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1000-MWE SGR
AND PROTOTYPE EVALUATION STUDY
VOLUME I
LARGE SGR DESIGN BASES, ECONOMICS,
AND DEVELOPMENT PROGRAM

AEC Research and Development Report

ATOMICS INTERNATIONAL

A DIVISION OF NORTH AMERICAN AVIATION, INC.

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ATOMICS INTERNATIONAL

A DIVISION OF NORTH AMERICAN AVIATION, INC.
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FOREWORD

Technological developments in the power industry point to the installation of larger units (400,000 kw and above) in utility systems. The sodium cooled, graphite moderated reactor (SGR) is most attractive at these ratings. This document, the first of two volumes prepared for the United States Atomic Energy Commission, discusses the sodium graphite reactor concept as applied to large central power stations. Volume I presents the design bases, economics, conceptual design, and development program associated with a 1000-Mwe SGR power plant. As an initial step towards the large SGR plant, a prototype SGR of 200-Mwe capacity is envisioned with intent to start in 1964. The design bases and development program for the prototype plant are also covered in Volume I. In Volume II are contained the preliminary design, description, and drawings of the 200-Mwe Prototype Plant.

These plants will be among the most efficient central power stations in existence. The SGR features the highest degree of inherent safety of any nuclear reactor concept and promises unmatched operational reliability and economics. Volumes I and II therefore cover the long range and immediate aspects of the SGR as a way to attain efficient and competitive nuclear power.

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I. INTRODUCTION AND SUMMARY

Continued national economic growth is perhaps best forecast by demands for a basic commodity — power. The electrical utility industry continued to grow by doubling capacity every 10 yr. In its ever-increasing search for low cost power, technological developments such as regional interties, extra-high-voltage (EHV) transmission, and nuclear power are becoming an integral part of utility planning. Interconnected system grids coupled with EHV transport of large blocks of power are bringing about a reorientation of utility philosophy. Planning is being carried out in terms of regional needs rather than the needs of a particular system. As a result, power generating facilities are becoming larger. Preliminary information from the forthcoming National Power Survey confirms that trends toward unit size of 500 Mwe and larger must be considered for future power expansion programs. A major turbine manufacturer has stated that half of the new machines now on order are 350 Mw or larger; in 1955, the average size was 150 Mw. This same manufacturer predicts units of 1200 Mw within 3 yr.

These developments present an optimistic picture for nuclear power and in particular for the SGR which is most attractive in large plant sizes. The low operating reactor pressure, essentially atmospheric, obviates the need of thick-walled pressure vessels; the largest proposed SGR vessels are well within the fabrication capabilities of American industry. The SGR is capable of providing modern steam conditions, either subcritical or supercritical. The limitation on steam pressure and temperature for the SGR is the same as for fossil-fuel fire plants, namely development of materials for advanced steam conditions. If materials are made available which permit an increase in steam pressure and/or temperature, the SGR will show the same thermodynamic gains experienced by fossil fuel-fired plants. In this respect, the 1000-Mwe SGR and the 200-Mwe prototype, discussed in this report, offer the highest thermal efficiencies of any power plant currently in operation or under construction. The very best fossil-fueled plant with steam at 3500 psig and two stages of reheat has a net plant efficiency of just over 40%. The large SGR offers an efficiency of 43.6%, the prototype 41.8%.

The SGR has inherent safety characteristics. One of the most outstanding characteristic is that molten sodium absorbs iodine quantitatively. This

characteristic should "substantially ease the siting problem by minimizing the dispersion of radioactive matter in case of a reactor accident".* Other safety aspects are as follows.

- a) The reactor, constructed of materials which provide reliable radioactivity containment, operates at low pressure. The inert gas cover over the reactor is maintained at a pressure essentially atmospheric. The highest pressure in the primary system is about 50 psig at the primary coolant pump discharge. Reactor vessel pressurization from chemical energy releases cannot occur in this type of reactor.
- b) The heat transfer properties of sodium permit high heat fluxes well below the burnout heat flux.
- c) A prompt negative temperature coefficient of reactivity provides the reactor with a self-limiting feature which tends to prevent a nuclear excursion. The reactor is also guarded by the plant protective system, designed for high availability without compromising safety.
- d) The difference between the operating temperature (1150°F) and the boiling point of sodium (1620°F at atmospheric pressure) provides a large thermal storage capacity which prevents bulk coolant boiling within the reactor vessel, even if unexpected surges in reactor power should occur.

In addition to the technical attributes of the SGR, the economic aspects of this concept are highly favorable. The estimated cost of the 1000-Mwe SGR represents an investment of \$141/kw, including customer costs. At 14.5% fixed charges and 80% plant factor, total power generation costs of 4.60 mill/NKwh, including fuel cycle costs of 17.1×10^6 Btu, or 1.34 mill/NKwh, are expected.

To reach the economic goals attainable with a 1000-Mwe SGR, demonstrate the reliability of large SGR systems, and establish the confidence of the utility industry in liquid metal technology, the following 2-step program is envisioned.

- a) To initiate in 1964 the construction of a 200-Mwe prototype plant, incorporating the larger plant design features basic to the improved

*Civilian Nuclear Power, A Report to the President, AEC (November 19, 1962)

SGR concept, and which require large scale demonstration of component performance and reliability.

- b) To continue development programs in the area of fuel element, control element, steam generator, and reactor physics, to meet the more advanced conditions of the 1000-Mwe plant. A realistic target date to initiate design of the 1000-Mwe plant is 1969.

This volume, the first of two, describes the general features of the SGR, presents a conceptual design of the 1000-Mwe plant, relates the large plant features to those of the prototype, and discusses economics of the large plant and the R & D programs required for both plants. The preliminary design of the 200-Mwe SGR prototype is detailed in Volume II.

The large SGR power plant complex consists of a reactor building, a steam generator structure, a turbine generator building, and other operating and service facilities. The reactor, fuel-handling system, radioactive sodium heat transfer, and service systems are contained within the reactor building structure. Adjacent to this building are housed the nonradioactive sodium systems and steam generator facilities, and adjoining this structure is the turbine generator plant.

The reactor is sodium cooled, graphite moderated and uranium carbide fueled. Liquid metal sodium passes through the reactor where it is heated to 1150°F. This is illustrated in the elementary flow diagram, Figure 1. This high temperature sodium is pumped through a shell-and-tube sodium-to-sodium heat exchanger, transferring thermal energy to a secondary sodium loop. Primary sodium is returned to the reactor at 750°F. Since the primary loop is radioactive, all portions of the loop are contained in inert-gas filled, steel-lined cells. Secondary sodium is nonradioactive and, as such, the routing of lines and location of equipment in the secondary loop is treated as a conventional plant layout design configuration.

Secondary sodium at 1060°F transfers heat to the steam generators and is returned to the shell side of the IHX at 660°F. A combination primary and secondary sodium loop form one heat-transfer circuit. The 1000-Mwe plant consists of five heat transfer circuits, each rated at approximately 200 Mwe. The 200-Mwe prototype plant consists of one 90% circuit and a 10% circuit; the latter is for decay heat removal.

Figure 1. Simplified SGR Flow Diagram

Steam is generated at 3500 psig/1000°F and 2400 psig/1000°F for the 1000-Mwe and prototype plants, respectively. In each case, reheat steam temperature is 1000°F. The thermal cycle is conventional in every respect, with modern high-pressure, high-temperature plants.

Modern design features of the prototype and 1000-Mwe plant which differ from SRE or HNPF plants, other than in size, are as follows.

- a) Calandria core, which permits operation at high sodium temperatures, high core power density, and provides improved neutron economy over the canned moderator designs of SRE and HNPF.
- b) Uranium carbide fuel, which is capable of high temperature, high power density operation with extended burnup, leading to low fuel cycle costs.
- c) Once-through, modular steam generator design, which affords low fabrication costs and increased reliability at high steam conditions.

The calandria is a single, large vessel which contains the graphite moderator. Process tubes penetrate through the graphite and are joined to the top and bottom heads of the calandria. These joints, which may be brazed or welded, permit replacement of the thin-wall process tubes and assure the maintainability of the calandria.

The currently preferred nuclear fuel material is hypostoichiometric uranium carbide, clad in stainless steel and assembled into multirod clusters. Experimental results to date indicate that this fuel material, well suited to sodium cooling, will operate at peak temperatures of 1750 to 2000°F with an average fuel exposure of 25,000 MWD/Mtu. Also, UO_2 appears to be suitable, with a small penalty (0.1 to 0.2 mill/kwh) in fuel cycle cost.

The steam generators consist of high-temperature stainless steel superheater modules and low-temperature chrome-moly steel evaporator modules. Stainless steel reheater modules are employed. The modules are of vertical, once-through design with sodium on the shell side and steam or water on the tube side. The utilization of the module concept provides a reliable, easily maintainable, low-cost steam generator. The concept is readily adaptable to subcritical or supercritical steam conditions, and to plants of any size.

Station performance characteristics for the prototype and the reference 1000-Mwe SGR plant are shown in Table 1.

Design of the 200-Mwe prototype plant is based on construction and operating experience with the Sodium Reactor Experiment (SRE) and the Hallam Nuclear Power Facility (HNPF). Major advances from these plants are based on AEC development programs in the areas of fuel, core structure and components, fuel handling, steam generator, sodium pumps and valves, and other related plant components. With few exceptions, the prototype plant development programs are designed to provide the necessary information for the design and construction of the 1000-Mwe plant. Only in the areas of fuel, control rods, steam generator and reactor physics will additional development be required for the larger plant. In these areas, continued development is required because of the somewhat more advanced operating conditions for the 1000-Mwe SGR.

This minimal development effort for the larger plant arises from judicious selection of component configuration and design. Most components in the prototype plant are representative of size requirements for a 1000-Mwe plant, and function under operating conditions similar to those of the large SGR. Where size cannot be economically duplicated in the prototype, such as with the calandria and reactor structure, prototypical design features have been incorporated and, in these cases, the component design configuration has been carefully selected to permit scaleup on sound engineering principles.

The development programs for the two plants have been conceived to permit design and construction of the prototype plant for 1968 operation, and to follow up with the initiation of design for the 1000-Mwe plant in 1969. Approximately \$21,575,000 are required during GFY's 1965, 1966, and 1967 to develop components and design criteria directly related to the 200-Mwe plant; about \$15,385,000 are required during GFY's 1965 through 1969 to develop information leading to the design of a 1000-Mwe SGR. In addition, a general development program, not necessarily related to a specific plant but connected directly to technology employed to verify the technical limits for a safe and efficient SCR, is envisioned. This program will cost \$12,160,000 during GFY's 1965 through 1969. Although these costs do not include the SRE-PEP modifications, it is assumed that this important modification will be supported at a level consistent with current planning.

TABLE 1
DESIGN AND PERFORMANCE CHARACTERISTICS FOR THE
200-MWE AND 1000-MWE PLANTS

Plant	200 Mwe		1000 Mwe
Net plant electrical output (kw)	198,700		1,019,100
Net plant heat rate (Btu/kwh)	8,160		7,824
Net plant efficiency (%)	41.8		43.6
Steam conditions			
Main steam pressure (psig)	2400		3500
Main steam temp. (° FTT)	1000		1000
Reheat steam temp. (° FTT)	1000		1000
Turbine Generator			
Type of turbine-generator	TCDF-30 in.-3600 rpm		CC4F-43 in.-3600/1800 rpm
Rated gross elec. output (kw)	210,049		1,050,600
Main steam flow (lb/hr)	1,351,700		7,236,700
Reheat steam flow (lb/hr)	1,183,800		5,092,300
Condenser pressure (in. Hg abs.)	1.5		1.5
Number of feedwater heaters	7		8
Final feedwater temp. (° F)	473		512
Nuclear Steam Generating System			
Reactor			
Core thermal power (Mwt)	480		2,350
Fuel	uranium carbide		uranium carbide
Active core size (ft:dia.xht)	13.5 x 14		25.1 x 14
Number of fuel elements	168		595
Core loading (kg U)	20,830		73,800
Enrichment (wt% U ²³⁵)	3.80		3.60
Average fuel burnup (Mwd/MTU)	25,000		25,000
Average specific power (kwt/kg U)	23		32
Lattice spacing (in. ²)	10.25		10.25
Number of control elements	24		84
Fuel elements			
Geometry	18-rod cluster		18-rod cluster
Fuel slug diameter (in.)	0.50		0.50
Active length (ft)	14		14
Maximum hot-channel fuel temp. (° F)	1750		2000
Maximum fuel slug power (kw/ft)	24		33
Sodium heat transfer system	Main Circuit	Auxiliary Circuit	Per Circuit
Thermal capacity (Mwt)	432	48	470
Number of primary loops	1	1	5
Number of secondary loops	1	1	5
Number of intermediate heat exchangers	2	1	2
Number of steam generator banks	2	1	1
Primary Loop			
Sodium temp. (° F)			
Reactor inlet	750	750	750
Reactor outlet	1150	1150	1150
Sodium flowrate (lb/hr)	12.2 x 10 ⁶	1.35 x 10 ⁶	13.2 x 10 ⁶
Secondary Loop			
Sodium temp. (° F)			
Steam generator inlet	1075	1075	1060
Steam generator outlet	675	675	660
Sodium flowrate (lb/hr)	12.2 x 10 ⁷	1.35 x 10 ⁶	13.2 x 10 ⁶

The 1000-Mwe SGR plant, described in this report, is a high performance nuclear power station that can fulfill the need for low cost power in the near future. In addition to serving this useful function in our growing economy, the sodium graphite reactor utilizes the liquid sodium technology necessary for fast breeder reactors. Hence, the SGR has strong future implications and lends itself to orderly near-term and long-term nuclear power development.

II. LARGE SGR DESIGN BASES

The selection of design parameters, components, reactor internals, fuel and control elements for the prototype and 1000-Mwe SGR plants has been the result of extensive engineering studies and tests. A review of the many aspects of the bases for the SGR design is covered in this section. The approaches discussed herein have led to the reference design of the 1000-Mwe plant, Section IV, Volume I, and to the preliminary design of the 200-Mwe prototype plant, Volume II. A continuing R & D program will serve to confirm the analytical studies, remove areas of uncertainty, and establish the SGR as a firm competitive producer of power.

A. UNIT SIZE AND CONFIGURATION OF SGR PLANT

Based on past performance and current trends in the electrical utility industry toward larger blocks of power to provide lower total energy cost, the reference large SGR power station was selected to be 1000 Mwe. In terms of technical and economic feasibility, the plant could be larger or smaller. The 1000-Mwe capacity is representative of a plant size required for the early 1970's.

In selecting the configuration for the 1000-Mwe SGR, a study was made of the number of heat-transfer circuits vs capital cost and plant availability. This study indicated that the cost differentials were small in the 3-, 4-, and 5-circuit region, and that an optimum is probably 4 circuits. More detailed results from this study are included in the Appendix, Section A. A 5-circuit 1000-Mwe SGR was selected as the reference design for the large SGR in view of component size scaleup from current HNPF components. For example, the rated capacity of the HNPF sodium pump is 7200 gpm. In conventional plants, scale factors of 3 to 5 are not uncommon but larger factors are generally not prevalent in industry. Hence, the maximum pump capacity for the 1000-Mwe plant was selected to be approximately 35,000 gpm, which indicated the 5-circuit plant would be most practical at this stage of development. Other parallel examples supported the 5-circuit selection.

In establishing the prototype plant capacity, consideration was given to component sizes and capacities representative of those in the 1000-Mwe plant. In addition, economic aspects of the smaller plant were reviewed. To this end,

the prototype plant of nominal 200-Mwe rating was considered to have one main heat transfer circuit, approaching the capacity of one circuit in the 1000-Mwe plant, plus a small auxiliary heat transfer circuit as a backup for reactor decay heat removal.

The exact rating of the plant was based on the selected turbine performance. The 200-Mwe turbine selected for 2400 psi/1000°F/1000°F steam conditions was a tandem compound, double flow - 30-in. last stage blading machine. This machine has a rated gross capability of 210,000 kw and an ultimate capability of approximately 221,000 kw. To drive this machine, a heat source of approximately 480 Mwt (rated) and 505 Mwt (ultimate) is required. In the prototype plant, the main power circuit is designed for 90% capability and the auxiliary circuit for 10% capability. On this basis the auxiliary circuit is nominally rated at 48 Mwt and the main circuit is rated at 432 Mwt, with an ultimate capability of 457 Mwt to match that of the turbine. The capability of the main prototype circuit is within 10% of that desired for one of the reference loops in the 1000-Mwe design and is within 5% when operated at ultimate capacity. These differences are small scale factors and would not radically affect the design of the major heat transfer system components.

A 200,000-kw plant will enable components in the main heat transfer system to have the characteristics desired for prototyping a 1000-Mwe plant at a minimum plant cost. Other systems in the 200,000-kw plant such as the sodium service systems, inert gas systems would demonstrate requirements and characteristics commensurate with those predicted for the future 1000-Mwe SGR. These systems are less dependent on capacity or output of the plant.

B. OPERATING CONDITIONS - STEAM AND SODIUM

Operating conditions for throttle and reheat steam and sodium temperatures have been established for large SGR systems through optimization studies, which evaluate and compare major sodium and steam system equipment first costs and plant operating costs. Earlier studies, prepared for a 360-Mwe SGR plant, provided a base for the initial design of large SGR plants. Details of the parametric studies are presented in the Appendix, Section B.

Based upon these studies, the following process conditions were selected for the 1000-Mwe plant and the 200-Mwe prototype.

	<u>1000-Mwe Plant</u>	<u>200-Mwe Plant</u>
Steam conditions (psig/°F/°F)	3500/1000/1000	2400/1000/1000
Final feedwater temp. (°F)	512	473
Reactor sodium outlet temp. (°F)	1150	1150
Reactor sodium inlet temp. (°F)	750	750
Intermediate heat exchanger log mean temp. difference (°F)	90	75

The selection of supercritical steam conditions of 3500 psig/1000°F/1000°F for the 1000-Mwe SGR plant is based on current trends in the industry, economics, and anticipated time schedule for construction of large plants. The use of supercritical steam cycles for power plants is becoming more common in the utility industry as technology in this field progresses. This is particularly true for large power stations in the range of 400 to 1000 Mwe. The turbine cost break point between 2400-psig and 3500-psig pressure is approximately 400 Mwe. The 3500 psig units become progressively cheaper than the 2400 psig units as machine size increases above 400 Mwe. The economics for a 1000-Mwe plant, as discussed in the Appendix, Section B are decidedly in favor of the 3500-psig conditions.

Steam conditions of 2400 psig/1000°F/1000°F were selected for the 200-Mwe prototype, based on industry practice for this size unit and economic evaluation. Standard supercritical machines are manufactured for ratings of 400,000 kw and above. In the 200-Mwe range, the gain in efficiency at 3500-psig vs 2400-psig operation is more than offset by the increased capital expenditure of the supercritical plant. As important as the economic considerations are, the availability of a subcritical steam generator test facility also has bearing on the selection of 2400-psig throttle pressure. This facility, the Sodium Component Test Installation (SCTI), is capable of testing steam generators of the type proposed for the prototype. The steam generator test program is discussed in more detail in Section V.

The use of the SCTI will be one of the factors that permit meeting the objective of a 1968 operational prototype SGR. The construction of an advanced SGR approaching the 1000-Mwe range is contemplated after successful operation of a prototype plant. Based on this assumption, the design of such a plant would be initiated in 1969. At this time, not only will the SGR concept have been

successfully demonstrated, but in addition the technology of supercritical steam generation will have been further developed by the large utility companies. These companies are also the potential customers for the large SGR. The step to supercritical steam generation using an SGR will then be a modest one, requiring only the proof-testing of a supercritical steam generator. Other heat transfer components will have operational histories to provide design criteria.

Although 1000°F steam temperature was selected for the reference design of both the prototype and 1000-Mwe plant, economic parametric studies in Section B of the Appendix indicate that 950°F could also be used efficiently. The higher steam temperatures were selected at this time to reflect the capability of the SGR system and the confidence to design to this higher condition. Final prototype steam temperatures may well be decided by economics associated with a particular utility's financing and operating conditions. The sodium process conditions in the prototype and 1000-Mwe plant are similar.

The maximum reactor sodium outlet temperature was fixed at 1150°F. Higher sodium temperatures were not considered because of structural uncertainties associated at this time with the fuel element, calandria, and reactor structure. The current development program which is investigating core structural creep problems at elevated temperatures, operation of the SRE-PEP at 1200°F, and the prototype experience may lead to higher sodium temperatures in the future SGR plants. Higher sodium temperatures would result in the consideration of higher steam temperatures, leading to improved plant cycle efficiency; or with the same steam temperatures considerable savings in heat transfer surface costs can be realized with higher sodium temperatures. However, the prototype and 1000-Mwe plants have been limited to the more conservative reactor sodium temperature of 1150°F.

The reactor inlet temperature of 750°F was set by parametric studies which indicate an optimum ΔT across the reactor of 400°F (Appendix, Section B). A slight difference in main secondary sodium temperatures (15°F) exists between the two plants, because of the selected log-mean temperature difference for the intermediate heat exchangers. An LMTD of 90°F appears more economical for the supercritical steam conditions, because of the reduced steam generator surface requirements, whereas an LMTD of 75°F is more economical for the subcritical plant. Sodium flow rate in each main heat transfer loop is 33,000 gpm

in the 1000-Mwe plant and 30,700 gpm in the prototype. The difference arises from the auxiliary power loop in the prototype which removes 10% of the plant power. These slight differences in sodium temperature and flowrates are not considered significant.

C. SODIUM HEAT TRANSFER SYSTEM COMPONENTS

1. Steam Generators

An important aspect of the large SGR concept is the development of reliable and economical sodium-heated steam generators for high quality steam (above 900 psi/900°F).

Desirable design criteria for sodium heated steam generators may be outlined as follows.

- a) Detail design features must minimize the possibility of caustic and chloride stress corrosion.
- b) Units should allow the generation of 950 to 1050°F steam with the lowest possible cost consistent with reliable performance.
- c) The unit design should be adaptable to high pressures to exploit the associated lower plant heat rates.
- d) The unit design should allow for flexibility of operation and ease in accessibility and maintenance.
- e) The unit design should allow fabrication and installation schedules for the steam generators which will not govern the plant construction schedule.

Existing steam generators do not lend themselves to economic configurations at the prototype and 1000-Mwe operating conditions. Caustic and chloride corrosion problems have been circumvented previously by the use of expensive noncorrosive linings, double-wall-tube construction and the reduction of steam temperatures to levels where lower strength alloys, which are not subject to the corrosion problem, may be used. These solutions are exemplified by the following considerations.

The Hallam Nuclear Power Facility generates 825 psig/833°F steam with 2-1/4 chrome - 1 moly and 5 chrome - 1/2 moly steam

generators of the drum evaporator-superheater concept, double-wall construction with helium as the third fluid. This design has been successful; however, the use of drum evaporators is generally limited to the lower steam pressure applications and the design is expensive. Future steam generators developing higher steam pressures will tend towards the more advanced once-through concepts.

The Sodium Reactor Experiment generating 625 psig/825°F steam utilizes 304 stainless steel and once-through, double-wall construction with mercury as the third fluid. Successful operation was obtained once a water-flow maldistribution problem due to horizontal mounting was overcome.

The Los Alamos Scientific Laboratory 2000-kw steam generator produced 900 psig/925°F steam with a chrome-moly evaporator and stainless steel super-heater of the single wall, once-through design. Initially some instability was observed when the unit (horizontally mounted) was operated at less than 50% of the design load. Installation of orifices at the evaporator inlet tubes improved the operating characteristics and the experiment was terminated as planned after 9500 hr of operation.

The Enrico Fermi plant generating 900 psig/780°F steam utilizes 2-1/4 - chrome - 1 moly with the once-through concept and single wall construction. Some problems have been encountered during startup and successful full power demonstration has yet to be achieved.

The AEC has long recognized that the development of steam generators involves testing of various concepts in order to fully realize the potential of the high pressure, high temperature steam cycle associated with the SGR. Accordingly the Sodium Component Test Installation (SCTI) was constructed at the AI Field Laboratory. The SCTI is now capable of testing steam generator units up to 2400 psig/1050°F steam conditions. Performance data can be obtained for units up to 35-Mwt capacity. Thermal transients can be imposed on the unit with sodium temperature transients as rapid as 30°F/sec. Several subcritical steam generator units are scheduled for testing; the first unit will start tests in July 1964.

The steam generator concepts, all of the once-through design, scheduled for testing at the SCTI include the following.

- a) Alco Products — this exchanger incorporates an evaporator and superheater in one shell and is designed for the generation of 2200 psi/1050°F steam. The exchanger is rated at 35 Mwt. Materials are 316 stainless steel for the shell and tubes. In addition, the tubes are provided with an Inconel overlay on the water side. Fabrication of the duplex tubes has proven difficult and several delays have occurred in the construction schedule.
- b) Babcock & Wilcox Co. — this steam generator combines the evaporator and superheater in one shell and is rated at 30 Mwt. The superheater section is 316 stainless steel and evaporator section, 2-1/4 chrome. Design is currently under way on this company's concept.
- c) Atomics International — AI's approach incorporates separate modules for the evaporator and superheater. Details of construction are covered below.

The R & D effort for testing of steam generators is covered in Section V.

For the prototype and 1000-Mwe plants, the AI modular concept was selected as the reference design. This does not obviate the use of either the Alco or B & W units if the test results indicate an economic and technical superiority over the modular steam generators. Briefly, the modular concept utilizes a shell-and-tube heat exchanger of the once-through design with sodium on the shell side and water/steam on the tube side. Separate modules, each rated 12 to 15 Mwt, are used for evaporator, superheater, and reheat duties. This concept was selected for the following reasons.

- a) Minimizes possibility of caustic stress corrosion by avoiding crevices, and minimizes the possibility of chloride stress corrosion by the use of ferritic steels in the evaporator region where concentrations of chlorides could build up.
- b) Allows generation of 950 to 1050°F steam with high reliability and low cost by using ferritic materials in the low temperature

evaporator to lower the cost and provide the maximum in corrosion protection and by using austenitic materials in the superheater where high temperature strength is required.

- c) Is adaptable to high pressures because of the relatively thin tube sheets, thus permitting the exploitation of improved plant heat rates.
- d) Allows isolation and removal from service without a major outage because of the small unit size. A hydrogen detector located at the sodium outlet of each module will identify which module is faulty. Its repair, removal, or replacement can be accomplished in the minimum of time.
- e) Allows fabrication and installation schedules for the steam generators which will not govern the plant construction schedule. The modules because of their size can be made by a large number of manufacturers and the production would be in units per month which could be installed in the plant as they become available.

A development program for the AEC is presently under way to design and construct a set of modules for test. The design proposed for the 1000-Mwe and prototype plants employs 61 tubes in a vertical (counterflow) shell-and-tube design with spherical heads. Except for length and material, the superheater and evaporator units are the same.

The low temperature (evaporator) module consists of two tubesheets interconnected by straight tubes butt welded to nipples on the tube sheets. The water or steam is on the tube side with the sodium on the shell side. The use of nipples extending from the tube sheets gives a crevice-free joint and removes the possibility of any caustic corrosion commonly associated with crevices at the tube joint. Support for the tubes is by means of an investment-cast "cookie cutter" design which gives minimum restriction to the sodium flow. The countercurrent flow design allows full utilization of the tube areas with very little if any reduction of heat-transfer coefficient relative to cross-flow concepts. The "cookie cutter" type of tube support is desirable since, in the event of a tube rupture, high acceleration forces are applied to the sodium. The restriction of sodium flow resulting from the use of conventional baffles would

accentuate the pressure rise. The clearances between the tubes and tube supports are according to TEMA, and the clearances would be continually washed with sodium so as to prevent any possible buildup of a caustic solution usually associated with crevices.

The shell of the modules does not use bellows and the tubes are straight. This concept is satisfactory, since the temperature difference between the shell and tubes is about 80°F under steady-state operating conditions. During transients, the difference in response times of the shell and the tube walls does not lead to excessive thermal stress.

The selection of shell thickness is predicted on containing full steam pressure for a short period of time. The shell design in combination with three rupturable membranes (spaced axially along the shell length), designed to rupture at a much lower pressure, provides a conservative and safe design. This conservatism is warranted since there is very little data on sodium-water interaction at these temperatures and pressures. As more data on sodium-water interaction become available it is hoped the shell thickness and the number of rupturable membranes can be reduced.

Sodium entrance and exit nozzles are enlarged and deflector plates are provided around the tubes to avoid impingement problems recently experienced in other sodium-heat exchangers. The design of these deflection plates and the reduction of the sodium velocities entering the tube bundle are in accordance with TEMA specifications.

The design of the high-temperature (superheater) module is essentially the same as that of the low-temperature module (evaporator) except for material changes and the use of one rupturable membrane. The reduction of rupturable membranes to one is predicted on existing data which indicate the sodium-steam interaction is much less severe than that for sodium-water.

Design of the modules for 3500 psi or 2400 psi will be essentially the same, except for length and materials. A 61-tube module design will be used for both the evaporator and superheater. Type 321 austenitic steel will be used in the superheater for both the 1000-Mwe and prototype plants. Type 5 chrome - 2 moly ferritic steel will be used in the prototype evaporator, and 2-1/4 chrome - 1 moly will be used in the 1000-Mwe plant. This material change is

- 1 HIGH TEMP STEAM OUTLET
- 2 REHEAT STEAM OUTLET
- 3 REHEAT STEAM INLET
- 4 FEED WATER INLET
- 5 MAIN SODIUM INLET
- 6 AUXILIARY BANK HIGH TEMP MODULES
- 7 AUXILIARY BANK LOW TEMP MODULES
- 8 MAIN BANK HIGH TEMP MODULES
- 9 MAIN BANK LOW TEMP MODULES
- 10 MAIN BANK REHEAT MODULES
- 11 AUXILIARY PRESSURE RELIEF TANK
- 12 MAIN PRESSURE RELIEF TANK
- 13 MAIN SODIUM OUTLET
- 14 DUMP STACK
- 15 AUXILIARY SODIUM INLET
- 16 AUXILIARY SODIUM OUTLET

Figure 2. Modular Steam Generator

required since a larger fraction of the heat load for a subcritical unit is in the evaporator section resulting in a higher sodium inlet temperature to the evaporator. Higher inlet sodium temperatures accentuate the carbon transport problem with 2-1/4 chrome - 1 moly; therefore, the use of 5 chrome - 1/2 moly is required.

Reheat modules of a larger size are used in both plants because of the reduced reheat steam pressure. Approximately 187 tubes of Type 304 stainless steel are used in the reheater.

The prototype 200-Mwe plant would represent one of five banks of steam generators in the 1000-Mwe plant. The configuration of a bank of modules to form the steam generator is noted in Figure 2. The high temperature (superheater), low temperature (evaporator) and reheater modules for the main sodium circuit are noted. The corresponding modules for the auxiliary sodium circuit of the prototype can also be seen. A detailed description of this steam generator bank is given in Volume II, Section V.

Subcritical steam generation in the 200-Mwe prototype plant would demonstrate modular steam generator operation, which is expected to be less severe in the supercritical plant because of the absence of phase change between water and steam. The higher operating pressure of the supercritical plant primarily affects metal thicknesses.

The operation of a bank of modular steam generators in conjunction with a secondary sodium heat transfer system is envisioned to be essentially the same whether the bank of steam generators is in a 1000-Mwe plant or a 200-Mwe prototype plant. Hot secondary sodium is supplied to both the high temperature modules and the reheat modules. The sodium leaving the reheat modules is reintroduced into the sodium stream leaving the high temperature modules prior to entering the low temperature modules. A flow control valve in the sodium stream leaving the reheat modules regulates sodium flow through these units.

2. Intermediate Heat Exchanger (IHX)

The design bases for sodium-to-sodium intermediate heat exchangers have progressed in recent years to the point where functional operation and design are readily understood. Further experience will be gained through operation of HNPF, EBR-II, Fermi, and through test of the Alco unit in the SCTI.

Figure 3. Typical Intermediate Heat Exchanger

The same basic configuration used in the HNPF plant, a vertical counterflow shell and tube unit, is proposed for the large SGR. Because of increased size, two units in each heat-transfer circuit are to be employed. In this case, each unit transfers approximately 8.8×10^8 BTU/hr thermal power, requiring a 48-in.-diameter tubesheet and approximately 19-ft-long tubes. The same basic units will be used in the prototype heat transfer circuit with a duty reduced by about 10%. A typical IHX is shown in Figure 3.

3. Sodium Pumps

Sodium pumps, although larger in capacity by a factor of 4 to 5 over the HNPF pumps and ~3 times that of the Enrico Fermi plant, are not anticipated to present a problem. A free-surface centrifugal pump, with variable speed control, similar to the HNPF design, will be used; this is illustrated in Figure 4. Increased capacity will be achieved by a larger impeller size and reduced speed. Standard design bases for conventional pumps will be utilized, with modifications for sodium containment and bearing lubrication. The principal uncertainty in pump design is thermal expansion problems during startup and temperature transients. Particular attention must be given to the thermal diffusivity of thick vs thin structural members subjected to sodium temperature changes. The R & D effort for sodium pumps is covered in Section V, Volume I.

4. Valves and Flowmeters

Other components in the main heat transfer system, such as valves and flowmeters, require reexamination of similar components used in HNPF, SRE, and other sodium reactor installations. Because of the large pipe sizes, butterfly shutoff and flow control valves appear to be the most promising design. Across the seat leakage is expected with this valve, as indeed any valve type of the size contemplated, 26 to 28-in. diameter. The system is designed with downstream drains to take care of leakage across the seat during shutdown conditions. A single butterfly valve in the cold leg of each loop will be used as a combination shutoff and flow control valve. Flow control with the valve is performed mainly in the thermal convection range (below 5% flow) during reactor decay heat removal. The checkvalve is expected to be of the usual swing-gate design. Design bases for these valves are fairly well understood; major questions which require further development are: (1) leakage as a function of back

Figure 4. Typical Primary Sodium Pump

pressure after thermal cycling; and (2) forward pressure required to open the check valve disc. The latter is necessary to determine resistance of the valve to thermal convection flow.

Flowmeters have not been built for sodium systems in the sizes required for the large SGR (26 in.). The largest electromagnetic flowmeters in use are 14-in. components at HNPF and the Fermi plant. Other types of flow-measuring devices might be used, such as orifice plates, venturi meters or flow-tubes. The design bases for each of these devices are known, but accuracy and reliability for sodium service have not been established.

Valves and flowmeter performance for the 5-circuit 1000-Mwe plant would be demonstrated in the prototype plant, since the requirements are essentially the same. The R & D effort for valves and flowmeters is covered in Section V, Volume I.

D. REACTOR

The reactor for the 1000-Mwe plant, described in Section IV, embodies a calandria type core structure in which the graphite is contained. Stainless steel process channels penetrate the calandria heads to provide coolant paths through the core and insertion of fuel and control elements. This type of core structure was selected to minimize the effect of radiation-induced shrinkage of graphite and to improve neutron economy. The same type of core structure is proposed for the 200-Mwe prototype plant, differing primarily in their respective diameters. The fuel and fuel-handling fixtures are similar for both plants, assuring prototyping features.

The reactor structure and a simplified process channel are shown in Figures 5 and 6. The reactor structure, moving radially outward from the center of the core, consists of the calandria, reactor vessel with outer liner, thermal shield, insulation, and cavity liner.

1. Core Parameters and Reactor Physics

Optimization of core parameters has been a continuing process, and the detailed results of a recent study of the 1000-Mwe SGR for minimum cost are presented in the Appendix, Section C. The selection of core parameters for the 1000-Mwe SGR, such as lattice spacing, fuel rod diameter, number of rods per

Figure 5. Reactor Structure

Figure 6. Simplified Process Channel

element and maximum fuel temperature, was based on results from these studies, qualified by fabrication and technical feasibility within a realistic time schedule.

The incremental total cost, defined as fuel cycle cost plus incremental capital cost, decreases with increasing maximum fuel temperature over the range studied, which was 1750 to 2250°F. Furthermore, the optimum fuel rod diameter varies from 0.470 to 0.600 in. for maximum fuel temperatures of 1750 and 2250°F, respectively. The optimum number of fuel rods per cluster appears to be close to 18, but somewhat larger clusters may prove attractive with more refined analyses. One cost factor, not included in this study, that favors larger clusters involves plant downtime due to fuel-handling. Increased number of rods in an element will proportionally reduce the handling time with all other factors equal. However, this type of comparison requires more study to properly weigh all cost factors involved.

The near-optimum core parameters for the 1000-Mwe SGR appear to be an 18-rod cluster of 0.500-in.-diameter fuel rods on a 10.25-in.-square lattice pitch with a maximum fuel temperature somewhat higher than 2250°F. Nevertheless, the peak fuel temperature selected for the 1000-Mwe SGR is 2000°F, which is consistent with a conservative fuel development program goal for 1970.

The prototype reactor core parameters are the same as for the 1000-Mwe reactor, except for peak fuel temperatures. The prototype peak fuel temperature was selected to be 1750°F, which is consistent with near-term goals of the fuel development program.

The core parameters tentatively selected were based on presently used analytical techniques. To confirm and resolve the uncertainties in the analytical approaches for the current design concepts of the 200-Mwe and 1000-Mwe sodium-graphite reactors, a strong experimental program in the area of reactor physics is planned. This physics program is detailed in Section V, of Volume I. Emphasis will be placed on measurements which will directly implement the design effort or provide data in areas where analytic means are presently inadequate. The program should significantly reduce the overdesign of the reactors required to allow for uncertainties in the present physics design effort. The main areas requiring experimental study are (1) clean core reactivity uncertainties, (2) fuel lifetime, and (3) control rod effectiveness, programming and resulting power distributions.

2. Reactor Structure

a. Selection of Calandria

In the SRE and Hallam cores, the moderator and reflector graphite is separated from sodium by cladding individual graphite logs with thin zirconium or stainless steel. The logs are hexagonal and, when installed in the reactor vessel, form a right circular cylinder core. Initial design studies of large SGR core structures included the canned moderator concept and the calandria concept. The calandria is a single, large vessel that contains the moderator and reflector graphite and is penetrated by process tubes which provide a flow path through the core for the sodium coolant. The fuel and control elements are located within the process channels.

After extensive study of the two concepts, the canned moderator core was eliminated from further consideration for the large SGR for the following reasons.

- 1) Thin moderator can cladding relies on structural support from the graphite. Data from Hanford Laboratories on graphite dimensional changes under irradiation indicate cladding support from the graphite could not be relied upon in the high power SGR for a 30-yr design life.
- 2) Creep deformation and creep buckling of thin zirconium alloy or stainless steel cladding at the higher sodium temperatures in the large SGR resulted in a marginal design.
- 3) Maintenance of a canned moderator core necessitates the use of a rotatable top shield to position removal plugs in the shield over any can position in the core. Large rotatable top shields for power plants in the range of 500 to 1000-Mwe result in expensive design details for the top shield and in questionable feasibility.
- 4) Construction costs for a calandria core are less than the canned moderator core.
- 5) The use of stainless steel in the calandria core design readily permits higher sodium outlet temperatures without adversely affecting neutron economy.

Data on graphite damage due to irradiation have been reported by Hanford Laboratories as a result of irradiation experiments for the EGCR Project at Oak Ridge National Laboratory. These data represent the longest exposure information available. Graphite irradiated in a temperature range of 475 to 850°C increases in density, which results in dimensional contraction. If theoretical density is reached, distortion at constant volume may continue.

A theoretical study of graphite behavior in a large SGR, based on an extrapolation of Hanford data^{*}, was conducted to appraise the general problem. The study was based on an SGR core with a peak fast-neutron flux of 3.0×10^{14} n/cm²/sec and maximum graphite temperature of 800°C. At the time of the study, which was conducted over a year ago, the maximum exposure of graphite specimens was 5.7×10^{21} nvt. This exposure would be accumulated in the SGR studied in less than one year of operation at 80% plant factor. The graphite contraction data were extrapolated at a linear rate for five years operation. All conclusions from this study are subject to question, due to the unknown behavior of graphite under long term irradiation. Five-year operation was used for this study. The general conclusions are as follows.

- 1) Bowing of graphite logs will probably occur, due to uneven contraction caused by flux and temperature variations across the logs. Theoretical analyses indicate the degree of bowing is a function of the square of the log length. Maximum graphite bow (h) in an unrestrained moderator log (l) feet long has been estimated as follows.

<u>l</u> (ft)	<u>h</u> (in.)
18	3.92
6	0.710
3	0.180
1	0.021
1/2	0.005

*"The H-3 Irradiation Experiment: Irradiation of EGCR Graphite, Interim Report No. 1," by J. M. Davidson and J. W. Helm, HW-71500A (October 16, 1962)

This maximum bow will occur in graphite logs positioned near the core-reflector interface, where the effective neutron flux gradient is at a maximum.

- 2) Contraction of graphite logs in the direction parallel to extrusion (normally in the vertical direction of an SGR) may be large. A central log 18 ft long may contract as much as 6 in. after 5 yr operation. If the logs are segmented, this contraction would be divided between segments in proportion to their relative positions in the core neutron flux field.
- 3) Contraction of graphite logs in the direction transverse to extrusion (normally in the horizontal direction of an SGR) may be as large as 0.045 in./in. of diameter. That is, a 4-in.-diameter process channel hole in the central graphite log at core mid-plane may contract 0.180 in. on the diameter, assuming graphite irradiation contraction acts in a manner analogous to thermal contraction.

General conclusions for the calandria and moderator can cores may be drawn from the graphite irradiation damage data and the analysis of graphite dimensional changes.

- 1) Bowing of graphite logs is of particular concern for a canned-moderator concept. Moderator elements are closely spaced to minimize sodium absorption of neutrons, yet positioned to provide sufficient sodium cooling of the graphite and cladding material. Spacing between moderator cans in the Hallam reactor is a nominal 0.160 in. Sodium temperatures in this area, and therefore cladding temperatures, are sensitive to dimensional changes; thus channel restrictions due to log bowing would be a major concern.

The major concern from graphite log bowing in a calandria core is the local loads on the process tube, which could result in creep deformation and carburization of the tube. Due to the flexibility of a calandria log design, these loads can be minimized.

- 2) Graphite contraction parallel to the process tube and cladding of a moderator can will result in gaps between segmented logs. Thin-wall cladding spanning these gaps will have a tendency to creep into the gaps, and under these conditions thermal cycling may cause cladding fatigue failure. If a single graphite log is used instead of segmented logs, the bowing problem would be more severe and axial contraction would still open gaps between the moderator log and the top head cladding. Axial contraction of graphite logs in a calandria core is not a problem, because the head is supported by external hanger rods. The hanger approach for moderator can heads is not attractive because of the additional complexity involved with moderator-can removal and replacement.
- 3) Radial contraction of graphite logs may result in interference with the process tube for either core concept. However, the calandria logs can be slit axially to provide the flexibility required to relieve radial loads. If this approach is pursued for the moderator can concept, the moderator can cladding would have to span the axial gaps.

Since this particular evaluation of graphite dimensional changes was made, additional data have been accumulated by Hanford. The latest graphite irradiation data carry exposure to 11.5×10^{21} nvt in the parallel direction to extrusion and 13.8×10^{21} nvt in the transverse direction. Contraction rate in the parallel direction appears to be increasing, whereas the rate of dimensional change in the transverse direction has not only decreased but reversed direction and is beginning to show signs of returning to normal dimensions or possibly to expansion from the initial dimensions. These changes in trend do not negate the earlier analyses, but rather tend to support the selection of this calandria. Since the moderator can cladding relies on structural support from the graphite, and since graphite dimensions cannot be predicted with any reliability, the moderator can concept is at best a marginal design. The calandria structure does not rely on graphite support and can accommodate graphite dimensional changes. For these reasons, the calandria was selected for the large SGR core structure. Other design studies associated with the economic comparison of

moderator cans and calandria indicated a cost advantage for the calandria. In addition, design studies of rotatable top shields for a large moderator-can core revealed major design problems in that area.

The primary concern with the calandria from graphite dimensional changes is possible local buckling of the process tube by pressure from the contracting graphite cylinder associated with each tube or by bowing or a graphite log. To prevent possible deformation of the process tube due to a radial contraction of the graphite log, axial slits are machined in both ends of the log to provide flexibility. Tests to determine stress and deformation behavior of slotted logs have shown that the log will not exert a radial pressure on the tube in excess of 1 psi, which is well below pressure that causes buckling. Tests are under way to evaluate local loads on process tubes from bowed graphite logs and are discussed in the development section of this report. These effects are expected to be small.

b. Vessels and Support

The reactor structure configuration for large SGR plants evolved through several design concepts. In early design studies, the calandria and reactor vessel were supported from the bottom, as was previously done for the SRE and Hallam reactors. This method of support required a self-supporting top head grid plate for the calandria. A calandria design of this type was investigated and rejected due to high costs, uncertainties due to stress from thermal gradients, and an inflexible design for scaleup to larger reactors.

In the present reactor structure design, the reactor vessel and the calandria are supported from the top by pinned hanger rods that allow for radial expansion without bending. Top support was selected for the following reasons.

- 1) Most of the high temperature supporting structure is in simple tension.
- 2) Supporting the thin wall, flat head of the calandria by hanger rods from the loading face shield results in a lightweight system.
- 3) Supporting the weight of the calandria by hanger rods arranged around the shell circumference and hanging the reactor vessel from the loading face shield minimize differential expansions between the two vessels.

Figure 7. 1000-Mwe SGR Structural Loadings with Fuel, Coolant and Operating Condition

- 4) Top hanging of the reactor vessel eliminates the necessity for a large diameter bellows seal that goes with a bottom-supported vessel, as well as the problems associated with relative sliding between the reactor vessel and its support.
- 5) The design features can be scaled up for larger size reactor structures.

The latter advantage is perhaps most significant. The calandria shell diameters for the 1000-Mwe SGR and the 200-Mwe prototype are 32 and 19.5 ft, respectively. In this instance, the smaller reactor structure does not prototype the size requirements for the 1000-Mwe plant. However, because of the top-support hanger rod design, loads in the structural members for both reactors, in general, are approximately the same. These similarities may be seen in the loading diagrams shown in Figures 7 and 8. The same geometry is used for the 1000-Mwe SGR and the prototype reactor structures. By increasing the number of hangers or stay rods in proportion to the increase in support area between the two reactor structures, the load per hanger remains approximately the same, except in the case of the reactor vessel hanger. The reactor vessel support hanger load increases proportionally to the increase in the cross-sectional area within the vessel, while the available space for attaching hangers to the vessel only increases proportionally to the circumference. In this case the cross-sectional metal area of the hangers is designed so that unit stresses are equivalent for the two reactors. The upper limit for hanger loads has not been reached even for the 1000-Mwe reactor.

In addition to the calandria top plate hangers and bottom plate stays, the calandria shell is stayed to the reactor vessel to prevent collapse of the thin-wall shell due to the external sodium static head. Because of the larger shell diameter for the 1000-Mwe reactor, more stays are required to maintain the same unit loads.

The calandria process tubes are basically the same for both reactors. Because the higher specific power core for the 1000-Mwe SGR requires a larger sodium flow area for the same core pressure drop, the process tube ID is increased from 3.950 to 4.250 in. Both reactors utilize 0.025-in.-thick process tubes on a 10.25-in.-square lattice.

Figure 8. 200-MWe SGR Structural Loadings with Fuel, Coolant and Operating Condition

The 1000-Mwe reactor, due to its size, presents a fabrication problem not encountered in the prototype. The prototype vessels can be shop-fabricated and shipped to the site. These components for the 1000-Mwe plant are designed for field fabrication either in place or on the site. The reactor structure components for both plants have been designed to avoid the necessity of machining pieces which require large diameter vertical boring mills as required for SRE and HNPF. Even the grid patterns in the calandria heads and in the loading face shield plate can be machined with portable boring equipment. For these reasons, the work is no longer limited to the few manufacturers who have large machines but is now open to the entire vessel fabrication field. Details of the field erection sequences are given in the 1000-Mwe SGR Conceptual Design contained in Section IV, Volume I.

The calandria and reactor structure are constructed with Type 304 stainless steel and operated in a sodium environment with a maximum nominal temperature of 1150°F. Type 316 stainless steel was also considered, but there is less operating experience with this material than with Type 304 stainless steel in a high temperature sodium environment. The latter material has been extensively evaluated from a metallurgical viewpoint in sodium, helium, and air up to 1200°F, with some studies at higher temperatures.

At the reference operating temperature of 1150°F, stainless steel is subject to creep under long-term loading conditions. Because present analytical predictions of creep deformation and creep buckling of structures cannot be completely relied upon, structural tests are required. A development program is under way to confirm the adequacy of the present design, and includes testing the following items: calandria stayed head, calandria stayed shell, reactor vessel hangers, and process tube structure.

Detail descriptions and cost estimates for these tests are given in the Development Programs contained in Section V of this report.

c. Nozzles and Flow Baffle

An antivortex flow baffle is a feature in the reactor upper plenum design. The baffle consists of a cylindrical outlet manifold around the inner circumference of the reactor vessel. Openings in the baffle are located around the entire circumference. This baffle eliminates the possibility of vortexing at the reactor outlet nozzle. Without the baffle, a sodium height of 18 ft above the

Figure 9. Fuel Element

center line of the outlet nozzle would be required to eliminate vortexing. The use of the baffle reduces the height to 10 ft. By reducing the upper sodium plenum height, substantial cost savings are realized in the reactor vessel, containment vessel, cavity lines, control rods, fuel element guide tubes, fuel-handling machine and other reactor components.

In addition to preventing vortexing, the flow baffle provides for uniform radial flow of sodium in the upper plenum. Sodium flow in a single direction across the upper plenum to the single outlet nozzle would result in relatively high velocities around the large number of orifice tubes in the upper plenum and would cause a hydraulic gradient variation in the upper pool surface. These variations in free surface level around the outer periphery of the vessel would upset the convective head requirements for flow in the annulus between the calandria and reactor vessel and could result in undesirable circulatory flows. Individual outlet nozzles to each heat-transfer loop is a possible alternate design; however, when one loop is shut down variations in the sodium level would still exist. The single outlet nozzle appears to be the best solution and provides considerable cost savings in piping arrangement and gallery space over alternate scheme. Individual heat transfer loop piping is connected to the single outlet nozzle outside the reactor structure.

Elements of the baffle design, such as size and location of openings, vortexing conditions, and flow velocity effects upon the free surface, will be confirmed by hydraulic model tests, which are discussed in the Development Program contained in Section V of this report.

3. Fuel Elements and Fuel Performance

The design of UC fuel elements for the large SGR is based on available materials technology with some extrapolation to the irradiation environment of the 200 Mwe prototype plant and a 1000-Mwe reactor. The supporting fuel development for both plants is discussed in Section V. Typically, the fuel elements are clusters of rods around a single dummy center tube which supports the fuel rod and spacers, Figure 9. The fuel rods are approximately 17-1/2 ft in length and are made up of 14 ft of uranium carbide slugs assembled in stainless steel tubes. The fuel slugs are 1/2 in. in diameter and vary in length from 1 to 8 in. The slugs are bonded to the tubes by sodium to improve heat transfer. The sodium annulus is 0.025 in. thick and allows for 10% local diametrical

swelling. A 40-in.-long gas space is provided above the fuel within each tube for expansions and fission gas collected. All structural materials are Type 304 stainless steel. Design operating conditions have been selected for the two plants as follows.

	<u>Prototype</u>	<u>1000-Mwe Plant</u>
Mixed mean sodium outlet temp. (° F)	1150	1150
Peak central fuel temp. (° F)	1750	2000
Peak clad temp. (inner surface ° F)	1250	1250
Peak burnup (Mwd/MTU)	37,000	37,000
Peak specific power (kw/kgU)	52	72

The design conditions selected are based on limiting temperatures at the center of the fuel and at the inner surface of the cladding. The maximum permissible fuel temperature has not yet been determined for hypo UC but is expected to be somewhat above 2000°F, based on melting of the free uranium in the grain boundaries at about 2070°F. The 2000°F selected for the 1000-Mwe plant is the hot-spot temperature; the nominal or expected probe temperature is about 1870°F, and only a small fraction of the fuel in the core would be near this value.

Most of the irradiation data obtained to date on UC falls into two temperature regions: (1) below 1600°F, with very good radiation stability and low fission gas release; and (2) well above 2000°F, with swelling to about 20% density decrease and substantial gas release. The hot-spot temperature selected (1750°F) for the prototype with a nominal temperature peak of about 1650°F is thus based essentially on the low temperature data obtained to date, and the current fuel program emphasizes investigation at about 1800°F. The 2000°F value selected for the 1000-Mwe plant is expected to be validated by irradiations in this temperature region within the next few years.

The cladding temperature limit was selected on the bases of strength of the stainless steel and possible compatibility difficulties. The creep strength of the austenitic stainless steels decreases rapidly above about 1200°F; the stress-to-rupture in 10,000 hr in Type 304 stainless is about 9700 psi at 1200°F and 5700 psi at 1300°F.

Hypostoichiometric UC, deficient in carbon, was selected as the fuel material to avoid difficulty from carburization of the stainless clad, which occurs when the UC is hyperstoichiometric (surplus of carbon). Use of the hypo material, which contains some free uranium, raised the question of possible U-stainless reactions. Compatibility tests run at temperatures of 1200, 1400, and 1600°F indicate no difficulty with cladding-fuels interface temperatures below about 1300°F.

Type 304 stainless steel was selected for cladding and all fuel element hardware because more information is available on this material in a sodium environment than with other austenitic steels. Type 304 stainless steel clad has been used extensively in SRE and HNPF, and for capsule irradiations. The primary design criterion for this cladding in the large SGR is 1/2% maximum allowable strain at end of life. The factors contributing to the internal pressure which produces the cladding load are: (1) temperature increase of the gas in the free volume above the fuel as the element operating temperatures are reached, (2) sodium bond expansion, (3) fission gas release from the fuel, and (4) irradiation swelling of the fuel. While the internal pressure buildup is occurring, the properties of the stainless steel clad are changing under neutron irradiation, generally in the direction of increased strength and reduced ductility. The magnitude of these changes under sodium reactor conditions is being investigated. Meanwhile the 1/2% strain limit has been accepted in the design of the UC fuel rod as a reasonably conservative figure.

4. Fuel Handling

During the initial period when the large SGR system performance characteristics were being established, it was known that the present fuel-handling systems, such as those used in the SRE and HNPF plants, would require modifications in order to handle the higher powered fuel elements for the large SGR. Modifications are necessary to accommodate the higher decay heat generation in the SGR fuel element. The basic HNPF fuel-handling machine concept, as shown in Figure 10, will be modified for the large SGR.

The higher power SGR fuel element decay heat characteristics are shown in Table 2.

Figure 10. Hallam Fuel-Handling Machine

TABLE 2

PROTOTYPE AND 1000-MWE SGR
DECAY HEAT CHARACTERISTICS

Time After Shutdown (hr)	Prototype Element (kw)	1000-Mwe SGR Element (kw)
1	46	64
3	35	48
5	30	42
10	23	35
20	20	27

For comparison purposes, the Hallam U-Mo fuel element decay heat generation after 10-hr shutdown is 12 kw. The large SGR fuel element cooling problem is magnified approximately 2- to 3-fold.

Due to the high power output, the large SGR fuel-handling system is designed to provide for continuous gas cooling of the fuel elements. Cooling is provided to the fuel element as it moves through the loading face shield and into the fuel-handling machine. The helium cooling system is designed to prevent the outer fuel rod cladding temperatures from exceeding 1200°F during fuel-handling operations. A standby helium cooling system will be provided and will be in operation during fuel-handling operations. The main cooling system power will be connected to the plant emergency power supply.

Fuel will not be removed from the reactor until 10 hr after shutdown. As shown in Table 2, the decay heat loads (after 10-hr shutdown), for the prototype and 1000-Mwe fuel elements are 25 and 35 kw, respectively. The 10-hr time delay does not greatly affect the overall fuel-handling schedule, since several hours are required after shutdown to prepare the reactor loading face shield for fuel-handling operations.

After removal of the fuel elements from the reactor, they will be placed in a large sodium pool so that the decay heat can be readily removed.

The basic fuel-handling system is adaptable to both the prototype and 1000-Mwe plants. The higher fuel performance characteristics in the

1000-Mwe SGR requires a larger helium heat transfer system; however, all component configurations are identical. The current fuel-handling machine development program, described in Section V will investigate the heat-transfer characteristics of the machine for heat loads up to 40 kw. No additional fuel-handling development program would be expected for the larger plants.

5. Control Elements

Existing control element designs for SGR application incorporate the necessary features to provide adequate absorption material, vernier-type rod motion control, and quick insertion of poison by release and free fall with retardation action. Thus, control elements serve as combined shim and safety rods. The concept utilizes an absorber column which moves vertically within a gas-tight, helium-filled thimble positioned in selected core positions. Thus, the absorber element is "dry" and does not contact the sodium coolant. A helium gap between the absorber and thimble permits free movement of the absorber during reactor operations. Control elements are cooled by conduction and radiation of heat across the helium gap to the thimble wall, which is cooled by sodium flowing up through the annulus between the thimble and process tube.

In looking for a specific rod design for the advanced SGR conditions, the specific Hallam control element design was analyzed because of previous experience with this component. It was found unacceptable for the following reasons.

- a) In order to achieve a reasonable in-core life expectancy with gadolinium-samarium oxide, an excessive amount of material is required because of the rapid burnup of the neutron absorbing isotopes in the SGR neutron flux density.
- b) Structural members of the absorber assembly would exceed temperature limits of available materials because of higher heat generation rates in the SGR.

Alternate poison materials, such as europium, boron, and cadmium, which would allow an incore life of 10 yr, were investigated. Of these, dysprosium oxide (Dy_2O_3) appeared to be the most economically attractive material because absorptions in the naturally occurring dysprosium isotopes breed highly absorbing daughter isotopes, leading to a slow decrease in rod worth

with time. Although europium oxide would provide a longer reactivity life time for equivalent thicknesses, the material cost makes it prohibitive for use at this time. Compared to a gadolinium-samarium oxide rod of the same thickness, the rate of rod worth loss with exposure for a dysprosium oxide rod is only one-fourth that of a gadolinium-samarium oxide rod. Thus, dysprosium oxide was selected for the reference material.

The structural members of the poison column exceed temperature limits of available materials because of: (1) high heat generation rates in the absorber element, and (2) comparatively restrained and limited heat transfer across the helium-filled gap between the absorber assembly and thimble.

Two approaches were studied to eliminate the temperature restrictions. Modification of the basic Hallam design to improve heat transfer characteristics appears feasible, within limits, by reducing the helium gap between the absorber and thimble and by locating absorber structural members closer to the cooling source. The improved absorber element design utilizes an annular ring of dysprosium oxide between an internal solid cylinder of graphite and an external Hastelloy-X tube which acts as a structural member. Thus, the Hastelloy-X tube is located as close as possible to the thimble. The greatest improvement in heat transfer properties is through reduced helium clearances, however, there is a practical limit to such an approach. Smaller clearances introduce other potential problems, such as reduced rod scram rates, interference due to bowing and misalignment, wear and galling of materials, and fabrication. Reactor control analyses determined that scram rates of only 0.1 g minimum are required, compared to 0.95 g specified for Hallam. Preliminary experimental data indicate scram rates of 0.5 to 0.6 g can be obtained with clearances as small as 0.020 in. Hence, the slower scram rates due to reduced clearances appear to be adequate. Additional control rod development tasks, which are reported in Section V, are currently under way to appraise the practical limits of the improved design, specifically, the smallest clearance and the maximum temperature for the absorber structural member. These limits are not known precisely at this time, but are believed to be 0.030 to 0.035 in. clearance and 1800 to 1900°F material temperature. Beyond these limits, the "dry" control element concept is considered unsuitable for reliable operation. Results of detailed studies of this problem are given in the Appendix, Section D.

It is apparent that the temperature limitation of present control rod designs can be circumvented by elimination of the thimble, to allow generated heat to be transferred directly to the reactor coolant. This alternate approach has been studied in some detail. Design studies of "wet" control rods have yielded no practical alternative to mechanisms which operate in the sodium coolant; this difficulty arises from the distance, approximately 13 ft, which the absorber element must move in an SGR. The major part of the development of a wet control rod therefore involves the finding of materials for bearings and gears or screws that can operate for long periods of time in sodium at about 1200°F. Two alternate designs were studied: one concept employs a telescoping rack-and-pinion actuator with a bellows seal; the other concept utilizes a screw-and-roller-nut actuator mechanism for translation of the absorber element and incorporates a rotary seal which uses a sodium vapor condensing zone below an O-ring seal. A program for design and development of the wet rod was initiated for the AEC but was discontinued in favor of the modified dry rod concept in view of an estimated 3-yr program to complete the wet rod work. The dry rod design was selected for development, rather than the more attractive wet rod concept, because development could be completed at an earlier date for the prototype reactor.

The dry rod concept is satisfactory for a near-term SGR, which is limited in specific power by the assumption of conservative operating conditions for the UC fuel element. It is expected that the development program on UC will demonstrate that this fuel is capable of substantially higher specific power, which would permit a reduction in size and cost of the reactor. The most economic specific power, particularly in large reactors of 500- to 1000-Mwe power, will require improved means of control rod cooling. The prototype dry control rod in the 1000-Mwe SGR, operating at a peak core specific power of 72 kw/kgU as compared with a peak specific power of 52 kw/kgU for the prototype reactor, has a peak temperature in the absorber structural member of 2035°F. This temperature is not acceptable for reliable operation. Several alternatives are available to provide a reliable control rod for the 1000-Mwe SGR.

- a) Direct cooling of the neutron absorber by liquid sodium, i. e., a "wet" control rod, is an obvious way to eliminate the heat transfer problem. This approach involves additional development.

- b) Temperatures in the current dry rod concept could be lowered by further reduction of the helium gap, provided other operations and fabrication difficulties are not encountered. This approach would require additional testing of the current design.
- c) The dysprosium oxide density in the current dry rod concept could be reduced by blending an inert material with the Dy_2O_3 , and thereby reduce the absorptions per rod and corresponding heat-generation rates to an acceptable level. This approach would require more control rods in the core as well as additional development of material compatibility and irradiation stability.

Of these choices, the wet rod concept is by far the better long-range approach, because it would solve the control rod heat transfer problem for all SGR plants in the foreseeable future and would result in the most economical and reliable control system. For these reasons, the wet control rod concept has been selected for the reference 1000-Mwe plant. Another benefit of this design approach is that the driver units can be readily located in the loading face shield so that the top of the shield would be clear for fuel handling.

6. Safeguards

Nuclear safety features of the SGR plants minimize design and regulatory considerations in the safeguards field. Inherent safety features, normal design features, and engineered safeguards features should allow construction of a large SGR on any site otherwise suitable for a power station. Volume II includes the results of accident analyses for the 200-Mwe prototype plant and comparison with AEC regulatory requirements for the AEC standard site. Safety features, design criteria, the 200-Mwe study results, modification of the results to a 1000-Mwe plant and conclusions for the larger plant are summarized below.

Significant favorable inherent safety features of an SGR are low vapor pressure of sodium, operating temperature of the plant about 500°F below the boiling point of sodium, high burnout heat flux, and chemical affinity of sodium for fission production. Unfavorable inherent features are radioactivity of sodium and sodium chemical reaction with both air and water.

Several design features incorporated in the plant as a result of other engineering decisions significantly enhance safety characteristics. Some of these are:

- a) The large heat capacity of graphite provides delay time for restoration of cooling following loss of normal cooling;
- b) The use of stainless steel to contain sodium reduces chance of brittle failure of the reactor vessel;
- c) The use of high conductivity and high melting point fuel provides substantial prompt negative temperature coefficient of reactivity prior to meltdown; and
- d) Use of massive shielding around the primary heat transfer system provides a convenient structure for backup containment if rupture of the stainless steel envelope should occur.

Other plant design features are related principally to safeguards. Where necessary, these are varied to suit postulated accidents and site conditions. The main items in this category are building design and ventilation. For most sites carefully designed industrial buildings, coupled with ventilation systems to ensure inward leakage and stack release, are sufficient to handle the consequences of accidents normally associated with SGR's. Air filtration and stack height are varied to control radioactive release following a postulated accident. Buildings designed for 145%/day at 1/4-psi differential pressure, using charcoal filtration in the event of a fuel-handling accident, should meet all the requirements for operating large SGR's on the AEC standard site. Other engineered features of SGR plants that contribute to plant safety include a stainless steel reactor vessel liner, guard piping, and natural convection cooling. The liner will contain radioactive sodium in case of a reactor vessel failure. In addition, the primary inlet and outlet lines are enclosed by guard pipes from the reactor to the primary cell. If an outlet or inlet line should rupture near the reactor, primary sodium is contained. The sodium systems are designed for natural convection flow for decay heat removal in the event of a pump failure. The elevation of primary loop piping is set such that a line rupture will not drain the reactor sodium pool below the outlet nozzle. Thus the design features are such that the core will remain covered with sodium if emergency conditions should occur.

Three potential accidents have been analyzed in some detail for the 200-Mwe prototype and are presented in Volume II of this report. These are:

(a) core meltdown, (b) fuel handling, and (c) sodium fires. The probability of these accidents occurring is extremely low, and requires multifailures of the reactor system and protection system components.

The core meltdown accident is radiologically inconsequential unless the malfunction of a seal is concurrently assumed in order to permit the release of fission products from the core to the building area. The fuel-handling accident assumes that an irradiated fuel element falls on the building floor and completely burns in 15 min. Radiological effects of sodium fires are a function of sodium inventory, gallery or vault floor area, available oxygen and fission material inventory.

The analytically determined siting distances for the 200-Mwe are shown in Table 3 for each of the three postulated accidents.

TABLE 3
REQUIRED SITING DISTANCES FOR A 200-MWE SGR

Accident	Exclusion Area	Low Population Zone	Population Center
Core meltdown	0.06*	0.07	0.09
Sodium fire	0.06*	0.07	0.09
Fuel handling	0.06*	0.21	0.28

*The codes used are not applicable to determining siting distances less than 100 meters d(0.06 mi)

The most restrictive siting distance is 0.28 mi. This falls within the site boundary for the AEC standard site and should be satisfactory for most locations considered. If required, this distance can be reduced by shielding for direct radiation from the dropped fuel element radiation.

A comparison of the physical differences between the 1000-Mwe and the 200-Mwe plant, in regard to the referenced accident conditions, will afford a good indication of the safety of the larger plant. Even though the 1000-Mwe SGR core volume is more than three times as large as the 200-Mwe plant, the fractional percentage of core meltdown with all rods out is about the same for the two plants. Approximately 3-1/3 more fission products will be released to a core cover gas volume, but this volume is about three times greater than in the

200-Mwe reactor. Thus, only a slightly greater $\mu\text{c/cc}$ concentration will exist in the 1000-Mwe case. Since element sizes are the same, a seal malfunction would release contaminated core cover gas at the same rate with the result that with a core meltdown accident, the exclusion area would be about the same as the 200-Mwe case. The low population zone distance would be slightly greater (about 10%). Since the total activity released would be increased by the factor of 3-1/3.

The fuel-handling accident for the 1000-Mwe plant involves the same fuel element but with less enrichment — 3.6% compared with 3.8%; thus, with the building volume more than twice as large, the exclusion radius and the low population zone distances are slightly smaller for the larger reactor.

Sodium spills and sodium fire accidents could be nearly twice as severe for the larger plant. The radiological effect, however, is still somewhat smaller than that for the fuel-handling accident. It is thus concluded that, in general, there is very little difference in the relative operational safety of the two plants. Analysis of all conceivable accidents to an SGR system has failed to produce a single accident with enough energy storage to breach the primary containment and as a result, release large quantities of fission products. To confirm these analyses, an experimental program in the area of reactor safety is desirable. The planned program is detailed in Section V, Volume I.

III. ECONOMICS

A. SUMMARY

Power generation costs for the reference 1,000,000-kw plant are based on AEC evaluation ground rules as specified in the Guide To Nuclear Power Cost Evaluation (TID-7025). These ground rules include format, percentages for indirect construction costs, expenditure curve for construction costs, percentages for working capital, capital charge rates, and fuel reprocessing costs. Additional assumptions to estimate total power generation costs are 80% annual plant factor and representative fixed charges for both investor-owned and municipally-owned utility financing. The net electrical generating capacity of the plant is 1,019,100 kw. These costs are summarized in Tables 4 (investor owned) and 5 (municipally owned). The estimated costs are as of November 1, 1963 and escalation beyond that date is not included. The plant terminates at the low side of the main transformer.

B. CAPITAL COSTS

The power station is assumed to be the first plant constructed on an undeveloped hypothetical site, as defined in TID 7027, Guide to Nuclear Power Cost Evaluation. The total capital cost for the reference plant design is \$143,493,000 as an investor-owned plant, and \$138,270,000 as a municipally owned plant. Based on a net generating capacity of 1,019,100 kw, the unit costs are \$141/kw and \$136/kw as an investor-owned plant and a municipally owned plant, respectively.

A summary of the total construction costs is given in Table 6. Details of the direct construction costs are tabulated according to the preferred AEC account systems in Table 7. Direct construction cost estimates are based on detailed takeoff and vendor quotes for major equipment items established for the 200-Mwe prototype plant and appropriately extrapolated for the 1000-Mwe power station.

Indirect construction costs which were added to the total direct construction estimate of \$93,942,000 include: administrative, miscellaneous construction, engineering, design and inspection, startup, land and land rights, contingency and interest during construction. The general and administrative miscellaneous

TABLE 4
ENERGY COST OF 1000-MWE SGR INVESTOR-OWNED PUBLIC UTILITY

Cost Item	Capital Cost (\$1000)	Rate (%)	Annual Cost (\$1000)	Unit Cost (mill/kwh)
Fixed charges				
Depreciating capital	143,133	14.5	20,754	2.91
Total capital cost (less land and land rights)				
Nondepreciating capital				
Land and land rights	360	13.0	47	0.01
Working capital				
a) Plant operation and maintenance	512	13.0	66	0.01
b) Fuel cycle operations	4,399	13.0	572	0.08
Nuclear liability insurance	-	-	330	0.05
Subtotal — annual fixed charges	-	-	21,769	3.06
Operating costs				
Fuel cycle cost			9,534	1.34
Operating and maintenance cost			1,427	0.20
Subtotal — operating cost			10,961	1.54
Total energy generation cost			32,730	4.60

construction, engineering, design and inspection costs were determined by applying fixed percentages — a function of the total direct construction cost — to the total direct construction cost in an accumulative manner. These percentages are indicated in parenthesis in Table 6. Startup charges were estimated at 35% of the annual operating and maintenance cost, including supplies but excluding fuel. Contingency was calculated at 10% of the total direct and indirect construction cost, including cost of land and land rights.

Interest during construction was based on a construction period of 54 months and the assumed expenditure curve contained in the AEC Guide. The construction interest charge for the plant depends on the type of utility ownership. The

TABLE 5
ENERGY COST OF 1000-MWE SGR MUNICIPALLY OWNED PUBLIC UTILITY

Cost Item	Capital Cost (\$1000)	Rate (%)	Annual Cost (\$1000)	Unit Cost (mill/kwh)
Fixed charges				
Depreciating capital	137,910	7.7	10,619	1.49
Total cost (less land and land rights)				
Nondepreciating capital				
Land and land rights	360	5.5	20	0.005
Working capital				
a) Plant operating and maintenance	512	5.5	28	0.005
b) Fuel cycle operations	4,399	5.5	242	0.03
Nuclear liability insurance			330	0.05
Subtotal — annual fixed charges			11,239	1.58
Operating costs				
Fuel cycle			9,534	1.34
Operating and maintenance			1,427	0.20
Subtotal — operating costs			10,961	1.54
Total energy generation cost			22,200	3.12

annual interest rate applicable to investor-owned and municipally owned utilities was assumed to be 6 and 4%, respectively.

C. FUEL CYCLE COSTS

Fuel cycle costs were computed for "equilibrium" fuel-cycle conditions. These conditions would be reached after approximately 3 yr of operation. During the initial period, the estimated average fuel cycle cost is around 0.2 mill/kwh higher than the equilibrium cost.

The design operating and economic factors used in calculating the fuel cycle cost are summarized in Table 8. A detailed breakdown of the fuel cycle cost is

TABLE 6
SUMMARY OF CONSTRUCTION COST ESTIMATE FOR 1000-MWE
SGR NUCLEAR POWER PLANT

Acc't No.	Description	(\$ in 000)		
		Material and Equipment	Labor	Total
21	Structures and improvements	5,534	4,562	10,096
22	Reactor plant equipment	45,439	10,253	55,692
23	Turbine generator equipment	22,054	2,543	24,597
24	Accessory electric equipment	1,753	947	2,700
25	Misc. power plant equipment	657	200	857
	Total direct construction cost	75,437	18,505	93,942
Indirect construction costs				
	General and administrative at 6.3%			5,198
	Subtotal			99,140
	Misc. construction costs 1%			991
	Subtotal			100,131
	Engineering, design and inspection services at 14.8%			14,819
	Subtotal			114,950
	Startup costs			500
	Subtotal			115,450
	Land and land rights			360
	Subtotal			115,810
	Contingency at 10%			11,581
	Subtotal			127,391
	Interest during construction (54-month schedule — 12.2%)			15,542
	Subtotal			142,933
	Sodium and gases (initial charge)			560
	Total evaluated cost — investor-owned			143,493
	Cost per net kw (plant net output — 1,019,100 kw) = \$141			
	Adjustment for municipally owned			
	Interest during construction (54-month schedule — 8.1%)			10,319
	Total evaluated cost — municipally owned			138,270
	Cost per net kw (plant net output — 1,019,100 kw) = \$136			

TABLE 7
DETAILS OF DIRECT CONSTRUCTION COST ESTIMATE FOR
1000-Mwe SGR NUCLEAR POWER PLANT

Account Number	Description	(\$ in 000)			Account Number	Description	(\$ in 000)			Account Number	Description	(\$ in 000)		
		Material	Labor	Total			Material	Labor	Total			Material	Labor	Total
20	Land and land rights				212-F	Gate house	12	8	20	23	Turbine generating plant equip.			
201	Land and privilege acquisition			360	212-G	Misc. outdoor foundations	10	15	25	231	Turbine generators			
202	Relocate highway and railroads			-						.1	Foundations-pedestal	151	185	336
203	Relocate phone and power lines			-	218	Stack	60	20	80	2	Turbine-generator	17,767	780	18,547
	Total acc't 20			360		Total acc't 21	5,534	4,562	10,096	.3	Standby exciter	517	30	547
21	Structures and improvements				22	Reactor plant equipment					Subtotal acc't 231	18,435	995	19,430
211	Ground improvements			-	221	Reactor equipment				232	Circulating water system			
.1	Access roads (off-site)			-	.1	Reactor vessel w/internals	1,042	1,500	2,542	.1	Pumping and regulating equip.	670	53	723
.2	General yard improvements			-	.2	Reactor controls	3,980	30	4,010	.2	Circulating water lines	434	197	631
.21	Grading and landscaping	180	120	300	.3	Reactor shielding	1,548	234	1,782	.3	Intake and discharge structures	552	342	894
.22	Roads, sidewalks, and parking areas	60	30	90	.4	Reactor cool. and heat. systems	1,930	892	2,822	.4	Corrosion cont. and water treat. sys	34	6	40
.23	Retaining walls, fences and railings	90	75	165	.5	Reactor plant containers	685	773	1,458		Subtotal acc't 232	1,690	598	2,288
.24	Outside water dist. systems	60	40	100	.6	Calandria moderators and reflectors	2,832	566	3,398	233	Condensers and auxiliaries	1,000	340	1,340
.25	Sewers and drainage systems	120	80	200	.7	Reactor plant cranes and hoists	203	12	215	234	Central lube sys.	37	23	60
.26	Roadway and general lighting	10	6	16		Subtotal acc't 221	12,220	4,007	16,227	235	Turbine plant instr. and control	360	340	700
	Subtotal acc't 211.2	520	351	871	222	Heat-transfer systems				236	Turbine plant piping	500	213	713
.3	Railroad 5 mi	200	100	300	.1	Reactor primary coolant system	2,228	323	2,551	237	Auxiliary equip. for generators	32	34	66
.4	Waterfront improvements	200	100	300	.2	Intermediate coolant system	7,174	436	7,610		Total acc't 23	22,054	2,543	24,579
	Subtotal acc't 211	920	551	1,471	.3	Stm. gen. S.H. and R.H. system	9,597	1,586	11,183	24	Accessory electric equipment			
212-A	Turbine-generator bldg.				.4	Reactor cool. rec. sup. and treatment	1,727	443	2,170	241	Switch gear	750	223	973
.1	Excavation and backfill	70	75	145		Subtotal acc't 222	20,726	2,788	23,514	242	Switch boards			
.2					223	Fuel, C.R. and mod. hdlg. and storage equip				.1	Main control board	55	11	66
.3	Substructure	330	360	690	.1	Cranes and hoist equipment	821	31	852	.2	Aux. power, battery and signal bds.	68	11	79
.4	Superstructure	470	290	760	.2	Special tools and services	300	0	300	.3	Motor control centers	84	16	100
.6	Building services	210	155	365	.3	Storage cool. clean. and insp. services	808	230	1,038		Subtotal acc't 242	207	38	245
	Subtotal acc't 212-A	1,080	880	1,960		Subtotal acc't 223	1,929	261	2,190	243	Protective equip.	18	34	52
212-B	R/A waste bldg.				225	R/A waste-treated disposal				244	Electrical structures	26	92	118
.1	Excavation and backfill	3	9	12	.1	Liq. waste system	138	48	186	245	Conduit	60	229	289
.3	Substructure	65	75	140	.2	Gaseous waste system	139	65	204	246	Power and control wiring	140	255	395
.6	Building services	13	10	23		Subtotal acc't 225	277	113	390	247	Station service equip.	552	76	628
	Subtotal acc't 212-B	81	94	175	226	Inst. and control					Total acc't 24	1,753	947	2,700
212-C	Administration bldg.				.1	Cont. rm. console and panels	1,200	375	1,575	25	Misc. power plant equip.			
.1	Excavation and backfill	7	28	35	.2	Locally mtd. instr.	2,550	500	3,050	251	Crane and hoisting equip	365	35	400
.3	Substructure	47	28	75	3	H.P. instr.	76	0	76	252	Compressed air and vacuum clean. sys.	125	70	195
.4	Superstructure	103	84	187	.4	Control valves	1,000	in piping	1,000	253	Other power plant equip.	167	95	262
.6	Building services	47	30	77	.5	Connect. and supports	263	400	663		Total acc't 25	657	200	857
	Subtotal acc't 212-C	204	170	374		Subtotal acc't 226	5,089	1,275	6,364					
212-D	Reactor building w/aux. bay and steam generator bldg.				227	Feedwater sup. and treat.	3,111	659	3,770					
.1	Excavation and backfill	73	273	346	228	Stm. cond. and feedwater sys.	1,575	975	2,550					
.3	Substructure	1,500	1,458	2,958	229	Other reactor plant equip.								
.4	Superstructure	950	607	1,557	.1	Equip. decontam. sys.	158	79	237					
.6	Building services	600	450	1,050	.2	React. plant maint. sys.	354	96	450					
	Subtotal acc't 212-D	3,123	2,788	5,911		Subtotal acc't 229	512	175	687					
212-E	Warehouse					Total acc't 22	45,439	10,253	55,692					
.1	Excavation and backfill	2	10	12										
.3	Substructure	8	4	12										
.4	Superstructure	24	16	40										
.6	Building service	10	6	16										
	Subtotal acc't 212-E	44	36	80										

TABLE 8
FUEL CYCLE DATA FOR 1000-Mwe SGR

	Units	Value
A. Design parameters		
1. Fuel composition		Hypostoichiometric UC
2. Cladding material		Stainless steel
3. Fuel enrichment, initial	%	3.60
4. Fuel enrichment at discharge	%	1.45
5. Average fuel exposure	Mwd/MTU	25,000
6. Plutonium concentration in fuel		
(a) Initial	g/kg U	0.00
(b) At discharge	g/kg U, ^a	6.78
7. Rated thermal power	Mwt	2,336
8. Rated net electrical power	Mwe	1,019
9. Reactor fuel loading, initial	MTU	73.8
10. Ratio of final to initial U loading		0.967
11. Fuel management program		4-zone, semigraded partial, out-in
B. Operating parameters set by industry		
1. Plant factor	%	80
2. Total time for fuel fabrication and shipping to reactor site	Months	5.0
3. Preirradiation storage time	Months	1.0
4. Batch size charged to reactor per refueling	MTU	18.45
5. Batch size discharged from reactor per refueling	MTU	17.84
6. Number of discharge batches accumulated for chemical reprocessing	-	1
C. Operating parameters set by AEC		
1. Minimum decay cooling	days	90
2. Irrecoverable loss during chemical separation, U	%	1.0
3. Irrecoverable loss during chemical separation, Pu	%	1.0
4. Irrecoverable uranium loss during conversion, UNH → UF ₆	%	0.3
5. Irrecoverable plutonium loss during conversion, PuNH → Pu metal	%	1.0
6. Chemical separation plant processing rate	MTU/day	0.956
7. Chemical separation plant turn-around time	days	8.0
D. Economic parameters set by industry		
1. Conversion and fabrication processing cost (excludes shipping and use charge)	\$/kg U	90.00
2. Shipping charge, AEC to conversion plant	\$/kg U	1.50
3. Shipping charge, fabrication to reactor	\$/kg U	1.50
4. Shipping charge, reactor to chemical processing site	\$/kg U	10.00
E. Economic parameters set by AEC		
1. Use charge rate	%	4.75
2. U price at initial enrichment	\$/kg U	320.90
3. U price at final enrichment	\$/kg U	90.35
4. Conversion charge, UNH to UF ₆	\$/kg U	5.60
5. Pu price (credit)	\$/g	9.50
6. Separation plant daily charge	\$	17,000

^a per kg U charged to core

given in Table 9. It may be noted that the total equilibrium fuel cycle cost is estimated at 1.34 mill/kwh.

The fuel fabrication cost of \$99.35/kg U is consistent with demonstrated fabrication costs on the ESADA UC fuel fabrication program. In fact, this cost is believed conservative since it does not assume any further process improvements, which are bound to result from the fabrication of the HNPF second core. In addition, process improvements gained from the fabrication of the 200-Mwe prototype first and possibly second core would further reduce fuel fabrication costs before fuel is required for the 1000-Mwe plant.

Use charges are based on lease of fuel material from the government at an annual rate of 4.75% of its value. The amount of fuel in process and its value are computed during all phases of the fuel cycle, including allowance for losses and spare fuel elements. For fuel elements in the core, the mean of the initial and final fuel value is used to establish its use charge.

Uranium losses and burnup charges are computed using the July 1, 1962 schedule of Base Charges for Enriched Uranium, as UF_6 . The uranium loss computation includes 1% for irrecoverable loss of U during chemical separation, and 0.3% for irrecoverable loss of U during chemical conversion from uranyl nitrate hexahydrate (UNH) to UF_6 . Uranium losses that occur during conversion and reduction are included in the fabrication cost.

Chemical processing cost includes those incurred from the time irradiated fuel is shipped from the reactor site until processing is completed and chemical separation of fissile isotopes occurs, and for conversion of UNH to UF_6 and Pu nitrate to metal. The conversion charge for converting UNH to UF_6 is \$5.60/kg U and for converting Pu nitrate to Pu metal is \$1.50/g Pu. The separations plant daily charge was taken at \$17,000/day.

Plutonium credit is computed on the quantity of Pu produced less Pu losses during chemical separation and conversion. The value of Pu is taken at \$9.50/g as metal.

Private ownership of special nuclear materials is currently under serious consideration both by Government and industry. Under the current AEC proposal on private ownership there are three major factors which would affect the fuel cycle cost. These are: (1) increase in fuel inventory charge rate in

TABLE 9

FUEL CYCLE COSTS OF 1000-MWE SGR

Fabrication	\$/KgU Charged into Reactor						Totals (mill/kwh)
	Processing	Shipping	Use Charge	U Loss or Consumption	Pu Loss or Production	Totals	
1. Transit to conversion site and fabricator	-	1.50		-	-		
2. Conversion and fabrication	90.00	-		*	-		
3. Transit to reactor and recycle	-	1.50		-	-		
4. Subtotal (1 + 2 + 3)	90.00	3.00	6.35	-	-	99.35	0.380
<u>Reactor</u>							
5. Preirradiation inventory	-	-	1.27	-	-		
6. Irradiation	-	-	26.06	233.53	(64.41) [†]		
7. Decay	-	-	1.03	-	-		
8. Subtotal (5 + 6 + 7)	-	-	28.36	233.53	(64.41)	197.48	0.755
<u>Chemical processing</u>							
9. Transit to process site	-			-	-		
10. Separation	24.28			0.88	0.66		
11. Uranium conversion	5.35			0.26	-		
12. Plutonium conversion	10.07			-	0.65		
13. Transit to receiving point	-			-	-		
14. Subtotal (9 + 10 + 11 + 12 + 13)	39.63	9.67	0.80	1.14	1.31	52.55	0.200
15. <u>Totals</u> — KgU charges	129.63	12.67	35.51	234.67	(63.10)	349.38	-
16. <u>Totals</u> — mill/Kwh	0.495	0.049	0.136	0.896	(0.241)	-	1.335

*Included in fabrication cost

†Values shown in parentheses are credits for net plutonium production

accordance with individual utility financing, (2) availability of toll enrichment of privately owned uranium in AEC facilities, and (3) guaranteed buy-back price for plutonium based on its relative value to U^{235} in thermal reactors. Accordingly, the fuel cycle cost may either increase or decrease depending on the relative effect of each of these factors. Figure 11 shows these effects for the reference plant, assuming fuel inventory charge rates to equal working capital

Figure 11. Effect of Private Ownership on SGR Fuel Cycle Cost (1000 Mwe SGR)

charge rates as recommended by the AEC Guide. Also shown for comparison is the effect based on an inventory charge rate of 10% which may be more representative of the average cost of working capital to most private utilities in this country. It may be noted from Figure 11 that the effect of private ownership on the SGR fuel cycle cost would be relatively small, so that it could not materially affect its economic attractiveness.

D. OPERATION AND MAINTENANCE COST

A summary of plant operation and maintenance cost are shown in Table 10. The plant operating and maintenance personnel requirements and cost are based on the staff, salary, and wage rates indicated in Table 11. An allowance of 20% is added to the total direct labor cost for fringe benefits. These include vacations, shift premiums, sick leave, and other company-financed benefits. The annual costs of operating supplies and maintenance materials and services are tabulated in Table 12. Adding this expense to the total labor gives a total annual operating and maintenance cost of \$1,427,000.

TABLE 10
SUMMARY OF ANNUAL OPERATION AND MAINTENANCE
COSTS ON 1000-Mwe SGR

Direct payroll costs	\$ 468,800
Fringe benefits at 20%	93,800
Total labor	562,600
Operating supplies and maintenance	864,400
	<hr/>
	\$1,427,000

E. NUCLEAR LIABILITY INSURANCE

The amount of financial protection required of AEC reactor licenses was determined in accordance with provisions of "Title 10 - Atomic Energy, Chapter 1 - Atomic Energy Commission, Part 140 - Financial Protection Requirements and Indemnity Agreements, Sub-part B, amended, published in 25 Federal Register 2944, April 7, 1960."

For the 1000-Mwe SGR, the maximum financial protection requirements of \$60,000,000 are applicable. The estimated annual premium for this protection is \$260,000.

In addition to the above coverage, a fee for government indemnification was charged at the rate of \$30/yr per thermal megawatt. This fee provides protection for liability in excess of \$60,000,000 in the amount of \$500,000,000. The annual fee for this protection is about \$70,000.

TABLE 11
ANNUAL LABOR COST OF 1000-Mwe SGR

	Salary or Wage Rate	Personnel Required	Annual Cost (\$)
<u>Plant Management</u>			
Superintendent	1,500/mo	1	18,000
Ass't superintendent	1,100/mo	1	13,200
Office Manager	3.00/hr	1	6,240
Secretary	500/mo	1	6,000
Senior clerk	3.00/hr	1	6,240
Clerk-typist	3.00/hr	2	12,480
Clerk	2.10/hr	1	4,368
		<u>8</u>	<u>66,528</u>
<u>Technical Staff</u>			
Reactor engineer	1,000/mo	1	12,000
Efficiency engineer	1,000/mo	1	12,000
Instrument engineer	900/mo	1	10,800
Mech-elec. engineer	1,000/mo	1	12,000
Chem. technician	3.00/hr	1	6,240
Health and safety engr.	1,000/mo	1	12,000
Health and safety tech.	3.00/hr	2	12,480
Junior engineer	750/mo	-	
		<u>8</u>	<u>77,520</u>
<u>Operating Staff</u>			
Shift supervisor	725/mo	5	43,500
Control operators	3.50/hr	5	36,400
Ass't control operator	2.70/hr	5	28,100
Equip. attendants	2.50/hr	14	72,800
Plant guards	2.25/hr	5	23,350
		<u>34</u>	<u>204,150</u>
<u>Maintenance</u>			
Maintenance supervisor	3.75/hr	1	7,800
Boilermaker	3.15/hr	1	6,550
Boilermaker helper	2.55/hr	-	
Fitter-welder	2.75/hr	1	5,720
Fitter helper	2.55/hr	-	
Machinist	2.90/hr	1	6,040
Utility man	2.25/hr	3	14,040
Instrument tech.	3.15/hr	8	52,400
Plant mechanic	2.65/hr	2	11,020
Electrician	2.95/hr	1	6,140
Electrician helper	2.55/hr	1	5,310
Warehouseman	2.70/hr	1	5,620
		<u>20</u>	<u>120,640</u>
Total direct payroll		70	468,838

TABLE 12
ANNUAL COST OF OPERATING SUPPLIES AND MAINTENANCE
MATERIALS AND SERVICES ON 1000-Mwe SGR

Item	Annual Cost (\$)
1. Makeup sodium, 10,800 lb at 0.30 \$/lb (Replacement of sodium removed in changing cold traps)	3,300
2. Helium, 1,000,000 ft ³ /yr at 0.05 \$/ft ³ (Normal use plus estimated purge requirement)	50,000
3. Nitrogen, 8,000,000 ft ³ /yr at 0.007 \$/ft ³ (Normal use plus estimated purge requirement)	56,000
4. Diesel oil, 3,000 gal at 0.15 \$/gal (Diesel generator tests and startup boiler operation)	500
5. Miscellaneous lubricants (greases, oil, etc)	600
6. Filters (H & V, R/A vent)	6,000
7. Chemicals (Demineralizers, water treatment)	3,000
8. Health physics supplies (cartons, wipers, gloves, coveralls, etc)	6,000
9. Chart paper (recording instruments)	36,000
10. Clerical supplies (paper, pencils, reproductions, etc)	3,000
11. Contract and outside services	100,000
12. Replacement parts	600,000
Total	<u>864,400</u>

The total estimated annual premium and fee for financial protection, therefore, is \$330,000.

F. WORKING CAPITAL

Working capital requirements were divided into two categories, one associated with operation and maintenance, and the other with fuel cycle operation.

The working capital for operation and maintenance was estimated at 2.7% of the annual operating and maintenance expense including fuel, plus 25% of the

annual cost of maintenance materials and supplies. The working capital for fuel cycle operations was estimated at 60% of the equilibrium cycle core fabrication cost. Table 13 summarizes the working capital requirements for the plant.

TABLE 13
WORKING CAPITAL FOR 1000-Mwe SGR

	\$
Plant Operation and Maintenance	
Average net cash required	
(2.7% of annual operating expense, including fuel = $0.027 \times \$10,961,000$)	295,957
Materials and supplies in inventory	
(25% of annual cost of maintenance, materials and supplies = $0.25 \times \$864,400$)	216,100
Fuel cycle operations	
Core fabrication	
(60% of core fabrication = $0.60 \times \$7,332,000$)	4,399,200
Nuclear materials	
(assumed leased from U.S. AEC at 4.75%/yr)	None
Total working capital	4,911,257

G. FUTURE SGR FUEL CYCLE ECONOMICS

There are several areas where sodium graphite reactor fuel cycle costs can be further reduced. One example is the reduction of the overall power peak-to-average and a corresponding increase in the core average specific power. An overall power peaking factor of 2.25 is currently used for design purposes. It is possible to significantly reduce this value (to about 1.8) by the use of improved control rod and fuel element programming and by the use of graded fuel enrichments in each fuel element. The resulting increase in average specific power will yield reduced fuel inventory charges. This is particularly attractive if the fuel is privately owned. This gain is partially counterbalanced by reduced neutron economy due to slightly increased neutron leakage and equilibrium

xenon concentration. The net effect, however, is a reduction in fuel cycle cost as well as a significant capital cost reduction due to a smaller core size.

Fuel cycle costs may also be reduced by more frequent shutdowns for fuel reloading. This does not significantly affect the percent down-time for refueling, but results in less reactivity required for fuel burnups, and hence in a reduced feed enrichment and reduced fuel burnup cost. An additional advantage is a reduction in the number of control rods required.

In the future, zirconium alloys may be developed which are suitable for use in SGR's. If the stainless steel structure materials in the reference core were replaced with zirconium, a significant reduction in parasitic neutron capture would result. Thus, neutron economy would be improved, enrichment requirements would be reduced, and fuel burnup costs would be lowered. Although fabrication costs would increase slightly due to the additional cost of zirconium over steel, the total fuel cycle cost would be reduced significantly.

Another potential means of improving SGR performance is the use of varied lattice spacings across the reactor core. Fresh fuel is inserted into a region of the core having a tight lattice spacing and a hard neutron energy spectrum, resulting in high conversion. When the element reaches half its discharge exposure, it is shifted to a region having an open lattice and a softer, more thermal spectrum. This increases the reactivity of the element by reducing resonance capture in U^{238} , Pu^{240} , and the fission products. In comparison with a uniform lattice core, the mixed spectrum SGR results in less capture in fission products, a reduction in nonfission capture in Pu^{239} , and an increase in conversion of U^{238} to Pu^{239} . Enrichment requirements are thereby reduced and Pu production is increased.

Significant reductions in fuel fabrication costs are also expected to occur. On the basis of a production volume of 600 kg U per day, it is expected that fuel fabrication costs can be reduced from \$100/kg U to approximately \$80/kg U.

All of these potential improvements, if realized, are expected to reduce the fuel cycle of SGR's by 0.2 to 0.4 mill/kw-hr.

IV. 1000-Mwe SGR CONCEPTUAL DESIGN

The following section describes the conceptual design of a 1000-Mwe SGR. The design is representative of a base-load power plant to be constructed in the 1970's. It incorporates advanced technology indicated by goals of current development programs to be commensurate with the target construction date.

The reactor features the calandria vessel for graphite containment. Its advantages are low capital costs, neutron economy, and a tolerance for graphite shrinkage. The reference fuel element is uranium carbide clad in stainless steel. It is capable of high temperature and high power density operation with extended burnup leading to low fuel cycle costs. A wet control rod design has been incorporated to meet the higher core performance conditions.

The plant utilizes a supercritical steam cycle of 3500 psig/1000°F with a single reheat to 1000°F. The steam generators are once-through modular construction leading to low fabrication costs and increased reliability at the higher steam conditions. The plant utilizes five sodium heat transfer circuits to transfer heat from the reactor to the steam generating system.

The 1000-Mwe plant design is presented as being located on the standard AEC hypothetical site, and the plant arrangement is made accordingly. The reactor description includes the core structure, fuel element, control rod, fuel handling, and core performance. The sodium heat transfer system and turbine generator plant are described and the required auxiliary services are indicated.

A. SITE AND PLANT ARRANGEMENT

The 1000-Mwe plant is assumed to be located on the hypothetical site defined in TID 7025, Guide to Nuclear Power Cost Evaluation.

The reactor plant consists basically of (1) the reactor building, (2) the steam generator structure, (3) the control room area, (4) turbine generator building, (5) the administration building, (6) the warehouse and shop facility.

The reactor plant arrangement is shown on drawings:*

GA001 Reactor Plant Arrangement – Plan Below Operating Floor

GA002 Reactor Plant Arrangement – Plan Above Operating Floor

GA003 Reactor Plant Arrangement – Sections

*All drawings referred to in this section follow IV. F., Technical Data

The reactor building has a high-bay and a low-bay area. The reactor, the fuel-handling system, the primary sodium heat transfer system, the primary sodium service system, the maintenance cell, the cell-cooling equipment, and the radioactive waste disposal area are in the reactor building. The reactor, the primary heat transfer and service systems, and the fuel storage facilities are normally radioactive and are contained in cells mainly below the reactor operating floor, surrounded by concrete shielding. The cells are steel-lined and, during operation, are filled with an inert atmosphere. Access to these shielded areas is provided by removable shield plugs in the floor.

Five intermediate heat-exchanger (IHX) cells are provided, one for each of the five primary loops. The dividing wall between cells affords shielding for access to a loop prior to decay of an adjacent filled loop.

The reactor operating area, the primary sodium pump drives, the fuel-handling and maintenance areas, and equipment are in the high bay of the reactor building. The high bay is adjoined north and west by the elevated portions of the intermediate heat exchanger cells. The auxiliary low bay with the cell-cooling, service, and radioactive waste disposal areas is attached to the east side of the high bay.

The steam generator structure, enclosed by standard sheet metal siding, is attached to the north and west ends of the reactor building in a L-shaped form. Besides the steam generators, this structure encloses the secondary sodium heat transfer and service equipment. The secondary sodium pumps and expansion tanks are near the top of the structure; and the secondary fill and drain tanks, cold traps, and service pumps are below the reactor operating level. The building heating and ventilating equipment is also in this structure.

The control room area adjoins the reactor high bay and the steam generator structure in the southwest corner of the reactor plant adjacent to the turbine generator room. This area includes the control room, various offices and laboratories at the reactor floor level. The switchgear room, battery room, diesel engine compartment, and instrument repair shop are at grade level. An elevator and a staircase between the steam generator structure and control room will serve both areas.

The turbine generator building adjoins the west end of the control room and also includes the feedwater heaters, pumps, and all other steam power equipment.

The long axis of this building is parallel with the long axis of the reactor building high bay. This arrangement will facilitate future addition of a second plant utilizing present facilities, such as the control room, fuel-handling machine, turbine gantry crane, maintenance cells, and other equipment for both units.

Normal access to all structures is provided through the administration building which houses the offices, chemical laboratory, locker rooms, and personnel monitoring station. Uncontrolled access is provided to all nonradioactive areas. Controlled access through the monitoring station is provided for all potentially radioactive areas.

B. REACTOR

1. Description

The reactor is fueled with uranium carbide, cooled by liquid sodium and moderated and reflected by graphite, thereby operating in a thermal neutron spectrum. The rated core output is 2350 Mwt. The reactor is housed in a steel-lined concrete cavity in the operating floor of the reactor building. The cavity is capped with the loading face shield consisting of 7 ft of dense concrete and reinforcing steel, which forms a continuation of the operating floor over the reactor and provides radiation protection for operating personnel.

Beginning at the outer perimeter of the concrete cavity and moving radially inward, the reactor structure consists of: (a) cavity liner, (b) thermal shield, (c) reactor vessel and its external liner, and (d) calandria vessel and internals. The reactor structure is shown on Drawing RA001. Each component of the reactor structure and its function are described in more detail in B.1.a.

The calandria vessel contains the reflector and moderator graphite and is penetrated by stainless-steel process channels which provide a flow path through the core for the sodium coolant. Within the process channels are located the fuel elements and control elements. Above the calandria vessel is the upper sodium plenum, an inert gas blanket, reflective insulation, and the loading face shield; below the vessel is the sodium inlet plenum.

The active core contains 595 fuel elements and 84 control elements, and forms a right cylinder 25.1 ft in diameter and 14 ft high. Initial fuel enrichment is 3.60 wt % U^{235} . The core is surrounded by a 2-ft thick axial and 2.25-ft-thick radial graphite reflector. Design data for the core are tabulated in Table 14.

TABLE 14

REACTOR CORE DESIGN DATA

Thermal power rating (Mwt)	2350
Active diameter (ft)	25.1
Active height (ft)	14
Fuel elements (number)	595
Control elements (number)	84
Lattice spacing (in. ²)	10.25
Core loading (Mt U)	73.8

Primary sodium is brought to the reactor at 750°F through a double-walled pipe which enters the bottom of the reactor vessel. Sodium flows upward into the inlet plenum, then through the process channels and fuel elements, and into the upper plenum where it is removed from the reactor vessel through a single outlet line. Mixed mean sodium outlet temperature is 1150°F. From the reactor vessel it flows through a sodium-to-sodium intermediate heat exchanger (IHX) where the heat assimilated in passing through the core is transferred to secondary sodium. The primary sodium then returns to the reactor to repeat the cycle.

Power production in the core is varied by control rod movement according to the turbine-generator load demand. A change in heat removal requirements of the core is met by varying sodium flow rates while maintaining the reactor ΔT approximately constant. Reactor system response time to load changes is shorter than for the conventional part of the plant, and therefore is not limiting.

Nuclear instrumentation for power operation is located in thimbles outside the reactor cavity liner. Incore startup nuclear instrumentation is placed within thimbles located in the process channels. This instrumentation is temperature-limited and is used during initial startup and testing when the reactor coolant temperature does not exceed 600°F.

Fuel element transfer and movement of other core components are performed with a fuel-handling machine carried on a gantry crane which travels over the reactor on rails embedded in the reactor room floor. The machine consists of component handling mechanisms, a control console, lead shielding,

and support equipment to cool a spent fuel element and maintain an inert gas atmosphere in the machine. Storage facilities are provided for both spent and new fuel.

The fuel management program is based on an equilibrium cycle average fuel burnup of 25,000 Mwd/MTU. Refueling of one-fourth of the core is required every 6 to 7 months. Shutdown time required for a refueling is approximately 10 days.

a. Core and Reactor Structure

The loading face shield for this reactor does not have the rotational requirements which were necessary on the Hallam reactor concept. For this reason it is fixed and becomes simply the upper closure for the reactor cavity and the reactor vessel. All components of the reactor core are removable and replaceable through the fuel holes.

The shield weldment consists of a plate penetrated by 944 8-in. -ID sleeves spaced on a 10.31-in. pitch. The plate forms the bottom of the shield. Its outer edge is welded directly to the top of the cavity liner walls. The plate is reinforced with vertical ribs which pass in one direction only between the rows of sleeves. Proper sleeve alignment and lattice spacing is held by the plate at the bottom of the sleeves and by contact between sleeve castings at the top of the sleeves. High density concrete is placed on the weldment and between the sleeves to form the shield. Nitrogen cooling coils are embedded in lead on the top surface of the plate so that the concrete temperature can be maintained below 150°F. Reflective insulation is incorporated below the shield to reduce the heat flow from the reactor. The loading face shield is 7 ft 4 in. thick and spans the 42 ft 0 in. diameter cavity.

The reactor cavity is a steel-lined pit which is shielded from the plant process system by 5 ft of concrete. The heat which is generated in the liner and in the concrete from the reactor is dissipated by a low pressure nitrogen duct system which is attached to the liner. The cavity liner and the nitrogen ducts are made from low carbon steel. The leak integrity of the liner is that attained by the normal soap bubble leak test. The duct system is not sealed.

The thermal shield is made up of two laminations of 2-in.-thick carbon steel plate stock. It is used in the flat form rather than being forged into cylindrical shape. Very little machining is required on the stock as received from

the mill. The thermal shield is supported on a steel structure with H-beams as columns at the joints between plates. Twelve panels make up the full circle of shielding.

The thermal insulation is fastened directly to the outside of the thermal shield. The insulation is 12-in. thick. It extends from the bottom of the loading face shield to the bottom of the cavity liner. The cavity liner floor is also covered with this insulation.

The reactor vessel provides a container for the calandria core and the sodium coolant. It is made of 1-3/4-in. -thick austenitic stainless steel and is 47 ft high with a 34-ft OD. The reactor vessel is supported from the loading face shield by hangers fastened to its upper edge.

There are two large nozzle openings into the vessel, one 70-in.-diameter sodium inlet and the other a 76-in. diameter for main sodium outlet. The five primary process loops are taken off this larger line outside of the reactor shielding. The inlet nozzle enters the vessel through the axial center of the bottom head; the outlet nozzle enters through the side of the vessel 5 ft above the calandria top plate. A cylindrical outlet manifold is provided around the upper plenum which uniformly distributes the outlet sodium flow. In addition, there are other smaller penetrations for fill, drain, and venting purposes.

Under normal operating conditions the reactor vessel contains sodium to a level about 15 ft above the calandria top plate. The lower plenum area operates at 750°F and the upper plenum at 1150°F. Maximum operating pressure in the vessel is the static head of sodium, which is about 16 psig.

The purpose of the reactor vessel external liner is to provide secondary containment for the sodium in the reactor vessel and thus maintain the sodium level above the core in the event of a leak in the reactor vessel. The liner is made of 3/8-in. stainless steel and is fastened and supported at the top of the reactor vessel. The space between the liner and vessel is 5/8 in. and is filled with helium. The liner is shaped around the nozzle openings and its extensions connect to the guard pipes. A line is provided to drain the space between the reactor vessel and the liner.

The primary purpose of the calandria vessel is to provide support and containment for the graphite moderator and reflector. The calandria and

the process channels provide the means for separating sodium and graphite, while utilizing a minimum of structural steel within the active portion of the core.

The calandria vessel is 32 ft in diameter, 28 ft in height, and has an integral elliptical lower head. It is constructed from 1-in.-thick, Type 304 stainless steel, and is supported from hanger rods fastened to the under side of the loading face shield. The top of the vessel is capped by a 1/2-in.-thick top plate which is also supported from hanger rods fastened to the loading face shield. A 1-in.-thick bottom plate is located at the intersection of the cylindrical vessel wall and elliptical lower head, and is supported and fastened to the lower head by vertical rods.

The lower head and the bottom plate of the calandria provide a pressure plenum for the inlet sodium coolant. Since the dynamic pressure is contained in this plenum, the reactor vessel and the calandria outer shell are relieved of all the static sodium pressure and the cover-gas pressure which is essentially atmospheric. Normal external operating pressures are 5 psig on the top plate and 33 psig on the bottom plate. Operating temperatures are 1150°F on the top plate and 750°F on the bottom plate.

The moderator and reflector utilize nuclear grade graphite logs. Three types of logs are used: (1) a process tube center log, cylindrical in shape and with a 4.300-in.-diameter hole through its center to accommodate a process tube, (2) a reflector center log, also cylindrical in shape, with a 2-in.-diameter hole through its center to permit a handling tool to reach its lower end, and (3) a filler log nearly square on three sides, and cylindrical and concave on the remaining side to mate with a center log.

The graphite unit cell which is repeated throughout the core is made up of four filler logs and either a process tube center log or a reflector center log. The process tube center log is assembled with and is part of the process tube.

The five graphite logs needed to make up a unit cell can be removed and replaced through the loading face shield hole associated with the cell. The replacement of process tubes and graphite is not expected to be a periodic maintenance item; however, the design provides for such maintenance.

Within the calandria vessel, the space between the graphite logs is filled with helium. The helium is used to provide a medium in which sodium vapor can be detected to indicate possible sodium leaks into the calandria vessel. The helium is circulated to an externally located unshielded sodium vapor detector. The circulating loop is tied into the plant helium supply system and the radioactive vent.

There are 680 process channels installed in penetrations through the moderator graphite, and 264 braze-joint sealed access holes in the reflector region. The process channels house the fuel elements and control elements, and provide a flow channel for the sodium coolant through the core. The process channel is divided into three functional parts: (1) the process tube through the core, (b) the orifice tube which is an extension of the process tube above the core and which provides flow control, and (3) the fuel shield plug and orifice casting which extends down from the loading face shield into the orifice tube.

The process tube is a 4.300-in. -OD by 0.025-in. -wall Type 304 stainless steel structure. It is installed with brazed joints at both ends to allow remote removal and reinstallation of process tubes and graphite. The calandria vessel is designed so that any individual process tube or group of process tubes and associated graphite can be removed and replaced. Each process tube has a bellows near the calandria top plate just below the brazed joint to accommodate differential expansion between tubes and calandria shell, and between adjacent tubes operating at slightly different temperatures. The top of the process tube also incorporates a guide tube which protects the bellows from stress caused by sodium flow or external lateral loads.

Flow through the fuel process channels is controlled by orifice tubes which extend up from inside the process tubes and are suspended from the loading face shield. This tube rests on the step of the opening through the loading face shield and makes a sliding fit into the process tube below. The orifice tube is the stationary member of the variable orifice and contains two opposing 1/2 in. slots 40 in. above the calandria top plate. These slots match up with triangular openings in the orifice casting extension of the fuel shield plug assembly which rotates inside the orifice tube to produce an adjustable flow area through which the sodium coolant discharges from the process channel into the reactor upper plenum. The rotating fuel shield plug and orifice casting are fastened together

below the loading face shield and function as one piece. This component is the adjustable portion of the variable orifice.

Coolant flow adjustments are made at the operating floor by a rotating device designed specifically to provide a vernier-type control over the range of rotation of the plugs. Adjustments are necessary only as required by changes in core power distribution.

The fuel shield plugs are equipped with quad-ring seals to provide a gas-tight seal to the top shield. There are two rings per plug. The second is for backup purposes and both are replaceable. Reflector shield plugs are similarly sealed. Thermocouple leads pierce the fuel shield plugs and extend to the orifice area to measure the sodium coolant temperature. In addition, each plug contains a small tube from which samples of reactor cover gas contained in the orifice tube can be monitored for fission products. By this means, a ruptured fuel element can be detected and located.

A single axial sodium inlet line supplies the coolant to the calandria plenum, where it is uniformly distributed to the process channels by a distribution nozzle. The sodium passes up through the core and exits via the orifice into the upper plenum. From the upper plenum it flows into a single outlet line through a perforated box header. The header eliminates vortexing and creates more uniform radial flow paths from the process channels.

A portion of the sodium is also supplied to the annulus between the reactor and calandria vessels for cooling of the vessel walls. The annular coolant flow is controlled by a valve automatically regulated to maintain uniform temperatures between the upper plenum sodium and the outlet coolant from the annulus. The annulus terminates in standpipes which protrude just above the sodium surface in the upper plenum in order to separate the annulus from possible variations in the free surface level that could cause adverse coolant circulation paths.

All large components of the reactor structure have been designed as field-erected rather than complete shop-fabricated units. The requirements for massive machining operations have been eliminated from the design. Shop-fabricated parts of components can be readily shipped and field-erected with normal erection equipment.

b. Reactor Structure Field Erection

The planned field fabrication and erection sequence for the 1000-Mwe reactor structure are described below.

The reactor structure components will arrive at the site as follows:

- 1) Top and bottom heads of the calandria vessel will be in two pieces. A single field-weld joint across the diameter will be required. Hole patterns will be machined in pieces for field welding of brazed-joint fittings. All other fittings will be shop-welded to heads, except those interfering with the field-weld. The halves of the bottom head will include the shell portions of the elliptical head with the staybolts in place. The nozzle section of the elliptical head is shop-fabricated for field welding to the head. The vessel courses will arrive as formed rectangular plates prepared for field-welding.
- 2) The reactor vessel and liner heads will also arrive in halves and the vessel courses as formed rectangular plates. The nozzle will be shop-fabricated for field-welding. The large outlet nozzle is shop-welded into its vessel plate.
- 3) The sodium outlet manifold is shop-fabricated but is shipped in four sections. The large outlet nozzle liner is shop-welded into one of these sections.
- 4) The loading face shield bottom plate sleeves and reflective insulation will be shop-fabricated for final assembly at the site. The bottom plate will be fabricated in three sections. The stiffeners will be welded to the sections prior to matching the hole pattern for the sleeves. Two longitudinal welds will be required in the field for assembly. Loading face shield sleeves will be welded into the bottom plate and the reflective insulation attached in the field before putting the bottom plate in place above the reactor vessel.
- 5) The thermal shield slabs and supporting structures are all shop-fabricated.

- 6) All material for the cavity liner will be shop-formed rectangular plates and sketch plates. Field fabrication will include the gas-cooling ductwork.

Reactor structure components will be field-erected in the following sequence:

- 1) Erect the cavity liner in place with concrete surrounding it to an elevation of 90 ft 6 in. (see Drawing RA001). During erection of the inlet pipe tunnel, the inlet pipe and guard pipe are placed in position.
- 2) Construct a temporary structure from which the reactor can be supported on the cavity liner floor.
- 3) Weld the halves of the reactor vessel head and reactor vessel liner head together, lower the assembled head onto the temporary support structure.
- 4) All weld passes will be qualified by dye-penetrant and x-ray-checked according to ASME Code, Section VIII.
- 5) Connect the inlet pipe and guard pipe to the reactor vessel and reactor vessel liner.
- 6) Complete fabrication of reactor vessel by lowering the pre-formed plates into the cavity and welding in place.
- 7) After reactor vessel has been completed and inspected, assemble and weld the reactor vessel liner around the reactor vessel.
- 8) Install the gas-cooling ducts on the interior of the cavity liner.
- 9) Install the thermal shield structure and the thermal shield.
- 10) Install thermal insulation to thermal shield.
- 11) After completion of field fabrication, lower the calandria vessel into the reactor vessel and support on the alignment keys.
- 12) Lower the sections of sodium outlet manifold and vessel liner shields and insulation into reactor vessel for later assembly.
- 13) Field fabricate the loading face shield steel structure (including reflective insulation) and lower into place on the cavity liner. Make seal weld.

- 14) Place dense concrete in loading face shield.
- 15) Install the reactor vessel and liner seals. Install the thermal insulation on reactor vessel liner seal.
- 16) Install the reactor vessel turnbuckle assemblies.
(Do not lift weight of reactor vessel from temporary support structure).
- 17) Install half of the calandria vessel support rods and lift the calandria vessel off the calandria vessel key support.
- 18) Install link bars between reactor vessel and calandria vessel.
- 19) Place half of the sodium outlet manifold into final position.
- 20) Install the remainder of the calandria vessel support rods through the manifold.
- 21) Remove the support rods that were installed in Step 16.
- 22) Finish installation of the sodium outlet manifold and reinstall support rods.
- 23) Install calandria top plate hanger rods and adjust to level the the top plate. Install the process channel top joint.
- 24) Remove the temporary supporting structure from the reactor vessel.
- 25) Install all the graphite in calandria vessel except that which is over the manhole.
- 26) Close the manhole in calandria bottom plate and finish installation of graphite from above.
- 27) Close manhole in the sodium outlet manifold at outlet nozzle location.
- 28) Finish the installation of cooling ducting on cavity liner floor.
- 29) Install the thermal insulation to floor of cavity liner.
- 30) Install the thermal insulation to area below thermal shield.
- 31) Connect the pipe within sodium inlet nozzle for cooling the annulus between the reactor vessel and the calandria.

32) Install the orifice tubes and shield plugs.

33) Install the instrumentation harnesses and lead shot in loading face shield.

c. Fuel Element

The fuel element proposed for the 1000-Mwe SGR consists of a bundle of 18 fuel rods and a dummy center tube to support the fuel rods and fuel rod spacers. The fuel rods are 17 ft 4 in. in length and are made up of 14 ft of uranium carbide slugs assembled in stainless steel tubes. The fuel slugs are 1/2 in. in diameter and vary in length from 1 to 8 in. The slugs are bonded to the tubes by sodium to improve heat transfer. The sodium annulus is 0.025 in. thick and allows for 10% local diametral swelling. A 40-in.-long gas space is provided above the fuel within each tube for expansion and fission gas collection.

The elements are designed to fit inside the 4.250-in. ID process tubes. Although the elements are heavy enough to offset the hydraulic uplift of the coolant, for additional safety they are latched into a seat at the bottom of the process channel. The latching mechanism is of the over-center link bar type. Each of three link bars actuates a dog or tab to engage a detent at the bottom of the process channel. A rod and tube assembly connects the lifting knob at the top of the element with the link bars near the bottom. The fuel-handling machine has a double lift system so that the fuel element can be lifted from the guide assembly to check the holddown latch or by the knob which unlatches the dogs. The weight of the fuel element is taken on a shoulder at the bottom of the channel rather than on the latch detent. All structural material is 304 stainless steel. Design data for the fuel elements are listed in the following Table 15.

d. Control Element

The control element concept for the 1000-Mwe reactor is shown on Drawing RM001. This combination shim-safety rod is referred to as a "wet rod" design because the absorber element is immersed directly in the primary sodium coolant. Because of its better heat dissipation capability, the wet rod structural materials run cooler in reactors with higher specific power.

The design of the rod drive places all component parts below the loading face, making it unnecessary to remove the drives for fuel handling.

TABLE 15
FUEL ELEMENT DESIGN DATA

<u>Materials</u>		
Fuel		Hypo UC 4.6 to 4.9 wt % C
Clad		304 SS
Bond		Na
Enrichment (wt % U ²³⁵)		3.60
Uranium inventory (Kg U/element)		124
<u>Fuel Rod</u>		
Number of rods		18
Fuel slug diam. (in.)		0.500 ± 0.005
Active fuel length (ft)		14
Fuel weight (Kg U)		6.88
Sodium annulus (in.)		0.025
Clad thickness (in.)		0.010
Fuel rod diam. (in.)		0.570
Total fuel rod length (ft)		17.3
<u>Fuel Rod Design Criteria</u>		
Clad strain (%)		0.4
Fission gas release (%)		3
Na outlet temperature (max. °F)		1150
Hot channel fuel temperature (max. °F)		2000
Burnup average (MWD/Mtu)		25,000
Design average fuel slug diam. increase (%)		5.3
Design max. fuel slug diam. increase (%)		8.0

This also simplifies decontamination and maintenance of the loading face area and manual adjustment of the variable orifice.

The parts which operate in sodium are the latch mechanism, the splined shaft, the drive screw, and the roller nut to which the absorber column is attached. For normal shim rod movement the rod is latched and the drive screw is rotated by the drive socket. This raises and lowers the roller nut (and absorber column) in its guide slots. The roller nut does not rotate.

To scram the rod the latch mechanism is released by the pull tube and magnet plunger, allowing the drive screw, roller nut, and absorber column to drop together as one assembly. The splined shaft remains attached to the drive shaft as the drive screw slips down. Snubbing action at the bottom of the fall is obtained by a piston and cylinder section located between the roller nut and poison column.

After scram, the rod is latched for normal operation by rotating the shaft to cause the drive screw to raise against the roller nut until the latch cup engages the latch. The holding magnet is energized to raise the pull rod and set the latch. The absorber column can then be raised by rotating the drive screw. The 156-in. travel of the drive screw will raise the absorber to the top of the active core region.

The reactor atmosphere is separated from the drive section by a lubricated O-ring seal where the rotating shaft enters the drive section. This seal is maintainable. All points of rolling or sliding friction are either in the liquid sodium or in the ambient atmosphere of the drive section. There are no bearing surfaces in the sodium vapor atmosphere of the upper plenum. This feature avoids the collection of sodium oxide and condensate at critical points.

The absorber section of the control rod is made up of 18 tubes filled with Dy_2O_3 at a density of 5.5 gm/cc. The tubes have a 5/16-in. OD with a 0.010-in. wall thickness. They are 13 ft 2 in. long. These absorber-filled tubes or pins are placed in the annulus between two larger tubes. Sodium coolant is allowed to fill the volume not occupied by the absorber pins.

The control rod assembly is placed in a standard process channel, and is cooled in the same manner as the fuel except that the sodium coolant flow is controlled by a fixed orifice at the bottom of the channel rather than by an adjustable orifice at the top.

e. Fuel-Handling System

In the fuel-handling procedure, an irradiated fuel element is taken directly from the active core into a lead-shielded cask which is an integral part of the fuel-handling machine. The machine then transports this fuel element either to another process tube or to the spent fuel storage pool. This is the same design philosophy successfully demonstrated in SRE and Hallam. Newly

fabricated fuel is transferred to the fuel-handling machine from a pickup cell. This cell is not expected to handle radioactive fuel or to become contaminated.

The fuel-handling machine consists of a lead-shielded cask 60 ft high which incorporates fuel-element and shield-plug handling grapples and other auxiliary handling components, a fuel-element cooling system, and an integral control console. The machine is mounted on a gantry spanning the high bay area of the reactor building and travels over the length of that area on rails embedded in the floor. The gantry and all cask operations are controlled from the control console which is also mounted on the gantry.

The cask contains three cylindrical storage barrels, two for handling fuel elements and one for handling shield plugs or a control rod and thimble. The fuel-handling grapples can be extended to the top of a fuel element where they can latch to or from the fuel element and actuate the latch at the base of the element. The shield plug grapple reaches only to the reactor loading face where it latches to or from a shield plug or control rod thimble.

The fuel-handling barrels each contain a guide tube which is lowered through the loading face shield with the grapple to the top of the process channel during the fuel transfer operation. The fuel element is drawn into the guide tube and the guide tube is then drawn into the fuel-handling cask by the grapple.

The fuel-handling machine has a helium cooling system capable of removing 35 kw from each of the fuel elements contained in the cask. The fuel element is cooled while in the guide tube by passing cool helium down the annulus between the orifice tube and the guide tube. On the downmost position of the guide tube, the helium passes into the guide tube through holes just above the sodium level and then passes around the fuel rods and up the inside of the guide tube to a manifold outlet. It is then piped to a sodium vapor separator, heat exchanger, and helium blower for return to the cask. The blower system operates whenever an irradiated fuel element is loaded, discharged, or in the cask. Heat is transferred from the helium to the reactor room atmosphere by use of a 5 - hp air blower. Two such systems are required for normal operation, and a third system is provided for standby purposes.

Three independent drives are used to raise or lower the grapples. Two of the units are identical for fuel-element handling, while the third has a larger capacity to accommodate the shield plug and control rod. Two motors

are provided for each drive unit. The second motor in each case is used for standby purposes only.

The operation of all three drive units is identical. The grapple is supported and actuated by two tapes. These tapes travel over an idler wheel assembly to the driver pulleys. The pulleys are driven by a brake motor through an integral eddy current coupling.

The idler wheel assembly consists of two free-spinning pulleys mounted on an eccentric shaft. A 180° rotation of the shaft displaces the pulleys vertically. This action displaces the tapes, raising one slightly and lowering the other, thereby actuating the grapple. The idler assembly is supported by two load-cells which register the load on each tape. An upper and lower limit on the load-cell automatically stops the drive system.

Spent fuel is stored below the operating floor on a rotating rack in a sodium-filled vessel. Decay heat is removed by a convective NaK system which jackets the storage vessel. The sodium is not circulated. The spent fuel storage is serviced by the fuel-handling machine in much the same manner as in the reactor.

New fuel can be stored either in the spent-fuel storage vessel or in the new fuel storage room which is located below the operating floor. New fuel must be transferred to a pickup cell with the building crane before the fuel-handling machine can accept it.

The control-rod storage facility consists of 90 thimble sleeves suspended in a shielded cavity below the reactor floor. Control rods removed from the reactor with the fuel-handling machine are stored in these thimbles under a helium atmosphere. Cooling of control rods in storage is not required.

In addition to control rod storage, the cavity contains special sleeves for miscellaneous reactor maintenance tools and reactor source element storage.

2. Core Performance

The primary nuclear and heat transfer characteristics for the reactor core and fuel elements are given in Tables 16 and 17.

The core produces a nominal thermal power of 2350 Mw. There are 595 fuel elements and 84 control rods. The mixed mean coolant outlet temperature is 1150°F, and the coolant inlet temperature is 750°F. The maximum

TABLE 16
CORE PERFORMANCE DATA

Thermal power (Mwt)	2350
Average specific power (kw/kgU)	32.0
Overall maximum to average power ratio	2.25
Total core coolant flow (lb/hr)	66.0×10^6
Core pressure drop (psi)	21.0
Mixed-mean core outlet coolant temperature (°F)	1150
Inlet coolant temperature (°F)	750
Reactivity control requirements; hot, clean core ($\% \Delta k/k$)	
Burnup	8.0
Equilibrium xenon	2.5
Equilibrium samarium group of fission products	0.9
Xenon transient override	0.7
Total	12.1
Reactivity worth of control elements ($\% \Delta k/k$)	16.0

TABLE 17
FUEL ELEMENT PERFORMANCE DATA

Initial enrichment (wt % U ²³⁵)	3.60
Final enrichment (wt % U ²³⁵)	1.45
Net plutonium production (g Pu/kgU)	6.78
Average fuel burnup (Mwd/MTU)	25,000
Average incore residence time - 80% p.f. (yr)	2.68
Maximum element power (Mwt)	5.95
Maximum to average element power ratio	1.50
Maximum linear heat rate (kw/ft)	33.25
Maximum specific power (kw/kgU)	72.0
Maximum nominal heat flux at clad OD (Btu/hr-ft ²)	760,000
Maximum hot channel clad temperature (°F)	1,250
Maximum hot channel fuel temperature (°F)	2,000
Maximum element flow rate (lb/sec)	46.8
Maximum coolant velocity (ft/sec)	14.2

fuel centerline temperature is 2000°F and the maximum inside surface cladding temperature is 1250°F. Approximately 94% of the core power is produced in the fuel and 6% outside the fuel. Almost all of the power generated in the core is removed by the coolant in the process channels. The hot-channel coolant velocity is 14.2 ft/sec and the core pressure drop is 21 psi across the fuel element and orifice.

The feed enrichment for fuel elements inserted into the equilibrium cycle core is 3.60 wt % U^{235} . Discharged fuel elements have an average exposure of 25,000 Mwd/MTU, an enrichment of 1.45 wt % U^{235} , and a Pu concentration of 6.78 gm Pu/Kg U. The steady-state fuel management program calls for a refueling shutdown every 6.4 full-power months. At this time, 1/4 of the fuel elements in the core (148 elements) will be replaced and 50 elements will be relocated within the core. The average incore residence time for a fuel element is 2.68 years at 80% plant factor. The total reactivity control requirements for the equilibrium core are 12.1 % $\Delta k/k$. The 84 control rods provided give a total reactivity control capability of approximately 16 % $\Delta k/k$.

The problems of reactor power level control in a large SGR plant will not be appreciably different from those of the smaller SGR plants currently in operation; namely, SRE and HNPF. The principal parameters governing short- and long-term gross power transients are quite similar due to material and geometric similarities in core components, with minor effects attributable to core size. Short-term response is controlled principally by the fuel and fuel coolant temperature coefficients of reactivity whose coupled effect produce the prompt power coefficient. In the current SRE design, the nominal fuel and fuel coolant temperature coefficients are $-1.1 \times 10^{-5} \Delta k/^{\circ}F$, and $+0.5 \times 10^{-5} \Delta k/^{\circ}F$, respectively, to produce a prompt coefficient varying from $-1.0 \times 10^{-5} \Delta k/Mwt$ to $-5.0 \times 10^{-5} \Delta k/Mwt$ over the power range. In the HNPF, these coefficients are essentially identical, differing only in the second decimal values. The HNPF power rating is eight times that of the SRE design, demonstrating the minor, if not negligible, effect of the core size on these coefficients. Values to be expected for these coefficients in a 1000-Mwe design would not differ greatly from those given.

The long-term or quasi-steady-state response of SGR's is determined by all core-component temperature coefficients of reactivity using the combined effects producing the steady-state power coefficient. In the SRE and HNPF

plants, this coefficient becomes positive at reduced power levels, implying steady-state power instability. This is not a limiting characteristic of SGR's, since the major positive contributor to the coefficient is the moderator temperature coefficient of reactivity. As a result of long thermal time constants in delivering the heat generated in the moderator to the moderator coolant, the reactor control system has considerable time in terms of its own time constant to compensate for reactivity disturbances, and reestablish the demand power level. This power coefficient characteristic is also anticipated for large SGR cores.

The principal control problem to be encountered with a large SGR core is the spatial power distribution oscillation or instability characteristic of all large cores due to the many critical mass configurations contained in the core assembly. This condition permits relatively independent power operation in regions of the core by local reactivity imbalances generating power oscillations axially, radially, or azimuthally. The likelihood of such oscillations is further enhanced by high flux levels, large core size, flat power distribution, and weak power coefficients.

The analytical treatment of spatial Xe^{135} oscillations in a large power reactor is a subject in need of further study and exploration. Approximate methods exist to determine the threshold for such oscillations, if the temperature coefficients are known. Furthermore, proper manipulation of control rods can minimize the effects of such oscillations and keep them under control. However, no methods are currently available to determine the economic penalties associated with controlling spatial Xenon oscillations. Additional reactivity, control rod worth, and fuel elements may be required, due to increased peaking factors to minimize and control these transients.

C. SODIUM HEAT TRANSFER SYSTEM AND EQUIPMENT

The sodium heat transfer system removes the heat generated in the reactor core and delivers it to steam generators for steam production. Heat transfer is accomplished by circulating a liquid sodium heat transfer medium in five parallel sodium circuits. Each of these circuits consists of a radioactive primary loop which transfers thermal energy from the reactor to the intermediate heat exchangers, and a nonradioactive secondary loop which carries heat

from the intermediate heat exchangers to the steam generators. The intermediate heat exchangers are utilized to separate the radioactive sodium of the primary loop from the steam generators. This is done to (1) permit access to the steam generating facilities during operation, and (2) protect the reactor from pressure and temperature surges which could result from a steam or water leak in the steam generator.

Each of the five circuits incorporates separate steam generators, which consist of sodium-heated low-temperature modules, sodium-heated high-temperature modules, and sodium-heated reheaters. The system has a nominal capacity of 2350 Mwt. Each circuit transfers 470 Mwt. A diagram of the sodium heat transfer system is shown in Drawing SA001. Major characteristics of the system are tabulated in Table 18.

The intermediate heat exchangers and steam generators are arranged vertically and elevated with respect to the core so that a natural convection head will exist in each circuit to remove decay heat in case of power loss or other malfunction.

The sodium purification will be performed by cold traps and hot traps. Separate purification systems are provided for the radioactive and nonradioactive sodium system.

1. Primary Heat Transfer Loops

The primary heat transfer system takes suction from the pool of sodium above the core in the reactor vessel. The sodium in each loop is pumped through the intermediate heat exchanger and through the reactor by a 33,000-gpm centrifugal pump. The pump is driven by a 2250-hp motor converted to a variable-speed drive for flow control. The pump motor and speed drive will be horizontally mounted because of its size.

The primary sodium pump is a vertical free-surface, single unit consisting of an upper barrel assembly which contains both shielding and thermal insulation, an expansion space for sodium to rise into the pump barrel, a hydraulic radial bearing, and the pump impeller and diffuser. The entire pump inner assembly is arranged so that it may be removed without disturbing the outer pump casing or piping connections. The upper radial and thrust bearings are conventional bearings, cooled and lubricated by the pump seal oil system. A

TABLE 18

1000-Mwe SODIUM HEAT TRANSFER SYSTEM

<u>Primary System</u>		
Number of loops		5
Capacity of each loop (nominal Mwt)		470
Sodium temperature (°F)		
Reactor inlet		750
Reactor outlet		1150
Total sodium flowrate (lb/hr)		66.0×10^6
Pump capacity per loop (gpm)		3.3×10^4
<u>Secondary System</u>		
Number of loops		5
Capacity of each loop (nominal Mwt)		470
Sodium temperature (°F)		
Steam generator inlet		1060
Steam generator outlet		660
Total sodium flowrate (lb/hr)		66.0×10^6
Pump capacity per loop (gpm)		3.3×10^4
<u>Intermediate Heat Exchanger</u>		
Type construction		Modular shell and tube unit
Number of units per circuit		2
Duty per unit (Btu/hr)		8.75×10^8
<u>Steam Generators</u>		
High-temperature module		
Type of construction		Shell and tube
Number of modules per circuit		20
Duty per module (Btu/hr)		36.1×10^6
	<u>Tube</u>	<u>Shell</u>
Fluid	Steam	Secondary Sodium
Flow per module (lb/hr)	72,400	481,000
Temperature in (°F)	725	1060
Temperature out (°F)	1005	833
Outlet steam pressure (psia)	3600	—
Low-temperature module		
Type of construction		Shell and tube
Number of modules per circuit		28
Duty per module (Btu/hr)		27.2×10^6
	<u>Tube</u>	<u>Shell</u>
Fluid	Water/steam	Secondary Sodium
Flow per module (lb/hr)	51,700	471,500
Temperature in (°F)	512	833
Temperature out (°F)	725	660
Inlet feedwater pressure (psia)	3800	—
Reheater module		
Type of construction		Shell and tube
Number of modules per circuit		6
Duty per module (Btu/hr)		44.7×10^6
	<u>Tube</u>	<u>Shell</u>
Fluid	Steam	Secondary Sodium
Flow per module (lb/hr)	169,000	600,000
Temperature in (°F)	603	1060
Temperature out (°F)	1005	833
Outlet steam pressure (psia)	705	—

double mechanical seal is used at the upper, cold end of the pump for atmospheric separation.

The pump suction lines of the main primary loops are 30-in. pipe coming from a common 76-in. header. The discharge lines are 28-in. pipe. The cold returns are 26-in. pipe returning to a common 70-in. header. Piping material is Type 304 stainless steel. The piping is routed to and from the reactor in such a way as to reduce neutron streaming into the equipment spaces and resultant activation of equipment. A high point is provided in the suction and return piping for each loop, which is above the reactor core and reactor vessel outlet nozzle, to permit draining of one or more loops without draining the remaining loops. This feature allows operation of the reactor with less than five loops, removal of decay heat during shutdown with one loop, and access after a decay period to the four loops without removing fuel from the reactor. The suction and return piping is double-walled from the reactor to the high point in the piping system, so that, in the event of a major line rupture, exposure of the core is prevented.

Check and butterfly valves are provided in the return piping to the reactor. Because of the parallel operation of the pumps in the five loops, the check-valve is necessary to prevent flow reversal in the event of loss of a pump or mismatch of pump speeds. The throttle-valve is used for flow control following a scram in order to prevent overcooling of the reactor and rapid loss of the reactor temperature gradient.

Two intermediate heat exchangers (IHX) operate in parallel in each primary circuit. The heat exchangers are vertical shell-and-tube modules approximately 49 in. in diameter and 28 ft long. Each unit has a surface area of 9890 ft². An expansion joint in the shell allows for differential thermal expansion between the shell and tube bundle.

2. Secondary Heat Transfer Loops

Each of the five secondary heat transfer loops has the same capacity as the primary loops. In each secondary loop, sodium is pumped in a closed circuit through the shell side of the intermediate heat exchanger and steam generator units. An expansion tank floats on the line to absorb volume changes in the sodium due to temperature changes, and to remove any gases that have accumulated in the system.

The secondary sodium pumps are of the same type as the primary sodium pumps. The piping in the hot-leg of the main loop (IHX to steam generator) is 28-in. pipe, and the cold-leg is 26-in. pipe.

3. Steam Generator

In the steam generator, the secondary sodium transfers its heat to water and steam to produce superheated steam at turbine throttle conditions of 3500 psig and 1000°F. The steam generator is made up from three types of modular units: (1) low-temperature modules, (2) high-temperature modules, and (3) reheater modules. These units are mounted vertically in a once-through design with sodium on the shell side, and steam on the tube side. Secondary sodium enters the top of the high-temperature and reheater modules, and flows downward through the shell. The sodium leaving the reheater modules mixes with the sodium leaving the high-temperature module, and the total combined sodium flow enters the low-temperature modules. The sodium exits from the low-temperature modules and returns to the intermediate heat exchanger.

The steam generator modules for each heat transfer circuit are arranged in a circular array, with high-temperature modules above the low-temperature modules. The reheater modules are located at the side of the high-temperature modules. The modules are mounted vertically in order to minimize horizontal space requirements in the plant and to avoid sodium stratification problems during free convection flow. The steam generator modules and the steam, water, and sodium header systems are supported in a steel structure which is enclosed by sheet metal siding. Pressure relief headers run vertically from the modules to pressure relief tanks.

Steam generators in each heat transfer circuit consist of 28 low-temperature modules, 20 high-temperature modules, and 6 reheater modules. The high-temperature and low-temperature modules are similar in size and construction, being nominally 8-in. in diameter and utilizing 61 tubes. The reheater module is similar in construction but is larger, with a nominal diameter of 16 in. and with 110 tubes.

The low-temperature module tubing is standard-drawn, 2-1/4 % chrome-1% moly, and the high-temperature module tubing is Type 321 stainless steel. The tubing in both units is 5/8 in. OD with a 0.095-in. wall thickness; reheater tubing is 1-in. OD with a 0.049-in. wall.

Bosses are bored in the tube sheets to form attachments for the tubes, The tubes are then welded to the tube sheet using "inside out" techniques now well developed by several manufacturers. This type of tube-to-tube sheet attachment eliminates the crevice formed in a rolled-and-welded joint. End-bells for feedwater and steam are formed by welding forgings to the tube-sheet forgings. The tube bundle is enclosed by a shell of standard piping with sodium inlet and outlet nozzles formed by standard piping tees. To minimize thermal shock during sodium temperature transients, an inert-gas blanket is maintained at the top of the module and a series of thermal baffles is provided at the bottom.

Each high-temperature, low-temperature, and reheater module has a hydrogen detector mounted in the sodium outlet piping to detect possible water leakage into sodium. In the unlikely event of a leak, it will normally be detected and the affected bank shutdown before the leak becomes significant. However, in the event of a large leak, rupture discs are provided to vent the effluent products. The gaseous products are vented from the blowdown stack and the sodium is collected in pressure-relief tanks.

D. TURBINE-GENERATOR PLANT

Supercritical advanced reheat cycle steam conditions of 3500 psig/1000°F/1000°F were selected for the proposed 1000-Mwe plant, which results in a station efficiency of 43.6%. The cycle characteristics are illustrated on the heat balance shown in Drawing SA011.

1. Turbine-Generator Unit

The rated capability of the main cross-compound turbine-generator is approximately 1,050,603 kw with turbine steam conditions of 3500 psig/1000°F/1000°F and a turbine back-pressure of 1-1/2-in. Hg abs. The maximum expected capability is approximately 1,110,000 kw with throttle valves wide open and the same steam conditions and back pressure. Other turbine-generator performance data are indicated in Table 19 based on operating with eight stages of regenerative feedwater heating as illustrated in Drawing SA011. The estimated net turbine cycle heat rate for the 1,000,000-kw turbine selected for the 1000-Mwe plant is shown in Figure 12.

The turbine is a cross-compound, four-flow unit with 43-in. last-stage blading. Steam from the high-temperature superheater modules of the steam

Figure 12. Net Turbine Cycle Heat Rate vs Load for 1,000,000-kw CC4F-43 in. LSB Turbine

TABLE 19
TURBINE-GENERATOR DATA

Type	CC4F-43 in. -3600/1800 rpm
Gross electrical output, rated (kw)	1,050,603
Gross electrical output, max. (kw)	1,110,000
Net turbine cycle heat rate (rated capability)(Btu/kwh)	7514
Throttle flow (lb/hr)	7,236,600
Reheat flow (lb/hr)	5,092,300
Main steam pressure (psia)	3515
Reheat steam pressure (psia)	691
Initial and reheat steam temperature (°F)	1000
Turbine back pressure (in. Hg abs.)	1-1/2
Short circuit ratio	0.64
Exciter ratings, each (kw)	2410
Exciter voltage (vdc)	500
Generator ratings (kva at 45 psig H ₂ each @)	640,000
Generator terminal voltage (preferred)(kv)	24.0
Power factor	0.85

generator enters the high-pressure turbine and exhausts as cold reheat at 768 psia and 603°F. The hot-reheat steam leaving the reheater modules enters the intermediate pressure turbine, crosses over to the four-flow low-pressure turbine and exhausts to the condensers.

The high-pressure and intermediate-pressure turbines are sealed at the turbine shaft penetration through each turbine shell element by steam against leakage to the atmosphere by means of labyrinth-type seals. A steam supply for seal steam during plant startup is provided. Steam-seal auxiliary equipment includes an automatic steam-seal pressure regulator, a gland-seal condenser with motor driven exhauster, and remote indicating instrumentation for steam-seal pressure.

The two generators are hydrogen-cooled, each rated 640,000 kva at 45-psig H₂ pressure. The output of the 3600-rpm high-pressure unit is 550,793 kw; output of the 1800-rpm low-pressure unit is 499,810 kw.

A turbine lube oil conditioning system is provided consisting of a dry-type oil conditioner including a precipitator, filter, and storage compartments, circulating oil pump, transfer oil pump, and accessories. The system can be used on a continuous or a batch type of operation.

2. Condensing System

The four-flow low-pressure turbine is connected to 335,000 ft² of horizontal, single-pass condenser, comprising two separate 167,500 ft² condensing sections, with a separate hotwell for each section. The condensers will contain 7/8-in. -OD 18-bwg Admiralty tubes, of 38-ft overall length rolled into Muntz metal tube sheets. The units have a fabricated steel shell and fabricated steel water boxes, lined with Epoxy. Performance is based on 85% cleanliness factor and a water velocity of 7.5 fps. Additional condenser data are provided in Table 20.

TABLE 20
CONDENSER DATA

Steam flow to condenser (total lb-hr)	4,516,000
Condenser duty (total Btu/hr)	4,290 x 10 ⁶
Condenser vacuum (in. Hg abs.)	1.5
Circulating water inlet temp. (°F)	57
Circulating water flow (total gpm)	429,000
Circulating water temp. rise (°F)	20

Four package-type, motor-driven rotary vacuum pump units are provided for air removal. They have a holding capacity of 7-1/2 cfm at 1 in. Hg abs., and a hogging capacity of 900 cfm at 15 in. Hg abs.

Circulating water to condense the 4,516,000 lb/hr of steam entering the condenser from the low-pressure turbine exhaust at rated load is supplied by four 107,250 gpm, 30-ft tdh vertical, single-stage mixed-flow circulating water pumps. Each pump is driven with a 900-hp, 235-rpm motor, each pump discharge through a 78-in. circulating water line to the condenser water boxes.

3. Condensate System

Three half-capacity 1800-rpm, vertical-type condensate pumps are provided, each producing a flow of 5600 gpm at a tdh of 350 ft. The condensate pumps discharge from the condenser hotwell through the condensate scavenging

demineralizers and then through two banks of four closed low-pressure U-tube feedwater heaters into the deaerator. Each bank consists of four 50%-capacity feedwater heaters.

Full-flow, mixed-bed demineralizing equipment for condensate scavenging is provided for the once-through steam generator design. The demineralizer units provide high quality feedwater and protect the steam generator and main turbine, in the event of a condenser leak, from impurities and carryover with resultant scale deposition, loss in temperature, and loss in turbine efficiency.

4. Feedwater System

Two-half-capacity, steam-turbine-driven boiler feedwater pumps take suction from the deaerator and discharge through a double bank of three closed high-pressure U-tube feedwater heaters into the low-temperature modules of the steam generator. Each bank contains three 50%-capacity high-pressure feedwater heaters.

Each boiler feedwater pumping unit consists of a 6000-rpm pump, driven by an eight-stage dual pressure noncondensing steam turbine with variable speed control for flow control. Additional boiler feedwater pump data are tabulated in Table 21.

TABLE 21
BOILER FEEDWATER PUMP DATA

Pump capacity per unit (gpm)	8000
Feedwater temperature (°F)	325
Pump TDH (ft)	11,400
Steam turbine driver input, each (kw)	16,970

5. Extraction Steam

Eight turbine extraction points are provided for feedwater heating. Reverse current trip valves which automatically close on turbine trip or feedwater high liquid level are provided in the steam extraction lines to the high-pressure heaters.

E. AUXILIARY SERVICES

The following auxiliary services are provided to support both the reactor plant and turbine-generator plant.

1. Service water supply is provided from the river to deliver makeup feed-water and cooling water to the heat-exchangers in the closed-loop cooling water system.

2. Closed-loop cooling water system circulates treated water in two branches, one for each plant. The reactor branch provides cooling of reactor instrument thimbles, cell nitrogen cooling system, reactor loading face shield cooling system, radioactive waste disposal system equipment and other components. The turbine branch serves the turbine-generator hydrogen cooling equipment, lube oil coolers, sample coolers, miscellaneous pump bearing and coupling cooling, air compressors, control room air conditioning units, and other equipment.

3. Instrument and service air systems supply compressed air to instrumentation and service equipment in both plants.

4. Gas supply is furnished for the following four services:

- a) Helium maintains an inert gas blanket in the reactor and in all piping, vessels, or equipment where a free surface of sodium exists in order to prevent oxidation of the coolant and maintain constant gas pressures required for proper operation. Helium is used as the displacement gas during draining, filling, and transferring sodium and other operations.
- b) Nitrogen is used to maintain an inert gas atmosphere in the shielded cells in the reactor building and provides cooling for these cells and the primary equipment therein, as well as for the reactor loading face shield. Nitrogen is supplied for purging and draining operations where helium is not necessary for this purpose.
- c) Hydrogen is supplied as required to the generator coolers for purging and makeup.
- d) Carbon dioxide is supplied for fire fighting and turbine cooling purge.

5. Loading face shield cooling system removes heat transferred from the upper sodium plenum and heat generated by the attenuation of radiation from the reactor, by circulating 250-psig nitrogen gas in a closed loop through tubing embedded in the lead above the bottom seal plate of the shield.

6. Cell nitrogen cooling system maintains proper operating temperatures in the inert gas atmosphere for the equipment in the cells and for the concrete shielding. The system consists of two separate circuits. One circuit cools the reactor cavity, the primary pipe shaft, and the fuel storage pool, while the other circuit cools the rest of the cells where the nitrogen is not expected to be activated. Each of the two circuits is provided with two sets of cooling units consisting of nitrogen to water heat exchanger, blower, and a motor mounted inside a leaktight pressure chamber having common atmosphere with the respective cells.

7. Radioactive vent system collects, filters, monitors, and disposes all gases that may be radioactive. Gases are released through the plant stack to atmosphere if the activity is below the maximum permissible concentration, or stored in decay tanks for subsequent controlled release to the atmosphere if the activity is greater. Analysis of samples from decay tanks determines the time of disposal and allowable release rates.

8. Radioactive liquid waste system collects, monitors, segregates, stores, and disposes of all liquid wastes, including drains from the radioactive gas system, the sodium component wash cells and the maintenance cell. Samples are taken from the collection tanks and monitored for activity. Liquid wastes with activity below the maximum permissible concentration for direct release are transferred directly to the circulating water discharge. If the waste activity is above the maximum permissible concentration, it will be transferred to a decay tank and discharged after the activity has decayed to the permissible level. Any waste containing high level, long half-life isotope contamination would be placed in the concentrated waste tank and held for ultimate disposal by land burial or sea disposal.

9. Sodium service facilities are provided to fill, drain, and flush the primary and secondary systems with sodium, and for the purification of the sodium. Initial filling and makeup for both systems from tank cars, trucks, or drums takes place through a melting and filtering station. In addition, the system is capable of individually filling and draining each loop of the sodium heat transfer system and the reactor vessel. The purification equipment consists of hot traps for carbon removal and oxide polishing, cold traps for sodium oxide removal, and plugging meters to determine oxide concentration. Sodium sampling connections are also provided in the service system.

10. Component maintenance

- a) Maintenance cell is designed for the disassembly, reassembly, packaging, and shipment of radioactive components, such as fuel elements, control rods, pumps, and the internals of the fuel-handling machine. These operations are performed in two areas, designated the operating and incell areas. Sealed access doors, viewing windows, special equipment, and special tools are designed for use in the maintenance cell.
- b) Wash cell is provided to remove sodium from components requiring maintenance. This operation consists of washing, drying, and purging and they are performed and controlled from a central operating board.

11. Sodium systems preheating is required to raise the temperature of the entire sodium system prior to initial filling with molten sodium and for maintaining the sodium in a fluid state prior to start of nuclear power generation. To accomplish this, the sodium system is preheated and maintained at a temperature approximately 350° F by use of tubular type electric resistance heaters installed on the piping, underneath the insulation. The steam generators are preheated using boiler feedwater and steam.

12. Steam dump and startup system is provided to dispose of the steam produced by the decay heat after reactor shutdown and furnish preheat steam to the steam generators at startup. This system is entirely independent of the normal plant hardware to allow turbine plant maintenance immediately after shutdown.

13. Fire protection

- a) Warning system is provided to initiate an alarm in the case of fire in any area subjected to a fire potential and not under continuous supervision.
- b) Turbine plant fire protection will be provided by a fixed fire water system, low-pressure CO₂ system, portable CO₂ and dry chemical extinguishers, and special design features incorporated in potentially hazardous systems. The fire water system serves sufficient hose connections, yard hydrants, water spray system in the turbine-generator area and the automatic deluge water spray system for the protection of transformers.

- c) Reactor plant equipment and piping containing radioactive sodium are sealed in compartments containing a nitrogen atmosphere. There is no fire water in the reactor building. Bulk dry calcium carbonate applied by hand scoops may be used to extinguish small fires resulting from small sodium spills, while movable chemical extinguishers are provided for larger fires.

14. Heating, ventilating and air conditioning

- a) Reactor building heating and ventilating is furnished by a once-through system, continuously circulating outside air through the building. In order to ensure that leakage of air is inward, toward areas of potential radioactivity, the building atmosphere is maintained at a negative pressure relative to outdoor barometric pressure. All air exhausted from the building is continuously monitored for activity before release from the stack.
- b) Turbine room heating will be supplied by steam type unit heaters, roof exhaust fans will provide ventilation.
- c) Control room is provided with year around air conditioning unit for controlling temperature and humidity as necessary for proper operation of the instrument system and operator's comfort.
- d) Administration building is provided with a standard industrial heating, ventilating, and air conditioning system.

F. PLANT DESIGN DATA

1. Reactor System

Thermal power (Mwt)	2350
Active core diameter (ft)	25.1
Active core height (ft)	14.0
Axial reflector thickness (ft)	2.0
Radial reflector thickness (ft)	2.25
Core structure	
Cavity liner diameter (ft)	42.0
Thermal shield thickness (in.)	4
Reactor vessel liner diameter (ft)	34.2
Reactor vessel diameter (ft)	34
Calandria vessel diameter (ft)	32

a. Fuel Element

Fuel material	UC
UC slug diameter	0.500 ± 0.005
UC slug length (in.)	1 to 8
UC slug stack height (ft)	14
Number of spaces per ft length	1
Thermal bond (Na)(in.)	0.025
Cladding material	304 SS
Cladding tube (OD, in.)	0.570
Cladding tube wall (in.)	0.010
Sodium annulus (in.)	0.025
Gas space length (in.)	40
Number of rods	18
Central support tube (OD, in.)	0.570
Central support tube wall (in.)	0.030
Process tube material	304 SS
Process tube (OD, in.)	4.300
Process tube wall (in.)	0.025
Square lattice pitch — cold at 68° F (in.)	10.25
Number of fuel elements	595
Number of control rods	84
Number of active core positions	680
Fuel per rod (kg U Nominal)	6.88
Fuel per element (kg U)	124
Core loading (Mt/U)	73.8
Design maximum fuel slug diameter increase (8%) (in.)	0.04
Design average fuel slug diameter increase (5.3% (in.)	0.0265
Design fission gas release fraction (%)	3
Design gas plenum pressure (end-of-life operating temperature) (psig)	250
Design maximum fuel clad strain (%)	0.4
Initial enrichment (wt % U ²³⁵)	3.60
Final enrichment (average)	1.45
Pu content of spent fuel (average)	6.78
Average exposure (Mwd/Mtu)	25,000

b. Core Heat Transfer and Coolant Flow

Nominal sodium inlet temperature (° F)	750
Nominal sodium outlet temperature (° F)	1150
Maximum hot-channel fuel temperature (° F)	2000
Location of hot-spot fuel temperature from midpoint (ft)	1.12
Maximum hot-channel inner clad temperature (° F)	1250
Maximum hot-channel outer clad temperature (° F)	1211
Maximum hot-channel Na outlet temperature (° F)	1166
Maximum nominal fuel temperature (° F)	1869
Maximum nominal clad temperature (° F)	1190
Maximum nominal linear rod power (kw/ft)	33.25
Maximum nominal heat flux (Btu/hr-ft ²)	760,530
Average nominal heat flux (Btu/hr-ft ²)	338,350
Maximum nominal element channel power (kw)	5949
Coolant temperature rise hot-channel factor	1.04
Clad temperature hot-channel factor	1.11
Fuel temperature hot-channel factor	1.14
Element disadvantage factor	0.90
Axial peak-to-average power ratio	1.35
Radial peak-to-average power ratio	1.50
Overall peak-to-average power ratio	2.25
Average sodium velocity (ft/sec)	13.7
Maximum sodium velocity (ft/sec)	14.2
Coolant flow area (in. ²)	9.514
Maximum element Na flow rate (lb/sec)	46.76
Total sodium flow rate (lb/hr)	66.0 x 10 ⁶
Core pressure drop (psi)	21
Reactor thermal power output (Mwt)	2350
Heat generated outside fuel (%)	6.0
UC fuel thermal conductivity (Btu/hr-ft-° F)	13.3
Sodium bond thermal conductivity (Btu/hr-ft-° F)	38.4

Average heat transfer film coefficient (Btu/hr-ft ² -°F)	6390
Average specific power (kw/kg U)	31.95
Average fuel power density (kw/in. ³)	6.68
Average core power density (kw/ft ²)	34.5
Average incore residence time (yr)	2.679
Fuel cross-sectional area (in. ²)	3.534
Thermal bond Na area (in. ²)	0.742
Stainless steel area (in. ²)	0.805
Sodium area (in. ²)	9.718
Void area (in. ²)	4.000
Graphite area (in. ²)	86.365

c. Control Data

Total number of control rods	84
Rods with actuators and drives	84
Rods with scram latches	84
Total control rod reactivity worth (minimum required) (%)	16.
Equilibrium xenon and samarium (%)	3.4
Xenon and samarium override (%)	0.7
Fuel depletion (%)	8.0
Rod length and travel	
Poison column length (in.)	156
Rod travel (in.)	156
Uppermost rod position places bottom of poison column at core reflector interface	X
At full-down position bottom of poison is 1 ft above core bottom	X
Rod speed, nominal (in. /min)	15
minimum (in. /min)	7.5
maximum (in. /min)	30
Reactivity rates	
Maximum allowable (%/sec)	0.033
Minimum required (%/sec)	0.0012
Nominal (%/sec)	0.005

Drive mechanism shall be torque-limited	X
Any of the rods may be selected for automatic or manual control	X
Interlocks	
Interlock shall prevent the rods from being driven out of the core until all the scram latches have been engaged.	X
No interlocks (except the lower limit switches) shall prevent the rods from being driven into the core.	X
Rod response time (sec)	0.25
Rod positioning and indication requirements	
Absolute position indication with respect to a frame of reference in the core (in.)	+1.0
Accuracy of differential rod position requirements (in.)	+0.1
Minimum rod motion possible by "inching" or jogging drive motors (in.)	0.05
When motor is deenergized rod shall not move	X
Rod position indication does not have to follow scram but must be reset before restart	X
Scram-latch engagement shall be indicated	X
Full-down position to be indicated coincidentally with reaching that position	X
If torque limiting is provided by a slip clutch, slippage indication shall be provided	X
Limits and stops	
An upper limit switch shall stop the drive motor and provide indication when the rod is at the uppermost position	X
A physical stop shall prevent motion of more than 1/2 in. beyond the uppermost position	X
A physical stop shall prevent motion beyond the lower position	X
All switches used in instrumentation or control shall be adjustable in 3 axes	X
All switch installations shall be dual	X

Flexibility

All electrical maintenance shall be possible without reactor shutdown. The rod concerned shall be capable of being dropped when a scram occurs. X

Capability shall be provided to test each rod circuit and rod drive without the poison column provided the reactor is shut down X

Safety system requirements

Scram delay time (sec) 0.5

Rod-drop characteristics

Average minimum rod-fall acceleration (g) 0.1

Rods shall be capable of being dropped from any position in the core X

Safety duty cycle requirements

Average number of rod-drops (per yr) 50

Maximum number of rod-drops (per yr) 150

Maximum number of complete withdrawals and insertions (per yr) 1000

Shim duty cycle requirements

Number of cycles (per yr) 3300/A

Amplitude of cycles (in.) $0.05 < A < 2.5$

Source

Material (capsule) Sb_2O_4

Material (cylinder) Be

Source strength (n/sec) 1×10^9

Operating temperature ($^{\circ}\text{F}$) 1300

Containment thimble 304 SS

Number required 1

2. Heat Removal Systems

a. Sodium Systems

(1) Primary System

Number of loops each 5

Capacity of each loop (nominal Mw) 470

Sodium temperature ($^{\circ}\text{F}$)

Reactor inlet 750

Reactor outlet 1150

Total sodium flowrate (lb/hr)	66.0×10^6
Sodium flowrate/loop (lb/hr)	13.2×10^6
Pump capacity (gpm)	3.3×10^4
Motor (hp)	2250
Total dynamic head (ft)	190
NSPH available (ft)	28
Pipe sizes (OD in.)	
Suction	30
Discharge	28
Cold leg	26

(2) Secondary System

Number of loops	5
Capacity of each loop (nominal Mwt)	470
Sodium temperature (°F)	
Steam generator inlet	1060
Steam generator outlet	660
Total sodium flowrate (lb/hr)	66.0×10^6
Per loop	
Pump capacity (gpm)	3.3×10^4
Motor (hp)	2000
Total dynamic head (ft)	190
Pipe sizes (in.)	
Hot leg	28
Cold leg	26

(3) Intermediate Heat Exchanger

Type construction	Modular shell and tube unit
Number of units in circuit	2
Main circuit unit	
Duty per unit (Btu/hr)	8.75×10^8
Surface area per unit (sq ft)	9890
Tube data per unit — 2880 tubes, 5/8-in. OD by 0.049-in. wall, 21 ft long	

	<u>Tube Side</u>	<u>Shell Side</u>
Fluid	Primary Na	Secondary Na
Flow (lb/hr) per unit	6.6×10^6	6.6×10^6
Temperature in (°F)	1150	660
Temperature out (°F)	750	1060
Design pressure (psi)	100	100
Design temperature (°F)	1200	1200
Material	304 SS	304 SS

(4) High Temperature Module

Type of construction	Shell and tube
Total number of modules	100
Number of loops	5
Number of modules per loops	20
Duty per module (Btu/hr)	36.1×10^6
Surface area per module (ft ²)	498
Tube data per module – 61 tubes, 5/8-in. -OD by 0.095-in. wall, 50 ft long	

	<u>Tube</u>	<u>Shell</u>
Fluid	Steam	Secondary Na
Flow per module (lb/hr)	72,400	481,000
Temperature in (°F)	725	1060
Temperature out (°F)	1005	833
Outlet steam pressure (psig)	3600	-
Material	321 SS	321 SS

(5) Low Temperature Module

Type of construction	Shell and tube
Total number of modules	140
Number of loops	5
Number of modules per loop	28
Duty per module (Btu/hr)	27.2×10^6
Surface area per module (ft ²)	476
Tube data per module – 61 tubes, 5/8-in. OD by 0.095-in. wall, 45.4 ft long	

	<u>Tube</u>	<u>Shell</u>
Fluid	Water/steam	Secondary Na
Flow per module (lb/hr)	51,700	471,500
Temperature in (°F)	512	833
Temperature out (°F)	725	660
Inlet feedwater pressure (psig)	3800	-
Material	2-1/4 Cr - 1 Mo	Same

(6) Reheater Module

Type of construction	Shell and tube
Total number of modules	30
Number of loops	5
Number of modules per loop	6
Duty per module (Btu/hr)	44.7 x 10 ⁶
Surface area per module (ft ²)	1710
Tube data per module — 110 tubes, 1-in. OD by 0.049-in. wall, 38.7 ft long	

	<u>Tube</u>	<u>Shell</u>
Fluid	Steam	Secondary Na
Flow per module (lb/hr)	169,000	600,000
Temperature in (°F)	603	1060
Temperature out (°F)	1005	833
Outlet steam pressure (psia)	706	-
Material	321 SS	321 SS

b. Steam System

(1) Main Turbine Generator

Turbine type	1,050,603-kw cross-compound 4-flow 43-in. LSB
3600 rpm turbine element	Single-flow high-pressure section and a double-flow reheat section
1800 rpm turbine element	2 identical double-flow low-pressure turbines

Main steam pressure (psig)	3500
Main steam temperature (°F)	1000
Main steam flow (lb/hr)	7,236,650
Reheat steam pressure (psig)	691
Reheat steam temperature (°F)	1000
Reheat steam pressure (°F)	700
Reheat steam flow (lb/hr)	5,092,300
Feedwater temperature (°F)	512
Turbine back-pressure (in.)	1-1/2
3600-rpm generator element	
Output (kw)	550,793
Rating (kva)	640,000
Exciter (kw/volt)	2410/500
Terminal voltage (kv)	24.0
1800-rpm generator element	
Output (kw)	499,810
Rating (kva)	640,000
Exciter (kw/volt)	2410/500
Terminal voltage (kv)	24.0
Short-circuit ratio	0.64
Power factor	0.85

(2) Feedwater System

(a) Main Condensers (2 required)

Data per condenser

Operating pressure (in. Hg abs)	1-1/2
Steam condensed (lb/hr)	2,258,000
Condenser duty (Btu/hr)	2,145,000,000
Entering circul. water temperature (°F)	57
Leaving circul. water temperature (°F)	77
Circulating water flow (gpm)	214,500
Surface required	167,500
Tube OD, gauge, length	7/8-in. tube 18 bwg 38-ft length

Single-pass condenser, divided water box, horizontal unit	
Circulating water tube velocity (ft/sec)	7.5
Cleanliness factor	0.85
Tube material	Admiralty

(b) Condenser Cooling Water Pumps (4 required, 2/condenser)

Data per circulating pump

Water flow (gpm)	107,250
Pump total dynamic head (ft)	30
Pump efficiency (%)	85
Motor rating	
Horsepower (hp)	900
Speed (rpm)	235
Inlet water temperature (°F)	57

Condensate pumps (3 required, vertical centrifugal)

Data per condensate pump

Condensate flow (gpm)	5600
Pump total dynamic head (ft)	350
Pump efficiency (%)	80
Motor rating	
Horsepower (hp)	750
Speed (rpm)	1800
Inlet water temperature (°F)	92

Circulating water piping 4 pipes from each circulating water pump at 78-in. OD each

(c) Main Boiler Feedwater Pumps (2 required)

Data per pump

Feedwater flow (gpm)	8000
Pump total dynamic head (ft)	11,400
Pump efficiency (%)	80
Pump rpm	6000
Steam turbine driver fixed loss, each (kw)	165
Inlet feedwater temperature (°F)	325
Barrel type	6-stage

Mechanical steam turbine drives for main boiler
feedwater pumps (2 required)

Data per turbine

Steam throttle pressure (psig)	730
Steam throttle temperature (°F)	605
Steam throttle flow (lb/hr)	590,000
First extraction steam pressure (psig)	310
First extraction steam temperature (°F)	460
First extraction steam flow (lb/hr)	215,000
Second extraction steam pressure (psig)	157
Second extraction steam temperature (°F)	365
Second extraction steam flow (lb/hr)	104,000
Third extraction steam pressure (psig)	90
Third extraction steam temperature (°F)	326
Third extraction steam flow (lb/hr)	130,000
Exhaust steam pressure (psig)	43
Exhaust steam temperature (°F)	286
Exhaust steam flow (lb/hr)	138,000
Turbine rpm and rating (rpm)	6000
Boiler feed pump input (kw)	16,970
Throttle design pressure (psig)	800
Throttle design temperature (°F)	750
8-stage dual pressure turbine, noncondensing type with variable speed control for flow control

Drawing 1. Reactor Plant Arrangement Plan Below Operating Floor for 1000-Mwe SGR (L160-GA001)

Drawing 2. Reactor Plant Arrangement Plan Above Operating Floor for 1000-Mwe SGR (L160-GA002)

Drawing 3. Reactor Plant Arrangement Sections for 1000-Mwe SGR (L160-GA003)

Drawing 5. Control Rod Assembly
 ment for 1000-Mwe SGR
 (L160-RM001)

TURBINE TYPE: 1000,000 KW @ 3.5" hg ABS, 3% M.U.
 CC4F-43 L5B 3600/1800 RPM
 3500 PSIG/1000" P - 1000" F

GENERATOR OUTPUT: 1,050,603 KW
 STATION AUX POWER: 31,503 KW (~3%)
 NET STATION KW: 1,019,100 KW

NET STATION ON: $7,237,650(424.8-501.0) + 5,078,700(1516.4-278.6) = 7824 \text{ BTU/KWH}$
 HEAT RATE: $\frac{7,237,650(424.8-501.0) + 5,078,700(1516.4-278.6)}{1,019,100} = 7514 \text{ BTU/KWH}$

NET CYCLE: $\frac{7,237,650(424.8-501.0) + 5,078,700(1516.4-278.6)}{1,050,603} = 7514 \text{ BTU/KWH}$

Drawing 7. Heat Balance Program
 for 1000-Mwe SGR
 (L160-SA011)

V. DEVELOPMENT PROGRAM

Studies of sodium as a reactor coolant were started at North American Aviation in 1949. The general investigation of sodium technology and sodium cooled reactor systems continued until 1954, when R & D was started for the Sodium Reactor Experiment (SRE). The development of sodium technology by AI has continued since that time, including the R & D for the Hallam Nuclear Power Facility (HNPF). This AEC-sponsored program, coupled with development by other AEC contractors and the SRE and HNPF experience, establishes the basis for the prototype design and construction.

Development programs required for the 200-Mwe prototype plant and the 1000-Mwe plant are discussed in this section. With few exceptions, the prototype plant development programs are designed to provide the necessary information for the design and construction of the 1000-Mwe plant. Only in the areas of fuel, control rods, steam generators, and reactor physics will additional development be required for the larger plant. In these areas, continued development is needed because of the somewhat more advanced operating conditions for the 1000-Mwe plant. This minimal development effort for the larger plant arises from judicious selection of component configuration and design. Most components in the prototype plant are representative of size requirements for a 1000-Mwe plant and function under operating conditions similar to those of the large SGR. Where size cannot be economically duplicated in the prototype, such as the calandria and reactor structure, prototypical design features have been incorporated and, in these cases, the component design configuration has been carefully selected to permit scaleup on sound engineering principles.

The development programs for the two plants have been conceived to permit design and construction of the prototype plant for 1968 operation and to follow up with the initiation of a design for the 1000-Mwe plant in 1969.

In addition to the specific development programs for the 1000-Mwe and prototype plants, a more general development program for sodium cooled reactors is presented. Projects in this category are not necessarily related to an SGR, but are connected directly to advancing technology of sodium cooled reactors, primarily in the areas of components and safety.

A summary of development costs by government fiscal year for the 200-Mwe prototype, the 1000-Mwe SGR Plant, and the general development program is shown in Table 22. The costs shown are operating and equipment funds. Facility operating costs have been assigned to each development program according to specific component tests utilizing these facilities. Construction costs for operating facilities are not included, but are assumed to be available at a level consistent with current plans.

The development costs presented in this section represent estimates for planning purposes only. At the time each development task is planned in detail, the costs may vary somewhat from those presented in this report, depending on past progress and more accurate cost information.

TABLE 22

LARGE SGR DEVELOPMENT PROGRAM COST SUMMARY

		\$1000							\$1000						
		FY 1965	FY 1966	FY 1967	FY 1968	FY 1969	Total			FY 1965	FY 1966	FY 1967	FY 1968	FY 1969	Total
A.	200-Mwe Prototype Plant														
1.	Component development														
a.	Calandria structure	380	-	-	-	-	380								
b.	Calandria maintenance	460	420	130	-	-	1,010								
c.	Control rod	260	115	-	-	-	375								
d.	Fuel and cladding	2,450	2,500	2,250	-	-	7,200								
e.	Fuel handling	110	-	-	-	-	110								
f.	Reactor hydraulics	90	-	30	-	-	120								
g.	Sodium pumps	-	-	325	-	-	325								
h.	Sodium valves	-	120	50	-	-	170								
i.	Steam generator	170	-	-	-	-	170								
j.	Sodium instrumentation	220	-	50	-	-	270								
k.	Design and safety evaluation	200	100	-	-	-	300								
	Subtotal	4,340	3,255	2,835	-	-	10,430								
2.	Physics program														
a.	Burnup studies	255	95	-	-	-	350								
b.	Doppler coefficient	75	80	-	-	-	155								
c.	Critical experiments	230	150	-	-	-	380								
d.	Reactor physics analyses	100	100	-	-	-	200								
	Subtotal	660	425	-	-	-	1,085								
3.	Facility operation														
a.	Sodium Reactor Experiment (SRE)	2,640	2,540	2,540	-	-	7,720								
b.	Sodium Component Test Installation (SCTI)	900	205	-	-	-	1,105								
c.	Large Component Test Loop (LCTL)	125	125	125	-	-	375								
d.	Sodium Pump Test Facility (SPTF)	-	-	860	-	-	860								
	Subtotal	3,665	2,870	3,525	-	-	10,060								
	Total: 200-Mwe Prototype Plant	8,665	6,550	6,360	-	-	21,575								
B.	1000-Mwe SGR Plant														
1.	Component development														
a.	Fuel and cladding	370	1,720	1,700	1,000	930	5,720								
b.	Control rod	90	180	125	-	-	395								
c.	Steam generator	175	30	-	-	-	205								
d.	Design and safety evaluation	200	300	400	400	-	1,300								
	Subtotal	835	2,230	2,225	1,400	930	7,620								
2.	Physics program														
a.	Burnup studies	-	-	90	90	90	270								
b.	Doppler coefficient	-	-	55	55	-	110								
c.	Critical experiments	145	20	60	55	-	280								
d.	Hot exponential	130	75	125	-	-	330								
e.	Fuel fabrication	-	50	180	-	-	230								
f.	Reactor physics analyses	30	30	40	40	40	180								
	Subtotal	305	175	550	240	130	1,400								
3.	Facility operation														
a.	Sodium Reactor Experiment (SRE)	-	-	-	2,540	2,540	5,080								
b.	Sodium Component Test Installation (SCTI)	120	-	-	245	920	1,285								
	Subtotal	120	-	-	2,785	3,460	6,365								
	Total: 1000-Mwe SGR Plant	1,260	2,405	2,775	4,425	4,520	15,385								
C.	General Development Program														
1.	Reactor safety program														
a.	Fuel element failure mechanism	100	150	150	100	50	550								
b.	Fuel element transient heating	150	150	200	150	50	700								
c.	Transient heat transfer in sodium	100	150	100	50	-	400								
d.	Boiling effects during sodium reactor accidents	150	150	100	50	-	450								
e.	Reactor kinetics	150	150	100	50	50	500								
f.	Fission product retention in sodium	100	100	50	-	-	250								
g.	Contamination removal techniques	175	100	50	-	-	325								
h.	Energy release and absorption following component failure	150	200	250	200	75	875								
i.	Fission product release from sodium fires	100	125	75	-	-	300								
j.	Atmospheric decontamination in reactor accidents	155	125	50	-	-	330								
	Subtotal	1,330	1,400	1,125	600	225	4,680								
2.	Advanced development														
a.	Effects of heat transfer on steam generator corrosion	100	100	-	-	-	200								
b.	Dispersed fuel experimental study	100	160	210	-	-	470								
c.	Development and evaluation of ceramic fuels	100	160	210	-	-	470								
	Subtotal	300	420	420	-	-	1,140								
3.	HNPF program														
a.	Fuel element surveillance and examination	520	350	350	350	350	1,920								
b.	Reactor operations analyses	310	275	250	250	250	1,335								
	Subtotal	830	625	600	600	600	3,255								
4.	Facility operation														
a.	Sodium Component Test Installation (SCTI)	-	655	860	655	-	2,170								
5.	Coordination, reports and special projects	215	175	175	175	175	915								
	Total: General Development	2,675	3,275	3,180	2,030	1,000	12,160								
	Program Total	12,600	12,230	12,315	6,455	5,520	49,120								

A. 200-Mwe PROTOTYPE PLANT DEVELOPMENT PROGRAM

The development program for this plant is divided into three sections:

1) component development, 2) reactor physics program, and 3) facility operation.

In general, each component involves two stages of development:

- 1) concept development – development required to establish the bases for the component design; and
- 2) prototype development – development and test of a component made to drawings and specifications for the plant.

In most cases, the concept development program is funded under the current AEC-sponsored SGR program. Where development is already under way, a description of progress to date is presented, and funds required to continue the concept development to a conclusion are given. Prototype development is assumed to commence in GFY 1965.

1. Component Development

a. Calandria Structure

Development of the calandria concept has been under way since 1962. The calandria structure operates in a sodium environment with a maximum nominal temperature of 1150°F. The structure is Type 304 stainless steel. At 1150°F, stainless steel is subject to creep under long-term loading conditions. Because present analytical predictions of creep deformation and creep buckling of structures cannot be completely relied upon, structural tests are required. Parts of the calandria structure subjected to thermal transient conditions must also be tested.

(1) Concept Development

The objective of the calandria structure development is to demonstrate the long-term reliability of the calandria core structure for the large SGR by: performing simulated reactor environmental tests with large scale sections of the reference calandria structure; and conducting experiments with small-scale calandria hardware to provide additional information on structural behavior.

The following concept-oriented tests associated with calandria reliability are in progress or will be conducted.

(a) Calandria Stayed Head

The calandria core top head structure will be suspended from the loading face shield by hanger rods. In order to arrive at the most satisfactory design for the hanger connection details and for the overall structural behavior of the perforated top head plate, several structural tests will be conducted. One or more hanger joint designs will be tested at room and elevated temperatures to determine the optimum joint design. Two large sections of the calandria top head plate, one section representative of an internal hanger cell and another section duplicating peripheral hanger geometry, will be tested to obtain measured stresses and deflections under simulated operational loads and temperatures. Initial elevated temperature tests will be carried out for 6 months at reactor operating temperatures. These tests will be followed by accelerated creep tests at higher temperatures or with higher loads to simulate 30-yr operation. The fabrication of test equipment has been started, and initial testing will be completed about January 1965. Accelerated long-term creep tests will be completed about June of 1965.

(b) Calandria Stayed Shell

The calandria shell is stayed to the reactor vessel to prevent buckling of the large diameter-thin wall shell due to the external pressure of the sodium pool in which the calandria is immersed. Structural analyses will be performed to determine the number and location of the stays. Several stay joint designs will be tested at room and elevated temperatures to obtain measured local peak stresses and deflections under simulated operating loads and temperatures. The design of test equipment will be started in January 1964 and testing will be completed about June 1965.

(c) Reactor Vessel Hangers

The reactor vessel will be suspended from hanger bolts which are fastened to the bottom of the loading face shield. The structural detail which transfers the load from the vessel wall to the circular sections of the hanger will be tested to demonstrate its structural adequacy. Tests to be performed on vessel hangers include accelerated creep tests to rupture and measurement of the peak stresses, stress distribution and deformations. Design of the test equipment will be initiated in January 1964 and testing will be completed about July 1965.

(d) Process Tube Axial Buckling

A unique axial buckling problem potentially exists within the calandria process tubes because of the presence of bellows at the end of the tubes. Internally pressurized, thin-wall tubes, not solidly restrained axially, will buckle at some pressure. This buckling phenomenon, under conditions where metal creep flow exists, appears as gradually increasing curvature, and is difficult to predict analytically. An apparatus will be designed and fabricated to test full length process tubes for instability due to internal pressure. Testing will be done at elevated temperatures and with internal pressure level as a variable to determine the margin of safety in the reference design. Fabrication of the test equipment has been started. Short-term tests will be completed about August 1964; long-term creep tests will be completed about June 1965.

(e) Process Tube - Graphite Structure

Tests will be performed to determine the effect of radiation-induced graphite log bowing on process tubes. Local loads on the process tube due to bowed graphite logs could cause carburization of permanent deformation of the stainless steel process tubes if the log is not properly designed. Design of the test equipment has been started. Testing will be completed about June 1965.

(f) Process Tube Radial Buckling

Normally the calandria vessel will contain a helium atmosphere at a pressure of 1 psig, which will create no buckling problem. However, during leak detection operations, the calandria internal atmosphere may be pressurized to 15 to 20 psig, which would result in an external pressure on the process tubes. Tests to date have shown that the unsupported tube will withstand only 9- to 10-psi external pressure. Testing will continue to determine the effect of the fuel element and process tube bracing on the collapse pressure. Short-term tests will be complete about June 1964; long-term creep tests will be complete about January 1965.

(g) Thermal Cycle

Several tests will be performed to evaluate the thermal shock resistance of brazed and welded process tube joints and of the calandria grid plate. Testing will be in two phases. The first phase, which is currently under

way, will thermal cycle single brazed and welded joints. Based on the results of these tests, on structural analysis of the joints, and on final selection of brazed or welded joints for reactor application, the second phase of the program will be initiated. The second phase is to test a large section of the top grid plate and several brazed (or welded) joints, in the same geometrical pattern as the reference calandria. The test would duplicate reactor sodium thermal transient conditions in the vicinity of the top brazed (or welded) joints and grid plate. The LCTL facility will be used for the test. The first phase will be complete about February 1964. The second phase will be completed about June 1965.

(h) <u>Concept Development Cost – Calandria Structure</u>	(in \$)	(in \$)
	<u>FY 1965</u>	<u>FY 1966</u>
Operating Funds		
Calandria stayed head	\$ 60,000	
Calandria stayed shell	60,000	
Reactor vessel hangers	20,000	
Process tube axial buckling	40,000	
Process tube graphite structure	20,000	
Process tube radial buckling	20,000	
Thermal cycle	<u>140,000</u>	
	360,000	75,000
Equipment Funds	<u>20,000</u>	
	\$380,000	75,000

(2) Prototype Development

Development and testing of a full-scale calandria structure is neither economically justified nor technically warranted. The concept development phase has been conceived to provide design information from large calandria sections, which represent prototypical features of the calandria structure. These sections are large enough to provide at least one typical cell which is subjected to dynamic loads expected in the full-scale structure. The calandria structure has been designed with the cell principle in mind, and tests will be conducted on the typical cell theory.

b. Calandria Maintenance

Maintenance philosophy with the calandria concept is considerably different from canned moderator cores utilized in the SRE and HNPF reactor concepts. Because the calandria is such a large vessel, removal and replacement of the entire assembly are impractical. Therefore, design features must be incorporated in advance to facilitate repair of the calandria in place. Furthermore, provisions in the design are required to minimize the extent of damage to the moderator graphite in the event a structural failure exposes the graphite to sodium coolant. The calandria structure for the 1000-Mwe SGR and prototype plants have been designed to meet these maintenance criteria.

The calandria shell and grid plates are designed for 30-yr operation with no expectation of failure. Components comprising this section of the structure are designed on an ultra conservative bases, similar to other reactor structure components that are categorized as nonreplaceable after reactor operation, such as the reactor vessel and sodium inlet and outlet nozzles. The SGR concept can afford this conservative approach because of the low pressure, resulting in reasonable metal thicknesses. Economic compromises are not entertained in the design of these components. Design of the remaining structural component of the calandria, the process tube, is an economic compromise between the objective of minimizing neutron poison in the active core region and providing a reasonably safe design for 30-yr operation. Due to this compromise, there is a small but finite probability of process tube failure. The calandria design provides for remote removal and replacement of process tubes and associated graphite logs. In addition, special features for detecting and locating a leak and for minimizing the extent of damage to the graphite have been incorporated.

(1) Concept Development

The objectives of the current development program are to demonstrate remote maintenance of calandria process tubes and graphite and to provide the design bases for the tooling and equipment required for actual reactor maintenance operations. This activity covers the following equipment: process tube removal and installation equipment, graphite log removal and replacement equipment, and calandria leak detection equipment.

Figure 13. Process Tube Brazed Joint Configuration

(a) Process Tube Maintainable Joints

A substantial amount of work has already been accomplished in the development of techniques to render the calandria structure easily maintainable. Primary among these efforts has been the development of induction brazed process tube joints and the equipment required to make and break the joints by remote operation. The initial development work was metallurgical in nature and directed toward the selection of suitable braze alloys. By using furnace brazed specimens and joint mockups, it was established that Coast Metal No. 52, 53, and 60 braze alloys are suitable candidates for the reactor braze materials. Results from mass transfer experiments in a 1200°F/650°F thermal harp loop provided a partial basis for this alloy selection. Additional basis was obtained from creep rupture tests performed at 1200°F. With loads in the 20,000 to 27,000 psi range, it was found that furnace brazed joints were more creep resistant than the base metal.

Concurrently with the alloy selection program, problems which are unique with induction heating such as coil size, configuration, location, and heating and cooling rates were resolved. This work was performed using a modified glove box assembly with both upper (8-in. nominal diam.) and lower (4-in. nominal diam.) process tube joints. The current reference joint designs are shown in Figure 13. At this stage, the general design of the coils and coaxial cabling was established, as shown in Figure 14. In addition, the tooling required to operate in conjunction with the induction heating equipment was devised. From the glove box experiments there evolved a whole series of joint and coil configurations. Sixteen bottom and fourteen top joints have been brazed in these experiments. Some joints were multiple rebrazes in which the same female section (fixed incore components) was brazed to as many as three successive male (process tube component) sections. Other tests investigated the effect of contaminants, such as sodium and graphite (powder), on leak tightness of brazed joints. In order to verify that a newly brazed process tube joint is leak tight, equipment was designed for incore helium leak detection. Development testing with subcomponent parts has demonstrated that the sealing technique (expandable seals) is satisfactory and that the assembly configuration is adaptable to reactor length proportions.

In addition to brazed joint approach to the calandria maintainability problem, an alternate effort utilizing welded joints is well under way.

Figure 14. Brazed Joint Induction Heating Coaxial Cable and Coil

Reweldable joints were designed for both the upper and lower joints of the process tube. The joint designs are shown in Figure 15. The design and manufacture of the equipment for remote installation, removal, and rewelding of the process tube assembly was completed. Preliminary weld tests have been performed with excellent results. Current tests on removal equipment, which employs a milling machine cutter to part the joint, are encouraging. Figure 16 shows the milling machine cutter tool and Figures 17 and 18 show the welding head tool.

When the development stages of both the braze and weld approach to calandria maintainability have reached a point where decided advantages of one over the other are readily apparent, the superior approach will be selected for advanced development on a prototype basis. Selection between the two processes is expected to be made in April 1964.

(b) Calandria Remote Maintenance

The current development effort on maintenance of the calandria core is concerned with all aspects of process tube and graphite replacement. In addition to development of techniques and equipment for brazing and welding process tube joints, development of techniques and equipment for remote removal and replacement of process tubes and graphite in the calandria is necessary. Remote maintenance tools and viewing devices, required to repair the calandria, are being developed.

A full-scale section of the calandria, incorporating four process tubes and associated graphite, has been fabricated for development of the remote maintenance tooling. A photograph of the calandria mockup, prior to assembly and installation of graphite, is shown in Figure 19. The facility will contain provisions for filling and draining sodium in the process tubes to duplicate sodium film deposition. In addition, simulated process tube failures will be possible with this facility to demonstrate leak detection equipment and handling tools for sodium saturated graphite.

Welding or brazing equipment, developed for glove box experiments, will be scaled up in length to suit the four-channel calandria mockup. Removal of process tubes with this test setup will involve a reach of 30 ft under remote operating conditions.

Figure 15. Process Tube Welded Joint Configuration

Figure 16. Welded Joint Milling Machine Cutter

Figure 17. Joint Welding Head

Figure 18. Welding Head Located in Sectional Mockup Joint

Figure 19. Calandria Mockup Sections Before Assembly

Schemes have been devised to remove and replace the graphite filler logs surrounding the process channel assembly. Simple bench tests have been performed on key aspects of the tooling to establish a measure of feasibility before proceeding with detail design.

During removal of the process tube assembly from the calandria, it is possible for filler logs to tilt into the open channel and obstruct the insertion of log removal equipment. This obstruction can be in the form of intact logs or irregular parts of a fractured log. Tooling such as that used so effectively in the SRE fuel slug recovery program can be utilized to grasp smaller pieces or to knock them into the debris bucket installed in the lower grid plate. Typical tools for remote grappling of graphite are shown in Figure 20. Large unwieldy sections may have to be pulverized by power-driven nibblers or choppers to reduce them to a size convenient for the gripping tools.

A remote viewing device, utilizing a miniature TV camera, was designed and fabricated to permit visual observation of the process tube internals and of the graphite after removal of the process tube. This unit, in addition to viewing equipment available from commercial borescope manufacturers, will serve as a core inspection device.

The components of the four-channel mockup have been fabricated. The assembly and maintenance equipment are being fabricated. Assembly of the mockup will start about February 1964 and testing and development will be completed about May 1965.

(c) Sodium-Graphite Interaction

Several scoping tests are required to determine the extent of damage to core graphite from a sodium leak through a failed process tube. Results from these tests will provide the design bases for graphite removal tooling. Laboratory scale tests will be started to determine the diffusion rate of sodium through graphite logs and to ascertain whether sodium will diffuse into adjacent logs. If diffusion into adjacent logs occurs, tests of graphite coatings, which will minimize diffusion, will be conducted. The feasibility of removing the sodium from graphite by reactor heat will be investigated. The test equipment is being fabricated, and testing will be completed about January 1965.

Figure 20. Typical Grappling Tools

(d) Calandria Leak Detection

Methods and equipment for determining that a leak exists in the calandria and for locating the leak are being developed. A system to detect sodium vapor in the calandria atmosphere and the requirements for bringing the calandria gas (helium) to the detecting instrument are being investigated. This system utilizes a spectrophotometric instrument which responds to the absorption of light (sodium D lines) by sodium vapor. Initial results indicate that the leak detector will respond to less than 0.4 ppm of Na vapor in helium at 1150°F and atmospheric pressures. Development of this system will be completed about June 1964.

Techniques for determining which process tube has failed are under investigation. One system requires pressurizing the calandria with helium gas and observing bubbles breaking from the surface of the reactor sodium pool. The position of the bubbles will indicate which process tube contains the leak. A series of experiments was performed to determine depth pressures and hole sizes required to force helium gas into a 600°F sodium pool. The smallest orifice used to date was a drawn Vycor glass tube of 0.00063-in. ID. With the orifice submerged 3 ft in a sodium pool, bubbles could not be observed with a gas pressure of less than 3.8 psig. At 3.8 psig, a bubble broke the surface every 35 sec, and at 9.8 psig a bubble appeared every 14 sec. These and other results indicate that the bubble leak location scheme may be feasible. A leak location scheme which utilizes the detection of A^{41} escaping from the calandria is being considered to supplement the bubble scheme. In this scheme, the calandria atmosphere is filled with a total or partial pressure of argon, which becomes activated (A^{41}) by neutrons. The reactor power and sodium flowrate are lowered to 20% of rated power to reduce sodium pressure, and the calandria gas (argon) pressure increased until the gas escapes into the process tube sodium stream. Gas entering the sodium stream will rise and be trapped in gas space enclosed by that particular channel orifice tube. Monitoring of each orifice tube gas space is accomplished through a sample tube leading to the loading face shield floor level. This system is being investigated for testing in a water loop, modified for a fuel channel and orifice assembly. The exact test system has not been designed to date, but testing of this or similar schemes is expected to be completed by May 1965.

(e) <u>Concept Development Cost —</u> <u>Calandria Maintenance</u>		(in \$) <u>FY 1965</u>
Operating Funds		
Process tube maintainable joints		—
Calandria remote maintenance		130,000
Sodium-graphite interaction		30,000
Calandria leak detection		<u>70,000</u>
		230,000
Equipment Funds		<u>20,000</u>
		250,000

(2) Prototype Development

The design bases for tools required for initial installation and for remote replacement of calandria process tubes and graphite will be established under the concept development program currently under way. Full scale prototype tools and associated gas locks, seals and special casks and grapples required to handle radioactive core components require design, fabrication and testing. These tools and equipment will be developed in two sequential steps in order to meet the prototype plant construction schedule. First, tooling required for initial process tube and graphite installation will be fabricated and tested, and secondly, tooling and equipment required for remote replacement of radioactive core components will be tested.

The current development program has been scheduled to provide a smooth transition into these two steps. Design bases for initial installation of process tubes and initial installation of graphite will be well established early in FY 1965. Design of prototype tools for these operations can begin at that time. Tools and equipment for removing broken graphite logs or logs saturated with sodium will not be developed until later in the fiscal year. Final design of prototype tools for these operations can be deferred until completion of the current development program. This equipment is not needed until the start of plant operation. In addition, gas locks, seals and special casks will not be tested under the concept development program, although reference design layouts will be prepared to assure tooling designs are compatible with other equipment requirements. Prototype equipment for these operations require fabrication and testing. The tasks required to test the core handling equipment are outlined as follows.

(a) Design and Procurement

Reference tool design layouts and associated casks, special grapples and gas locks will be prepared under the concept development program. These layouts will be reviewed and any final design changes incorporated. Fabrication drawings and specifications for reactor tools and casks will be prepared, and an installation for testing the equipment will be designed. Design of the following items is necessary.

- 1) Process tube brazing (or welding) and replacement tools with associated power supply, control console, gas locks and handling equipment
- 2) Optical viewing devices, gas locks and handling equipment for inspecting the process channel before and after removal from the reactor
- 3) Handling tools and equipment for removing graphite from the reactor and transferring it to a storage area
- 4) Handling tools for installing new graphite into the reactor
- 5) Handling equipment test installation

All components required for the remote maintenance of the reactor will be procured, and the test installation will be constructed.

(b) Testing

The test installation will consist of a pit approximately 50 ft deep which contains a reactor mockup, sodium service and an inert gas supply. A 50-ft-high superstructure will be built over the pit and will incorporate crane capability for handling casks and full length maintenance tools.

Components will be assembled at the handling equipment test installation and checked out for operability. Mockup operations for initially installing calandria process tubes and graphite will be conducted. Remote maintenance operations will then be performed as if the core mockup was indeed radioactive. Remote removal and replacement of process tubes and graphite will be demonstrated, using all the necessary tools, gas locks and handling casks.

(c) Prototype Development

Cost — Calandria Maintenance

(in \$)

	<u>FY 1965</u>	<u>FY 1966</u>	<u>FY 1967</u>
Operating Funds			
Test Installation			
Design	65,000	—	—
Procure and construct	80,000	55,000	—
Initial installation tools			
Design	55,000	—	—
Procure and fabricate	10,000	45,000	—
Test	—	70,000	—
Remote replacement tools			
Design	—	50,000	—
Procure and fabricate	—	115,000	—
Test	—	60,000	130,000
	<u>210,000</u>	<u>395,000</u>	<u>130,000</u>
Equipment funds	—	25,000	—
	<u>210,000</u>	<u>420,000</u>	<u>130,000</u>

c. SGR Control Rod

The control rod to be used in the prototype is the same basic design as the HNPF control elements. Because of the different operating conditions of the prototype some design changes are required. There are three main uncertainties resulting from these changes that must be investigated before the rod can be recommended for the prototype reactor. These are: 1) scram rate characteristics; 2) poison material properties under irradiation; and 3) absorber element material compatibility under temperature and wear environments of the reactor.

The uncertainty in scram rate characteristics is due to a reduced helium gap between the absorber column and the thimble. The Hallam control rod design utilized a 0.052 in. clearance, whereas the higher specific power in the prototype core requires a clearance of 0.035 in. to reduce absorber structural material temperatures to a maximum design value of 1900°F. Decreased clearances could cause a snubbing affect and change the scram rate.

Dysprosium oxide was selected as the reference poison material for the prototype control rod. Incore life of the gadolinium-samarium oxide material, used in the Hallam rod, would be in the order of 1 yr in the prototype reactor for reasonable poison thicknesses. For this reason, other poison materials with potential incore lives of 10 yr were investigated, resulting in the selection of Dy_2O_3 . Because very little information is available on radiation effects, thermal conductivity, and physical and chemical stability of Dy_2O_3 , irradiation tests are required. The absorber assembly is comprised of an outer Hastelloy X tube, which contains an annulus of Dy_2O_3 around a central graphite cylinder. Compatibility of these materials at elevated temperatures must be determined. Also, elevated temperature wear testing of absorber cladding materials against the thimble material will be required.

The control rod development program will determine, by testing, the unknown characteristics of the control rod absorber assembly. The modified Hallam control rod actuator assembly is considered adequate for the concept development.

(1) Concept Development

The program for control rod testing follows.

(a) Scram Rate Tests

A series of tests have been performed with a dummy absorber assembly and the existing prototype HNPF Mark B control rod to investigate the effect of a smaller clearance annulus (0.019 in. vs 0.052 in.) and the effect of thimble deflection on scram characteristics. With the smaller clearance the average acceleration was reduced from 0.97 to 0.68 G. The minimum permissible acceleration is 0.1 G. Due to other considerations, such as fabrication tolerances, material wear properties, and absorber and thimble alignment, a 0.035-in. gap was selected for the reference prototype control rod.

An absorber assembly and thimble of the reference SGR design configuration will be fabricated and tested with the HNPF Mark B actuator. Cold and hot cycling and scram tests will be made. These tests will run about nine months and will be completed about September 1964.

(b) Poison Material Irradiation

Capsule irradiation of a neutron absorber material will be performed in a test reactor. This test will be designed to determine the effects of radiation exposure at temperature on thermal conductivity and physical and chemical stability of the absorber materials. Design of the test has been started. The hot cell examination of the capsule will be completed about December 1965.

(c) Poison Column Reactivity Worth

Preliminary criticality tests in the SGR critical facility have been performed for one Dy_2O_3 thickness. At least one additional measurement will be made for another thickness in order to determine worth vs thickness. This will be completed about March 1964.

(d) Compatibility Tests

Elevated temperature compatibility tests between graphite, dysprosium oxide, and Hastelloy X are in process. Capsules containing (a) dysprosium oxide and graphite and (b) dysprosium oxide and Hastelloy X are being tested in a controlled atmosphere. Test temperature requirements are 2300°F for dysprosium oxide and graphite and 2000°F for dysprosium oxide and Hastelloy X.

A 20-in.-long section of absorber column with the center graphite hollowed to permit insertion of a heat source capable of achieving a 2800°F centerline temperature will be thermal cycled. The anticipated temperature profile is 2300°F at graphite-dysprosium oxide interface and 1900°F at the dysprosium oxide-Hastelloy X interface. The specimen will be thermal cycled between these temperatures and 650°F isothermal to simulate service conditions. After 1000 cycles, the specimen will be sectioned and subjected to metallographic examination. Testing will be completed about June 1964.

(e) Material Wear Properties

Elevated temperature testing has been started to establish the wear properties of absorber cladding (Hastelloy X) against thimble material (304 stainless steel). Small sections of the thimble and Hastelloy X material are rubbed together at operating temperatures (Hastelloy cladding at 1800°F and thimble material at 1100°F). The sections will be cycled 100,000 times and then examined for wear and galling. Testing will be completed about March 1964.

(f) Concept Development Cost – Control Rod

	(in \$) <u>FY 1965</u>	(in \$) <u>FY 1966</u>
Operating Funds		
Scram rate tests	25,000	
Poison material irradiation	40,000	50,000 Hot
Poison column reactivity worth	-	cell exam
Compatibility tests	-	(2) of SRE
Material wear properties	-	capsule
	<u>65,000</u>	<u>50,000</u>
Equipment funds	10,000	
	<u>75,000</u>	<u>50,000</u>

(2) Prototype Development

With the initiation of the prototype plant design, there will be a need to start design, fabrication and testing of a control rod of the final basic design established for use in the SGR. In order to establish basic design data expeditiously, a modified Hallam prototype control rod is presently being tested. While adequate preliminary information for determining design requirements will thus be obtained, a true prototype will not be tested, nor would other changes shown as desirable during the developmental tests. Changes to the shield plug and shroud, the actuator mechanism and drive assemblies are not now being made. Subsequent tests of an actual prototype will verify assumptions necessarily used in establishing the basic design and will provide information leading to economical production of units for reactor installation.

The program required to accomplish this objective will include:

(a) Detail and assembly drawings and final analyses (nuclear, thermal and stress) of the control rod design will be made. Design followup during prototype fabrication will be provided.

(b) A prototype of the production control rod design will be fabricated.

(c) The test facilities will be redesigned and modified. Obsolete or worn-out equipment and parts will be replaced. A new furnace will be fabricated and installed.

(d) A control console for automatic and manual cycling of the control rod in both shim and scram modes will be designed and fabricated. Instrumentation for test of the operating characteristics of the control rod will be provided.

(e) The prototype control rod will be assembled and tested at room temperature. The drive, actuator and neutron absorber assemblies will first be individually tested, then testing of the complete assembly will be performed. Testing of the prototype under simulated reactor operation conditions will be performed. Results will establish operating characteristics and demonstrate conformance to design and performance specifications.

(f) Prototype Development Cost – Control Rod

	(in \$)	
	<u>FY 1965</u>	<u>FY 1966</u>
Operating Funds		
Prototype design	45,000	-
Prototype fabrication	80,000	-
Test facility modification	40,000	-
Control console	20,000	-
Tests	<u>-</u>	<u>115,000</u>
	185,000	115,000
Equipment Funds	<u>-</u>	<u>-</u>
	185,000	115,000

d. Fuel and Cladding

(1) Concept Development

Uranium carbide containing 4.5 to 4.8 wt% C and operating at a peak temperature of 1750°F to an average burnup of 25,000 Mwd/MT are the general reference fuel specifications for the prototype plant. These specifications are based on experimental data produced from large scale processing development, experimental material property determination, fuel-bond-clad compatibility tests, and irradiation results. These operating conditions are all modifications from the basic Hallam conditions which were less severe. The fuel development program has reached the stage of development that the initial

results in all technical and cost phases are promising. Fuel element fabrication cost of \$100/kg U has been demonstrated. The current fuel and clad materials development program is concentrating on (a) refining the initial experimental data and (b) producing reliable statistical results. The fuel rod and fuel element testing phase has been started. Tests with rods and elements under the prototype reactor operating conditions are required for the preparation of fuel warranties. Conservative design parameters have generally been used.

The capsule irradiation results are sufficiently promising to permit extrapolation to full burnup conditions. An irradiation program now in progress will demonstrate the specific materials data and show initial results on large scale fuel rods to the full burnup. The irradiation results of the ten Hallam elements currently operating and the Hallam Core II UC elements, will provide supporting results. The emphasis in the development program is to define clearly the fuel and clad material characteristics, substantiate fuel fabrication costs, and to demonstrate large scale operating performance of fuel rods and fuel elements. The latter requirements will be strongly supported by the SRE-PEP fuel program. The fuel development program for the 200-Mwe prototype consists of three main areas (a) UC processing (b) physical properties and compatibility of UC fuel and 304 SS clad, and (c) irradiation results on fuel and clad materials, and fuel elements.

(a) UC Process Development

The fuel slug carbon composition specification is a compromise between the known irradiation performance characteristics, the tolerances of the chemical analysis techniques, and the ability of the processors and fabricators to produce materials with reasonable yield and cost.

Irradiation results indicate that the fission gas release increases with decreasing carbon content. On this basis it is anticipated that the carbon content range will be narrowed more closely approaching stoichiometry for fuel elements beyond Hallam Core II performance characteristics.

The present fuel slug specification of 4.5 to 4.8 wt% C is a 25% decrease in the allowable carbon content from the present Hallam specification. The minimum carbon content of 4.5 wt% C is planned to minimize the effects of the continuous free uranium grain boundary phase which occurs in compositions ≤ 4.4 wt% C.

In the casting of UC fuel slugs, the present charge to the arc melting furnace consists of about 40% new feed stock, 40% remeltable scrap and 20% U metal. The arc casting process has been successful in producing high quality as-cast slugs within dimensional specifications. Further increases in process yields are anticipated as a result of improved mold design, arc furnace operating environment changes, improved feed stock quality, and improved carbon analysis methods. Additional improvements in process engineering have been demonstrated on a laboratory scale which can further reduce as-cast slug costs. The processing of the first large quantity of approximately 25,000 kg for Hallam Core II will provide reliable costs throughout all phases of the process. It will also provide statistical data on the process capability to produce slugs with the tighter carbon specification.

Initial fuel fabrication costs of about \$100/kg U for a completed element on the basis of Hallam specifications have been demonstrated. The exact cost is a direct function of enrichment, throughput, facility, and contractual conditions. These costs require further large scale verification and adjustments with specification improvements.

Processing techniques are in the preliminary stages of development that can produce material to tighter carbon limits. A continuing process development effort is needed to meet the requirements of higher temperature and higher specific power in the prototype fuel material.

Tasks for UC process development are as follows: (1) process engineering analysis and evaluation of HNPF Core II processing methods and alternate processes for process capability, (2) alternate slug casting processes for controlling carbon content — such as semicontinuous small charge casting or hollow cathode discharge casting (no carbon electrode), (3) qualification of vendor feed stock and cast slugs, (4) UC product specifications, (5) carbon analysis technique improvements, and (6) demonstration test with 2000 kg.

(b) Physical Properties and Compatibility

The main physical properties of interest for basic fuel rod design are thermal conductivity and thermal expansion. Initial design values of these properties have been determined for both hyper and hypo stoichiometric material. A problem requiring further investigation is the change in thermal expansion on thermal cycling at high temperature for hypo UC.

Measurements of thermal conductivity have indicated relatively minor changes with carbon composition. The conductivity appears to be a straight line function with essentially zero slope from 0 to 1000°C. The thermal instrumentation from the irradiation capsules indicate that the inpile thermal conductivity value of 0.055 ± 0.002 cal/°C-cm sec (13.3 Btu/hr-ft-°F) is not significantly changed from out-of-pile measurements. Since the thermal conductivity is an important value when the peak temperature is limited, it is of primary importance to have an accurate value with minimum tolerance. Additional supporting information is required on the change in thermal conductivity at higher burnup.

The effects of mechanisms of carburization of the 304 SS clad by hyperstoichiometric UC have been established. All UC development and testing is currently based on hypo UC except for a few control specimens. Extensive tests have shown that hypo UC is compatible with 304 SS at the fuel-clad temperatures of interest to all SGR programs, e.g. <1300°F. Under extreme temperature gradient conditions it has been shown that some free uranium in hypo UC will migrate to the surface. This condition creates a potential U-Fe eutectic reaction at 1340°F. Since all coolant temperatures will be in the range of 1150°F and all peak fuel surface temperature and the peak clad temperature do not occur at the same longitudinal position on a fuel rod, no problems are anticipated. Verification tests are currently in progress to determine the limiting conditions for uranium migration under a thermal gradient in hypo UC.

The out-of-pile 304 SS clad material program is investigating the effect of cold work, stress to rupture properties, heat to heat composition and physical property variations in 304 SS. This effort directly supports the irradiation clad work which has been started. There are two primary related clad problems: (1) to determine the effect of radiation on 304 SS under prototype operating conditions; and (2) to determine the optimum design criteria for these conditions. A general review of all AEC clad programs has shown that the available data from water and gas cooled reactor programs are not directly applicable to the sodium reactor program for two reasons. The test media environment and the thermal to fast flux ratio greatly effect the results. Each of these conditions is distinctly different in the sodium system. This phase of the clad development program directly supports the irradiation test program.

Tasks for physical properties and compatibility are as follows: (1) effect of burnup on thermal conductivity, (2) effect of thermal cycling on coefficient of thermal expansion, (3) determine conditions and mechanisms for U migration in hypo UC, and (4) determine stress rupture properties of 304 SS clad as affected by composition, cold work, and sodium environment.

(c) Irradiation of Fuel and 304 SS Clad

The primary objective of the short term fuel program is to develop and test economical UC fuel elements for the large SGR. The short term emphasis is being placed on (1) hypo UC fuel, (2) large scale fuel rod and fuel element performance, and (3) 304 SS clad. A summary of the irradiation program shown in Table 23 illustrates the magnitude of the program. All three areas of fuel element technology have been put into full scale hardware test e.g. instrumented capsules, fuel rods, and fuel elements. Pre and post irradiation surveillance programs on the fuel and clad will provide statistically significant data. The detailed schedules for all irradiations for the next two years are shown in Figures 21 and 22. The long burnup capsules, rods and elements extend into 1967.

TABLE 23
UC IRRADIATION PROGRAM SUMMARY

*Capsules planned	2
*Capsules operating	8
Capsules examined	17 (13 hyper UC and 4 hypo UC)
Capsules irradiated, not examined	7 (4 hyper and 3 hypo)
*Fuel rods planned	3 at NRX
*Fuel elements operating	12 (2 at SRE and 10 at Hallam)

*Hypo UC, < 4.8 wt % C

The emphasis on hypo UC (<4.8%) is based entirely on compatibility between the fuel and steel clad. One of the primary hypo UC problems is to determine with reasonable accuracy the UC radiation performance versus carbon composition versus temperature. The two instrumented capsule series for HNPF and LSGR shown in Figure 21 are expected to yield this data. The

Figure 21. Large SGR UC Capsule Irradiation Schedule

Figure 22. Large SGR Irradiation Schedule of UC Long Fuel Rods and Elements

use of the more stable environment of the NRX reactor is expected to produce more reliable data. Additional capsules will be required as the initial results are analyzed and evaluated.

It is extremely difficult to extrapolate accurately radiation effects from small capsules to full scale performance predictions. The principal problems are the wide fluctuations in temperature, neutron flux, thermal cycles, and burnup rates that mask the true material effects. Capsule results, unless carefully instrumented and environment controlled, will yield data only satisfactory for program scoping and planning.

The irradiation results to date from capsule tests indicate hyper UC to have low fission gas release <3%, and low dimensional changes <8%, to peak burnups of 25,000 Mwd/MTU. The hypo UC shows approximately the same results but the burnups to date are lower, ~ 8000 Mwd/MTU. A couple of hypo UC capsules at 12 to 16,000 Mwd/MTU sustained cladding ruptures apparently due to excessive temperatures, and thus high internal pressures. Even at these high temperatures, estimated to be >2800°F, the dimensional changes were within fuel rod design limitations. The fission gas release in these cases was high. Since the temperature greatly exceeded the melting point of the free uranium in the hypo UC, these effects were not unexpected. Typical irradiation performance data are plotted in Figure 23. These ruptured capsules provided the basis for setting the lower carbon composition limit and peak operating temperatures. For the Hallam requirements, a hypo range of 4.4 to 4.8 wt% C and 1600°F peak, hot channel factor included, will be utilized. Depending on the results of tests now in progress, it appears necessary to tighten the carbon range for the prototype SGR to 4.5 wt% C minimum. The SRE-PEP program which has the higher design central temperatures of 2000°F peak will definitely use a tighter carbon range. Resolving this problem is one of the major objectives of the current fuel program.

The long fuel rod and fuel element tests have only recently been initiated. This data is the main basis to support fuel warranties. Currently two hypo UC elements are operating in SRE, one at Hallam conditions and one at prototype SGR conditions. A peak burnup of 500 Mwd/MTU will be achieved by January 1, 1964. There are ten UC elements inserted in the Hallam core. The peak burnup to date is <1000 Mwd/MTU. In addition, three long rod irradiations under prototype SGR conditions are planned for test to burnups of

LEGEND
CENTRAL TEMPERATURE (°F)

CAPSULE	MAXIMUM	AVERAGE
◇ Al-3-1		
A	2220	1725
B	2450	1815
C	2260	1790
D	2350	1800
○ Al-3-5		
A	2780	2110
B	>2700	>2100
C	>2500	>1950
D	>2350	>1850
E	>2300	>1750
F	>2500	>1650
G	2780	2110
H	>2700	>2100
I	>2300	>1800
J	>2500	>1650
▽ BMI-23-5		
A-B	1440	1326
△ BMI-23-6		
A	1600	1300
□ W 1-1		
A-B	>1400	1400

◇ ▽ △ □ ○ MEASURED DIAMETER INCREASE
■ ▼ ● ONE-THIRD MEASURED DENSITY DECREASE

Figure 23. Measured Diameter Change vs Burnup and Fuel Central Temperatures for Hypostoichiometric Uranium Carbide

15, 30, and 40,000 Mwd/MTU respectively, in NRX, beginning February 1964. The schedule for all of these large scale irradiations and the future Hallam and SRE-PEP tests are shown in Figure 22. These long rod and fuel element tests will also provide excellent statistical performance data on the individual fuel slug and clad components. The current irradiations on 304 SS clad are limited in scope but of great importance to the program. It has been observed throughout the nuclear industry that structural and clad materials change properties as a result of reactor core exposure. The changes in ductility, creep, and strength properties are significantly great in some cases to be a major problem to establishing reliable clad design criteria. It is imperative that these criteria be established early in reactor design and to use design data specifically related to the reactor conditions. A survey of the radiation effects to 304 SS work at other sites showed that very limited work was being conducted that could be directly used by the sodium program. The significant parameters requiring investigation include coolant environment, fabrication history and chemical composition, thermal to fast flux ratio, annealing and the solid state mechanisms causing the change in properties. This effort may require expansion pending the results of tests now in progress. The irradiation tests consist of (1) short term tests in SRE, (2) longer exposures in another reactor and (3) examination of clad components from capsules, rods, and elements from the fuel irradiation program.

Other radiation effects and related problems currently receiving attention are: (1) post irradiation annealing of UC, (2) improved reliability of inpile thermocouple instrumentation, (3) inpile fuel rod pressure monitoring apparatus, (4) effect of O₂ and N₂ impurities on physical properties and radiation performance, (5) evaluation of UC radiation data from all sources, (6) effect of UC cracking on clad, and (7) UC phase equilibria.

This fuels and materials program represents a concentrated effort to solve the highest priority short term problems in all major phases of uranium carbide fuel technology. Successful completion of this program should allow realistic fuel warranties for the next large SGR or prototype. UC technology is in its infancy and further programmatic efforts are required to exploit the ultimate potential of this concept.

Tasks for irradiation of fuel and 304 SS clad are as follows:

(1) accurately determine UC material relationships between carbon composition and effects of radiation (e.g., fission gas release, burnup, specific power,

surface and central temperature, density change), (2) demonstrate large scale irradiation performance characteristics under prototype conditions, (3) determine effects of irradiation on clad design criteria; including defect tests, (4) develop and test alternate clad material or design, and (5) continue materials engineering in support of fuel and clad, e.g., UC phase equilibria, effect of impurities, inpile properties, support technology.

(e) Concept Development Cost — Fuel and Cladding

	(in \$1000)		
	<u>FY 1965</u>	<u>FY 1966</u>	<u>FY 1967</u>
Operating Funds			
UC process development	200	200	50
Fuel and clad materials properties and compatibility	150	150	100
UC capsule irradiation and examination	700	550	400
Long rod irradiation	250	250	400
Special SRE element fabrication and examination	250	400	450
Special HNPF element fabrication and examination	200	200	200
Clad irradiation and examination	<u>450</u>	<u>550</u>	<u>500</u>
	2,200	2,300	2,100
Equipment Funds	<u>250</u>	<u>200</u>	<u>150</u>
	2,450	2,500	2,250

(2) Prototype Development

Unfortunately, no reactor facility exists which can accept a full-scale prototype fuel element and operate under the prototype reactor conditions. However, PEP modifications to the SRE will permit testing of shorter fuel elements at operating conditions somewhat more severe than contemplated for the prototype reactor. And, fuel elements in HNPF will provide data with full length elements operating below prototype performance conditions. Results from these two facilities, coupled with results from long-rod and capsule irradiation tests in other reactor facilities, will provide the information and confidence to proceed directly with the prototype fuel element design.

e. Fuel Handling

(1) Concept Development

The fuel-handling machine for the prototype plant is basically a Hallam machine modified to accommodate the higher decay heat generated in the fuel. Design features that require testing are: (a) fuel element hold-down latch and grapple, and (b) machine cooling system.

The fuel element in the prototype reactor is latched and supported at the bottom of the calandria core. This feature differs from the Hallam fuel element, which is supported from the top shield plug, and is necessary because of the method of cooling fuel elements during fuel handling. The fuel element latch and grapple must be tested for reliable operation under fuel-handling conditions.

The prototype fuel element maximum decay heat after 10 hr shut-down is 25 kw. The power generation is sufficient to melt the fuel cladding if cooling is not provided as soon as the element is removed from the reactor sodium. A helium cooling system passes helium gas over the element at all times during fuel handling operation. The cooling system must be tested to assure fuel cladding temperatures do not exceed 1200°F during fuel handling. In addition, effects of Na and Na vapor carry-over in the heat transfer system must be determined.

The fuel-handling program will develop and test the important features of the fuel handling scheme, through the design, fabrication and testing of a simplified composite fuel-handling machine. The current development program involves the following tests.

(a) Fuel Element Latch and Grapple

A fuel element hold-down latch and a fuel element grapple have been fabricated and tested. These two components were tested for 1200 typical refueling cycles in 750°F sodium. Test refueling cycles included the operation of inserting a fuel element with the grapple into a simulated process tube and locking the element to the bottom of process channel, releasing the grapple and checking for positive locking of the element latch, withdrawing the grapple into the cask and leaving the fuel element locked in place, lowering the grapple to engage the fuel element and release the hold-down latch, and raising the

grappled fuel element into the fuel-handling cask mockup. This test was completed in FY 1963 and successfully demonstrated the reliability of the latch and grapple.

(b) Composite Fuel-Handling System Installation

A single-barrel fuel-handling machine, cooling system and reactor process tube mockup have been designed, and fabrication of equipment completed. The components will be installed into a composite assembly for testing. Installation will be completed about April 1964.

(c) Composite Fuel-Handling System Test

The fuel element cooling system, without sodium in the simulated reactor vessel, will be tested. Using an instrumented heat fuel element mockup as a power source, the heat transfer characteristics of the overall system will be determined. Gas pressure and temperature transients, which may occur during withdrawal of the heater from the process channel into the cask, will be observed.

Sodium will be introduced to the simulated reactor vessel and maintained at the reference refueling temperature of 650°F. Handling tests will be repeated to determine the degree of sodium carry-over to the cooling system. Repeated immersions of the heater in the sodium pool will be performed to accelerate carry-over affects. If problems occur, such as fouling of heat transfer surface and gas circulator, vapor and mist elimination techniques will be devised and tested.

A reference fuel element will be substituted for the heater element, and fuel element latch and grapple operations will be performed to check out the grapple tape and drive assembly and the fuel element hold-down latch. This test will combine all fuel handling operations; however, fuel element heat generation cannot be duplicated in this test. Completion of the composite fuel-handling assembly test is expected about January 1965.

(d) Concept Development Costs — Fuel Handling

	<u>(in \$)</u>
	<u>FY 1965</u>
Operating funds	
Fuel element latch and grapple	-
Composite fuel-handling system installation	-
Composite fuel-handling system test	<u>110,000</u>
	<u>110,000</u>
Equipment funds	<u>-</u>
	<u>110,000</u>

(2) Prototype Development

The fuel handling composite assembly, constructed for the concept development phase, incorporate full-scale sections of important features of the prototype machine. This assembly will suffice for all fuel-handling development required for the prototype and 1000-Mwe plant.

f. Reactor Hydraulics

The prototype reactor will operate at higher temperatures and at higher power density than previous SGR plants. This imposes more severe hydraulic requirements on fuel elements and sodium plenum designs. The main uncertainties in the prototype reactor hydraulics involve: (1) fuel element channel and variable orifice, (2) control element channel, and (3) reactor upper sodium plenum.

The maximum fuel element channel velocity is 13 ft/sec with a core pressure drop of approximately 21 psi. The vibration characteristics of the fuel element and thin-wall process tube require experimental investigation to assure the frequency and amplitude of vibrations will not cause failure in either component. Experimental verification of theoretical predictions of fuel channel pressure drop and variable orifice tube flow control characteristics are also required. Present analytical techniques for predicting these parameters are not adequate.

Cooling requirements to the control channel during steady-state operation and following a reactor scram vary for each rod position in the core.

Orifice designs to control flow changes, required to minimize sodium outlet temperature changes during various operations, must be developed and tested.

Sodium flow from each of the process channels discharges into an upper plenum from which sodium is pumped around the heat transfer loops. A circumferential baffle with openings is installed around the outer region of the plenum to collect sodium from the plenum in a radial direction and to avoid the possibility of vortexing, which could occur by nonuniform removal of sodium from the plenum through the single outlet nozzle. Experimental determination of the hydraulic behavior of the fluid in the upper plenum and in the baffle area is required.

All of the reactor hydraulic tests will be conducted in an existing water loop which has been modified recently to a pressurized hot water system. Now tests can be conducted over a water temperature range of 60 to 250°F under conditions geometrically and dynamically similar to high temperature — high velocity sodium. Water test data may be related to reactor coolant conditions (sodium between 750 to 1150°F) by adjusting water temperature and flow-rate to provide a Reynolds number similarity to reactor sodium conditions.

(1) Concept Development

(a) Fuel Element and Process Tube Vibrations

A full-scale fuel element will be mechanically vibrated to determine the individual fuel rod vibration frequency and amplitude. A short mockup fuel bundle will be fabricated and vibrated at the frequency and amplitudes obtained from the previous experiment to determine the number of cycles required to fail a fuel rod or process tube by fretting or fatigue. Confirmation of vibration frequency and amplitude will be gained during hydraulic testing of a fuel element in a water loop.

Preliminary analysis and planning of the experimental program is complete. Testing will be completed about May 1964.

(b) Fuel Channel Hydraulics

A full scale fuel element in a process channel will be installed in a water loop and pressure drop measurements will be made through the flow range expected in the reference reactor. In addition, the fuel channel variable

orifice will be calibrated in terms of coolant flowrate, pressure drop across the orifice, and orifice setting. Bypass leakage flow between the orifice and process tube will be measured. Fabrication of the test assembly is underway. Testing will be completed about December 1964.

(c) Control Channel Hydraulics

The orifice design for a control channel will be installed in the water loop to determine pressure drop versus flowrate through the orifice. Since velocities in the control channel will be small in magnitude, vibration experiments will not be required. The schedule and detailed plan of action will be established as soon as the design is set. Testing will be completed early in FY 1965.

(d) Reactor Upper Plenum Hydraulics

A model or full-scale section of the upper plenum circumferential baffle will be installed in a water loop and velocity distributions measured for various flowrates into the upper plenum. The mockup will be designed for hydraulic similitude to the reference reactor characteristics. Work will start and be completed in FY 1965.

(e) Concept Development Cost Reactor Hydraulics

	<u>(in \$)</u>
	<u>FY 1965</u>
Operating Funds	
Fuel element vibrations	-
Fuel channel hydraulics	15,000
Control channel hydraulics	10,000
Reactor plenum hydraulics	<u>65,000</u>
	90,000
Equipment Funds	<u>-</u>
	90,000

(2) Prototype Development

A prototype fuel element and variable orifice will require testing prior to fabrication of fuel element and orifice hardware for installation in the plant. Tests will be conducted in an existing water loop and will consist of fuel

element pressure drop and vibration measurements and variable orifice calibration. Prototype component fabrication and tests will be conducted in FY 1967 and are estimated to cost \$30,000.

g. Sodium Pumps

It is expected that free surface type pumps will be used in the large SGR plants. The largest sizes used to date are the EBR II and HNPF pumps which are about 7200 gpm capacity and the Enrico Fermi pumps which are about 11,800 gpm capacity. No difficulty is anticipated in scaling pump size up to 30,000 gpm required for the prototype.

(1) Concept Development

The design bases for free-surface, centrifugal sodium pumps have been well established from previous development programs. Sufficient information is available for this component to proceed directly to the prototype development phase.

(2) Prototype Development

Assuming a large sodium pump test facility is available, it would be desirable to operate a prototype pump in sodium at temperature prior to plant startup. Pumps will be tested in water by the manufacturer to confirm pump operating characteristics, however, hot sodium will provide advanced data on possible differential thermal expansion problems. The hot sodium tests could be performed in the plant, but prior testing would probably result in savings in time and money during plant startup.

Procurement of a 30,000-gpm pump and installation in a test facility would cost approximately \$325,000. Funds for procurement and installation of the pump will be required in FY 1967. These costs do not include the pump loop facility or operating cost of the loop during the test. Loop operating costs have been estimated for this test and are included under the Facility Operation section.

h. Sodium Valves

The 1000-Mwe SGR and the single main loop in it's prototype require large valves in the main sodium piping system. Each loop will contain a combination throttle-shutoff valve and a check valve in the cold leg piping, which is

a 26-in.-diameter line. No valves of this size have ever been built for sodium service, and testing is highly desirable.

The combination throttle-shutoff valve is employed to control thermal convection sodium flow below 5% of full flow during reactor decay heat removal operations. Sodium flow control during power operation is accomplished with the variable speed sodium pump. In addition, this valve will be used to shut off a loop from the reactor and other heat transfer circuits. Across the seat leakage is expected after thermal cycling, therefore the system is designed with downstream drains to take care of this leakage. Because of size, the combination throttle-shutoff valve will be of the butterfly type with a freeze seal on the stem. The check valve is expected to be of the usual swing gate design.

(1) Concept Development

The valve concepts proposed for the prototype plant are standard industrial designs with minor modifications for sodium application. The design bases for these components are available and require no further concept development.

(2) Prototype Development

A prototype throttle-shutoff valve and a check valve will be fabricated and tested in the Large Component Test Loop (LCTL). Tests will be performed with the throttle-shutoff valve to answer questions pertaining to:

(a) across the seat leakage after thermal cycling as a function of back-pressure; (b) throttling characteristics below 5% of full flow; and (c) maximum torque requirements for valve operation after build-up of sodium oxide in the freeze seal stem. The latter test is necessary to determine the valve operator torque requirements. Uncertainties associated with the swing check valve are: (a) operational reliability in sodium service; and (b) amount of forward pressure required to open the swing gate. Information on the latter uncertainty is required to determine the resistance of the valve gate to thermal convection flow during removal of reactor decay heat.

Design and procurement of the test units will require a period of 18 months and will be initiated in FY 1965. Preparation of the LCTL for the tests will be done concurrently. Testing and evaluation of results will require

manpower for 6 months, in addition to the regular LCTL operating staff. LCTL operating costs are included under the Facility Operation section.

Prototype Development Cost – Sodium Valves

	(in \$)	
	<u>FY 1966</u>	<u>FY 1967</u>
Operating Funds		
Check and block valves	75,000	-
LCTL modification	45,000	-
Testing	-	50,000
	120,000	50,000
Equipment Funds	-	-
	120,000	50,000

i. Steam Generator

An important phase in the development of the prototype plant is the development of a reliable and economical sodium heated steam generator for high-quality steam. Problems encountered due to caustic stress corrosion from Na-water reaction and chloride stress corrosion resulting from condenser leakage have been circumvented previously by the use of expensive noncorrosive linings, double-wall-tube construction and the reduction of steam temperatures to levels where lower strength alloys, which are not subject to the corrosion problem, may be used.

The AEC has undertaken an extensive program to develop a reliable and inexpensive steam generator concept for sodium application. Currently three concepts are being pursued: (1) the ALCO steam generator; (2) the B and W steam generator; and (3) the AI module steam generator. These concepts are scheduled for testing in the SCTI. Since the B and W design has just been initiated, testing of this unit will probably not be completed in time for use in the 200-Mwe SGR prototype plant. Therefore, the ALCO and AI designs are the leading contenders for prototype application. Either one of these units may prove to be the most desirable; however, the modular steam generator concept appears to have important features not contained in the ALCO unit. These features are as follows.

- 1) Design details should minimize the possibility of caustic and chloride stress corrosion by elimination of crevices. The module unit is crevice free because of the unique tube-to-tube sheet butt-weld feature.
- 2) The unit design should be adaptable to supercritical pressures in order to minimize development of a steam generator for the supercritical 1000-Mwe Plant. The module design provides this feature.
- 3) The concept should be inexpensive. The module concept is small enough to permit a large number of manufacturers to bid on a contract, thus introducing the cost benefits of competitive bidding. The module concept contains this feature, and cost estimates received to date verify the economy of this approach.

Because of these attractive features, a program was proposed to the U.S. AEC to design and develop the module concept.

In May of 1962, authorization was received to design and develop fabrication techniques for supercritical modular steam generators. Steam generator designs for 37 and 61 tube modules were completed. Tube-to-tube sheet, internal butt-weld equipment was developed to the point where internal welds at a depth of 8 in. can be successfully made. The welding operation is completely automatic. The welds are acceptable in accordance with Section VIII of the ASME Boiler and Pressure Vessel Code. An experimental stress program on a 37 tube module tube sheet was initiated. A three-dimensional photoelastic stress model was studied for critical stress areas. In addition, a machined module tubesheet of 4130 steel was made for experimental stress data. Strain gages were applied and the unit pressurized as high as 3000 psig. Stresses were within allowable limits in all areas. Data from the experiment determined the optimum wall thickness and tubesheet geometry of the spherical head configuration.

In April of 1963, an R & D Proposal to continue the Modular Steam Generator Program in accordance with AEC requests was developed. The scope of work for FY 1964 outlined in the proposal was as follows.

"A low temperature (evaporator) and a high temperature (superheater) module capable of generating 2400 psi steam at 1050°F will be designed and fabricated. In addition, the necessary design work for installation of the modules in the SCTI will be completed and the necessary equipment for installation of the modules will be ordered. The modules will be available for installation in the SCTI as a backup to the ALCO unit."

The above project was funded in July 1963, and work has been initiated except for the design and procurement of equipment for installation of the modules in the SCTI. A 37-tube module design will be used for both the evaporator and superheater. The development work, which was completed with the 37-tube supercritical module head and the welding equipment, is directly applicable to the 2400-psi module design. In fact, the identical head design developed for the supercritical modules will be used for the 2400-psi modules.

The remaining development tasks required to demonstrate the adequacy of the module concept for the prototype plant are as follows.

(1) Concept Development

(a) Cyclic Fatigue Experiment

A 37-tube module tubesheet is being fabricated to investigate fatigue from pressure cycling. Strain gages will be applied to the tubesheet at critical locations and the unit will be cycled according to expected plant transient conditions. Testing will be completed about April 1964.

(b) Operational Experience

Under the present program it is planned to fabricate two 37-tube evaporators and two 37-tube superheater modules for testing in the SCTI. The units, with a combined rating of 21.6 mwt, will provide operational experience with modular steam generators mounted in parallel, the normal mode for prototype operation. The units have been designed, and the fabrication was started in January 1964. Design modifications to the SCTI for module installation is currently under study. The first set of evaporator-superheater modules will be ready for installation in the SCTI in June 1964. The second set will be ready for installation in August 1964. Although the ALCO steam generator unit is also scheduled for delivery to the SCTI at approximately the same date, previous fabrication difficulties with this unit may delay delivery. In any event,

the primary interest for the prototype plant is early testing of the module concept, and it is recommended that this unit be tested first. Operating costs of the SCTI during a 15-month test period for the modules have been included in the Facility Operation section.

(c) 61-Tube Module Head Experiment

Testing of the 37-tube modules will provide the basic data for design of the more economical 61-tube modular steam generator. One additional experiment is required to verify the design of the slightly larger unit. An experimental stress program for a 61-tube module head will be required to verify that stress concentrations in a larger head do not exceed allowable limits. The experiment will be similar to that conducted with the 37-tube head.

(d) Concept Development Costs -- Steam Generator

	(in \$)
	<u>FY 1965</u>
Operating Funds	
<u>37-Tube Modules</u>	
Cyclic fatigue experiment	-
Design and fabrication of SCTI modules	30,000
SCTI modification and module installation	90,000
	<u>120,000</u>
<u>61-Tube Modules</u>	
Module head experiment	<u>50,000</u>
	170,000
Equipment funds	<u>-</u>
	170,000

(2) Prototype Development

The steam generator concept proposed for the prototype plant, the 61-tube module design, will be adequately developed and tested with a 37-tube module. Although an exact prototype will not be utilized in this test the scale-up factor to a 61-tube unit is considered to be well within engineering principles,

particularly with the experimental stress data on a 61-tube head. No prototype testing is required for this component.

j. Sodium Instrumentation

Several instrumentation items require development for the 200-Mwe prototype because of higher operating temperatures and larger component size.

Flowmeters have not been built for sodium systems in the sizes required for the prototype (26 in.). The largest electromagnetic flowmeters in use are 14-in. flowmeters at HNPF and the Fermi plant. Other types of flow measuring devices which might be used are orifice plate, venturi meter, or flow tube. Flow might also be determined using the pressure drop through one of the system components such as the IHX.

The SRE and HNPF employ sodium level gauges based on induction coil techniques. This type level gauge is not accurate or reliable at sodium temperatures generated in the prototype reactor (1150°F). Previous development programs have investigated other level measuring concepts, such as differential pressure transducers and "J" tube level indicators, which appear satisfactory for measuring sodium level. Prototype testing of one of these concepts is required.

Electrical resistance heaters are contemplated for system preheating. Preliminary analysis indicated that a considerable cost saving may be made by using only one heater on the bottom of the pipe, rather than several distributed around the circumference as in HNPF. Tests will be required to confirm the calculated temperature distributions and effectiveness of heating.

(1) Concept Development

Previous development activities with improved sodium level gauges and with electrical resistance heating have provided sufficient information to complete the concept development phase for these components.

(a) Sodium Flowmeters

Large sodium flowmeters require further concept development before a specific flowmeter device can be selected for the large SGR. Prime candidate for flowmeters in 26-in. pipe sizes are orifice plate, venturi flow tubes, and possibly electromagnetic devices. Design studies of each

flow-measuring device will be conducted, followed by testing of small units to appraise sensitivity and reliability in a sodium environment. Tests will be conducted in the LCTL facility at temperatures and velocities commensurate with prototype conditions, but size and mass flowrate cannot be duplicated because of limited flowrate capacity. The tests will provide sufficient information to select the unit embodying the most favorable characteristics.

(b) Concept Development Costs – Sodium Instruments

	(in \$)
	<u>FY 1965</u>
Operating Funds	50,000
Equipment Funds	<u>-</u>
	50,000

(2) Prototype Development

(a) Sodium Level Gauge

The most severe condition for sodium level measurement in the prototype plant is the reactor upper sodium plenum. The level indicator must monitor the sodium level over a range of 15 ft and over a range of temperatures from 350 to 1150°F.

Through design studies and review of previous development results, one concept will be selected for the prototype reactor and a full-scale unit designed and fabricated for prototype testing. The LCTL will be used to test the instrument under simulated reactor environmental conditions. Tests will determine the units sensitivity with level and temperature changes and its overall reliability.

(b) Electrical Resistance Heating

A typical section of 26- to 30-in. pipe run in the prototype plant will be constructed with a single heater on each pipe leg and elbow. Tests will consist of preheating the piping and elbows from ambient to 350°F and measuring temperature distributions around the pipe circumference. Strain gages will be employed to measure strains in the pipe wall due to nonuniform heating.

Further testing will be conducted to determine the minimum electrical heating required to prepare for sodium filling. Calculations of pipe stresses from filling a cold piping system with 350°F sodium indicate this approach may be feasible with only preheating of equipment and local piping near the equipment. The facility will be built to include sodium service, and testing of this concept will be pursued.

(c) Sodium Flowmeter

The concept development phase will evaluate alternate flow measuring devices and select one concept for the prototype plant. Assuming a large sodium loop is available by 1967, a full-scale prototype flowmeter will be fabricated and tested for instrument characteristics and calibration. Costs for the large loop construction are not included in this project, but operating costs of the loop during the test are included under the Facility Operation section.

(d) Prototype Development Costs — Sodium Instrumentation

	(in \$)		
	<u>FY 1965</u>	<u>FY 1966</u>	<u>FY 1967</u>
Operating funds			
Sodium level gauge	50,000	-	-
Electrical resistance preheat	105,000	-	-
Sodium flowmeter	-	-	<u>50,000</u>
	155,000	-	50,000
Equipment funds	<u>15,000</u>	-	<u>-</u>
	170,000	-	50,000

k. Design and Safety Evaluation

A design and an analytical effort is required to establish the plant design criteria and description, to provide the test criteria for component development and to integrate the results of the development program into final plant operating characteristics and design configuration. This activity assures continuity in the direction of the development program and the design of the 200-Mwe prototype plant. In addition, this project will review requirements of the 1000-Mwe plant and, whenever possible, incorporate these requirements

into the prototype plant design and establish development criteria to meet conditions for both plants.

Design studies of the reactor structure and components will continue, with particular emphasis on structural analyses of the calandria, reactor vessel and support features. Reference design drawings, loading diagrams and thermal performance characteristics for steady-state and transient conditions will be prepared. These reference design and operating conditions will be collected into a final set of test criteria for the development program.

Reactor physics analyses will be conducted, utilizing the methods developed in the reactor physics development program, to determine reactor characteristics. Particular attention will be directed to fuel burnup characteristics, to stability and control of large SGR cores, due to xenon shift and temperature and power coefficients, and to fuel and control rod management schemes.

Further study is needed in the area of instrumentation and control. A major effort is required to analyze the instrumentation and controls associated with steam dump, cold startup, hot startup and load following system requirements. The general approach to this problem will involve familiarization with system part load operating conditions for various parameter control points, followed by specific instrumentation design. Initial work will be directed to the steam generator and feedwater system and gradually work through the sodium system control and instrumentation and thence to the integrated reactor control and protective system. Results from this study will establish the steady-state and transient operating conditions of plant components, such as steam generators, intermediate heat exchangers and sodium pumps. Test criteria, where ever applicable, will be prepared.

General design studies of component configuration and arrangements will be conducted. Stress analyses of components will be made, particularly from a thermal stress standpoint, to determine the limiting component which will set the maximum rate of temperature change during load-following and startup operations.

Development Costs

	(in \$)	
	<u>FY 1965</u>	<u>FY 1966</u>
Operating funds	200,000	100,000
Equipment funds	<u>-</u>	<u>-</u>
	200,000	100,000

2. Reactor Physics Program

An experimental program in the area of reactor physics is necessary to confirm and optimize the current design concept of the 200-Mwe prototype reactor. Emphasis will be placed on measurements which will directly implement the design effort or provide data in areas where analytic means are presently inadequate. This program should significantly reduce reactor overdesign, required to allow for uncertainties in the present physics design effort. The main areas requiring experimental study are (a) fuel burnup lifetimes, (b) Doppler coefficient measurements and (c) specific critical experiments. A strong analytical effort will be required to ensure that information obtained from the experimental program will be interpreted adequately. A detailed discussion of the needs relating to each of these areas will be found under each task heading.

a. Burnup Studies

Accurate determination of fuel lifetime is necessary to minimize the risk involved in establishing guarantees of fuel cycle costs for the large SGR. It is apparent that general present-day techniques are not adequate in this area. Lifetime predictions and experiments vary from 10 to 40% and the calculations may predict either longer or shorter core lifetimes. The limited experimental information, which is available from water-moderated reactors, is not the best basis on which to predict burnup of the long-lived SGR cores (37,000 to 40,000 Mwd/MTU peak). The water-moderated cores use UO_2 fuel in a well thermalized spectrum and in most cases have reached only modest burnups ($\sim 10,000$ Mwd/MTU). Present sodium graphite cores are UC fueled and operate with a considerable epithermal neutron contribution.

An adequate experimental program must treat the problem from two different approaches. One of these is the spectroscopic examination of the

irradiated fuel slugs. Currently, these fuel rods are disected, polished, and examined carefully by microscope to observe changes in the microstructure. In addition, radiochemistry can study the burnup of several Pu isotopes and the distribution of 2 or 3 fission products in the rods. With very little or no further sample preparation, a vast amount of additional information may be obtained by a mass spectrometer using isotope dilution techniques. This would include isotopic analysis for total mass and spatial distribution (due to migration as well as flux shapes) of U^{235} , Pu^{239} , 240 , 241 , 242 , and the complete range of fission products. Analysis of isotopic ratios can yield important information on the variations in the plutonium capture-to-fission ratio. Because of the faster neutron spectrum, the effective values of this ratio are expected to differ considerably from those in water moderated reactors.

The second approach uses oscillator techniques to perform reactivity worth measurements of irradiated fuel slugs in an SGR type spectrum. Such measurements, made at a variety of exposure points, would reveal the magnitude of any error in the analytical prediction of state of loss of reactivity for such fuel. Information will be obtained in the thermal and epithermal cross sections of the gross fission products as well as on the reactivity changes.

The initial phase of this task will be the construction, checkout and calibration of both the mass spectrometer and the oscillator. Mockup samples of burned fuel will be used for comparison purposes and for checkout of the experimental method. During this period it will be possible to develop and improve computer codes for the analytical treatment of burnup. A wide range of irradiated fuel sampels will be available for detailed studies. These include full UC elements irradiated in Hallam to ~ 5000 Mwd/MTU, as well as capsule irradiations at other facilities giving exposures as high as 40,000 Mwd/MTU. The results from this program should clearly form a basis for an adequate calculational program previously impossible for sodium graphite reactors.

Development Costs – Burnup Studies

	(in \$)	
	<u>FY 1965</u>	<u>FY 1966</u>
Operating Funds	115,000	90,000
Equipment Funds	<u>140,000</u>	<u>5,000</u>
	255,000	95,000

b. Doppler Coefficient

One of the major contributions to normal control and stability of SGR's, as well as the main mechanism for shutdown, is the prompt temperature coefficient of reactivity. A major component of this is due to the Doppler broadening of resonances in the fertile and fissile materials present. Doppler measurements to date have been limited to U metal or UO_2 fuel, with measurements being made in single slugs in a 1/E spectrum. The agreement between theory and measurement is acceptable for U metal, but has been poor for UO_2 . To guarantee an adequate treatment of the problem for sodium graphite reactors, it will be necessary to make measurements using UC fuel located in a typical SGR spectrum.

The method of measurement that will be used is the reactivity oscillator, which can have orders of magnitude more sensitive than any other. Unfortunately, it is much more susceptible to systematic error because it does not measure the capture rate directly, but infers it through changes in reactivity. However, the work currently being performed at AI, in this area, has as one of its major objectives to find and evaluate the magnitude of systematic errors present in this measuring technique. The samples to be oscillated may be studied at as many as five temperatures ranging from room temperature to 1000° F.

Since Doppler measurements have not been made on UC to date, it will be necessary to begin these measurements using small samples in a 1/E spectrum. This is necessary to place the analytic effort on a firm foundation. Once this is accomplished, the measurements may then be made in the spectrum to be expected in a sodium-graphite power reactor. The spectrum shapes will be investigated, both analytically and experimentally.

Development Costs – Doppler Coefficient

	(in \$)	
	<u>FY 1965</u>	<u>FY 1966</u>
Operating Funds	70,000	70,000
Equipment Funds	<u>5,000</u>	<u>10,000</u>
	75,000	80,000

c. Critical Experiments

A series of critical experiments will be carried out over a period of two years in support of the 200-Mwe SGR development program. The purpose of these experiments is to provide data which will serve to establish a basis for checking and verifying the design of the reactor. The program will be directed towards reducing uncertainties in present calculational techniques and towards investigations of safety and stability problems.

Since uncertainties of as much as 3 to 4% still exist in predicting k_{eff} for some of the UC criticals studied to date, additional studies are necessary. These studies will include measurements of gross effects such as critical mass, k_{eff} and cell parameters. A knowledge of the variation of neutron importance with energy and position in the core is extremely useful in the design of power reactors, which are not well thermalized as in the case of SGR. The experimental determination of the neutron spectrum by use of resonance activation detectors and fission counters will aid in developing and confirming calculational models used in the description of the neutron flux spectrum at a point in a cell for heterogenous reactors.

Measurements to determine control rod worth, control rod heating for partially inserted rods off center and the determination of the overall effect of control rod programming will greatly reduce uncertainties which presently exist in these areas. It has been estimated that the uncertainty in calculating the heating rate, from a given neutron absorption rate, in a dysprosium oxide control rod, may be of the order of $\pm 25\%$. In addition, the neutron capture rate, which forms a source term in the heating calculation, is believed to be much more uncertain.

Safety and stability problems will be investigated. An important consideration in reactor safety is the formation of voids in the coolant during an excursion. Measurements to determine the effects of such voids will provide the necessary data for the improvement of the calculational techniques. A dynamic instability as manifested in severe temperature oscillations, has occurred in a presently operating sodium-graphite reactor due to fuel rod displacement or bowing. In order to predict these effects in present fuel element design, it is necessary to determine the reactivity changes for various fuel pin and element positionings within the fuel process channel.

A summary of the critical measurements to be performed in support of the 200-Mwe reactor development program is here itemized.

- 1) Intracell parameter measurements
- 2) Measurements of critical mass and k_{eff}
- 3) Neutron spectrum measurements
- 4) Neutron importance measurements
- 5) Power distribution measurements in support of control rod programming studies
- 6) Zoned fuel loadings to support fuel element programming as the elements are depleted by burnup
- 7) Control rod studies
- 8) Fuel element bowing studies
- 9) Sodium void measurements

Development Costs – Critical Experiments

	<u>(in \$)</u>	
	<u>FY 1965</u>	<u>FY 1966</u>
Operating Funds	200,000	150,000
Equipment Funds	<u>30,000</u>	<u>-</u>
	230,000	150,000

d. Reactor Physics Analysis

An effective experimental program requires a parallel analytical effort. This will involve the design of experiments, development of theoretical methods, analysis of data, and correlation of theory with experiment. All of these areas are important to the development program if a 200-Mwe prototype sodium-graphite reactor is to be designed with confidence. Proper design of experiments will assure that the correct range of nuclear parameters is explored and the maximum useable information is obtained. Correlation of theory and experiment must be made in order to understand and resolve any discrepancies, to evaluate the accuracy of proposed design methods, and to condense the data to its most useful form.

Burnup studies are the most important uncertainty in the prediction of fuel cycle economics of large sodium-graphite reactors. Studies will be made of actual operating data on reactivity and isotopic fuel compositions from the HNPF, as well as from water reactor programs, Yankee, Dresden, etc. Nuclear data for the plutonium isotopes and for fission products will be reviewed. Particular attention will be devoted to Pu^{240} resonance selfshielding and secondary fission product chains. Results from the SGR physics development program will be utilized as data become available.

Doppler coefficients will be calculated for uranium carbide using the most recent resonance parameters.

Critical experiments will be correlated with a modified 2-group theory if possible, since this is easy to use for design studies. Two-dimensional burnup studies on a computer are also less expensive if done in two groups rather than more. Monte Carlo calculations will be made and a collision probability theory developed for 18-rod element and perhaps extended to other geometries. These developments should be necessary to assess any fuel rod bowing effects on reactor stability.

Sodium void coefficients and neutron importance effects will be studied in order that the safety and stability of large cores can be predicted with confidence. Sodium cross sections and resonance parameters will be compiled and put into a code library so that these effects can be calculated theoretically in detail. Perturbation theory techniques utilizing adjoint fluxes calculated from multigroup codes will be developed and compared with experimental measurements of neutron importance.

Development Costs — Reactor Physics Analyses

	<u>(in \$)</u>	
	<u>FY 1965</u>	<u>FY 1966</u>
Operating funds	100,000	100,000
Equipment funds	<u>-</u>	<u>-</u>
	100,000	100,000

3. Facility Operation

a. Sodium Reactor Experiment (SRE)

The fuel materials and fuel component development programs are strongly dependent upon the continued operation of both the SRE and Hallam facilities. Each of these plants will have uranium carbide cores by 1965. The preparation of these large quantities of fuel will provide realistic fuel fabrication costs for uranium carbide. The irradiation and examination of the UC fuel elements currently in the SRE and Hallam, as well as the surveillance of the full cores of UC, are vital factors in the successful development of UC fuel elements. In terms of testing the exact 200-Mwe prototype conditions of mechanical design, power density, and burnup rate, neither reactor can simultaneously duplicate all of these physical configurations and conditions. However, the combination of the fuel element performance data with the specific fuel materials development program data, from well known conditions, will provide a sound basis for fuel warranties. The SRE-PEP program is actually designed for higher design capability than for the prototype conditions. The results from the SRE-PEP program will strongly contribute to the 200-Mwe prototype fuel program.

The use of the Hallam plant to test mechanical design and extended power tests on special elements is being considered. The costs of operating the SRE are included as part of this development program while the operation of the Hallam plant is not a part, due to the utility contract obligations. Funds for SRE-PEP modifications have not been included, but are assumed to be funded according to current plans. SRE operating costs through FY 1967 are required for the prototype fuel and cladding development program.

Operation Costs

	(in \$)		
	<u>FY 1965</u>	<u>FY 1966</u>	<u>FY 1967</u>
Operating funds	2,600,000	2,500,000	2,500,000
Equipment funds	<u>40,000</u>	<u>40,000</u>	<u>40,000</u>
	2,640,000	2,540,000	2,540,000

b. Sodium Component Test Installation (SCTI)

This facility has been constructed by the AEC to provide operational experience with intermediate heat exchangers and steam generators for sodium application. The facility will be required to test the module steam generator concept, which will involve approximately 15 months of SCTI operation.

Operation Costs – Sodium Component Test Installation

	(in \$)	
	<u>FY 1965</u>	<u>FY 1966</u>
Operating funds		
IHX installation - ALCO unit	40,000	-
SCTI operation	<u>850,000</u>	<u>200,000</u>
	890,000	200,000
Equipment funds	<u>10,000</u>	<u>5,000</u>
	900,000	205,000

c. Large Component Test Loop (LCTL)

Component development programs involving the LCTL operation include the calandria structural, large sodium valve, sodium flowmeter, and sodium level gauge. These tests will be conducted over a 3-yr period and will require an operating and maintenance staff during this time. Operating costs of this facility are in addition to the costs required for the individual component development activities utilizing this facility

Operation Costs – Large Component Test Loop

	(in \$)		
	<u>FY 1965</u>	<u>FY 1966</u>	<u>FY 1967</u>
Operating Funds	125,000	125,000	125,000
Equipment Funds	<u>-</u>	<u>-</u>	<u>-</u>
	125,000	125,000	125,000

d. Sodium Pump Test Facility (SPTF)

A large sodium test loop is required to develop pumps and flowmeters for the prototype plant. There is no such facility available at this time,

although construction of such a facility is currently under consideration by the AEC. Assuming a facility is available in time for use in the prototype plant development program, testing of a sodium pump and flowmeter would require approximately one year of loop operation.

Operation Costs — Sodium Pump Test Facility

	<u>(in \$)</u>
	<u>FY 1967</u>
Operating funds	850,000
Equipment funds	<u>10,000</u>
	860,000

B. 1000-Mwe SGR PLANT DEVELOPMENT PROGRAM

The development program for the 1000-Mwe SGR is basically an extension of the prototype program in the areas of fuel and cladding, control rod, steam generator, and reactor physics. All other development work for the prototype applies to the 1000-Mwe plant. Continued development in these areas is necessary because of the higher core performance and steam conditions for the larger plant.

1. Component Development

a. Fuel and Cladding

The major differences in the fuel specifications for the 1000-Mwe plant are the increased central temperature to 2000°F and the higher specific power of 72 kw/kg U. These changes require statistically demonstrated large scale test results for use in fuel warranties. The fuel and clad development program for the 200-Mwe prototype plant provides the general basic data for UC technology. In addition to this basic development program, a program is required to specifically satisfy these new requirements. It is assumed that the current development program will be continued.

The available fuel materials data indicate that as the fuel temperature increases the fission gas release increases, the fission gas release decreases as the UC carbon content increases, and the free uranium in hypo UC migrates under high thermal gradients. The 1000-Mwe plant fuel temperature conditions of 2000°F, and almost a 50% increase in peak specific power to 72 kw/kg U, enhance the mechanisms for fission gas release and density changes in the hypo UC on the low side of the carbon composition. It is well known that uranium metal undergoing phase changes and approaching a molten phase releases a higher percentage of fission gases. Thus, one of the main aspects of the fuel development program is to determine the fuel composition specifications to achieve the required irradiation conditions and to develop high yield low cost fabrication methods based on these specifications.

The higher capability conditions require a development program in two specific areas: (1) UC process development and or an alternate clad design or development, and (2) irradiation performance demonstration at 1000-Mwe fuel conditions. All of the tasks and results required by this program will be directly

related to and dependent upon the results from each phase of development for the 200-Mwe plant. The products of these combined programs will yield reliable data for fuel design criteria and fuel warranties.

(1) Concept Development

(a) UC Processing

The carbon composition required to meet these conditions is expected to be in the range of 4.6 to 4.9 wt % C. The exact composition range will be a result of (1) the evaluation of the irradiation results, (2) the process capability to achieve the required composition with reasonable yields and cost, and (3) the tolerance of the chemical analytical methods. In order to produce material of this composition, process development will be required primarily in the fuel slug casting operation. The preparation of UC feed stock with minor composition changes will require verification. Analytical precision and accuracy for both the feed stock and final product will require improvement. It is important that the development work for the 200-Mwe prototype be well coordinated with this effort since some of the objectives are quite similar.

(b) Alternate Clad

One method that could alleviate the effects of producing fuel slugs to a tight carbon specification near stoichiometry, is to develop a clad material that will resist the gettering action of the clad for carbon from the UC₂ source. One promising material resulting from sponsored research at Armour Institute is a high strength corrosion resistant zirconium dispersion alloy. This material would require complete fabrication, physical properties, compatibility, and radiation testing on both small and large scale. Zirconium takes up an initial surface layer of available carbon but this thin carbide layer reacts or diffuses slowly through the remaining material. A second approach to the clad problem is to develop and test a zirconium sleeve within the stainless steel jacket. The Zr sleeve could be either separate or integral with the steel jacket depending upon technical feasibility and cost. The purpose of the sleeve is the same as the dispersion clad objective. The use of the zirconium base materials will also improve the neutron economy. A third method which shows conceptual promise is the vented fuel design. An analysis of plant cover gas activity problems is required to be followed by small scale tests and proof tests. This technique could replace other clad efforts.

Depending upon the schedule, budget, and initial experimental results it may be feasible to combine some of the UC process development effort for 200-Mwe and 1000-Mwe plants at an overall reduced total program cost.

Tasks for UC process development or alternate clad are as follows.

- 1) Fabricate UC feed stock for slug casting specification.
- 2) Develop method to produce cast slugs with small carbon tolerance near stoichiometry, e.g., eliminate the carbon electrode by electron beam melting, hollow cathode discharge melting, plasma arc melting.
- 3) Develop quality control procedures for feed stock and cast slugs.
- 4) Develop alternate Zr clad material; e.g., Zr dispersion alloy or Zr sleeve in 304 SS.
- 5) Analyze, evaluate, fabricate and test vented fuel concept.

(c) Irradiation of Fuel and Clad for 1000-Mwe Plant

The general status and current irradiation development program described under the 200-Mwe section directly applies and supports this 1000-Mwe plant fuel and clad development program. It is assumed that the present program will be continued. The general problem areas are somewhat broader in scope due to the change in performance requirements. For instance, all three phases of irradiation development will be tested and evaluated; namely, capsules, rods, and elements. The irradiation tests of various carbon compositions performed in multi-compartmented capsules will be used to provide the basic fuel materials data for fuel rod design. This performance requirement will likewise provide the broad guidance for the UC process development. The fuel rods and fuel elements must then be tested in large scale size to provide the statistical results required for fuel warranty preparation. The mechanical aspects of the fuel design require proof tests as well as the individual materials and components.

The clad irradiations will follow the same basic plan. Capsule tests will be used for technical feasibility and fuel rod and fuel element tests will provide the reliability data.

Tasks for irradiation of fuel and clad are as follows.

- 1) Capsule fuel irradiations at 2000 to 2100° F with 72 kw/kg U specific power to determine optimum carbon composition, fission gas release and density changes with burnup.
- 2) Fabricate, irradiate and examine, fuel rods and fuel element hardware under 1000-Mwe plant conditions; use NRX, SRE, Hallam.
- 3) Fabricate and test alternate clad materials and designs; include defect tests and vented clad design test.
- 4) Prepare specifications and fuel element design criteria for 1000-Mwe plant.

(d) Concept Development Cost – Fuel and Cladding

		(in \$1,000)				
		<u>FY 1965</u>	<u>FY 1966</u>	<u>FY 1967</u>	<u>FY 1968</u>	<u>FY 1969</u>
Operating funds						
UC process development		50	200	150	30	30
Alternate clad development		50	200	200	50	30
Irradiation and examination fuel and clad capsules		200	550	500	300	300
Irradiation and examination fuel rods and fuel elements		40	700	750	550	500
		<u>340</u>	<u>1,650</u>	<u>1,600</u>	<u>930</u>	<u>860</u>
Equipment funds		<u>030</u>	<u>70</u>	<u>100</u>	<u>70</u>	<u>70</u>
		370	1,720	1,700	1,000	930

(2) Prototype Development

The experimental data gained from fuel elements operating in SRE-PEP and HNPF, coupled with long rod and capsule irradiations in other reactors, will provide necessary information for designing the fuel elements for the 1000-Mwe plant. Of course, information gained from fuel elements operating in the 200-Mwe prototype plant will greatly add to the knowledge of fuel element performance under the SGR conditions. The prototype reactor will be capable of accepting special fuel elements for testing under the 1000-Mwe reactor conditions. However, because of possible contractual difficulties, use of the prototype reactor for test purposes has not been considered in this development program.

b. Control Rod

The reference control rod for the prototype SGR has a neutron absorber element which operates dry in a helium-filled thimble. Heat generated in the absorber element is transferred to the sodium coolant by radiation and conduction across a helium gas gap to the thimble, which is cooled by flowing sodium. This design is satisfactory for the prototype SGR, which is limited in specific power by the assumption of conservative operating conditions for the UC fuel element. It is expected that the development program on UC will demonstrate that this fuel is capable of substantially higher specific power, which would permit a reduction in size and cost of the reactor. The most economic specific power, particularly in large reactors of 500-Mwe or higher power level, will probably require improved means of control rod cooling. Direct cooling of the neutron absorber by liquid sodium, i. e., a "wet" control rod, is an obvious way of eliminating the heat transfer problem.

Design studies of wet control rods have yielded no practical alternative to mechanisms which operate in the sodium coolant; this difficulty arises from the distance, approximately 13 ft, which a control rod must move in an SGR. A major part of the development of a wet control rod therefore involves the finding of materials for bearings and gears or screws which can operate for long periods of time in sodium at about 1200°F. A program for development of the wet rod was initiated for the AEC in FY 1963 but had to be discontinued in favor of the dry rod in view of inadequate funding to pursue both approaches for the near term prototype plant. However, preliminary results before terminating the tests were encouraging, and confirmed that the wet rod was a feasible approach.

The proposed program includes: design and analytical studies of mechanisms and control elements; study and testing in support of the development of materials for mechanisms to be operated in 1200°F sodium; further testing of seals; design, fabrication and assembly of a full-scale prototype control rod; design and modification of test facilities to provide sodium-test capability for prototype testing; and functional testing under environmental conditions (other than nuclear) of the prototype control rod.

(1) Concept Development

Before producing fabrication drawings of a prototype from the conceptual control rod designs which have been made, the most important task

is recognized to be the development of materials for mechanisms that will resist wear and galling when subjected to the service conditions to be encountered. In this regard, two new types of materials testing assemblies will be utilized in addition to the type which has been used in the past for materials evaluation on this program. The devices on hand utilize two matched spur gears mounted on individual shafts which are loaded torsionally while being rotated. The torsional load transmitted to the gears is also imparted to replaceable sleeve-type bearings. Testing is performed with the gears and bearings immersed in 1200°F sodium. The second type of materials test device will be used for testing of journal bearings of various sizes at various speeds and loads; simultaneous gear testing will be avoided. The third type will be for test of thrust bearings. The system design for the two new devices will include provision for measurement and control of oxygen content of the sodium, an important variable in the friction between surfaces of materials tested in the past. For optimum results, both in use of facilities and personnel and in the production of test data, two each of the new devices will be fabricated.

Testing of both rotary and linear seals for sodium and sodium vapor service will be performed. Testing already performed, while incomplete, indicates that acceptable designs of both types of seals have been produced; additional testing will establish endurance limits and verify indicated adequacy.

The absorber assembly for a wet control element utilizes a circular array of 18 rods containing the poison material. The reference poison material for the control elements in the prototype plant, dysprosium oxide, is entirely suitable for the wet rod approach. Hence, irradiation experiments performed in the prototype plant development program are applicable to the 1000-Mwe control element.

Concept Development Costs — Control Rod

	(in \$)		
	<u>FY 1965</u>	<u>FY 1966</u>	<u>FY 1967</u>
Operating funds			
Design and analyses	25,000	—	—
Test assembly design and fabrication	25,000	15,000	—
Testing	40,000	50,000	25,000
	<u>90,000</u>	<u>65,000</u>	<u>25,000</u>
Equipment funds	—	—	—
	<u>90,000</u>	<u>65,000</u>	<u>25,000</u>

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(2) Prototype Development

Once suitable materials, components and nuclear design have been selected, fabrication drawings of the prototype will be prepared. The conceptual design presently favored (designated the SGR Mark II) employs a sodium-immersed screw-and-roller-nut actuator mechanism with rotary seal. The reference seal uses a sodium-vapor condensing zone below an O-ring seal. An alternate rotary seal employing sodium-immersed, bellows-mounted face seals has also been designed. An alternate design approach under consideration (designated the SGR Mark I) utilizes a bellows-sealed, sodium-immersed, telescoping rack-and-pinion actuator. The final choice between the designs is dependent upon results of the developmental test efforts.

During design of the prototype wet control rod, design for modification of the test facility will also be performed. The present facility, while adequate in size, does not have sodium service.

After fabrication of the prototype and modification of the test facility, environmental testing of the control rod will be started. The environmental test conditions will include all but those of, or those resulting from, the expected nuclear environment. During the first phase of testing, the operating characteristics will be determined, compared to performance specifications, and developmental changes made, as required, to assure conformance to the specifications. After the developmental testing phase, endurance testing will be performed to verify adequacy of the design to meet life requirements and the degree of reliability required by power plant operating practices.

Further testing of the prototype wet control rod could be achieved by inserting the rod in the prototype reactor, which is designed to accept either dry or wet type control rods. This test would depend on contractual agreement with the prototype plant customer. Costs for this test have not been included in this program.

Prototype Development Cost — Control Rod

	(in \$)	
	<u>FY 1966</u>	<u>FY 1967</u>
Operating funds		
Prototype design and fabrication	95,000	25,000
Prototype test facility modification	5,000	20,000
Prototype test	—	50,000
	<u>100,000</u>	<u>95,000</u>
Equipment funds	15,000	5,000
	<u>115,000</u>	<u>100,000</u>

c. Steam Generator

The modular steam generator concept, currently being developed for the 200-Mwe prototype plant, appears to be capable of operating at higher pressures, such as the 3500 psig/1000°F steam conditions for the 1000-Mwe plant. Assuming that a 61-tube modular steam generator for 2400 psig/1000°F steam has been successfully developed for installation in the prototype plant, verification of the units higher pressure capabilities will be required for the 1000-Mwe plant.

(1) Concept Development

The development program for the 200-Mwe prototype plant will provide sufficient information to design the modular steam generator for supercritical conditions. Therefore, no further concept development is required for this component.

(2) Prototype Development

The scope of the development program for supercritical operation will include the design and fabrication of two 61-tube high temperature modules and two 61-tube low temperature modules, capable of generating 3500 psig steam at 1000°F. The units will be of the same basic design as those proposed for the prototype plant. The supercritical module head design will be based on experimental stress data obtained with a 61-tube module head for subcritical pressures. Module lengths will be reduced because of improved heat transfer properties of the supercritical fluid. Tube wall thickness may increase slightly because of the pressures.

The units will be installed in the Sodium Component Test Installation (SCTI) for testing under steady-state and thermal transient conditions. Since the SCTI steam-feedwater system is not designed for supercritical pressures modification of the facility will be required. These modifications and SCTI operating costs are covered in the Facility Operation section.

Prototype Development Costs — Steam Generator

	(in \$)	
	<u>FY 1965</u>	<u>FY 1966</u>
Operating funds		
Module design	25,000	—
Module fabrication	150,000	30,000
	<u>175,000</u>	<u>30,000</u>
Equipment funds	—	—
	<u>175,000</u>	<u>30,000</u>

d. Design and Safety Evaluation

A design and an analytical effort is required to establish the plant design criteria and description, to provide the test criteria for component development and to intergrate the results of the development program into final plant operating characteristics and design configuration. The project will guide the equivalent effort for the prototype plant to assure prototypical features of the larger SGR are incorporated in the prototype plant design. Improved features, which might not be incorporated in the prototype, may be designed into the 1000-Mwe SGR because of the longer development time for the later large SGR. Current thoughts along these lines are not expected to greatly alter the development programs.

Design and reactor physics studies will be continued to evaluate the economic incentive for larger fuel area per cluster. Studies will investigate twenty-four and thirty 0.500-in.-diameter fuel rods per element and other alternatives. Variable fuel-to-moderator ratios for the large SGR will be evaluated. A variable neutron spectrum might be achieved by irradiating a fuel element in a tight lattice space to 12,500 Mwd/MTU and then moving the fuel element to a more open lattice for the remainder of its core exposure. Preliminary calculations indicate that a 20% gain in k_{eff} is feasible. More detailed physics

calculations will be performed, and core and reactor structure layout studies will be made. Reactor physics calculations will be performed initially with existing analytical techniques, followed by improved methods developed from the reactor physics program describe later in this section.

The present 1000-Mwe SGR core utilizes 84 combination shim-safety control rods. A reduction in number of regulating rods appears feasible with a significant cost saving, by properly employing stationary rods. Physics and heat transfer studies are required to evaluate this concept. Reactor control problems, due to xenon shift and temperature coefficients, will be studied for the 1000-Mwe SGR core. Also fuel management and control rod programming requirements will be evaluated.

Part-load operation with the supercritical steam cycle will be studied, as well as cold and hot startup problems. Plant control and instrument requirements will be more clearly defined.

General design studies of component configurations and arrangements will be made, and a reference plant description will be prepared and kept up to date when design studies indicate desirable changes. Reference drawings and plant performance characteristics will be prepared and will be employed as test criteria for the development programs.

Development Cost — Design and Safety Evaluation

	(in \$)			
	<u>FY 1965</u>	<u>FY 1966</u>	<u>FY 1967</u>	<u>FY 1968</u>
Operating funds	200,000	300,000	400,000	400,000
Equipment funds	—	—	—	—
	<u>200,000</u>	<u>300,000</u>	<u>400,000</u>	<u>400,000</u>

2. Reactor Physics Program

An experimental program in the area of reactor physics is necessary to confirm and optimize the current design concept of the 1000-Mwe SGR. Emphasis will be placed on measurements which will directly implement the design effort or provide data in areas where analytic means are presently inadequate. The main areas requiring experimental study are (a) clean core reactivity, (b) fuel lifetime, (c) control rod effectiveness, programming and resulting power distributions, and (d) reactor safety. Analyses in conjunction with the experimental program will be required to insure that information

obtained from the experimental program will be interpreted adequately. A detailed discussion of tasks relating to each of these areas is presented.

a. Burnup Studies

The burnup study program begun for the 200-Mwe prototype will be continued in support of the 1000-Mwe reactor. Both the mass spectrometer and oscillator techniques will continue to be used. During this period fuel will be available from the SRE Core III. This fuel will be the first to have achieved high burnup (~ 30 to $40,000$ Mwd/MTU) in an equilibrium SGR spectrum. The distribution and quantity of fission products are expected to differ from high burnup capsules obtained from special irradiation facilities. Fuel with U^{235} enrichments of 6.5 to 15% will be available. This information will result in refinements of the analytic model, which will enable firm calculations of core lifetime to be made for the 1000-Mwe core.

Development Cost — Burnup Studies

	(in \$)		
	<u>FY 1967</u>	<u>FY 1968</u>	<u>FY 1969</u>
Operating funds	85,000	85,000	85,000
Equipment funds	<u>5,000</u>	<u>5,000</u>	<u>5,000</u>
	90,000	90,000	90,000

b. Doppler Coefficient

Measurements under the 200-Mwe program will have given values of the Doppler coefficient for UC slugs in both 1/E and SGR type spectrum. It will now be necessary to examine in detail the coefficient associated with fuel slugs arranged as an entire fuel element to investigate the Dancoff effect, which accounts for the interaction of the closely-spaced slugs within the fuel element.

The fuel element geometry may be investigated in several ways. Initially, only short sections (~ 6 in. or less) may be investigated. In order to eliminate end effects, it will be necessary to look at an entire fuel element. Several alternative techniques are possible here. The method used in the HTGR is to oscillate entire "cold" and "hot" fuel elements into the reactor. This method may present some problems, as the critical assembly fuel elements are contained in a process tube which also holds the Na coolant mockup. An alternative to this technique would be to oscillate single rods within the entire element.

Development Cost - Doppler Coefficient

	<u>(in \$)</u>	
	<u>FY 1967</u>	<u>FY 1968</u>
Operating funds	50,000	50,000
Equipment funds	5,000	5,000
	<u>55,000</u>	<u>55,000</u>

c. Critical Experiments

In addition to the types of measurements discussed for the support of the 200-Mwe plant, additional experiments will be performed for the 1000-Mwe SGR.

Two-region critical studies will be carried out to perfect techniques required to analyze various fuel element designs. Variable fuel-to-moderator ratio studies will be performed. Calculations from 1000-Mwe SGR optimization studies indicate a 20% gain in k_{eff} for a fuel element with 25,000 Mwd/MTU irradiation when the lattice pitch is increased from 9.25 to 14.6 in. These calculations will be checked by experiment.

Reactivity worth measurements, neutron flux distribution measurements, and intracell parameter measurements will be carried out in a small test region, requiring a minimum amount of fuel element and control rod fabrication, for the purpose of optimizing designs. This program will be supported by a design effort and theoretical calculations.

Additional fuel element and control rod designs will be studied. The following types of designs appear to be worthy of further experiments.

- 1) Fuel elements having a greater number of fuel pins and increased fuel area.
- 2) Central moderator in the fuel cluster.
- 3) Fixed sleeve control rods.
- 4) Possible use of burnable poisons.

Intracell and k_{eff} measurements will be performed, using UC fuel elements containing Pu and mock fission products, to supply information relating to a reactor core under equilibrium conditions. These measurements, along with burnup studies, will be used to check the calculations pertaining to core lifetime.

Development Costs — Critical Experiments

	(in \$)			
	<u>FY 1965</u>	<u>FY 1966</u>	<u>FY 1967</u>	<u>FY 1968</u>
Operating funds	145,000	20,000	55,000	50,000
Equipment funds	—	—	5,000	5,000
	<u>145,000</u>	<u>20,000</u>	<u>60,000</u>	<u>55,000</u>

d. Hot Exponential

In order to more accurately predict the operating characteristics of a reactor, at temperature, subcritical experiments at elevated temperatures will be carried out. To date, physics experiments in support of sodium-graphite reactors have been carried out only at room temperature and the results have been extrapolated to operating temperatures. This extrapolation leads to uncertainty in reactor design.

For sodium-graphite reactors, three temperature coefficients of reactivity are important. They are the temperature coefficients for the fuel, the sodium, and the graphite. The fuel and sodium effects are required for safety considerations, since they are prompt in nature. Even when the fuel Doppler coefficient is accurately known, the resonance capture in the lattice must be accurately known. Due to the spectral hardening, as the temperature increases, the graphite temperature coefficient is difficult to calculate. The combined effects of these temperature coefficients will be measured by determining material buckling values at various temperatures. Neutron flux distributions at high temperature will be measured. In order to provide information related to the above mentioned problems, the customary lattice parameters ($B_{\frac{2}{m}}$, L , ρ , f , δ^{28} , and ICR) will be measured at various temperatures in the hot exponential lattice.

The most economical approach to solving these problems is to perform subcritical measurements. Even though a hot critical program may be desirable in the future, in order to determine critical mass and small reactivity changes, the low cost hot exponential experiments should be carried out first. This approach will allow for the development of measuring techniques and will determine whether the additional and more accurate information available in a critical experiment is worth the additional expense.

Development Costs – Hot Exponential

	(in \$)		
	<u>FY 1965</u>	<u>FY 1966</u>	<u>FY 1967</u>
Operating funds	130,000	70,000	120,000
Equipment funds	<u>—</u>	<u>5,000</u>	<u>5,000</u>
	130,000	75,000	125,000

e. Fuel Fabrication

Approximately 700 kg of UC fuel will be required to support the experimental program. From this fuel, approximately 14 fuel elements will be fabricated. About one-half of the elements will be fabricated into various configurations, such as 24- and 30-rod clusters, to determine the optimum fuel element design for the 1000-Mwe reactor. The remaining elements will contain Pu and mock fission products, to support studies of SGR core equilibrium conditions.

Development Costs – Fuel Fabrication

	(in \$)	
	<u>FY 1966</u>	<u>FY 1967</u>
Operating funds	50,000	180,000
Equipment funds	<u>—</u>	<u>—</u>
	50,000	180,000

f. Reactor Physics Analysis

Continued analytical effort will be necessary to support the experimental program for the 1000-Mwe reactor concept. Extension of current methods of nuclear analysis and the development of new methods will be necessary to account for physical effects previously unevaluated, but probably important, to the design of the larger-sized long burnup core.

The burnup studies will be based on experimental results from SRE Core III fuel elements, and will add considerably to the actual operating data on reactivity, fission product buildup, and Pu concentrations available for SGR cores.

The hot exponential experiments will yield lattice parameter measurements in addition to bucklings. Flux distributions in the fuel rod cluster and

in the graphite will be calculated as a function of temperature for comparison with experiments. Also, conversion ratios and U^{238} to U^{235} fission ratios will be determined.

Evaluation of critical experiments performed during this program will lead to improved analytical treatment of the equilibrium and near end-of-life core. Previous critical experiments yielded information on beginning core life. With the long lifetimes expected from the SGR cores, the equilibrium cores with their fission product poisonings and Pu buildups play a more important role.

Fuel element and control rod design optimizations require the ability to predict flux distributions in complex, heavily-absorbing fuel elements and relative self-shielding effects of different control rod poisons with complex resonance structures. Effective two-group boundary conditions for different control rod materials in different thicknesses will be determined. Thermal neutron spectrum variations in a cell can be analyzed with the THERMOS code or with a multigroup transport code with up-scattering.

Development Cost – Reactor Physics Analysis

	(in \$)				
	<u>FY 1965</u>	<u>FY 1966</u>	<u>FY 1967</u>	<u>FY 1968</u>	<u>FY 1969</u>
Operating funds	30,000	30,000	40,000	40,000	40,000
Equipment funds	—	—	—	—	—
	<u>30,000</u>	<u>30,000</u>	<u>40,000</u>	<u>40,000</u>	<u>40,000</u>

3. Facility Operation

a. Sodium Reactor Experiment (SRE)

The SRE-PEP facility will provide the primary irradiation data for this development program. The purpose of the facility is to test large scale elements to 25,000 Mwd/MTU burnup at 2000°F peak central temperature within a two year period. The driver elements in this facility will operate at somewhat lower temperatures and will operate to lower burnup due to reactivity requirements. The SRE-PEP operating costs for FY 1968 and 1969 are included in the program costs below. Operating costs for FY 1965 to 1967 are included in the 200-Mwe prototype development program.

The Hallam and NRX facilities are being considered for testing large scale rods for the 1000-Mwe plants. The NRX program would be a continuation

of the prototype program tests currently in the final design stage. An evaluation of plant capability must be made to accommodate the higher heat output of these rods. The Hallam facility must be similarly evaluated for test capability. Initial conceptual designs use a multirod cluster containing some dummy rods. This element would then test rods under the planned conditions but the total heat output for the element would be within the allowable tolerance for individual elements. Local heat flux conditions would require analysis to insure safety and to minimize excessive stress conditions on core structural components. The operating cost for Hallam is not included.

Development Cost – Sodium Reactor Experiment

	(in \$)	
	<u>FY 1968</u>	<u>FY 1969</u>
Operating funds	2,500,000	2,500,000
Equipment funds	40,000	40,000
	<u>2,540,000</u>	<u>2,540,000</u>

b. Sodium Component Test Installation (SCTI)

This facility will be required for testing the supercritical modular steam generator for the 1000-Mwe plant. Since the steam-feedwater system of the SCTI is not designed for supercritical pressures, modifications in the following areas would be required.

- 1) Installation of new feedwater heat and feedwater piping to the steam generator.
- 2) Installation of new steam piping for 3500-psig operation with two steam pressure reducing stations to be introduced into existing steam header system.
- 3) Install new set of gears in boiler feedwater pump speed increaser.
- 4) Modular steam generator steam safety valves.
- 5) Pressure-temperature-flow instrumentation and control changer on steam and sodium systems.

- 6) Minor changes to steam side startup and shutdown systems.
- 7) Minor changes to main sodium piping to and from modular steam generator, plus modular support provisions.

The schedule for testing the supercritical module steam generator is dependant on AEC plans for steam generator development. Steam generator concepts currently in the design, construction or planning state are: (1) the AI subcritical module; (2) the ALCO subcritical unit; (3) the B and W subcritical unit; and (4) the AI supercritical modules. One possible sequence for testing is as follows.

- 1) AI subcritical module installation and test – June 1964 to September 1965.
- 2) ALCO subcritical unit installation and test – September 1965 to December 1966.
- 3) B and W subcritical unit installation and test – December 1966 to March 1968.
- 4) SCTI modifications for supercritical steam – March 1968 to September 1968.
- 5) AI supercritical module installation and test – September 1968 to December 1969.

Regardless of what test sequence is pursued, it appears prudent to start design and procurement of equipment for modification of the SCTI in FY 1965. Equipment would be on hand for immediate installation in the event premature failure occurs in one of the scheduled steam generators and other units were not available for testing. The SCTI would be modified to operate at either subcritical or supercritical steam conditions.

Based on the assumed test schedule, the SCTI would be shut down for modifications in FY 1968, and the supercritical steam generator would be tested in FY 1969 and 1970, which is compatible with the 1000-Mwe SGR construction schedule.

Development Cost — Sodium Component Test Installation

	(in \$)					
	<u>FY 1965</u>	<u>FY 1966</u>	<u>FY 1967</u>	<u>FY 1968</u>	<u>FY 1969</u>	<u>FY 1970</u>
Operating funds						
SCTI modification — design and procurement	110,000	—	—	—	—	—
SCTI modification — installation	—	—	—	100,000	100,000	—
Steam generator module — Installation	—	—	—	—	60,000	—
SCTI operation	—	—	—	120,000	750,000	425,000
	<u>110,000</u>	<u>—</u>	<u>—</u>	<u>220,000</u>	<u>910,000</u>	<u>425,000</u>
Equipment funds	10,000	—	—	25,000	10,000	5,000
	<u>120,000</u>	<u>—</u>	<u>—</u>	<u>245,000</u>	<u>920,000</u>	<u>430,000</u>

C. GENERAL DEVELOPMENT PROGRAM

This development program includes reactor safety, advanced component development, HNPF fuel surveillance, reactor operations analyses (ROAP), facility operation, and project coordination. Technical information gained from these projects is not only useful to the large SGR program but also supports other types of sodium cooled reactor systems.

1. Reactor Safety Program

The safety features of sodium graphite reactors must be well understood in order to take full advantage of the inherent safeguards of a low pressure reactor system. Since most of these features are related to the use of sodium as a coolant, a strong experimental safety program is necessary to confirm the expected behavior of sodium during and following accidents. Emphasis will be placed on measurements which are necessary for the design of the reactor system and which provide basic data necessary for the analysis of the safety behavior of sodium systems. The proposed program is an integrated series of tasks which will lead ultimately to an understanding of the causes which may lead to the initiation of accidents through the effects or consequences of malfunction of the reactor system. The results of this investigation will lead to the lowest cost sodium graphite reactor system at any site consistent with the highest standards of safety required for any reactor system.

There are two main areas requiring experimental study: (1) the kinetics of the reactor which could lead to failure of the core or primary system; and (2) the subsequent sequence of events which determine the dispersion of radioactivity to the environment. Although the Doppler and void coefficients of reactivity are essential ingredients in understanding the kinetic behavior of the sodium graphite reactor system, they have been included under the reactor physics program. One of the most important, if not the most important, accident affecting the safety of sodium graphite reactor systems, the dropping of a fuel element on the floor and subsequent burning or oxidation, is not included here because it is one of the ingredients of the fuel element safety program currently under investigation at ORNL. It is an essential part of the sodium graphite reactor safety program.

The kinetics tasks in this reactor safety program are designed to lead to a complete understanding of the ultimate shutdown mechanism in sodium cooled,

ceramic fueled graphite moderated reactor systems and to establish that core meltdown will not lead to rapid and violent failure of the primary reactor system. Experiments involving fuel element transient heating are desirable to establish the ultimate shutdown mechanism, as well as experiments to determine that no boiling effects during reactor accidents can lead to the generation of shock waves which might rupture the primary system.

Fission product retention and removal mechanisms and techniques are necessary for the ultimate exploitation of any sodium cooled reactor concept. Experiments will establish the design requirements for construction of primary system containment, a fundamental safety feature of sodium reactor systems. The data obtained in these experiments would also be necessary for the accurate analysis of the safety of the system. Also important for the design of sodium reactors is the rate of energy release of burning sodium, the self-extinguishing characteristics of large full scale fires and the release of fission products from those fires. Since it is anticipated that natural and inherent airborne depletion mechanisms will result in reducing the availability of sodium and any attendant fission products, and in addition reducing the requirements of an air cleaning system, it is desirable to study atmospheric decontamination following sodium fires. The details of the individual tasks for the complete safety program are outlined in the following pages.

a. Fuel Element Failure Mechanisms

The ultimate shutdown mechanism for a sodium cooled reactor during a severe uncontrolled, power excursion is postulated to be meltdown of the core into a subcritical geometry. The mechanisms associated with the entire spectrum of events from the initiation of melting within a portion of the fuel to complete meltdown and gross movement require general, basic understanding. The task of defining and understanding the behavior of fuel elements during transient heating failure is formidable. This is because a large number of independent variables may play significant roles in fuel failure. In addition, possible interaction of several variables increases the complexity of the problem.

By means of small scale experiments this task will examine parameters which govern fuel element failure and develop analytical models which will be tested for extrapolation in engineering scale TREAT experiments. The small experiments will be conducted in the KEWB reactor facility in instrumented,

relatively inexpensive inpile capsules containing small fuel specimens which can experience varying degrees of failure as a function of initial parameters when exposed to short-period neutron bursts.

One of the major objectives of this task is the investigation of failure characteristics of uranium carbide fuel specimens as a function of material and coolant parameters and initial system conditions of coolant temperature and pressure. The material parameters of interest are fuel composition (carbon content), bonding, cladding material, cladding defects, cladding thickness, and preirradiation history. The coolant parameters include absence of coolant-inert atmosphere, stagnant sodium, flowing sodium, and sodium vapor. Other specific objectives include investigation of the migration and/or redistribution of fuel and changes in metallurgical character of the fuel following transient heating up to the condition of gross failure; characteristics of fuel movement, dispersion and reassembly for transient heating conditions involving gross melt-down and failure; and dynamic system behavior for various degrees of transient heating fuel failure such as pressure and volume changes.

The establishment of correlations between these results and the TREAT test results will enhance the future evaluations of fuel safety aspects. That is, possible fuel composition and design modifications could be rapidly screened or compared by small scale tests in advance of larger scale testing. The techniques developed under this project will be equally applicable to studies with other fuel materials.

Small-scale experiments will be conducted using capsules of about 1-1/2- to 2-in. -diameter containing a fuel pin of about 1/4 in. diameter or less and a few inches in length, prepared with clad or bond conditions as required. The capsules will have provisions for filling with sodium, inert gas or vacuum. Instrumentation will include fuel, clad and coolant thermocouples, a pressure transducer, and a linear displacement transducer. The capsules will also be provided with a preheating system capable of heating the capsule to temperatures as high as 1500°F.

A survey of potential fuel failure mechanisms and related parameters will be conducted to establish the detailed requirements of the experimental method and an analytical transient heat transfer model required for the evaluation

of experimental results will be prepared. Transient instrumentation techniques required for the experimental program will be evaluated and developed for high-temperature sodium conditions.

Development Costs — Fuel Element Failure Mechanism

	(In \$)				
	<u>FY 1965</u>	<u>FY 1966</u>	<u>FY 1967</u>	<u>FY 1968</u>	<u>FY 1969</u>
Operating funds	95,000	145,000	145,000	95,000	48,000
Equipment funds	<u>5,000</u>	<u>5,000</u>	<u>5,000</u>	<u>5,000</u>	<u>2,000</u>
	100,000	150,000	150,000	100,000	50,000

b. Fuel Element Transient Heating

The principal long-range objective of this task is to experimentally examine the entire spectrum of events which occur during various conditions of nuclear transient heating of typical SGR fuel element configurations, and from such studies develop the foundation upon which to base the understanding of SGR core behavior during power transients. This information is also required in order to define the operating limitations of the fuel element design.

A major objective of this task is to experimentally determine the transient characteristics of SGR fuel element configurations. An analytical model, which most nearly represents the nuclear-thermal-mechanical features of the meltdown of a uranium carbide fuel element, will be developed from the experimental information procured in the process of the TREAT experiments. This analytical model will permit the prediction of the ultimate shutdown mechanism of the SGR.

Studies of the behavior of uranium carbide fuel rods under transient irradiation in the TREAT reactor will be directed toward determination of the extent of cladding deformation or failure for transient thermal conditions in which no melting occurs in the fuel. In addition, the project will evaluate metallurgical and gross structural changes (fuel expansion cracking) of UC associated with nonmelting power transients; investigate the magnitude of transient pressures which may be developed within the fuel rod for conditions up to and including fuel melting; examine changes in density and character of a particular uranium carbide fuel specification as a result of partial melting and re-solidification; and determine gross structural changes in fuel rod assembly

during power transients in which various degrees of fuel melting occurs. The experiments will be carried out in capsules inserted in the TREAT facility. These capsules will include uranium carbide pins with appropriate cladding and bonding material to simulate actual fuel pin configurations. The fuel pins will be immersed in sodium and the entire assembly will be instrumented to obtain transient temperature and pressure data. The capsules will be irradiated in TREAT under varying conditions of energy release, temperature, and geometry. Postirradiation examination will include visual examination, determination of dimensional changes in cladding and fuel. Metallurgical tests of cladding and fuel, check on any redistribution of fuel, gamma and neutron radiography, and radiochemical analysis.

Development Costs — Fuel Element Transient Heating

	(In \$)				
	<u>FY 1965</u>	<u>FY 1966</u>	<u>FY 1967</u>	<u>FY 1968</u>	<u>FY 1969</u>
Operating funds	150,000	147,000	195,000	147,000	50,000
Equipment funds	-	3,000	5,000	3,000	-
	<u>150,000</u>	<u>150,000</u>	<u>200,000</u>	<u>150,000</u>	<u>50,000</u>

c. Transient Heat Transfer in Sodium

The purpose of this task is to investigate and define for use in safety studies the transient heat transfer characteristics of liquid sodium. The experimental information will be obtained over a range of conditions which will allow application of results to any sodium cooled reactor using analytical methods developed under studies of boiling sodium. Transient boiling of sodium at the fuel element surface as a result of sudden power surges may be an important reactivity factor under conditions approaching and following fuel melting. The formation of voids in the sodium coolant prior to fuel melting introduce positive reactivity feedbacks in large, ceramic-fueled reactors. In thermal systems the reactivity is due to competing changes in thermal utilization and streaming leakage; in fast systems the reactivity is due to spectrum changes which reduces the stored Doppler poison and negative prompt Doppler coefficient and increases the average number of neutrons per fission. Various regimes of transient boiling are likely to be encountered, particularly in the case of the high melting point ceramic fuels, e.g. UO_2 and UC. The conditions leading to possible generation of dynamic pressures during transient boiling must be investigated.

The investigation and subsequent understanding of the mechanisms associated with the transient thermal behavior of sodium will enhance the area of general sodium cooled reactor dynamic analyses, and in addition provide a strong foundation for planning, analysis, and evaluation of engineering scale experiments and integral reactor tests. The mechanisms and parameters to be investigated under this task include; the effect of coolant flow on transient conduction as a function of transient heating rate and coolant velocity, the effect of transient heating rate and subcooling on initiation of transient boiling, void formation rates in transient boiling of sodium, the onset of stable film boiling in sodium during transient heating, initiation of boiling at a liquid metal-liquid interface, determination of the significance of radiation heat transfer as a transient heat transfer mechanism in sodium, and investigation of the existence of dynamic pressure generation during transient boiling in sodium.

A thorough analytical evaluation of the various transient heat transfer mechanisms in sodium will be conducted utilizing the experimental results and analytical models available from Reactor Safety Program studies of reactor transients as a guide. This evaluation will permit an assessment of those areas requiring emphasis, and in addition will determine the parameters of interest and their ranges.

The various experimental methods required in the overall study will be determined on the basis of parameter limitations with respect to the various objectives. The various types of experiments to be used will include; inpile capsule experiments for very short period transient heating in the KEWB reactor facility, short period out-of-pile transient heating experiments with and without flow using recently developed pulse heating techniques such as electron beam heating, and modified inpile capsule experiments with flowing coolant for convective effects at short periods if justified on the basis of results from out-of-pile loop experiments.

A research for suitable transient instrumentation techniques for high temperature sodium service will be conducted. The major item here is the pressure transducer. Surface temperature measurement with bare junction thermocouples will require techniques for electrical insulation of the wires from the sodium.

Inpile capsule transient heating experiments with sodium will be conducted to compare results with analytical models developed for other fluids. Verification of analytical conclusions on significance of radiative heat transfer will be attempted.

Analysis and evaluation of inpile experimental results will be performed to establish some parameter effects for transient sodium heat transfer and to determine analytical models for the extension of those for ordinary liquids which are most appropriate.

Development Costs – Transient Heat Transfer in Sodium

	(In \$)			
	<u>FY 1965</u>	<u>FY 1966</u>	<u>FY 1967</u>	<u>FY 1968</u>
Operating funds	95,000	145,000	97,000	50,000
Equipment funds	<u>5,000</u>	<u>5,000</u>	<u>3,000</u>	<u>-</u>
	100,000	150,000	100,000	50,000

d. Boiling Effects During Sodium Reactor Accidents

The overall objective of this task is to generate reliable experimental information for use in predicting coolant and fuel behavior during sodium cooled reactor excursions which are severe enough to result in sodium boiling in the core. This information is essential to the safety evaluation of sodium cooled reactors because of the key roles that sodium boiling and two-phase hydrodynamics play in the sequence of events leading to fuel melting and ultimate shut-down of the reactor. This task, together with information obtained on the transient heat-transfer in sodium, will permit the development of analytical models which represent the onset of sodium boiling in the reactor system.

Specific objectives are to measure steady-state burnout heat flux and boiling heat transfer coefficient as a function of pressure, subcooling, and local quality; measure steady-state pressure drop and void fraction in sodium two-phase-flow as a function of mass flow and power; study the hydraulic stability and dynamics of sodium two-phase flow; investigate heat transfer to sodium under conditions of unstable two-phase flow; measure transient pressures, generated by the introduction of various quantities of saturated sodium vapor into a volume of subcooled liquid sodium; and measure heat transfer rates and temperature histories during very high intensity heating of small simulated fuel slugs in a sodium environment.

Specific objectives in the program for development of analytical methods to complement the experimental work include development of analytical models and theories which adequately predict both the steady-state and transient results of all sodium boiling experiments and development of a digital computer program which incorporates these analytical methods into a general reactor model. Such a program will calculate voids, flow rate, heat transfer, and fuel temperatures during pressure or power transients.

Experimental measurements will be made to obtain information in the areas of forced convection, burnout heat flux, subcooled boiling heat transfer, chugging and choked flows, heat transfer under unstable flow conditions, and bubble collapse shock waves. Experiments will be conducted in a boiling sodium loop with various simulated fuel elements. The loop will be instrumented for pressure, temperature, flow rate, and possibly void measurements.

Specific experiments will be carried out in a range of parameters which will permit the generation of an analytical model and will permit the determination of the ultimate shutdown mechanism of the reactor.

<u>Development Costs – Boiling Effects During Sodium Reactor Accidents</u>	<u>(In \$)</u>			
	<u>FY 1965</u>	<u>FY 1966</u>	<u>FY 1967</u>	<u>FY 1968</u>
Operating funds	150,000	147,000	98,000	50,000
Equipment funds	-	3,000	2,000	-
	150,000	150,000	100,000	50,000

e. Reactor Kinetics

The objective of this task is to investigate the general area of reactor dynamics in order to seek out potential difficulties concerning the safety of reactor systems. Theories will be formulated where none exist and, with present theories, tested or verified by experiments or application of previous measurements where feasible. This work will not be limited to specific goals but will look toward potential problems to provide direction for the work.

It is also important to have realistic models of the dynamic behavior of reactor systems in order to understand their basic safety. It is the intent of this task to develop models which can lead to an understanding of the dynamic behavior of reactors. Computer programs will be written to provide tools for use of these models and to allow the theories developed to be used for hazard

analysis by engineering personnel. Current practices in analyzing the safety of reactors will be scrutinized and improved where possible.

Theoretical study will be initiated with the object of showing that the kinetic behavior of multiplying and nonmultiplying media is contained in the dispersion relation (the space-time Fourier transform of the spatial reactor equation). This work will include the planning of experiments to test the theory.

The multi-channel reactor transient simulation model with moving coolant segments (OSCAI) integrates reactor kinetics with heat transfer and fluid dynamics. The model incorporates a differential solution of the reactor kinetics equations which will permit taking incremental time steps consistent with the time involved for sensible changes in thermal conditions rather than being limited by consideration of the prompt neutron lifetime. Simulation of multiple channels and moving coolant segments will permit investigation of nonuniformities between various reactor regions and their resulting interrelationships. As examples: temperature and flow distributions can be established; transient temperature and flow can be analyzed; and fuel melting and channel choking can be simulated. Simulation of external heat transfer, control and safety systems will be incorporated with OSCAI and AIREK III models to provide an engineering tool for accident analyses required for hazards evaluation. In addition, it will be possible to study the dynamic behavior of any reactor system in order to establish control and safety system requirements.

The multi-channel reactor transient simulation program (OSCAI) will be improved by incorporation of a 10-channel reactor model with 10 axial fuel segments per channel, 20 axial coolant segments per channel at base density condition, 10 axial power factors, a separate designed reactivity coefficient for each channel, quadratic formulations for imposed inlet velocity and temperature changes and imposed reactivity changes, the capability of making calculations at constant power without reactivity feedbacks and with reactor kinetics bypassed when desired, and representation of boiling coolant which will provide a linear volumetric expansion with heat absorption during vapor formation at any boiling temperature. Development of all of the digital kinetics codes under this task will draw heavily on experimental data produced in other tasks. The integrated kinetics model will include appropriate coefficients determined over

a wide range of parameters in the experimental work of the first four tasks. By this means the model will be updated and will represent the latest available experimental information.

Development Costs — Reactor Kinetics

	(In \$)				
	<u>FY 1965</u>	<u>FY 1966</u>	<u>FY 1967</u>	<u>FY 1968</u>	<u>FY 1969</u>
Operating funds	145,000	145,000	97,000	50,000	50,000
Equipment funds	<u>5,000</u>	<u>5,000</u>	<u>3,000</u>	<u>-</u>	<u>-</u>
	150,000	150,000	100,000	50,000	50,000

f. Fission Product Retention in Sodium

The efficiency with which fission products are trapped by a reactor coolant is a function of the properties of the coolant and the particular fission product under consideration. The purpose of this task is to determine the ability of sodium to retain specific fission products released to the coolant stream as a result of a fuel-cladding failure or fuel meltdown. If it can be shown that the most radiologically damaging fission products are retained in the coolant stream following a fuel meltdown, then the problems associated with containment of the sodium cooled reactor are considerably improved. Evidence produced to date indicates that certain fission products are indeed selectively retained by sodium. These fission products include iodine, barium-lanthanum, cerium, and possibly strontium. Much has already been learned concerning the ability of sodium to retain iodine.

Existing information will be extended to obtain information on the retention characteristics of the more radiologically damaging isotopes. Information sought will pertain to a determination of those parameters most significant in effecting the retention characteristics of sodium. Plating effects, temperature effects, geometric effects, thermodynamic effects, and other controlling parameters will be studied to obtain a basic understanding of the retention characteristics of sodium.

Both the inpile and out-of-pile techniques will be used. The KEWB reactor will be utilized in conjunction with the STF reactor to provide an appropriate means of injecting fission products into high temperature sodium under varying parametric conditions. In addition, varying electrical heating methods

will be used. Comparisons will be made of the experimental data obtained with simulated analytical models of similar thermal and geometric conditions in an actual reactor situation. Retention of fission products in sodium following release from long-term burnup fuel will be studied by capsule experiments in the SRE or by examination of existing Hallam fuel elements.

Development Costs — Fission Product Retention in Sodium

	(In \$)		
	<u>FY 1965</u>	<u>FY 1966</u>	<u>FY 1967</u>
Operating funds	100,000	97,000	50,000
Equipment funds	-	3,000	-
	<u>100,000</u>	<u>100,000</u>	<u>50,000</u>

g. Contamination Removal Techniques

The objective of this task is to develop techniques for removing harmful impurities from sodium reactor coolant. Contaminants to be considered in this phase include carbon and fission products.

The first objective is to develop techniques to reduce the carbon concentration in sodium to a safe level for high temperature reactor operation. Contamination of sodium reactor coolant with carbon and subsequent carburization of stainless steel has serious adverse effects on reactor components during high temperature operation. Failure of fuel cladding or other structural material may occur due to reduction in mechanical properties by carburization. The consequence may be shutdown of the reactor to make repairs, or possible reduction in reactor performance level to avoid overstressing structural material. Preliminary calculations indicate that the carbon content of "as received" sodium poses a potential carburization problem for reactor operation. Also, introduction of carbon into the sodium coolant after reactor startup can result from rupture of moderator clad or inleakage of foreign matter. To combat these problems a carbon removal technique is necessary.

Sodium manufacturing processes will be studied to determine sources and concentration levels of carbon contamination. In cooperation with suppliers, an effort will be made to eliminate or reduce the introduction of carbon during sodium production.

A series of experiments will be conducted to determine whether the carbon contaminant in sodium is in solution, particulate form, or colloidal form. This information is needed to develop the most effective carbon removal equipment. Cold-trapping or filtration may be further investigated depending upon the results. This will be done for both "as received" reactor grade sodium and for sodium contaminated by dissolution of moderator grade graphite.

Existing carbon in sodium solubility data will be extended by dissolving carbon at various temperatures and oxide concentrations in sodium previously purified by distillation or hot-trapping. The improved analytical technique will provide better solubility data than now available. Dissolution rate will be measured in terms of the variables of temperature, velocity, free stream carbon concentration, and shape and form of graphite specimens. The solubility and dissolution data will provide a basis for estimating the rate at which carbon will transfer into the sodium stream in case of a moderator cladding rupture.

Although hot-trapping has been successfully demonstrated as a carbon removal technique, its application in large sodium systems may be uneconomical. Therefore, an economic study will be made to compare hot-trapping with other possible carbon removal techniques. Possible means of reducing hot-trap fabrication costs will be studied.

A secondary objective is to develop techniques to maintain the fission product level in the sodium at a safe level. Current plans include the operation of the SGR with some fraction of the fuel elements having cladding defects. Such cladding defects will permit the accumulation of fission products in the coolant stream. If allowed to accumulate in an unattended fashion, the fission product activity over a period of years would make maintenance of the equipment quite hazardous and indeed could influence siting criterion. Therefore, methods for reducing fission product activity to a given level are required.

The removal of fission products will be investigated experimentally in small bench-scale apparatus specifically designed for the tests. The tests will be conducted with sodium reactor coolant which contains fission products or with synthetic contaminated coolant prepared by exposing sodium to unclad fuel which is irradiated. Promising removal devices and techniques to be tested include precipitation and coprecipitation. When sodium is cooled, separation of fission product impurities may occur because of favorable solubility relationships

or coprecipitation phenomena. The precipitated or coprecipitated fission products can then be removed from the sodium by filtration, centrifugation, or other means. Fission products precipitate when the amounts present exceed the solubility limits. In coprecipitation a substance precipitates from a solution, and carries with it trace impurities which are also present in the liquid. While the precipitation concept assumes that fission products have a changing temperature-solubility relationship, the coprecipitation mechanism does not require such a relationship. Coprecipitation has been well demonstrated in aqueous systems and has been applied in the purification of water. Possible coprecipitation in sodium was indicated by experience at the SRE which showed greater accumulation of radioactivity in cold traps than in other portions of the reactor system. The carrier precipitate may be formed from other impurities present in the sodium (e.g., sodium oxide or carbon) or from components added specifically for this purpose. At this early stage, a semiempirical approach based on precipitation and coprecipitation will be taken. Alternate fission product removal techniques to be considered include absorption, adsorption, and filtration. Data will be used where possible to assist in the evaluation of fission product removal techniques.

Development Costs — Contamination Removal Techniques

	(In \$)		
	<u>FY 1965</u>	<u>FY 1966</u>	<u>FY 1967</u>
Operating funds	175,000	95,000	50,000
Equipment funds	-	5,000	-
	<u>175,000</u>	<u>100,000</u>	<u>50,000</u>

h. Energy Release and Absorption Following Component Failure

Most designs for fast and thermal sodium cooled reactors include a primary sodium piping system and intermediate heat exchanger, a secondary sodium system, and a boiler and water-steam tertiary system. Energy-release accidents in such systems can be categorized into two general classifications. They are: (a) failure of a component in the primary system, resulting in a release of coolant and possible radioactive materials directly to the building or containment cell. The escaping coolant transfers energy (and in some cases reacts and creates additional energy) with the surrounding atmosphere and building or cell walls; and (b) failure of a boiler-tube and intermixing of

high-pressure water and steam with sodium. The resulting sodium-water reaction creates pressure which is relieved through venting devices, or a secondary sodium system component fails. If the secondary sodium system component failure is the intermediate heat exchanger, there is the possibility of the release of primary system sodium and radioactive materials.

Objectives of this task are: (1) to develop basic data which will permit the construction of an analytical model which accurately represents the energy release and transfer characteristics of the sodium cooled reactor system under various parametric conditions, (2) to perform large cellular experiments so that a safe design of the primary system cell containment can be effected.

Initial work will consist of literature surveys and bench experiments in conjunction with analytical models to better define basic energy release characteristics (magnitude and rate of pressure and temperature rise of heat sinks) as a function of various parameters. A complete parametric survey is not required but rather the extremes and the required intermediate information on key parameters are needed to define energy release rate characteristics for formulation of a general analytical model. Information developed by other contractors and data from programs currently under way, pertaining to energy release from sodium-water interactions, will be fully utilized to assess sodium pressure temperature parameters as they effect primary system cell containment.

A test installation which will mockup a primary equipment cell will be fabricated to study energy release mechanisms. It is probable that the test installation will take the form of modular test cells with various sized injection nozzles and vent nozzles. A heating and pressurizing source and storage tanks will provide for coolant injection control. Mixing tanks or nozzles will permit coolant mixing and energy transfer experiments. Considerable effort will be extended in the equipment design phase to assure that data obtained is reliable and extends over an appropriate range of parameters.

Information essential to demonstrate the safety of the cells under coolant spill conditions will be studied. This will include the development of a automatic mechanical hatch closing device. Under certain conditions of operation, it is possible that manholes can be opened to the cells normally maintained under an inert atmosphere. If a coolant spill should occur during the time when

a hatch is opened, it is presently planned that an automatic device will close and limit the extent of any fire. The safety of this hypothesis will be tested in the experiments.

Development Costs — Energy Release and Absorption Following Component Failure

(In \$)

	<u>FY 1965</u>	<u>FY 1966</u>	<u>FY 1967</u>	<u>FY 1968</u>	<u>FY 1969</u>
Operating funds	140,000	190,000	245,000	195,000	75,000
Equipment funds	<u>10,000</u>	<u>10,000</u>	<u>5,000</u>	<u>5,000</u>	<u>-</u>
	150,000	200,000	250,000	200,000	75,000

i. Fission Product Release from Sodium Fires

In the analysis of radiological hazards associated with sodium cooled reactors, a primary mechanism of potential radioactivity dispersion is usually postulated to be through sodium spillage and a subsequent fire. If the sodium is assumed to contain significant fission-product contamination the principal radiological hazard could depend on the magnitude of the airborne release of certain isotopes (I^{131} , SR^{89} , SR^{90} , and Cs^{137}) by the burning sodium. Recent experimental evidence has indicated that a large fraction of such activity would be retained in the solid residue left by burning sodium. The objectives of this task are to quantitatively determine the release fractions and transport of several fission-product elements of prime radiological significance from contaminated sodium fires.

One of the specific goals of this task is to determine the effect of geometry and ambient conditions on sodium burning rate and surface crust formation, measure the release of stable tracer elements simulating fission products from intermediate size fires, measure the release of radioactive tracers from intermediate size fires, perform basic chemical studies of the interaction of fission products in the surface crust, and study the release and transport of fission products and sodium oxide smoke from a very large fire.

A second specific goal of this task is to determine the characteristics of a large fire in a gallery of the size for the 200-Mwe prototype. Information desired pertains to the feasibility of using a cell with an automatic closing device to limit the extent of the combustion of released sodium. This task will

correlate effort with that of the preceding task to obtain information necessary to safely design the primary system cell.

Sodium fires will be carried out in varying sizes and under different geometric conditions in equipment presently available. This equipment permits control of the air flow rate over the surface of the fire and permits measurement of the quantity of effluent oxide from the fire. Measurements will be made of those parameters which are important to the release of fission products. In particular, the self-extinguishing characteristics of the fire will be studied. These characteristics have been observed in previous experimental work and appear to be due to the formation of a large sodium oxide crust on the surface of the unburned sodium.

Data will be developed which will permit the prediction of the quantity of sodium burned prior to auto-extinction and the fraction of fission products released from the burned sodium.

Development Costs — Fission Product Release from Sodium Fires

	(In \$)		
	<u>FY 1965</u>	<u>FY 1966</u>	<u>FY 1967</u>
Operating funds	96,000	120,000	75,000
Equipment funds	<u>4,000</u>	<u>5,000</u>	<u>-</u>
	100,000	125,000	75,000

j. Atmospheric Decontamination in Reactor Accidents

Studies conducted at Atomics International of various types of reactor buildings and of alternative hypothetical sites have demonstrated that an important criterion in establishing siting distances for sodium-cooled reactors is the Na^{24} radioactivity released during a fire. Of particular importance is the immersion and direct dose from Na^{24} leaking from the building. Additionally, the direct dose from Na^{24} within the building is important for certain arrangements and sites. In the event of a large scale sodium-air fire, the possibility exists that radioactive oxides of sodium may be released within the building structure. These may occur in such locations as a heat exchanger cell, a pipe gallery, or in an air-filled reactor building.

If industrial type buildings are used to house the reactor, such as the Hallam Nuclear Power Facility (HNPF), the integrity of the building and thus

the environmental release may depend on the ability of the air-cleaning system to remove large quantities of sodium oxide particles and air without clogging.

The primary objective of this task is to study the production, distribution, and availability for dispersion to the environment of sodium oxide particles produced in large sodium fires. To most effectively carry out this primary objective, areas requiring study are: to study particle production mechanisms, size distributions, velocities, etc.; to determine natural agglomeration effects, fallout, deposition, and other removal phenomenon in order to determine the remaining particulate suspension as a source term for environmental dispersion and loading requirements for an air-cleaning system; and to determine possible artificial means of enhancing removal by increasing particle growth, fallout, and deposition.

A secondary objective of this task is to determine the most effective air-cleaning arrangement for removing sodium oxide particles which remain suspended in the air stream.

Bench scale experiments will be performed in which small quantities of sodium are oxidized completely in bombs, bell jars, or quartz tubes. These oxidation studies will be made in controlled N_2 and O_2 atmospheres. Techniques will be developed and equipment purchased for measuring the particle size distribution of the highly deliquescent sodium monoxide and sodium peroxide. These experiments will be used to develop simple sizing techniques for future tests. The effects on particle size distributions will be observed due to variable oxygen concentrations, water vapor pressure, and burning rates. Liquid sodium temperature will also be varied.

The particles collected in dry atmospheres will be used in conjunction with settling chambers to determine the absolute concentrations of particles which remain airborne. Minimum particle sizes will be generated when the sodium is burned in inert, dry atmosphere and thus, minimizing the effects of particle agglomeration. In these experiments, it is expected that a maximum fraction of airborne particles will result. Such experiments will establish an upper limit to the airborne components. Agglomeration and fallout of oxide particles under large scale fire conditions will be studied. The major parameters involved will be moisture content, air flow rate and the vectorial distance from the source.

Large scale experiments on sodium fires and particulate distributions in air, on walls, and on ducting will be conducted. Experimental studies of devices that enhance the agglomeration and fallout of sodium oxide particles will be performed. Studies will be made of the manner in which the fission products are distributed in space and time; the relationships of such distributions to those of sodium oxides will be determined. A test installation will be designed and built for measuring the efficiency of various air cleaning systems for sodium oxide particles. Surveys will be conducted in order to select the most efficient and economic systems. The parameters which will be controlled are: air flow rates, sodium oxide feed rates, sodium oxide particle distribution, and efficiency for trapping fission products.

Building design criteria will be established in order to determine the minimal leak rate capability of conventional materials and the economics of low-leakage construction. It has been demonstrated through component and model leakage studies on concrete and insulated metal panel buildings that low-leakage construction can be obtained by using carefully controlled and improved construction techniques. These studies have shown that estimates of the leakage of a full size structure can be predicted from the data obtained from laboratory measurements, using test cells.

Specific goals are directed towards clarifying existing questions concerning the economic comparison of improved low-leakage, low-pressure, conventional structures with absolute containment vessels and to establish the permanency and reliability of the caulking and sealing compounds which have been recommended for reducing the leak rate of conventional structures. The present experimental program to satisfy these objectives and goals will be completed. A practical design manual will be published.

Development Costs — Atmospheric
Decontamination in Reactor Accidents

	(In \$)		
	<u>FY 1965</u>	<u>FY 1966</u>	<u>FY 1967</u>
Operating funds	145,000	120,000	50,000
Equipment funds	<u>10,000</u>	<u>5,000</u>	<u>-</u>
	155,000	125,000	50,000

2. Advanced Development

a. Effects of Heat Transfer on Steam Generator Corrosion

Previous experience with liquid metal heated steam generator evaporators has shown unexpected damage on the water side of the tubes due to corrosion at the hot end of the tubes or other areas of high heat flux. These effects are not normally encountered in fossil fueled steam generators since the heat flux in these units is limited to prevent tube burnout on the combustion gas side. However, instances of on-load pitting have been reported although the cause is not yet known. Sodium heated evaporators do not have the limitation imposed by hot gases and may operate at peak heat fluxes, an order of magnitude higher than those of fossil fueled boilers.

Previous investigations have indicated that the effect of heat transfer rates on corrosion could be a potential limitation on the allowable heat fluxes in sodium heated evaporators. Several corrosion mechanisms exist in which excessive heat fluxes could accelerate the rate of corrosion.

Since the cost of heat transfer area in sodium heated steam generators is relatively high, the tendency is to utilize the maximum heat flux allowed for economic reasons. The occurrence of corrosion would promote tube leaks, and become an important safety consideration. The major objectives of this project therefore, are: to determine whether corrosive effects will impose limitations in sodium heated steam generator design, determine the practical limitations on heat flux as demonstrated by experience to date, determine the experimental work required to provide data in areas not covered by previous experience, and obtain actual experimental corrosion data on the effect of heat flux and thermogalvanic effects on heater tube materials for planned sodium cooled reactor operating conditions.

This program is required to provide design limitations essential for the efficient and safe design of future sodium heated steam generators.

Work on this project during FY 1963 was limited primarily to literature surveys and theoretical studies on the causes and mechanisms of corrosion. A contract was negotiated with Battelle Memorial Institute of Columbus, Ohio, for a literature search and theoretical investigation of the mechanisms whereby high heat fluxes could cause damage to steam generator tubes.

Several potential mechanisms were suggested in the literature including effect of increasing the concentration of dissolved salts in the water at the heat transfer surface, thermal stresses in high heat flux areas causing separation of the protective oxide film from the metal surface, and thermogalvanic corrosion due to localized overheating. Quantitative data, however, on the effect of heat flux on corrosion as applied to the specific problem are not available and will need to be obtained by directed experimental work under controlled conditions.

Work on this project was terminated at the end of FY 1963. Continuation of the program is proposed for FY 1965. Literature surveys will be continued and contact will be maintained with others working in the field of corrosion in steam generators. The work of others will be carefully studied and the results, if applicable, will be related to this project.

An experimental effort will be initiated to determine the effect of heat flux on the corrosion of heater tubes under controlled conditions of heat transfer rates and water chemistry. Heat fluxes of 1, 2, and 5×10^5 Btu/hr-ft² are planned for Cr-Mo tubes with constant surface temperatures.

An experimental program will be initiated to determine the effect of thermogalvanic corrosion on Cr-Mo and austenitic steels under conditions of controlled temperature differences and water chemistry expected in future SCR design. It is planned that this phase of the experimental work would be accomplished on a research subcontract to an independent laboratory under the supervision of this project.

Development Costs — Effects of Heat Transfer
on Steam Generator Corrosion

	(In \$)	
	FY 1965	FY 1966
Operating funds	98,000	98,000
Equipment funds	2,000	2,000
	100,000	100,000

b. Dispersed Fuel Experimental Study

This effort is directed toward the development of advanced fuels, which will provide improved thermal performance and burnup and reprocessing characteristics. The specific objectives of the project are to determine the

apparent thermal conductivity of various fuel-liquid phase-container combinations at various fuel/liquid metal ratios, perform engineering studies on selected fuel concepts and establish burnup limits, and to study the rate of fission gas release and fission product transport.

Metals or ceramics will be studied for liquid-bonded dispersed fuel elements suitable for either an enclosed static fuel system or a mobile fuel system which permits continuous or discontinuous fuel movement through the core at very low rates. These fuel elements should be capable of operating at a container surface temperature of 1200°F to burnups in excess of 40,000 Mwd/MTU. The liquid bonds considered include Na, Pb, and Sn. The fuels include UC, UO₂, and U-16 Mo.

The static capsule concept holds promise as a high heat flux-high burnup-high temperature fuel system. The fuel swelling problem in such a system is reduced and the conductivity of the solid-liquid mixture may be greatly enhanced over that of solid fuel.

The dynamic concept with continuous fuel changing would eliminate the need to shut down for refueling and simplify reactor control because of the continuous reactivity adjustment. Furthermore, this concept permits continuous fission gas removal. For the dynamic application, UO₂-sodium-austenitic stainless steel systems appear feasible. UC-sodium-austenitic steel systems appear practical so long as the carbon content of the UC is retained at or below stoichiometric composition; however, confirming experiments are required.

One of the possible problem areas in this concept is the potential growth and sintering of the fuel particle. The tendency for growth or sintering must be determined, and if significant, it will be necessary to evaluate the effect on conductivity, irradiation stability, fuel mobility, etc.

The rate of fission gas release and transport is another area that requires study since there is a decided advantage in continuous fission gas removal. Continuous fission gas extraction would permit higher burnup.

The detailed study of characteristics of promising combinations of container materials, UC or UO₂ particles of various sizes, and liquid metal bonds under environmental conditions will be initiated. Properties to be studied in addition to compatibility will include but not be limited to density, pressure

drop, thermal conductivity, fission gas nucleation, and adherence at fuel surfaces. Preliminary to any experiments, extensive analytical effort will be conducted to identify areas of most promise and to establish a sound basis for experimental design.

Engineering studies will be initiated to establish preferred geometries, types of inpile irradiation measurements to be included, and methods of evaluating the performance of the dispersion fuel.

Several capsule irradiation tests will be performed to determine the effect of particle size on the quantity of gas released, particle disintegration, gas migration, sintering, and heat transfer.

Development Costs — Dispersed Fuel Experimental Study

	(In \$)		
	FY 1965	FY 1966	FY 1967
Operating funds	98,000	150,000	200,000
Equipment funds	2,000	10,000	10,000
	100,000	160,000	210,000

c. Development and Evaluation of Ceramic Fuels

The overall objective of this project is to study a limited number of advanced ceramic and cermet fuel concepts for the purpose of developing improved fuel materials for high temperature, high burnup applications. Specific objectives are: to determine the physical and mechanical properties of selected fuel materials; investigate the compatibility between fuel, bonding, and cladding materials; make preliminary investigations of techniques for the preparation of the fuel materials; and determine the behavior and operating limits of ceramic fuels for high temperature reactor service.

Major fuel development efforts on the SGR program are currently centered on uranium carbide as the most promising fuel for high temperature, high burnup application. It is necessary to continue investigation of other promising ceramic and cermet fuel combinations to develop improved fuels from the overall economic consideration of preparation, fabrication, handling, reprocessing, and reactor performance.

Recent developmental efforts on this project have been concentrated on U_3Si_2 . Physical properties, such as thermal conductivity, uranium density,

melting point, absence of transformation, and compatibility with bonding and cladding materials indicate promise for U_3Si_2 as a high temperature, long burn-up fuel. The application of induction melting and casting techniques has been demonstrated for economic fabrication of uranium silicide fuels. In addition, U_3Si_2 exhibits favorable resistance to oxidation up to 200°C which makes it attractive from the standpoint of handling and storage. Further studies are required to establish the time-temperature limits of application.

Favorable properties of high melting point, high thermal conductivity, high uranium density, and inertness to atmospheric reactions make uranium mononitride (UN) attractive for high temperature fuel applications. Disadvantages related to high thermal neutron cross section of nitrogen and to the complexity of preparation methods may be overcome to some extent by suitable solid solution alloying with carbon. Studies are required in the area of preparation methods, physical and mechanical property determinations, and compatibility with bonding and cladding materials.

Development Costs – Development and Evaluation
of Ceramic Fuels

	(In \$)		
	<u>FY 1965</u>	<u>FY 1966</u>	<u>FY 1967</u>
Operating funds	98,000	150,000	200,000
Equipment funds	<u>2,000</u>	<u>10,000</u>	<u>10,000</u>
	100,000	160,000	210,000

3. Hallam Nuclear Power Facility (HNPF) Program

a. Fuel Element Surveillance and Examination

The purpose of this project is: (1) to analyze and evaluate the performance of the U - 10 Mo fuel elements in Core I, 10 UC fuel elements in Core I; (2) to recommend fuel element removal as a result of the evaluation and UC fuel elements in Core II; (3) to design, fabricate, and install the required fuel-handling equipment in the HNPF maintenance cell; and (4) to provide the cask necessary to ship the fuel rods from the HNPF site to a hot cell for purposes of making detailed examinations. The program includes nondestructive examinations at HNPF maintenance cell and destructive examinations at CDHC.

Considerable technology relating to U - 10 Mo and UC irradiation behavior has been derived from small-scale capsule tests. The capsule tests only

approximate the conditions existing in a power reactor and the results must be extrapolated in an effort to predict the performance of full scale fuel elements. Preirradiation measurements made in conjunction with periodic examinations of actual fuel elements are required to provide sufficient statistical data for accurately evaluating the fuel element behavior. This project will provide the data and thus will permit continuing evaluation of predicted fuel element life. Liaison and support efforts necessary to handle and ship fuel rods and periodically transmit and interpret pertinent records of operating data will also be provided.

Maintenance cell equipment is required to permit necessary operations to be performed in the HNPF maintenance cell including removal of individual fuel rods from HNPF fuel elements, performing non-destructive examination of these fuel rods, and canning one to four fuel rods for shipment to a hot cell for destructive examination. The equipment will be capable of handling both Core I and Core II fuel elements. A suitable shipping cask will make possible rapid and safe transportation of HNPF irradiated fuel in accordance with 10 CFR 72.

Development Costs — Fuel Element Surveillance and Examination

	(In \$)				
	<u>FY 1965</u>	<u>FY 1966</u>	<u>FY 1967</u>	<u>FY 1968</u>	<u>FY 1969</u>
Operating funds	500,000	325,000	325,000	325,000	325,000
Equipment funds	<u>20,000</u>	<u>25,000</u>	<u>25,000</u>	<u>25,000</u>	<u>25,000</u>
	520,000	350,000	350,000	350,000	350,000

b. Reactor Operations Analyses (ROAP)

This project supports the primary HNPF objective of demonstrated high plant availability factor consistent with low power generation costs and safe operation. Plant operating data will be analyzed to detect any changes in operating parameters and safety factors caused by anomalies in the reactor and process system performance. The analyses of HNPF operating data are primarily to assure safe operating conditions, but will provide information on SGR performance characteristics and equipment operation and reliability which will be utilized in the design and operation of the prototype and 1000-Mwe plant.

Development Costs — Reactor Operations Analyses (ROAP)

	(In \$)				
	<u>FY 1965</u>	<u>FY 1966</u>	<u>FY 1967</u>	<u>FY 1968</u>	<u>FY 1969</u>
Operating funds	310,000	275,000	250,000	250,000	250,000
Equipment funds	<u>-</u>	<u>-</u>	<u>-</u>	<u>-</u>	<u>-</u>
	310,000	275,000	250,000	250,000	250,000

4. Facility Operation

a. Sodium Component Test Installation (SCTI)

This project is to provide services, supplies, and personnel for the operation of the Sodium Components Test Installation. The objective of this test operation is to obtain data on the performance of 35-Mwt intermediate heat exchangers (IHX) and steam generators in order to verify the thermal design, determine transient behavior, evaluate thermal stress design under transient conditions, evaluate pressure stress design, demonstrate structural integrity to repeated transients, determine startup methods and limitations, and evaluate compatibility of materials to the test environment. After these objectives have been met and evaluated, the units may be subjected to continued operation to destruction.

The need for engineering evaluation data for sodium-to-sodium and sodium-to-water-steam heat exchangers arises from peculiarities of the fluids involved, the structural materials used, and their application to nuclear reactors. Some of the factors involved are: (a) sodium reaction with water which requires that the structural integrity of a single-wall tubing steam generator be greater than that required by conventional types; (b) chloride stress corrosion in the area of water evaporation to dryness, which occurs when stainless steels are used; (c) the use of stainless steels, or more expensive materials, which are required to provide reasonable designs for high-pressure steam and high-temperature sodium; (d) the thermal characteristics of sodium which result in temperature changes being imposed on the containing materials rapidly while those of the stainless materials result in high thermal gradients across their thickness; (e) the transient characteristics of power reactors which impose severe temperature transient requirements on the intervening heat transfer systems; and (f) the integrity of the intermediate heat exchanger and the

performance of the steam generator relief system which must assure that any casualty condition in the steam generator cannot be reflected into the primary reactor system.

This facility will be required to test the modular steam generator concept for the 200-Mwe prototype plant and the 1000-Mwe SGR plant. Facility operating costs during the test periods associated with each module unit is covered under their respective plant development program. Steam generators and intermediate heat exchanger concepts, other than the module design, are scheduled for testing in this facility. These units include the ALCO steam generator and the B&W steam generator. Facility operating costs for testing these concepts are covered in this section.

Development Costs — Sodium Component Test Installation

	(In \$)		
	<u>FY 1966</u>	<u>FY 1967</u>	<u>FY 1968</u>
Operating funds			
ALCO steam generator	650,000	425,000	-
B&W steam generator	-	425,000	650,000
	<u>650,000</u>	<u>850,000</u>	<u>650,000</u>
Equipment funds	5,000	10,000	5,000
	<u>655,000</u>	<u>860,000</u>	<u>655,000</u>

5. Coordination, Reports, and Special Projects

The objectives of this project are to provide overall coordination and integration of all ASCR projects, provide for the publication and submission of ASCR Program progress reports, and provide for publication of topical reports on work accomplished on discontinued or cancelled projects.

This project will also provide for liason with the AEC on all ASCR projects, evaluation and analytical work as necessary on advanced sodium cooled reactor concepts and components not immediately associated with any current ASCR project, and the accomplishment of minor tasks requested by the AEC which are not specifically identifiable with other ASCR projects, and which do not merit separate funding.

The ASCR projects will be planned and scheduled so as to attain early achievement of program goals. The project is necessary because of the complexity and variety of projects encountered in the ASCR program. By providing coordination between projects and guidance of specific projects, a considerable savings in time, manhours, and cost is realized.

Development Costs — Coordination, Reports, and
Special Projects

(In \$)

	<u>FY 1965</u>	<u>FY 1966</u>	<u>FY 1967</u>	<u>FY 1968</u>	<u>FY 1969</u>
Operating funds	213,000	174,000	174,000	174,000	174,000
Equipment funds	<u>2,000</u>	<u>1,000</u>	<u>1,000</u>	<u>1,000</u>	<u>1,000</u>
	215,000	175,000	175,000	175,000	175,000

VI. SCHEDULE

The proposed schedule for the 200-Mwe Prototype is shown in Figure 24. The time from award of contract to on-line full power operation is expected to be 52 months. Initial criticality will be attained at the end of the 49th month, followed by 3 months of low power operation, approach to full power, and full power acceptance tests.

Key construction items and regulatory requirements are indicated in the schedule. The first 11 months are devoted exclusively to engineering. Ground breaking is scheduled for the 12th month, with major reactor plant construction essentially completed by the 38th month. The turbine generator plant requires less system checkout than does the reactor plant and is therefore scheduled for a later start to minimize interest costs during construction. The turbine-generator plant will be completed by the end of the 47th month, sufficient time for initial generation of steam, which will occur after criticality.

The project schedule for the 1000-Mwe SGR plant is anticipated to require 54 months. Engineering, construction, and preoperational testing are expected to follow the same sequence as outlined for the prototype plant. An additional 2 months are required for the 1000-Mwe plant because of field erection of the larger reactor structure.

Design of the prototype plant is scheduled to start in July 1964, resulting in full power commercial operation in November 1968. Eight months of commercial operation are anticipated before design of the 1000-Mwe plant is initiated, which is scheduled for July 1969. Commercial operation would begin in January 1974.

200 Mwe PROTOTYPE AND 1000 Mwe PLANT CONSTRUCTION SCHEDULE

Figure 24. 200-Mwe Prototype and 1000-Mwe Plant Construction Schedule

A. SODIUM HEAT TRANSFER CIRCUIT EVALUATION

In establishing the design of the sodium heat transfer system, attention must be paid to parameters such as temperature, size of individual components and previous experience, number of repetitive systems and their effect on plant availability and cost. The selection of five circuits for the 1000-Mwe plant represents an engineering compromise among all of the above factors.

Sodium reactor outlet temperatures of 1150°F represent a reasonable driving force to minimize heat transfer surface with steam conditions of 1000°F, while at the same time not overtaxing the practical design temperature limitations of the reactor core structural materials. Detail studies have indicated that the optimum ΔT for the sodium system is approximately 400°F with the selected steam pressure and temperature conditions.

The choice of number of circuits is influenced by component size, previous experience, cost, and effect on plant availability. The major components which must be considered are the pumps, valves, piping, intermediate sodium-to-sodium heat exchangers, and steam generators.

Actually utilizing the modular steam generators as proposed herein eliminates the need for considering the size of these items when determining optimum number of circuits. Each module generator is the same size. It is merely a question of how many are piped together to form the generator for a circuit. To a lesser extent the same thing is true for the intermediate heat exchanger. More than one unit can be utilized per loop, so that size of a unit is not a real governing factor in selection of number of circuits. When examining the cost of circuits, it is found that 55 to 65% of the cost is the heat exchanger surface (IHX and steam generator), which because of their modular design will not vary significantly in \$/kw for a given plant, regardless of number or size of circuits.

The size of the pumps and piping and valves for a 1000-Mwe plant is of course directly related to the number of circuits. As shown in Table 24, these sizes can vary from 33,000 gpm and 30-in. pipe for a 5-circuit system to 83,000 gpm and 48-in. pipes for a 2-circuit system.

TABLE 24
1000-MWE PLANT COMPONENT SIZES

Number of Circuits	Nominal Circuit Capacity (Mwe)	Pump Capacity* (10 ³ gpm)	Pipe Size† (in. OD)
6	167	28	27
5	200	33	30
4	250	41	34
3	333	55	39
2	500	83	48

*Based on delta T of 400°F in circuit

†Based on suction side velocity of about 15 ft/sec with nominal pipe sizes.

The large piping cost is almost a direct function of weight and, as pipe size increase, required wall thicknesses and hence weight also increase. Of course, the fewer the number of circuits, the less the cost of pump, structure, and instrumentation costs. Cost scoping studies for a 1000-Mwe plant as summarized in Table 25 indicate that three circuits give minimum capital cost; however, total circuit costs are rather flat as a function of number, varying by only about \$3/kw between maximum costs and minimum costs.

TABLE 25
ESTIMATED HEAT-TRANSFER SYSTEM COST VARIATION
WITH NUMBER OF CIRCUITS
(1000-Mwe Plant Size)

Cost in \$1000	2 Circuits	3 Circuits	4 Circuits	5 Circuits	6 Circuits
Structure	1,180	1,555	1,925	2,230	2,435
Heat exchangers	17,400	17,400	17,400	17,400	17,400
Pumps and piping	9,200	5,970	6,360	6,930	7,110
Instruments	2,850	3,200	3,550	3,900	4,250
Total	30,630	28,125	29,235	30,460	31,195
Dollars per kw	30.6	28.1	29.2	30.4	31.2

In choosing proper number of circuits, attention must be given to their effect on the availability of the plant since capital costs differences are not decisive.

The common definition of availability factor is the ratio of the time a generating unit or piece of equipment is ready for or in service to the total time interval under consideration. In applying this definition to a multicircuit reactor plant, where partial capacity may be generated with one or more circuits out of service, consideration must be given not only to the availability of the reactor to generate power, but to the capability of transferring the reactor power to the steam generators. One way to do this is to state availability in terms of availability to generate equivalent full power hours.

Without any circuit-forced outages or scheduled outages, a plant would be available for full-power generation 100% of the time. Scheduled or forced outages of the entire plant or portions thereof would reduce the total amount of energy the plant would be capable of generating. This measure of plant reliability is one of loss of energy probability. The results are stated in terms of the probable ratio, over an interval, of load energy that could be provided to the amount of load energy that could be provided plus the total potential energy lost because of outages, forced or scheduled.

The percent of maximum equivalent full power operation possible, considering the probability of forced circuit outages, as a function of number of circuits is shown in Figure 25. This is based on an assumed probability of in-service time of 0.95, or in other words 5% forced outage per circuit. As may be seen, the greater the number of circuits, the greater the availability, although the change becomes progressively smaller as more circuits are added.

Large reactor plants will be built because of their low incremental fuel cost as compared to conventional plants. A forced outage of a reactor will mean that spinning reserve and peaking capacity will have to be used and this capacity will generally be high incremental cost units. If the differential incremental fuel cost because of a forced outage is assumed to be 2 mill/kwh (extra cost of alternate power), and the differential incremental capital cost of additional circuits is as shown in Table 25, then the effect of number of circuits on total power cost will be as shown in Figure 26. Incremental additional capital costs and fuel cost

Figure 25. Effect of Number of Sodium Heat Transfer Circuits on Equivalent Full Power Availability

Figure 26. Incremental Power Costs vs Number of Sodium Heat Transfer Circuits

penalties are additive for 2 circuits or less. Three circuits represents minimum capital cost but the fuel cost penalties are higher than for a greater number of circuits. As number of circuits increases above three, incremental capital costs increase but fuel cost penalties decrease.

As a result of the cost and availability studies described above, four circuits would appear optimum but three or five circuits appear to be reasonable selections. Within the accuracy of the scoping studies, there is not a great deal to choose between three, four, or five circuits. A 5-circuit 1000-Mwe SGR was selected as the reference design for the large SGR in view of component size scaleup from current sodium operating components. As may be seen from Table 24, it is quite feasible to ultimately upgrade the prototype circuit to suit a 4-circuit 1000-Mwe plant or possibly even a 3-circuit 1000-Mwe plant.

B. STEAM CYCLE OPTIMIZATION STUDIES

In recent years, optimum steam cycle and sodium process conditions for an SGR have been a subject for continual investigation at Atomics International. Since most nuclear power stations involve steam pressures in the reactor complex, technical limitations prevent exploitation of modern steam conditions. This limitation is not imposed on the low-pressure SGR system; hence, selection of steam conditions is basically governed by an economic balance between net station heat rate, fuel cycle cost, and equipment cost. The primary restriction to high pressure-high temperature steam conditions for SGR application is similar to those with present fossil-fueled power stations — namely, material structural limitations due to temperature.

The main function of the steam cycle optimization studies has been to determine the most economical parameters for the conceptual design state of a particular size SGR plant. Having established the initial basic data, selection of the final design values, based on additional detailed ground rules and cost information as the initial design progresses, would then involve a minimum of effort to upgrade the final design.

This section of the appendix summarizes results of a steam cycle optimization study which was conducted for an early SGR plant design with a nominal power rating of 360 Mwe. Most of the results of this study are still pertinent for smaller or larger SGR plants. Perturbations to the basic data for specific

application to the 200-Mwe Prototype and the 1000-Mwe SGR plant are discussed in this section.

The general trend that is apparent from these optimization studies for SGR nuclear power plants are as follows.

- 1) For plant ratings up to approximately 350 Mwe, 2400-psig steam pressure would be the most economical operating condition.
- 2) For plant ratings above 350 Mwe, 3500-psig steam pressure is the economic selection.
- 3) The highest justifiable single reheat steam temperature condition for 1150°F reactor outlet temperature is 1000°F/1000°F, the final selection being either 950°F/950°F or 1000°F/1000°F.
- 4) For all plant ratings, the best range for the value of the sodium temperature difference across the reactor is from 375 to 425°F.
- 5) The best range for the value of the temperature approach of the intermediate heat exchanger for the 2400-psig steam cycle is 65 to 80°F, while for the 3500-psig steam cycle the range is 85 to 95°F.

1. 360-Mwe SGR Plant Study

The purpose of the first phase of the study was to establish the optimum steam pressure conditions. Four standard steam pressures, namely, 1450, 1800, 2400, and 3500 psig were studied. The type of steam cycles were non-reheat, single reheat, and double reheat. The steam temperature was varied from 830 to 1050°F, the reactor outlet sodium temperature from 950 to 1150°F, and the sodium inlet temperature to the reactor from 650 to 850°F. Steam conditions and steam cycles initially studied are tabulated in Table 26.

The feedwater temperatures and number of heaters are based on steam cycle information published in NAA-TDR-6524 for use in cycle optimization studies. This working paper provided information at that time to illustrate minimum heat rates which might be practically obtained in nuclear power plants. The question as to whether the added expenditure for extra feedwater heaters plus associated piping and controls warrant the savings in operating cost is covered in a later portion of this study.

TABLE 26
STEAM CYCLE CONDITIONS EXAMINED
FOR SGR APPLICATION

	Steam Pressure (psig)			
	3500	2400	1800	1450
Nonreheat Cycle (°F)				
950	-	-	-	x
1000	-	-	-	x
Single Reheat Cycle (°F)				
830/830	-	-	-	x
900/900	-	-	-	x
950/950	x	x	x	x
1000/1000	x	x	x	x
1050/1050	x	x	x	-
Double Reheat Cycle (°F)				
950/950/950	x	-	-	-
1000/1000/1000	x	-	-	-
1050/1050/1050	x	-	-	-
Feedwater Temperature (°F)	600	550	530	500
Number of Feedwater Heaters	9	8	7	7

Ground rules for the first part of the study are as follows.

- a) Tandem compound 4-flow, 29-in. LSB turbine generator units operating at 3600 rpm were used as reference machines for 3500-, 2400-, 1800-, and 1450-psig single reheat cycles. Costs of these units for various steam temperature are shown in Figure 27. The 1450-psig nonreheat cycle is based on a tandem compound double flow unit with 43-in. blades operating at 1800 rpm.
- b) The differential cost of the reactor was based on \$50,000 for each 1% thermal power rating above a base reference size reactor.
- c) The annual differential capitalized fuel costs were based on using a $20¢/10^6$ Btu fuel cost and an 80% annual plant load factor. To

Figure 27. Plot of Turbine Generator Costs for 350-Mwe Nominal Rating for a Tandem Compound 4-Flow Machine with 29-in. Buckets for Different Inlet Steam Temperature

cover the general area of privately vs publically financed power plants, a capital charge rate of 14 and 7%, respectively, were used in this study.

- d) Heat-exchange equipment was sized for 395-Mw electrical capacity; net plant capability is 360 Mwe.
- e) Use 5/8-in.-OD tubing, 37-tube bundle in 6-in. pipe shell for superheater and evaporator modules, and 102-tube reheater in 10-in.-pipe shell. For high-temperature module, use 347 stainless steel; use 9 chrome-1/2 moly for low-temperature and reheater modules; use 304 stainless steel in intermediate exchanger.

- f) Limit sodium velocities to about 15 ft/sec, steam velocities to about 150 ft/sec for piping.
- g) Turbine cycle heat rates are based on NAA-TDR-6524. The net turbine cycle heat rates include boiler feed pump power but exclude condensate, pump, circulating pump, and main sodium pump power requirements.
- h) Main steam condenser sizes are based on: (1) \$4.50/ft² of required surface, (2) 7/8-in.-OD tubes, 18 BWG, 36-ft length, (3) 7.5 ft/sec water velocity; inlet/outlet circulating water temperature = 57°F/85°F, and (4) cleanliness factor = 0.85; condensing at 1.5-in. Hg abs.
- i) Feedwater heater and piping, steam and condensate piping, plus associated pumps and drives were extrapolated from costs shown in Kaiser Report KE-60-31.

The cost of heat exchanger surface derived from information available at the time is as follows.

	Surface Cost in \$/ft ² (psig)			
	1450	1800	2400	3500
Superheater module (HTM)	45	50	55	60
Evaporator module (LTM)	40	45	50	50
Reheater module	35	35	35	40
Intermediate heat exchanger (IHX)	35	35	35	35

The first part of the study was based on four major cost items which are: (1) cost of superheater, evaporator, reheater and intermediate heat exchanger modules; (2) cost of main turbine generator; (3) differential cost of reactor; and (4) differential annual capitalized fuel costs.

The approach was to arrive at the total cost of the foregoing items for each steam cycle condition listed in Table 26. The case with the lowest total cost was then selected as representative for that particular steam cycle. This essentially covered a total of 60 cases calculated at 14% capital charge rate, and 60 cases calculated at 7% capital charge rate. These optimum cases are shown in Table 27.

TABLE 27

COST OPTIMUMS FOR HEAT-EXCHANGER SURFACE ITEMS

Capitalization Rate (%)	Throttle Steam Pressure (psig)	Throttle Steam Temperature (°F)	First Reheat Steam Temperature (°F)	Second Reheat Steam Temperature (°F)	Reactor Outlet Temperature (°F)	Sodium Temperature to Steam Generator (°F)
14	3500	950	950	-	1150	1050
	3500	950	950	950	1150	1050
	2400	950	950	-	1150	1050
	1800	950	950	-	1150	1050
	1450	950	950	-	1100	1000
	1450	950	-	-	1100	1000
7	3500	1000	1000	-	1150	1050
	3500	950	950	950	1150	1050
	2400	1000	1000	-	1150	1050
	1800	1000	1000	-	1150	1050
	1450	950	950	-	1100	1000
	1450	950	-	-	1100	1000

One typical set of curves for the 2400-psig steam cycle at 7% capital charge rate is shown in Figure 28 to illustrate how the temperature conditions were arrived at as listed in Table 27. The lowest combined cost as shown in Figure 28 is for (1) reactor outlet temperature of 1150°F, (2) sodium inlet temperature to steam generator of 1050°F, and (3) a steam temperature of 1000°F. Therefore, this selection is shown in Table 27 for the 2400-psig steam pressure condition at 7% capital charge rate.

The next tabulation, Table 28, is based on the results of Table 27, plus the following items:

- a) Costs of main steam condensers
- b) Costs of feedwater and condensate systems

A review of the differential comparative cost items from Table 28 indicates that:

- a) The 2400-psig steam condition at 14% capital charge rate is the lowest cost, the next better pressure condition being 3500 psig.
- b) The 3500-psig steam condition at 7% capital charge rate is the lowest cost, the next better pressure condition being 2400 psig.
- c) At the 14% capital charge rate for all steam pressures concerned, the final steam temperature trend was 950°F in all cases.
- d) At 7% capital charge rate, the temperatures varied at 950°F, or 1000°F.

It is apparent that either 2400 psig or 3500 psig is the optimum steam pressure for a large SGR, depending on a utility's capitalization rate. At the time of this study, greater interest centered about sites where the prevailing capital charge rates ranged from 11.5 to 14.0%, therefore the 2400-psig conditions were selected for more refined analyses. In addition, supercritical steam conditions were considered somewhat too advanced for the next SGR plant, and the predominant development programs on steam generators were directed to 2400-psig conditions. These factors defined a more narrow range of steam conditions to be studied, namely, 2400 psig/1000°F/1000°F and 2400 psig/950°F/950°F.

Figure 28. Differential Capital Costs for 2400-psig Single Reheat Steam Cycle Including Steam Generators and IHX's

TABLE 28
COST COMPARISON OF STEAM CYCLES, INCLUDING MAIN CONDENSER, CONDENSATE AND FEEDWATER SYSTEMS

Capitalization Rate	Throttle Steam Pressure (psig)	Throttle Steam Temperature (°F)	1st Reheat Steam Temperature (°F)	2nd Reheat Steam Temperature (°F)	Reactor Outlet Temperature (°F)	Sodium Temperature to Steam Generator (°F)	Net Turbine Cycle Heat Rate* (Btu/kw-hr)	Change in Heat Rate (Btu/kw-hr)	① Value of Annual Capitalized Differential Fuel Costs (\$)	② Differential Heat Exch Costs (HTM, LTM, RHT and IHX) (\$)	③ Differential Main Steam Turbine Costs (\$)	④ Differential Reactor Cost (\$)	⑤ Differential Feedwater and Condensate System Costs (\$)	⑥ Differential Condenser Costs (\$)	⑦ Differential Comparative Cost of Items 1 through 6 (\$)
14%	3500	950	950	—	1150	1050	7754	Base	Base	Base	Base	Base	Base	Base	Base
	3500	950	950	950	1150	1050	7519	-235	-847,000	+139,000	+500,000	-151,000	+213,000	-39,000	-185,000
	2400	950	950	—	1150	1050	7885	+131	+469,000	-33,000	-203,000	+85,000	-759,000	+20,000	-421,000
	1800	950	950	—	1150	1050	8081	+327	+1,179,000	-45,000	-135,000	+210,000	-1,300,000	+53,000	-38,000
	1450	950	950	—	1100	1000	8208	+454	+1,636,000	+443,000	+5,000	+293,000	-1,671,000	+77,000	+783,000
	1450	950	950	—	1100	1000	8260	+506	+1,824,000	-55,000	+75,000	+325,000	-1,751,000	+99,000	+517,000
7%	3500	1000	1000	—	1150	1050	7640	-114	-822,000	+413,000	+193,000	-74,000	+102,000	-21,000	-209,000
	3500	950	950	950	1150	1050	7519	-235	-1,694,000	+232,000	+500,000	-152,000	+121,000	-39,000	-1,032,000
	2400	1000	1000	—	1150	1050	7746	-8	-58,000	+362,000	-10,000	-5,000	-659,000	-2,000	-372,000
	1800	1000	1000	—	1150	1050	7968	+214	+1,543,000	+371,000	+57,000	+138,000	-1,220,000	+32,000	+921,000
	1450	950	950	—	1100	1000	8208	+454	+3,272,000	+443,000	+5,000	+293,000	-1,670,000	+77,000	+2,420,000
	1450	950	—	—	1100	1000	8260	+506	+3,647,000	-55,000	+75,000	+325,000	-1,751,000	+99,000	+2,340,000

+Indicates additional cost over Base

-Indicates savings in cost over Base

*Based on Hegetschweiner paper (BER 1222) which includes BFP power but excludes other pumping powers such as condensate, circulating water, also main sodium pumping

The objectives of the second phase of study were to determine the effect of various parameters on the selection between these two steam conditions. The parameters to be studied were:

- a) Different type turbines: (1) cross-compound double flow, 43 in.; (2) cross-compound double flow, 38 in.; (3) tandem compound 4-flow, 29 in.; and (4) tandem compound 4-flow, 26 in.

- b) Low, medium, and high set of steam generator surface costs

	<u>(\$/ft²)</u>		
	<u>Set 1</u>	<u>Set 2</u>	<u>Set 3</u>
High temperature module	50	75	110
Low temperature module	45	55	90
Reheater module	35	40	40
IHX module	35	40	35

- c) Variation in feedwater temperatures from 425 to 550°F
- d) Plant factor: from 80 to 60%
- e) Fuel costs: from 20¢/10⁶ Btu to 25¢/10⁶ Btu
- f) Include capitalized sodium and feedwater pumping power costs
- g) Use 11.55% capitalization rate
- h) Variation of the intermediate heat exchanger LMTD between 45 to 125 °F.

The purpose of including these further details for the 2400-psig steam condition was to determine if there was a definite trend to 1000°F or 950°F final steam temperature. In addition, it was necessary to determine if the more expensive turbine with the better heat rate and the feedwater cycle with the highest feedwater temperature, coupled with higher cycle efficiency, were worth the added equipment first costs to justify the lower operating costs.

The curves shown in Figures 29, 30, and 31 indicate the trend as steam generator surface costs are progressively increased. The lowest surface costs in Figure 29 show a trend to the 1000°F steam temperature condition (about \$120,000 lower than 950°F). As the steam generator costs are increased (Figures 30 and 31), there is a standoff between the 950 and 1000°F steam

Figure 29. Differential Plant Costs vs Reactor ΔT for Low Heat Exchanger Surface Costs

Figure 30. Differential Plant Costs vs Reactor ΔT for Medium Heat Exchanger Surface Costs

Figure 31. Differential Plant Costs vs Reactor ΔT for High Heat Exchanger Surface Costs

temperature. In all cases, the optimum reactor sodium temperature difference is in the 350 to 400°F range. The optimum temperature approach for the intermediate heat exchanger is in the range from 65 to 80°F. Calculations for intermediate heat-exchanger temperature approaches at 45, 95, and 125°F resulted in much higher overall costs than for the 65 to 80°F range.

In Figure 32, the feedwater temperature was varied from 550 to 475°F and 425°F for one steam condition at 2400 psig/1000°F/1000°F using a cross-compound double-flow 43-in. turbine. The trend indicates that 550°F feedwater

Figure 32. Differential Plant Costs vs Reactor ΔT for Different Feedwater Temperatures

temperature cannot be justified on an economic basis. Since the curves for the 475 and 425°F feedwater temperature basically overlap in the 350 to 400°F reactor ΔT range, the selection of either is a matter of judgment. The same trends apply for the 2400 psig/950°F/950°F steam condition.

In Figure 33, four different type turbines are compared for 2400 psig/950°F/950°F steam condition, and in Figure 34 the same turbines for the 2400 psig/1000°F/1000°F steam condition. The trends for the type turbine in each figure are definitely toward the tandem compound 4-flow 26-in. turbine, the lowest cost turbine of the four selected. The most expensive turbine, costing approximately \$400,000 more on a total comparative cost basis, would be the cross-compound double-flow 43-in. unit.

In Figure 35, the tandem compound 4-flow 26-in. turbine, the best turbine selection from Figures 33 and 34 is plotted for each steam condition; the trend indicating that the 2400 psig/950°F/950°F steam condition is slightly more economical than 2400 psig/1000°F/1000°F.

Figure 33. Differential Plant Costs vs Reactor ΔT for Different Types of Turbines at 2400 psig/950°F/950°F

Figure 34. Differential Plant Costs vs Reactor ΔT for Different Types of Turbines at 2400 psig/1000°F/1000°F

Figure 35. Differential Plant Costs vs Reactor ΔT for Tandem Compound 4-Flow 26-in. Turbine at 2400 psig/1000°F/1000°F and 2400 psig/950°F/950°F

In Figure 36, the capacity factor was changed from 80% (as used in Figure 35) to 60%. The result is a stronger trend to favoring the 2400 psig/950°F/950°F steam condition.

Figure 36 . Differential Plant Costs vs Reactor ΔT for 60% Capacity Factor

In Figure 37, the fuel cost was changed from 20¢/10⁶ Btu (as used in Figures 29 through 36) to 25¢/10⁶ Btu. The trend now is towards the 2400 psig/1000°F/1000°F steam condition.

Figure 37. Differential Plant Costs vs Reactor ΔT for 25¢/10⁶ Btu Fuel Cost

Based on the foregoing results the following final conditions were recommended for the 360-Mwe SGR plant.

- a) The reactor sodium temperature differential was selected to be 400°F. Although the 400°F differential is slightly higher in total costs than 350°F, the higher ΔT offers a more conservative temperature drive across the cold end of the evaporator module and reduces the pressure drop through the core.
- b) The intermediate heat-exchanger temperature approach was selected as 75°F.
- c) The turbine selection was the tandem compound 4-flow 26-in. machine since, regardless of steam conditions, it offered the lowest total plant cost.
- d) The feedwater temperature was selected as 475°F rather than 550°F, since it showed a definite cost advantage over 550°F and no noticeable disadvantage to 425°F. In addition, the 475°F temperature follows the nominal temperature of 470°F for this size and pressure turbine.
- e) As to the selection of the cycle steam temperature conditions, it is evident that changes in heat-exchanger costs, fuel costs, and load factors, alter the trend as follows.
 - 1) As fuel costs rise, the trend is toward the cycle with the better heat rate; namely, 2400 psig/1000°F/1000°F.
 - 2) As load factor increases the trend is to the cycle with the higher heat rate; namely 2400 psig/1000°F/1000°F.
 - 3) As heat-exchanger costs increase the trend is toward the cycle with the lower heat rate; namely, 2400 psig/950°F/950°F.

The final steam conditions selected for the 360-Mwe SGR plant are 2400 psig/1000°F/1000°F, based on the best judgement at that time for general conceptual design purposes. Detailed analysis, prior to final design based on current costs and customer conditions, would be required to verify this selection. This completed the effort on the 360-Mwe plant study.

2. Steam Cycle Studies for 200-Mwe Prototype and 1000-Mwe SGR Plant

The plan of action for establishing the plant parameters for the 200-Mwe prototype and 1000-Mwe SGR plant was to evaluate those items that most influenced the plant parameters, as determined from past steam cycle studies.

The ground rules that were established for this study are: (1) 14% capital charge rate, (2) $20\text{¢}/10^6$ Btu fuel cost, and (3) 80% annual plant load factor.

a. 200-Mwe Prototype Plant

Since the 360-Mwe SGR study did indicate that the 3500-psig steam condition was second-best to the 2400 psig/1000°F/1000°F cycle for the 14% capitalization rate, this pressure was not considered economical for the 200-Mwe prototype. A 200-Mwe turbine constructed for 3500-psig steam operation would cost approximately \$150,000 to \$200,000 more than a 2400-psig-200 Mwe machine. The prices shown in the latest turbine handbook establish the breakpoint at approximately 400 Mwe between 2400- and 3500-psig steam conditions, with the 3500-psig unit becoming progressively cheaper than the 2400-psig unit as machine size increases above 400 Mwe.

The following tabulation of turbine first costs and erection plus the value of the annual differential capitalized fuel costs for a 200-Mwe turbine rating indicates that the tandem compound double-flow 30-in. LSB turbine is the economic selection.

<u>200-Mwe Rated Type Turbine</u>	<u>Differential Turbine Cost (\$ x 10³)</u>	<u>Annual Differential Capital Fuel Cost (\$ x 10³)</u>	<u>Combined Differential Turbine Plus Differential Fuel Cost (\$ x 10³)</u>
<u>3600/3600 rpm</u>			
TCDF-30 in.	+ 50.0	+256.0	+ 306.0
TCTF-23 in.	Base	+387.0	+ 387.0
TCTF-26 in.	+ 240.0	+217.0	+ 457.0
TCFF-23 in.	+ 510.0	+170.0	+ 680.0
TCFF-26 in.	+ 606.0	+113.0	+ 719.0
CCFF-23 in.	+ 760.0	+170.0	+ 930.0
CCFF-26 in.	+1260.0	+113.0	+1373.0
<u>3600/1800 rpm</u>			
CCSF-43 in.	+ 660.0	+106.0	+ 766.0
CCSF-52 in.	+1160.0	+ 18.0	+1178.0
CCDF-38 in.	+1710.0	Base	+1710.0

Requirements for the rated flow guarantee conditions follow:

1) Main steam superheater outlet:	flow	1,351,741 lb/hr
	pressure	2500 psia
	temperature	1005 °F
	enthalpy	1461.7 Btu/lb
2) Steam reheater outlet:	flow	1,183,827 lb/hr
	pressure	515 psia
	temperature	1005 °F
	enthalpy	1520.4 Btu/lb
3) Steam reheater inlet:	flow	1,183,827 lb/hr
	pressure	540 psia
	temperature	630 °F
	enthalpy	1312.2 Btu/lb
4) Feedwater evaporator inlet:	flow	1,351,741 lb/hr
	temperature	472.5 °F
	enthalpy	456.6 Btu/lb
5) Generator output		210,049 kw
Net station output		198,700 kw
6) Net turbine cycle heat rate		7803 Btu/kwh
Net station heat rate		8160 Btu/kwh
7) Reactor thermal power		478.4 Mw

Since the individual duty requirements of the steam generator and reheat units for the 200-Mwe plant are essentially proportional to those for the 360-Mwe plant, the curves plotted for the 360-Mwe SGR study (Figures 29 and 37) would be relatively parallel for the 200-Mwe prototype, and would result in the same trends. The discrepancy could be compensated for by a proportioning factor applied to the units of the ordinate in each figure. It therefore follows, that the same reactor sodium temperature differential, temperature approach for the intermediate heat exchanger and steam pressure/temperature conditions would apply for the 200-Mwe prototype. These results applied to the 200-Mwe prototype are as follows: (1) reactor outlet temperature, 1150°F; (2) reactor sodium temperature differential, 400°F; (3) intermediate heat-exchanger approach temperature, 75°F; and (4) 2400 psig/1000°F/1000°F. This summarizes the basic

approach in establishing the plant parameters for the 200-Mwe SGR prototype plant conceptual design.

b. 1000-Mwe SGR Plant

The same ground rules as outlined for the 200-Mwe prototype plant were used for the 1000-Mwe SGR plant.

Since the turbine represents one of the major cost items, a similar comparison of combined differential turbine costs plus differential annual capitalized fuel costs based on 14% capital charge rate is shown in Table 29. The table compares 2400-psig with 3500-psig turbines. Since the standard 2400-psig turbines are manufactured up to ratings of 600 Mw, two 500-Mw units were used. A 1000-Mwe, 2400-psig turbine is being designed for installation but it consists essentially of two 500-Mwe units connected in parallel.

TABLE 29
COMBINED DIFFERENTIAL TURBINE COST PLUS
DIFFERENTIAL ANNUAL CAPITALIZED FUEL
COST BASED ON 14% CAPITAL RATE

Type of Turbine (in.)	Number of Turbines	Rating Per Turbine (Mw)	Steam Conditions (\$)	
			2400 psig/1000°F/1000°F	3500 psig/1000°F/1000°F
CC8F-30	One	1000	-	+ 188,000
CC4F-43	One	1000	-	Base
CC6F-38	One	1000	-	+ 252,000
CC4F-52	One	1000	-	+ 484,000
CC6F-43	One	1000	-	+ 1,308,000
TC4F-26	Two	500	-	+ 4,056,000
TC4F-30	Two	500	-	+ 4,090,000
TC6F-26	Two	500	-	+ 4,652,000
TC6F-30	Two	500	+4,074,000	+ 5,436,000
CCDF-43	Two	500	-	+ 6,812,000
CCDF-52	Two	500	+6,054,000	+ 7,466,000
CC4F-38	Two	500	+6,768,000	+ 8,250,000
CC4F-43	Two	500	+9,636,000	+11,188,000
CC4F-52	Two	500	-	+15,146,000
CC6F-26	Two	500	+5,690,000	+ 8,492,000
CC6F-30	Two	500	+6,214,000	+ 8,966,000
CC8F-26	Two	500	+6,184,000	+ 8,436,000
CC8F-30	Two	500	+8,822,000	+11,584,000

The results of Table 29 indicate that the 3500-psig, 1000-Mwe cross-compound 4-flow 43-in. LSB turbine machine is the economical selection. In addition, the best selection of two 500-Mwe 2400-psig turbines would cost \$4,074,000 (two TC6F-30-in. machines) above the 3500-psig cross-compound 4-flow 43-in. LSB unit.

Based on the past steam cycle studies, there is no particular item or group of items, representing the major costs for the 1000-Mwe SGR plant, which could provide savings in excess of \$4,074,000 that would allow one to select the 2400-psig steam cycle over the 3500-psig condition. Since the steam side of the plant, as far as the reactor is concerned, stops at the steam side of the steam generator, the reactor core main primary sodium and main secondary sodium systems are not affected by the 2400- or 3500-psig pressure selection. The steam side of the steam generator will have higher design pressures but the comparative steam generator exchanger costs will not increase proportionally. The added steam generator exchanger costs for thicker pressure sections for 3500-psig operation will be compensated for by less total surface required because of improved heat transfer coefficients for supercritical steam.

Rated flow requirements for the 1000-Mwe guarantee condition are as follows.

1) Steam superheater outlet:	flow	7,237,650 lb/hr
	pressure	3600 psia
	temperature	1005 °F
	enthalpy	1424.8 Btu/lb
2) Steam reheater outlet:	flow	5,078,700 lb/hr
	pressure	706 psia
	temperature	1005 °F
	enthalpy	1516.4 Btu/lb
3) Steam reheater inlet:	flow	5,078,700 lb/hr
	pressure	745 psia
	temperature	603 °F
	enthalpy	1278.6 Btu/lb
4) Feedwater evaporator inlet:	flow	7,237,650 lb/hr
	temperature	511.8 °F
	enthalpy	501.0 Btu/lb

5) Reactor thermal power	2336 Mwe
Primary sodium flow	66.0×10^6 lb/hr
Secondary sodium flow	66.0×10^6 lb/hr
6) Generator output	1,050,603 kw
Net station output	1,019,100 kw
7) Net turbine cycle heat rate	7514 Btu/kwh
Station heat rate	7824 Btu/kwh
Station efficiency	43.6%

This summarizes the basic approach in establishing the plant parameters for the 1000-Mwe SGR plant conceptual design.

C. REACTOR CORE OPTIMIZATION STUDY

1. Introduction

a. Purpose of Study

The proposed 200-Mwe SGR is the prototype for a 1000-Mwe SGR to be built in 1970's. It is therefore desirable to design the 200-Mwe plant so that it will prototype as many features of the 1000-Mwe plant as possible. In particular, it is desirable that the fuel element and lattice cell of the prototype be similar or identical to those to be used in the large reactor. The technical and economics characteristics of a 1000-Mwe reactor must therefore be known. To this end, a series of parametric optimization studies on a 1000-Mwe advanced SGR has been completed. A description of these studies and the results obtained are presented here.

b. Ground Rules

The ground rules for the study and the parameters held constant throughout the study are given in Table 30.

Fixing the coolant inlet and outlet temperatures at reasonable values obtained from previous studies of the ex-core heat transfer system is a useful expedient. This allows the core to be optimized by itself, since the core and the external system are connected almost exclusively through the coolant inlet and outlet temperatures.

TABLE 30

GROUND RULES FOR 1000-MWE SGR OPTIMIZATION STUDIES

Fuel element type	Multirod cluster with central support tube
Fuel material	Hypostoichiometric UC
Structural material	304 SS
Coolant	Na
Moderator	Graphite
Type of moderator containment	Calandria tank
Lattice type	Square pitch
Coolant inlet temperature	750°F
Coolant outlet temperature	1150°F
Maximum cladding temperature	1250°F
Core length	14 ft
Core pressure drop	21 psi
Average fuel discharge exposure	25,000 Mwd/MTU
Core thermal power	2350 Mw
Fuel cladding thickness	0.010 in. for fuel rods up to 0.500-in. diameter; proportional to fuel diameter above 0.500 in.
Process tube thickness	0.025 in. for process tubes up to 4.25-in. ID; proportional to process tube diameter above 4.25-in. ID
Fraction of fuel elements in core replaced at each refueling shutdown	0.25
Power peaking factors: radial	1.50
axial	1.35

It is desirable that the fuel element length of the prototype be the same as that of the large plant. This eliminates redesign of the fuel handling machine, control rods and drives, fuel element fabrication procedures, etc. For this reason, 14 ft was selected as a reasonable core height for both the 200-Mwe and the 1000-Mwe SGR. It is believed that total cost is a relatively weak function of core height.

c. Independent Variables

It was convenient to take the four independent parameters shown in Table 31 as variables for the study.

TABLE 31
INDEPENDENT VARIABLES AND RANGE OF EACH

Variable	Range
Number of fuel rods per element	12, 18, 30
Cross-sectional fuel area per element (in. ²)	3.4 to 6.0
Moderator-to-fuel (c/u) atom ratio in cell	62 to 95
Maximum fuel centerline temperature (°F)	1750, 2000, 2250

d. Dependent Variables

Given the ground rules tabulated above, and given a value for each of the independent variables tabulated above, all other significant core parameters become dependent variables whose value may be determined. Some of the major dependent variables found for each case studied are given in Table 32.

TABLE 32
PRIMARY DEPENDENT VARIABLES

Fuel rod diameter
Lattice Spacing
Feed fuel enrichment
Discharge fuel enrichment
Number of fuel elements
Number of control rods
Core diameter
Fuel specific power
Fuel linear heat rate
Coolant area per element
Process tube diameter
Maximum coolant velocity
Fuel in-core residence time
Cycle time between refueling shutdowns

2. Technical Analysis

a. Nuclear Survey

Not all of the independent variables have a strong effect on core physics. The two most significant variables from the nuclear standpoint are the fuel area and the moderator-to-fuel ratio. It was assumed that these two variables are the only ones having significant interaction; i.e., two clusters having the same fuel area but different number of rods and rod diameter will have similar nuclear characteristics, all other factors being approximately equal. There are reactivity differences due to different sodium and steel areas, but this difference is small and relatively independent of other variables. Therefore, a complete parametric survey was performed on the 18-rod cluster only. One case was calculated for each of the other clusters studied to determine the reactivity variation due to the number of rods in the cluster. Similarly, changes in fuel centerline temperature and in the core dimensions result in small reactivity changes. These effects were accounted for by making reactivity adjustments in the curves of reactivity vs burnup for an infinite core and a single fuel temperature. Thus, a complete analysis including reactivity lifetime calculations was made for a total of 29 cases. Twenty-seven of these cases were for an 18-rod cluster.

Enrichments from 2.7 to 4.8% U^{235} were considered for each of the following nine geometries.

<u>Case No.</u>	<u>Fuel Rod Diameter (in.)</u>	<u>Fuel Area (in.²)</u>	<u>C/U Atom Ratio</u>	<u>Square Lattice Pitch</u>
1	0.500	3.534	62	9.25
2	0.500	3.534	78	10.25
3	0.500	3.534	95	11.25
4	0.575	4.674	62	10.64
5	0.575	4.674	78	11.79
6	0.575	4.674	95	12.94
7	0.650	5.973	62	12.02
8	0.650	5.973	78	13.32
9	0.650	5.973	95	14.62

In addition, the following two cases were studied in order to compare different clusters having the same fuel area:

Case No.	Number Rods in Cluster	Enrichment (% U ²³⁵)	Fuel Area (in. ²)	Rod Diameter (in.)	C/U Atom Ratio
1	12	3.9	3.534	0.612	78
2	30	3.9	3.534	0.387	78

The results of the physics calculations on the 18-rod cluster calculations are given in Table 33. It was assumed that the values of most nuclear parameters other than reactivity (i.e., burnup characteristic, discharge Pu concentration, etc.) were independent of the number of rods per element, provided that the fuel area, initial enrichment, and C/U ratio are not changed.

b. Heat Transfer Survey

The following nine geometrically different fuel clusters were analyzed.

Number Rods in Cluster	Fuel Rod Diameter (in.)	Fuel Area (in. ²)
12	0.600	3.39
12	0.700	4.62
12	0.800	6.03
18	0.500	3.53
18	0.575	4.67
18	0.650	5.97
30	0.400	3.77
30	0.450	4.77
30	0.500	5.89

For each of these cases, limiting fuel centerline temperatures of 1750, 2000, and 2250°F were studied. For each case, the coolant area, and hence the process tube diameter, was adjusted so that the core pressure drop for each case was limited to 21 psi. Since core thermal performance is not affected by lattice spacing, this variable was not considered in the heat transfer analysis.

The results of the heat transfer analysis, as well as some of the basic fuel element dimensions and areas, are given in Table 34.

TABLE 33

PHYSICS DATA FROM SURVEY OF 18-ROD CLUSTER CASES

Fuel Rod Diameter (in.)	Lattice Spacing (in.)	C/U Atom Ratio	Feed Enrichment, E_o (wt %)	ΔE ($E_o - E_s$) (wt %)	Discharge Enrichment, E_f (wt %)	Discharge Pu Concentration (gm Pu/kg U)	Average Burnup Characteristic (% ΔK per 1000 Mwd/MTU)	Migration Area (M^2)	K_{∞} , 18-Rod Cluster, 2000°F Maximum Fuel Temperature
0.500	9.25	62	3.3	1.985	1.315	8.16	1.253	435.9	1.201
0.500	9.25	62	4.0	2.155	1.845	8.48	1.242	425.8	1.245
0.500	9.25	62	4.8	2.307	2.493	8.72	1.213	414.2	1.282
0.500	10.25	78	3.0	1.989	1.011	6.50	1.328	482.8	1.224
0.500	10.25	78	3.7	2.175	1.525	6.82	1.290	470.0	1.277
0.500	10.25	78	4.4	2.315	2.085	7.07	1.240	456.9	1.316
0.500	11.25	95	2.7	1.970	0.730	5.40	1.455	538.4	1.228
0.500	11.25	95	3.4	2.185	1.215	5.71	1.394	521.6	1.292
0.500	11.25	95	4.0	2.317	1.683	5.91	1.310	507.2	1.331
0.575	10.64	62	3.3	1.995	1.305	7.87	1.289	457.6	1.232
0.575	10.64	62	4.0	2.160	1.840	8.19	1.268	447.5	1.273
0.575	10.64	62	4.8	2.309	2.491	8.42	1.227	435.9	1.309
0.575	11.79	78	3.0	2.000	1.000	6.33	1.372	511.9	1.250
0.575	11.79	78	3.7	2.185	1.515	6.64	1.310	499.0	1.301
0.575	11.79	78	4.4	2.322	2.078	6.86	1.258	486.1	1.339
0.575	12.94	95	2.7	1.981	0.719	5.30	1.507	576.0	1.250
0.575	12.94	95	3.4	2.195	1.205	5.59	1.420	559.3	1.312
0.575	12.94	95	4.0	2.326	1.674	5.77	1.335	544.9	1.351
0.650	12.02	62	3.3	2.000	1.300	7.69	1.312	481.0	1.256
0.650	12.02	62	4.0	2.168	1.832	7.98	1.283	471.0	1.298
0.650	12.02	62	4.8	2.311	2.489	8.20	1.235	459.6	1.332
0.650	13.32	78	3.0	2.008	0.992	6.22	1.405	543.4	1.272
0.650	13.32	78	3.7	2.192	1.508	6.51	1.321	530.6	1.323
0.650	13.32	78	4.4	2.326	2.074	6.71	1.269	517.8	1.352
0.650	14.62	95	2.7	1.990	0.710	5.25	1.545	616.6	1.270
0.650	14.62	95	3.4	2.204	1.196	5.51	1.448	600.1	1.332
0.650	14.62	95	4.0	2.331	1.669	5.67	1.350	585.9	1.367

TABLE 34

SUMMARY OF DATA FROM HEAT TRANSFER CALCULATIONS

Maximum Fuel Temperature (°F)	Number of Fuel Rods per Element	Fuel Rod Diameter (in.)	Clad Thickness (in.)	Process Tube Thickness (in.)	Process Tube OD (in.)	Fuel Area (in. ²)	Sodium Area (in. ²)	Clad and Spacer Area (in. ²)	Process Tube Area (in. ²)	Number of Fuel Elements in Core	Number of Control Rods in Core	Fuel-to- Control Element Ratio	Number of Fuel Rods in Core	Fuel Mass per Rod (kg U)	Fuel Mass per Element (kg U)	Core Fuel Inventory (MTU)	Average Specific Power (kw/kg U)	Maximum Linear Heat Rate (kw/ft)	Hot Channel Coolant Velocity (fps)	Average Fuel Element In-Core Residence Time (fp yr)	Cycle Time Between Refueling Shutdowns (fp month)	Total Down Time per Refueling Shutdown (days)	% Down Time for Refueling (80% Load Factor Assumed)
1750	30	0.400	0.010	0.027	4.580	3.77	10.93	0.554	0.386	525	70	7.54	15,750	4.41	132.3	69.5	33.8	23.9	13.5	2.02	6.06	8.12	3.66
1750	30	0.450	0.010	0.028	4.798	4.77	11.20	0.636	0.420	528	79	6.66	15,840	5.58	167.4	88.4	26.6	24.6	13.1	2.58	7.74	8.16	2.81
1750	30	0.500	0.010	0.029	5.021	5.89	11.44	0.720	0.455	532	91	5.82	15,960	6.89	206.7	110.1	21.3	25.2	12.7	3.21	9.63	8.22	2.28
1750	18	0.500	0.010	0.025	4.042	3.53	7.67	0.412	0.316	783	101	7.74	14,090	6.89	124.0	97.1	24.2	25.2	12.9	2.83	8.49	11.71	3.68
1750	18	0.575	0.012	0.025	4.311	4.67	7.91	0.555	0.337	789	118	6.68	14,020	9.11	164.0	129.8	18.1	25.9	12.4	3.78	11.34	11.80	2.77
1750	18	0.650	0.013	0.027	4.587	5.97	8.08	0.677	0.387	799	138	5.81	14,380	11.64	209.6	167.6	14.0	26.4	12.0	4.89	14.67	11.93	2.17
1750	12	0.600	0.012	0.025	3.646	3.39	5.60	0.384	0.284	1133	142	7.96	13,600	9.92	119.0	135.0	17.4	26.1	12.2	3.94	11.82	16.57	3.74
1750	12	0.700	0.014	0.025	3.936	4.62	5.78	0.516	0.307	1145	170	6.72	13,740	13.50	162.0	185.7	12.7	26.7	11.7	5.42	16.26	16.73	2.74
1750	12	0.800	0.016	0.025	4.234	6.03	5.92	0.658	0.331	1163	203	5.73	13,960	17.64	211.6	246.4	9.5	27.2	11.2	7.18	21.54	16.98	2.10
2000	30	0.400	0.010	0.029	4.921	3.77	13.42	0.554	0.446	398	58	6.86	11,940	4.41	132.3	52.6	44.7	31.6	14.5	1.54	4.62	6.57	3.79
2000	30	0.450	0.010	0.030	5.129	4.77	13.72	0.636	0.481	399	66	6.05	11,970	5.58	167.4	66.8	35.2	32.6	14.1	1.94	5.82	6.58	3.01
2000	30	0.500	0.010	0.031	5.343	5.89	13.99	0.720	0.517	402	76	5.29	12,060	6.89	206.7	83.2	28.2	33.4	13.7	2.42	7.26	6.62	2.43
2000	18	0.500	0.010	0.025	4.312	3.53	9.42	0.412	0.337	591	84	7.04	10,640	6.89	124.0	73.4	32.0	33.3	13.9	2.14	6.42	9.25	3.84
2000	18	0.575	0.012	0.027	4.572	4.67	9.68	0.555	0.386	595	98	6.07	10,710	9.11	164.0	97.8	24.0	34.3	13.4	2.85	8.55	9.30	2.90
2000	18	0.650	0.013	0.028	4.838	5.97	9.90	0.677	0.423	602	114	5.28	10,840	11.64	209.6	126.2	18.6	35.1	13.0	3.68	11.04	9.40	2.27
2000	12	0.600	0.012	0.025	3.863	3.39	6.86	0.384	0.301	854	118	7.24	10,250	9.92	119.0	101.7	23.1	34.6	13.2	2.97	8.91	12.90	3.86
2000	12	0.700	0.014	0.025	4.139	4.62	7.05	0.516	0.323	862	141	6.11	10,340	13.50	162.0	139.9	16.8	35.5	12.7	4.08	12.24	13.01	2.83
2000	12	0.800	0.016	0.026	4.429	6.03	7.21	0.658	0.360	875	168	5.21	10,500	17.64	211.6	185.3	12.7	36.2	12.2	5.40	16.20	13.19	2.17
2250	30	0.400	0.010	0.030	5.222	3.77	15.78	0.554	0.489	322	52	6.17	9,660	4.41	132.3	42.6	55.1	39.0	15.2	1.24	3.72	5.72	4.10
2250	30	0.450	0.010	0.032	5.426	4.77	16.12	0.636	0.542	323	59	5.44	9,690	5.58	167.4	54.1	43.5	40.3	14.9	1.58	4.74	5.74	3.23
2250	30	0.500	0.010	0.033	5.630	5.89	16.40	0.720	0.580	325	68	4.76	9,750	6.89	206.7	67.2	35.0	41.3	14.5	1.96	5.88	5.76	2.61
2250	18	0.500	0.010	0.026	4.551	3.53	11.05	0.412	0.370	477	75	6.34	8,586	6.89	124.0	59.2	39.7	41.3	14.7	1.73	5.19	7.88	4.05
2250	18	0.575	0.012	0.028	4.800	4.67	11.32	0.555	0.420	480	88	5.46	8,640	9.11	164.0	78.9	29.8	42.5	14.2	2.30	6.90	7.92	3.06
2250	18	0.650	0.013	0.029	5.057	5.97	11.57	0.677	0.458	485	102	4.75	8,730	11.64	209.6	101.7	23.1	43.5	13.8	2.97	8.91	7.99	2.39
2250	12	0.600	0.012	0.025	4.053	3.39	8.03	0.384	0.316	689	106	6.52	8,268	9.92	119.0	82.0	28.6	42.9	14.0	2.39	7.17	10.82	4.02
2250	12	0.700	0.014	0.025	4.322	4.62	8.26	0.516	0.338	694	126	5.50	8,328	13.50	162.0	112.6	20.9	44.1	13.5	3.28	9.84	10.89	2.95
2250	12	0.800	0.016	0.027	4.604	6.03	8.43	0.658	0.388	703	150	4.69	8,436	17.64	211.6	149.0	15.8	45.0	13.0	4.34	13.02	11.01	2.25

TABLE 35

NUCLEAR AND HEAT TRANSFER VARIABLES - 1000-MWE SGR

Number of Fuel Rods per Element	Fuel Rod Diameter (in.)	C/U Atom Ratio	Lattice Pitch (in.)	Maximum Fuel Central Temperature											
				1750°F				2000°F				2250°F			
				E _o Feed Enrichment (wt %)	E _f Discharge Enrichment (wt %)	Discharge Pu Concentration (gm Pu/kg U)	Core Diameter (ft)	E _o Feed Enrichment (wt %)	E _f Discharge Enrichment (wt %)	Discharge Pu Concentration (gm Pu/kg U)	Core Diameter (ft)	E _o Feed Enrichment (wt %)	E _f Discharge Enrichment (wt %)	Discharge Pu Concentration (gm Pu/kg U)	Core Diameter (ft)
30	0.400	62	9.56	4.54	2.28	8.65	21.9	5.04	2.69	8.80	19.2	5.55	3.13	8.88	17.4
30	0.400	78	10.59	3.84	1.63	6.88	24.3	4.22	1.94	7.01	21.3	4.57	2.22	7.12	19.3
30	0.400	95	11.63	3.56	1.34	5.77	26.7	3.85	1.57	5.86	23.4	4.11	1.77	5.94	21.2
30	0.450	62	10.76	4.00	1.84	8.18	24.9	4.37	2.13	8.31	21.8	4.73	2.43	8.40	19.8
30	0.450	78	11.92	3.47	1.34	6.55	27.6	3.75	1.55	6.65	24.2	4.03	1.78	6.75	21.9
30	0.450	95	13.09	3.28	1.12	5.55	30.3	3.54	1.31	5.64	26.5	3.76	1.48	5.71	24.0
30	0.500	62	11.94	3.59	1.51	7.82	28.0	3.88	1.74	7.94	24.6	4.18	1.98	8.04	22.3
30	0.500	78	13.23	3.23	1.15	6.32	31.0	3.44	1.31	6.41	27.2	3.65	1.47	6.49	24.7
30	0.500	95	14.52	3.13	1.00	5.42	34.1	3.32	1.14	5.48	29.8	3.50	1.27	5.54	27.1
18	0.500	62	9.25	3.95	1.80	8.46	25.9	4.26	2.04	8.57	22.6	4.62	2.34	8.68	20.4
18	0.500	78	10.25	3.39	1.29	6.69	28.7	3.62	1.46	6.79	25.0	3.86	1.65	6.88	22.6
18	0.500	95	11.25	3.17	1.05	5.62	31.5	3.37	1.19	5.70	27.5	3.57	1.34	5.77	24.8
18	0.575	62	10.64	3.52	1.47	7.98	30.1	3.80	1.68	8.11	26.7	4.06	1.88	8.21	23.8
18	0.575	78	11.79	3.12	1.08	6.39	33.4	3.32	1.23	6.48	29.6	3.50	1.36	6.56	26.4
18	0.575	95	12.94	3.00	0.92	5.44	36.7	3.17	1.03	5.50	32.5	3.32	1.15	5.56	29.0
18	0.650	62	12.02	3.21	1.23	7.64	34.6	3.43	1.39	7.75	30.2	3.63	1.55	7.84	27.4
18	0.650	78	13.32	2.96	0.97	6.21	38.3	3.12	1.08	6.28	33.5	3.23	1.15	6.33	30.3
18	0.650	95	14.62	2.89	0.83	5.32	42.1	3.01	0.91	5.38	36.8	3.13	1.00	5.42	33.4
12	0.600	62	9.07	3.68	1.60	8.34	30.5	3.92	1.78	8.44	26.6	4.15	1.96	8.53	24.0
12	0.600	78	10.05	3.18	1.14	6.59	33.8	3.35	1.26	6.68	29.5	3.52	1.38	6.75	26.6
12	0.600	95	11.03	3.01	0.93	5.55	37.0	3.15	1.03	5.61	32.3	3.28	1.13	5.66	29.2
12	0.700	62	10.58	3.25	1.27	7.84	36.1	3.44	1.41	7.94	31.5	3.62	1.54	8.03	28.5
12	0.700	78	11.72	2.93	0.95	6.30	40.0	3.06	1.04	6.36	34.9	3.18	1.13	6.42	31.6
12	0.700	95	12.87	2.84	0.81	5.37	43.9	2.95	0.88	5.42	38.3	3.05	0.95	5.46	34.7
12	0.800	62	12.08	2.99	1.08	7.51	42.0	3.15	1.19	7.61	36.7	3.29	1.29	7.68	33.2
12	0.800	78	13.39	2.80	0.86	6.12	46.5	2.90	0.92	6.17	40.7	3.00	0.99	6.22	36.8
12	0.800	95	14.69	2.74	0.74	5.26	51.0	2.83	0.79	5.30	44.6	2.92	0.85	5.34	40.4

c. Integration of Heat Transfer and Nuclear Analysis

Combining the nuclear and heat transfer analyses results in a total of 81 core configurations (4 variables, 3 values of each). For each case, all the dependent variables were determined and tabulated. The values of these variables, not given in previous tables, are given in Table 35.

3. Economics Analysis

a. Fuel Cycle Cost

For each of the 81 cases studied, the equilibrium core fuel cycle cost was calculated. Fuel cycle cost was calculated for the case of private fuel ownership as well as the case of government fuel ownership. The economics ground rules and bases for the fuel cycle cost calculation are given in Table 36.

TABLE 36

GROUND RULES FOR FUEL CYCLE COST CALCULATION

- 1) Current price schedule (as of July 1962) used for the value of UF_6 .
- 2) Value of Pu as metal: \$9.50/gm Pu
- 3) Lease rate for U: 10%/yr for private fuel ownership, 4-3/4%/yr for government ownership of fuel.
- 4) Cost of money for items other than fuel: 6%/yr
- 5) Fuel fabrication cost as follows:

Fuel Rod Diameter (in.)	Fabrication Cost (\$/kg U)
0.4	114.06
0.5	100.00
0.6	92.36
0.7	87.76
- 6) Cost of shipping fuel: \$3.00/kg U unirradiated
\$10.00/kg U irradiated
- 7) Ex-core inventory: 6 mo during fabrication plus 160 days during cooling and reprocessing of irradiated fuel
- 8) Reprocessing plant charge: \$17,000/day
- 9) Conversion to UF_6 : \$5.60/kg U
- 10) Conversion of PuN to Pu metal: \$1.50/gm Pu
- 11) Plant capacity factor: 80%

b. Incremental Capital Cost

For each case, the incremental capital cost associated with the reactor core was estimated. The cost of the plant equipment external to the reactor proper (e.g., the sodium system, heat exchangers, pumps, turbines, etc.) is essentially constant for all cases, since the coolant inlet and outlet temperatures and the core power level was held constant. Thus, the incremental capital cost is a small fraction of the total cost of a 1000-Mwe reactor plant. The cost of all capital equipment was written off at 14% per year and converted to mill/kwh assuming a load factor of 80%. The factors considered in determining the incremental capital cost are given in Table 37.

TABLE 37
FACTORS FOR ESTIMATING INCREMENTAL CAPITAL COST

	Core Parameters Item Depends Upon
Calandria vessel	Core diameter
Reactor vessel	Core diameter
Reactor vessel liner	Core diameter
Thermal shield	Core diameter
Thermal insulation	Core diameter
Cavity liner	Core diameter
Concrete in top shield	Core diameter
Lead in top shield	Core diameter
Steel structure in top shield	Core diameter
Outlet manifold	Core diameter
Sodium in reactor	Core diameter
Reflector graphite	Core diameter
Orifice tubes	Number of channels, channel diameter
Shield plug assemblies	Number of channels, channel diameter
Process tubes	Number of channels, channel diameter
Top shield sleeve	Number of channels, channel diameter
Moderator graphite	Core diameter, number of channels, channel diameter

4. Results and Conclusions

The criterion used to compare the relative economic potential of the cases studied is the incremental total cost, equal to the fuel cycle cost plus the incremental capital cost.

Figures 38 through 40 show the incremental total cost for each of the 81 cases, plotted vs core diameter. Curves are given for private fuel ownership as well as government fuel ownership. Figure 41 shows incremental total cost vs fuel slug diameter. Only the minimum cost lattice spacing is used in this figure. As may be seen from this figure, incremental total cost decreases with increasing maximum fuel temperature over the range studied. Furthermore, the optimum fuel rod diameter varies from about 0.470 in. for 1750°F maximum fuel temperature to about 0.600 in. for 2250°F maximum fuel temperature.

A maximum fuel temperature of 2000°F was selected as the most probable extrapolation of current data consistent with a 1969 plant construction date and an average fuel discharge exposure of 25,000 Mwd/MTU. For this fuel temperature, these studies indicate that a fuel slug diameter of about 0.500 in. is near the optimum and that fuel area variations between 3.5 and 6.0 in.² make a negligible difference in the incremental total cost. These results are essentially unchanged if government ownership of fuel is assumed, instead of private fuel ownership.

Prior to this study, analytical and design efforts were concentrated on the reference 200-Mwe fuel element, an 18-rod cluster of 0.500-in. fuel rods set on a 10.25-in. square lattice pitch. The results of this study indicate that this element is close to optimum for a 1000-Mwe plant with 2000°F limiting fuel temperature. For this reason, this element, with the process tube opened up from 3.95-in. ID to 4.28-in. ID to accommodate the higher specific power, was selected as the reference element for the 1000-Mwe plant.

Nevertheless, the possibility of using other clusters of fuel rods of about 0.500 in. in diameter should not be ruled out at this time. Particularly for the larger plant sizes, 24 or 30 fuel rods per fuel element may prove attractive. Further studies to explore this narrow range of element types will be initiated in the future, but for the present, the reference 1000-Mwe element will remain an 18-rod cluster of 0.500-in.-diameter fuel rods set on a 10.25-in. square pitch.

Figure 38. Incremental Total Cost vs Core Parameters for Maximum Fuel Temperature of 1750°F

Figure 39. Incremental Total Cost vs Core Parameters for Maximum Fuel Temperature of 2000°F

Figure 40. Incremental Total Cost vs Core Parameters for Maximum Fuel Temperature of 2250°F

Figure 41. Increment Total Cost vs Core Parameters at Optimum Lattice Spacing

D. CONTROL ELEMENT DESIGN CRITERIA

Extensive nuclear and heat transfer studies of the control element have been conducted for the 200-Mwe reactor operating conditions. The reference rod employs the dry-rod concept and contains dysprosium oxide as the poison material which is capable of attaining a 10-yr life. The main problem encountered with the dry rod is the large heat generation that must be transferred across a helium gap between the poison column and the thimble without creating excessive temperatures in the rod structural material. Figure 42 illustrates the sheath

Figure 42. Hastelloy X Sheath Temperature as Function of Linear Heat Rate with Initial Cold-Gap Clearance as Parameter

temperature for various gaps over the linear heat range of interest for the thermally expanded operating conditions of the control rod. The operating hot-gap thickness over the linear heat rate range as a function of the initial cold gap is shown in Figure 43. Figure 44 is a special plot of the former figure, depicting the clad sheath temperature for an initial cold gap of 35 mils over a larger linear heat rate range.

The axial distribution of heating in the rod from rod-born gammas is calculated for rod insertions of $1/4$, $1/2$, and $3/4$, and full-in, based on the flux shapes on the surface of the rod and the maximum thermal absorption rates. The contribution to heating from core-born gammas is neglected since it is less than 1%. The heating effect from thermal neutron absorption in the rod is also negligible.

Figure 43. Hot-Gap Thickness vs Linear Heat Rate with Initial Cold Gap as Parameter

Figure 44. Hastelloy X Sheath Temperature vs Linear Heat Rate for Initial Cold Radial Gap of 35 Mils

For every neutron absorbed by the dysprosium, with neutron captures by the graphite and clad neglected, an energy spectrum of gamma rays is promptly released. The total energy of the released photons is equal to the binding energy of the neutron, which in the case of dysprosium is 6.66 Mev. The fraction of the 6.66 Mev per neutron capture which leaks out of the control rod is calculated and, in turn, the fraction that is absorbed. Although dysprosium oxide acts more nearly as a surface source, it can be approximated by a uniform volume source with a small percentage error in leakage probability. Applying this calculational method to the dysprosium oxide control element proposed for the 200-Mwe SGR, a heat-generation rate due to capture gamma rays is determined. To this value is added the decay energy from $\text{Dy}^{165\text{M}}$ (dysprosium in the meta-stable state). By conservatively assuming that all of the gamma and beta decay

energy will be absorbed in the dysprosium, a total heat generation rate is found for normal reactor operation.

The total energy release was calculated from the average neutron absorption rates and the energy released per absorption. Heat generation rates were calculated by GRACE-II, a gamma ray attenuation code. It was assumed that the beta energy remains in the rod and that in the case of the Dy_2O_3 rod, all decay gammas were absorbed in the rod and that all neutron absorptions occurred in the dysprosium.

Figure 45 shows the relative thermal neutron flux on the surface of the rod, which is also the relative axial distribution of absorptions of thermal neutrons. Although there exists a sizeable number of epithermal absorptions, it is assumed that this configuration is a reasonable representation of their distribution. The absorption data, given in Table 38, are the basis of the calculations. Table 39 summarizes the detailed results of the calculations.

Figure 45. Relative Thermal Neutron Flux on Surface of Black Control Rod for 1/4, 1/2, 3/4, and Full Insertion of Rod

TABLE 38

NEUTRON ABSORPTION RATES FOR 110% OF
CORE THERMAL POWER

Rod Insertion	Full	3/4	1/2	1/4
Maximum neutron absorption rate (neutron/cm-sec x 10 ⁻¹⁵)	1.01	1.66	1.44	0.863
Axial peak-to-average	1.45	2.47	2.14	1.25
Effective rod length (ft)	14.0	12.5	9.0	5.5

TABLE 39

DYSPROSIUM OXIDE CONTROL ROD SUMMARY FOR
110% OF CORE THERMAL POWER

Capture gamma energy (Mev/absorption)	6.67
Beta plus gamma decay energy at saturation (Mev/absorption)	0.44
Effective half-life of decay power (hr)	3.4
For saturation of decay isotope	
Peak power in rod, full in (kw/ft)	16.0
Average power in rod, full in (kw/ft)	10.4
Peak power in rod, 3/4 in (kw/ft)	20.9
Average power in rod, 3/4 in (kw/ft)	7.1
Total power in rod, full in (kw)	146.0
Total power in rod, 3/4 in (kw)	89.0
Total energy release, full in (kw)	337.0
Total energy release, 3/4 in (kw)	292.0
Estimated leakage probability of rod, full in	0.54
Estimated leakage probability of cell	0.12

For the case of saturation of the decaying isotope, the heat generation rate is plotted as a function of axial location in Figure 46. Based upon the gamma energy flux from the fuel elements into the moderator, 15 kw was added to the control rod channel cooling load.

Figure 46. Heat Generation Rate vs Distance from Tip of Dy_2O_3 Rod

Based on the rod heat generation rates, heat transfer calculations were performed to determine the peak temperature in the Hastelloy-X clad inner surface. The peak heat generation rate is 21 kw/ft, and yields a maximum clad temperature of 1953°F for a cold gap of 35 mil. This case applies to a central control rod, three-fourths inserted in the core, with the peak heat generation rate located 5 cm from the rod tip on a very narrow spike as shown in Figure 46. This spike is due to the "lightning rod effect" and can be reduced somewhat by the use of a gray absorber material or tapered poison thickness at the rod tip. At the base of the spike, the linear heat rate is approximately 15 kw/ft and produces a temperature of 1800°F. Thus, for a 35-mil cold gap, the clad temperature will be in the range of 1800 to 1900°F, which is high but within material design limits. As can be seen from Figure 42, clad temperature is sensitive to gap width, and larger gaps would result in unacceptable operating temperatures. Smaller gaps would reduce temperatures; however, as indicated in Figure 43, the initial cold gap of 35 mil will reduce to 27 to 28 mil at peak operating temperatures. There is a practical limit to gap reduction for operating clearance, and 35 mil appears to be a minimum gap.

The 1000-Mwe SGR reactor, operating at a specific power of 32 kw/kg U as compared to a specific power of 24 kw/kg U for the 200-Mwe prototype, has a peak control element heat generation rate of 28 kw/ft and would produce a peak temperature of 2080°F for a 35-mil cold gap. Assuming the rod tip can be designed to completely eliminate the heating spike due to the lightning rod effect, the temperature in the clad would still be in the order of 1950°F. The control element with structural Hastelloy-X at this temperature would be marginal. Without suffering an economic penalty that would be incurred in lowering reactor specific power or increasing the number of rods with a gray poison to limit the dry-rod temperatures, the only other feasible alternative appears to be the wet-rod approach. In this design, the heat generated in the absorber would be directly transmitted to the reactor sodium coolant without having to go through a helium gap. As a result, the temperatures would be significantly less at the same power level. This is the primary factor which led to the adoption of the wet control element as the reference design for the 1000-Mwe SGR.