ESADA GROUP OF UTILITIES AWARDS FUEL DEVELOPMENT CONTRACT TO AI

Proposed SRE Modifications

The Sodium Reactor Experiment (SRE), originally built to demonstrate the feasibility of the sodium graphite reactor concept, has succeeded in this objective, and has produced valuable engineering information on fuel, sodium systems, and the interpretation of a sodium-cooled reactor plant with a steam generating station.

Advances in sodium reactor technology in the area of components, materials, and fuel have demonstrated a substantial increase in capability in sodium graphite reactors. Consequently, under the terms of the general development program, it has been proposed that the thermal power level of the SRE be raised from its current 20 thermal megawatts to about 15 thermal megawatts.

Specific power in the fuel elements, planned as uranium carbide, will be doubled from the present 17.5 W/kg of fuel to about 50 W/kg. Heat flux from fuel elements to coolant will be increased from the present average of 100,000 BTU/sq. ft/hr. Thermal neutron flux in the fuel will be increased from its present value of 2 x 10¹⁶ to 4 x 10¹⁶ neutrons/sec cm².

Pluss would include increasing the reactor outlet temperature to 1200°F in order to provide steam at 1050°F as a demonstration of the capability of this type reactor to match modern steam conditions. The uranium carbide fuel elements would be operated at surface temperatures of 1250°F and central temperature approaching 2200°F. High strength zirconium alloy, developed especially for sodium service, would be used as graphite cladding. An increased heat transfer surface would be installed in the radioactive-to-nonradioactive sodium heat exchanger, and the best dissipation capacity of the reactor system would be increased by combining an air blast heat exchanger with an existing turbo-generator installation.

Completion of these proposed modifications would make the SRE essentially a new experimental reactor capable of testing design conditions, fuels, and materials for advanced sodium graphite reactor systems.
AETR Program Attaining Major Objectives

For Southwest Atomic Energy Associates

Considerable progress has been made in the Advanced Epithermal Thorium Reactor (AETR) program for the Southwest Atomic Energy Associates since the epithermal critical assembly safely attained its first criticality on December 13, 1960.

Changeovers from the first to the second, third, and fourth cores have now been made. Nuclear analysis is more clearly defining the optimum energy spectrum which will be sought, and advantageous for sodium-cooled, Th-U fuelled reactors. More complete nuclear data are being developed in the Epithermal Critical Assembly (ECA) at Argonne National Laboratory.

The AETR development program is being carried out by Atomic International for the Southwest Atomic Energy Associates (SAEA). A group of 15 investor-owned utility companies in the southwestern part of the United States. The program comprises a 3-year period of research and development at an estimated cost of $5.3 million which is being financed entirely by SAEA.

First Two Phases

Two phases to the program have been authorized. Phase I, completed in June 1959, included the conceptual design and analytical studies of possible core and blanket configurations, engineering problems, performance requirements, and the economical feasibility of the concept.

Operations of the critical assembly machine comprised the major part of Phase II. The assembly machine is housed in the epithermal critical experiments laboratory in Atomic International's Bettendorf test area. The first core duplicated a ZPR-II critical assembly core at Argonne National Laboratory, and was used to check the usefulness and validity of AETR multi-region critical assembly and theoretical calculations.

Second Core

The second core was fueled with U-233 and went critical on June 2, 1961. Second core nuclear data compilations were successfully completed after 38 reactor runs in 2 months. The successful use of a core with U-233 marked another milestone in epithermal critical experiments. This was the first core with a test region similar to full-scale reactor compositions and the first using U-233. This material will ultimately be the fuel for an AETR. Successful operation with it was one of the goals of Phase II of the AETR program.

The change from the first to the second core involved only the test regions at the center of the assembly. Plates of U-233 clad with welded, air-tight stainless steel replaced the U-235 first core. Critical assembly drawings were placed in a shielded cask and the 2 x 2 x 4-inch plates of U-233 were remotely inserted. The addition of thorium, a fertile material, and carbon (as the moderator) was another distinguishing factor of the second core.

Third Core

A third core, using U-235, was installed on July 28, 1961. The U-235 plates of the second core were removed remotely for the substitution of the U-235 plates. Each plate had 29 grams of U-235 metal enclosed between aluminum covers. Measurements made on the second core were repeated on the third core to study the correlation of fuel performance. For the third core, the compositions of both the second and third cores, almost twice as many, were U-233 or U-235 as needed to go critical. Criticality of the third core was predicted within 0.01%.

Experiments on the third core have been completed and results have verified the theoretical calculations.

Fourth and Fifth Cores

The fourth core was installed on September 22, 1961, and was moderated with 50 volume percent beryllium. The core had a very low thermal energy of 10 watts to establish the lower limit to the energy range which is being considered in AETR studies. Beryllium shows promise because of its good nuclear mechanical properties.

Loading of the fifth core was started on November 6, 1961. It contained a larger test region than previous cores and will be used in studying graphite moderated AETR systems. The next series of three or four cores will have the larger test region.

Continuing Programs

During the coming months, effort will be concentrated on the following programs. The critical experiments will yield results on several critical assembly configurations containing various amounts of graphite or beryllium, and could then be used to evaluate better known [12] systems. As improved information becomes available from initial experiments and other sources, corrections will be made in cross section data and calculational methods. Nuclear analysis of full-scale cores and studies of control and safety will establish bases for core design. Fuel developmental emphasis will be on graphite and other ceramic type fuels. Samples of fuel designs will be fabricated to study processing problems and material quality. Compatibility of fuel materials will be observed under laboratory conditions. To reduce fuel cycle costs, fuel element designs will be evaluated for ease of reprocessing and refabrication. Heat transfer studies and economic optimization of the full-scale system will be based on the results of the analytic studies of burnup and control, fuel cycle economies, and the experimental physics. Information obtained from all portions of the project will be integrated into a preliminary design of a full-scale AETR. Design will cover the core, external system, and reactor control and instrumentation.

500 Mwe Plant

Conceptual plant design and parametric studies are being developed for a 500 Mw electric generating power station based on a sodium cooled, Th-U fuel cycle nuclear energy source. Modern steam conditions of 1100°F heat, 1000°F reheat, and 2300 psi with an overall plant efficiency of 42% are feasible. High plant utilization factors result from reliably developed plant components, and core fuel conversion. High plant capital and operating costs are counteracted by lower fuel costs and good plant utilization.

AETR Potential

The Advanced Epithermal Thorium Reactor nuclear-electric plant will be economically competitive with fossil fuel power stations by 1972. Capital costs of $150 million/kw and operating and maintenance costs of 0.4 mills/kw are projected, based on experience and costs previously offered. Fuel costs could approach 1.0 mills/kw by 1972 if development is started soon. Thus, total cost of generating energy should be about 4.7 to 5.3 mills/kw. This would be economically competitive with highly efficient fossil-fuel plants in many areas of the United States. Over a thirty-year life the total energy cost for nuclear stations might be considerably less than for fossil-fuel plants if fossil fuel costs continue to rise as they have in the past and expected improvements in processing and reutilization result in lower nuclear fuel costs.

SOUTHWEST ATOMIC ENERGY ASSOCIATES is made up of 15 investor-owned electric utility companies in Louisiana, Arkansas, Mississippi, Montana, Idaho, and Texas. The group are the principal suppliers of electric power in the states they serve, and represent a total installed electrical capacity of 6,250,000 kilowatts.

TRUSTEES AND BOARD MEMBERS of SAEA included AF facilities during their annual meeting held last summer and the significant progress of the AETR program was reviewed. The program comprises a 5-year period of research and development at an estimated cost of $5.3 million which is being financed entirely by SAEA.

FIRST U-233 FUELED CORE GOES CRITICAL-Use of U-233 fuel in the second core of the epithermal assembly on June 2, 1961, marked another milestone in epithermal critical experiments. Successful operation with U-233 was one of the goals of Phase II of the AETR program.

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"Space Flight Report to the Nation" Features AI's Compact Reactor Models

At the "Space Flight Report to the Nation" meeting held at the Coliseum in New York City by the American Rocket Society, October 9-15, 1961, three SNAP (Systems for Nuclear Auxiliary Power) reactor models were featured at the Atomics International exhibit. The SNAP 2 display was a full-scale, 13-foot high model, with its interior animated by special lighting techniques to simulate liquid NaK. The SNAP 8 display was a one-third scale model of the SNAP 10A space power system. In addition, a motion picture on SNAP, titled "Nuclear Reactors for Space," and produced by Atomics International for the United States Atomic Energy Commission, was shown at the display booth.

**Space Reactor Power**
A paper titled "Space Reactor Power," co-authored by J. R. Wetch and M. G. Coombs of Atomics International, was presented as part of the "Space Flight Report to the Nation." The paper outlined the work being done on SNAP by Atomics International for the U.S. Atomic Energy Commission, the objectives of the SNAP program, the current systems under development, the probable time schedule on the availability and various applications of these systems, and the feasibility of some of the higher power, more advanced systems under consideration for development.

**SNAP Program Objectives**
The basic reactor program now consists of three systems: development projects, each leading to flight tests programmed by the Department of Defense and the National Aeronautics and Space Administration (NASA). These three systems range in electrical power outputs from 500 watts to 30 kilowatts.

The basic objectives of the SNAP program are to produce nuclear systems with these advantages:

a) Minimum size and weight per energy (kw-hr) output
b) Unaffected by environment (sun-shade-orientation-vacuum-radiation-vibration-shock, etc.)
c) Unattended operation, remote startup and shutdown

d) Complete public safety

e) Extreme reliability and endurance
f) Maximum flexibility towards various vehicles, missions and power outputs
g) Economically attractive for intended applications
h) Development and qualification time scale compatible with requirements.

The application of all these criteria has resulted in a reactor powered range from a few watts to a few hundred watts to radioisotope; the power range from several hundred watts to 100 kilograms to the liquid NaK-coupled metal hydride reactors; and the power range from several hundred watts to several megawatts to the fast, high-temperature ceramic reactors.

**Current Systems**
The three SNAP reactor systems currently being developed in the space program for the U.S. Atomic Energy Commission by Atomics International are the SNAP 10A, SNAP 2 and SNAP 8.

**SNAP 10A**
The initiator reactor in the space system, is a compact, metal hydride fueled and moderated, temperature coefficient controlled nuclear reactor. It is coupled to a regenerative conversion system, cooled with liquid NaK, and utilizes thermionic electric power conversion to avoid the use of moving parts for operation. The system will produce 500 watts of electricity and weigh about 525 lbs unshielded.

**SNAP 2**
The second space reactor power system, uses this same reactor system, but utilizes thermionic power conversion which allows smaller weight and higher temperature operation by the addition of a high temperature control system. Also, the use of a regenerative conversion system is eliminated. The SNAP 2 reactor will be capable of delivering up to 100 kilowatts to a system producing a total of 60 Kilowatts. The unshielded SNAP 2 reactor will weigh about 1000 lbs. and the 60-kilowatt system about 2200 lbs. SNAP 2 is currently scheduled to be flight-tested by the National Aeronautics and Space Administration in 1965.

**SNAP 8**
The SNAP 8, the third space reactor power system, will use a scaled-up reactor similar to the SNAP 2 and SNAP 2 reactors. This is a 30-kilowatt conversion system, but the reactor is capable of driving two 30-kilowatt conversion systems to produce a total of 60 Kilowatts. The unshielded SNAP 8 reactor will weigh about 1000 lbs. and the 60-kilowatt system about 2200 lbs. SNAP 8 is currently scheduled to be flight-tested by the National Aeronautics and Space Administration in 1965.

**Advanced Systems**
When early SNAP 2 satellite experiments and SNAP B flights have proved the feasibility of electric propulsion, higher power reactor-electronic systems in the 300 kilowatt and 1 megawatt power range should immediately be developed. To be useful in the space power plant weight of these electric propulsion systems should be reduced below 10 lbs per kilowatt of electric output. Studies conducted to date indicate that there are two conceptual concepts which appear promising for this power range.

One is a 2000°F fast reactor cooled with a liquid alkali metal, such as sodium or lithium, which is currently being considered in the SNAP 2 and SNAP 8 systems. The second is the thermionic reactor. In this system, the electric power output system, powered by a thermionic core by electron emission from a fuel element cathode surface. This approach would not require moving parts and offers the potential of system weights as low as 3 to 5 lbs. per kilowatt. The reactor fuel must operate between 2000 and 3500°F. Ceramic electrolyte fuel mixtures appear to be the most promising fuel material for this concept.

The development of either the thermionic or the high temperature Rankine system will depend upon extensive fuel and materials research and development. Overall technology is currently better understood for the 2000°F turbogenerator system. However, the rapid progress in thermionic conversion technology has in greater potential for space applications may favor that system.

**Availability of High Power Systems**
Both Department of Defense and the National Aeronautics and Space Administration have initiated research and development work in the high temperature turbogenerator power conversion techniques, and the U.S. AEC is conducting an extensive thermionic conversion research program.

When the basic feasibility problems in high power space nuclear electric generating systems are established, and then the general characteristics and specifications of high power systems and space vehicles can be determined, then either the turbogenerator or the thermionic system, or possibly both, will be technically and economically developed. It is expected that either system might be available by 1970.

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“Atoms at Work” Exhibit in Lebanon Features L-77 Reactor in Operation

The U.S. Atomic Energy Commission’s “Atoms at Work” exhibit at Beirut, Lebanon, from October 8 to November 8, 1961, featured an operating L-77 Laboratory Reactor designed and built by Atomics International. The reactor was the second time an L-77 had actually operated before the general public at a Commission exhibit. Previously, in September 1958, the U.S. AEC’s exhibit in Geneva, Switzerland, had also featured an operating L-77.

The Beirut exhibit was organized in cooperation with the Lebanese Atomic Energy Commission, and was the first exhibit of this kind to be held in Lebanon. Its purpose was to demonstrate the increasingly important role nuclear energy is playing in our everyday lives, how it is serving as a working laboratory for scientists and engineers, and as a training school for students.

The exhibit was housed in a specially designed, transportable, steel and aluminum pavilion. A space-frame covered with white nylon and suspended by cables from a 116-foot aluminum mast in the center, formed the roof. The aluminum pavilion had previously housed U.S. AEC exhibits in Geneva (twice), Rome, Tokyo, New Delhi, Cairo, Lahore, Buenos Aires, and Rio de Janeiro. Lebanon was the 10th occasion the exhibit was displayed.

The U.S. Atomic Energy Commission will again feature an operating L-77 reactor at upcoming exhibits which are being planned for the spring and fall of 1962 at Athens, Greece, and Bangkok, Thailand.

Use of L-77 at Exhibit

During the Beirut exhibit many Lebanese scientists and technicians were trained in the operation of the L-77 reactor and received certificates to show they had taken part in the training program.

Several experiments were also performed during the period the exhibit was held. The purpose of these experiments was to observe the effects of electrons on red blood cells. Neutrons produced by the L-77 and rays from the gamma facility, another feature of the exhibit, were used to perform several tests for comparison studies.

L-77 Installations

Three other L-77 reactors are currently in operation and are being used for various purposes. The first of these has operated in Atomics International’s facilities at Canoga Park, California, since May 1958. It has been used extensively for specific experiments in the company’s research and development programs.

A second L-77 has operated for the University of Wyoming at Laramie, Wyoming, since March 1959. This reactor is being used in graduate study programs in reactor engineering and technology, radiation chemistry, and reactor physics.

The third L-77 was installed in August 1959 in the Puerto Rico Nuclear Center established by the United States Atomic Energy Commission at Mayaguez, Puerto Rico. The reactor is used for instruction in the characteristics and operation of nuclear reactors, and in the application of radioactivity.

Low-Power Unit

Designed early in 1955 by John W. Flora of Atomics International and protected by United States patent number 2,951,127, the L-77 is a 10-watt solution-type research reactor system specially suitable for college and research laboratories.

The complete L-77 consists of a reactor assembly unit and a desk-type console which can be shipped assembled and ready for installation. The entire unit can be accommodated in a 20 x 20-foot floor area in any typical college laboratory or research institution. The reactor is approximately 8 feet in diameter by 7 feet in height. A water source to fill the shield tank and conventional electric power are the main requirements.

Easy to Operate

The L-77 is simple to operate and only requires one operator. All controls and instruments are centralized at the console and conveniently arranged for observation and use. It is one of the safest reactors built. The large negative temperature and bubble coefficients of reactivity allow the reactor to shut itself off if the power level rises above prescribed limits.

The shutdown mechanism functions even if the mechanical safety rods fail to operate. Enriched uranium in a water solution is used as fuel. Distilled water serves both as solvent for the fuel and as the moderator. At its design power level of 10 watts, a peak thermal neutron flux of slightly more than 2 x 10^12 n/cm^2/sec is available for sample exposures.

Other Solution-Type Reactors

In addition to the L-77, two other solution-type reactors are manufactured by Atomics International. These are the 50-kilowatt L-54, and the 1500-watt L-55. Reactors of these two types are currently operating at the Armour Research Foundation in Chicago; The Japanese Atomic Energy Research Institute at Tokai; the Johann Wolfgang Goethe University in Frankfurt; the Nuclear Research Institute at Tokai; the Johann Wolfgang Goethe University in Frankfurt; the Nuclear Research Institute at Tokai; the Danish Atomic Energy Commission center at Risoe, Denmark. Another is being installed for the Walter Reed Army Medical Center in Washington, D.C.

L-54 in FRANKFURT—In addition to the L-77, Atomics International manufactures the 50-kilowatt L-54, and the 1500-watt L-55 solution type reactors. These two types are operating in Chicago, Tokai, Japan; Frankfurt, Germany; Moscow, U.S.S.R.; and Black Balsam, Louisiana; and the Danish Atomic Energy Commission center at Risoe, Denmark. Another is being installed for the Walter Reed Army Medical Center in Washington, D.C.

“Atoms at Work” Exhibit in Lebanon