power of Hoover Dam or the installed electrical capacity of the country of Denmark. It has a chamber temperature of about 4500° R which is about the temperature we are working on today.

Figure 16 shows this engine incorporated into the whole rocket. The reactor has a graphite fuel matrix. The hydrogen is contained in the large tank shown in the picture. It flows first through a pump, then into the walls of the nozzle for a regenerative heating cycle. The cold liquid hydrogen cools the nozzle shell and then flows into the reactor, through the reactor and exits the reactor at 4500° R. The turbine that is required to operate the pump is driven by a small bleed of gas from within the high-temperature plenum.

Several reactor and engine tests have been conducted. A typical test is illustrated in Fig. 17. It takes only about 30 minutes of total propulsion time for each reactor in a manned Mars mission. Several reactors have been run for 10 to 40 minutes under conditions that will give, in the large thermal reactor which we are now developing, the 800-850 seconds of specific impulse that is wanted. Hydrogen is pumped into the nozzle shell first and from there through the reactor and out vertically. All together eight or nine reactor tests have been run, mostly at power outputs of about 1100 MW.

The propulsion module concept with the NERVA engine behind a large tank of propellant is shown in Fig. 18. In these views modules are individual or clustered for use in the several typical candidate missions shown. The basic module can be used with a fairly small tank of hydrogen for a lunar direct mission or for an unmanned space probe. For a manned Mars flyby one NERVA engine is used.

The manned Mars landing requires much more velocity change. Five NERVA modules are planned: three for the first stage which would be the stage which leaves the earth orbit, one for the slowing down at Mars, and one for returning to earth.

In summary, several nuclear power systems have been used in space. Nuclear systems technology is now competitive with solar systems in certain specialized space applications. We are now developing a family of long-life isotopic and reactor power systems which will supply up to 50 kW for use in the early to mid-seventies, and we are working on the better technology required for systems in the more distant future. This technology effort is concentrated in several areas designed to give us a higher temperature capability and higher efficiency power conversion for better performance and higher reliability.

In the nuclear rocket program we are now preparing to enter into a program to develop a NERVA engine, and in due time, a nuclear propulsion module--a complete nuclear stage--to be used in a wide variety of space missions.

NERVA ENGINE

- THRUST
200,000 - 250,000 LB.
- POWER
4,000 - 5,000 M.W.
- CHAMBER TEMPERATURE
4,500°R
- SPECIFIC IMPULSE
825 SEC.

NASA NPO67-49
REV. 1-23-67

Fig. 15--The evolving NERVA engine.

Fig. 16--Propulsion module concept using NERVA engine.

Fig. 17--Stationary NERVA reactor and engine test system.

VERSATILITY

BASIC PROPULSION MODULE

NASA NPO 65-1874
12-14-65

Fig. 18--Proposed propulsion systems for various missions utilizing basic NERVA module.

DISCUSSION

QUESTION: Mr. LaFleur, can you comment on the use of thulium in isotope powered devices?

ANSWER: There are two isotopes of thulium that are potentially attractive. Thulium-170 with a half-life roughly like polonium-210, about 128 days, is attractive potentially for shortlife missions. It normally requires more shielding than polonium-210 but may not under some conditions of use. Until recently, it has not been available because stable thulium-169 from which it is made has not been available. Today, there are companies that claim they can furnish enough thulium-169 to make it possible to make significant quantities of thulium-170. This thulium-169 is apparently a byproduct of the color television industry. One of the big advantages, as compared to polonium-210, is that it has a stable high-temperature oxide form that makes it potentially safer.

Thulium-171 resulting from adding a neutron to thulium-170 with about a 2-1/2 year half-life is almost an ideal isotopic fuel but is practically impossible to generate and to separate economically.

QUESTION: Is there any concern with the radiation damage in the thermoelectric materials themselves?

ANSWER: Not much. Of the two classes of thermoelectrics--the lead telluride which is good at low temperatures and the silicon-germanium at higher temperatures--the silicon-germanium is more prone to damage than the lead telluride. Both telluride and silicon-germanium are quite impure as compared to the semiconductors used in transistors. For that reason they can sustain a large amount of radiation reactions in their matrix without deteriorating in performance.

If there are transistors in the payload, the shielding requirements are orders of magnitude lower for the generator than for the payload.

QUESTION: You showed the graphite reactor in a propulsion system. What is the status of the competing or advanced fast reactor propulsion system?

ANSWER: We could possibly get higher performance with a tungsten-family fuel. However, it would not be much higher and would be much more expensive to develop. Because we have had so much experience with graphite, we have not started a large-scale program with tungsten.

Argonne did a lot of good work a few years ago to develop techniques for using the tungsten fuels but has no program right now and no plan to leave the graphite and start with the faster reactor systems.

ASTRONUCLEAR ENGINEERING EDUCATION
at the
UNIVERSITY OF VIRGINIA

J. L. Meem
University of Virginia

The title of this paper, Astronuclear Engineering Education, is taken from the subject of two short conferences held several years ago and sponsored by the Nuclear Engineering Division of the ASEE and the Division of Nuclear Education and Training of the AEC. The first conference was held at Purdue University, April 13-15, 1964, with Philip Powers as Chairman; and the second was held at the University of Florida, March 8-10, 1965, under the chairmanship of Robert Uhrig, who reported the results of these conferences in a paper.¹ A conclusion of both conferences was that education in the nuclear space field was needed at the universities. It was also concluded that research of a fundamental nature at the universities was needed to support the educational programs. Unfortunately, the sources of financial support for such research programs was uncertain several years ago and still is today. I would like to describe the development of the program in nuclear space power and propulsion at the University of Virginia, which is probably typical of the development of similar programs at several other schools.

Most nuclear engineering graduate students at Virginia are interested in nuclear power plant technology. However, at the doctoral level the student is permitted a rather wide choice in choosing his major area of emphasis. One such option is in nuclear space propulsion. A course in nuclear rockets is offered in the Nuclear Engineering Department. In cooperation with other departments, the student may select topics in magnetohydrodynamics (MHD) and other forms of energy conversion, as well as courses in advanced space propulsion.

Historically, the program developed at Virginia because of the interest in nuclear space propulsion which resulted from my previous work in the Aircraft Nuclear Propulsion program. I attended the summer institute on nuclear rocket propulsion sponsored by the National Science Foundation at the University of Florida in 1962. It was at this institute that I was able to learn enough about the various aspects of a nuclear rocket to enable the University to offer a reasonably well integrated one-semester course on the subject. The course was offered for the first time in 1963, again in 1965; and there was sufficient demand (six students) to offer it again in the fall of 1966.

Three students have chosen nuclear rockets as an area of research, and a first year graduate student has expressed interest in the area. One

student dropped out after completing the master's program, but we anticipate that the other two working in the field will go on to the doctorate.

Since the nuclear rockets course forms the backbone of the program, it should be described in some detail. The student should have the necessary first year graduate program including elementary reactor theory, shielding heat transfer and fluid flow behind him. The following is a summary of the course:

Summary of Nuclear Rockets (NE 615) Course

Chapter I. Introduction (1/2 week)

History of nuclear flight and nuclear space propulsion and power; NEPA, ANP, ROVER, PLUTO, RIFT, NERVA, SNAP (even), SNAP (odd); thermoelectric concepts, thermionic concepts, MHD.

Chapter II. Fundamentals of Rocketry (1 week)

Thrust, specific impulse; the VIS-VIVA equation, orbital velocity, escape velocity; thrust equation in free space, thrust equation with gravity; the mass ratio; comparison of chemical and nuclear rocket performance, examples.

Chapter III. Conceptual Design of Nuclear Rockets (1/2 week)

Nozzle, rocket engine, turbopump and feed system, hydrogen tank, payload, shielding; the overall vehicle. Solid core reactors; graphite, tungsten-water, fast tungsten. Advanced concepts; gaseous cores.

Chapter IV. Powered Flight in Space (2 weeks)

Distance traveled in free space at constant thrust.
Distance traveled in free space at constant acceleration.
Distance traveled in vertical takeoff from the earth's surface at constant thrust.
Determination of the average value of g, the acceleration of gravity during flight.
Principles of staging.

Chapter V. The Rocket Nozzle (2 weeks)

Fundamental thermodynamic relationships; choked flow; throat area and area ratio; thrust coefficient and thrust correction factor; characteristic velocity.

$$P_c = \text{Const} \times m \sqrt{T_c}$$

Chapter VI. Heat Transfer and Fluid Flow (3 weeks)

Description of average channel and power distribution in the unit cell; enthalpy and temperature rise of coolant; temperature at coolant channel wall and in the fuel matrix; pressure drop; Mach number in a coolant channel and pumping power; the FORHEX code.

Chapter VII. Nucleonics (3 weeks)

Two group theory and the ROVER code; reactor startup kinetics; temperature and hydrogen reactivity feedback; nuclear kinetics in the FORHEX code.

Chapter VIII. Shielding and Propellant Heating (3 weeks)

Propellant heating in the tank; gamma dose at payload while emptying the tank; the tapered tank; energy release in the core during operation and on shutdown; shutdown coolant requirements; propellant consumption during startup; full power operation and shutdown; space radiation and crew shield weight; design and weight of the shadow shield.

Chapter IX. Vehicle Design and Performance (1 week)

Postulated mission; weight of propellant and structure; design and weight of engine; shield weight estimates; total weights and performance.

Total (16 weeks)

The FORHEX code is mentioned several times. This was a code developed by R. J. Bohl at Los Alamos to analyze the startup transients of the KIWI reactors. The pump, the nozzle, the reflector and the core are each treated as a single lumped channel. Some quite gross simplifying assumptions are made which work surprisingly well. Temperatures, pressures and densities of the hydrogen at inlet and outlet of the nozzle, reflector and core are predicted as a function of time during startup. From the nucleonics part of the code, the reactor power, the temperature and hydrogen reactivity feedbacks, and the control rod motions are calculated. Again, some quite gross simplifying assumptions are made that work quite well. We go through the code in detail in class and assign different problems to the students. It is an excellent educational tool.

Returning to the research that is being done, a few words should be said on the financial support or lack of such that the University has received. A graduate student at the University of Florida had also attended the Advanced Institute on Nuclear Rocket Propulsion and upon completing the requirements for the master's degree at Florida joined me at Virginia in the fall of 1963 to follow some mutual research interests. A proposal for "Analog Simulation of Nuclear Rocket Dynamics" was prepared and submitted to the Office of Contracts and Research Grants of NASA in November, 1964. That office sug-

gested the proposal be submitted to the Division of Nuclear Education and Training of the AEC. After bouncing back and forth between NASA and AEC several times, it was suggested that the proposal be submitted to the Space Nuclear Propulsion Office, which it was.

The proposal was for a two-year research contract amounting to about \$120,000 of which \$72,000 was for analog computer equipment to supplement the computer at the University. It was then learned that an analog computer facility at NASA's Lewis Laboratory was being assembled at a cost of between one-quarter and one-half million dollars which would be able to perform analog simulation of nuclear rocket dynamics. This was pretty stiff competition. There were few universities offering advanced training in nuclear space applications. Accordingly, SNPO should support a proposal from a university on the basis of encouraging graduate student training in the field as well as on the basis of obtaining useful results of the research. The arguments were received with sympathy but not much else. The AEC has its Division of Education and Training, and NASA has its Office of Contracts and Research Grants with funds for supporting university research. SNPO has no such funds and can support only direct mission-oriented research contracts. We didn't get a contract, and the graduate student left to work for Westinghouse Astro-nuclear Laboratories.

I spent the summer of 1965 as a Visiting Staff Member working for Los Alamos at the Nuclear Rocket Development Station at Jackass Flats, Nevada. While there, much of my time was spent on nuclear safety, and it was apparent that there was a great deal to be done in the area of accident analysis for nuclear rockets. In particular I became interested in the problems of the liquid hydrogen insertion accident, the water insertion accident and the deliberate nuclear destruct for the ROVER-type reactor. In September, 1965 a proposal for Accident Analysis of Nuclear Rocket Reactors was submitted to SNPO. This proposal was definitely mission oriented. The proposal was for about \$19,000 for a period of one year with the work to be done on the digital computer at Virginia. Depending on the progress during the first year, it is hoped that the contract will be continued. Again, SNPO advised that funds were not available to support work of this type at a university; but that it might be possible to obtain a contract directly with Los Alamos. Members of N Division at Los Alamos were contacted, and we were told that funds were unavailable for an outside contract; but if funds became available, the work probably would be done in N Division. Again, we didn't get a contract.

By this time another graduate student was deeply interested, and we decided to undertake the project without financial aid. He has written a code called IGNITE which includes a transient heat transfer analysis of the nuclear rocket core which is the most accurate description available. This is a true transient heat transfer code and not a quasi steady state heat transfer code like many others. Coupled with this he has written an accurate nuclear kinetics code for the rocket core. We use the FORHEX code from Los Alamos to predict hydrogen pressures, temperatures and densities from the pump through the nozzle and reflector to the core inlet chamber. We have analyzed the KIWI-B4E runs on which we have some unclassified data, and we can predict the KIWI-B4E runs with remarkable precision. We are now ready to vary the startup conditions and investigate such things as the hydrogen pump runaway to ascertain under what conditions we could obtain a liquid hydrogen insertion accident.

The problem now is digital computer costs. A single run on the KIWI-B4E takes several hundred dollars of computer time. In the past two years approximately \$10,000 of machine time has been used. The graduate student is an AEC Fellow, and we have been able to pay about \$3,000 out of his cost of education allowance. The other \$7,000 of machine time has been donated by the Computer Center, and obviously the Computer Center is beginning to resist. We can only get time on the midnight shift, and that time is becoming increasingly unavailable.

Another marked disadvantage of working alone on a project like this is the lack of access to classified data. Without a research contract clearance is not available for the student, and he does not have access to information on Phoebus. With some changes our code could perform a transient analysis and predict accident conditions on Phoebus. Instead we must work with data from the now obsolete KIWI. While the results are worthy of a doctoral dissertation, it is frustrating to feel that the results are not of immediate interest in the field. Having these results and an excellent transient code, it may be possible to go back again to SNPO and possibly Los Alamos with a contract proposal.

An interesting offshoot of this research project led to the involvement of another graduate student in predicting the proper nuclear parameters for the reactor kinetics. Assuming a chopped cosine power distribution along the axis of the core, we can obtain the temperature, pressure and density profiles of the hydrogen coolant as it passes through the core. The transient code IGNITE will also give us the temperature profile in the graphite fuel matrix along the axis of the core. As the hydrogen density increases with increased flow rate, there is a positive hydrogen coefficient of reactivity; and as the graphite fuel matrix heats up, there is a negative temperature coefficient. These reactivity coefficients affect the rate of increase in power during startup. Also, the nonuniform temperature and hydrogen density distributions cause a deviation from a pure chopped cosine power distribution. We are using the DTF code to predict these nuclear parameters. So far we find that the deviation from a chopped cosine power profile is not pronounced. This is fortunate, for otherwise, we would have to iterate back and forth between DTF and IGNITE again increasing the costs of computer time. In computing the temperature and hydrogen reactivity changes, we do not obtain agreement with those calculated from the experimental runs on KIWI-B4E. We obtain considerably smaller reactivity changes from temperature and hydrogen individually. Since the temperature coefficient is negative and the hydrogen coefficient is positive, it is the difference that is important; and we do obtain a net reactivity change of about the same magnitude as was found experimentally on KIWI. We may have stumbled on some information of fundamental importance to the ROVER project although the results are pertinent to the obsolete KIWI reactors. The results will appear in a master's thesis.

In summary we have an option in Nuclear Space Propulsion in the Nuclear Engineering Department at Virginia which is designed to meet the needs of the growing nuclear space industry. The success of the program is limited only by the research support we can obtain. The AEC encourages university research through its Division of Education and Training, and NASA does the same through its Office of Contracts and Research Grants. However, the Space Nuclear Propulsion Office which is joint between the two parent

agencies does not have the means for promoting education and research at the universities.

About 1950 the Oak Ridge School of Reactor Technology was formed, largely to meet the needs for trained manpower in the nuclear submarine program. Similar needs for trained manpower in the nuclear space field exist today. The universities were not prepared to give the training in 1950, but they are prepared today. They are limited only by the lack of support for graduate student research.

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FUSION AND PLASMAS*

J. L. Dunlap
Oak Ridge National Laboratory

I. Plasmas

By definition a plasma is a collection of free electrons and ionized atoms. At anything other than small densities, coulomb forces dictate an auxiliary criterion that there be nearly equal numbers of positive and negative charges.

The early Greeks maintained that there were four basic materials in the universe: earth, water, air and fire. As it developed, they were not so very wrong; their classification compares well with the four states of matter: the solid, liquid, gaseous and plasma states. We are concerned here with the plasma state, which the ancients observed in only three forms: fire, lightning or the aurora. In modern times it is observed in a multitude of forms.

This sequence in the states of matter is a consequence of the average kinetic energy of a structural unit. Solids remain solids only so long as the average kinetic energy of the molecules does not exceed the binding energy of the crystals. If the average kinetic energy is raised somewhat above this level, the crystal bonds are broken; then the material becomes liquid or goes directly to the gaseous state. In a similar fashion a liquid is changed to a gas if the average kinetic energy of molecules is large enough to overcome the van der Waal forces. Gases become plasmas when the kinetic energy exceeds the ionizing potential--this generally occurs at a few eV, so if we speak in terms of kinetic temperature, plasma is produced at about 10^4 or 10^5 °K.

If we follow along in this sequence of states, the fifth state of matter does not occur until the average kinetic energy is sufficient to break nuclear bonds. This step requires a minimum of some 2 MeV kinetic temperature. To study the properties of this fifth state, one requires a significant collection of free nucleons and electrons, not simply the single isolated ones available from nuclear physics experiments. So this state remains unexplored and plasma presently represents the final accessible state.

The significant differences between the behavior of plasma and the closest accessible state, gas, are primarily due to two considerations: (1) plasma behavior is dominated by long-range coulomb forces rather than short-range molecular forces as in a gas, so strongly coherent behavior is possible; (2) plasma can carry electrical current whereas gases are insulated. As a consequence of these considerations, the most characteristic and specta-

*Research sponsored by the U. S. Atomic Energy Commission under contract with the Union Carbide Corporation.

cular plasma behavior is exhibited by plasma in a magnetic field.

Let us return for a moment to the energy range appropriate for the plasma state. This range was from a few eV to 2 MeV. The point to be stressed here is that this is by far the widest energy range for any of the four accessible states. It is perhaps then not surprising to find estimates that more than 99.9 per cent of the matter in the universe is ionized and is, therefore, in the plasma state. It is as though plasma is the normal state of matter, and that almost everywhere except on earth matter is either hot enough or is diffuse enough to be plasma.

Since so much of the universe outside the earth is plasma, it is not surprising that the study of natural plasma has formed an important part in astrophysical research since about 1900. Here one could cite the many plasma effects contained in observations of sun spots and of solar prominences and corona. Plasma also enters in less direct celestial measurements. For one example, radioastronomy is affected by the plasma dielectric constant. For another, there is evidence that much of the interstellar material is ionized, and that interstellar magnetic fields are of the order of 10^{-5} gauss. Thus there is the possibility of interstellar plasma activity; activity that is locally weak because of low density, but which can be active over immense distances. It has been suggested that the electric fields associated with this activity may be the acceleration mechanism for cosmic ray particles.

Closer to the earth there are other significant natural plasma effects. The ionosphere is the ionized layer in our outer atmosphere, and the structure of this layer and its effects on transmission of radio waves have been studied to a significant extent. The magnetic storms, which are spectacular disturbances of the earth's magnetic field, are attributed to plasma beams from the sun. These same beams are associated with the aurora. Finally, there are the van Allen belts which consist of charged particles trapped in the earth's magnetic field.

In turning from the study of natural plasmas to the study of laboratory plasmas, the beginnings of these studies were the result of passing electrical current through gases at reduced pressures. In fact it was in writing of some of his discharge experiments that Crookes said in 1897, "the phenomena in these exhausted tubes reveal to physical science a new world, a world where matter may exist in a fourth state..." This statement appears to be the first identification of the material created in gaseous discharges as the fourth state of matter.

These initial discharge studies were made with quite elementary electrical and vacuum equipment, and the plasmas which were created were only partially ionized. The discharges still contained a lot of neutral gas, and as a consequence the behavior of the medium was dominated by collisions between charged particles and neutrals. There was abundant emission of atomic and molecular spectral lines. The study of these lines introduced atomic and molecular physics. Aside from the spectroscopic studies, there was considerable interest in the electrical properties of the discharge itself. It was eventually recognized that the properties of weakly ionized gases were dominated by complex collisional processes such as ionization, excitation, recombination, inelastic and resonant scattering--in fact most of the fundamental processes that are still the subject of atomic collision work.

The laboratory study of plasma as a state, however, was not possible until the development of the more adequate electrical and vacuum equipment that was required to reproducibly make highly ionized media. These became available about 1920; and shortly thereafter, Langmuir developed a number of diagnostic techniques still in common use, developed the basic theory of an ionized gas and gave the medium the name "plasma." During the period from 1920 to the beginning of the second World War, the subject developed considerably due to the efforts of Langmuir, Tonks, von Engel, Loeb, Penning, Townsend, Rogowski and others.

More recently, plasma research has been spurred by the desire for technological applications which range from specific problems such as rapid switching and high current rectification to some exciting technological possibilities. Two of these, ion space propulsion and direct conversion, are the subjects of other papers at this Conference. The third, the possibility of controlling the energy release of nuclear fusion reactions, is considered in the following section.

II. Fusion Research

In the United States work toward a fusion reactor is supported primarily by the AEC. This work is concentrated at four laboratories: the Los Alamos Scientific Laboratory, the Lawrence Radiation Laboratory, the Princeton Plasma Physics Laboratory and the Oak Ridge National Laboratory. The University of California is the operating contractor for LASL and LRL; PPL is operated by Princeton University, and ORNL is operated by the Union Carbide Corporation.

The AEC support for this work has averaged between 20 and 30 million dollars a year for the past 10-12 years. This is about 20 per cent of the world's fusion budget. Russia is the most active followed by the U. S. and Britain. The European Atomic Energy Community (EURATOM) is also quite active, particularly in France and West Germany.

The fusion reaction having the most favorable cross section is the DT reaction,

Even this cross section is insignificant below energies of some 10's of keV, and even there it is much smaller than the cross sections for ionizing collisions and for cumulative small angle coulomb scattering.

From these considerations we see that the fuel must be a fully ionized plasma with mean collision energies of 10's of keV, and that it must be confined against coulomb scattering. For a working reactor an appreciable density of plasma must be confined for an appreciable time. There is a simple relationship between the density and mean confinement time required,

$$n \tau > 10^{14} \text{ cm}^{-3} \text{ sec.}$$

This is known as the Lawson Criterion. If it is not satisfied, the plasma will radiate away more energy from the reaction volume than it produces by thermonuclear reactions.

The present plasma experiments fail to satisfy the Lawson Criterion by orders of magnitude. They are almost without exception limited by faulty confinement, and the understanding of this faulty confinement is the basic research goal. It is obvious that the desired plasma cannot be contained by contact with material walls, and the only confinement schemes that show significant promise involve magnetic fields. Magnetic confinement is possible because, to a first approximation, charged particles follow helical paths along magnetic field lines. This introduces the problem of what to do about the ends of the reactor. If the field lines leave the reactor, the plasma may follow them out. There are two solutions to this problem.

1. One is to crowd the field lines together at the ends to create regions of high magnetic field that will reflect back plasma particles that don't have too large a velocity component parallel to the field lines. This would be called an "open-ended" or "magnetic-mirror" machine.
2. The other solution is to produce a field whose lines never leave the reactor. This would be "closed geometry" and generally means that the machine must have the shape of a torus.

The following are three facets of confinement which, though not absolutely necessary, are clearly desirable.

1. Single plasma particles must be confined.
2. An equilibrium must exist. Generally, there will be nonzero charge densities and current flows due to the plasma. These produce electric and magnetic fields which along with the external magnetic field exert forces on the plasma. There must exist a plasma configuration in which these charge densities and current flows are in equilibrium with these forces.
3. The equilibrium must be stable. This implies that if a small perturbation is made in the plasma, the plasma must return to or oscillate about the equilibrium state.

The questions that arise about the necessity of these three conditions relate primarily to allowable time scales for plasma loss. For example, it can be argued that the equilibrium need not be completely stable if instabilities do not limit plasma accumulation below densities appropriate for reactors.

We will review first the status of research in open-ended geometry and will use as the first example the simple magnetic mirror field shown in Fig. 1. There has been extensive work in this field configuration. Typical central field values (B_c) have been 5 to 25 kG with typical ratios of B_m/B_c of 2 to 3. Various means of creating plasma in these traps have been used. One method is to inject an energetic beam of H_2^+ ions and to use dissociation

Fig. 1--A simple magnetic mirror field. Field lines along with an approximate particle trajectory are shown in the upper sketch. A variation of axial field strength is shown in the lower sketch.

reactions ($H_2^+ \rightarrow H^+ + H^0$ or $\rightarrow H^+ + H^+ + e$) to trap resulting energetic protons. A second method is to inject energetic atomic hydrogen beams and to use the $\underline{v} \times \underline{B}$ force or collisions to ionize these atoms. Other researchers have fired in plasma from guns, using pulsed rather than dc magnetic fields to trap and to heat the plasma.

In this simple mirror field, as in more complicated forms of open-ended traps, it is not difficult to satisfy the requirements of single particle confinement and existence of a plasma equilibrium state. There remains the important question of plasma stability.

It was found that this simple mirror does not provide stability against a low frequency magnetohydrodynamic (MHD) mode called the "flute" or the "interchange instability," which involves a surface perturbation of plasma in the form of a flute or a bump (Fig. 2a). The influence of a magnetic field gradient is to cause ions and electrons to precess in different directions about the magnetic axis; so the two charge components separate to the edges of the perturbation and by doing so establish the $\underline{E}$ field shown in Fig. 2b. If the magnetic field has a negative radial gradient, i.e., if the field lines are concave toward the center as in the simple mirror, the $\underline{E} \times \underline{B}$ drift of plasma is outward. The situation is then unstable because the perturbation grows; plasma is dumped to the wall, and accumulation is limited by this instability.

Obviously what was needed was a configuration with field lines concave away from the center. Here the $\underline{E} \times \underline{B}$ drift from the flute perturbation would be directed inward, and the plasma would be stable against this mode.

A way of obtaining the desired configuration while retaining single particle confinement is to combine the mirror field with a multipole field from a set of linear conductors running parallel to the axis of the mirror coils. For example, viewing the mirror coil system from one end, add a quadrupole set as shown in Fig. 3. With appropriate currents in the mirror and quadrupole conductors, this system provides a nonzero field minimum at the center and fields which increase as one moves outward from the center either radially or toward the mirrors. Such field configurations are called "minimum-B."

The first experiments in this geometry were performed by Ioffe of the USSR about 1961. He showed that plasma as a minimum-B trap is stable against the flute. The results have been confirmed by numerous investigators since that time.

The discovery of these minimum-B traps constituted an important step in open-ended research. Most of the current and projected experiments incorporate this idea. Figure 4 shows the general scheme for use of such a trap in connection with energetic neutral injection. Here an array of four coils makes the quadrupole contribution to the magnetic field. Currently there are two such experiments in the USSR and one in England. The Lawrence Radiation Laboratory has done experiments with similar coil configurations and is now pursuing experiments with coils which look like the seams on a baseball. These baseball coils also set up a minimum-B field. Oak Ridge is planning a series of experiments in mirror-quadrupole geometry using electron cyclotron plasmas in the trap as an ionizing target for the fast neutral beam and for enhanced plasma stability.

Fig. 2--Flute instability. The perturbation is shown in cross section in (a). Distribution of surface charges and the resulting electric field are shown in (b).

Fig. 3--A mirror-quadrupole conductor system (end view).

Fig. 4--General scheme for energetic neutral injection into a minimum-B trap. A mirror-quadrupole trap is shown here.

Though flutes do not occur in minimum-B, hot plasma confined there is far from stable. At this point it is necessary to make a distinction between two general classes of instability.

1. The first class are those like the flute which are at low frequency, cause a motion of macroscopic elements of the plasma and for which stability is expected in minimum-B. These are called macroscopic instabilities and are adequately described by MHD treatments.
2. The second class are called microinstabilities. These require a much more detailed treatment than is possible with the fluid-like equations of MHD. Instead, it is necessary to employ the phase-space distribution functions for electrons and ions along with Maxwell's equations for the electric and magnetic fields. Microinstabilities involve a much wider range of phenomena than do the macroinstabilities, and in general these are not stabilized by minimum-B. They, therefore, constitute the major concern of present open-ended experiments.

Generally speaking, microinstabilities lead to oscillating high frequency electric and magnetic fields which have much the same effect on confinement as collisions between plasma particles. Cross field diffusion (radial losses) should be negligible compared with end losses; so it is end losses that may block open-ended reactor configurations.

There is ample evidence for microinstabilities in minimum-B. Many of the experiments exhibit density limitations because of them. A number of the observed modes have been identified with theoretically predicted unstable modes--enough so that there is confidence in the ability of theory to guide future experimental work. Fortunately, theory suggests the possibility of an adequately stable regime of fusion interest. The goal of present mirror research is to fulfill these theoretically suggested criteria, which involve control of electron and ion distribution functions, boundary conditions and magnetic field geometry.

Let us consider closed systems. My own experience has been limited to open-ended geometry, and therefore, I am relatively unfamiliar with closed systems. There has not been nearly as close connection between experiment and theory in closed systems; so the unification of the results of different experiments, which this connection is expected to supply, is not so evident.

Magnetic field configurations in toroidal geometry are quite complicated. This is illustrated by Fig. 5 which shows one configuration of the Princeton Model C Stellarator. The "regular magnetic field coils" are simple solenoid windings. These do not provide confinement since the field is stronger at the inside of the turnarounds than at the outside, and particle drifts to the wall would result. It is necessary to add the "corkscrew" windings in order

Fig. 5--One coil configuration of the model C Stellarator.

to introduce rotational transform--to provide that a field line does not close upon itself after passing once around the torus but instead is rotated through some angle. The rotational transform is made a function of radius in order to introduce shear, which has a stabilizing effect on some instabilities.

A variety of methods of generating plasma have been used in Model C. All of these experiments have exhibited "pumpout," an anomalous diffusion of plasma to the walls. There is evidence for both macroinstability and microinstability. The theoretical studies indicate that MHD modes are probably not at fault; furthermore, the microinstability modes are also probably excluded since particle losses from these would be primarily along the field rather than across the field. Certain drift modes of instability peculiar to toroidal confinement are believed to cause pumpout, though it has been suggested that an equilibrium state may not exist. These modes require strong shear for stabilization. They were not theoretically recognized until long after the Stellarator was built, and Model C unfortunately has weak shear.

The assignment of difficulties to these drift modes may be supported by Russian work with their TOKOMAKS, which are Stellarator-like machines. These were fortuitously built with large shear, and loss rates there are only one-tenth those expected from scaling the Princeton results. However, the Russian experiments are extremely difficult to analyze; and the assignment is somewhat uncertain.

Progress in toroidal theory has indicated important stabilizing factors for these modes, and there is now a gradual shift of emphasis toward smaller scale experiments to verify these indications and toward design of new toroidal devices that will incorporate them.

The most promising such devices are the toroidal multipoles which were pioneered at General Atomic and at the University of Wisconsin. These use a number of conductors suspended within the torus, with the direction of current flow the same in all conductors. Figure 6 shows the conductor arrangement and field lines for a toroidal quadrupole. Along the axis of the torus B is a minimum, in fact zero. Near the axis, plasma is confined by field lines which are concave away from the plasma; hence they are stable against flutes. As in the minimum- B mirror machines field lines leave this area, but as particles follow a line out they encounter higher magnetic fields which give a mirror-like reflection to those with sufficiently small $v_{||}$. Those with larger $v_{||}$ follow field lines around to the back side of the conductor, where the field curvature would allow flutes to develop. However, the distance along field lines between these regions of good and bad curvature is small, and if this connection is good enough stability should result.

An obvious difficulty is how to support and supply current to the conductors. So far, the experiments have used mechanical supports; and the plasmas collided with these in a short time. In spite of this the plasmas lasted much longer than would be expected from Stellarator experiments without evidence of strong instability or losses other than these collisions.

Princeton is now designing a large experiment along these lines, one which will use superconducting levitated conductors in order to avoid the collision losses and the ambiguities of interpretation which they introduce.

Fig. 6--Conductor arrangement and field lines of a toroidal multipole.

Two other major experiments in which there is interest on the part of both open-ended and closed geometry camps warrant some description. One of these is the θ -pinch work at Los Alamos. There a single turn sheet coil is positioned around the plasma chamber and energized rapidly by a capacitor discharge. The induced electric field ionizes gas in the chamber to form the plasma which is heated and compressed as the magnetic field builds up. There is no attempt to prevent end losses so the plasma lives for only the length of the field pulse, which is measured in microseconds. Still the density accumulates to about 10^{17} cm^{-3} . The $n \tau$ product is about 10^{12} , a record for fusion experiments. Los Alamos is now securing additional power supplies that will allow experiments with mirrors for end confinement and eventually plan to bend the θ pinch into a torus for full closure.

The second experiment is the Astron at LRL. This is diagrammed in Fig. 7. As the upper sketch shows, external coils set up a mirror configuration. A relativistic electron beam is injected from one end. In going down the field these electrons lose enough energy to an external electrical circuit to insure that they are not lost out the opposite mirror, but rather they remain and reflect between the mirrors for some time. If enough current can be built up with these electrons, the "E layer", the field produced by their circulation will produce the set of magnetic field lines shown in the lower sketch, where one quadrant is represented. Obviously the field configuration is closed at the interior.

If cold deuterium gas is now admitted to the system, it will (in principle) be ionized and heated to thermonuclear temperatures.

The Astron experiments so far have involved building up the E layer. As yet destructive instabilities in the E layer have not been observed, and enough electron current has been accumulated with the present inadequate accelerator to cause a few per cent depression of the central field from that of the external coil set.

I want to return to the basic difference between open and closed systems--the matter of end losses--and the relevance of this difference to reactor economics. If radial instability losses in closed geometry can be controlled, then there seems to be no reason in principle why confinement will not be adequate enough for reactors. You must remember that although I have emphasized the effects of instabilities on confinement (in keeping with current research), coulomb scattering collisions in a stable plasma also affect confinement. The scattering is mostly along field lines. This does no damage to confinement in an adequate closed geometry, and in fact, by making the plasma distribution more isotropic even helps by reducing the chances of microinstability.

However, these scattering collisions do cause end losses in a mirror machine. Estimates of the magnitude of these losses from stable plasma suggest that the possibility of obtaining an open-ended reactor is marginal. It may or may not be possible.

Recently some mirror reactor configurations have been suggested, which when detailed calculations are finished, may result in positive affirmation of the possibility of open-ended reactors. In any event continuation of a

(a)

(b)

Fig. 7--The Astron experiment.

strong mirror effort is certainly indicated by the understanding of high temperature plasma that this effort can contribute, and by the fact that the eventual reactor may well be a hybrid system composed of a mirror section plus end ties to relieve the losses.

It should be clear that there are many technological problems in fusion research. The research is highly demanding of vacuum, power supplies, beam sources, structural design, magnetic field synthesis and coil construction. Perhaps some idea of the magnitude of these problems will be evident from Figs. 8-10 which are photographs of some equipment at ORNL.

III. University Research

It is clear that there is a wide range of interest in plasma and plasma effects on the part of numerous engineering and scientific disciplines, and that as a consequence, plasma research presents fertile opportunities for university participation. Furthermore, the fact that plasma studies simultaneously embrace a wide variety of disciplines makes them particularly valuable in satisfying the primary objectives of university research, the teaching of the general research method.

The four agencies most active in support of plasma studies at universities are the AEC, the Department of Defense, NASA and the National Science Foundation. I have not been able to locate statements of such funding on the part of all these agencies, so I can't really say in what order they contribute. However, B. J. Eastlund of the Division of Research, AEC Headquarters, Washington, has very kindly provided details of the AEC off-site funding that is being made in connection with the fusion program. Most of these funds involve contract support for university research. A breakdown is provided as Table I.

Organizations that receive support for plasma theory work, i.e., support in Category I, are the University of California at San Diego, Cornell University, the University of Michigan, New York University, Stanford University, Stevens Institute of Technology, the University of Tennessee and the University of Texas. The theoretical investigation of microinstabilities is the dominant field in this work. Strong footholds in the areas of quasilinear and nonlinear theory have developed in the past few years, primarily at Tennessee, Texas and the University of California. A variety of topics are pursued at NYU, where 15 senior scientists contribute various fractions of their time to this research, totaling 10 man years of effort. The NYU contract is the largest off-site theoretical contract both in terms of dollars and in terms of man years of senior effort. A small program has begun this year in high-beta plasma research at Cornell University. This is in line with the general intent of intensifying the total Controlled Thermonuclear Research effort in high-beta research. Computer simulation of plasma confinement at Stanford University is a unique element in the list of theoretical areas of study.

Under Category II, plasma experiments at high temperature, support is provided to the University of Wisconsin, the University of Maryland, Stevens Institute of Technology, United Aircraft Corporation, Westinghouse Research Laboratories and MIT. The major element here is the multipole program at Wisconsin. This is the largest of the off-site programs devoted to

Fig. 8--The Oak Ridge DCX-1 experiment which uses 600 keV H_2^+ injection into simple mirror geometry.

Fig. 9--The Oak Ridge DCX-2 experiment. Injection of 600 keV H_2^+ is also used here.

Fig. 10--The 600 keV 1 ampere dc power supply that is used for injection into the DCX-2 trap.

TABLE I

Contract Support for University Research

	<u>GM Y*</u>	<u>SM Y**</u>	<u>\$K***</u>
I. Plasma Theory	13	19	529
II. Plasma Exp - High Temperature	13	12	589
A. Multipole			
B. Theta pinch			
C. Laser-produced plasmas			
D. Hot electron plasmas			
III. Plasma Exp - Low Temperature	25	9	853
A. Discharges			
B. Alkali plasmas			
C. Collisionless shocks			
IV. Atomic and Surface Physics	7	8	335
A. Atomic physics			
B. Surface physics			
	<hr/>	<hr/>	<hr/>
TOTALS	58	48	\$2.3 x 10 ⁶

*GM Y - Graduate Student Man Years Charged to Contract

**SM Y - Senior Scientific Man Years Charged to Contract

***\$K - Thousands of Dollars

Notes: 1. Subcategories are in decreasing order of funding.

2. Data are for fiscal year 1967.

a single type of experiment. It also is the largest contract in terms of numbers of graduate students engaged in the research. For this one contract there are nine graduate man years, three senior scientific man years and a funding of \$215,000 for fiscal year 1967. The study of laser-produced plasmas is another major area of high-temperature research. That at Westinghouse emphasizes early time behavior--the absorption of radiation and the initial plasma expansion. The research at United Aircraft emphasizes late time behavior--plasma confinement in mirrors, cusps and minimum-B fields. A novel program on colliding high-beta plasmas and their confinement in cusps is carried on at Stevens. Hot electron plasmas are studied at MIT and also at Wisconsin.

In Category III the experiments at low temperature, support is given to the Environmental Sciences Service Administration, MIT, the University of Miami, Stanford University, the University of Texas, the University of California, the University of Maryland and Stevens Institute of Technology. The major emphasis is in the area of discharge plasmas. Arcs are studied at MIT, ESSA and Miami. Beam-plasma experiments are performed at MIT and Stanford. RF plasmas are studied at Stanford and Texas. A smaller area is

that of the alkali plasma, on which research projects are underway at California (Berkeley), Maryland and Stevens.

The final category, atomic physics and surface physics, gives support to the Georgia Institute of Technology, the Joint Institute for Laboratory Astrophysics (which involves the University of Colorado), the National Bureau of Standards, Roanoke College and Washington State University. Selected atomic collision phenomena are studied at Georgia Tech and the Joint Institute for Laboratory Astrophysics. Personnel at these sites are also participating in activities of the Atomic and Molecular Processes Information Center at Oak Ridge. The surface physics program has declined recently by virtue of the termination of the two largest contracts in this area, which were at Stanford Research Institute and Westinghouse. However, small university efforts in this area are continuing at Washington State and Roanoke. Vacuum ultraviolet bombardment of solids is studied at NBS.

I am sure that you are interested in the prospects for continuing support in these areas. Insofar as I have been able to determine, the official AEC posture is contained in the document TID-23277 which is entitled "AEC Policy and Action Paper on Controlled Thermonuclear Research." This report is dated June, 1966. It was prepared by the AEC in response to a request by the Joint Committee on Atomic Energy that the Commission develop a report on its controlled thermonuclear program similar to the policy paper prepared on the high-energy physics program. Contained in the document is the report of a panel of leading scientists and engineers who carried out a detailed review of the AEC program on CTR. From the review panel's report is cited the following paragraph:

"Work at the major AEC-supported university plasma laboratories is yielding excellent returns in plasma physics and plasma diagnostic techniques for funds expended. The programs at Berkeley, MIT, Wisconsin and New York University which were reviewed by panel members appear to be especially fruitful, with excellence in output of research results and of gifted plasma scientists and engineers. Work at Stanford and Stevens Institute of Technology is contributing to our understanding of plasma physics."

The Policy and Action Paper continues then to read (now on the part of the AEC):

"Research carried out at university plasma laboratories constitutes an important part of the AEC's program. Among other things, universities clearly play a critical role in the training of scientists and engineers needed to invigorate the CTR effort. While an expanded university program is an integral part of the intensified CTR effort, special attention must be given to maintaining high quality research work, with emphasis on theoretical studies of plasma behavior."

In reading elsewhere in the Action section of the document, we find:

"The action planned by the AEC includes a significant expansion of the AEC-supported effort at universities. Specifically it is planned to expand the support of these activities at a more rapid rate than that at the national laboratories. The university support would extend to basic plasma studies, recognizing that a thorough understanding of the field will be required for achievement of the ultimate objective."

The Policy and Action Paper reviews problem areas in CTR, and for this reason those of you who may wish to investigate the possibility of CTR research funding for your own universities would probably find it useful reading. It is available from the Clearing House for Federal Scientific and Technical Information, National Bureau of Standards, U. S. Department of Commerce, Springfield, Virginia.

REFERENCES

Author's Note: In reviews such as this, there seems to be little or no reason to attempt an original work. The introductory material reflects liberal use of the excellent treatments of W. B. Thompson ("An Introduction to Plasma Physics") and J. G. Linhart ("Plasma Physics"). A recent unpublished review by E. G. Harris was useful in preparing parts of Section II. The primary sources for Section III were cited in the text.

PLASMA PHYSICS EDUCATION
at
PENNSYLVANIA STATE UNIVERSITY

E. H. Klevans
Pennsylvania State University

Historically, the field of nuclear engineering has been associated with fission and fission reactors. Research and coursework have evolved along the lines of neutron behavior in reactors and engineering analysis of reactors. Radioactivity and the interaction of radiation with matter have also had an important part. The nuclear reactor has been a successful device for producing power, and the need for people educated to understand and build new kinds of reactor systems is sure to increase. This must be an important (perhaps the most important) role for the Nuclear Engineering Departments.

Many nuclear engineers feel that plasma physics and fusion research are too far outside the main thrust of the nuclear engineering field. They may feel that fusion power is too far in the future, or that plasma research belongs in some other department such as physics or electrical engineering. The latter feeling no doubt is derived in part from the strong emphasis in plasma physics on electricity and magnetism. This subject is almost exclusively the domain of electrical engineering and physics, and many students going into nuclear engineering have no background or an inadequate background in electricity and magnetism. This means that the student, in addition to his reactor and related courses, must add electricity and magnetism to the list.

If we focus our attention on fission, radiation interaction and reactors, we will certainly not come to view plasma physics as a natural part of the nuclear engineering program. This view is too narrow. We should broaden our view to assert that, within the domain of nuclear engineering, we will include all possible energy production sources associated with the nucleus. With this view, fusion reactions and devices, and research associated with such devices, are included.

There is a definite education gap which the Nuclear Engineering Departments are in a position to fill. In many institutions Physics Departments have totally shied away from plasma physics as being too "applied," and few Electrical Engineering Departments have shown an interest in fusion-oriented work. This leaves fusion work and the training for it without a home.

This need for graduate education in plasma physics has been emphasized in a recent article discussing the teaching of plasma physics in American universities.¹ I would strongly recommend this article to all of you.

This article points out that publications in plasma physics are increasing exponentially with a three-year e-folding time. This includes work in ionospheric physics, gas discharges, fusion, magnetohydrodynamics, etc. The field is broad, and it is of an interdisciplinary nature. I want to stress this for it has considerable bearing on the plasma programs at Pennsylvania State, Columbia, Wisconsin and other institutions.

I. Plasma Physics Curriculum

The plasma physics curriculum has been formalized as a body of courses for about one year, but actually it is older. Alan Jacobs of the Nuclear Engineering Department offered coursework in plasma physics in 1962 and 1964 based on the courses taught by R. K. Osborn at the University of Michigan. The expansion of this effort during the past year was to a large extent related to the addition of Robert Quinn to the Electrical Engineering Department.

We joined forces, and with Tom Seliga of the Electrical Engineering Department began to map out a joint program in plasma studies including a series of courses and a research program. To get some experience, we started out separately; but after one year, a concrete program for graduate work in plasma physics was developed. Actually we have shied away from the term plasma physics so as not to become involved in any political dispute with the Physics Department. Thus, such terms as "plasma studies," "plasma phenomena," "plasma theory" and "thermonuclear engineering" are used.

The curriculum which was finally adopted has six courses as follows: NE 490 (EE 490), Introduction to Plasma Theory; EE 491, Plasma Phenomena and Diagnostics Laboratory; NE 503, Thermonuclear Engineering; NE 590 (EE 590), Plasma Waves; NE 591 (EE 591), Advanced Plasma Theory; and EE 592, Magnetohydrodynamics.

As indicated, three courses are cross-listed for both nuclear engineering and electrical engineering. Cross-listing enables staff from either department to teach these courses, emphasizing our belief that some of the basic courses form a common core to any specialized area of plasma research, which is again the interdisciplinary idea. Three other courses are listed in either nuclear engineering or electrical engineering. This corresponds to specialized interests or specific capabilities of one of the departments. For example, the laboratory course is being set up by Quinn in an Electrical Engineering Laboratory using funds obtained on a matching grant basis from the National Science Foundation. Consequently, the plasma laboratory course falls into the domain of electrical engineering rather than nuclear engineering.

One other point is that 400 level courses can be taken by advanced undergraduates as well as graduate students. Thus, the introductory theory and the experimental course are accessible to undergraduates. This is of great importance. It is anticipated that undergraduates from electrical engineering, engineering science and perhaps physics will take advantage of the opportunity.

A brief summary of content, prerequisite and a syllabus are given for each course in Tables I-VI.

TABLE I

NE 490 (EE 490) Introduction to Plasma Theory

Properties of plasmas; orbit theory; derivation of kinetic equation; macroscopic equations; basic transport processes.

Prerequisite: EE 43

1. Occurrence and Properties of Plasmas
2. Basic Equations and Conservation Laws
3. First Order Orbit Theory; Adiabatic Invariants; Cosmic and Geophysical Applications
4. Introduction to the Boltzmann Equation; Moments of the Boltzmann Equation; Diffusion; Mobility; Dielectric and Conductivity Tensor
5. Particle Interactions
 - a. Electrical neutrality
 - b. Debye shielding distance
 - c. Plasma frequency
 - d. Fokker-Planck equation
 - e. Relaxation times

TABLE II

EE 491 Plasma Phenomena and Diagnostics Laboratory

Methods of plasma generation and diagnosis; experimental investigation of basic plasma phenomena.

Prerequisite: EE 490

A number of experiments will be performed including:

1. Plasma Probes
2. Microwave Spectroscopy
3. Visible Spectroscopy
4. Laser Operation
5. Electron Beam Orbit Theory Studies
6. Experiments Demonstrating Fundamental Plasma Properties
 - a. Transport properties
 - b. Wave properties

TABLE III

NE 503 Thermonuclear Engineering

The problems of controlled fusion, including containment, heating, radiation and heat losses; study of devices.

Prerequisite: NE 490

1. Survey of Fusion Reactions, Problems and Devices
2. Containment
 - a. Equilibrium configurations
 - b. MHD stability
 - c. Microinstabilities
3. Plasma Heating
 - a. Magnetic pumping
 - b. Cyclotron heating
 - c. Injection
4. Radiation and Heat Losses
 - a. Charge exchange
 - b. Bremsstrahlung
 - c. Cyclotron radiation
 - d. Impurity radiation
 - e. Wall losses
5. Power Balance
6. Examination of Current Fusion Devices
 - a. Torus machines
 - b. Mirror machines
 - c. Pinch devices
 - d. Other fusion machines

TABLE IV

NE 590 (EE 590) Theory of Plasma Waves

Solutions of the Boltzmann equation; waves in bounded and unbounded plasmas; radiation and scattering from plasmas.

Prerequisite: NE 490 (EE 490)

1. Solution of the Boltzmann Equation
 - a. Transport properties in weakly ionized gases
 - b. Chapmann-Enskog theory
2. Wave Phenomena in Cold Plasmas
 - a. Analysis of the spectrum of plasma oscillations in infinite media
 - (1) without external magnetic fields
 - (2) with external magnetic fields

- b. Criteria for instabilities
 - (1) two stream instability
 - c. Spectrum of oscillations for bounded plasmas
 - d. Transmission and scattering from bounded plasmas
- 3. Wave Phenomena in Warm Plasmas
 - a. Solution of the Vlasov equation
 - b. Nyquist theory of stability
- 4. Radiation from Plasma
 - a. Bremsstrahlung
 - b. Cyclotron radiation

TABLE V

NE 591 (EE 591) Advanced Plasma Theory

Advanced topics in kinetic theory, fluctuation theory, microinstability and turbulence.

Prerequisite: NE 590 (EE 590)

- 1. Plasma Waves in a Hot Plasma with External Magnetic Fields
 - a. Analysis of spectrum
 - b. Microinstability theory
- 2. Fluctuation Theory in Equilibrium and Non-equilibrium Plasmas
 - a. Charge fluctuations
 - b. Current fluctuations
- 3. Incoherent Scattering from Plasmas
- 4. Derivation of the Plasma Kinetic Equation
 - a. Statistical mechanics of charged particles
 - b. Fluctuation viewpoint
- 5. Theory of Turbulent Plasmas
 - a. Quasilinear theory
 - b. Scattering theory

TABLE VI

EE 592 Magnetohydrodynamics

Fluid properties; derivation of MHD equations; channel flows; MHD waves; stability theory; power generation techniques.

Prerequisite: EE 438

- 1. Review of Fluid Properties of Plasmas and Liquid Metals
 - a. Mass, momentum and energy conservation
 - b. Stress and strain
 - c. Dielectric properties

2. Derivation of the MHD Equations
 - a. MHD approximations
 - b. The simplified equations
 - c. MHD parameters; Reynolds number, Mach number, Hall parameter, etc.
3. Incompressible MHD Flows
 - a. The Hartmann problem
 - b. Couette flow
 - c. Boundary value problems
4. Plane Waves in Fluids
 - a. Alfvén waves
 - b. Transverse waves
 - c. Longitudinal waves
 - d. Wave parameters
5. MHD Power Generation Techniques
 - a. Hall generator
 - b. MHD pumps
 - c. Vortex generator
 - d. Propulsion phenomena

The syllabus of the course, Plasma Phenomena and Diagnostics Laboratory, deserves further consideration. The development of such a course is an experiment in itself. We have set for ourselves an ambitious task--to develop a set of experiments that can be performed and understood by both graduate students and undergraduate students. These experiments must serve the role of enhancing the understanding of the student in the behavior of plasmas; of indicating the methods of investigating plasmas and the limitations of these methods; and of preparing the experimental students for independent research.

Hopefully, the six chosen experiments will do a reasonable job of fulfilling these goals. We just recently received a grant from the National Science Foundation for over \$30,000 on a matching grant basis to purchase the equipment for this laboratory. We hope to offer the course for the first time in the spring term.

In the first three experiments which will each be accompanied by several hours of theory, several of the basic diagnostic methods will be investigated and used to obtain plasma properties. In the gas laser experiment the laser is treated as a plasma with some peculiar properties. The influence of the lasing on the plasma and the diagnostics will be explored. In the introductory course the students study orbit theory, thus we have an experiment using an electron beam and a plasma in a strong magnetic field that will visually exhibit the orbiting properties of the charged particles. Drift velocities, Larmor radii and other variables will be measured. We will also study the phenomenon of trapping in a magnetic mirror. In a final experiment wave propagation parallel and perpendicular to a magnetic field will be investigated.

II. Research Programs

The current plasma research projects at Penn State are discussed in the following sections.

A. Wave Propagation in Bounded Plasmas

In 1958, Trivelpiece and Gould performed theoretical and experimental investigations on surface spacecharge waves in cylindrical plasma discharges. Such waves exist only in bounded plasmas. In the theoretical analysis the plasma was treated as a dielectric with the dielectric constant obtained from cold plasma theory. Ions served only as a neutralizing background.

There is a phenomenon called collisionless or Landau damping which is associated with plasma oscillations in an infinite plasma. This Landau damping, which was successfully measured not long ago, is associated with an energy exchange between particles, moving almost with the phase velocity of the plasma wave, and the plasma wave. It requires a kinetic theory formulation of the plasma oscillation problem to obtain Landau damping. We are interested in obtaining the collisionless damping for surface waves. A kinetic theory treatment is required. Some work has been done on this problem by A. N. Kondratenko in Russia. Work on the problem is far from complete, however, even for a plasma without an external magnetic field.

In order to prepare ourselves for the kinetic theory, a student and I decided to study the problem of surface waves in a cold plasma slab, using the dielectric model. We were surprised to find that when a magnetic field was present, the problem of finding the mode structure had not been thoroughly analyzed. We are just finishing this part of the work. Another student has been studying the effect of including ions and allowing the electron temperature to greatly exceed the ion temperature. He is using a two-fluid model to describe the plasma. Support for this work is obtained from a laboratory NASA grant.

B. Theoretical Study of an Inertial Confinement Device

Late last year, ITT Farnsworth Corporation, a research division of International Telephone and Telegraph Company, revealed that for several years they have been working on an inertial confinement fusion device which does not require magnetic fields. A schematic diagram of the device is shown in Fig. 1. A tentative, simplified description of how the device works is as follows: Ions are injected radially through a cathode which is biased 100 keV negative. As a consequence of space charge effects, the ions come to rest inside the cathode and form a virtual anode. Ions are then accelerated back out; and as a result of ion collisions with the cathode, electrons are released. These electrons are accelerated by the potential established by the virtual anode. Neglecting collisions, the electrons pass through the ion virtual anode and form a virtual cathode in the center. Ions are accelerated into this region, collide and fuse. The potential should look something like the bottom figure.

There have been some interesting experimental data obtained from this device as shown in Fig. 2. These fusion neutron outputs have been obtained under steady state conditions from seconds to minutes. No instabilities have been observed.

There are numerous questions about how this machine really does work. Are stable potential wells really formed? Are ions trapped in these

Fig. 1--The general arrangement and the potential distribution when ions only are present.

Fig. 2--Neutron output as a function of applied voltage, gas pressure and gas mixture. $I = 10$ mA.

wells? This is necessary for a fusion device. Otherwise, the ion densities will not get high enough near the center. What is the effect of coulomb collisions? A student is performing theoretical studies to try to answer some of these questions. This student is supported by a fellowship, and his computer calculations are being funded by the University.

C. Investigations of a Laboratory Plasma Trapped in a Dipole Magnetic Field

The experimental arrangement is essentially a glow discharge with a uniformly magnetized sphere serving as the cathode. The residual gas pressure is typically of the order of a few microns. A suitable voltage is then applied between the cathode and the anode to break the gas down. When the breakdown occurs, a number of plasma configurations may exist. One is the existence of a luminous plasma belt confined to the equatorial regions of the sphere and extending for a few radii outward. A second configuration is the onset of instabilities, given the proper experimental conditions. These instabilities consist of intense arcs which originate in the belt, traverse a field line and precipitate on the sphere in the polar regions. They last for a few milliseconds. While they occur, the luminous belt dims markedly and may even be extinguished. The rate at which these instabilities occur can be adjusted by a number of methods.

The density, temperature and electric field in the stable belt have been measured using Langmuir probes under varying conditions. Evidence of drifting in the equatorial plane has also been observed. The instabilities have been separately studied using electrostatic probes. It has been found that the laboratory instabilities are quite similar to those described in a theory proposed for a method of energizing electrons in auroral phenomena.

The use of electrostatic probes has thus been of considerable value but is nevertheless limited. Other diagnostic techniques which are applicable are measurement of the microwave scattering and measurement of the visible spectrum. Two students are engaged in these studies. Measurements of attenuation and phase shift are now being made. These will provide some of the first laboratory measurements of the scattering cross section for a plasma clad body of spherical geometry.

A modification of this plasma device leads to a new plasma configuration. If a second magnetized ball is used as the anode, recent photos indicate that a neutral point is formed between the spheres. The plasma bears some resemblance to the front of the magnetosphere, although no solar wind is present.

All three of these research efforts are being supported by the Space Science and Engineering Laboratory at Penn State, and the latter is funded by a NASA Sustaining Grant to the University. The funds are being administered by the Ionosphere Research Laboratory.

D. Characteristics of the Langmuir Probe in a Strong Magnetic Field

One of the basic diagnostic tools for the determination of electron density and temperature in the plasma is the Langmuir probe. Despite the great importance of this device, a completely satisfactory theory of Langmuir

probes is not yet available for a variety of experimental conditions. Further study of these areas is needed. In the current investigation both theoretical and experimental research is being conducted for the case of a plasma in a strong magnetic field. This work is also being performed at the Ionosphere Research Laboratory under a NASA grant.

E. Other Research Projects

Several research problems not listed above deserve to be mentioned. In the Ionosphere Research Laboratory one student has almost completed kinetic theory calculations on the impedance of a grid pair in a plasma beam. Another student is performing an experimental investigation of an ac plasma probe. Also, there is a substantial program, both theoretical and experimental, associated with wave interactions with the ionosphere. The above research is funded on a variety of NSF, Navy and Air Force contracts.

III. Conclusions

The final important question is: Where do we go from here? It is my hope that as further interest in fusion work develops in the Nuclear Engineering Department, some experimental research will begin. There is also the possibility of establishing a new interdisciplinary Plasma Research Laboratory. In the recent round of Project Themis proposals, we unsuccessfully bid for the establishment of such a laboratory. If sufficient funding can be found, this is still a possibility.

Except for the research mentioned on the inertial confinement fusion device, all the research and student support has come from various contracts of the Ionosphere Research Laboratory. The rapid development of our program has been intimately related to this funding situation.

As far as the Nuclear Engineering Department is concerned, plasma and fusion research and education will not, and should not, be a dominant activity. However, it can still be an important activity which will attract the interest of a number of students. If this program helps prepare students for a useful, interesting professional career and produces some useful research results, I will be satisfied.

DISCUSSION

QUESTION: Where do you anticipate the engineering students will acquire a background in the classical kinetic theory for gases and in ordinary discharge physics, including methods of generating plasmas?

ANSWER: The kinetic theory background basically has to be supplied in the courses, both in the introductory course and in the advanced course. That is generally true for all engineering students. The vast bulk of the students coming into the program are from electrical engineering and nuclear engineering, although we anticipate we may have some small interest from aerospace engineering.

As far as the gas discharge work goes, that will be, to a large degree, coupled with the experimental course. It is also mentioned in the introductory course where we study transport properties. This part of the course is, to a great extent, oriented toward discharge studies. In the experimental laboratory most of the diagnostic work will be done using discharge tubes. So the physics of discharges will be discussed in the beginning lecture material in the laboratory.

REFERENCES

1. "Report of Panel to Discuss Teaching of Plasma Physics in American Universities," Am. J. Physics, 33:104 (1966).

NUCLEAR SAFETY RESEARCH AND DEVELOPMENT*

W. B. Cottrell
Oak Ridge National Laboratory

If the stated purpose of this meeting were paraphrased in order to relate it to my talk it would go something like this: "Report the significant research activities in nuclear safety which are not directly related to reactor design and identify those areas in which more research is needed." Although the scope of this meeting includes nuclear safety education as well as nuclear safety research, I have addressed myself only to the latter.

In order to make the subject more tractable and easier to grasp, I have chosen to limit myself mostly to water power reactor safety research. Thus research dealing with waste disposal, space safety, as well as that on gas-cooled reactor safety and liquid-metal-cooled reactor safety has been excluded. At the present time this limitation amounts to less than 20 per cent of the total safety program sponsored by the AEC Division of Reactor Development. As this talk has not been cleared with the Atomic Energy Commission, which is the largest single sponsor of such activities, what I have to say must be regarded as my own opinion, although much of the talk is largely taken from the AEC documents--principally from a recently released informal report on the water-reactor safety program.¹ Following the general treatment of the subject, I will then illustrate it in detail by reviewing some of the specific projects at Oak Ridge National Laboratory together with some comments on the role of universities in our research programs. Following that, I will conclude with my opinion of where the safety business appears to be headed.

I. Objectives of the Nuclear Safety Program¹

The fundamental objective of the water-reactor safety program is to provide the Commission and industry with the safety-related information required to develop and regulate the safe application of nuclear energy to power production. The program accomplishes this by performing the necessary kind and amount of research, development and testing. In this program attention is focused on protection against accidents (both major and minor) rather than on the control of normal operational problems associated with the routine operation of a nuclear power plant. The central concern is the control of the radioactive fission products under accident situations. The problems of control of fission products and other radioactive materials under the normal

*Research sponsored by the U. S. Atomic Energy Commission under contract with the Union Carbide Corporation.

plant and effluent conditions, while closely related to the reactor safety program, is regarded as an operational and not a safety problem.

In order to assure that nuclear plants may be designed, built and operated without undue risk to the health and safety of the public, the nuclear safety program sponsors activities which may be divided into six general categories as follows:

1. Design Reactor Systems to Prevent Accidents:
to reduce to a negligible level the probability of occurrence of accidents.
2. Develop Operating Procedures to Prevent Accidents:
to reduce to a negligible level, the risk of operator actions or errors leading to the initiation of an accident.
3. Design Safeguards to Limit Consequences:
to modify and limit the consequences which might otherwise result from the already unlikely occurrence of an accident.
4. Develop Capability for Accident Recovery:
to provide capability for recovery from the consequences of accidents, should they occur.
5. Assess Adequacy of Safety Margins:
need to develop techniques and essential data for this purpose.
6. Develop Inspection and Testing Techniques:
to audit, inspect and test reactor and plant systems to ensure that safety margins are maintained.

The ultimate objective of each of the above is the development and regulation of safe power plants. The specific research projects are established in order to provide information on the accident phenomenology for reactor accident analyses. Of principal concern are, first the behavior of accident released fission products, and second the safeguards measure which may be employed to prevent or limit such releases. These basic concerns and objectives of the safety program are shown schematically in Fig. 1. The research and development program (the top three boxes) must develop the technical base in the form of technology and data which will in turn provide the basis for criteria standards and codes to assure that the objectives can be met.

The three principal problem areas within the research and development program are interrelated; each feeds information to the others and each provides direct program output in the form of the technical basis for guides, criteria, standards and codes for the design, location, construction, testing and operation of nuclear power test facilities and plants. The program output has significance to, and impact on, each of the six specific objectives shown in Fig. 1, which in turn all contribute to the general objective to develop and regulate the safe application of nuclear power.

UTILIZATION OF SAFETY PROGRAM

Fig, 1--Schematic diagram showing utilization of the AEC Nuclear Safety Program.

The first, "Accident Understanding" includes all those activities which lead to the ability to predict and analyze the major postulated reactor accidents in all their phases--the modes and credibility of initiation, the system response and consequences.

The second, "Fission-Product Behavior" includes all efforts necessary to understand the nature and magnitude of potential flows of fission products from their normal location within the fuel, through the cladding, primary system and containment barriers, and thence into the environment; all under realistic accident conditions.

The third, "Safeguards Measures" is concerned with measures which are provided in nuclear plants to interrupt the course or to limit the consequences of potential accidents, no matter how unlikely. In this activity the technical bases for designing and assessing the performance, testability and reliability of proposed engineered safeguards are developed.

II. Scope of Research Within Safety Program

Reactor designers must provide assurance that the design of the reactor and plant equipment is adequate to make negligible the probability of failures which could lead to initiation of an accident involving the risk of significant release of fission products from their normal location within the fuel cladding barrier. To accomplish this objective, it is necessary to identify the possible modes of accident initiation and to understand the response of the system to them so that either the probability of the initiating failure can be made negligible, or the response of the system can be made to be such as to prevent breach of fuel cladding barrier. The safety program provides technical information important to this design activity, both directly and through its application to the development and implementation of criteria, standards and codes. The program includes studies of materials properties, including radiation effects; studies of piping, pressure vessel and other component failure modes and likelihood of failure; studies of the effects of blowdown in pressurized systems; crack propagation and nil ductility studies; studies of transient strain rates and heat transfer; studies of reactor kinetics, including response to reactivity perturbations in large and small cores with and without the presence of control systems; etc.

In order to develop procedures for the operation of nuclear power plants which reduce to a negligible level the risk of operator actions or errors leading to the initiation of an accident, it is necessary to identify and understand the nature and course of possible accidents. The program contributes to this subject by providing studies of operating experience; increased understanding of reactor and system responses in abnormal situations which may affect operator actions; studies of operator qualification and evaluation techniques; etc.

A large effort is directed toward the incorporation into nuclear power plants of so-called engineered safeguards systems which will limit accident consequences or prevent abnormal conditions from causing an accident. An understanding of the overall response of the reactor and plant to the accident initiation is needed to determine the steps which are necessary to limit these consequences to acceptable levels. Information necessary to the

design of such systems and to the assessment of their performance under accident includes studies of reactor kinetics; transient heat transfer; pressure generation; fuel damage mechanisms; fission-product release and transport; blowdown; core heatup; emergency cooling systems; containment systems; air cleaning systems; protective and control systems; emergency power systems; etc.

In addition to the extensive measures which must be taken to prevent the occurrence and limit the consequences of accidents, the industry and the Commission also have the obligation to be prepared to deal with the aftermath of potential major accidents, unlikely as they may be. Suitable plans and equipment must be available should an accident occur, to ascertain whether and assure that the fission products are under control and will remain so; that the reactor has been secured in a safe condition; that any injured personnel can be rescued; and that the plant can be restored to a condition (whether dismantled or operable) which constitutes no significant hazard to the public. The program provides substantial input to the planning of such a recovery program by providing realistic estimates of the situations which might exist following postulated accidents, and where necessary, by developing special equipment or techniques for coping with these situations.

The Commission is also charged with a responsibility to regulate the applications of nuclear energy so as to ensure that no undue risk is presented to the public. The central output from the safety program in this regard is again an understanding of the course and consequences of hypothesized accidents, both with and without the action of engineered safeguards devices. The safety assessment made for regulatory purposes must ascertain that the accident-preventive and consequences-limiting design features have been accomplished to a sufficient degree and that the remaining level of risk presented to the general public by the construction and operation of a given plant is acceptably small. Information from the safety program assists in establishing the technology necessary to accomplish safe designs, and also provides information for use by the regulatory bodies in evaluating the efficiency and sufficiency of the specific safety-related design features provided in given plants.

Once a reactor plant is constructed and placed in operation, both the operator and the Commission must ensure that the protection provided is not compromised by equipment deterioration or malfunction, by inadequate maintenance, or by operator negligence. This audit, inspection and testing function is the sixth major mode of utilization of the safety program output indicated in Fig. 1. The design of meaningful tests of the reliability and performance of the original protective features provided in the design, and of the engineered safeguards systems requires that substantial information be provided with respect to the various abnormal environments to be expected in accident situations, the manner in which emergency equipment can deteriorate or malfunction, etc. The safety program includes several projects for the development of meaningful testing methods for auditing the performance and reliability of safety features, and the development and validation of practical techniques for revealing the presence of incipient failure of equipment or of procedures.

III. Program Rationale and Accident Sequence Structure

In the preceding the safety program activities have been described in general terms and the major activities which utilize the program output have been identified. The interrelationships of these activities were shown in simplified form in Fig. 1.

As previously noted, the prime concern in reactor safety is to ensure the control of fission products (and plutonium where applicable) because of the potential hazard to the public of the release of fission products to the environment. The fission-product-barrier concept is a particularly appropriate one upon which to base an exposition of the overall reactor safety problem as shown in Fig. 2. Fission products build up in the fuel and are kept there by the first barrier, the fuel cladding. If for any reason the cladding is violated, some fraction of the fission-product buildup will be released to the primary system which forms the second barrier. If then, the primary system is breached, fission products encounter the third barrier, the secondary containment structure; finally, if this fails, fission products can move into the environment. What must be considered, then, are the factors that enter into the possible violation of the integrities of the fuel cladding, primary system, and secondary containment structure, and the relationship of these factors to each other and to plant design including specifically the various engineered safety features that are provided.

Massive failure of the cladding barrier (as contrasted to the failure of a few fuel elements such as might occur as a result of loss of flow or a temporary overpower condition) can come about either as a result of a reactivity accident which gives rise to a severe power excursion; or as a result of a loss-of-coolant accident, in which the primary system is breached with the subsequent loss of the cooling medium. The latter is the one most commonly considered. Pipe rupture, vessel rupture, etc., could stem from inadequate criteria, codes and standards for reactor design or from improper operation or maintenance. The other conceivable cause of primary system failure is a severe reactivity accident. In this accident the reactivity excursion could cause cladding stresses high enough to result in cladding rupture. This in turn could result in expulsion of hot fuel into the coolant, extremely rapid heat transfer from fuel to coolant, a "steam explosion" and resultant pressure pulse, and subsequent mechanical effects which could conceivably include breaching of the primary system.

The containment structure, the third barrier to fission-product escape, can fail because of improper design, construction, operation or maintenance; as a result of seismic loads or externally generated missiles; or as a direct consequence of primary coolant system failure. The effects of primary system failure on the containment include: (1) missile generation; (2) over-pressurization of the containment (which would be primarily a result of the coolant blowdown, but could also be affected by core afterheat, the energy released from possible metal-water reactions, or the heat from the combustion of the hydrogen evolved from metal-water reactions); and (3) a potential melt-through which might be caused by the migration of molten fuel.

In the development of more detailed objectives for the individual projects which are designed to fulfill the safety program objectives, a logical

Fig. 2--Life history of fission products during a reactor accident.

framework is needed. An exposition of the course and consequences of the major accidents which can be postulated for reactor plants provides a useful and consistent basis for the delineation of program objectives and the interrelationships of individual projects. Figure 3 is a simplified diagram of the generalized accident sequence for water-cooled reactors which has been specifically constructed to provide the needed framework. It follows the accident sequence through failure of the various barriers and the potential path of fission products from their normal location to the external environment. Some interrelationships, principally feedback loops such as the possible role of coolant blowdown in reactivity insertion, have been omitted for clarity and simplification.

With the growth in size and power level of power reactors, the necessity for and use of engineered safeguards has become an increasingly vital and integral consideration in reactor safety design and evaluation. For the purpose of this talk, an engineered safeguard is defined as any design feature which is not required for the normal operation of the system as a power-producing unit. This permits a distinction to be made between normal safety factors, such as designing to prescribed code requirements, and those which are specifically used to cope with possible accident situations. In the accident sequence diagram shown in Fig. 3, the electrical diode symbols have been used to indicate graphically the points in the accident sequence where engineered safeguards can reduce the probability of initiation of an accident, control the response of the system if the accident is initiated, and/or limit the consequences of an accident if the accident runs its course.

As indicated in Fig. 3, a number of measures are taken to prevent the occurrence of inadvertent reactivity insertions. Design criteria such as limitations on a single-rod or total reactivity availability, design features such as physical barriers to control rod ejection, and the use of special codes and standards applicable to nuclear systems are typical of these safeguards. Their basic purpose is to prevent initiation of an accident leading ultimately to release of radioactive material.

If, despite these measures, an inadvertent reactivity addition occurs, the next safeguards affect the reactor response through the action of the control and protective systems. These systems can, in general compensate the reactivity insertion before damage can result, provided that the power excursion does not develop so quickly that system time-response capability is the limiting factor. Incorporation into the reactor design of inherent reactivity shutdown features, usually nuclear, such as a prompt negative power coefficient of reactivity, provides negative reactivity feedback to the system which can, for some range of reactivity perturbations, quench the power excursion before fuel cladding failure occurs.

Safeguards are also provided to prevent primary system failure from leading to the fuel cladding failure. If the primary coolant system fails at elevated temperature and pressure, there will inevitably be a loss of some fraction of the primary coolant by the blowdown, or depressurization, process. Forces resulting from the depressurization may cause core damage and the loss of the primary coolant will lead to a loss, or at least a significant lowering, of cooling capacity. The safeguards interposed at this point include adequate design of the core structure and the use of auxiliary cooling systems

Fig. 3--Accident sequence diagram with engineered safeguards for a reactor.

such as in-core sprays or special core-flooding systems to ensure that the cooling capacity remains great enough that fuel cladding will remain intact. The principal problems are those of the performance of the safeguard system, i.e., is it sufficiently effective and will it operate as intended under particular accident situations?

If violent fuel cladding failure occurs, there is no realistic way of preventing a rapid heat transfer to the coolant with the accompanying possibility of the generation of a severe pressure pulse. The energy release must be dissipated somewhere in the system, and some mechanical effect on the system will result. Some precautions can be taken to ensure that these effects do not damage the pressure-retaining part of the system by backing the pressure vessel with water, concrete, crushable shields, etc., or by making the inner structure such that it will absorb much of the energy before it reaches the barriers.

In considering the engineered safeguards which can be included to prevent failure of the secondary containment, three contingencies must be protected against--missile damage, overpressurization and melt-through of the structure. It is customary to design reactor systems such that the possible generation of missiles is held to a minimum, but it is not possible to guarantee that there will be none. Missile shield safeguards are usually provided in the form of massive concrete walls or other energy-absorbing structures interposed between the primary system and the containment.

Containment pressurization resulting from blowdown of the primary system can be controlled in a number of ways. One is to design the containment structure to withstand the maximum pressure which could be encountered. Another is to vent the escaping coolant into and through a condensing pool, as is done in so-called "pressure-suppression" systems. With the conventional containment shell, a containment spray system can be used to cool down the enclosed atmosphere. These three safeguards also act to control the contributions of core afterheat, metal-water reactions, and hydrogen combustion to containment pressurization. In addition the exothermic metal-water-reaction contribution can be controlled by limiting the temperature of the core through the action of in-core emergency cooling systems. If, however, the metal-water-reaction does take place, hydrogen will be released and could possibly form an explosive mixture. Possible safeguards against this eventuality are the use of combustion inhibitors and/or inert atmosphere in the containment structure.

If the various safeguards designed to prevent core meltdown fail and the molten core forms a pool in the bottom of the primary vessel, it is conceivable that a melt-through of the primary vessel will occur. Protection against this can be provided by external cooling of the vessel through the use of cooling coils, submergence in a pool, or by some other heat removal scheme. If the molten core melts through the vessel, it can conceivably continue to melt its way through the substructure until the containment itself is violated. Safeguards to prevent melt-through of the ultimate containment could take the form of some type of catch basin, possibly with its own cooling system.

Safeguards are also provided which act directly on the fission products. If fuel cladding fails, there will be a release of fission products to the

coolant; and therefore, to the primary system. There is no practical engineered safeguard which will prevent this fission-product release to the coolant. If, in addition, the primary system is breached, there are two possible situations which result: The secondary containment could retain its integrity and allow the escape of the available fission products to the environment at only a "near normal" rate; or the containment could be violated with a resulting high leak rate. In either case safeguards which operate on the fission products can be of value in either reducing the quantity of fission products available for release to the environment at the design rate, or in competing with abnormal release rates, as almost any conceivable violation of the containment would still require some time for fission-product migration, due to the large volume which is contained. The safeguards which can be used here include containment spray wash-down systems, trapping and filter systems, and/or fission-product retentive coatings.

Fission products which are released to the environment are not operated on directly by a safeguard. Siting guides, regional meteorology, evacuation plans, etc., constitute important factors which must be considered in safety assessments of specific reactor systems.

IV. Summary of AEC Nuclear Safety Research Tasks

We have defined the scope and objectives, examined the accident sequence, causes and effects, as well as philosophized, theorized, and perhaps even rationalized about the overall nuclear safety program. The details of which the program consists have not yet been shown. This is partly because a program which involves about 100 prime contracts plus many times that number of subcontracts is difficult to condense in a meaningful manner. Nevertheless, with the help of the next series of tables, I will attempt to do just that. Table I groups the research activities into 14 categories which unfortunately do not have a one-to-one correspondence with the safety categories which have just been discussed, although they are close enough to be useful.

TABLE I

AEC Terrestrial Nuclear Safety Programs

1. Reactor Kinetics Research
2. Primary System Integrity
3. In-core Thermo- and Hydrodynamic Accident Transients
4. Chemical Reactions
5. Fission Product Release and Transport in Containment Systems
6. Characterization and Filtration of Fission Product Aerosols
7. Containment Systems Behavior
8. Fission Product Release to Environment
9. Engineered Safeguards--Preventive
10. Engineered Safeguards--Consequence Limiting
11. Fast Reactor Safety Studies
12. Gas Cooled Reactor Safety Studies
13. Environmental Studies
14. Miscellaneous

Ideally, I would have 14 more tables--one for each category which lists the research project by title and principal contractor. However, my file is prejudicial in that there is a table for only those categories in which ORNL has at least one project. While this information is somewhat out of date, this review is intended to be illustrative rather than complete. Thus we can see from Tables I-IX that the second grouping on Primary System Integrity consists of eight projects with the contractors as indicated.

TABLE II
Primary System Integrity

Project	Installation
Primary System Pipe Rupture Studies	General Electric--San Jose
Fatigue Tests of Pressure Vessels	Southwest Research Institute
Irradiation Tests of Model Vessels	Pacific Northwest Laboratories
Nonradial Nozzle Connection in Spherical Shells	Oak Ridge National Laboratory
Reinforced Openings in Pressure Vessels	Battelle Memorial Institute
Variability of Impact Data in Reactor Vessel Steels	Oak Ridge National Laboratory
Review of Present Pressure Vessel Technology	Oak Ridge National Laboratory
Proposed Modifications to ASME Nuclear Vessel Code	Oak Ridge National Laboratory

TABLE III
Chemical Reactions

Project	Installation
Metal-Water Reactions	Argonne National Laboratory
Development of Metal-Water Reaction Models	Phillips Petroleum Company
Steam-Graphite Reactions	Oak Ridge National Laboratory
Air-Graphite Reactions	Oak Ridge National Laboratory
Sodium Reactions	Atomics International

TABLE IV

Fission Product Release and Transport in Containment Systems

Project	Installation
Behavior of Fission Products on the In-Pile Destruction of Reactor Fuels	Oak Ridge National Laboratory
Release of Fission Products Under Transient Reactor Conditions	Oak Ridge National Laboratory
Nuclear Safety Pilot Plant (NSPP)	Oak Ridge National Laboratory
Application of Existing Data to Analysis	Battelle Memorial Institute
Machine Program for Fission Product Release	Phillips Petroleum Company
Code for Transport and Deposition of Fission Products	Phillips Petroleum Company
Generalized Fission Product Transport Code for Prolonged Release	Phillips Petroleum Company
Containment Research Installation Mock-Up	Oak Ridge National Laboratory

TABLE V

Characterization and Filtration of Fission Product Aerosols

Project	Installation
Properties of Fission Product Aerosols Produced by Overheated Reactor Fuels	Oak Ridge National Laboratory
Characterization, Control and Simulation of Accident Released Fission Products	Oak Ridge National Laboratory
Fission Product Release Studies	Brookhaven National Laboratory
R & D New Air-Cleaning Techniques and Testing Methods	Harvard Air Cleaning Laboratory
Filter Evaluation Program	Oak Ridge National Laboratory

TABLE VI

Containment Systems Behavior

Project	Installation
Containment Systems Experiment	Pacific Northwest Laboratories
Loss-of-Fluid Test	Phillips Petroleum Company
Off-Shore Siting of Power Reactors	Oak Ridge National Laboratory
Leakage of Air through Building Components	Atomics International
Fast Reactor Containment	Argonne National Laboratory

TABLE VII

Gas-Cooled Reactor Safety

Project	Installation
Kinetic Studies Related to Gas-Cooled Reactors	Phillips Petroleum Company
Steam-Graphite Reaction and Fission Product Release and Transport Studies	Oak Ridge National Laboratory
Air-Graphite Reaction and Fission Product Release and Transport Studies	Oak Ridge National Laboratory
Engineering-Scale Experiment Steam-Graphite Reaction	Oak Ridge National Laboratory
Fission Product Desorption in Steam	Oak Ridge National Laboratory
Effect of High Temperature Operation on Fission Product Migration and Release	Oak Ridge National Laboratory

TABLE VIII
Environmental Studies

Project	Installation
Demonstration of Ultimate Disposal of High Level Solid Wastes in Salt Formations	Oak Ridge National Laboratory
Waste Management Engineering and Safety Evaluation Studies	Oak Ridge National Laboratory
Characterization of Fluoride Volatility Effluents under Accident Conditions and Development of Handling Processes	Argonne National Laboratory
Dispersion of Radioactive Aerosols and Gases in the Atmosphere	U. S. Weather Bureau
Sorptive Properties of Earth Materials for Radioactive Gases	Oak Ridge National Laboratory
Removal of Organic Iodides from Reactor Containment Atmospheres	Pacific Northwest Laboratories
Development of Analytical Model to Predict Travel of Radioactive Gases Injected into the Ground	Pacific Northwest Laboratories
Dispersion of Radioactive Gases Injected into the Ground	University of California
Disposal of Waste by Hydraulic Fracturing Technique	Oak Ridge National Laboratory
Behavior of Radionuclides in the Ground	Oak Ridge National Laboratory
Columbia River Sediment Studies	Pacific Northwest Laboratories
Soil Moisture Flow in Waste Burial Areas	Savannah River Laboratories
Geohydrologic and Gas Injection Studies at NRTS	ID Health and Safety Division (NRTS)
Studies of Oceanographic and Thermal Factors Affecting the Uses of Nuclear Power Sources in or Adjacent to the Sea	Chesapeake Bay Institute, Johns Hopkins University
Model River System Studies	University of Texas
Dispersion of Radioactive Effluents Released to the Ground and Surface Streams; Geohydrologic Aspects of Nuclear Facility Siting	USGS

TABLE IX
Miscellaneous

Project	Installation
Nuclear Safing and Recovery	Idaho Operations Office
Reactor Operator Study	American Institute for Research
Reactor Safety Analysis Methodology	Planning Research Corporation
A Study of Power Reactor Safety	
Experience and System Reliability	Holmes and Narver, Inc.
Nuclear Safety Information Center	Oak Ridge National Laboratory
Radioactive Shipment Hazards	Pacific Northwest Laboratories
Transportation Accident--Full Scale,	
Field Collision Test	Joint AEC-DA
Nuclear Facility Monitoring Guide	Pacific Northwest Laboratories
Laboratory and Analytical Studies of	
Different Shipping Cask Types and	
Materials	Franklin Institute
Reactor Fuel Transport Safety	Oak Ridge National Laboratory
In-Plant Testing Program	Phillips Petroleum Company

This resume of the AEC safety program is incomplete and has of necessity been brief. For those who are interested in more complete and detailed information, you should examine a published summary report² covering all nuclear safety research and development or follow the subject in the journal Nuclear Safety, each issue of which contains a brief summary of all AEC sponsored work.

V. The ORNL Nuclear Safety Program

Since the ORNL program represents about one-sixth of the Commission budget for nuclear safety, it does not reflect the breadth of the Commission program although each of the ORNL research projects is directed to a specific problem in the accident sequence chart which we previously discussed. The ORNL program is composed of the eight general research areas shown in Table X, each of which consists of an average of five projects.

There are now over 30 specific research projects ranging from \$25,000 to several hundred thousand dollars each in the ORNL program alone, including several large projects which have recently been initiated. However, it is obvious from Table X that the ORNL program has been heavily oriented toward fission product behavioral studies.

Before describing a few of the facilities which are employed in the fission product program, there is one additional concept which must first be appreciated in order to understand the program. This concept concerns the geometric size of the various facilities. The results of the small scale experi-

TABLE X

ORNL Nuclear Safety Program

1. Release and Transport of Fission Products upon In-Pile Overheating of Reactor Fuels
2. Chemical Reactions and Fission-Product Behavior with Overheated Fuel Materials in Out-of-Pile Facilities
3. Filtration and Adsorption Technology
4. Characterization and Behavior of Accident-Released Fission Products
5. Analytical Models of Fission-Product Behavior
6. Steel Pressure Vessel Technology
7. General Nuclear Safety Studies
8. Nuclear Safety Information

ments must be extrapolated to large sizes, in order to provide useful information for reactor safety analyses. The extrapolation of small scale experiments must be extensively tested to ascertain that the predominant mechanisms do not change with the size of the experiment. In the discussion that follows, some overlapping is involved in work performed in various facilities. This is not accidental. It is done on purpose in order to test the validity of extrapolations with scaleup.

Such scaleup is necessary because the large experiments which are planned in the U. S. Nuclear Safety Program (such as the LOFT experiment) will be exceedingly complex, costly and necessarily few in number. The interpretation of their results will, therefore, depend heavily on a wealth of information obtained from many smaller scale experiments. Our experiments are conducted in both in-pile and out-of-pile facilities. The in-pile experiments involve the greatest geometric extrapolation because of their small size, but introduce the least uncertainty regarding the adequacy of the heat source simulation. We have three out-of-pile facilities with containment volumes ranging in size from 50 feet³ for the Containment Mock-up Facility, 140 feet³ for the Containment Research Installation, to 1350 feet³ for the Nuclear Safety Pilot Plant. However, in these facilities the fission product release from the UO₂ fuel must be effected in a non-nuclear furnace. The first two facilities employ induction heating and the last a plasma torch.

Figure 4 shows the in-pile reactor facility for simulation loss-of-coolant fuel destruction experiments. This 2-1/2 inch diameter capsule contains a miniature fuel element and is inserted into one of the fuel holes in the Oak Ridge Research Reactor where the fuel element melts under its own fission heat. During the experiment, gas (which may be inert, or may be air or steam, etc.) passes over the fuel element and helps carry away some of the released fission product aerosol which also contains some vaporized fuel and stainless steel or zircaloy fuel cladding. The aerosol is characterized as it goes through the small exit tube which doubles as a diffusion tube and then to a small holdup vessel. The reactor furnace is positioned in the ORR Reactor, shown at the bottom of the figure. The in-core assembly is inserted into the proper flux in the reactor by a hydraulic mechanism.

Fig. 4--In-pile reactor facility for simulating loss-of-coolant fuel destruction experiments at ORNL.

These and other in-pile destruction studies are highly successful; and they have included experiments to study effect of atmosphere and condensing steam, effect of burnup, gas flow rate, nature of fuel and method of fuel destruction. In addition data have been obtained on fission product fractionation and freezing point depression of UO_2 by oxidized stainless steel, which causes depressions as large as 1000°C in the melting point of UO_2 .

Going on to the out-of-pile programs, Fig. 5 shows the diagram of the Containment Mock-up Facility. It is built in a hot cell in order to handle large quantities of activity released when previously irradiated fuel is melted. The facility has a furnace to melt the fuel and generate an aerosol cloud, which passes directly into a 180 liter containment vessel. In the containment vessel the aerosol is aged for a predetermined period; and during the aging period, samples are taken of the containment atmosphere to determine how the concentration of aerosols changes with time. After the aerosol is aged, it is displaced by argon introduced at the bottom. As the aerosol is displaced either by pressure release or by argon, it is passed through sampling and/or filtering devices depending upon the experiment.

Figure 6 shows a schematic of the Containment Research Installation which has recently been placed into operation. This installation has (1) a primary vessel of 120 liter capacity that can be pressurized and heated to 800°C ; (2) a containment vessel of 4000 liter capacity that can be pressurized and supplied with different surfaces, i.e., painted or concrete surfaces; (3) an aerosol generator capable of handling 100 grams of fuel; and (4) capabilities for simulation of top or bottom brakes in the primary reactor piping. This facility was designed with a number of experimental capabilities including (1) release of fission products from the fuel; (2) retention in primary systems; (3) transport to the containment vessel; (4) behavior of fission products in the containment vessel; (5) recycle filter systems; (6) pressure suppression devices; and (7) spray and dousing techniques which may be simulated in the containment vessel.

Figure 7 shows an isometric cutaway of our largest facility--the Nuclear Safety Pilot Plant. The Nuclear Safety Pilot Plant is used to study the behavior of fission product aerosols in a containment vessel of substantial size (about 40,000 liters) to determine whether a change in scale has had a significant effect in scaleup extrapolation. It can handle fuel elements that are 1 inch in diameter x 6 inches long and 250 grams in weight and uses a plasma jet as a source of heat. It employs many advanced techniques in sequential sampling and has all the experimental capability of the CRI except that for changing the inner surface of the vessel.

These are but a small sampling of research facilities associated with four of the forty programs at ORNL. Detailed consideration of the data produced by these facilities is beyond the scope of this paper. Those interested in more details of the program in its entirety can do so from published sources which include annual progress reports, **, ³ bimonthly progress reports, ***, ⁴ as well as the reports in Nuclear Safety.

**The last one in this series is ORNL-4071

***The last one in this series is ORNL-TM-1913.

Fig. 5--Containment mockup facility showing a schematic of the scaled up fission product release experiment.

Fig. 6--Schematic of the containment research installation.

Fig. 7--Cutaway diagram of the nuclear safety pilot plant used to study the behavior of fission product aerosols.

The ORNL safety program alone has at least 11 research sub-contracts, totaling over \$600,000 with seven different universities, not including approximately 15 direct consultant contracts with individuals in universities. While this list is not complete, it includes those shown in Table XI.

TABLE XI

University Research Subcontracts

University	Amount FY68	General Subject
Auburn	\$ 15,000	Nozzles Attached to Spherical Shells
Auburn	4,000	Nozzle Stresses
Cornell	19,000	Prestressed Concrete Pressure Vessel Model
Cornell	18,195	Effect of Cyclic Loading on Pressure Vessels
Illinois	15,150	Prestressed Concrete Pressure Vessel
Illinois	226,000	Failure Phenomena of Prestressed Concrete
Tennessee	33,500	Nozzles Attached to Flat Plates
Tennessee	146,000	Stress Analysis of Nozzles
California	100,444	Stress-Strain Criteria for Concrete
N. C. State	20,000	Fission Product Deposition
Minnesota	<u>22,000</u>	Aerosol Technology
	\$619,289	

VI. Research Problems of Tomorrow

Safety problems, which are most likely to constitute the nucleus of the safety programs of tomorrow, fall into two general categories. First, those associated with reactor design and second, those associated with the reactor site.

The factors in the "design" category which may introduce further safety problems include (1) changes in reactor design; (2) higher reactor power; and/or (3) higher reactor power density; (4) design for less probable accidents; (5) an inadequacy in present criteria or assumptions; and (6) the development of a different reactor type.

The factors in the "site" category which will introduce further safety problems include (1) siting in more populous areas, and (2) siting in seismic areas. Evidences of these trends are already quite prevalent. Power

levels of proposed reactors are becoming higher and their fuel cycles longer. Some of these reactors are being proposed for construction closer to large metropolitan centers. The increase in reactor power and the lengthening of the fuel cycle in terms of percentage fuel burnup increase the fission product inventory accumulated in the fuel of the reactor. However, advances in technology contribute increased safety in the design and construction of reactors.

In view of these developments, an important reactor siting question for the future is how much emphasis should be placed, relatively speaking, on (1) distance from population centers; (2) engineered safety features designed to prevent accidents; and (3) engineered safety features designed to limit their consequences. This question applies to all sites.

The basic safety issue presented by metropolitan sites for large power reactors is whether, in lieu of partial dependence on distance from population centers, full reliance can or should be placed on the inherent safety of the reactors and their engineered safety features.

The AEC has already answered some of these questions for the immediate future in recent testimony by Commissioner Seaborg before the Joint Committee on Atomic Energy when he said, "While progress has been made in improving the safety of reactors we need to obtain more experience and data in order to develop criteria for the siting of these units in urban areas." By "these units" he means nuclear power plants of the type now being constructed and sited elsewhere. Advances in these reactor types along the line previously indicated or the introduction of new types will continually broaden the horizon for nuclear safety research. In my view, not yet shared by the Commission, one of the more attractive prospects for nuclear safety analyses, as required for evaluating safety, is the coupling of safety evaluation with a probabilistic approach to safety analyses.

VII. Postscript

For those interested in keeping up with all AEC sponsored nuclear safety research, I can think of no better source than Nuclear Safety. Every issue of Nuclear Safety contains "Progress Summary of Nuclear Safety Research and Development Projects" which, in addition, identifies the responsible individual and all reports issued during the reporting period. Anyone with more than a passing interest in nuclear safety will find this publication extremely valuable for many other reasons as well.

For those of you who in the future may desire information on many of the technical specialities within nuclear safety, I would refer you to the Nuclear Safety Information Center which was established by the AEC at ORNL for the purpose of serving the nuclear community in this area. The Center's staff not only collects and analyzes nuclear safety literature, but is also available for consultation and to render assistance. The Center also has available a free selected information service which anyone interested in nuclear safety information would find invaluable.

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REACTOR DYNAMICS - A FOUNDATION FOR NUCLEAR SAFETY

D. L. Hetrick
University of Arizona

The nuclear safety program at the University of Arizona is actually not a single formal program but rather a collection of small efforts under a variety of sponsors; and often these are secondary parts of some broader program, for example, nuclear rocket dynamics, desalination or in-core thermionic reactors.

The fact that the safety interests at Arizona have developed primarily as parts of the other programs is probably good, because that is where safety belongs; namely, everywhere. Not that every graduate student is supposed to learn how to prepare a complete hazards evaluation, but rather safety should be an integral part of his personality as has been the tradition in sound design practice for some time.

The University has not thought it wise to institute separate required courses in safety practices and procedures. The graduate course "Nuclear Safety" is intended rather for specialists who will some day be responsible for detailed safety evaluations, or who will do research in nuclear safety.

This course, which is not required, is for second- and third-year graduate students. Emphasis is placed on reactor safety, which includes topics such as introduction to reactor dynamics; reactor siting, containment, startup and shutdown; supercritical excursions; loss-of-coolant accidents; fission-products dispersal; biological hazards; and criticality safety.

Several of the above topics represent specific research areas at the University of Arizona. Because of their importance in nuclear safety, I should like to discuss the results which have been obtained in three areas. These are: (1) numerical calculations in reactor dynamics, (2) analytical approximations for calculating reactor excursions, and (3) nonlinear reactor stability. All are supported by the University with the exception of the latter which is supported in part by the AEC contract on dynamics of thermionic reactor systems. These represent only the neutron dynamic parts of reactor safety, and that is a small part of the field of reactor safety. The University is expecting in the immediate future to achieve considerable broadening in two directions. These are space-dependent dynamics and stability and overall plant accident calculations in safety evaluation models.

I. Numerical Calculations in Reactor Dynamics

Figure 1 shows the familiar equations of reactor dynamics. If one thinks of a column vector whose first element is neutron density and whose other elements are precursor concentrations, then the equations can be written in simple form using the coefficient matrix A . The one-one entry is a complicated time-dependent function, depending on reactor power and its past history through whatever feedback mechanism one would want to use. Suppose the feedback is known, and that one desires a numerical calculation of reactor power during a transient that is not particularly fast, i.e., time scales of many minutes. Many of the important problems in reactor safety are not fast, superprompt-critical excursions. In those cases when the neutron dynamics enters the picture, one often has a relatively slow transient. In a fast reactor the numerical calculation is going to be troublesome because of the one-one entry in the matrix:

The mathematician would say the trouble is due to the largest eigenvalue. Physically it is because the neutron generation time is short. If one sets this up as a problem in finite differences or uses other, more sophisticated methods like Adams or Runge-Kutta, immediately numerical instability is encountered. The problem cannot be solved without talking about a million time steps to get to one second.

There are two recognized ways to avoid this.¹ One is called the prompt-jump approximation; one of the objectives has been to put this approximation into a more formal context for better understanding.

Figure 2 shows a system of equations in vector-matrix form. The shortness of the neutron generation time means large entries in the first column of the matrix; and one of the variables, X_1 (neutron density or reactor power), will be an extremely small number compared to the other entries in the column vector. The differential equation for X is then rewritten with the X_1 term taken out of the summation. The equation is then multiplied through by the small parameter ϵ (in reactor dynamics it is convenient to choose $\epsilon = \lambda l / \beta$). Defining the new variables Y_j , there is a set of variables which have the same order of magnitude.

In Fig. 3 the first differential equation for X_1 is solved and dropping the term in epsilon yields the first approximation. One order from the set of differential equations is dropped; and one initial condition, which is replaced by a discontinuity, is lost. Use the approximation to compute the derivative, put it back into the first equation and generate an asymptotic series, in physical terms an asymptotic series for the solution of the problem in powers of the small parameter l . The first term of that series is the prompt-jump approximation, and the other terms are correction terms.

Figure 4 presents this in familiar notation. Garrett Birkhoff of Harvard wrote down the first equation about a year ago.² For one delay group it becomes the second equation, and from square brackets an explicit criterion for the range of validity of the prompt-jump approximation in the one-group case is obtained. The last equation summarizes the approximate neutron dynamics--a simple, first-order, nonlinear differential equation. This method is valid only for reactivity below prompt critical.

$$\dot{n} = \frac{\rho - \beta}{\ell} n + \sum_i \lambda_i c_i$$

$$\dot{c}_i = \frac{\beta_i}{\ell} n - \lambda_i c_i$$

$$A = \begin{bmatrix} \frac{\rho - \beta}{\ell} & \lambda_1 & \lambda_2 & \lambda_3 & \dots \\ \frac{\beta_1}{\ell} & -\lambda_1 & 0 & 0 & \\ \frac{\beta_2}{\ell} & 0 & -\lambda_2 & 0 & \\ \frac{\beta_3}{\ell} & 0 & 0 & -\lambda_3 & \\ \vdots & & & & \ddots \end{bmatrix}$$

Fig. 1--Equations of reactor dynamics.

$$\dot{\underline{x}} = A \underline{x} \quad (1)$$

$$\dot{x}_1 = \sum_j a_{1j} x_j = a_{11} x_1 + \sum_{j>1} a_{1j} x_j \quad (2)$$

$$\epsilon \dot{x}_1 = b_{11} x_1 + \sum_{j>1} a_{1j} y_j \quad (3)$$

$$b_{11} = \epsilon a_{11} = O(a_{ij}), \quad i > 1 \quad (4)$$

$$y_j = \epsilon x_j = O(x_1), \quad j > 1 \quad (5)$$

Fig. 2--System of equations with a large eigenvalue.

$$\dot{x}_1 = -\frac{1}{b_{11}} \left(\sum_{j>1} a_{1j} y_j - \epsilon \dot{x}_1 \right) \quad (6)$$

$$\dot{x}_1 \approx -\frac{1}{b_{11}} \sum_{j>1} a_{1j} y_j \quad (7)$$

$$\dot{x}_1 \approx -\frac{1}{b_{11}} \sum_{j>1} \left(\dot{a}_{1j} y_j + a_{1j} \dot{y}_j - \frac{\dot{b}_{11}}{b_{11}} a_{1j} y_j \right) \quad (8)$$

$$\dot{x}_1 = -\frac{1}{b_{11}} \sum_{j>1} a_{1j} y_j \left[1 + \epsilon f(y_j, \dot{y}_j) + O(\epsilon^2) \right] \quad (9)$$

Fig. 3--Development of the asymptotic series.

$$n = \frac{\sum_i \lambda_i \ell c_i}{\beta - \rho} - \frac{\ell}{(\beta - \rho)^2} \left[\frac{\sum_i \lambda_i \ell c_i \dot{\rho}}{\beta - \rho} + \sum_i \lambda_i \ell \dot{c}_i \right] + O(\ell^2)$$

$$n = \frac{\lambda \ell c}{\beta - \rho} \left[1 - \frac{\ell}{(\beta - \rho)^2} (\dot{\rho} + \lambda \rho) \right] + O(\ell^2)$$

$$\beta - \rho \gg \sqrt{\ell |\dot{\rho} + \lambda \rho|}$$

$$(\beta - \rho) \dot{n} \approx (\dot{\rho} + \lambda \rho) n$$

Fig. 4--The prompt-jump approximation.

To get through prompt critical (a start-up accident to be followed into the superprompt-critical range with a single method of computation), there are a number of other devices one can use. The iterative method in Fig. 5 is the method developed by Dick Cohen of Atomics International about 10 years ago.³ It is well known as the basis of the codes called AIREK and AIROS.⁴ There are two steps. The first step is to convert to a quasilinear differential equation by moving the time-dependent coefficient out of the differential operator and putting it on the right side; then performing a formal integration, one realizes that in the integral, the source of trouble in slow transients is localized in the rapid variation of the integrating factor.

The numerical algorithm is therefore constructed by writing just the slowly-varying factor as a Taylor series. Integrating term by term generates elementary functions as shown in the last equation. This has been extremely successful. The code will cover the complete range of generation times and time scales with reasonable time step sizes.

Figure 6 refers to the other two major types of methods that will perform the same function. The variational method is what Ray Brittan used in the RE-29 code at Argonne about 12 years ago.⁵ In the integrals the slowly-varying part is represented as a linear, or in this case a quadratic trial function and is integrated term by term. This trial function has two unknown coefficients, and the coefficients are evaluated by demanding that the integral equation be satisfied exactly at the middle and at the end of the time step. This is known as the collocation method as further developed by Jerry Kaganove⁶ at Argonne and Horace Flatt⁷ at IBM. In many ways it is the neatest and best method.

Hansen⁸ at MIT has developed what might be called the largest eigenvalue method. He evaluates the integrals by assuming that the unknown functions are always exponential in time, and for each time step the exponential has a time constant that is determined by the average reactivity appropriate to that time stop. Physically, one uses the largest root of the inhour equation.

The equations in Fig. 7 are in neutron dynamics notation. The first two equations are used in AIREK, although Felix Adler⁹ suggested an improvement which involves a partial integration and a little rewriting which improves it a great deal. The other two equations are Hansen's method for one delay group. The mathematical form is simple. The only problem is that one has to solve the inhour equation over again in every time step.

Figure 8 shows the test problem.¹ Representatives of each of these three integral methods were taken, and the neutron generation time and the time step size were varied. We wanted to choose the most simple problem we could think of and still have a peak power as in a self-limiting excursion. The feedback computation was avoided by taking the first half cycle of a sine-wave reactivity. The peak power always comes in the second quarter-cycle, and the reactivity amplitude and time scale were adjusted to simulate roughly a self-limiting feedback. In the one-group case the prompt-jump approximation can be solved exactly, so for a short neutron generation time there is a convenient analytical comparison. An example is shown in Fig. 9. This is typical of a large amplitude reactivity oscillation problem with the sine wave started from equilibrium.

ITERATIVE

$$\dot{x}_1 - a_{11}x_1 = \sum_{j>1} a_{1j}x_j \quad (10)$$

$$\dot{x}_1 + \alpha x_1 = (a_{11} + \alpha)x_1 + \sum_{j>1} a_{1j}x_j \quad (11)$$

$$x_1(t) = x_1(t_0) e^{-\alpha(t-t_0)} + \int_{t_0}^t \left\{ [a_{11}(t') + \alpha] x_1(t') + \sum_{j>1} a_{1j}(t') x_j(t') \right\} e^{-\alpha(t-t')} dt' \quad (12)$$

$$\int_0^h f(t_0 + \tau) e^{\alpha\tau} d\tau = \sum_n f_n \int_0^h \tau^n e^{\alpha\tau} d\tau \quad (13)$$

Fig. 5--Iterative integral method.

VARIATIONAL

$$x_j(t') = x_j(t_o) \left[1 + A_j(t' - t_o) + B_j(t' - t_o)^2 \right] \quad (14)$$

LARGEST EIGENVALUE

$$\dot{x}_1 - \bar{a}_{11} x_1 = \sum_{j>1} \bar{a}_{1j} x_j \quad (15)$$

$$x_1(t) = x_1(t_o) e^{\bar{a}_{11}(t-t_o)} + \sum_{j>1} \bar{a}_{1j} \int_{t_o}^t x_j(t') e^{\bar{a}_{11}(t-t')} dt' \quad (16)$$

$$x_j(t') = x_j(t_o) e^{\omega(t' - t_o)} \quad (17)$$

$$x_1(t) = x_1(t_o) e^{\bar{a}_{11}(t-t_o)} + \sum_{j>1} \bar{a}_{1j} x_j(t_o) \frac{e^{\omega(t-t_o)} - e^{\bar{a}_{11}(t-t_o)}}{\omega - \bar{a}_{11}} \quad (18)$$

Fig. 6---Collocation-variational and largest-eigenvalue methods.

$$\dot{n} + \alpha n = \left(\frac{\rho - \beta}{\ell} + \alpha \right) n + \sum_i \lambda_i c_i$$

$$\alpha = \frac{\beta - \rho(t_o)}{\ell}$$

$$n(t) = n(t_o) e^{(\bar{\rho} - \beta)(t - t_o)/\ell}$$

$$+ \frac{\lambda c(t_o)}{\omega - (\bar{\rho} - \beta)/\ell} \left[e^{\omega(t - t_o)} - e^{(\bar{\rho} - \beta)(t - t_o)/\ell} \right]$$

$$c(t) = c(t_o) e^{-\lambda(t - t_o)} + \frac{\beta n(t_o)/\ell}{\omega + \lambda} \left[e^{\omega(t - t_o)} - e^{-\lambda(t - t_o)} \right]$$

Fig. 7--Reactor-dynamics integral equations.

$$\rho = \rho_o \sin \frac{\pi t}{T}$$

$$\frac{n}{n_o} = \frac{\beta}{\beta - \rho_o \sin \frac{\pi t}{T}} \exp \left[-\lambda t + \frac{\beta \lambda T}{\pi \sqrt{\beta^2 - \rho_o^2}} \cdot \right.$$

$$\left. \left(\sin^{-1} \frac{\rho_o}{\beta} - \sin^{-1} \frac{\rho_o - \beta \sin \frac{\pi t}{T}}{\beta - \rho_o \sin \frac{\pi t}{T}} \right) \right]$$

$$\rho_o = \frac{\beta \omega}{\omega + \lambda}$$

$$\frac{T}{2} = \frac{4}{\omega}$$

Fig. 8--Reactivity oscillations.

Fig. 9--Reactivity oscillations.

Figure 10 shows typical results for peak power as a function of the interval size (the time-step size), the upper part represents a fast reactor and the lower part a thermal reactor. The dotted line is predicted by the prompt-jump approximation, and it looks as though the fast reactor cases converge to it as they should. As long as the reactivity is under 99 cents, the error is about 10^{-4} or less, and that doesn't show in this figure. The little crosses represent the collocation variational methods; and with respect to time-step size, they seem to converge most quickly. Hansen's method also converges well. The method of Adler is really Cohen's method with Adler's modification. In this particular case it is not the best for larger time steps, but it converges fairly rapidly as the interval is reduced.

Note especially the upper set of results representing a fast-reactor transient that takes about 500 seconds of real time. With a time-step size of one second, that is a reasonable calculation requiring only 500 steps. Using simple finite difference, it will take about 2×10^8 time-steps and by Runge-Kutta, 5×10^7 steps.

Slightly faster transients are shown in Fig. 11. In this case Adler's method seems best for large time steps. Also shown are a few points obtained with Cohen's method without Adler's modification. In this case the modification clearly helps.

The small squares labeled LT (Laplace Transform) represent the method originally suggested by Ash¹⁰ and developed by Keepin¹¹ into the RTS code. The method works for the thermal reactor cases, although it is rather less efficient, requiring 10^{-3} sec time step for good accuracy. For the fast reactor cases it fails completely, and the points in Fig. 11 were obtained using a special modification in which fast-decaying transients were neglected. Good accuracy still requires too many time steps.

II. Analytical Approximations for Calculating Reactor Excursions

In 1964 in a thesis at Harvard, Canosa¹² showed that the problem has a first integral as shown in Fig. 12. Using one-delay group dynamics with a prompt linear shutdown coefficient and without heat loss, he obtained an n, T , power-temperature phase-plane portrait for this problem. It is a third-order problem, and this represents a projection of a three-dimensional trajectory.

Typical phase planes are shown in Fig. 13. On the left is the subprompt-critical reactivity step input, and on the right is the superprompt-critical step input. The dotted line labeled ZSI is the zero-slope isocline, and the trajectory must cross the dotted line at the instant of peak power. It turns out that in the subprompt-critical case the ZSI is identical to the prompt-jump approximation so the solution stays near the dotted line.

For a superprompt-critical step, the hyperbola flips and two branches are showing. The parabolic part of the trajectory is the old Nordheim-Fuchs model. When the trajectory comes down, it switches over following the dotted line. One can indeed start with a simple Nordheim-Fuchs model; and when the fast burst is over, switch to the prompt-jump approximation and carry it the rest of the way--a complete slide rule calculation for a fast excursion including the delayed-neutron tails.

Fig. 10--Peak neutron density vs. computing interval size for various computing methods: $n_0 = 1$, reactivity amplitude = 23 cents, and oscillation period = 700 sec.

Fig. 11--Peak neutron density vs. computing interval size for various computing methods: $n_0 = 1$, reactivity amplitude = 68 cents, and oscillation period = 100 sec.

$$\dot{n} = \frac{\rho - \beta}{\ell} n + \lambda c$$

$$\rho = \rho_0 - \alpha T$$

$$\dot{c} = \frac{\beta}{\ell} n - \lambda c$$

$$\dot{T} = K n$$

$$K \ell n \frac{dn}{dT} = (\rho_0 - \beta - \lambda \ell) n + \frac{\lambda \rho_0}{K} T - \frac{\lambda \alpha}{2K} T^2 - \alpha n T$$

Fig. 12--Adiabatic point reactor with one group of delayed neutrons and the first integral.

Fig. 13--Power-temperature phase planes for step-induced excursions.

Canosa did not consider the transitional case (prompt-critical). The hyperbola degenerates to the straight lines as shown in Fig. 14. As the neutron generation time decreases, the trajectory is pushed into the corner and looks like a triangle. The energy produced by the compensated reactivity at the time of peak power can be extremely small.

This was observed at Phillips in the SPERT I project and also at AI in the homogeneous reactor safety experiment. If one calculates instantaneous reactivity from a power trace and plots the part compensated up to peak power as a function of the initial step, it goes through a shallow but low minimum near a prompt-critical step input. For a number of years it was thought that one might learn something about the mechanism of shutdown by studying this minimum, but it is now clear that this is a delayed-neutron phenomenon.

Figure 15 shows peak power and energy at peak power in terms of the reactivity step with the subprompt-critical case shown first followed by the superprompt case and finally the transitional case.¹³ Note the approximate square-root dependence of energy at peak power on ϵ (neutron generation time) for a prompt-critical step. Figure 16 shows the asymptotic expansions rewritten in terms of the initial (maximum) inverse period ω by means of the inhour equation.

Figure 17 shows peak power as a function of the initial inverse period for a step input. The straight line of slope two on the right is the Nordheim-Fuchs fast-transient peak power. Note the two different neutron generation times representing a thermal and a fast reactor.

At the other extreme in the lower left are two dotted lines which overlap. These are the prompt-jump approximation solutions. The one labeled λ is computed using the one-group slow-transient equivalent decay time, and the one labeled λ' has the weighted mean decay constant for a fast transient. The little circles representing digital computer results drift from one curve to a second and to a third as the time scale decreases. Furthermore, the asymptotic series of Figs. 16 and 17 locate the transition from slow to fast. This is a simple recipe for predicting peak power over the full range of transients.

Figure 18 shows the energy at the peak as a function of the initial inverse period. The parameter is β over beta, and for a fast reactor there is the deep minimum in the neighborhood of prompt-critical. The circles were computed by one of the three digital methods described earlier, and this shows that the same mathematical algorithm is suitable from a slow transient to a fast transient.

As suggested in connection with Fig. 15, if all those valleys are connected together, one sees roughly a square root dependence for the height of the minimum as a function of β as shown in Fig. 19. This is a graph of energy at peak power for the prompt-critical step as a function of neutron generation time. The lower dotted line is a square root, the first approximation to the solution of the transcendental equation shown in the figure; and the upper dotted line is just a second approximation. The black dots are digital computer results.^{1, 14} The white circles are estimates obtained by representing the power burst empirically by a series of exponentials in time.¹⁵

Fig. 14--Power-temperature phase plane for prompt-critical excursion.

$$\hat{n} = \frac{\lambda}{\alpha K} \left[\beta - \sqrt{\beta^2 - \rho_o^2} - \beta \left(\frac{\beta}{\sqrt{\beta^2 - \rho_o^2}} - 1 \right) \epsilon + O(\epsilon^2) \right]$$

$$E(\hat{n}) = \frac{1}{\alpha K} \left[\rho_o - \beta + \sqrt{\beta^2 - \rho_o^2} - \sqrt{2\beta(\beta - \sqrt{\beta^2 - \rho_o^2})} \epsilon + O(\epsilon) \right]$$

$$\hat{n} = \frac{\lambda}{\alpha K} \left[\frac{(\rho_o - \beta)^2}{2\beta\epsilon} + 2\beta \ln 2 + O(\epsilon) \right]$$

$$E(\hat{n}) = \frac{1}{\alpha K} \left[\rho_o - \beta + \frac{\rho_o + \beta}{\rho_o - \beta} \beta \epsilon + O(\epsilon^2) \right]$$

$$\hat{n} \approx \frac{\lambda \beta}{\alpha K} \left[1 - \frac{1}{2} \sqrt{\epsilon \left(\ln \frac{2}{\epsilon} - 2 \right)} \right] \quad E(\hat{n}) \approx \frac{\beta}{\alpha K} \sqrt{\epsilon \left(\ln \frac{2}{\epsilon} - 2 \right)}$$

Fig. 15--Peak power and energy at time of peak (subprompt-critical, superprompt-critical, prompt-critical).

$$\hat{n} = \frac{\lambda \beta}{\alpha K} \left\{ 1 - \frac{\lambda}{\omega + \lambda} \sqrt{1 + 2\omega/\lambda} + \left[1 + \frac{(\omega/\lambda)^2 - (\omega/\lambda) - 1}{\sqrt{1 + 2\omega/\lambda}} \right] \epsilon + O(\epsilon^2) \right\}$$

$$E(\hat{n}) = \frac{\beta}{\alpha K} \left[\frac{\lambda}{\omega + \lambda} (\sqrt{1 + 2\omega/\lambda} - 1) - \sqrt{2(1 - \frac{\lambda}{\omega + \lambda} \sqrt{1 + 2\omega/\lambda})} \epsilon + O(\epsilon) \right]$$

$$\hat{n} = \frac{\lambda \beta}{\alpha K} \left[\frac{1}{2\epsilon} \left(\frac{\ell \omega}{\beta} \right)^2 + 2 \ln 2 - 1 + O(\epsilon) \right]$$

$$E(\hat{n}) = \frac{\beta}{\alpha K} \left[\frac{\ell \omega}{\beta} + \left(1 + \frac{\beta}{\ell \omega} \right) \epsilon + O(\epsilon^2) \right]$$

Fig. 16--Asymptotic expansions with initial inverse period as parameter (subprompt-critical, superprompt-critical).

Fig. 17--Peak power vs. initial inverse period.

Fig. 18--Energy at peak vs. initial inverse period (normalized to shutdown coefficient of one dollar per unit energy).

Fig. 19--Energy at peak in prompt-critical excursion vs. neutron generation time.

Triangles pointed up are from SPERT I data,¹⁶ and the square is from KEWB reactor data.¹⁷ Those reactors have complicated feedback mechanisms, and yet they fall right in line suggesting that this really is a delayed neutron phenomenon. Also shown is a point for TREAT, the thermal graphite pulse reactor¹⁸ and a point from the SNAPTRAN experiment.¹⁹ Some recent fast-reactor data falls a bit off-scale just above the dotted line in the lower left corner.

III. Nonlinear Reactor Stability

Figure 20 shows a formulation of the two-temperature feedback problem but including qualitative effects of delayed neutrons by using for G the zero power transfer function in the prompt-jump approximation. This changes the picture considerably from what has often been done in stability problems, i.e., using the effective lifetime model which contains no qualitative effect of delayed neutrons. Using the prompt-jump approximation, there is still a third-order system instead of fourth or higher; and in the linear case there results a simple equation²⁰ shown in Fig. 21. The hyperbola in the parameter space of the reactivity coefficients serves as a boundary between stable and unstable regions (the resonance line); it is the boundary in the conditionally stable region which may be crossed just by raising the power level.

Figure 22 shows a typical hyperbola, the line CDE, obtained with the prompt-jump approximation; and the straight line AB is the resonance line for the effective lifetime model. "Everyone knows" that delayed neutrons improve reactor stability, and it indeed appears to be the case here. The resonance line is moved to the left. This is typical if the heat transfer coefficients are not too large.

Another case²¹ is shown in Fig. 23 illustrating reactor stability. The shaded region can exist if the heat-transfer time constants are short enough. It is a region of stability for the effective lifetime model but instability for the prompt-jump approximation. This is illustrated by the Nyquist plots of Fig. 24, which are for Point F in the shaded region of Fig. 23. As pointed out recently by Smets,²² delayed neutrons do not always improve reactor stability.

So far we have been talking about decoupled regions. Now they are coupled and heat goes from one to the other. Intuitively if both coefficients are negative in the two-region problem, the reactor is always stable; this is not true if the regions are coupled, and the example of Fig. 25 shows this. The reactor will go unstable if the negative coefficients are too large, as predicted by Lellouche.²³ Near the origin is the region that is stable with positive delayed coefficient and negative prompt coefficient. This example is used to remind the students that intuition doesn't always serve; coupled systems are not always stable if the feedbacks are negative.

Figure 26 illustrates with a nonlinear problem how well the prompt-jump approximation works in these stability problems. The trajectory labeled "exact" is a complete calculation in that it contains the neutron generation time. Note the pip at the start, which is the initial fast transient. It decays quickly, and the trajectory closely follows the asymptotic curve (the prompt-jump approximation).

$$G = \frac{n_o}{\beta} \frac{s + \lambda}{s}$$

$$H_1 = \frac{a_1 K_1}{s + \gamma_1}$$

$$H_2 = \frac{a_2 K_2}{s + \gamma_2}$$

$$1 + GH = 1 + G(H_1 + H_2) = 1 + K \frac{(s + \lambda)(s + \xi)}{s(s + \gamma_1)(s + \gamma_2)}$$

$$K = (a_1 K_1 + a_2 K_2) \frac{n_o}{\beta}$$

$$\xi = \frac{a_1 K_1 \gamma_2 + a_2 K_2 \gamma_1}{a_1 K_1 + a_2 K_2}$$

Fig. 20--Linear two-temperature reactor model.

$$K\xi > 0$$

$$K + \gamma_1 + \gamma_2 > 0$$

$$\begin{aligned} & (\lambda + \gamma_2) x^2 + (2\lambda + \gamma_1 + \gamma_2) xy + (\lambda + \gamma_1) y^2 \\ & + [\gamma_1(\lambda + 2\gamma_2) + \gamma_2^2] x + [\gamma_2(\lambda + 2\gamma_1) + \gamma_1^2] y \\ & + \gamma_1 \gamma_2 (\gamma_1 + \gamma_2) > 0 \end{aligned}$$

$$x = \frac{a_1 K_1 n_0}{\beta}$$

$$y = \frac{a_2 K_2 n_0}{\beta}$$

Fig. 21--Conditions for asymptotic stability in the linear two-temperature reactor model.

Fig. 22--Reactivity-coefficient parameter space (region 1 fast; heat transfer times long compared to mean lifetime for delayed neutrons).

Fig. 23--Reactivity-coefficient parameter space (region 1; heat transfer times short compared to mean lifetimes for delayed neutrons; shaded regions destabilized by delayed neutrons).

Fig. 24--Nyquist plots for system at Point F in Fig. 23.

Fig. 25--Reactivity-coefficient parameter space for a thermally coupled two-temperature reactor model.

Fig. 26--Phase-plane trajectories illustrating use of prompt-jump approximation.

Figure 27 shows how the linearly unstable system represented by Point F in Fig. 22 is converted by nonlinearity into a stable limit cycle.²⁰

Figure 28 shows some further nonlinear examples.²⁴ Case A is linearly stable but becomes unstable for sufficiently large displacements; the transition is an unstable limit cycle. Case B is on a resonance line (sustained oscillation of a linear system); the stable curve for Case B is a highly magnified trajectory inside the unstable limit cycle. Case C is in the linearly unstable region. Case F is linearly stable, but its region of nonlinear stability in state space is much larger than for Case A.

Case G is noteworthy because it is linearly unstable both with and without delayed neutrons. With nonlinear neutronics it is a stable limit cycle in the absence of delayed neutrons. With delayed neutrons, as shown, the oscillation is not bounded suggesting further cases of "destabilization by delayed neutrons" that are outside the scope of linear analysis.

It is not possible to discuss nonlinear analytical studies here. Work is being extended elsewhere by including delayed neutrons with complicated feedbacks using Liapunov's second method and various series expansion methods for describing limit cycles.

IV. Some General Comments on Nuclear Safety

As the educational program is broadened, one of the big assets is the past experience of university faculty members in safety research and evaluation.

Safety is an integral part of good design and is taught in engineering courses. The training of nuclear safety technicians could also be conducted, but this has not been tried at Arizona. This, like the training of reactor operators and processing plant technicians, could be done by colleges to some extent; but it will be taken care of mostly in junior colleges, trade schools and in the industry itself.

One needs to think about the perspective of the graduate nuclear engineer--the perspective he should have as he looks at the world and at the role power reactors will play in the world. Safety in the nuclear business is not thought of in the same way as is safety in other technological industries. The main reason for this is not the nature of the hazard but the present poor state of public acceptance, which may be partly our own fault. For 20 years the case has been overstated by pointing with pride to impressive extremes of safety precaution, and that is enough to scare most laymen who say, "If it is really as safe as all that, then why these extreme precautions?" Maybe it is our fault to some extent that the nuclear industry, which has a clearly superior safety record, has less social acceptability than the explosives industry with its occasional instances of public damage. Perhaps we became overly conservative partly because of weapons psychology and partly because of fear of the unknown. Even as recently as 1965, Popular Science magazine quoted the minimum critical mass for a bomb, compared it with the mass of fissile material in a fast breeder, used this ratio to scale the bomb's explosion damage and then asked, "Is atomic industry risking your life?"²⁵ As for fear of the unknown, it might be kept in mind that radiation is only one of many modern

Fig. 27--Stable limit cycle in linearly unstable system (Point F in Fig. 23).

Fig. 28--Nonlinear phase plane trajectories for the system of Fig. 23.

industrial hazards that could conceivably be fatal without advance warning to unaided human sensory organs.

The problem is one of public acceptance. The U. S. Atomic Energy Commission, American Nuclear Society, Atomic Industrial Forum and others are trying to do something about it in the world of commerce and industry; but we can contribute on the campus. However, a new required course in public acceptance would be just waving a red flag, and there is not room for it in the curriculum. We can help the ANS student chapters to sponsor programs on this topic, and we can try to include bits of myth--exploding information--in lectures in all courses and in other contacts with the students. We are responsible for keeping informed about what is happening in the battle for public acceptance, and we are doubly responsible for keeping students up to date and interpreting this information to them so they may profit from the history that is being made while they are still in school. In particular the students will have a much better chance of preventing or correcting public misunderstandings (like the superbomb of Popular Science magazine) if they have the background to be able to discuss such things knowledgeably and confidently.

The following is another public misunderstanding: The myth of thermal pollution was circulated recently in Southern California,²⁶ aided and abetted by Ramparts magazine.²⁷ The supposition was that nuclear plants are worse than conventional coal-fired power plants because nuclear plants continually dump heat into the ocean. The coal industry has suddenly learned how to circumvent the second law of thermodynamics.

There is the earthquake myth, too. Faculty and students should know that the Advisory Committee on Reactor Safeguards recommended in favor of Bodega Bay only after they were convinced that a containment breach could not occur without an earthquake severe enough to solve the overpopulation problem in California and other western states for many years. In the context any resultant fission product dispersal would be irrelevant, and yet the ACRS has been cast as a major villain by the California conservationists.

All students should know a little history, particularly of the "fable of the deadly wristwatch." When Edward Teller and Linus Pauling were principals in the H-bomb fallout controversy some years ago,^{28, 29} each of them calculated, or their followers calculated, the relative danger of the bomb test fallout as compared with wearing a radium-dial wristwatch. Teller found in favor of the fallout by about a factor of 10, and the other people found the wristwatch to be less damaging by several orders of magnitude. From that day until the bomb test treaty, SANE had a field day with the "fable of the deadly wristwatch." Secondary consequences of a massive reactor accident are probably just as difficult to calculate accurately.

Dr. Teller agrees, and it is interesting that he also says in the August, 1967 issue of Nuclear News,³⁰ "For the fast breeder to work...you probably need something like half a ton of plutonium...I do not like the hazard involved." This may be good news to the promoters of thermal breeders, but not to the Public Relations Department at Detroit Edison.

Figure 29 would illustrate the ideal reactor-safety predictor, as suggested by the late Mark Mills and others, if it only had some numbers on

Fig. 29--The ideal "reactor safety predictor."

it. The dashed line is the boundary between acceptable and unacceptable conditions. All reactor licensing decisions would be greatly simplified if it were not that there are uncertainties in both ordinate and abscissa by factors of at least 1000.

Progress will come as people try harder to calculate numerical probabilities for the occurrence of unlikely coincidences. Missile men are used to doing this under the labels "operations research" and "reliability analysis." It is not a new idea in the nuclear industry. Technical Operations, Inc. was working on probabilities in reactor safety more than 10 years ago, and Planning Research Corporation of Los Angeles has recently been studying power reactor reliability in terms of probabilities. Atomics International--notably with the work of Griffin³¹ and Willis and Carlson³²--is concentrating on the application of "fault trees" to nuclear accident probability evaluation. More recently, Hoglund and Kelber of Argonne National Laboratory have begun work in this area. Maybe all this has been slow to catch on because it is more fun to calculate probabilities for unlikely successes, as in rockets, than for improbable disasters, as in commercial power reactors.

After the calculations are made, the location of that boundary line in Fig. 29 is ultimately a political decision. In the meantime everyone is arguing about which of those two arrows we are on as the power of successive reactors is raised. We are somewhere in between, but who knows?

By now it should be clear that I am pleading for realism in reactor safety. There is a great deal yet to be done: more data gathering, development of evaluation methods, prediction of risks. There is also a lot to be done in the area of public education. If the objective is to save the beauty of the beach, that is good; but if the beauty of the beach is to be saved only as a result of unprincipled attacks on the safety of the reactor and the integrity of its designers, then someone is letting the ends justify the means in a vicious and irresponsible way.

Everyone must realize that there is a big stake in this type of public controversy and that it is not enough to laugh off the opposition as know-nothing do-gooders. I hope I have illustrated what I have meant by "perspective that a graduate nuclear engineer should have." Maybe what I really mean here is that it is a question of social awareness or savoir-vivre. I would like to think there is more to an engineering college than technical courses, assistantships and thesis writing; and that we are not so far estranged from the liberal arts and humanities, and from just plain people, as it sometimes seems.

Figure 30 illustrates what I meant when I said that some of our troubles may be our fault.

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Fig. 30--"Gridley, how many times do I have to tell you? We do not refer to the reactor core as a ground zero!"

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DIRECT ENERGY CONVERSION PROGRAMS AT NASA

Roland Breitwieser
National Aeronautics and Space Administration

The usual justification for direct energy conversion is illustrated in Fig. 1. It is an artist's comparison of a 300 kW photocell system to an advanced reactor system. The point is that a large amount of surface is involved if the radiant energy from the sun is to be captured. At some point the deployment of these surfaces becomes somewhat unmanageable. The scaling factors are quite approximate as are the comparative weights. The weight is well over 100,000 pounds for the photocell and battery system. Much of the weight is based on off-duty cycle assumption and a high-peak load requirement for the batteries. With these constraints the justification for the advanced reactor comes through. Optimistically, the advanced reactor is on the order of one-fifth of the weight of a photocell battery system.

Research on photovoltaic cells at NASA's Lewis Research Center has attempted to reduce the size and weight of the solar conversion approach. Although it is desirable to improve the efficiency of the silicon cell, only small gains from the present 10 to 15 per cent range can be anticipated.

The primary emphasis is now placed on providing a greater degree of performance stability in terms of space environment. Also, some emphasis is being placed on easing the deployment of the large surfaces by using photovoltaic cells in the form of flexible films. Flexible film cadmium sulfide cells have been developed with efficiencies that are interesting, although below that of the silicon cell. Deployment will probably be improved as the flexible cells evolve, but size due to efficiency problems will remain large. With these problems photovoltaic cells are and will remain the primary electrical space power source for some time, particularly at low power levels. What is desired, of course, is a compact system using a concentrated energy source such as a nuclear reactor.

Several approaches may be used for converting nuclear energy as heat to electricity. Figure 2 shows an advanced nuclear system using a liquid-metal Rankine cycle. The long term requirements of 10,000 hours of space, completely unattended, is a bit challenging for a system having this many moving parts. Some savings in terms of complexity can be made by using the Brayton cycle, shown in Fig. 3. However, simplicity is achieved at the expense of a less desirable thermodynamic cycle for space application, which results in the requirement for a larger radiator.

MANNED ORBITING STATION

300 KW POWER SYSTEMS

ADVANCED
REACTOR SYSTEM

PHOTOCELL
SYSTEM

PHOTOCELL	WEIGHT, LB	AREA, SQ FT
PHOTOCELLS AND BATTERIES	138,000	69,000
ADVANCED REACTOR SYSTEM	24,000	1,200

Fig. 1--Artist's comparison of a 300 kW photocell and an advanced reactor system for a manned orbiting station.

POTASSIUM-RANKINE SYSTEM

Fig. 2--Advanced nuclear system using a liquid metal Rankine cell.

BRAYTON CYCLE SPACE POWER SYSTEM

Fig. 3--Space power system using the Brayton heat cycle.

Thermoelectric and thermionic conversion systems have considerable appeal because they do not require complex electrical power generation equipment. There has been a tremendous amount of money and time spent in the attempt to develop more efficient thermoelectric converters that have a greater high temperature capability. Unfortunately, as the temperature capability is extended by going to materials such as silicon-germanium, the efficiency drops. The probability of a major breakthrough in terms of efficiency and high temperature operation is slight. The phonon contribution to conduction in solids and diffusion in solids are the factors that limit the application of thermoelectric devices.

As one looks for approaches to the conversion of heat to electricity that possess better physical stability and can possibly provide better efficiency, a converter that incorporates a physical barrier such as a solid-gas interface looks appealing. The thermoelectric plasma diode is such a device. Upon further examination, junction rectification appears more effective than electrical power generation through the thermoelectric power of a plasma. Thus the evolution into thermionic conversion.

Some estimates of the relative weights of the nuclear Rankine, the nuclear Brayton and the nuclear thermionics system's are shown in Fig. 4. The systems are compared at several temperatures. The point of the comparisons is this: Not enough is really known of the details of the systems to establish a clear superiority for any approach. The thermionic system with converters in the reactors may be competitive with other approaches, and indeed it might be a bit better. Reliability, development and testing costs can easily sway the selection from one system to another.

This is as far as comparative system analysis is warranted. We shall now examine the actual thermionic device in more detail and evolve the background of the uncertainties associated with extrapolation of thermionic-converter information of today to projected systems.

I. Principles of Thermionic Conversion

The basic principle of thermionic conversion is illustrated in Fig. 5. Heat is applied to one metal electrode from an energy source such as a nuclear fuel. As the metal temperature increases, electrons are boiled out and are captured on the cooler electrode. The emission and capture of electrons provide the current flow through the external load. The voltage that can be obtained is depicted in the potential diagram. The hotter electrode (the emitter) has a relatively large population of electrons in the higher energy levels compared with the cooler electrode (the collector). The variation in the electron density is the driving force that permits the development of a useful voltage. Characteristically this is about three-fourths volt. The thermionic converter is a unique heat engine since it is one of the few energy conversion methods that produces electric power directly from heat over a large temperature interval in a single step. It is also unique in its ideal efficiency. A simplified expression for the efficiency is also given in Fig. 5, where the output term--electric power--is the net current, times the difference in the emitter and collector voltage barriers. The input term is the energy involved in elevating the electrons to the higher energy levels so that the current can flow over the potential barrier. If the potential barrier is large in comparison with

300-KW POWER SYSTEMS

UNMANNED, 10^{11} NVT, 10^7 RAD, 50-FT SEPARATION
(EXCLUDES POWER COND.)

Fig. 4--Estimates of the relative weights of the nuclear Rankine, Brayton and thermionics systems, having a power output of 300 kW.

IDEAL THERMIONICS

Fig. 5--Simplified expression for the efficiency of an ideal thermionic system.

the kinetic energy of the electrons in transit, and if the barriers V_1 and V_2 are adjusted to match the absolute temperature ratio, the Carnot cycle efficiency is approached. These highly idealized conditions neglect the ever present loss terms. Figure 6 removes one from the never-never land of ideality into the reality of the actual thermionic device.

The dictates of the energy source require high power densities and thus high current densities. At the present time the high current densities require high emitter temperatures to produce the necessary flow of electrons. In order to sustain the high current densities, a partially ionized gas must be introduced into the interelectrode space. Electron scattering then exists, and radiation transport becomes significant because of the large temperature difference. Gaseous thermal conduction in the interelectrode space also results. Ion flow negates some of the electron current and also transports energy. In order to extract the electric power, leads must be attached to the hot and cold electrodes. Excessive heat is conducted if the lead is too big, or excessive voltage losses if the lead is too small.

The expression for efficiency departs from the Carnot-like expression to give the practical expression of efficiency shown in Fig. 6. The difference between the expressions in the numerator involves a voltage drop and a correction to the net current that accounts for electrical leakages. The denominator increases because of the need to recognize kinetic energy of the electrons and the summation of the many losses enumerated above.

In practice because of some yet inescapable plasma voltage losses, practical converters are achieving only a bit over one-third of the Carnot cycle efficiency. Even though the present day fraction of Carnot efficiencies of thermionic energy conversion are not among the highest, certain features of thermionics are attracting interest for the potential application to a nuclear energy source, particularly for space application.

Some of these qualities are illustrated in the next few figures. An elaborate nuclear thermionic design is not illustrated. A simplified design has been selected to demonstrate the basic principles and problems of in-pile nuclear thermionics. The three main parts of the system are the reactor, shield and radiator (Fig. 7). The radiator is considerably smaller than for other space power systems since the operating temperatures are higher. Segmented radiators coupled with similar segmented reactor elements are suggested. Each of the reactor segments is further subdivided into unit thermionic cells (Fig. 8). The cells are mounted in a tray and are cooled by liquid metal flowing across the tray similar to the usual crossflow heat exchanger. The arrangement shown has two cells connected in series in each cylindrical station. Each double converter module should develop about 1-1/2 volts. Conceptually, one can visualize the thermionic reactor as a group of 1-1/2 volt batteries connected in some series--parallel network to provide both the necessary voltage and current and to provide reliability through redundant cross connections.

A typical individual cell consists of encapsulated uranium dioxide in the center, which is electrically isolated by an insulator from the collector. The fuel cladding serves as an electron emitter. The electrons flow to the collector, which is typically a trilayer tube constructed of a thin layer of

ACTUAL THERMIONICS

$$\eta_{EL} = \frac{(J_{NET})(V_1 - V_2 - \Delta V)}{J_1(V_1 + 2kT_1) - J_2(V_2 + 2kT_2) + \sum \dot{Q}} \neq \eta_{CARNOT}$$

Fig. 6--Practical expression for the efficiency of an actual thermionic system.

IN PILE THERMIONICS

Fig. 7--Simplified sketch of an in-pile nuclear thermionic system showing the reactor, shield and radiator.

IN PILE THERMIONICS

CROSS FLOW HEAT EXCHANGER

Fig. 8--Cross flow heat exchanger for the in-pile nuclear thermionic cells.

aluminum oxide sandwiched between layers of niobium. The trilayer tube simultaneously serves as the collector surface current lead and heat exchanger surface, which is cooled by liquid metal. Cesium gas is introduced into the interelectrode space by use of a cesium reservoir or cesium dispenser material.

The main advantage of the in-pile thermionic conversion system is the high source temperature of this heat engine. The nemesis of most space power is the temperature limit imposed by the liquid-metal corrosion limit and/or the high temperature strength characteristics of materials. This limitation is avoided by not exposing the liquid metal to the high temperature source. The liquid metal is only exposed to the collector, which is at the point of heat rejection. The high temperature zones can be lightly stressed since the high temperature surfaces are exposed only to a low pressure ionized gas. The other attractive feature of this system is the modular design. The modular design offers reliability through redundancy and eases testing problems. The system subdivides into a series of building blocks such as radiators, reactor compartments and individual cells. Designs like the one illustrated permit out-of-pile testing of the individual cells and groups of cells. The individual thermionic cells can also be introduced into a test reactor for studies of the nucleonic interactions. It has been said that the scaling-by-multiplicity feature of thermionics permits the development of a power system with a nuclear source that has many of the advantages that photovoltaics has provided the solar source. The disadvantages are obvious. The reactor design and fabrication are not simple.

The reactor complexity can be avoided by an alternate approach in which the thermionic converters are external to the reactor, as shown in Fig. 9. A conventional compact liquid-metal-cooled reactor is used. The hot liquid metal is circulated through a thermionic heat exchanger. The heat is rejected through an additional liquid-metal loop to the radiator. Thermionic conversion is achieved in a multipassage heat exchanger. The system illustrated uses an electron emitter (Fig. 10). The second trilayer tube houses the ionized gas needed for the converter, serves as a collector and is sectioned to provide a series of individual thermionic converters. The annulus formed by the outside tube and the second trilayer tube contains the liquid-metal coolant. Many variations of external thermionics exist. The system shown merely illustrates one of the more obvious features of external thermionics. The advantages of a simple reactor approach coupled with modular thermionics exist, but unfortunately, the liquid-metal loop is exposed to the reactor fuel temperature. Thus, the system is limited to low source temperatures, a region somewhat unattractive to the electronics of thermionics.

If a system analysis is made for either the in-pile or out-of-pile thermionic approach, the performance levels for the converters required to make the system competitive to other power systems becomes established. Roughly, the net power should exceed 5 watts per square centimeter at 2600° F and should exceed 10 watts per square centimeter at 3000° F. The efficiency should approach or exceed 30 per cent of the Carnot cycle efficiency as shown in Fig. 11.

OUT OF PILE THERMIONICS

Fig. 9--Out-of-pile thermionic converter system.

OUT OF PILE THERMIONIC CONVERTER

Fig. 10--Details of multipassage heat exchanger for the out-of-pile thermionic converter.

ACTUAL THERMIONICS

$$\eta_{EL,AV} = \frac{(J_{NET})(V_1 - V_2 - \Delta V)}{J_1(V_1 + 2kT_1) - J_2(V_1 + 2kT_2) + \sum \dot{Q}} \gg 30\% \text{ OF CARNOT}$$

Fig. 11--Expected Carnot cycle efficiency for nuclear thermionic system.

II. Thermionic Conversion Research

The remainder of this discussion concerns the NASA program directed toward achievement of the required efficiency, the power densities and the operating life of thermionic converters.

A. Electron Emission

The physics of thermionic energy conversion reduce to the problem of getting the electrons out of the emitter, across the interelectrode gap and into the collector efficiently. Figure 12 shows the problem of getting the electrons out of the emitter. The high temperatures involved in thermionics require the use of refractory metals. Unfortunately, the same interatomic forces that hold a metal together also bind the electrons to the surface. This electron affinity is described quantitatively by the electron work function. The electron emission rate equation in Fig. 12 is the Richardson-Dushman equation. Using a typical work function for tungsten and plugging in a temperature of 1923°K results in an emission current density of only 0.4 ma. Therefore, the work function must be lowered to provide for more emission. This is achieved usually by depositing a partial layer of a low-work-function metal over the surface of the refractory emitter. Practical thermionic converters usually use cesium. The low-work-function layer is maintained by continuously flooding the emitter surface with cesium gas.

An interesting and fortunate characteristic of the adsorption process is that the more refractory the metal and the higher the work function, the stronger the cesium attachment is to the surface. An experimental correlation recently noted by the Field Emission Corporation illustrates this effect in Fig. 13. The bare work function, i.e., for the metallic surface free of any adsorbed particles, is the abscissa. The maximum change in work function that can be produced by adsorbing cesium on the surface is given in the upper ordinant. The change in work function overcompensates for the initial high value of the substrate, so that the lowest values of work functions of cesiated surfaces are achieved with the surfaces that have the highest initial bare work function. The field emission experiments used in obtaining these data permitted the identification of individual crystal planes. It was noticed that this correlation held for individual crystal index planes of the same metal, and indeed a tungsten surface, oriented to expose a 110 Miller index plane, is one of the more desirable surfaces for the cesium adsorption and retention process. This field emission study and other similar research indicate that high work functions are desired not only for the minimum work function but at the intermediate work function values used in practical thermionic converters.

It appears that the 110 oriented tungsten surface can be produced and maintained in vacuum if the bulk orientation is 110. Although these encouraging results can be obtained from small zones refined monocrystalline samples, the question remains Can we reduce this approach to actual practice? For example, can the oriented surfaces be established on the surface of a cylinder?

Recently, developments in tungsten fabrication by vapor phase deposition of tungsten suggest an approach that is quite promising in this regard. Tubes such as shown in Fig. 14 have been routinely, though not always repro-

GETTING ELECTRONS OUT OF EMITTER

$$J = 120T^2 \text{EXP} - \phi/kT \text{ A/CM}^2$$

Fig. 12--Illustration of the problems involved in getting electrons out of an emitter.

Fig. 13--Experiment work function correlation of bare and cesiated surfaces.

Fig. 14--Vapor deposited tungsten emitters showing variation in crystal structure resulting from reduction of WCl_6 and WF_6 .

ducibly, fabricated by vapor deposition. The vapor-deposited tubes are near theoretical density, are leak tight, exhibit good structural properties and can be deposited in quite complex shapes. The vapor deposition involves hydrogen reduction of a tungsten halide on the heated mandrel. It has been observed that if tungsten hexachloride is used, the bulk sample exhibits a strong 110 orientation parallel to the surface of the mandrel. For tungsten hexafluoride, a 100 bulk orientation results. Photomicrographs (obtained by L. Yane, Gulf General Atomics) of the chloride and fluoride tungsten products after heat treatments are shown in the lower part of Fig. 14. The tungsten hexafluoride sample with its small columnar grain structures should exhibit superior mechanical properties, except that the noninterrupted columnar grain boundaries that propagate through the metal may enhance mass diffusion. A series of preliminary tests were performed on a group of vapor-deposited disks to see if the bulk orientation affected the surface orientation in the same manner as observed in the small zone-refined monocrystalline studies. The results were encouraging.

Additional adsorption studies are being conducted as part of the NASA thermionic program as follows:

1. Analytical
 - a. Binding energies
 - b. Two-species adsorption theory
 - c. Multiple-species adsorption theory
2. Experimental
 - a. Field emission studies
 - b. Molecular beam research
 - c. Diode studies

The analytical work includes calculations of the binding energies of various adsorbates, two-species adsorption theory (tungsten on cesium) and multiple-species adsorption, where the adsorption of cesium, barium or strontium on oxidized and fluorided surfaces has been analyzed. Experimentally, adsorption has been studied by field emission. Molecular beams of cesium, oxygen and fluorine have been utilized to examine the details of dual-species and multispecies adsorption.

The final evaluation still resides in tests in the actual environment of the thermionic converter. Figure 15 illustrates experimental results of the well documented tungsten cesium adsorption system obtained in a research-type thermionic converter. A tungsten emitter whose bulk orientation is basically 110 was used. Current density is plotted as a function of cesium pressure. The curves presented for two emitter temperatures are based on experimental data and are paired through the use of an analytical expression. Current densities as high as 100 amperes per square centimeter have been observed by pulse measurements at the higher cesium pressures.

The unambiguous demonstration of the stability of a 110 tungsten surface at high temperature and high current conditions has not been established. It has been verified, however, that more than adequate emission currents are available in a diode if the cesium pressure is sufficiently high. In terms of relative particle flux, it is interesting to note that, at the high current conditions, 3 to 6 cesium atoms must arrive at the surface per unit time for every

BASED ON DATA FROM DIODE USING 110 ORIENTED W EMITTER

Fig. 15--Effect of cesium pressure on electron emission based on experimental data from diode using 110 oriented tungsten emitter.

electron that is emitted from the same surface. Spacings in a converter, therefore, must be small to avoid electron-atom scattering. Methods exist for getting electrons out of the emitter. The problem remains, however, of getting electrons across the interelectrode space. Figure 16 illustrates some of the problems in getting electrons across, and the following outline presents part of the associated research program:

1. Space Charge
 - a. Application of Langmuir theory to electrons and ions
 - b. Two-species theory
 - (1) ion and electron current-voltage curves
 - (2) stability analysis
2. Elastic Scattering Including Space Charge
 - a. Isotropic collisions
 - b. Energy dependent and angular dependent collisions
3. Cross Section
 - a. Atom-atom
 - b. Atom-ion
 - c. Atom-electron

If no ions are present in the interelectrode space, a large space-charge barrier develops so that the electron flow is reduced to a mere trickle. Ions formed at the surface will reduce the space-charge barrier, but with the exception of thermionic converters which use high source temperatures, the rate of surface ionization does not produce enough ions to permit the transfer of significant electron currents. The surface ion formulation mechanism is now quite well understood as are the modifications to the space-charge theory which includes ions as well as electrons.

The transmission of a substantial fraction of the emitted current requires that ions be produced in the interelectrode space. It follows, therefore, that some of the energy of the emitted electrons must be diverted to produce the ions. A voltage drop results because of both elastic and inelastic electron scattering. The elastic scattering randomizes the directed velocity of the electron, whereas the inelastic collisions absorb energy of the incoming electron by the excitation or ionization of the target atoms. The elastic scattering process is at a reasonable level of understanding. A new technique exists by which the complexity of the simultaneous effects of space charge, energy-dependent cross sections; and angular-dependent elastic cross sections can be treated to generate theoretical current-voltage curves for thermionic converters. A variation of a Monte Carlo transport analysis is used.

As mentioned, experimental and analytic information on the various collision cross sections are being obtained in support of this effort. The atom-atom, atom-ion and atom-electron cross sections are under study. The application of cross section data can be seen in the effect of argon on the elastic scattering of electrons presented in Fig. 17. Argon exhibits an electron-neutral cross-section minimum in the near thermal energy range; this effect is called the Ramsauer effect. This strongly energy-independent cross section provides a stringent analytic exercise when used in electron transport analysis as well as an interesting possible gaseous additive for a thermionic converter.

Fig. 16--Illustration of some of the problems of getting electrons across the interelectrode space.

Fig. 17--Effect of argon on transmission of electrons in planar diode, including angular and velocity dependent elastic cross section.

Argon minimizes ion losses by reducing mobility and thus wall combination. Relatively large ion-neutral cross sections compared with the small electron-neutral cross sections in the Ramsauer region provide the needed mechanism.

The current voltage characteristics (in dimensionless units) have been compared with the noncollisional space-charge solution. The collision parameter refers to the product of pressure (in Torr) and distance (in cm). These data were calculated by the aforementioned Monte Carlo analytical method. The analyses in their present forms are merely a first step in the attempt to provide a rigorous, analytical treatment for electron transport that can account for the detailed nature of the electron-neutral collisions and space-charge interactions.

B. Electron Collection

Figure 18 summarizes a portion of the previous discussion and introduces the problem of getting electrons into the collector. It indicates the approximate type of collector work function required to achieve the required electron current density from the emitter. Also indicated, are the type of plasma sheath, the minimum plasma drop for the ignited or volume-ionization mode of operation involving inelastic collisions and the type of collector sheath needed for the output voltages of a practical thermionic diode. The collector problems relate first to the nature of the collector sheath and are outlined below:

1. Collector Sheath
 - a. Emission spectroscopy (line shift)
 - b. Laser diagnostics (improved spatial resolution)
 - c. Langmuir probe studies
2. Adsorption Processes
 - a. Analytical (emphasis on low work function region)
 - b. Experimental (field emission, molecular beam and diode work)
 - c. Contamination (collector surface "cold" trap impurities)

At times the ineffectiveness of thermionic conversion is attributed to the collector surface, although it is actually caused by the properties of the plasma sheath adjacent to the collector. Attempts to strengthen the analysis of the sheath problem are being made by evolving new diagnostic tools that provide the spatial resolution needed to probe the sheath. This first step relates to a definition of the central plasma properties. The line shift method of emission spectroscopy suggests some simplifications in this regard. The laser, with its excellent optical resolution, is being developed as a probe. Advances in Langmuir probes through miniaturization are also being made. In addition to the problems related to the uncertainties of the sheath at both the collector and emitter, all the adsorption processes must be controlled to produce the desired collector work function. The emphasis is on a lower work function for the collector. If the allowed operating temperatures, the various potential barrier heights required for the emission of the current densities of interest and the plasma drops present in the group of converters of the day are considered, the collector work function should be less than 1.8 eV to prevent serious inroads into the output potential of the devices; for low temperature

Fig. 18--Illustration showing the problem of getting electrons into the collector.

thermionics, the lower the work function, the better. Since a single gas pressure exists in the interelectrode space, it is necessary to attempt to optimize the coverages on both the emitter and collector surfaces with a closely linked arrival rate of the adsorbing species. An additional collector problem is that the collector not only collects electrons, but the various contaminants in the diode as well. The collection of diode debris may seriously affect the adsorption processes as illustrated in Fig. 18. The debris is presented by the black dots.

Visual examination of many thermionic converters gives strong evidence of collector deposits. The severity of the collector deposits is related to diode preparation, emitter and collector materials and possible nuclear fuel clad interactions.

Diffusion of the nuclear fuel through the emitter clad is a source of collector contamination that has received major attention since the fuel cladding may not exhibit long-time, high temperature stability with some of the proposed fuels.

The electron reception characteristics of the collector are affected by extremely small amounts of contamination since a fraction of an atomic monolayer of contaminant may influence the collector adsorption process. Carefully controlled out-of-pile tests have been used to obtain quantitative information on the rates of diffusion through simulated fueled emitters (Table I). The amount of uranium deposited on a simulated collector surface, expressed in terms of equivalent atomic monolayers is shown for 90 mole per cent uranium carbide-10 mole per cent zirconium carbide and uranium dioxide. Two tungsten clads were evaluated for the uranium dioxide. The greatest rate of diffusion was observed for the uranium carbide-zirconium carbide fuel. A possible problem in the use of vapor-deposited tungsten is suggested by the higher rates of uranium diffusion as compared with the powder-metallurgy-product tungsten. The vapor-deposited tungsten was produced from tungsten hexafluoride and possessed the usual columnar grain structure. The large number of columnar grain boundaries propagating through the clad undoubtedly enhanced fuel diffusion rates. Proper selection of fuel type and cladding promises to minimize the amount of contamination to levels where the use of "getters" or the natural absorption characteristics of collector materials can accommodate low rates of material transport.

C. Converter Evaluation

Having established some of the thermionic converter problems, the next few figures deal with the reduction to practice of the aforementioned research. Figure 19 shows the latest type of research converter, which is undergoing preliminary orientation tests prior to in-pile evaluation. The planar configuration was chosen to provide for ease in instrumentation and ease in evaluating new design approaches. The emitter consists of vapor-deposited tungsten bonded in tantalum. The collector is a niobium-niobium alumina cermet-niobium trilayer. The niobium cermet serves as an electrical insulator between the collector and the heat sink. The cermet is made by hot isostatic pressing of niobium spheres that were precoated with high-purity alumina. The vapor-deposited alumina coatings were graded to provide for an increasing alumina concentration toward the center of the cermet. The dark

TABLE I

U DIFFUSION THROUGH W-CLAD EMITTER **20-MIL CLAD, 1000-HR TEST**

FUEL	TEMP,		CLAD	AMOUNT (MONOLAYERS)
	⁰ K	⁰ F		
90UC-10ZrC	2170	3432	VAPOR DEPOSITED W (G.A.)	268
UO ₂	2070	3252	VAPOR DEPOSITED W (G.E.)	12
UO ₂	2100	3320	POWDER METALLURGY W	3

Fig. 19--Advanced type of research converter showing photomicrograph of the niobium-niobium alumina cermet-niobium trilayer collector.

areas in the photomicrograph in Fig. 19 correspond to the alumina, and the light areas correspond to the niobium. The cold resistance of the cermet is typically greater than 10 megohms for a 1-square-centimeter sample. The collector properties, in terms of structural reliability and thermal and electrical characteristics, are encouraging. The emitter has indicated favorable structural stability during the initial series of thermal annealing and degassing processes. To date, performance in the presence of cesium vapor has not been obtained.

Figure 20 presents results of the electrical performance of converters that reflect the technology of the 1964 to early 1966 time period. Power densities observed at conditions corresponding to maximum efficiency are shown as a function of emitter temperature for the converters. The higher performance corresponds to data obtained in a research-type converter built in late 1965 and early 1966. The lower performance data, indicated by data points, are typical of converters fabricated about two years ago and used in tests evaluating the life of thermionic converters. Tungsten emitters and niobium collectors were used for both groups of converters. Insufficient documentation of the detailed properties of the emitter and collector processing, as well as some questions of data accuracy in the life-test converters, precludes a quantitative explanation of the difference in performance. The performance curves are typical of the carefully processed thermionic converter being tested presently and are conservative examples. The ability to assemble large groups of high performance diodes, as projected in system studies, remains to be proved.

The problem of reliable operation has undergone examination in both in-pile and out-of-pile thermionics. Table II presents typical out-of-pile results. The results summarized in the table are from cylindrical diodes with a power density of 7 to 9 watts per square centimeter and an efficiency of 11 to 13 per cent throughout the tests.

These tests explored the effect of unenriched fuel sandwiched in cylindrical emitters. Tests LC-1, LC-5 and LC-6 incorporated uranium carbide-zirconium carbide fuel and suffered from significant performance degradation. Tests LC-2 and LC-3 contained a tungsten-uranium dioxide cermet, and LC-4 was a fuel-free control diode. Although performance levels could be maintained for the noncarbide tests, the cesium reservoir temperature had to be reset to maintain peak performance. No self-consistent major failure mode was observed. The collector deposits resulting from the uranium carbide emitter interaction increased radiation heat transfer and increased the collector work function and thus degraded performance consistently.

So far we have only briefly treated the nuclear fuel problems of thermionics. One obvious problem is the small interelectrode space (about 10 mils) and possible dimensional instability of the encapsulated fuel. Emitter clads for thermionics operate in the temperature range where the clads have low strength. It is necessary, therefore, to consider methods for minimizing the pressure buildup of the fission gases. One possible solution is shown in Fig. 21.

Uranium dioxide releases its fission gases at high temperatures in the plastic flow range. At these conditions, however, vapor loss of uranium

Fig. 20--Comparative thermionic converter performance for tungsten emitters.

TABLE II

THERMIONIC CONVERTER LIFE TESTS

T_E , 2000⁰ - 2100⁰ K (3140⁰ - 3320⁰ F); POWER DENSITY, 7 - 9 W/CM²;
 η , 11 - 13%; ELECTRICALLY HEATED

CO.	DESIGNATION	DURATION, HR	FAILURE MODE	DEGRADATION, %
G.A.	LC-1	7,174	METAL-CERAMIC SEAL	60
	-2	3,235	CRACK IN EMITTER STEM	NONE
	-3	10,406	SHORT	NONE
	-4	7,345	BRAZE JOINT	NONE
	-5	7,558	BRAZE JOINT	15
	-6	1,200	CONTINUING	^a 40

^aLOW INITIAL POWER.

Fig. 21--Comparative UO₂ vapor losses for vented fuel capsules.

dioxide poses a problem. If a 2100° F cylindrical emitter fueled with uranium dioxide were only partially encapsulated so that one end was left open to vent fission gases, most of the fuel would be lost in an operating time of one year.

III. Conclusions

The discussion has attempted to show the reason for thermionics and some of the uncertainties present in thermionics. No fundamental road-blocks to a successful system development are apparent. There are many individual cases of excellent structural characteristics, fuel stability and converter performance which lead to logical system design and testing concepts. However, the small number of tests and the statistical scatter of results certainly indicate the need for more extensive and more controlled testing of many subparts of this potentially attractive space power system. An example of the cause of data scatter is the inability to document emitter temperatures (an all-important parameter) in in-pile tests.

DISCUSSION

QUESTION: Why are there different crystallographic orientations for the decomposition of tungsten hexachloride than for tungsten hexafluoride?

ANSWER: A group at Oak Ridge is trying to answer this question from both the experimental and analytical viewpoints. Since specific information is not available, I will give you a generality. The decomposition is related to a temperature-time relationship, and different surface orientations result. In general the low index planes that correspond to minimum free energy surfaces are the surfaces exposed.

QUESTION: Referring to the Monte Carlo program, is it able to account for nonlinear terms such as possible recombination?

ANSWER: It is possible, but now we have to worry about keeping the program restrained to a reasonable amount of machine time. The properties of the program can handle inelastic collisions. It can handle recombination processes. The technique used involves slicing the transport zones into small sections and then describing the nonlinear conditions in these sections.

QUESTION: In trying to write equations which describe the surface physics from basic first principles, have you obtained answers which are within 0.3 eV of the measured work function for a cesiated surface?

ANSWER: The state of the art in terms of trying to solve the problem quantum-mechanically, as it should be done, is reflected by my experience earlier. A young man involved in using this approach came to see me with a long face and tears in his eyes and said, "Unfortunately, when I put cesium on the surface the work function is going the wrong direction." It may be that he experienced the problem that you often run into in all the quantum-mechanical approaches. The solution is direct if the interaction under analysis does not distort the system excessively, i.e., a small perturbation. The nature of the "animal" in cesium adsorption is that a major perturbation exists. This is where the theory starts to break down. I have heard the comment if we can't (quantum-mechanically) analyze the tensile strength of solid material effectively, how can we quantum-mechanically analyze the complex gas-surface interface?

One of the things that worries me is a common feature of most surface adsorption models. The theories are based on a static building-block approach. The experimental work such as field emission microscopy gives a visualization of the process. Real life is not anything like what most of the theorists project. The adsorption gas at the temperatures of interest for thermionics behaves like a highly agitated two (or more) dimensional gas on the surface.

DIRECT ENERGY CONVERSION STUDIES WITH EMPHASIS ON THOSE INVOLVING NUCLEAR RADIATION

G. H. Miley
University of Illinois

The direct energy conversion activities at the University of Illinois are scattered throughout the following four departments that have related activities: Nuclear Engineering, Electrical Engineering, Mechanical Engineering and Aeronautical Engineering. In 1966 a committee was formed to answer the question of whether or not direct energy conversion should be organized as a separate interdisciplinary field, much the same as nuclear engineering. After looking at this question, the committee decided it should not be, but that an informal association and close communication should continue between the activities in the various departments.

There are three main courses at Illinois that deal with direct energy conversion. One titled Direct Energy Conversion is taught at the senior or first-year graduate level in mechanical engineering. A generalized magneto-hydrodynamics (MHD) course is offered in aeronautical engineering, and there is a graduate level special topics course in nuclear engineering. Both the direct energy conversion and the special topics courses have a modest number of laboratory experiments which were described at an earlier ANS meeting.¹

A basic problem in direct energy conversion, from an educator's point of view, is that it is as complex, as interdisciplinary, as is nuclear engineering itself. Someone who is interested in specializing in this field at the graduate level must consider a multitude of possible courses. Usually it boils down to a question of whether or not one wants to concentrate in solid state areas or in plasma areas.

At the national level the American Society for Engineering Education has formed a committee to study the educational aspects of energy conversion. Several sessions at the 1968 ASEE annual meeting in Los Angeles will grapple with the problem of "objective criteria" in energy conversion education and also with detailed course structure. Also, the ASEE committee is currently trying to inaugurate a summer practice school for graduate students at one of the NASA laboratories. It would be similar to the current Argonne practice school except it would be for students interested in energy conversion. Another external activity that is rather unique is the organization of the energy conversion activities at the University of Pennsylvania into an interdisciplinary laboratory called the Direct Conversion Institute under the sponsorship of NASA.²

Consider how direct energy conversion is related to nuclear engineering. One might ask why should nuclear engineers become involved in this? There is a motivation from a standpoint of converting energy from the reactor or a radioisotope power source into a convenient energy form, e.g., electrical energy. From the educator's point of view, one has to worry how this fits into the educational program. This involves identifying the fundamental principles that underlie the "Big Four," thermionics, MHD, fuel cells and photovoltaic cells. On the surface these may seem to be similar, but the basic mechanisms are quite different. One has to worry, particularly at a graduate level, about how to present this material in depth without ending up with a survey course. This is particularly important if one wishes to superimpose subject matter of this type on another broad area such as nuclear engineering which is itself an interdisciplinary study. (The details of nuclear engineering at Illinois were discussed at an earlier Gatlinburg meeting.³)

The problem of the proper place for direct energy conversion in the Nuclear Engineering Department was one which was faced several years ago when considering a research and teaching program involving nuclear radiation.⁴ The reasoning was as follows

The existing background of nuclear engineering students could be used as a base without adding additional fundamental courses. Thus, one could launch into a fairly high level graduate course in an area which represents an obligation for a nuclear engineer. The subject matter is such that only persons interested in nuclear engineering are likely to grapple with it. In contrast, other areas in direct conversion, such as thermionics, which involve a heat cycle draw people from a wide range of disciplines, because they have much broader applications.

Several other thoughts in the selection and definition of this work should be noted. The final form of energy in these studies has never been restricted. Electrical energy is not necessarily the ultimate goal. Instrumentation is not ruled out because ending up with power for some high-power operation is interesting. The differences between instrumentation and power concepts are not that great. The major difference lies in the objectives.

Looking back to thermionics, there is a 1957 article in one of the IEEE journals that said that thermionics would never be practical because of an inherently low efficiency. The author apparently based this conclusion on the then state of the art of electron tube technology. It was clear within a year that he was wrong. There is a subtle distinction, and it is often difficult to predict when conditions will be "ripe" so that instrumentation concepts can pass over into power concepts.

I. Research Involving Nuclear Radiation

The influence of the above philosophy will become apparent in the description of some of the activities in the area of energy conversion involving nuclear radiation.

TABLE I

Direct Energy Conversion Involving Nuclear Radiation

1. Direct Collection of Charged Particles

Examples	Fission Fragments - F. E. C
	Gamma Radiation - γ E. C. (Compton or Photoelectrons)
	Alpha Particles - α -Cell
	Beta Particles - Kr-85 Battery

2. Radiation Induced Ionization and Excitation

a. Direct Conversion

Examples - Ionization Electric Cell, Chemonuclear,
Laser

b. Space Charge Neutralization in Thermionics

c. Nonthermal Ionization in MHD

As outlined in Table I, two major areas are involved. One has to do with direct collection of charged particles, and the other has to do with radiation-induced ionization and excitation. One way to convert nuclear radiation energy might be to erect a large potential barrier in its path, and direct collection effectively does just that. As an example, visualize a vacuum capacitor with one plate coated with a charged particle emitter. These particles go to the other plate and charge it, building up an electrical potential. Additional particles leaving the emitting plate are forced to overcome this potential, thus converting their kinetic energy to potential energy. This is a brute-force method, but it is simple.

One might ask whether or not the same thing could be accomplished using a number of smaller barriers. This is in effect what happens in the ionization and excitation route. Essentially atomic level potentials represent, instead of one monstrous potential barrier, a number of smaller barriers which convert the radiation energy to potential energy or other forms of energy. The problem here is that each step typically turns out to be inefficient. The result is once the energy has passed from its original form to ionization-excitation, it has already been randomized, thus it ends up in several forms. If one is trying to build a cycle which operates on a single form, e.g., potential energy, the others are lost.

Examples of these concepts are also listed in Table I. Direct collection devices can be classified according to the source used, e.g., radioactive sources such as alpha or beta particle emitters, fission fragments from nuclear fission or Compton electrons from gamma interactions. Consider radiation induced ionization. One could actually convert radiation energy to electrical energy by producing ionization and then separating the ion pairs, i.e., ions and electrons. For example, consider the irradiation of a gas confined between two plates. This sounds like an ionization chamber. However, the ion chamber uses a voltage difference imposed by a battery or other external power supply to separate the ion pairs. It can easily be shown that

this does not produce a useful power output. What is needed is an intrinsic driving force to separate the ion pairs. One straight-forward way to do this is to use plates constructed of different materials, i.e., use the difference in work functions between the two plates to produce the voltage difference. This has been done, and Ohmart Corporation makes such devices commercially as part of a gamma gauge.⁵ Using the Illinois TRIGA as a radiation source, a similar device has been studied.⁶ The following is a quick mental calculation of the efficiency which might be obtained. Consider work function differences of the order of 1 or 2 volts. It takes approximately 30 eV to form an ion pair. Thus if one were able to collect all of the ion pairs produced, an ideal efficiency of $(1-2)/30$ or ~ 6 per cent might be obtained. This is, of course, an upper limit. Efficiencies have dropped in this multistep barrier concept. Still the concept may be valuable in special situations.

There are other ways of providing the driving force to separate the ions, for example, two plates made of the same material but held at different temperatures with a plasma between them. Because of the difference in temperature, the plasma potential relative to each plate will be different, and one can use this potential difference to extract electrical energy. Some experiments of this type have measured unexpectedly large potential differences, and this is known as the "Klein Effect."⁷

If electrical energy is not wanted, there are other possibilities. Ionizing radiation can be used to promote chemical reactions in order to derive chemical energy. This has led to the chemonuclear reactor concept under study at Brookhaven National Laboratory and elsewhere.⁸ One could pump a laser with nuclear radiation and obtain optical or infrared radiation. This would amount to changing the radiation wave length or frequency.⁹ Instead of ionization this approach depends upon excitation of the medium.

Other categories which involve radiation have to do with using it more or less as a catalyst. In thermionic devices one could use radiation to produce ionization, i.e., a plasma in the region between the plates and hopefully neutralize the space charge which is one of the fundamental problems in a thermionic diode. This concept has been examined in some detail by Jamerson et al. at General Motors.¹⁰ The same type of thing could be considered in MHD to produce nonequilibrium ionization.¹¹ This would in concept lead to a reasonable conductivity without seeding or high temperatures.

A. Fission-Electric-Cell

The use of the Fission-Electric-Cell as a direct collection device is impractical, but if fission fragments are mentally replaced with alphas or betas, it is easy to envision an Alpha-Electric-Cell or a Beta-Electric-Cell that will work.

As shown in Fig. 1, the FEC is simple in principle. Fission takes place in a uranium layer on one plate. The charged fission fragments emerge with a net electrical charge of about $+20e$. One tries to collect them on the collector plate. A quick mental calculation for fission fragments having an average energy of approximately 80 MeV and an average charge of $+20e$ shows voltages of 4 million volts. This is a problem. These voltages cannot be controlled, especially in the radiation field that would be present in such a cell.

Fig. 1--Parallel plate fission-electric cell.

The FEC concept was first proposed in 1944 by Wigner. It was not considered seriously, however, because it appeared to be equivalent to building a vacuum tube reactor. All the space between plates had to be hard vacuum to provide electrical insulation and also to prevent energy losses from the charged particles traversing the space. In 1957 after some work on a cavity reactor, Safonov realized that it would be ideally mated to the FEC.¹² The cavity reactor concept involves the low density region in the center, and this could be envisioned as a space containing the cells. This would lead to practical weight and uranium requirements.

Figure 2 shows Safonov's early design for such a reactor. It was designed to produce ~ 200 MW with a critical mass of 8 kg of uranium-235. It illustrates a basic principle which should be stressed. The FEC does not represent a heat cycle. Thus it is not limited by Carnot efficiencies, so for efficiency purposes the temperatures are not important. Thus Safonov concentrated on the possibility of a low temperature reactor (<120° C) so he could select materials with optimum neutronic characteristics. He selected aluminum, magnesium, etc. for the structure.

The low temperature approach has not been pursued in recent years. The opposite extreme appears attractive for space applications. In this case the waste heat radiator for a nuclear power supply could run at almost the same temperature as the reactor without harming the cycle efficiency. Both temperatures would be high, approaching material limits. This would minimize the radiator weight which is important; since in space applications, the radiator is a key item in the overall system weight. A number of conceptual designs of this type have been investigated in some detail at both Battelle Memorial Laboratory and the Jet Propulsion Laboratory.^{13, 14} Their results were moderately encouraging.

Consider Figs. 3 and 4 as a simple calculation of the efficiency of a planar cell. Basically the problem is that during fission, the fission fragments are produced with an isotropic angular distribution. Thus half of the fragments go in the direction of the emitter (cathode) support plate and are absorbed or lost there immediately. Also particles with too large an angle θ will not have a large enough velocity component in the x-direction to reach the anode when it is at a high potential. The maximum allowed value for a simplified case is calculated in the figure. Following the simple energy and momentum balances indicated, the efficiency is found to be $1/2(1-\sqrt{\beta})\beta$ where β represents the voltage normalized by the maximum allowed voltage for an open circuit. This result is plotted in Fig. 5. A maximum efficiency of 7+ per cent is seen to occur at $\beta = 1/2$, or approximately 2 million volts.

This is a highly idealized calculation. In effect it gives an upper limit for a planar cell. It might be said that the whole concept seems low, but one should take a second look and ask what has happened? After all, this concept is not limited by Carnot efficiencies, so why hasn't a higher efficiency been found? The answer goes back to the emission characteristic; isotropic emission. As already noted, half of the fragments were automatically lost in the support plate. Others with the wrong angle just were not energetic enough to get to the collector. In contrast, if all of the fragments came out perpendicular to the fuel layer it is easy to see that this idealized calculation would show 100 per cent efficiency.

Fig. 2--A low temperature reactor for the direct production of electricity from fission (after Safonov, Ref. 12). Vacuum region contains ~ 20 positive and ~ 20 negative electrodes alternately spaced with ~ 10 cm vacuum gaps. Negative electrodes are coated with 2.4×10^{-3} g/cm² uranium-235. Electrodes may be laminar or concentric cylinders and are internally cooled by D₂O. The critical uranium-235 mass is about 8 kg.

Fig. 3--Idealized cell geometry.

a) MAXIMUM ANGLE OF EMISSION, $\theta_{\max}$

$$v_x = v_0 \cos \theta ; E_x = E_0 \cos^2 \theta \xrightarrow{\min} U^+ \bar{e}$$

$$\text{or } \cos^2 \theta \geq \left[\frac{U^+ \bar{e}}{E_0} \right] = \beta ; \underline{\underline{\theta_{\max} = \cos^{-1} \sqrt{\beta}}}$$

b) NUMBER REACHING ANODE, I

$$I = \frac{\text{AREA} \Big|_{\substack{\theta = \theta_{\max} \\ \theta = 0}}}{\text{TOTAL AREA}} S = \frac{\int_0^{\theta_{\max}} 2\pi r \sin \theta \, r d\theta}{4\pi r^2}$$

$$= \frac{1}{2} (1 - \sqrt{\beta}) S$$

c) EFFICIENCY, η

$$\eta = \frac{I \cdot (\Delta \text{ K. E. / PARTICLE})}{\text{TOTAL EMITTED} \cdot (\text{INITIAL K.E.})}$$

$$= \frac{\frac{1}{2} (1 - \sqrt{\beta}) \phi \cdot U^+ \bar{e}}{\phi \cdot E_0} = \frac{1}{2} (1 - \sqrt{\beta}) \beta$$

Fig. 4--Efficiency of an idealized cell

Assume: plane electrodes; zero fuel thickness;
all fragment to support plate loss;
single energy (E_0) and charge (e) fragments;
perfect grid; and neglect beta decay effects.

Fig. 5--Efficiency and current vs. reduced voltage.

This is perhaps not so far fetched as it seems. One way to approach this ideal would be to go to a spherical geometry. If a point emitter were used, regardless how the fragment came out, it would always have its velocity component directed toward the collector, and this too would give 100 per cent efficiency.

The effect of geometry on efficiencies is summarized in Table II.

TABLE II

Maximum Efficiencies

<u>Geometry</u>	<u>One-sided Emission</u>	<u>Two-sided</u>
Plane	7.4	14.8
Cylindrical	19.2	38.4
Spherical	50.0	100.0

Advantages

Space - nonthermal cycle → minimum radiator,
high power/wt.

- high voltage for ion-propulsion without
conversion equipment

- rugged, no moving parts

Other - possible low temp cycle → improve
neutron economy

- pulsed or ac operation possible

- topping cycle

In the spherical case, half of the fragments may still be lost to the support structure necessary for the emitter in which case the 100 per cent efficiency noted above drops to 50 per cent (this is labeled "one-sided" emission). Similar arguments hold for the cylindrical configuration.

This is an important result because one wants to look at the upper limit. While no one has come too close to these ideal efficiencies, they illustrate goals. The best efficiency was reported by Linder and Christian in 1952 using a Beta-Electric-Cell.¹⁵ This was a low-power cell in the milliwatt range, but it produced over 300,000 volts with a measured efficiency of approximately 30 per cent.

It is a tantalizing thought that quite respectable efficiencies are prevented only by ingenuity in the design and understanding of the phenomena that occur.

Some advantages for the concept are listed in Table II. They usually revolve around the nonthermal characteristics of the cycle, e.g., the minimum radiator weight, reliability, etc. The importance of these factors should not be overlooked. They may in fact be more significant in some applications than efficiency.

Figure 6 summarizes some of the complicating features which will cause deviations from the ideal case. One of the basic problems in direct collection has to do with the transport of the charged particles through the source material. The situation is this The fragments have a range of order of 10 microns in uranium. Thus even for a thin coating energy losses will be significant. The problem is then to understand how to get them out of the uranium layer most efficiently and predicting how much energy they have left after they get out.

In addition one has to worry about secondary electron emissions from the surface as the fragment leaves. It turns out that more secondary electrons emerge than do fragments, and in fact the net negative to positive charge ratio is approximately 20. If the secondary electrons are not stopped from reaching the collector, a negative charge will build up instead of the desired positive charge, which is undesirable. The secondary electrons have low energies, so if they are collected, the device won't work efficiently. Thus a grid or magnetic field must be added to turn the secondaries around.

One must worry about sputtering. When a fragment rams into a surface, pieces of the surface come off--both neutral and ionized particles--and this may also cause leakage currents, and in the extreme case voltage breakdown.

The experiments that have been done are complex, and to date an experiment that appears suitable for university research has not been devised.

Figure 7 shows a triode cell which was studied at the Battelle Memorial Institute. It is called a triode cell because of the squirrel-cage type grid structure which turns back the secondary electrons. This design has been run in the Battelle research reactor for approximately 200 experiments. A typical result is shown in Fig. 8. The net current should be positive for useful power production. However in this case the curve, for some unexplained reason, goes negative above about 1800 volts on the grid. In this and a number of similar attempts, output voltages were always less than a hundred or so volts out of a possible 4 million. This caused the investigators to question the validity of the basic concept. To test this and to explore new areas, similar cells were built with beta and alpha emitters and successful voltages were achieved. Thus it seems that there are some phenomena in the FEC which are not yet completely understood.

Figure 9 shows the cell used in experiments that were done at JPL by Krieve, Heindel and others.¹⁷ They used a magnet to turn around the secondary electrons, rather than a grid. This cell was put in a reactor,

Fig. 6--The Fission-Electric-Cell illustrating the requirements of (1) thin fuel, (2) hard vacuum, and (3) grid.

Fig. 7--Sketch of a cylindrical triode cell (after Plummer and Anno, Ref. 16).

Fig. 8--Typical current behavior of triode cell (after Plummer and Anno, Ref. 16).

Fig. 9--Fission-Electric Cell capsule (after Krieve, Ref. 17).

and again no voltage build up was observed. These studies continued for several years until the cell design shown in Fig. 10 finally evolved. A fairly successful voltage build up was achieved with it, as shown in Fig. 11.

This design presents several important points. First the emitter-collector configuration was inverted, the emitter being on the outside (compare to Fig. 9). Also additional electrodes called sweep rings were added. These electrodes were conceived based on the following logic Krieve proposed that voltage build up did not occur due to a sputtering problem. He suggested that sputtered material produced as the particles hit the collector might in turn be ionized by collision. Trapped electron densities may be high enough so that ionizing collisions would be favored. Specifically, secondary electrons suffering energy losses while in the magnetic field region cannot return to the emitter. They would be trapped in the interelectrode space, spin around for a while and perhaps ionize sputtered material while drifting toward the collector. This would create an unwanted plasma in the vacuum space between the two electrodes, resulting in leakage currents. The sweep rings were designed to provide a positive potential to sweep out trapped electrons. Also inverting the electrode geometry minimizes electron paths, hence, reducing the collision probability. As shown in Fig. 11, a potential of 20,000 volts was achieved with a 60 amp magnetic field current. This is still well below the million volts discussed earlier, but as illustrated in the figure, the calculated curves and experimental data agree fairly well, at least in slope. On this basis Krieve argued that had larger field currents been available, higher voltages could be obtained. While the above may explain the failure to achieve voltages in devices using magnetic secondary suppression, it still does not explain the trouble with cells using grids. Unfortunately, the latter may be the most practical from the point of view of size and weight.

This completes a brief discussion of the FEC. Alpha and beta cells have also been built which exhibited certain advantages and disadvantages. One major point to note is that the beta emitters involve lower energy particles, so they can operate efficiently at relatively low voltages, this gets around some of the voltage problems. As noted earlier Linder and Christian achieved 30 per cent efficiency at 300,000 volts.

The only active project involving a high power cell is a Beta-Cell study in progress at NASA Lewis Laboratories where the cell is of interest for electric propulsion schemes.¹⁸ A number of companies produce low power cells, commonly called "nuclear batteries."¹⁹

B. Direct Collection Studies

The approach at the University of Illinois has generally been to select some specific aspect for detailed study rather than attempt to study the device as a whole. (One exception to this rule is the work with the Gamma-Electric-Cell.²⁰)

A number of varied calculations in the area of charged particle transport were made.²¹ An illustration of some results related to fission fragment transport is shown in Fig. 12. Here n_j , s_j , and m_j are parameters related to the energy-charge loss law used in calculating fragment slowing. Without explaining this figure in detail, the main point illustrated is that the

Fig. 10--The modified FEC capsule (after Krieve, Ref. 17).

Fig. 11--Open-circuit potential as a function of magnetic field current, modified FEC capsule (after Krieve, Ref. 17).

Fig. 12--Efficiency calculations for a thick fuel (after Miley, Ref. 22).

treatment of particle transport is an important problem in efficiency calculations. Not only is this true in direct collection devices but also in the ionization cells where one is interested in obtaining charged particles from the fuel layer and putting them into the fluid to produce ionization and/or excitation. These problems are also of interest to other areas such as space shielding. Thus this area is one where significant work can be done in universities, and where there are still a number of remaining problems. As an example, there is a problem in doing calculations leading to Fig. 12. The calculations were compared with the available experimental data. There are two ways of doing the experiment. One is a transmission type measurement of the energy loss of the charged particle as it passes through foils of various thicknesses. The other is a measurement of the energy spectrum of the fragment current leaving fuel layers of various thicknesses. The latter in effect is an integral measurement while the transmission measurement might be viewed as a differential measurement. There are major discrepancies between the best experiments using these two techniques. Some careful studies are needed to resolve this problem.

Another area of transport theory that has received attention is secondary electron production. The problem is illustrated schematically in Fig. 13. Consider a primary particle such as a fission fragment entering (or leaving) a surface. It produces a number of electrons, ranging from low to high energies, and one is interested in their formation and then their transport or diffusion to the surface. The higher energy electrons (δ -rays) may produce further secondaries, and these must be considered also. Then there is the interesting question of the barrier for escape across the surface.

Most of the work to date in this area has been for low-energy incident particles. Fission fragments, alpha particles, etc. with their high energy and relatively high charge represent an open field for research studies.

Some feeling for the situation can be joined by reference to Table III. Several recent calculations and measurements of low energy (< 50 eV) secondary electron yields are given. Note that the calculation for the fragment yield is almost an order of magnitude different from that by Anno²³ although both use the Ionization-Diffusion Model. While the result is closer to experiment, there are sufficient uncertainties involved so that the agreement in this case might still be fortuitous. An older model for such calculations, the thermionic model, is also shown. It clearly gives erroneous results for fragments, but because of their lower charge, alpha particle calculations turn out to give reasonable results.

In conclusion secondary electron production is a complex subject. At present the theory and even the measurements are "fuzzy" when high energy-high charge particles are involved. It too looks like a ripe area for university research.

C. Nuclear Radiation for Laser Pumping

As indicated earlier, this has been selected as an example of conversion of nuclear radiation via ionization and excitation, i.e., following the earlier analogy by many small barriers rather than one large one.

Fig. 13--Formation of secondary electrons (SE) and δ -rays by a high energy charged particle. (The shaded area indicates the escape depth or "zone" for secondary electrons.)

TABLE III

Calculated and Experimental Yields

Target	Particle	Yield (Normal Incidence)		
		Measured ⁽¹⁾ (Ref. 21)	I-D Model	Thermionic Model ⁽⁴⁾
Au	Alpha (~ 3 MeV)	5.5	5.7 ⁽²⁾ 8.8 ⁽³⁾	5.8
UO ₂	Fission Fragment	118	800 ⁽²⁾ 196 ⁽³⁾	7.7 x 10 ⁴

- (1) Corrected for geometry, Δ^f obtained from Δ^α using $R_\Delta = 21.3$ (p. 58, Ref. 23).
 (2) Values reported by Anno based on $\left\langle \frac{dE}{dx} \right\rangle = 2.6 \times 10^{-10}$ and 3.4×10^{-9} cal/cm for Au and UO₂ respectively and $I = 690$ eV for Au.
 (3) Calculations by Miley, Ref. 21.
 (4) Calculations by Anno using the following constants

	Au	UO ₂
C_v (cal/gm ^o c)	0.32	0.078
K (cal/sec cm ^o c)	0.71	0.0063
(gm/cm ³)	19.3	10.1
$\left\langle \frac{dE}{dx} \right\rangle$ cal/gm	2.6×10^{-10}	3.4×10^{-9}

The laser may seem to be an odd example at first thought since it does not produce electrical power. Still by earlier ground rules, it is a perfectly valid subject. Basically, the scheme would convert nuclear radiation to either optical or infrared radiation.

The possibility was first considered in connection with military uses, but these reports have not been obtained. One of the members of the Program Committee for this meeting, Dr. Herwig, while at United Aircraft produced in 1964 what is a classic report on nuclear pumping of a gas laser.²⁴ While there were some vague ideas about the problem prior to then, certainly this report "cemented" our thoughts. A basic point that came to light in our studies was that the experiment should be feasible with neutron fluxes such as those achieved during a TRIGA reactor pulse. It was decided in 1965 in co-operation with Professor Verdeyen of our Electrical Engineering Department to begin an experimental program using the Illinois TRIGA. Several graduate students have been involved and a Ph.D. thesis is in progress. No concrete results have been obtained to date. We have in general attempted to identify specific problems which may be bottlenecks and concentrate on them. (Note added in proof Preliminary results from the in-core helium-neon laser study were reported at a conference in November, 1967.²⁵ Successful operation of a low-voltage-arc laser, described later, was first achieved during September, 1967.²⁶)

Figure 14 illustrates how a laser works. Consider an idealized two-energy state system. Radiation coming in "kicks" an electron from State 1 to State 2 corresponding to the conventional absorption process. Neglecting outside influences, this electron will eventually decay back to the lower level giving up a photon by spontaneous emission. However, stimulated emission can also occur. One can bring in radiation with an energy exactly matched with these two levels and get out two photons. Thus the possibility of building a "critical system" becomes evident. The term criticality is used because there is a direct analogy with nuclear reactor criticality. Stimulated emission is analogous to neutron multiplication. A reflector helps the reactor go critical, and in the case of the laser, mirrors are used for this purpose. In the jargon of the field, one places the laser in a resonant cavity.

The University of Illinois has been concentrating on direct methods for coupling lasers and reactors. Gas lasers, e.g., typical helium-neon lasers, are much less than 1 per cent efficient based on an overall energy-in, energy-out balance. (The CO₂ laser is a notable exception where efficiencies well over 10 per cent are possible, but its output is in the 10 μ range which restricts its application.) If one concedes that reactors are going to be in space, and also lasers are going to be in space (for communications, etc.) it seems it would be important to couple them in the most efficient, reliable manner possible.

Figure 15 illustrates two separate paths. The conventional laser cycle would use heat from the reactor in connection with a conventional stream or liquid metal-electrical generator cycle, and then a conditioning step to produce high voltage (kV range) or rf which would actually pump (supply energy to) the laser.

A more simple cycle, also shown in Fig. 15, involves heat generation, but the heat would then be used to drive a thermionic discharge. Such a discharge produces a nice plasma if it is designed properly, and there is a good chance this plasma can lase. There are some horrendous problems to actually building an in-core device. If people working on in-core thermionic devices can make these work, the extension to an in-core low-voltage-arc laser is not too great. Along this line it might be noted that cesium has lased (but under optical pumping), so there remains a possibility of a combined thermionic energy conversion--laser cycle.

The last cycle shown in Fig. 15 is even more direct. Here fission fragments, alphas or other charged particles produced by neutron interactions are used to ionize and excite the gas. In the conventional cycle electrons accelerated by an imposed voltage or rf signal produce the ionization and excitation. Thus, the fission fragments or alphas simply replace these electrons. This may not seem like much of a step, but there are some subtle differences. In order to appreciate this, let us first review the helium-neon energy level diagram illustrated in Fig. 16.

The lasing levels (three are possible 3.39, 1.15 and 0.63 μ) are actually located in neon. Helium is used as a convenient means of transferring energy to the desired neon levels. Helium is excited by collisions into 2¹S or 2³S states which are metastable. Subsequent collisions transfer this energy to neon as illustrated, so that lasing may occur between one or more of the

$$I = I_0 e^{\gamma x}$$

WHERE: $\gamma \propto (n_2 - n_1)$

$\therefore \gamma < 0 ; n_2 < n_1$ (ATTENUATION)

$\gamma > 0 ; n_2 > n_1$ (AMPLIFICATION)

$n_2 > n_1 \equiv$ POPULATION INVERSION

Fig. 14--The concepts of stimulated emission and population inversion.

Fig. 15--Laser pumping cycles.

Fig. 16--The helium-neon gas

- Key 1 Neon emits at desired frequencies, i.e., 1.15 or 3.39 μ (I.R.), or 0.63 μ (visible).
- Key 2 Method to invert the population is available helium collisions plus relatively long lifetimes of neon states.
- Key 3 Resonant cavity.

neon level pairs. Note that the neon then has to pass through a metastable state in order to return to the ground state. A dominate mechanism for this is a wall collision. This is significant because it requires that the neon migrate to the wall and in essence imposes a pressure-diameter restriction (pd ~ 3-5 Torr-mm for optimum performance).

Now consider substitution of alpha particles, possibly from a neutron-boron reaction, for the electrons involved in the excitation of helium. Some calculations are summarized in Table IV.

TABLE IV

Helium Metastable Production by Alphas

Pure He 43 eV/Ion pair
0.1% Ar 30 eV/Ion pair

Penning Effect $\text{He} + \text{Ar} \rightarrow \text{He} + \text{Ar}^+ + \text{e}^-$

~ 2 ion pair/86 eV in $\text{He} \xrightarrow{\alpha} \text{He}^+ + \text{e}^-$ ~ 3 pair/90 eV in $\text{Ar}^+ \text{He}$ conclusion
~ 1 He /90 eV

Summary of Calculations by Herwig²⁴

Calculated Upper Laser Density, Atoms/cc

$\Phi_{\text{th}} = 10^{13}$ (for 10 Torr)			Threshold, Ne	
³ He	Ne	6.7×10^4	.63	3.1×10^8
¹⁰ B		2.1×10^6	1.15	5.1×10^7
²³⁵ U		2.7×10^8	3.39	2×10^6
²¹⁰ Po		3.4×10^7		

The first question is how many excited states are created in helium per alpha particle? This is a key number, but it is somewhat hard to pin down. We are currently attempting a theoretical calculation, but an experimental observation is indicated here. Consider pure helium. It requires about 43 eV to produce an ion pair. If some argon is added, only ~ 30 eV are required per ion pair. The difference can be attributed to the Penning effect, i.e., ionization of argon by collision with metastable helium atoms. Thus about 86 eV would produce 2 ion pairs in pure helium whereas 90 eV would produce 3 ion pairs in the Penning mixture. This implies that roughly 1 metastable helium is produced per 90 eV lost by the alpha (This argument was first noted in the report by Herwig.)

Once this number is known, the threshold equation for lasing can be evaluated, and some typical results are shown in the same table. Also some estimates of the excited state densities are shown assuming that the charged particles are produced by neutron reactions with helium-3, boron-10 or uranium-235. The values shown assume a neutron flux of 10^{13} neutrons/cm²-sec, a 10 Torr gas pressure, an optimum laser tube diameter and an optimum coating thickness of uranium-235 or boron-10. Consider alphas from boron-10. The values in the "boxes" in the table indicate that a flux of 10^{13} should be sufficient to make a helium-neon laser "go" on the 3.39 μ line (near infrared). Since this assumes an optimum geometry, etc., a higher flux may be required for an actual "as built" laser. Incidentally, the threshold for a laser is roughly inversely proportional to the cube of the wave length, so as one goes from infrared to optical wave lengths, the job becomes more difficult.

These calculations are encouraging because neutron fluxes of over 10^{16} during a pulse with the TRIGA reactor can be obtained. This leaves an appropriate safety margin, so the experiment appears feasible. The characteristic width of the reactor pulse is 10-30 milliseconds. This is long relative to lifetimes involved in the lasing cycle, hence the laser would be in a quasi-equilibrium state during the pulse.

Figure 17 shows a schematic diagram of an experimental in-core laser in use at Illinois. It is exceedingly simple, in essence a normal laser tube coated with boron. Integral mirror-windows are used to obtain a rugged system which will hold alignment when lowered into the reactor core. The electrical discharge section is used to align the system when the laser is first placed in the reactor. The tube shown is designed to operate on the 3.39 μ line.

There are a number of developmental problems. First, the entire assembly must fit into a fuel element position so the diameter is limited to something under 1-1/4 inches. Secondly the materials of construction, mirrors, etc. must either be radiation resistant or located at a safe distance from the core. Finally, since the reactor core is some 17 feet below water, and since our detector has an active area of about 1 mm², and since bent tubes and mirrors are needed to prevent gamma and neutron streaming, alignment problems are indeed severe. Many hours have been spent at the top of the reactor moving the detector about in vain searching for the infrared laser beam.

While these experiments represent a direct attack on the problem, as noted earlier we are simultaneously attempting to dissect the problem into smaller studies. I want to mention one such problem to illustrate this. It has to do with the electron slowing-down spectrum in the gas, which is illustrated schematically in Fig. 18. This is a fundamental problem in radiation-produced plasmas, but there is not a complete solution for it in the literature. The problem is similar to the calculation of neutron slowing down spectrum in a thermal reactor. In the present case alpha particles produce electrons in the gas having an average energy of 25 eV, and they slow down. The resulting energy spectrum will be considerably different from that produced in rf or dc discharges. In the latter electric fields accelerate the electrons between collisions. As a result electron "temperatures" are typically 10,000 degrees or more. In the absence of fields, the plasma produced by alphas may have an

Fig. 17--Sketch of an experimental in-core laser tube (after Guyot, et. al.,
Ref. 25).

Fig. 18--Postulated electron energy distribution in helium.

effective temperature closer to the actual gas temperature.²⁷ The relative rates of electron production and recombination-leakage will in effect determine the detailed thermal spectrum. Of course, the concept of "temperature" applies only to the Maxwellian portion of Fig. 18. As illustrated, the slowing down region must also be accounted for.

The electron energy spectrum is important in the laser calculation. Some of the excitation may be due to direct alpha excitation, but a significant portion will come from the high energy electrons produced by the alphas. The importance of this spectrum can be illustrated by the following argument. The hot electrons in the standard electrical discharge tend to excite neon atoms via collisions from the $1S_2$ neon level into the lower lasing levels (recall Fig. 16). This stresses the importance of depopulating the $1S_2$ level as quickly as possible by diffusion to the wall, and this is no doubt an important factor in the pressure-diameter restriction noted earlier. Now, if the alpha-excited plasma has a lower electron temperature, the back excitation from the $1S_2$ level may not be so important, and it may be possible to relax the pressure-diameter requirement. This would in turn lead to larger power outputs, illustrating one speculative advantage of nuclear radiation pumping. Some other possible advantages, most of which were noted first by Herwig, are listed in Table V.

TABLE V

Speculative Advantages of Nuclear - Pumping

1. Increased Power via
 - a. improved coupling vs. rf
 - b. higher pressure
 - c. lower electron temperature
2. Improved Control
 - a. E & B fields independent of pumping
3. Higher Frequencies
 - a. excited ions
4. Compact, Light Weight, Reliable

The University of Illinois has a similar effort involving the low-voltage arc laser. The experimental laser is illustrated in Fig. 19. It essentially consists of a triode design using plane-parallel electrodes; namely, a heated oxide coated cathode, a plasma stabilizing grid and a movable anode. An electrically heated cathode is used, but the thought is that this could be replaced by heat from a fuel element in an in-core design. The feasibility of a large area cathode such as this along a low-voltage arc, in contrast to the typical small area, high-voltage discharge design, has not been demonstrated experimentally to date. Thus far there are plasma stability problems, but it is hoped that this difficulty can be overcome by some modification of the operating conditions. The initial studies involve helium-neon. This system is understood best so it is preferable for feasibility studies.

Fig. 19--An experimental low-voltage arc laser tube (after Herceg and Miley, Ref. 26).

The possibility of a low-voltage arc cycle has also been considered by F. M. Johnson.²⁸ While he did not carry out experiments, he does point out some interesting reasons for considering this approach, and these are summarized in Table VI.

TABLE VI

Advantages of Thermionic Laser

- high efficiency
- topping cycle concept
- possibility of alkali-metal lasing

also, it may

- aid understanding of low voltage arc mode (ignited mode)
- aid understanding of pressure-diameter relations

Problems are

- mechanical - similar to those for thermionic devices
 - designed for in-core or indirect heating
- sensitivity to impurities in the plasma
- plasma stability

II. Conclusions

The things about which I have talked are quite specialized. I have, as have most nuclear engineers, mainly earned my living to date from our "bread and butter" product--the nuclear reactor, and suspect we will continue to do so for a number of years. I do not think we should all immediately jump into a program like the one described here. At the same time, a certain fraction of the total effort should be put into areas such as this, which tend to diversify the field as a whole.

DISCUSSION

QUESTION: There is currently an unclassified, theoretical paper from Battelle Northwest on pumping the laser. Are you familiar with it?

ANSWER: We have talked some with the Battelle staff about this problem. I am not familiar with the details of the report you mention. (Note added in proof: The report under discussion appears to be BNWL-452 by B. H. Duane. Rather than dealing with laser pumping it suggests methods for using a laser output to accelerate electrons.)

I would like to reiterate that theoretical work has come fairly easily in this area. A number of people, including ourselves, have shown it works on paper. However, there have been no experimental verifications, and there have been many frustrations. On this connection I should point out that studies using a solid state laser have encountered a radiation "cut-off" effect (see Ref. 9). We ruled this approach out some time ago because of possible radiation damage. This was a fortunate decision. The "cut-off" effect is not

entirely understood; but essentially, an operating solid-state laser placed in a strong gamma radiation field will be quenched. While there may be a way around this problem, it is not going to be easy. Luckily, the same effect would not be expected in a gas laser. This has to do with the fact that the solid has a fixed structure whereas gas molecules are of course free to move. The same situation occurs in radiation damage; the gas system has a big advantage.

QUESTION: Do you mean that just a chemical binding system makes a difference?

ANSWER: That or the lattice structure. The energy levels that are typically involved in a solid-state laser come about because of the bond structure. The effect of nuclear radiation on this structure is different from the effect it has on the atomic levels in the gas.

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RADIATION EFFECTS IN SOLIDS

D. K. Holmes
Oak Ridge National Laboratory

I want to discuss the general subject of radiation damage¹--the current status of research, where we stand and where the problems are.

The investigation of the effects of radiation on the properties of solids is motivated from two quite different directions. The technological needs require that we have some understanding of how a given solid material will behave in a radiation field with respect to those properties which are most important for such things as electronics, reactor structural components, etc.

Conversely, we would like to study the fundamental properties of solids and particularly defects in solids. This may be approached from the radiation damage viewpoint by introducing controlled amounts of defects in solids by the use of radiation. In the early stages of a field of investigation such as this, the two different goals are not necessarily compatible. The person who wants to find out how a transistor will behave in a high flux of gamma rays at high temperatures will not necessarily learn very much from the studies of another person who is investigating the annealing of defects produced in germanium at liquid helium temperature. The use of materials in technology requires the knowledge of how these materials will behave in very complex situations, whereas the very goal of the fundamental studies is to simplify the system as much as possible in order that the very primary properties may be studied.

These two lines of procedure will eventually converge. However, this may take many years of study, and one of my purposes is to indicate where we stand in the process of convergence of these two methods at the present time. My own interests in radiation damage are primarily from the fundamental or theoretical side of the study of radiation damage in solids. Consequently, you may expect to note in this talk a bias toward the fundamental scientific investigation of solids and of defects in solids.

One of the roles of the Solid State Division at the Oak Ridge National Laboratory is to support fundamental studies of radiation damage in solids. Thus, you may also expect a bias in the direction of recent work of this kind performed in the Solid State Division. The larger effort of ORNL is directed toward those large-scale projects which demand immediate answers to materials questions.

The most fundamental scientific investigation of radiation damage can be pursued for its own sake, and its greatest direct contribution to the technological demands is still in the future. However, the indirect help given to project work from the basic side in terms of day-to-day advice and guidance based on ever advancing fundamental knowledge is invaluable.

I. Technological Effects

Looking first at the technological side, consider the data shown in Fig. 1. Graphite has been a material of interest in reactor development from the very beginning. Recently there has been renewed interest in the use of graphite, particularly as a structural material and a moderator, at high temperatures. Figure 1 shows the dimensional changes which have been observed in graphite by irradiating at high temperatures.² At high doses there are some high percentage changes in the dimensions of the sample.

The first change noted at early, low level radiations with fast neutrons is a decrease, shrinkage. This then reverses and is followed by a large expansion which seems to be continuing and even increasing in rate at the highest doses that have been observed.

The questions from the technological viewpoint are: How far does this go? Do we expect these dimensional changes to continue as we go to even higher doses? These are extremely important questions, because if the growth continues at the rate shown here, an obvious problem is created in the design of economically interesting thermal reactors.

The question which may be asked of the basic scientist is whether this expansion should be expected to continue to dose levels difficult to achieve in present reactors and beyond the highest nvt shown here. Consider what the expert in radiation damage is faced with if he should try to do a really precise calculation of the anticipated expansion of graphite.

II. Fundamental Studies

From the viewpoint of basic solid state physics, graphite is a complicated system. Graphite, when taken in a single crystal form, is a rather unfortunate substance in many ways. It has a highly anisotropic behavior showing layer planes which are close packed--the atoms are close packed in the planes, but the planes themselves are not close packed on top of each other. The observed irradiation behavior has always been very complicated. It appears that interstitial atoms tend to collect in layers between the layer planes, forming new layers, which is presumably what causes the c-axis expansion.

Furthermore, graphite as used in reactors is not a simple material. It consists of small crystallites with a certain amount of bonding material in between which has somewhat unknown properties. Thus, at the present time we would be in a bad position in trying to predict what will happen, from the basic solid physics viewpoint. The situation is so bad that we cannot predict the behavior under much simpler conditions than this. At the present time many technological needs are very far beyond anything that is available from the viewpoint of basic science.

One of the questions which must arise is why this situation should persist when fundamental radiation studies have now been under way for nearly 20 years. The answer is that the whole situation is much more complicated than anyone had ever realized in the early days of these investigations. The major point of this discussion will be to show where these complications have arisen, and where the greatest difficulties lie with respect to further progress in the fundamental studies.

A. Stages in the Radiation Damage Process

In order to provide a framework for the discussion of the difficulties in the fundamental studies of radiation damage, Fig. 2 shows the various stages in the radiation damage process. This is schematically illustrative of a "before and after" irradiation experiment.

The first frame shows the measurement of electrical resistivity of a good metal crystal. The second frame shows that the incident radiation has given some energy to the primary atom. In the third frame this primary atom has started to cause a cascade of moving atoms in the solid, and in the fourth frame the cascade has ended. We see various interstitials and vacancies scattered throughout the crystal, although not necessarily stable as they stand. In the fifth frame we have allowed annealing--either raised the temperature or possibly allowed annealing, if this could occur, at the temperature of irradiation.

In any case some defects annihilate and some interstitials may even leak out of the sample; when everything finally comes to rest we might even get some clustering of the various defects.

Finally, in the sixth frame when the electrical resistivity is re-measured, there is found extra electron scattering by the defects and thus a radiation induced increase in resistivity.

The transfer of energy from the incident radiation to an atom of the solid, or in some cases an electron of the solid which eventually ends up on an atom of the solid, is a point which is fairly well understood.

All the various types of radiation--fast neutrons, thermal neutrons, gamma rays, electrons, heavy charged particles, fission fragments--have been studied in considerable detail, and their method of transferring energy to the atoms of the solid is a known process in nuclear or atomic physics and can be calculated fairly well. Some of the details of the transfer of energy from radiation to atoms of the solid could use some further study. For example, in irradiating a specimen in a nuclear reactor--subjecting it to fast neutrons--it is true that in many instances the actual dependence of the neutron flux on energy is not sufficiently well known for truly precise calculations.

In reactor irradiations when the neutrons suffer elastic scattering from nuclei, the energy given to the atom including the nucleus depends upon the angular distribution of the scattering collision. The angular distribution of scattered neutrons is not well known for all energies and all nuclei and is still a rather large field of study in nuclear physics.

Fig. 1--The volume change (in per cent) of reactor grade graphite as a function of fast neutron dose at temperatures of the order of 600° C. (This curve is schematic, representing the general trend of many experimental results, e.g., Ref. 2.)

Fig. 2--Effect of irradiation on the electrical resistivity of a metal, in a "before and after" experiment.

Sometimes recoil atoms from nuclear reactions are used to induce damage. In the (n, γ) reaction several gamma rays are given out from the nucleus, so that the nucleus and its associated electrons recoil. The angular correlation of the gamma rays coming from the decay determines how much recoil energy is given to the nucleus, and the angular correlation at the present time is poorly known in such reactions.

It is very important to establish the amount and distribution of energy given to atoms in a solid under a given radiation field. If the total amount of energy dumped into the solid in terms of primary atom kinetic energy per unit time per unit volume is known, there is already a measure of the damage because this energy has to be dissipated in some way.

The energy of the primary atoms is not a very precise measure nor a very complete measure of the ultimate damage, but this point has been fairly well investigated and is not at the present time a limiting factor in our study of radiation damage. However, the next stage in the radiation damage process, which is the formation of the cascade of moving atoms caused by the primary atom as it loses energy moving through the solid, is one of the most poorly known processes at the present time.

As the primary is moving through the solid, if it has reasonably high energy, it loses energy in two different ways. Figure 3 gives a schematic representation of energy loss per unit distance of travel of a moving atom.³ The moving atom starts at high energies. As it loses energy it moves down toward lower energies, and there is different behavior for the two different forms of loss.

The two types of energy loss are the ionization losses to the electrons of the solid--either those which in the case of the metal are electrons of the whole solid or electrons bound to given atoms--and the loss by elastic collisions in which one atom scatters from another atom and neither changes its internal electronic state. At high energies the ionization losses dominate. As the energy decreases, these two curves cross; and finally at lower energies the elastic losses are dominant. The energy which is lost to the electrons of the solid through ionization or excitation for metals and some other solids will not produce any permanent residual damage, so if an atom starts with sufficiently high energy it will go through a considerable distance in the solid without producing many defects. As its energy falls below the crossing point, it will have a number of elastic collisions, knock many atoms loose from their lattice sites and create considerable damage. However, since its energy is low, the distance of travel is relatively small; and thus the cascade of struck atoms occupies a fairly small volume in the solid.

Neither one of these curves is well known. The ionization curve which is sketched in with dashed lines is totally a speculation and certainly could not be expected to hold except in a very average way. There must be all sorts of "structure" in this curve for any given solid.

Referring again to Fig. 2, Frame 3 shows the cascade at a point when the energy has been degraded to such an extent that each of the moving atoms has energy in the range in which the elastic losses are dominant. Suppose that atom A moving through the lattice with a relatively low energy (50 eV)

Fig. 3--Schematic representation of the energy loss per unit distance $\left(-\frac{dE}{dx}\right)$ of a moving ion (after Lindhard and Thomsen³).

strikes an atom of the lattice and shares its energy in some way with that atom, then does the struck atom move on and become a displaced atom, or does it remain at its lattice site to which it has a certain effective binding energy? If the struck atom does move on, does it do so in such a way that there are now two displaced and moving atoms?

The answers to these questions depend upon the nature of the binding of atoms in a solid and would be difficult to obtain at present. It is clear that the answers to the questions depend upon the lattice direction in which the momentum is given to the struck atom. If the struck atom is moving directly toward one of its neighboring atoms in the lattice, it is perhaps less likely to escape from its lattice site than if it is aimed toward one of the spaces between two of its neighboring atoms. The consequences of the orderly arrangement of atoms in the lattice can be very difficult to calculate.

Frame 4 of Fig. 2 pictures a situation in which all of the original kinetic energy of the primary atom has been dissipated, and all of these pictured atoms have come to rest leaving certain vacant lattice sites. This is a nebulous state and might be difficult to define, because it is not true that all of the damage is stable in the geometrical configuration which is indicated. A given vacancy-interstitial pair could be mechanically unstable and might collapse, thus annihilating both defects.

Again, this is obviously a solid state or defect-solid state question. How far apart do a vacancy and an interstitial have to be in a lattice in order to be stable at essentially the absolute zero temperature (neglecting zero point vibrations)?

Another question involved at this same point in the developing radiation damage process is what has happened to all of the energy of the primary? There must have been many cases in the development of the cascade in which a moving atom struck a lattice atom and did not transfer to it enough energy to tear it away from its lattice site but did transfer some energy to it. Much of this has happened in a very short time. Because of the high energies involved, the actual time between the transfer of the energy of the incident radiation to the primary in Step 2 and the cascade's end in Step 4 is very short on a laboratory scale.

The whole process might require 10^{-14} or 10^{-13} seconds--the time for one lattice vibration in the solid. The implication is that most of the energy transferred in subthreshold collisions to the atoms is still around as kinetic energy of the atoms but has not resulted in displacements. What happened to this energy? Calculations are very difficult in this area but tend to show that each agitated atom will share its energy with its neighboring atom, so that the energy moves away, and it may be that it even moves away in an organized fashion like a shock wave

To what extent can this extra energy, which appeared at the same time as the defects, act in the same way that high temperature does and promote the annealing or annihilation of these defects? This question is difficult to answer, because the region is relatively small (1,000 angstroms across), and it is extremely doubtful that equilibrium has been established as the energy moves out of this region. Therefore, it is improper to talk about an effective

temperature of the region; or at least it is difficult to calculate on the basis of our present knowledge.

In going from Frame 4 to Frame 5 in the radiation damage process illustrated in Fig. 2, the motion of at least one type of defect has been allowed. This is a diffusive type motion involving thermally activated motion of the defects, which results in annihilation of vacancies and interstitials to make a perfect lattice or in clustering of vacancies and clustering of interstitials or both. The study of this type of motion and the subsequent events is a part of the general study of defects in solids. Radiation damage studies may provide information about the distribution of these basic defects, which could not be obtained in a solid in any other way. However, it must be admitted that in any case, theoretically and experimentally, our knowledge of the defect properties--the motion properties of defects--is poor.

Frame 6 of Fig. 2, which is the measurement of the original property after the irradiation has been performed, is the most complicated part of the entire process from the viewpoint of theoretical and experimental knowledge. So many different properties of different materials can be used that it is almost impossible to talk about the situation in general. The effect of radiation on some properties might be easily predicted or the change in the property interpreted in terms of the basic underlying defects, whereas for other properties it might be difficult to correlate the basic atomic and electronic situation with the bulk measurement.

In the case that has been illustrated here, the electrical resistivity of a good metal, it is actually rather difficult to perform reliable calculations of the scattering caused by a point defect, which results from the deviation of the lattice from true perfection. Various calculations of this scattering in the past have varied by a factor of 10.

It is clear that the parts least understood in this entire radiation damage process are the basic physics of atomic collisions, the basic physics of the motion of defects in solids and the basic physics of the effect of defects on solid state properties. All these are important to the understanding of the solid state in general, and to the extent that radiation damage studies can contribute to the understanding of these basic processes, they are worthwhile as contributing to the basic physics of matter.

B. Radiation Effects on Electrical Resistivity

Continuing the discussion of the basic studies of radiation damage designed toward understanding the properties of solids in general and defects in solids specifically, one may move from a broad survey of the radiation damage process to a discussion of a rather specific system, the resistivity introduced into a good metal at low temperatures by various types of radiation. The idea is to take as simple a system as possible, i. e., to remove many of the complexities so that with theory and experiment moving together it becomes possible to establish precisely some of the basic properties in order to extend understanding of radiation damage. The resistivity is chosen as a property. It seems certain that in a good metal, as long as the defects are kept reasonably well separated from each other in terms of the mean free path of an electron, the contribution to the resistivity is additive. The choice of low temperature

is to keep the defects from moving around during the irradiation so that as more and more defects are put in, the resistivity introduced by the radiation rises linearly with time.

If a comparison of various metals and various kinds of irradiation is considered, it should improve the knowledge of such items as the energy transfer to the lattice from the radiation field, the atom-atom collisions in a cascade, and the lattice binding energy effects. The latter point is an extremely essential one, and it has been summed up in the past in terms of the threshold concept. Imagine that atoms are bound to their various lattice sites. Then, when they are struck either by irradiation or another atom, they can be knocked loose from their lattice sites only if the energy transferred is equal to or greater than some threshold energy which is called E_d . If less, this transferred energy must be dissipated in the lattice as heat.

This does not accurately represent the situation as may be seen from the fact that there must be directional effects in the lattice. There are different directions in which it is easier or harder than the average to get away from the lattice site. However, this is an approximation for an overall crude estimate of the situation. This binding energy of the atom to its lattice site enters directly into all estimates of radiation effects.

It can best be investigated by using high-energy electrons as bombarding particles. If it is thought that the threshold energy is 25 eV, which is an average number, then on copper 0.5 MeV electrons can provide just enough energy to displace the atom from its lattice site. If the incident energy of electrons is varied from 0.3 MeV up to 0.7 MeV and the resultant damage rate is measured, then a good measure of the general effects of lattice binding can be obtained.

If electrons of energies just above that needed to displace an atom are used, the cascade is quite simple consisting perhaps only of one displaced atom a few lattice distances away from the vacant lattice site which it left behind. However, in dealing with heavier particles and greater energy transfer such as fast neutron irradiation, there is a much larger and more complex cascade; and it becomes much more difficult to evaluate what the final results will be in terms of the number of defects eventually produced and also in terms of the effect on the final property measured.

If contribution to the electrical resistivity of a number of widely separated vacancy-interstitial pairs is known, it still might be difficult to say what would be the contribution to the electrical resistivity of a large number of vacancy-interstitial pairs, all clumped into a local region of 1,000 angstroms across.

The results obtained to date indicate that the best calculations for a large cascade of fast neutrons are a factor of 2 to 5 off using the same basis for the calculation as that which gives essentially correct results for electron bombardment. The factor of 2 to 5 is such that the theory seems to estimate a larger irradiation effect than is observed.

A theoretical calculation of the electrical resistivity contribution of a vacancy-interstitial pair is not used. This is avoided by taking the ratio

of the resistance put in by neutron irradiation to that put in by electron irradiation, assuming in both cases that the electrical resistivity introduced is proportional to the total number of pairs introduced at the low temperature. The threshold energy is not really calculated; it cannot be calculated to the kind of accuracy that would be needed in order to make any predictions.

The true form of the displacement probability function is not the threshold form, but rather that the probability of an atom leaving its lattice site increases in some way as the energy transfer to the atom increases, and finally becomes essentially unity. Such displacement probability functions must be taken from experiments using electron irradiation; they cannot be calculated as yet.

With respect to the total cascade made by the initial primary atom, we are dependent upon the knowledge of atom-atom scattering collisions. It has been shown that the actual results for the total number of displaced atoms produced by a cascade are not very sensitive to the form of the atom-atom scattering law. Our knowledge is not sufficient to be sure of the estimate within 20 per cent, perhaps not even 50 per cent. In addition we have no way of treating the so-called ionization range for the bombardment with more energetic particles.

If the atom is created above a certain energy, it loses energy without causing any defects until it slows down to a certain energy. The cascade is then treated as if it were purely of elastic collisions with no ionization below that energy--a crude model.

What has impeded our progress with respect to theory? Can a better job be done of calculating cascades and displacement energies? At the present time there are high-speed computers to help. Why can't they give us some of the answers?

C. Computer Simulation of Radiation Effects

Figure 4 shows an attempt to simulate a radiation damage event in a lattice as calculated on a computer. This is a calculation of George Vineyard and coworkers⁴ at Brookhaven.

For this type of calculation, a solid is simulated in the computer with a certain number of atoms, a total of possibly 1,000 to 4,000 atoms. The atoms are held together by forces; i. e., there are no artificially simulated lattice sites; it is a collection of atoms held together in such a way that the position of each atom depends upon the positions of all the atoms around it and the consequent forces. In this lattice one atom is given a certain amount of energy, such as the one illustrated at Point A. It moves out, strikes other atoms, and all motions are calculated on a short time scale, using Newton's laws.

The size of the array of atoms that can be treated in this way is relatively small, and it is impossible to represent large-scale radiation damage events in this manner. The event shown represents perhaps 100 eV of energy given to the primary atom. It transfers energy to other atoms, which perform small oscillations around their lattice sites, but altogether possibly one other atom is permanently displaced from its lattice site.

Fig. 4--Computer simulation of the motions of atoms in a solid as a result of the production of a low energy primary at A (after Gibson, et. al.⁴).

The forces chosen to represent the binding of one atom to another, and therefore eventually form the binding energy of the atom to the solid, are not particularly realistic. They are perhaps the best ones that can be chosen at the present time, but they are not considered to be good enough. The solid is not really a cohesive whole, but must be held together by artificial external surface forces which are included in the hope of simulating the properties of the solid itself. Consequently, when calculating the energy needed to eject an atom from its lattice site and make a permanent displacement, there is no reliability at all upon the actual numerical values achieved.

The movement of atom A between its two neighboring atoms depends very sensitively upon the details of the interaction of these atoms at fairly long distances--halfway between the two atoms--and yet these are short distances as compared with normal atomic spacings. When two atoms are crowded into half the normal atomic spacing, the distances between atoms cannot be achieved experimentally by application of high pressures nor in shock waves.

Furthermore, all such forces which have been used at the present time involving essentially two-body forces cannot take into account the effects of the electrons in the solid, i. e., the electrons in the whole solid (a metal) or such things as the valence electrons in a nonmetal. These are many-body forces, and there is not any good way known to include them. The results of this "computer experiment," while somewhat interesting as a guideline, are not really reliable with respect to overall results.

Figure 5 shows a much higher energy calculation, in which the cascade has been calculated in beryllium oxide for a 5 keV incident iodine atom.⁵ While a larger energy and a larger volume cascade is considered, this type of computer calculation does not have the atoms tied together with atom-atom forces. The atoms are simply tied to their lattice sites, which form an artificial lattice in space with a certain energy required to displace them. There is no attempt in this model to calculate the details of displacement type events as in the Vineyard model. There is no possibility of calculating displacement energies; they are put in artificially. The model is too crude to give anything but a suggestion as to how a cascade might appear, i. e., a large cascade containing many atoms. Even with these limitations in both calculations, the chosen events are such that the motion is always confined to one plane. This further simplifies the calculation and allows long-range events but doesn't simulate the reality of a three-dimensional cascade.

In the hope of doing somewhat better with respect to the basic properties in radiation damage, we turned a few years ago to considering the ranges of incident atoms in solids. This was stimulated by some experiments performed by several different groups of investigators, particularly those of Davies and his coworkers⁶ at Chalk River. In these experiments they allowed energetic ions (argon or xenon in the range of 10-100 keV) to be incident on the surface of a metal such as aluminum. These ions were radioactive, and the experiment was designed so that the argon atoms remained at the spot at which they came to rest in the metal. After the irradiation was over, it was possible to strip off small layers of about 20 angstroms thickness, and by a counting technique find the amount of radioactive incident atoms deposited in the layer. Thus, the distribution of atoms inside the solid could be determined.

Fig. 5--Calculated displacement cascade in a solid resulting from the passage of a high energy bombarding particle (after Beeler and Besco⁵).

This experiment is somewhat simpler, and perhaps represents the possibility of doing something with calculations. It certainly tests our knowledge of atom-atom scattering collisions. Although it doesn't test directly the question of energy transfer and a number of displacements in a cascade, it does bear directly upon the spatial extent of a cascade.

The subsequent displacements which are made by the initial atom are not observable. It is only the incident atoms that are observable and which might be compared in radiation damage with the primary atom.

As a first approximation, inside a material being subjected to bombardment the behavior of the primary is representative of the behavior of the whole cascade. If a copper primary is created in copper, then at the first scattering collision between a primary atom and a copper atom at a lattice point, the two atoms are really essentially indistinguishable, so that one cannot tell which atom is the primary. If a computer is allowed to follow a primary, making essentially random collisions, one can follow many histories and eventually find the position of an average primary and thus simulate the experiment of Davies and coworkers.

The geometry can be followed by a computer, which is designed for this purpose. The essential piece of physical information that has to be introduced into the calculation is the interatomic potential. Figure 6 shows two of the potentials which can be assumed.⁷ These are not what could be called true interatomic potentials, but simply mathematical forms which can be easily used, and for which there is justification at least in some ranges. This figure shows only the repulsive part of the potential, which is most important for this type of calculation. Both the Thomas-Fermi potential and the Bohr potential can be put into the calculation, and the results compared with the experiments. Both of them, especially the Bohr potential, have a certain amount of adjustability in terms of the assumed screening distance.

D. Comparison of Calculated and Experimental Data

Figure 7 shows a comparison of the theoretical results of such a calculation with some of the experimental data of Davies and coworkers. The median penetration is plotted as a function of the energy of the incident ion.

The calculated data are shown as solid curves. One of them, based on the Bohr potential, gives too much penetration at every energy with the particular Bohr screening length chosen. Conversely, the Thomas-Fermi potential, using the standard Thomas-Fermi screening length, gives too little penetration. The answer probably lies somewhere between the two, but it appears that there is a reasonable type of an interatomic potential or one can be guessed for such experiments.

If the scale values shown in the figure are somewhat relevant to the situation of radiation damage, then energies of 50 keV indicate penetrations of 300 angstroms, which might be representative of average cascade dimensions.

This work was extended greatly and compared with many other experimental results. It was felt that we were rather close to understanding

Fig. 6--Theoretical interatomic potentials as functions of the interatomic spacing (after Oen, et.al.⁷).

Median Penetration Depths of Xe^{133} Ions in Aluminum.

Fig. 7--Median penetration depths of xenon-133 ions in aluminum (after Oen, et. al. 7, 8).

something about the interatomic potentials needed in these relatively high energy ranges, recognizing that this was not the full answer with respect to calculating cascades, because in the cascade process the most critical events happen at the low energies. As Leibfried⁹ has shown, things are not too sensitive to the form of the potential at the higher energies. This work was useful, and it led to something even more interesting.

E. Channeling Phenomena

Both the theory and the experiment uncovered the phenomenon of channeling. In both the theory and experiment it was found that a small percent of all the incident ions moved extremely long distances in the solid--so long that both the theorists at ORNL¹⁰ and the experimentalists at Chalk River, independently, felt that probably something was wrong with their respective calculations or experiments. However, time has shown that both were really in good condition, and that these special atoms actually moved through extremely great distances. As compared with a few hundred angstroms in Fig. 7, these special atoms might move thousands and thousands of angstroms. These atoms appeared to be moving down easy channels in the lattice.

Figure 8 shows a model of a sodium chloride type lattice, in which it can clearly be seen how the atoms, when looked at in this direction, seem to be arranged in rows. If a moving ion were to hit the lattice in such a way that it were moving between and parallel to the rows, it might easily move a long way without coming to rest.

Figure 9 shows schematically the way a channeling event might occur. The moving atom follows the path indicated between two rows of lattice atoms, making collisions with the atoms, but collisions in which the angular deflection is small. As it moves from one side of the channel to the other, it is directed back into the channel by a series of collisions with the atoms which comprise the walls of the channel.

In spite of the current interest in channeling it does not seem at present that channeling will play a strong direct role in the radiation damage process itself. When copper is placed in a fast neutron flux of a reactor, it seems unlikely that the channeling of copper atoms resulting from the original knock-on atoms would significantly affect the final results. Few atoms knocked from a lattice site would find themselves making such fortunate collisions in which they would happen to be going down a channeling direction. Channeling occurs much more readily when ions are incident from outside the solid onto the surface, if the correct direction is chosen, and thus channeling is expected to be much more important in, e.g., sputtering. However, the interesting part about channeling is that the study of channeling might lead to a deeper knowledge of the atom-atom scattering process and of the inelastic scattering processes, i.e., the ionization losses.

In studying the energy loss of ions shot through single-crystal or nearly single-crystal foils of gold, a group of investigators at ORNL¹¹ found that when the gold foil was so oriented that the incident beam was essentially along the close-packed (110) directions in the gold, which would lead to open channels. The energy loss was significantly less, thus indicating that the moving ions spent more time in regions which are half way between atoms,

Fig. 8--A model of the sodium chloride lattice.

Fig. 9--Channeling path of an atom channeled between two close-packed rows of lattice atoms. Points indicate sites of binary elastic collisions.

where the electron density is presumably lower, than passing near the nuclei of atoms where the electron density is higher.

The investigators also found that there was an interesting spectrum energy loss. There is found not just one group of ions which lost less energy than usual but a number of groups of ions which lost different amounts of energy, all less than usual.

This at first was puzzling, but when the apparently true interpretation appeared, as made by Hans Lutz, it proved to point toward a possible tool for further study. Figure 10 shows how the process is envisioned at the present time. The important point is that the film may be rotated with respect to the beam, but the source and the detector remain lined up. As the ion moves back and forth in the channel, it has to come out essentially in the direction in which it was incident in order to be detected. Thus another ion, which made a smaller number of waves in the channel, would simply not be detected, and the only ions which would be detected are the ones which made an integer or half-integer number of waves in the channel. The interesting fact is that if we go to an ion which makes one more wave than the one shown as detected in Fig. 10, its path will have to move closer to the channel walls, because the motion is anharmonic and the wavelength depends upon the amplitude. Consequently, the second group of particles will test a different part of the interatomic potential and also will test a different electron density for the ionization losses. If a theory of the movement of a channeled ion in a channel is set up, one can put into this theory, first an electron density across the channel, and second a scattering potential across the channel. By varying these, if all kinds of experiments can be simulated, one might be able to get an approximation as to what these functions should really look like. This general line of investigation is being pursued both theoretically and experimentally, not only at ORNL but many other places and shows great hope for the future.

F. Studies at Low Temperatures

Returning to the discussion of damage cascades, consider the structure of a cascade at the end of the process but before any thermally activated motion of the defects is allowed. Suppose that the cascade consists of vacancy-interstitial pairs, where in principle it would be possible to say which vacant lattice site every interstitial came from, and interstitials and vacancies are present in equal numbers.

These vacancy-interstitial pairs will have different separations. If an interstitial moved away from a vacant lattice site and only passed one single atom and came to rest at a next nearest neighbor or next nearest non-neighbor site, it might be supposed that this whole situation would just be mechanically unstable, and that one would get collapse of the pair to heal the lattice without any need for thermal activation. If more were known about the structure of a solid and the way it is held together, and especially the way the atoms behave when they are out of lattice sites and in interstitial positions, it might be possible to predict just when this will happen. At the present time, it cannot be done. A calculation such as the one shown in Fig. 4 by Vineyard can be set up to simulate the condition of an interstitial, to see which pairs are stable and how far away the interstitial has to be before it becomes un-

Fig. 10--Two possible trajectories of channeled atoms passing through a solid (after Datz, et.al.¹¹).

stable. It is difficult to rely on these calculations because the interatomic forces included are not realistic. However, certain things are qualitatively probable.

When the interstitial is far enough away from the vacancy so that it does not collapse at low temperature, it still may be within the region of influence of the vacancy. This may be contrasted with the situation when the pairs are separated by a sufficient distance that there is essentially no interaction between them. If the temperature is raised, the closer pairs will tend to collapse by the movement of the interstitial toward the vacancy, whereas the more widely spaced noninteracting pairs will find the motion of the interstitial essentially random with respect to the vacancy. The lattice details of such structures cannot be calculated accurately. There is speculation that there may be different geometrical arrangements of the pairs in different lattices because the lattice is not isotropic, and a vacancy and an interstitial separated by a certain distance in one lattice direction may not be equivalent to that pair separated off in another lattice direction.

Consider large cascades that are produced by fast neutron bombardment. It is estimated that the maximum defect density expected in such a cascade is approximately 5 per cent, i.e., 5 per cent of all lattice sites are either vacant or have an interstitial by them. In such cascades it might be expected that the vacancies would tend to be in the center with the interstitials toward the outside, since it is the interstitials that are moving; and they leave the vacancies behind. However, actual calculations of cascades have not shown this effect clearly. Eventually as cascades are put into a lattice, i.e., as the lattice is damaged, the cascades will begin to overlap. When this is done, interstitials will accidentally fall near vacancies from other cascades resulting in pair collapse, which will heal some of the damage.

Thus, in a heavily damaged lattice there should be radiation annealing, i.e., the rate of introduction of the property deviates from linear toward smaller property change per unit flux. The damage or the property introduced will eventually saturate at some very high defect density. However, such a defect density has not been reached in any good metal at low temperature.

G. Annealing of Radiation Damage

From the study of damage in a simple system, a metal at low temperature, correlating theory and experiment, has not been satisfactory. The worst is yet to come. Consider the process of annealing of radiation damage. Having finished the cascade and allowed collapse of all unstable pairs, the temperature is raised and one asks what happens when the defects are allowed to move. This quickly becomes a complicated situation. There are two different methods of annealing.

One is called isochronal and the other isothermal. In isothermal annealing the temperature of the solid is raised to some previously selected value and held there. The changes of a given property are observed as a function of time. Isochronal annealing is essentially a warm-up type annealing. However, it is not convenient, generally speaking, to raise the temperature continuously; but rather the method uses "pulse annealing", i.e., the tempera-

ture is raised to a previously chosen value, held at this temperature for a certain time and then lowered to the reference temperature at which the property measurement is performed. In the next pulse the temperature is raised slightly and held for the same time--thus the word isochronal--and again lowered to the reference temperature for measurement.

Figure 11 illustrates the annealing process. The upper part of Fig. 11 shows the annealing of a property put in originally by irradiation. The fraction of the property remaining at any temperature is plotted as a function of the rising temperature. It is shown as a continuous curve, but it would really be a set of discrete points using the isochronal process. A rather rapid change in the property is occurring for the temperature ranges T_I and T_V .

The lower curve in Fig. 11 shows the derivative of the upper curve. The experimental data are now usually reported as the derivative. By taking the derivative, the regions in which the principal processes are occurring show up as peaks, and the important temperature ranges are clearly emphasized.

Figure 11 also shows an idealized annealing pattern. Suppose that a population of vacancies and interstitials is uniformly distributed through the lattice at low temperature waiting for something to happen. The interstitials become mobile at a lower temperature than the vacancies, i.e., their activation energy for motion is lower. As the temperature is raised the interstitials start moving, and perhaps some of them annihilate vacancies; others wander to dislocations or impurity atoms. After all of the interstitials are relocated, there would be a reduction in the original property, such as indicated at T_I , which is the temperature of motion of the interstitials, corresponding to the number of vacancies and interstitials of the original population which have annihilated. This leaves some vacancies frozen in the lattice with interstitials at some trapping sites where they cannot affect the vacancies. If the temperature is raised even further, a temperature is reached at which vacancies become mobile, whereupon they move and either move to trapped interstitials or themselves move to dislocations or other sinks and are annihilated. Finally, all of the defect structure disappears from the lattice and the original unirradiated condition has been restored.

This simple picture is nice to contemplate but has not to my knowledge been seen in any system. Figure 12 shows the annealing pattern of a few different metals following irradiation with thermal neutrons, as observed by Ralph Coltman and coworkers¹² at the Solid State Division of ORNL. The radiation damage results from the recoil of the nucleus after the capture of a thermal neutron and subsequent emission of a gamma ray. The energy of the recoiling nucleus is in the region of 100 eV to 300 eV, which from the radiation damage viewpoint is considered to be fairly low energy.

These annealing curves are like the lower curve of Fig. 11 and show the derivative of the property change with temperature in an isochronal annealing. Again the peaks should correspond to interesting regions and interesting underlying processes. There is an extremely complicated structure in all cases with nothing resembling the particularly simple pattern that one would like to see. Copper shown in the lower left corner has one large

Fig. 12--Annealing spectra for various metals after thermal neutron irradiation (after Coltman, et. al. ¹²).

Fig. 11--Idealized annealing pattern for interstitials and vacancies. (a) warm-up annealing, (b) derivative of (a).

peak at a low temperature, it also has additional peaks on the sides. There seems to be some structure in between, and the annealing is not completed at 300° K.

Consider the structures of defects to be expected in solids, particularly the simple metal. Think first of a vacant lattice site which in many ways should be the most simple possible defect. However, for a vacant lattice site there must be a certain relaxation of the atoms around the vacant lattice site, so that the structure is more complicated than the most naive picture would give. When the vacancy moves from one site to another (actually when a lattice atom jumps from its lattice position into the vacant lattice site), the displacement pattern around the vacancy must also shift with it. One may ask the question, "What is the energy of motion of a vacancy, i.e., what is the energy increase at the saddle point as the configuration moves from one equilibrium site to the next equilibrium site?" Considerable information can be obtained from nonradiation damage experiments about vacancies, and calculations can be of some help. The region around a vacancy is not nearly as crowded as the region around an interstitial, and thus we are testing our knowledge of interatomic forces in a region where we can obtain some information from other sources.

It is possible to quench-in a certain population of vacancies into a metal, and by the act of quenching alone some information about the energy of formation can be obtained. By raising the temperature and watching the change in some property, some information about the energy of motion can be gathered. In almost all good metals, the self-annealing energy is known, which is the sum of the energy of motion and the energy of formation for vacancies.

One of the things that has become fairly clear over the years is that vacancies like to cluster into groups of more than one vacancy, and they have a certain binding energy to each other. The lowest order cluster is the divacancy. It is presently believed that there is a certain binding energy of a divacancy, and some information on this can be obtained from quenching experiments.

It is also believed that in good close-packed metals the divacancies themselves are reasonably mobile. Higher clusters of vacancies can also be formed, and they will also have a binding energy. Presumably each succeeding vacancy that is attached to a cluster is held with smaller and smaller energy. It is reasonable to suppose that most of the vacancy clusters above the divacancy are themselves immobile, except perhaps in some very rare cases.

H. Interstitials and Vacancies

The interstitial is a much more complicated story. Figure 13 shows some possible forms for the interstitial which have been suggested and even calculated to be stable.¹³ One would think, in a face-centered cubic metal, that the natural position for an interstitial atom would be at the body-centered position, and some calculations indicate that this is the right place in some metals.

Fig. 13--Some possible interstitial configurations in a face centered cubic lattice (a) 100-split interstitial (dumbbell) (b) body-centered interstitial (c) 110 crowdion interstitial (after Simmons, et.al.¹³).

Conversely, some relatively early calculations¹⁴ indicated that it would be better, as shown at the upper left of Fig. 13, that the interstitial in the body-centered position move over a bit pushing an atom out of the face of a neighboring cell into a position some distance into the cell, so that the two atoms are sharing the interstitial properties around a face-centered lattice site. This configuration is called the split interstitial or the dumbbell interstitial, and the exact separation of this dumbbell depends upon the model of interatomic potentials chosen. As indicated earlier, whether it is stable or not compared with the simple body-centered interstitial depends upon the interatomic potential.

The bottom of Fig. 13 shows a crowdion.¹⁵ In this configuration an interstitial is pushed into a close-packed line of atoms, which is in a (110) direction in a face-centered cubic lattice, then the whole assemblage of atoms relaxes somewhat along the close-packed line and according to some calculations would remain stable.

If these static configurations of interstitials seem complicated, consider what happens when the interstitial starts to move and the situation becomes much more complicated. The motion of the split interstitial is quite complex. A crowdion motion is in principle simple because it moves along a close-packed line, but the determination of its energy of motion is very difficult, and the calculations may not be reliable. However, there is a general principle here. In good close-packed metals, the indication is that it is harder to form an isolated interstitial than it is an isolated vacancy, and also, that an interstitial configuration moves more easily--with a lower activation energy for motion--than does a vacancy.

Some calculations have also shown that interstitials may tend to cluster. At first, this would not seem to be a very probable event. Two overcrowded configurations, such as two interstitials, should repel each other. However, if they come together in certain lattice directions, it seems at least possible and even probable, according to present-day thinking, that the two interstitials may be slightly attracted to each other, and furthermore, might move as a di-interstitial configuration.

There is considerable evidence to indicate that if the temperature is high enough so that defects such as vacancies and interstitials are mobile, they will move and finally form large clusters. Large clusters of both kinds have been observed with considerable probability that the observations are indeed correctly interpreted. It is now believed that if these clusters become large enough, they may actually collapse, especially on close-packed planes, to give dislocation loops of different signs.

It should be pointed out that the final resting positions of defects which move are very complicated. If an interstitial moves directly into a vacant lattice site, complete healing of the lattice can be expected. Vacancies may cluster with vacancies and interstitials with interstitials, however, both interstitials and vacancies may annihilate at particular kinds of dislocations and effectively disappear as they add to or subtract from the edge of the extra plane in an edge dislocation. Again, they may both disappear at surfaces of the solid if they can reach it in time, either may certainly be trapped at impurity atoms.

As a final complication it should be pointed out that once an interstitial finds itself relatively close to a vacancy, so that it is within the region of the strain field of the vacancy and is strongly influenced by the vacancy, the structure that the interstitial chooses may be affected by the field of that vacancy.

It should be pointed out again that calculations are not reliable in any of these areas. At the present time the calculations do more to indicate the possible complexity of the situation rather than to resolve any of the difficulties.

Figure 14 shows the annealing behavior in copper following bombardment with various types of radiation.¹⁶ With all types of irradiation, the annealing shows a strong, large peak at about 400° K, with various structures on both sides of the peak and a tendency to show an additional peak at about 300° K.

Closing your eyes to the little wiggles, this general pattern seems to follow the simple kind of annealing that was mentioned earlier. Since the energy of motion of interstitials is supposed to be lower than that of vacancies, the 400° K peak may be taken to correspond to the motion of interstitials annihilating at some vacancies. Not all interstitials do that, some are trapped or go to dislocations or surfaces, leaving some vacancies which then move at around 300° K.

The trouble with this proposed simple scheme is that copper has undergone considerable study with respect to the properties of vacancies, and it now seems certain that the energy of motion of vacancies--single vacancies in copper--is sufficiently high that they could not be undergoing significant motion at 300° K in the time period for isochronal annealing. Under standard isochronal patterns, the vacancies should not move significantly until around 400° K. This may not seem like a large difference, and one might question the energy of motion values, but the experiments seem to be fairly precise. The energy of motion appears in an exponential when the number of jumps per unit time is considered. It would take a significant change in the energy, well beyond the errors in experiments at the present time, to allow for a peak to be seen at 300° K. Almost all investigators now agree that a peak at 300° K (in copper) has to be explained by something other than the motion of vacancies.

I. Irradiation with Electrons

In many ways electron irradiation should give the most simple results of all. If the energy of electrons high enough to make displacements and yet not give high energies to the knock-ons has been chosen, one might expect that the damage would be in the form of fairly simple pairs. Looking at the curve for 1.4 MeV electrons in Fig. 14, there are a number of prominent peaks. I especially call your attention to three peaks which appear at a temperature below that of the onset of mobility of an interstitial. It is now believed that this is below the onset of mobility of any defect in copper. How is the presence of these peaks explained? Some experiments performed a few years ago by Simmons and Balluffi¹⁷ indicated, at least in copper and probably other good metals, that the observation of annealing is an observation of

Fig. 14--Recovery spectrum in copper following bombardment with various types of radiation (after Schilling¹⁶).

annihilation. When an interstitial clusters or when an interstitial goes to a trapping site, not much resistivity is removed; most of the resistivity removal results when a vacancy and an interstitial collapse. One might assume that the peaks below 40° K correspond to annihilation of a vacancy by an interstitial.

How does this happen when nothing can move? Here the concept of the close pairs is introduced, i.e., some pairs are stable but close enough together so that they are in each other's strain field. As soon as the temperature is raised, it becomes possible for the interstitial to move toward the vacancy because the attraction energy of the vacancy lowers the energy of motion, and the pair collapses. These peaks have been studied in some detail by Sosin¹⁸ in California, and he seems to have been able to associate the various peaks with various lattice positions of the vacancy-interstitial pairs. The suggestion which was originally made by Corbett, Smith and Walker¹⁹ is that the main peak, Peak I_D at 40° K, is due to the onset of motion of the free interstitials themselves.

The low temperature part of Peak I_D is the range in which most of the interstitials find the vacancies from which they came. In electron irradiation the initial correlation should be high, but as the temperature is raised the interstitials wander away from their own vacancies and begin to annihilate other vacancies. The I_E peak that is seen at 50° K for electron irradiation, 5 MeV alpha particle irradiation and thermal neutron irradiation is supposed to be due to this second stage, corresponding to random diffusion of interstitials to other than their own vacancies and is well separated from the annihilation of correlated pairs. It is to be noted that fast neutron irradiation does not seem to show the I_E peak. This is usually attributed to the supposed complicated structure of the fast neutron resultant cascades in which many vacancies and interstitials are close together, and it becomes hard to observe the annihilation of uncorrelated pairs.

J. Annealing at Higher Temperatures

The annealing between 60° K and 200° K is called Stage II and shows different behavior for different irradiations. Fast neutron irradiation shows quite a bit of Stage II annealing, but with a complicated structure, difficult to analyze.

In Stage II annealing after thermal neutron irradiation shows considerable structure but not much is shown for the other two irradiations. What has been indicated strongly by certain experiments, especially those of Martin at Harwell, is that it is possible to find strong annealing in the Stage II temperature range, which results from the detrapping of interstitials from impurities. In moving around the low temperature range, 55° K, some of the interstitials become trapped at impurities. As the temperature is raised, they untrap from the impurities and find some stationary vacancies, causing more annihilation. This, in the case of doped copper, can produce some sharp peaks in the Stage II range and is supposed to account for some of the annealing.

The annealing from 200° K to about 300° K is generally called Stage III and shows itself prominently in the 5 MeV alpha particle electron irradiation, in the thermal neutron irradiation at apparently a slightly lower temperature and in the fast neutron irradiation.

K. Correlated-Unrelated Pair Annihilation Theories

Figure 15 shows results of a theoretical fit to experimental data, the experimental data being taken with the thermal neutron damage,²⁰ corresponding therefore to fairly low energy recoils. The theoretical work is that of Walter Atkinson,²¹ who used what is called the Waite theory of annealing,²² in which he allowed the original distribution of damage to be in pairs with a certain distribution of separation distances for the pairs, all the pairs being correlated, i.e., around each vacant lattice site an interstitial was sure to be found at some distance. When the temperature is raised in an isochronal anneal, the correlated annealing will be observed first, then as the interstitials escape from their vacancies and annihilate vacancies whose interstitials have also left, the higher temperature portion of the annealing is observed.

Both the experiment and the theory seem to indicate that the correlated part of the annealing is, as expected, first order. The temperature at which the peak occurs does not shift with increased concentration of the damage, whereas for the uncorrelated part the behavior is that of a second order process, and the temperature of the peak shifts to lower temperature as the concentration of the damage increases, as indicated in Curves A, B and C.

The theoretical fit with experiment seems convincing and leads to values for certain properties of the original pair distribution and capture radii for the defects. The Gaussian width of the pair distribution needed to fit the data was about 4 nearest neighbor distances. An atomic vibrational frequency of 10^{13} was put into the calculation and the energy of motion for the interstitial was found to be 0.12 eV. This calculation included not only the possibility of annihilating at vacancies but also that interstitials might cluster to form di-interstitials. It turned out that the interstitial-vacancy capture radius was about 2.3 nearest neighbor distances, whereas the interstitial-interstitial clustering radius was about 6.6 nearest neighbor distances, which indicates that the attraction of interstitials for each other was greater than the attraction of vacancies for interstitials.

Figure 16 shows more clearly the Stage III effect. The changes with defect concentration are apparent. As the defect concentration is increased, the I_E peak disappears into the side of the I_D peak, and the Stage III peak also shifts to the left. Many investigators have shown that at least part of Stage III is pure second order and shows the appropriate second-order shift. It has become necessary to assume that some defect is diffusing around in the otherwise perfect lattice, presumably (from the Simmons and Balluffi experiments) causing annihilation, with an uncorrelated free migration to annihilation sites. The identification of this defect is one of the central problems in the entire basic study of radiation damage at the present time.

It has been established that the interstitial moves in Stage I, and we have eliminated the vacancy as being of too high an energy of motion to move in Stage III (in copper). What is left?

There have been various speculations and a great deal of discussion and controversy. It has come effectively to two different models. The first is the interstitial-di-interstitial model. It has been shown by Corbett,

Fig. 15--The application of correlated pair theory to the I_D - I_E annealing stages in copper after thermal neutron irradiation. (Expt., R. R. Coltman, et.al.²⁰, Theo., W. E. Atkinson²¹).

Fig. 16--Annealing spectrum of irradiated copper.¹⁶

Smith and Walker that Stage III is intrinsic in the sense that it is created by the interaction of the irradiation with itself and does not depend on, e.g., impurity content. Having eliminated the vacancy as the diffusing entity, they originally suggested that Stage III was due to di-interstitials, which is at least consistent with the calculations of Atkinson. During Stage I_E the interstitials cluster, become immobile and then as the temperature is raised to the Stage III region, the di-interstitials thus formed moved as di-interstitials over to a vacancy, whereupon one of them is annihilated, thus removing damage.

It is equally possible that the di-interstitial, being itself immobile, breaks up at around 300° K, and the individual interstitials move over to vacancies and annihilate. Either way, the kinetics as observed would be all right.

The other school of thought has suggested that the observed annealing results from the presence of two different stable interstitials in copper. One interstitial is stable and another interstitial is metastable. This suggestion was originally proposed by Seeger in Germany. His candidate for the metastable interstitial was the crowdion, which he felt would be mobile around the I_E region (50°-55° K). Conversely, the split interstitial which would also be formed to some extent during the irradiation would not become mobile until about 300° K, i.e., Stage III.

Recently, Bauer and Sosin²³ in an experiment and Bauer, Seeger and Sosin²⁴ in an interpretation have somewhat modified and improved this whole model, and they now call it the two-interstitial conversion model. They postulate that in Stage I_E when the crowdion starts moving, it can annihilate at a vacancy; however, at many other different sites, such as another interstitial or an impurity or perhaps even dislocations, it suddenly converts to a normal or split interstitial. Consequently, when the temperature is raised to the Stage III region, there are not only the split interstitials created by the original irradiation but also those converted from crowdions ready to move and perform further annihilations in Stage III.

In order to test this, Bauer and Sosin did a rather clever experiment, which is illustrated in Fig. 17. The annealing is from 100° K up through Stage III. Two different irradiations were performed. (This general pattern of studying the annealing was initiated by Corbett, Smith and Walker.)

Consider a temperature range of mobility, such as that around 50° K in copper. One irradiation is performed well below this temperature, accumulating a large amount of damage. The temperature is then raised above 55° K to 70° K, and the sample is allowed to anneal. The other irradiation is done at 70° K, above the stage of motion I_E, and the amount of irradiation needed to introduce a given amount of damage is compared with the previous irradiation.

The general idea is that by irradiating at low temperature, a large population of interstitials and vacancies is built up and stuck in the lattice; then when the temperature is raised to 70° K, and the interstitials begin to move, they have a good chance of clustering because there is a large density of interstitials.

Fig. 17--The resistivity recovery and slope of recovery of copper as a function of annealing temperature after 1.3 MeV electron irradiation at 70° K (after Bauer and Sosin²³).

Conversely, irradiating at 70° K, a large population of interstitials in the field is never formed, so that there is little clustering of interstitials, but rather a large amount of annihilation and a great deal of trapping at impurities.

The contribution of Bauer and Sosin was not only to compare what happens in the two different irradiations but to anneal through Stage III to see if Stage III is any different after the two irradiations. If Stage III results from either the motion or the breakup of di-interstitials, then there should be quite a difference resulting from these two different irradiations, one of which forms many di-interstitials and one of which should form few di-interstitials. The result they obtained was that Stage III does not differ significantly, in their estimation, for the two different types of irradiation.

They came to the conclusion that the di-interstitial model for Stage III is not tenable, and in fact Stage III is the same in the two irradiations, which is perfectly consistent with the two-interstitial conversion model, since the two different irradiation-annealing procedures would achieve essentially the same final defect density.

Here we leave the story of the irradiation and annealing in the study of defect motion. In copper we have delved deeply into the situation, but we still haven't touched a great many of the finer points. However, it is clear that though some of the controversy may die down as a result of the Bauer-Sosin experiments, the story is complicated. The unraveling of this spectrum of annealing shows complications that were totally unexpected some years ago.

Theory has not played a heroic role in this story. Seeger's early calculations showed that the crowdion should be metastable as a defect in copper. However, it must be admitted that Seeger's calculations were quite crude compared with modern calculations, which indicate that the crowdion is indeed not metastable in copper. The only real contribution that theory has made is that some people believe these calculations so strongly that they refuse to accept the two-interstitial conversion model, and they look around for other explanations for this complicated behavior in copper; they may yet be proved right.

L. Comparative Annealing Spectra for Cadmium and Copper

Even in relatively simple systems the study of radiation damage is a complicated affair. Figure 18 shows the annealing spectrum in cadmium after thermal neutron irradiation.²⁵

The irradiation was performed at a rather low temperature--approximately 4° K. The first annealing seen is near 5° K, and it is a strong annealing--a huge peak. However, that is not particularly of interest. The fact that this corresponds to a migration energy of perhaps 0.02 eV instead of 0.1 eV or more cannot be simply dismissed; it is certainly possible. The rather fascinating or horrifying thing is that the amount of annealing in this early and predominant stage (and also as can be seen in some of the other stages at higher temperature) depends strongly upon the concentration of the damage originally introduced. The damage of the maximum and minimum doses varies by a factor of 100. The amount of annealing in that first big

Fig. 18--The annealing spectrum shown by cadmium after thermal neutron irradiation (after Coltman, et. al.²⁵).

peak certainly differs by at least a factor of 2, and yet the central temperature of the peak does not change.

This would indicate that in some strange way the concentration of the damage affects the amount of whatever defect is being annihilated in this original large stage, yet all of these irradiations fall on a linear damage rate curve at 4° K.

Contrast this with the behavior of copper in the low temperature range, where the change of concentration shifted the I_E peak into the side of I_D but otherwise made little change. The conclusion after two to three years of thinking about this data and study by experimentalists and theorists is that not a single aspect of it is understood and a really good speculation as to what might be going on in this metal cannot be offered. Cadmium is admittedly somewhat more complicated than copper, being a hexagonal metal whereas copper is a face-centered cubic. However such a vast difference and such a totally unexplained annealing spectrum is still hard to accept.

As a final illustration, consider Fig. 19 from recent work at ORNL. This is a picture taken using anomalous x-ray transmission through a relatively thin sample of copper which had been irradiated at the high temperature of about 400° C. This copper specimen is about two millimeters in thickness and was perfect copper. (It was grown by Fred Young in the Solid State Division.) After irradiation with fast neutrons, the copper specimen was then taken out and examined by means of x-ray transmission using a clever technique which is known as the Borrmann effect. There is some damage in this crystal, this is indicated by the nature of the contrast pattern shown. Copper before irradiation would look completely uniform over the entire field. Notice that the dimensions of the field are several millimeters. If the copper had been irradiated at low temperatures, then warmed up with the type of isochronal annealing previously discussed, and finally examined at room temperature, again nothing would be seen in this field. It would be essentially a uniform field, but after irradiation at 400° C, there is a peculiar pattern which shows some type of organized behavior over very long distances. At present there is no explanation of this effect.

III. Conclusions

We are asked to predict the effects of tremendous doses of fast neutrons-- 10^{23} --on a complicated system like commercial grades of graphite at high temperatures when we do not understand the behavior of simple copper under carefully controlled conditions. As fundamental investigators--both experimentalists and theorists--of solid state physics and defects in solids, we are a long way from contributing directly to the prediction of the changes of properties of technologically important materials under irradiation. At present studies of the latter type must be done for themselves, i.e., if one wants to know what happens to a piece of graphite at high temperatures and high doses, one must put it in a neutron flux and observe the damage.

I hope that some day the two forms of study, the empirical irradiation studies and the basic scientific studies, will meet, but I must conclude that I see this meeting as being many years in the future.

Fig. 19--The pattern of damage observed in copper by use of the anomalous transmission of x-rays on a sample irradiated at about 400° C with fast neutrons (10^{19} nvt) (after Hulett, et. al. 26).

DISCUSSION

QUESTION The first figure shows the dimensional changes in graphite as a function of neutron dose. Would you expect the same effects on single-crystal graphite?

ANSWER It seems that the effect is a crystalline effect resulting from the introduction of extra basal planes in the crystallites. Yet, it is also found that above a certain temperature ($\sim 500^{\circ}\text{C}$) the larger the crystallite size the smaller the c-axis growth. At sufficiently high temperatures (above 800°C) the growth would be negligible in single crystals. It is in the nuclear-grade, polycrystalline graphites that the growth rate, which is high at the temperatures below 300°C and there known as the "Wigner disease", is again high at temperatures above 500°C , after having reached a minimum in between.

QUESTION For the isochronal annealing experiments, has anyone ever identified any higher temperature peaks that they associate with clusters in annealing?

ANSWER Not to my knowledge. You can read papers on all kinds of speculations, but there is no general agreement at the present time for such an identification.

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RADIATION EFFECTS AND NUCLEAR SCIENCE PROGRAM
AT NORTH CAROLINA STATE UNIVERSITY

T. S. Elleman
North Carolina State University

The nuclear engineering program at North Carolina State University is concerned with both radiation effects and nuclear materials. Nuclear materials are included with radiation effects because they go hand in hand in the department, which is mainly interested in reactor problems. Many of the problems are much more closely allied with nuclear materials than with radiation damage so the discussion of these two subject areas will be carried through in parallel.

When one looks at the general field of radiation damage, several facts become apparent. One is that a large number of the experiments being carried out today require rather involved, expensive and complicated equipment. Examples are cryostats for irradiating at low temperatures, high flux reactors for obtaining high radiation exposures in relatively short times, in-pile instrumentation for making measurements in the reactor and environmental chambers for controlling environment.

A second observation, and one which tends to follow logically from the first, is that most of the work is being carried out in national laboratories and special materials laboratories where there is money and facilities to support the equipment required.

This raises the question of just what the role of a university is in radiation effects programs; and more specifically, what is the role of a nuclear engineering department in carrying out programs of this type. I do not propose to answer the question of what nuclear engineers should be doing in the area of radiation effects. I will try to outline the academic program and the type of research which is being undertaken at North Carolina State University. This will show the approach one university is taking in the area of nuclear materials.

Perhaps a good place to start is to ask the question: "Who is interested in radiation damage measurements?" In answering this question, one can identify the four general areas of interest shown in Table I. This table delineates the major fields, emphasis and some of the specific requirements of groups that are involved in radiation damage experiments.

TABLE I

Major Fields with Radiation Effects Interest

<u>Field</u>	<u>Emphasis</u>	<u>Requirements</u>
Solid State Physics	Fundamental properties of solids	High purity materials, well defined radiation fields, low temperature irradiation facilities
Nuclear Reactor Development	Properties of reactor fuels and structural materials in radiation environments	High neutron fluxes well defined energy spectra, in-pile instrumentation
Weapons Development	Radiation burst effects	High level radiation sources, burst capability
Space Systems	Space radiation effects	Electron, proton and x-ray radiation sources

The first field listed is solid state physics. The main interest in this area stems from the fact that radiation displaces atoms in solids, resulting in defects which produce physical property changes in materials. It may be possible to relate these property changes to the defects that are produced and gain some understanding of the importance of defects in solids. The interest here is not in irradiation per se, it is in using radiation as a tool to gain a better understanding of the physical properties of solids. The experimental requirements are usually fairly sophisticated, and one tends to work with high-purity materials, well defined radiation fields, low temperatures to prevent annealing of radiation damage, and materials in which the properties are well characterized.

Most experiments are carried out with copper, silver, gold and other materials which are well understood. This is reasonable from the viewpoint of the solid state physicist but does not help the nuclear engineer as these materials are rarely used in reactor construction.

The next group with a radiation effects interest are scientists and engineers in reactor development. They are involved because the materials used in reactor construction are often adversely affected by radiation fields. Examples are irradiation swelling of reactor fuel elements and embrittlement of reactor pressure vessels. An understanding of these effects is necessary in the design of safe reactors. This group has an interest in radiation damage, but there is more of a tendency to duplicate the radiation environment that the material is actually going to encounter in utilization. The requirements in this field tend toward high neutron fluxes, well defined neutron energy spectra and in-pile instrumentation where one can measure what is happening in the radiation field.

The third group includes nuclear weapons-development activities. Scientists in this area become involved with radiation effects because detonating nuclear weapons can produce intense radiation fields which interfere with electronic systems. This can become critical if a nuclear weapon is detonated near an electronic guidance system on a missile coming into the atmosphere, producing high radiation fields near the missile. The tendency in this field is to look at burst radiation effects to get some idea of what is happening in solids during radiation bursts.

The last group involves space systems. The motivation stems from the fact that the van Allen belts produce rather high radiation fields, and these can adversely affect semiconductor materials and other components in a satellite. Most investigators with an interest in this area work with charged particle radiation sources in evaluating the effects of space radiation on satellite components.

Since this is a nuclear engineering department, the activities are much more closely involved with reactor technology radiation effects and nuclear materials problems than with the other areas--although there is occasionally some overlap in solid-state physics problems.

I. Development of the Nuclear Engineering Department

Nuclear engineering programs have been active at North Carolina State for over a decade. The present department was split from the Physics Department in 1960, and our growth since that time has been rapid. There are now 13 full-time faculty members, of which four have materials interest.

In addition to the regular faculty, there is also an affiliated graduate faculty. These are faculty members who are attached to other departments, such as Chemistry, Physics, Electrical Engineering and Mechanical Engineering; but who also have nuclear interests. These people help supervise research topics, serve on committees and in general complement the faculty in nuclear engineering. The affiliated faculty in combination with the permanent faculty gives a large group on which to draw in supervising programs.

The department activities in materials and radiation damage research have been enhanced considerably in the last year by an NSF Science Development Grant, which funded nine new faculty positions in the Engineering School. All of these positions were designated for new faculty with materials interests and the Nuclear Engineering Department was awarded two positions. It is expected that there will be a considerable expansion of the materials program, as well as an expansion of materials research in other departments of the Engineering School.

There are about 45 graduate students at the present time, approximately equally divided between M.S. and Ph.D. candidates. Their backgrounds are similar to those encountered in other nuclear engineering programs. Physics and nuclear engineering graduates comprise the bulk of the graduate body, but occasionally mechanical engineers, electrical engineers and even a rare chemist or mathematician come into the program.

II. Facilities for Nuclear Education and Research

Some of the facilities for nuclear education and research at North Carolina State University are shown in Figs. 1-7. Nuclear Engineering is housed in Burlington Nuclear Laboratories, which is shown in Fig. 1. The rather rapid, recent expansion has produced severe strains on this building, so the faculty is scattered in a number of adjacent office and laboratory buildings. The tower visible on the back of the building is for exhausting gases from the reactor floor and for supporting the cooling tower which is visible at the top.

Figure 2 is the North Carolina State University heterogeneous reactor. It is a water-cooled, graphite moderated, heterogeneous reactor with MTR type fuel elements. Experiments can be lowered into the facility, either through the top where the two operators stand, or through beam ports in the side. Unfortunately, the maximum operating power level is 10 kW which is entirely too low for most radiation damage experiments. This means one has to be extremely selective in designing the kind of experiments to be carried out or else send the materials to other facilities for irradiation.

Figure 3 is the reactor console. The facility was constructed earlier than many, and there is probably more instrumentation than at many other reactors. However, this can be an advantage in using the reactor as a teaching tool.

Figure 4 shows one of the laboratories where the intermediate levels of radioactivity are handled. There are also a shielded cave and glove-box, as shown in the background.

Figure 5 is a 26,000 cu cobalt-60 source which is housed in a water-filled pool. This is particularly useful for radiation damage experiments on organic materials, chemical synthesis studies and other similar programs. The large, rectangular container in the foreground can be lowered between two plaques containing cobalt-60 which gives a fairly even radiation exposure over a relatively large volume. There is some flexibility in rearranging the cobalt strips, so that one can vary the size of the specimen irradiated and the dose rates.

Figure 6 is somewhat obscure because it was taken through the 20 foot water head of the cobalt-60 pool. This is a variable irradiator and irradiation turntable. The strips which can be seen contain cobalt-60 and are mounted in an iris diaphragm so that one can move them out or contract them down around the sample. This gives flexibility in varying the dose rate for a sample. By closing the strips down tight against a small sample, dose rates of around 5×10^5 rads per hour can be obtained.

The University also has access to a 1 MeV van de Graaf accelerator. This facility can accelerate either protons or deuterons. One can get beam currents of 1 to 20μ A with the facility and expose a sample of about 1 cm^2 in the area.

Fig. 1--Burlington Nuclear Laboratories, North Carolina State University.

Fig. 2--North Carolina State University 10 kW heterogeneous nuclear reactor.

Fig. 3--Console for research reactor shown in Fig. 2.

Fig. 4--Radioisotope handling area in Burlington Nuclear Laboratories.

Fig. 5--Specimen holder ready for lowering into pool containing 26,000 cu cobalt-60 source.

Fig. 6--Variable irradiator and canister turntables in cobalt-60 source pool.

One facility which is useful in materials work, as well as in other work, is an IBM system 360 model 75 computer. It is housed in a partially completed building at the Research Triangle Park. There are stations at North Carolina State, Duke University and the University of North Carolina for program input and obtaining output data.

Although experimental facilities in Burlington Nuclear Laboratories are currently limited, there are plans for the future as shown in Fig. 7. The North Carolina State Legislature has appropriated approximately a million dollars to be used in the construction of a new Nuclear Science and Engineering Building which is illustrated here. It will be a four-story building and will attach on the north face of the existing Burlington Nuclear Reactor building. Our department will share about half of the building and will obtain considerable expansion of our existing facilities. The large building in between Burlington Laboratories and the new building will house a 1 MW Pulsar swimming pool reactor, which will provide considerably higher radiation exposures than we have been able to obtain to date.

III. Nuclear Engineering Curriculum

North Carolina State awards a degree in nuclear engineering at the B.S., M.S. and Ph.D. levels and also provides a materials option in all three of these programs. Table II gives an approximate breakdown of a typical course distribution for a student in the materials option.

TABLE II
Nuclear Engineering - Materials Emphasis
(Typical Curricula)

<u>B.S.</u>	Mathematics	21 credits
	Physical Sciences	27 credits
	Liberal Arts	31 credits
	General Engineering	30 credits
	Nuclear Engineering	15 credits
	Materials Science	12 credits
<u>M.S.</u>	Mathematics	9 credits
	Nuclear Engineering	13 credits
	Materials Science	9 credits
<u>Ph.D.</u>	Mathematics	15 credits
	Physical Science and Engineering	9 credits
	Nuclear Engineering	22 credits
	Materials Science, Radiation Effects, Solid State	12 credits

At the B.S. level one of the graduates would take about 12 semester credits in materials science. This consists principally of courses emphasizing structure and properties of materials, radiation effects, and metals and ceramics tech-

Fig. 7--Proposed new building for nuclear engineering teaching and research.

nology. The rest of the student's curriculum is divided into 21 semester hours in mathematics, 27 hours in physical sciences, etc.

A nuclear engineering undergraduate actually takes few credits in nuclear engineering. The program corresponds more closely to an engineering science curriculum, so a nuclear engineering graduate has considerable flexibility in professional choice and is not committed only to nuclear engineering.

At the M.S. and Ph.D. levels a student would typically take 9 to 12 additional credits in materials courses. The major course emphasis is on the more basic courses in nuclear engineering including reactor theory. These programs are kept as flexible as possible to give maximum accommodation to specific student interests. There are a number of courses available to the student in the materials option, including solid state physics, quantum mechanics and a variety of courses on physical metallurgy, properties of ceramics, etc. In the Nuclear Engineering Department itself there are four courses:

Table III shows the two most basic of the Nuclear Engineering Department materials courses. One is Materials Problems in Nuclear Engineering. This is concerned with the factors that dictate the choice of specific materials in a reactor and the effects on these materials of reactor environments. The course emphasizes mechanical and electrical properties of metals, ceramics and polymers.

TABLE III

Nuclear Engineering Materials Courses

NE-562	<u>Materials Problems in Nuclear Engineering</u> Nuclear materials, reactor requirements, mechanical and electrical properties of metals, ceramics and polymers, minimization of deleterious effects.
NE-570	<u>Radiation Effects on Materials</u> Damage mechanisms, comparison between results and theory, measurement techniques, annealing phenomena, observed effects in metals, ceramics, fissionable materials and polymers.

The second course is directed more specifically to radiation effects on materials. It is concerned with damage mechanisms, calculating numbers of displaced atoms in solids, comparison between results and theory, annealing experiments, treatment of annealing data and in general a discussion of the kinds of property changes produced in metals, ceramics and polymers by radiation fields.

Both of these courses are designated as 500-level courses, which means they are open to both advanced undergraduates or graduate students in a nuclear engineering program.

The other two courses are oriented more specifically toward the research interests of the individuals who teach them. These are shown in Table IV.

TABLE IV

NE-591	<u>Mathematical Simulation of Physical Processes</u> Computer applications, experiment design, dynamical, variational and Monte Carlo methods, processing of results.
NE-671	<u>Transport of Matter in Nuclear Reactors</u> Mass transport theory, diffusion, fission product migration, aerosol formation, fission product release from reactors and countermeasures.

The first course is entitled Mathematical Simulation of Physical Processes. While this does not sound like a materials course, it actually has a useful application to materials problem. The course is oriented toward application of computer techniques to physical processes. The course is presented by Beeler of the Nuclear Engineering Department, whose research interest is in using computers to simulate the formation and behavior of defects in solids; so materials applications are well covered.

The second course is entitled Transport of Matter in Nuclear Reactors. This course is concerned with fission product migration in reactors, mass transport phenomena and discussion of the mechanisms by which material can be transferred within reactors.

These two courses are being offered for the first time this coming year, so it is not certain how these will fit into the curriculum, or just what the student response is going to be.

IV. Research Areas and Activities

There are three research programs that come under the general heading of fission-product behavior in reactors. When one looks at some of the newer reactor concepts that are being proposed, it is readily apparent that a number of the major problems that are anticipated are materials problems. If one starts looking into these materials problems, a significant number of these are concerned with how fission products behave in reactors.

This fission-products behavior problem can have a variety of facets. Since the fission recoils deposit considerable energy in the reactor fuel as they slow down, there can be a fission-fragment radiation-damage problem within

the fuel. After the fission products come to rest, they constitute foreign impurity atoms in the solid and can produce additional undesirable effects.

At the higher temperatures proposed in a number of future reactors, there is a greater probability of fission-product diffusion through claddings, segregation within the fuel and swelling from rare gas clustering. Problems such as these can be grouped under this general heading of fission-product behavior.

The first research program is concerned with the diffusion of fission products in solids, and more specifically, with the effect of radiation damage on diffusion of rare gases in solids. This is an area of research under funding by the Research Division of the AEC.

The motivation for this work is due to the recognition that an appreciable fraction of the fission products produced in fission are rare gases, and if temperatures are high enough, these gas atoms can come together in a reactor fuel and form bubbles. If these bubbles exceed certain critical sizes, it is possible to produce swelling in the fuel which can reach undesirable proportions.

Figure 8 illustrates how swelling can occur in a reactor fuel. Shown here is a piece of glass that has been embedded with uranium, irradiated and heated to produce coalescence of the gas atoms to form bubbles. Although this is done with glass, the effect is similar to what can be obtained in reactor fuels at high temperatures. The general problem of migration of fission gases in solids is of interest because it relates to the formation of gas bubbles and to the fission-product release behavior in reactor fuels.

Historically, fission-gas release from reactor fuels has been measured by lightly irradiating a fuel specimen in a reactor and then heating it to drive out the gases, thus permitting the rate of release of fission gases to be measured. The kinetics of this release would then be correlated with the diffusion properties in the fuel.

Most experiments of this type gave results which did not correlate with classical diffusion predictions. It was generally observed that gas release was too rapid at short heating times and dropped off much more rapidly than was predicted at longer heating times. A number of explanations have been advanced for this behavior including radiation damage, gas-atom trapping at defects in the lattice, pipe diffusion along grain boundaries and a variety of other explanations.

What has been done in the present work is to look at simpler systems than fissionable fuels and try to understand how gas behaves in these systems. The hope is that these results might be used as a means for understanding what is happening in reactor fuels. The solid which has been used is single crystals of cesium iodide. These are impregnated with radioactive xenon, heated, and the time rate of release of the xenon gas measured with time.

Xenon is introduced into these specimens by one of three different mechanisms. As shown in Table V, one way is to dope the cesium iodide with iodine-133 before growing the single crystals. Iodine-133 is radioactive and

Fig. 8--Gas bubble formation in irradiated solids.

decays with a 20.5h half-life to form xenon-133 which is also radioactive and has a half-life of 5.3 days.

TABLE V

Effect of Radiation on Rare Gas Diffusion in Solids - CsI

- I. ^{133}I Tagging - 10 eV/gas atom

$$^{133}\text{I} \xrightarrow[20.5\text{h}]{\beta^-} ^{133}\text{Xe}$$

$$^{133}\text{Xe} \xrightarrow[5.3\text{d}]{\beta^-} ^{133}\text{Cs}$$
- II. Fast Neutron Transmutation - 50 keV/gas atom

$$^{133}\text{Cs} (n, p) ^{133}\text{Xe}$$
- III. Fission Recoil Doping - 25 MeV/gas atom

$$^{235}\text{U} (n, \text{fission}) ^{133}\text{Xe}$$

If crystals are permitted to stand for several days after growing, the result is a homogeneous distribution of radioactive rare gas in the solid. The beta particle produced during the radioactive decay does not produce much radiation damage to the lattice, and the recoil to the iodine-133 atom is fairly low (in the range of 10 eV per gas atom). This gives essentially a radiation-damage-free procedure for introducing the gases in the solid.

Another technique is to irradiate the cesium iodide specimens with fast neutrons. Radiation damage from the fast neutrons is obtained and also an (n, p) reaction on cesium-133 which gives xenon-133, again giving a homogeneous rare gas distribution in the solid. The recoil energy from the transmutation process is considerably higher than the recoil energy for iodine-133 doping. The energies are in the range of 50 keV per gas atom generated.

A third technique is to introduce uranium impurity atoms in the solid or to surround the specimen with uranium and irradiate with neutrons. Fission-product rare gases are recoiled into the solid to give an energy disposition of around 25 MeV per gas atom.

The three techniques permit varying the amount of radiation damage over rather wide limits per gas atom of xenon-133 generated within the solid. If one grows the best crystals and uses the first technique which produces no radiation damage, classical behavior of the rare gas diffusing in the solid can be obtained. Both the kinetics of gas release at a given temperature and the temperature dependence of the diffusion coefficients corresponded well to solutions of Fick's law of diffusion.

If one introduces defects in these crystals by growing them rapidly, the crystals are cloudy and the x-ray patterns indicate considerable strain.

These defects serve as trapping sites to trap the gas atoms, and one can get release kinetics that look much like those obtained with reactor fuels. A plot of the fraction of gas released vs. time drops off much more rapidly than one expects for classical diffusion.

If the solid is surrounded with uranium and recoiled in the fission products, the fission products diffusing out of the surface layers have exactly the same diffusion coefficients as the classical values. This shows that enhanced diffusion back along the recoil path of the fission fragment or radiation damage effects are not occurring.

If these solids are doped with impurities that have different cation-to-anion charge ratios from the host lattice, the anion and cation vacancy concentrations can be changed. By making gas diffusion measurements on these solids, it is shown that gas release kinetics are unchanged. By implication diffusion is not occurring through a vacancy process, and gas atoms appear to be diffusing by an interstitial mechanism. Through experiments of this type, a better understanding of what factors affect fission product migration in solids may be obtained.

The current trend is to look at increasing radiation damage levels in these crystals and to go to higher and higher gas concentrations to see what effects can be observed.

The second program is concerned with the escape of fission products from porous ceramics at high temperatures. The motivation for this work comes from the increasing interest in ceramic fuels in gas-cooled reactors where the high temperatures of operation result in increased mobility of the fission products in the fuel.

If the typical gas-cooled reactor fuel element consisting of UO_2 or UC_2 in a graphite matrix is heated, certain fission products (such as barium and strontium) can diffuse rapidly through this graphite matrix. Others like zirconium and niobium diffuse at much lower rates. The question then arises regarding the mechanism which permits some fission product to diffuse rapidly while others move much more slowly.

The results show that diffusion in these porous bodies is not classical, and the best explanation at present is that there is a combination surface diffusion mechanism and an adsorption-desorption phenomenon within the graphite specimen that explains the fission-product transport through the fuel.

Lloyd Zumwalt of the Nuclear Engineering Department is continuing work that he began at Gulf General Atomics in this area and is trying to measure the adsorption equilibrium of fission products on graphite as a function of temperature. He introduces fission products into a system under vacuum, controls the temperature in the presence of a graphite body, allows the system to reach equilibrium, and then uses Knudsen-cell techniques to measure the fission-product vapor composition. From the vapor concentration, he is able to establish the equilibrium adsorption concentrations on the graphite.

The hope is that this information can be combined in some general theory to explain how fission products move through graphite fuel bodies. This information is of great use in evaluating the hazard from fission-product release in high temperature reactor fuels.

The third program in this area is concerned with how the fission products behave after they get out of the fuel element. This work is being carried out by Necatte Ozisik, who is one of the affiliate graduate faculty members, and who works quite closely with nuclear engineering. The program is sponsored by the Nuclear Safety Division of the AEC and more recently through the nuclear safety program at the Oak Ridge National Laboratory.

The importance of fission-product migration in gas streams was defined well in an earlier paper by Cottrell,¹ which considered the potential hazards resulting from fission-product-release accidents in reactors. The basic questions are where the fission products go and what are their concentrations following their release during a cladding rupture or fuel element meltdown. If all of the fission products were to deposit on the first surface they hit, this would pose a considerably less severe environmental hazard than if they were carried freely through the system with the possibility of getting out into the environment.

Professor Ozisik, by initially considering simple systems, is trying to predict what the distribution of fission products is going to be in these systems at some time after the release. First he considered cylindrical ducts with constant laminar flow of a fission product down the tube and calculated the expected distribution along the walls, assuming that all of the fission products stuck to the wall on the first impact. He then considered more complex cases in which only a fraction of the fission products stuck to the wall on impact, and the remainder continued down the tube. This approach permitted formulation of a sticking coefficient that represented the probability of an atom adhering to the surface upon impact. A comparison was then made between the predicted distributions and the experimentally measured distributions to see whether a correlation between prediction and experiment was possible.

The type of results obtained are illustrated in Fig. 9. This figure shows the normalized wall concentration of a specific fission product as a function of distance down a cylindrical tube. The quantity Z along the abscissa may be thought of as a measure of distance down the tube. The points representing the experimental data have been generated at ORNL, Battelle Northwest and other facilities. The straight lines represent the fit of the data that can be obtained using arbitrarily selected sticking coefficients for the equations.

The agreement obtained does not necessarily mean that the theory is correct, because arbitrarily selected sticking coefficients were used. Conversely, it is apparent that it is possible to correlate experimental data with some reasonable mechanism. It is next desirable to apply the results to other circumstances to see if this correlation continues to be valid.

Professor Ozisik currently is working on more complex systems such as fission-product deposition within reactor containment vessels. He

Fig. 9--Fission product deposition rates in cylindrical tubes.

is assuming an initial, homogeneous distribution of fission products in a gas phase and attempting to determine the deposition rates on the walls of the containment vessel. He considers perturbing influences such as convective flow, steam formation and other parameters to evaluate their effects on fission-product distributions.

Figure 10 illustrates the kinds of results which can be obtained. The ordinate is the mean concentration of iodine in the gas phase divided by initial concentration of iodine in the gas. The decreasing gas fraction with time reflects deposition on the walls of the vessel. The squares and circles represent the experimentally generated data taken from ORNL work. It is apparent that it is possible to obtain a fit between experimentally generated data and reasonable mechanisms for predicted deposition. The intent in this work is to try to develop predicted behavior patterns for the fission products which are released from fractured or melted fuel elements.

These three programs illustrate a continuity between behavior of fission products in a solid, diffusion out of a fuel specimen through a porous matrix and behavior once they have gotten out of the fuel.

There is a different program which utilizes computer techniques to predict what is happening in solids. This research is being carried out by J. R. Beeler, Jr. of the Nuclear Engineering Department and is a continuation of work he engaged in while at General Electric Company. He overcomes our limitations in reactor power by completely dispensing with any experimental equipment and instead uses a computer to simulate a solid system. He then introduces perturbations in the system in the form of displaced atoms and lets the computer tell him what is happening and how the energy is going to be redistributed. Holmes² outlined this method in sufficient detail, thus there is no need to discuss the principles involved here.

This technique of using a computer to simulate processes in a solid can have some decided advantages. The method can display defect distributions on a microscopic scale and provide a basis for predicting changes in the solid. An example is the replacement spike which is the displacement cascade resulting from energetic lattice atom displacements in solids. It is common practice among many investigators to interpret macroscopic property changes produced by radiation in terms of effects which occur within the displacement spikes. Computer techniques permit a better visualization of displacement spikes, their size and how they are distributed within a solid.

Channeling of energetic lattice atoms is another process which can be simulated by computer techniques. A computer analysis shows that atoms which conform to specific angle and energy requirements can channel between lattice planes of atoms and go long distances in solids before losing energy through scattering interactions. A computer program can be used to predict the distance an atom will travel when it meets the requirements for channeling. There is experimental evidence which shows that some atoms travel long distances in solids, apparently through channeling processes. A computer technique can also be used to predict that atomic collisions should become "focused" along specific lattice planes under certain conditions. One atom would hit another in a certain lattice direction, and the collisions would be focused along a row of atoms so that energy transfer would occur in specific directions. There is some experimental evidence to support this mechanism.

Fig. 10--Comparison of theoretical and measured deposition rates for iodine as a function of gas velocity

The limitations of using computer techniques have already been discussed by Holmes. It is possible to simulate only a certain size of a crystal, and the damage produced by a specific cascade may well extend outside of the volume simulated on the computer. Uncertainties in the interaction potential can strongly affect the results which are obtained.

There is one more program which differs from the others in that it is not related to reactor technology. This research is concerned with transmutation doping in semiconductors. It is being carried out by Kuruvilla Verghese of the Nuclear Engineering Department under an NSF grant. The electrical properties of semiconductors are dependent upon the impurities they contain, and these properties to a large extent can be controlled by doping the semiconductors with different impurities. If impurities are taken from Group V of the periodic table, there are more electrons in the outer shell than in the semiconductors. These impurities tend to function as donor-type impurities, and n-type semiconductors in which electrons are the majority-charge carriers are obtained. Correspondingly, if Group III elements are added as impurities, they have fewer outer electrons than the atoms in the host lattice, and p-type semiconductors are obtained. If n-type and p-type semiconductors are combined, junction devices are obtained which are useful as transistors, solid-state radiation detectors and solar cells.

In this program Verghese is irradiating silicon with thermal neutrons and using the neutron transmutation reactions to introduce n-type impurities. The reactions involved are shown in Table VI.

TABLE VI

Transmutation Doping of Silicon Semiconductors

If silicon is irradiated with thermal neutrons, (n,γ) reactions are obtained with silicon-28 and silicon-29 to give stable silicon-29 and silicon-30. An additional (n, γ) reaction is obtained with silicon-30 to give silicon-31 which is radioactive and decays through beta emission to give phosphorus-31, an n-type impurity.

If one starts with p-type silicon, irradiates it with thermal neutrons and waits a sufficient time period to obtain decay of the silicon-31, it should be possible to convert the p-type silicon to n-type.

This principle has been well known and has been used by a number of groups to produce junction devices. The problem Verghese is working on

concerns the behavior of the transmutation-generated phosphorus-31 which does not behave the same as if the same amount of phosphorus had been introduced by a diffusion process. The phosphorus produced by transmutation appears to be electrically less active than the diffusion-introduced phosphorus, and conversion to n-type silicon is not obtained at the expected neutron exposure. One possible explanation is that the recoil which occurs during the beta decay of silicon-31 has recoiled the phosphorus atom into some defect state in which it is not participating electrically in the solid. Verghese is looking at minority carrier lifetimes and resistivity measurements to better understand the behavior of the transmutation-generated phosphorus-31. He is also annealing specimens to see how their properties change as a function of temperature in order to get some insight into what is happening during the transmutation doping of solids.

V. Conclusions

The materials program at North Carolina State is quite recent. The longest research program is about 3-1/2 to 4 years old, and most of them are considerably more recent than this. As yet, there is not a good understanding of the problems and advantages of a materials option in a nuclear engineering curriculum. However, looking into future reactor needs, it appears that materials problems are going to be a significant part of the future reactor scene, and there is a definite place for a materials-oriented individual in the nuclear program.

DISCUSSION

QUESTION Have you had problems with solid state people and metallurgical people in setting up the programs which overlap their disciplines?

ANSWER No. There are many examples of different departments offering courses which are similar in content but directed toward the specific needs of one discipline. We have had good cooperation with other departments, and there is considerable interchange of students between departments to take courses of specific interest.

REFERENCES

1. W. B. Cottrell, "Nuclear Safety Research and Development," this conference.
2. D. K. Holmes, "Radiation Effects in Solids," this conference.

PROJECTIONS

J. A. Lane
Oak Ridge National Laboratory

If one looks at recent happenings in the nuclear industry, it is apparent that the age of nuclear power has arrived with a bang. During the last two years, more than 70 nuclear plants have been ordered or planned by U. S. utilities. These will dot the countryside in 23 states in the U. S. as shown in Fig. 1.

Although these comprise a total capacity of more than 40 million kW and a total investment of more than \$5 billion, a more important aspect of this sudden growth of nuclear power is that it represents about three-fourths of all new electric capacity ordered by the utilities for operation in the early 1970's. Thus the coal people are already saying, "We are now number two, but we will try harder."

I. Past Projections

This rapid growth was completely unanticipated as little as five years ago. In November, 1962, in its famous Report to the President, the AEC predicted an installed nuclear capacity only one-fourth as great for this same time period.

Going back a little further into history, 20 years ago the Chairman of the General Advisory Committee (Dr. Oppenheimer) recommended that the AEC not undertake the development of civilian nuclear power because he thought it would never be economically competitive with coal.

Finally, 34 years ago, even the technical feasibility of nuclear power was seriously questioned. In the September 12, 1933 issue of the Herald Tribune, Lord Rutherford scoffed at the entire idea of extracting energy from the atom. History thus shows us that the longer range the prediction, the less accurate it is in terms of real situations.

II. Current Projections

With that word of warning, I shall now project both the growth and costs of nuclear power over the next 34 years and attempt to analyze what the resulting impact will be on our way of life. Let us first look at some projections of the future growth of U. S. nuclear power shown in Table I and the cost projections shown in Table II.

CENTRAL STATION NUCLEAR POWER in the UNITED STATES & PUERTO RICO

Nuclear Reactor Capacity (kilowatts)

● Operable	(14)	1,882,200
▲ Under Construction	(13)	7,359,200
■ Planned*	(32)**	23,732,000
	TOTAL	32,973,400
◆ Announced	(11)**	-

*Contracted for.

**3 Sites Undetermined

TABLE I
Projections of U. S. Nuclear Power Growth

<u>Projector</u>	<u>Year of Projection</u>	<u>Millions of kWe</u>	
		<u>1980</u>	<u>2000</u>
Lane	1954	40	210
Lane	1959	78	600
AEC	1962	40	734
Lane	1966	150	--
AEC	1966	95	--
Lane	1967	150	1150*
AEC	1967	150	--

*70 per cent of total U. S. capacity.

TABLE II
Nuclear Power Cost Projections

<u>Projector</u>	<u>Date of Projection</u>	<u>Mills/kWh</u>
Lane	1955	4 - 5
Lane	1959	6 - 7
AEC	1959	8 - 9
AEC	1962	3.6 - 4.8
AEC	1966	3.5 - 3.9
Lane	1966	1 - 1.5
Lane	1967	0.5 - 1

A. Basis for Projections

The starting point for any projection as to the relative role of competitive energy sources is to find out what the competition really is. To do this one must estimate the future market for new electrical capacity as a function of plant size and decide which of each individual block of capacity is likely to be nuclear or conventional fossil fired. The competitive costs between nuclear and coal fired plants of one particular size are shown in Table III.

Without going into details on the complete analysis which was carried out in 1966, the results led to an installed nuclear capacity of 150 million kW by 1980.

TABLE III

Electricity Production Costs
(1000 MWe Plants Entering Service 1970)

	Mills/kWh	
	<u>Nuclear</u>	<u>Coal</u>
Capital Investment	2.3	2.4*
Fuel	1.4	2.2
Operation and Maintenance	<u>0.3</u>	<u>0.3</u>
	4.0	4.9

*Includes sulfur oxide pollution control equipment.

B. Outlook for Lower Cost Power

The competitive position of nuclear plants in 1970, however, is only the beginning because the trend in nuclear power costs will continue to be downward. Factors contributing to this trend are shown in Table IV. The effect of these factors on power costs in light water reactors is shown in Table V.

TABLE IV

Factors Leading to Reduced Nuclear Power Costs

<u>Factor</u>	<u>Current</u>	<u>Projected</u>
Plant Size, MWe	1000	3000
Process Plant Capacity, Ton/Day	1	10
Reactor Type	Converter	Breeder
Plant Factor, %	80	85 - 95
Cost of Money, %	12	6 - 12

As long as these low power costs prevail, nuclear plants should be able to capture the major share of the market for base and intermediate load plants. Peaking requirements constituting about 30 per cent of the total installed capacity, however, will probably be met by gas-turbine plants which cost about \$25/kW less than the nuclear plants. Thus the nuclear capacity saturation point should be about 70 per cent of the total electrical capacity rather than the 50 per cent postulated by the AEC.

TABLE V

Projected Power Costs
Light Water Reactors (1975-1990)

	Mills/kWh	
	Investor Owned <u>Utility</u>	Public Owned <u>Utility</u>
Capital Investment	1.4	0.7
Fuel Cycle	1.2	0.9
Operation and Maintenance	<u>0.2</u>	<u>0.2</u>
	2.8	1.8

Basis: 3000 MWe Nuclear Plant Capacity
10 T/Day Process Plant Capacity
90% Plant Factor
12%/6% Cost of Money

C. The Advent of Breeders

The costs given previously are based on the assumption that low cost uranium will continue to be available through the stated time period. Many people believe that lots of low cost uranium will be found; however, mining companies are currently drilling at a rate of about 12 million feet a year in order to keep up with the rush of orders for uranium.

The most logical way out of the dilemma is to develop high gain breeders which would replace mining as a source of fissile material. Although the AEC is currently spending on the order of \$100 million annually on breeder development, it is not yet clear when such breeders will be commercially available; nor is it clear what kinds of breeders will ultimately take over the future nuclear business in the same way light water reactors have taken over the current nuclear business. In any event if and when the technical problems are solved, power costs in breeders will be even lower than those for light water reactors, because breeders sell rather than consume fissile material. Typical costs for both thermal and fast breeders are shown in Tables VI and VII.

These power costs are in the attractive range of 1 mill/kWh, however, this is not the ultimate in low power costs. As part of the desalting program at Oak Ridge National Laboratory, we have been looking at more advanced breeder reactors with heat costs low enough to produce desalted water for agricultural purposes as shown in Table VIII.

The reactor with lowest heat costs uses unclad uranium metal fuel in direct contact with the sodium coolant. The technical feasibility of such an approach has not yet been explored in any great depth; however, the British fast breeder at Dounreay operates with coolant in contact with the metal fuel. So far this aspect has not caused trouble, but they are experiencing problems of sodium leakage from an undetected source.

TABLE VI

Power Costs in Molten Salt Breeders

<u>Fuel Cycle Costs</u>	<u>MSBR (Pa)</u>	<u>MSBR (Pa-Pb).</u>	<u>MOSEL (Pa-Pb)</u>
Fuel and Salt Inventory	0.08	0.05	0.09
Salt Makeup	0.07	0.07	0.01
Process Plant Capital	0.06	0.06	0.02
Process O + M	0.07	0.07	0.03
Bred Fuel Credit	<u>(0.11)</u>	<u>(0.12)</u>	<u>(0.19)</u>
	0.17	0.13	-0.04
<u>Plant Investment</u>	0.68	0.68	0.68
<u>Operating, Maintenance, Ins.</u>	<u>0.20</u>	<u>0.20</u>	<u>0.20</u>
Total Power Costs, Mills/kWh	1.05	1.01	0.84

Basis: 7%/5% Capital Charges, 0.94 Plant Factor.

TABLE VII

Power Costs in Fast Ceramic Fueled Breeders
(Public Financing)

Reactor Size, MWt	<u>2500</u>	<u>10,000</u>
<u>Fuel Cycle Costs</u>	mills/kWh	
Fabrication	0.33	0.21
Burnup and Losses	0.04	0.03
Shipping	0.10	0.02
Processing	0.08	0.05
Inventory	0.15	0.23
Bred Fuel Credit	<u>(0.23)</u>	<u>(0.47)</u>
	0.47	0.07
Plant Investment	0.90	0.80
Operating and Maintenance	<u>0.35</u>	<u>0.11</u>
Total Power Costs	1.72	0.98

Basis: 7%/5% Capital Charges, 0.90 Plant Factor, \$10/g Pu.

TABLE VIII

Relative Unit Heat Costs from
10,000 MWt Process Heat Plants
(Public Financing)

	Pu Credit = <u>\$5/g</u> <u>\$10/g</u> <u>\$15/g</u>		
<u>Reactor Type</u>	Cents per Million BTU		
Boiling Water (BWR)	15.2	14.8	14.5
Heavy Water Organic (HWOGR)	14.1	13.7	13.2
Very Large Fast Breeder (VLFBR)	8.3	7.1	6.0
Cast Metal Fast Breeder (CMFBR)	6.4	3.5	0.70

In order to utilize the unclad fuel breeder as a power reactor, it would be necessary to use a thorium metal core instead of uranium (because of temperature limitations on uranium). Such a reactor does not have quite as high breeding gain but does have the potential for power costs as low as 0.5 mills/kWh as indicated in Table IX.

TABLE IX

Power Costs in Unclad Metal Breeders (3500 MWe)

<u>Investor Financed</u>	<u>Mills/kWh</u>	
Pu Value, \$/g	<u>10</u>	<u>15</u>
Fuel Cycle	0	-0.2
Capital and Operating	<u>1.7</u>	<u>1.7</u>
Total	1.7	1.5
<u>Public Financed</u>		
Fuel Cycle	-0.2	-0.4
Capital and Operating	<u>0.9</u>	<u>0.9</u>
Total	0.7	0.5

III. The Future Outlook

Thus it would seem that the potential exists for producing electricity at costs "too cheap to meter." As a matter of fact, in these advanced breeders the sale of plutonium more than offsets all other fuel cycle costs. Under such conditions the utilities will tend toward the sale of capacity rather than energy, and metering becomes unimportant. There is one catch to the

situation, namely, that I have only been talking about the production costs of electricity. Consumers also have to pay for costs of transmission and distribution of the electricity, which are equally important (see Table X).

TABLE X
Consumer Electricity Costs
(U. S. Average)

Year	Mills/kWh	
	<u>1967</u>	<u>1980</u>
Production	7.9	6.3
Transmission and Distribution	<u>7.7</u>	<u>6.0</u>
Consumer Cost	15.6	12.3

What do we do about these costs? Obvious solutions are:
(a) Improve transmission technology via higher voltage, superconducting systems, etc. (b) Reduce transmission distances by bringing producer and user of electricity closer together.

One can achieve this latter objective by building mammoth nuclear power-industrial complexes close to urban centers. In such complexes, the electricity is used on site to produce iron and steel, aluminum glass, fertilizers, cement and other electricity intensive products. The major portion, however, will be used to support the individual and public needs of the super cities of the future. It has been postulated that by the year 2000 there will be 10 super cities containing one-third of the U. S. population. The power requirements of these cities, as shown in Table XI, will be enormous.

TABLE XI
Super Cities of the Future

<u>Super City</u>	<u>Population Millions</u>	<u>Electricity Required, Millions, kWe</u>
New England	7	35
New York	23	115
Delaware Valley	9	45
Chesapeake and Potomac	10	50
Southeast Florida	7	35
Detroit	10	50
Cayahoga Valley	5	25
Chicago	11	55
San Francisco	8	40
Los Angeles	<u>20</u>	<u>100</u>
	110	550

Under such conditions transmission and distribution costs will be greatly reduced. They can be brought down to the order of 1-2 mills/kWh. Consumer costs then from these gigantic nuclear power stations would be of the order of 2-4 mills per kWh or about one-fifth the present U. S. average. Assuming two-thirds of the plants are investor-owned and one-third are public-owned, savings in the cost of electricity in the year 2000 could amount to as much as \$80 billion per year (or \$240 per person per year).

Another way of stating the future situation is that by the year 2000 we will on the average use five times as much electricity per person, but the total family or industrial consumer bill will be no higher than at present. Electricity usage will account for more than 60 per cent of our energy needs, supplying us with most of our heat requirement and a good share of our energy for transportation. With such cheap electricity entire cities will be climatized, putting an end to the problems of winter.

This all may take place 35 years from now; but if history repeats itself, I have undoubtedly underestimated what will really happen.