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SOVIET POWER REACTORS-1970

Report of the United States of America
Nuclear Power Reactor Delegation Visit
to the Union of Soviet Socialist Republics,
June 15—July 1, 1970

Division of Reactor Development
and Technology
United States Atomic Energy Commission

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Cover Photograph

View of one of two banks of five desalination evaporators each in center. Fossil fired station at BN-350 site at left. BN-350 reactor building at left far rear. 7 sea water screen wells behind canal. Canal from the Caspian Sea in foreground. Desalination control room behind screens.

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August 1970

Division of Reactor Development
and Technology

United States Atomic Energy Commission

U.S. Delegation and Shevchenko Staff

Construction—BN-600

U.S. Delegation on BR-5

Novovoronezh Turbine-Generator Hall

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I. SUMMARY

A U.S. Nuclear Power Reactor Delegation toured U.S.S.R. reactor facilities and research institutions from June 15 through June 30, 1970, reciprocating a visit by a U.S.S.R. team in November 1969. This was the second round of reciprocal visits on nuclear power, the first information exchange by visitation occurring in 1964. The objective of the visit was to observe Soviet progress in development and construction of nuclear power reactors.

The material in this report has been arranged into two major sections, the first section (the body of the report) being a concise description of the technical disciplines for each reactor program. The second section, the appendixes, provide a detailed description of the reactor projects. For the most part, the description of the reactor disciplines is based upon firsthand knowledge gained by extensive discussions between the U.S. Delegation and the U.S.S.R. nuclear-power engineers and by observations of supporting facilities. The descriptions of the reactor projects, including the desalting plant at Shevchenko, are also based upon firsthand discussions and observations of the facilities involved. However, a substantial amount of reference background material on each project was reviewed and incorporated, as appropriate.

A. FAST-REACTOR PROGRAM

The U.S.S.R. has now focused its development programs exclusively on the development of the sodium-cooled breeder reactor. The objective of the U.S.S.R. fast-reactor program is to meet long-range power needs. The rate of increase in power demand is such that a doubling time of 7-8 years is sought.

The U.S.S.R. now has two small fast reactors in operation, BR-5 and BOR-60 (5 and 60 MWt, respectively). Construction of a third, BN-350 (350 equivalent* megawatts electric), at Shevchenko on the Caspian Sea, is nearing completion, and construction of the civil works for a fourth, BN-600 (600 MWe), is about half completed.

The schedule for BN-350 is to complete construction by end of 1970, achieve criticality by mid-1971, and approach full-power operation late in 1971. The reactor vessel is in place and major components and piping are being installed,

*Only 150 MW is electrical output; the remaining 200 MW will be used as steam to desalinate Caspian Sea water.

excluding pump drives and internals. The schedule for BN-600, which was started in 1968, is less firm, completion being estimated at between mid-1973 and mid-1975.

The Soviets have extrapolated the technology developed in BR-5 to the design and construction of BOR-60, BN-350, and BN-600. BOR-60, designed to be a fuel-test reactor, was recently started up and has reached 20 MWt. Prototype fuels for both BN-350 and BN-600 will be inserted before the reactor is taken to full power at the end of 1970.

Alternative design concepts for both reactors and major components are to be tested in the fast reactors now under construction. Thus, BN-600 is a tank-type reactor, while BR-5, BOR-60, and BN-350 are piped-type (loosely coupled) reactors. The steam generators and intermediate heat exchangers in BN-350 are of different design than those on BOR-60 and BN-600. Notable in the design of BN-600 is the use of a ring girder to support components from the bottom, thereby reducing the load the roof must support. Differential expansion between the top deck and a truss structure inside the reactor vessel which supports the main components is taken up by large removable bellows, which can also take 1.5 cm (0.59 in.) of lateral displacement. The objective of these variations in design concepts is to get large-scale operating experience with them for use in design of later "standard" fast reactors.

In BOR-60, BN-350, and BN-600, the cores are very tight. While the cores for BOR-60 and BN-350 are fixed, the core for BN-600 could still be changed to factor in design changes that might be required to accommodate swelling.

Annular fuel pellets (0.20- to 0.23-in. OD with a 0.072-in. hole) are used in the Soviet fast reactors, the internal hole eliminating central melting, providing a path for fission-gas release to gas plenums, and providing void space within the fuel pins. All cores initially are enriched UO_2 , excepting that in BR-5. BR-5 is now operating with an enriched uranium carbide core, but there appears to be no intent to pursue this fuel. After completion of the present work with uranium carbide, BR-5 will be modified to increase the power to 10 MWt. Operation will be resumed with a plutonium oxide core. There appears to be little experience with mixed oxides.

The accepted cladding is a Russian stainless steel (16 wt % Cr, 15 wt % Ni, 3 wt % Mo, and 0.8 wt % Nb).

whose out-of-reactor properties are better than those of Type 316 stainless steel.

The U.S.S.R. safety philosophy is that those accidents or failures that could be disastrous to the environs of their nuclear plants are made so unlikely as to be incredible. Reactor design is based on the assumption that there will be no failed-fuel-element propagation (none was observed as a result of many fuel failures in BR-5), and that there is no credible accident corresponding to a severe accident excursion or total loss of coolant. Credible initiating incidents are handled so they will not lead to extensive core damage.

The U.S.S.R. program in fast reactors can be characterized as aggressive, as shown by the current construction of large-scale units of different designs, including the biggest under way in the world. They are being built on the basis that any deficiencies in plant design, fabrication practices, and technology will be amenable to correction.

B. WATER-REACTOR PROGRAM

Three of the sites visited also included water-cooled reactors. To the extent to which the U.S. Delegation visit permitted such a determination, the water-reactor technology on PWR's and BWR's appeared to be well within the performance range of similar reactors in the West. One exception is the two boiling-superheating reactors at Beloyarsk, a 100-MWe unit operating since 1964 and a 200-MWe unit operating since 1967, which have overcome initial operating difficulties and are now operating at high capacity factors. A considerable number of technical innovations are evident in the design and operation of these reactors, relative to any water-cooled reactors in the U.S.

At Novovoronezh, two pressurized-water reactors (210 and 365 MWe) are in operation. A third unit, 440 MWe in capacity and now under construction, is scheduled to be on line in 1971. Two additional units (440 and 1000 MWe) are to be built—to be on line in 1973 and 1975, respectively,

thus completing the total planned capacity of 2500 MWe at Novovoronezh. For the near future the U.S.S.R. apparently has standardized the 440-MWe plant. Two such units are being built at Murmansk and two at Oktomberyan in Armenia. Such standard units are also being built in East Germany, Bulgaria, Czechoslovakia, Finland, Hungary, and Rumania.

The fuel for the PWR's has also been standardized—annular UO_2 pellets clad in zirconium-1 wt % niobium. A significant technical result in conventional water-reactor technology appeared to be the continuing good performance of the zirconium-1 wt % niobium alloy cladding for PWR fuel. Currently, fuel burnups are in the range of 10,000 to 15,000 MWd/ton, but target burnups for Units 4 and 5 at Novovoronezh are 40,000 MWd/ton. At the Kurchatov Institute in Moscow, where most of the fuel development work is done, experimental fuel assemblies have been carried to 63,000 MWd/ton.

Thermal and radioactive pollution are of concern. Stringent limits are placed on discharge activity to ground or river waters. At Novovoronezh, the allowable temperature rise in the Don River is 5°C (9°F) in the summer and 3°C (5.4°F) in the winter. To accommodate Units 3 and 4, seven natural-circulation cooling towers will be built. For the last unit, a 1500-acre cooling lake will be built. The radioactive release criterion from the station is 200 Ci/day. At Beloyarsk, a 10-km^2 (about 2500-acre) lake has been constructed for cooling the first three or four units. The nominal stack discharge limit is 1000 Ci/day, but the average rate is 250 Ci/day, of which 80% is ^{41}Ar (110-min half-life).

The safety philosophy for the light-water reactors is similar to that for the LMFBR's in the U.S.S.R. For example, a double-ended pipe break is considered incredible; therefore no separate emergency core cooling systems, no containment buildings, no ice condensers, and no pressure suppressant chambers are provided.

II. INTRODUCTION

The visit of the United States Nuclear Power Reactor Delegation from June 15 through June 30, 1970, was conducted under the terms of the 1970-1971 Memorandum on Cooperation in the Peaceful uses of Atomic Energy between the United States Atomic Energy Commission and the State Committee of the U.S.S.R. for the Utilization of Atomic Energy. This visit by a U.S. Power Reactor Delegation to the U.S.S.R. is the second one to have been made. The previous U.S. Delegation visited the U.S.S.R. in December 1964. Both that visit and this one were reciprocal to visits of U.S.S.R. Power Reactor Delegations made to the U.S. in the spring of 1965 and fall of 1969.

Selection of members of the 1970 U.S. Power Reactor Delegation was based upon their technical qualifications in AEC's high-priority Liquid Metal Fast Breeder Reactor Program, combined with their experience in the Light Water Reactor Program. Members of the Delegation were as follows:

Dr. G. W. Wensch (Delegation Leader)
Special Assistant to the Director
Division of Reactor Development and Technology
U.S. Atomic Energy Commission

Mr. A. Amorosi, Assistant Laboratory Director for
Reactor Development and Technology Programs
Argonne National Laboratory

Mr. L. Burris, Jr., Manager
Sodium Technology Program
Argonne National Laboratory

Mr. R. W. Dickinson, Division Director
Liquid Metal Engineering Center

Dr. E. A. Evans, Manager
Fuels and Materials
WADCO Corporation
Hanford Engineering Development Laboratory (HEDL)

Dr. P. Greebler, Manager
Nuclear Engineering Subsection
Breeder Reactor Development Operation
General Electric Company

Mr. W. B. McDonald, Manager
Plant Design Engineering Department
WADCO Corporation
Hanford Engineering Development Laboratory (HEDL)

Dr. P. Murray, Technical Director and Manager
Fuels and Materials
Westinghouse Advanced Reactor Division

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Division of Reactor Development and Technology
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Dr. E. L. Zebroski, Manager
Development Engineering Section
Breeder Reactor Development Operation
General Electric Company

II. INTRODUCTION

The U.S.S.R. Delegation visited seven sites plus USAEC Headquarters. The U.S. Delegation visited six sites and had a final summary discussion with the U.S.S.R. State Committee on Atomic Energy.

The U.S. Delegation toured facilities and had extensive discussions with technical staffs at the Melekh Research Institute for Atomic Reactors, Novovoronezh Power Plant, the BN-350 reactor and desalting plant at Shevchenko, the Obninsk Physics and Power Institute, and the Beloyarsk Power Plant. Some general discussions were also held with the staff at the Kurchatov Atomic Energy Institute, including a tour of the MR reactor, and with the U.S.S.R. State Committee.

Electrical and Nuclear Power in the U.S.S.R.

The utilization of nuclear power for producing electricity in the U.S.S.R. should be viewed within the overall framework of the entire U.S.S.R. electric-power program. In 1959, the U.S.S.R. State Production Committee for Power and Electrification predicted the production of electricity in accordance with Fig. II.1. Up to 1965, the prediction was met. Beyond 1965 to 1969, the actual production is shown by the dotted line.

Fig. II.1. Annual Production of Electricity in U.S.S.R.

TABLE II.1. U.S.S.R. Power-reactor Program in Operation by 1975

	MWe
1. Siberian Atomic Electric Station (Started 100 MWe in 1958) All are 100-MWe units, completed in 1964.	600 (Little known of this) (May be Pu producers)
2. PWR's	
Novovoronezh:	
December 1964	210
December 1969	365
1971	440
1973	440
1975	1000
Armenia	880 (two 440-MWe units)
Kola Peninsula	880 (two 440-MWe units)
Subtotal	4215
3. BWR	
Beloyarsk:	
April 1964	100
December 1967	260
	300
VK-50, Melekh	50
SM-1, Obninsk	5
Subtotal	355
4. Sodium-cooled Fast Breeders	
Melekh: BOR-60, Dec 1970	12
Shevchenko: BN-350, 1972	350
Beloyarsk: BN-600, 1973-75	600
Subtotal	962
5. Total, all reactors	6132

Dr. M. Sinev, Deputy Chairman of the State Committee, in 1964, at the Geneva Conference, predicted 900 to 1000×10^9 kWh in 1970 and 2700 - 3000×10^9 kWh in 1980. He then stated that atomic-energy stations would reach several thousand megawatts electrical by 1970, and several tens of thousands of megawatts electrical by 1980.

The following are in actual operation as of 1970:

PWR	575 MWe
BWR	355 MWe
Siberian Station	600 MWe
Total	1530 MWe

This total does not include portable reactors.

Table II.1 lists the reactors in operation and reactors known in construction stages. It can be assumed that the total will be on line around 1975. As of 1975, it is estimated for comparison purposes based on known data:

U.S.S.R.: 6,200 MWe in operation.

U.S.A.: 65,000 MWe in operation.

Committee members participating in the final discussions with the U.S.S.R. State Committee for Atomic Energy were I. I. Smolin (right), Deputy Chief, and D. D. Sokolov (left), Deputy Chief Engineer, Department of Foreign Relations. In the center is Mrs. D. F. Khoklova, of the Department of Foreign Relations.

This comparison is also reflected in plutonium production. By 1980, about 800 kg cumulative of fissionable plutonium will be available from the five Novovoronezh PWR reactors (2500 MWe). Between 7000 and 10,000 kg of plutonium may be available by 1980 from all U.S.S.R. reactors and from reactors sold to other countries. By 1980, the U.S. will have accumulated 100,000 kg of plutonium from light water reactors.

Estimates of nuclear-power-plant construction costs are as follows:

Beloyarsk Superheat Reactor No. 1: 100 MWe, 704 rubles (\$781)/kWe, including site and cooling lake.

Beloyarsk Superheat Reactor No. 2: 200 MWe, 224 rubles (\$249)/kWe.

Beloyarsk Fast Breeder Unit No. 3: 600 MWe, 225 rubles (\$250)/kWe.

Nuclear-power-plant energy costs are as follows:

Beloyarsk No. 2: 0.85 kopeck (0.94 cent)/kWh.

Beloyarsk No. 2 fuel-cycle costs = 0.60 kopeck (0.67 cent)/kWh.

Novovoronezh No. 1 PWR: initial cost in 1965 = 1.62 kopeck (1.80 cent)/kWh; No. 1 PWR, 1970 cost = 0.90 kopeck (1.00 cent)/kWh; No. 5 PWR cost = 0.60 kopeck (0.67 cent)/kWh.

The Soviets indicated that amortization (6-12%) is included in costs, but that the interest rate is not included.

Nuclear-power-plant labor requirements are as follows:

Beloyarsk No. 1 and 2 = 700 for operation and maintenance; Beloyarsk No. 3 = 450 for operation and maintenance; Total = 1150 for operation and maintenance of a plant of 900-MWe capacity.

Beloyarsk No. 3 = 200 for present construction force.

Beloyarsk No. 3 = 1150 for peak construction force in 1973-74.

BN-350 heat-transport complex

BN-600 heat-transport complex

III. SOVIET FAST-REACTOR TECHNOLOGY

A. PLANT DESIGN

1. Goals

The Soviets are celebrating their 50th anniversary of electric power. Their electrical demands have been increasing exponentially, approximately on an 8-yr doubling time. Accordingly, the U.S.S.R. is planning to develop fast reactors with a doubling time of 7-8 yr. They expect that a fair fraction of the power plants being built in the 1980's will be fast reactors. Since they plan to operate thermal reactors to a high burnup, a goal like 30,000 MWd/tonne, and since they have only a moderately scaled thermal program, they are not tying the pace of fast reactors to availability of plutonium from thermal reactors. They are satisfied to load their present fast reactors with ^{235}U and to limit the plutonium used to plutonium-fueled criticals. No overall criteria have been established. However, breeding ratio efficiency and plant cost are stated to be important. The final choice of plants and characteristics is the decision of the U.S.S.R. State Committee on Atomic Energy.

2. Status of Development

The Soviets now have two fast reactors in operation (BR-5 and BOR-60) and two (BN-350 and BN-600) under construction. BR-5 is a 5-MWt reactor which has been operating for 10 yr, with 25% availability to a fast fluence of 2×10^{22} nvt, and has provided valuable experience in components and system behavior as well as performance of oxide fuel. BOR-60 (60 MWt) has been operating since March 1970 at 20 MWt on forced-air heat exchangers, and is to be operated at 60 MWt with steam generators by the end of 1970. Power operation will include a core with some prototype fuel for BN-350 and BN-600 inserted in the reactor.

The preliminary design of BN-350 (1000 MWt, 150 MWe, and 200 MWe equivalent for desalinization) was completed in 1964. Construction is to be completed by the middle of 1971, and power operation is contemplated by the end of that year.

The preliminary design of BN-600 (600 MWe), except for the steam generators, is completed. This is the first tank-type reactor being built by the Soviets. Detailed design of the reactor area has *not* been completed. The schedule for completion of this plant is estimated as somewhere between 1973 and 1975. The civil works are about half constructed.

In view of the above the Soviets are ahead in the physical construction of their demonstration plants, notwithstanding their smaller operating experience.

The design of U.S.S.R. plants before that of BN-600 has been based on a maximum utilization of standard industrial (nonnuclear) design and manufacturing practices. These plants are of the pipe type, with components isolated one from the other, so that the various component industries can design and build components from performance specifications and a general space layout. The Soviets have designed BN-600 with only three loops (BN-350 having six)

BN-600 primary system

and have made other departures from their more conventional industrial practice. BN-600 is a more complicated plant. The radically different primary system will require considerably more design effort and manufacturing skill. This indicates they have made a major national commitment in developing the sodium-cooled fast breeder reactors and are prepared to pay the price for a major retooling for the fabrication of fast-reactor components on a commercial basis.

3. Design Practices

a. Introduction

In the fast-reactor field the Soviets have concentrated on sodium as the coolant, flow of sodium up through the core, an intermediate system, hexagonal fuel subassemblies, a layout that provides for natural-circulation cooling to take care of decay heat, single-wall steam generators, oxide fuel clad with stainless steel, refueling with a double rotating plug, an in-vessel fuel-handling machine, no failed fuel propagation (there are no provisions for such consequences as melt down of the core), and mill-type buildings. Some advanced work is being done on carbide and vented fuel.

b. Site Considerations

Many of the populated areas of the U.S.S.R. are far from large bodies of water, and there are not many large rivers in these areas, so cooling by rivers, lakes, or oceans is limited. Shevchenko, uses the Caspian Sea for cooling. At Beloyarsk a 2500-acre man-made lake will be used when the installed power rises above 1000 MWe. For rivers such as the Don, the problem of thermal effects has been recognized. The U.S.S.R. Bureau of Fisheries has limited the rise of temperature of the total river to 3°C in winter and 5°C in summer.

The Soviets locate their power plants in the general location of load demands and at the location of the coolant. They do not require an exclusion area around their reactors. Townsites are developed conveniently for personnel access to the industrial site or activity. Housing for atomic plant workers is usually located close to the plant.

c. Building Practices

The Soviets use more concrete and less steel in their building construction than is U.S. practice (see Figs. III.A.1 and III.A.2). This may be related to local material

Fig. III.A.1. Framework of Building for BN-600 Steam Generator.

Fig. III.A.2. Looking toward Roof of Building to House Unit No. 3 of PWR at Novovoronezh.

availability, type of labor supply, and standardization of constructional methods. The Novovoronezh Plant Unit 3, with a capacity of 440 MWe, used 60,000 m³ of concrete, 15,000 tons of total steel, with 4,000 tons of steel in the primary system. About 175,000 m² of steel has to be painted.

d. Plant Layout

The practice of isolating components and providing access room and headroom for ease of inspection, isolation, and maintenance is evident in all the Soviet fast breeder plants. Even BN-600 with its reactor tank and associated very compact primary system has a loosely coupled secondary system and power plant that is very large by U.S. practice.

The fact that a large-energy-release accident is not a design requirement simplifies the plant design. The elimination of a containment building, airlocks, and restraints on penetrations gives freedom in handling personnel, equipment, electrical leads, and layouts. The elimination of energy and missile absorbers greatly simplifies the design of the crowded area above the reactor.

The Soviet reactor tank concept is different from that used in other countries. Support of the tank and equipment in the tank is from the bottom (except for the rotating plugs). The pool and reactor vessel is maintained at the sodium inlet temperature. Differential vertical expansion of the IHXs and pump, and some lateral expansion is accommodated by vertical bellows located in the gas space at the top of the reactor compartment. These bellows are inspectable and removable.

All plants have primary-system flow on the outside of the tube bundle of the IHX. The piping layout has been handled in somewhat different ways in their three plants. BOR-60 is a two-loop piped system with eight-ply bellows in the secondary piping to take care of expansion. BN-350 is a six-loop piped system. BN-600 is a tank system with three loops and employs a tightly coupled primary system in which equipment is allowed to move.

e. Operating Conditions

The Soviets have chosen the same range of temperature and operating pressure as have other countries. They have designed BOR-60 for operation up to an outlet sodium temperature of 600°C (1112°F), are operating now at only 350°C (662°F), but intend to operate it over the next few years up to 500°C (932°F), a temperature they believe to be adequate for near-term objectives. The operating experi-

ence with BOR-60 will establish the operating conditions for BN-350 and -600.

The Soviets have designed BOR-60 for a 220°C (396°F) rise and their other plants to run at a 200°C (360°F) rise. However, these conditions will be checked out by operation of BOR-60. Although the Soviet transient specifications take into account a large thermal shock on load change and scram (see Part 2 of Appendix A), their operating practice is to minimize these transients, programming flow on scram to minimize shock. BOR-60 has encountered some thermal transient problems.

The major parameters for the U.S.S.R. reactors are given in Table III.A.1.

The Soviets use an electrically heated bismuth-tin seal in their rotating plugs and therefore design their primary system for essentially atmospheric pressure on the low-pressure side of the primary system. The Soviets use single-stage mechanical pumps, and they therefore design for only about 12 atm ΔP in the primary system.

The Soviets do not include many scram signals. Reactor shutdowns are programmed so that there are slow and fast trips, with most being slow to avoid excessive transients that could eventually compromise the integrity of the components. With a slow trip, the rate of temperature change of the primary-system piping and nozzles does not exceed 1°C/sec. For a fast trip, temperature changes are limited to 10°C/sec.

Turbine pressures are primarily dictated by thermal-cycle analysis, and steam pressures for Soviet plants correspond to American practice for sodium and sink temperatures. Reheat is used by Soviets in BN-600. System materials selected by Soviets correspond generally to U.S. practice. Stainless steel is used in the primary and unstabilized Croloy in the secondary system; Stellite is used for hard facing.

f. Fuel Handling

All Soviet designs use double rotating plugs, and a simple grapple for lifting the fuel with a shoe to prevent adjacent elements from lifting. In BOR-60, the fuel is removed directly through the plug, which is a feasible expedient for small subassemblies. In BN-350 and -600, the fuel is moved to decay-storage positions at the perimeter of the core (as is done in Fermi and EBR-II). In these reactors, the fuel is removed from the reactor by use of an inclined (17°) elevator (see Fig. A.3.12). Each component of the refueling system appears to be simple, reliable, and replaceable. However, there are many components in the scheme. The slit formed by the displacement of a vertical element

TABLE III.A.1. Soviet Fast-reactor Characteristics

	BR-5	BOR-60	BN-350	BN-600
Power output, thermal, MWt	5	20 ^a	1000	1470
Power output, electric, MWe	—	—	150	600
Date of power operation	1959	March 1970	Dec 1971 ^b	1973-1975 ^b
Operating conditions				
Inlet sodium temp, °C	400-430	360-480	300	380
Outlet sodium temp, °C	500	600	500	550-580
Steam pressure, atm	15.5	100	50	140
Steam temperature, °C	400	540	430	505-540
Plant characteristics				
Layout (pipe or tank)	pipe	pipe	pipe	tank
Number of primary loops	2	2	6	3
Primary system flow, gpm	~460	4000	~85,000	~95,000
Core and fuel characteristics				
Critical mass, kg	49.6 (PuO ₂)	176 (²³⁵ U) ^d	{ ~1200 (²³⁵ U) 850 (Pu)	{ 1700 (²³⁵ U) 1200 (²³⁹ Pu)
Size (L x D, in.)	11 x 11	15.75 x 15.9	32.8 x 41.8	80 x 29.6
Power density				
kW/ft max	5.5	16.7	13.4	16.1
kW/liter max	500	1100	840	
Peak flux, 10 ¹⁵ n/cm ² /sec	0.48	3.7	~10	
Design burnup, %	2 ^c	10	5	10
Fuel lifetime, days			200	

^aScheduled for 60-MWt operation in December 1970.^bPlanned date of operation.^c~5% burnup has been achieved.^dThis figure may be for a dry, clean core.

moving on an incline is kept narrow to eliminate neutron and gamma streaming. Test experience has been obtained on the operation of the inclined elevator. The Soviets are using two methods of cooling the fuel during removal from the primary system: forced and natural argon gas cooling.

Fuel handling from the reactor area exit port is somewhat different in the three plants. At BOR-60, the fuel cleaning is performed in a cask. BN-350 and -600 use a narrow refueling cell above the reactor floor, in which transfer of fuel to cleaning and storage takes place. A linear movement of fuel occurs at BN-350, whereas at BN-600 a rotary motion is used.

EBR-II fuel handling system

BN-350 refueling system

BN-600 refueling system

The Soviets are looking at two methods of cleaning the fuel. One is with nitrogen plus steam; the other is by lead displacement such as is used by the British. They have no actual experience with either at this time. The Soviets expect to have sufficient experience by the summer of 1971 to determine actual cleaning specifications.

g. Instrumentation and Control

The Soviets use dc driven mechanical pumps and therefore can vary flow over a wide range to minimize thermal transients. The Soviets vary in the degree they use computerized data-reduction equipment. Limited core-monitoring equipment is used. Even for the research reactor BOR-60, only in one-half of the subassemblies does each contain one nonreplaceable thermocouple. BN-600 uses only 20 outlet thermocouples. Propagation of fuel failure is not expected, and therefore extensive core instrumentation is not required.

The Soviets are developing instruments for the following applications:

Gross failed-fuel detection in sodium and in gas

Location of failed fuel by removing each subassembly and checking in special apparatus

Subcriticality measurement

Small thermocouples (seen in their heat-transfer stands at Obninsk)

High-temperature strain gauge

Sodium level and pressure measurement by gross flow instrumentation

Cable and connectors

Flux and cross-section determinations

Sodium technology instrumentation (see Sect. III.F)

h. Radioactivity Tolerance Levels

The basic standards for protection of public health and for facility design embody limits on the maximum permissible concentrations of radioactivity in the air and water. These limits are established by the Soviet Ministry of Health and are obligatory for all States of the Soviet Republic. Such standards form the fundamental criteria for process design of the reactor facilities. The limits are discussed on pages 94 and 98.

The Soviets stressed the point that their effluent releases from normal operation of their facilities are kept well below specified maximum allowable limits at the point of release. Effects that might be attributed to either atmospheric or water dilution were not credited in establishing operating effluent limits. There was no way of confirming such a practice, but their expressed interest in control of effluents and the evidence of a considerable amount of radiation monitoring in the plant access areas indicated that such matters are given considerable emphasis by the Soviets.

Plants having considerable amounts of wastes, such as the thermal-reactor facilities at Novovoronezh or Beloyarsk, have waste-burial facilities on site. Research institutions were reported to be sending their solid and liquid wastes to "specially assigned" burial centers.

4. Management Practices

Very little is known as to the detailed organization of the nuclear power industry in the U.S.S.R. Above all it is a centralized system stemming from the Council of Ministers—the central body which acts as the government of the U.S.S.R. This council includes in part: (1) all ministers, (2) all chairmen of state committees, (3) all chairmen of all republic councils of ministers. The Ministry for Power and Electrification, the Ministry for Heavy Industry, and the State Committee for Atomic Energy are included.

Figure III.A.3 is an oversimplification of the organizations involved in the design, construction, operation, and maintenance of nuclear power plants and can serve merely as an orientation as to possible directions. Omitted are tie-ins (approval and advisory) with the Ministry of Health, State Planning Committee (Gosplan), Academy of Sciences, and State Committee for Science and Technology and other organizations involved.

Power plants, other than those within the jurisdiction of the Research Institutes (BOR-60 at Melekhess, VK-50 at Melekhess, SM-1 at Obninsk), are under the jurisdiction of the Ministry of Power and Electrification. This ministry appoints the director of the power plant project who has

Fig. III.A.3. Simplified Organization of a U.S.S.R. Atomic Power Plant Project.

complete control of the construction, operation and maintenance of the power plant.

A laboratory makes a conceptual design (Melekes, Obninsk, etc.) which is forwarded to a Design Institute for detailing and preparation of specifications for fabrication. The Ministry of Heavy Industry procures the components from the "factory," which prepares working and shop drawings. Component drawings are delivered to the con-

struction site with the components. Quality assurance is performed by a branch of the State Committee on Atomic Energy. Welding procedures are prepared by the Welding Institute, but are not obligatory. The Director of the site at which the plant is being built is responsible for construction, personnel training and qualification, and subsequent operation. He is empowered to make field changes on the spot. The Institute performing the conceptual design visits the Design Institute and "factories" periodically for review. The construction-site Director relies primarily on drawings

and components furnished and approved by others. In the case of faulty design or fabrication, his avenue of relief is to the State Committee, although he may seek advice from the appropriate institute.

The U.S. delegates did not visit the Design Institutes. How well the plants were designed could not be established.

These things can be said about design philosophy:

U.S. designs have been carefully reviewed and extensively applied.

In many cases, design requirements are limited to likely events or malfunctions and unlikely events are not designed for. For instance, the steam generators have been designed for only one tube failure. Isolation of steam generators is used to handle an unlikely event greater than this.

It appears that the plant is designed in two places, the conventional part at the All Union Design Institute and the nuclear portion at the Design Institute of the State Committee for Atomic Energy, but this is an assumption and may not be a reality.

Model and proof testing of components is accomplished at the production factories which are set up with research and development facilities for such purposes.

B. CORE DESIGN

Both BN-350 and BN-600 will be started up with ^{235}U enrichment and will be converted later to plutonium-fueled cores with mixed oxide. On the present design, the breeding ratio for a mixed-oxide BN-600 core has been calculated as 1.37. The breeding ratio of the initial ^{235}U -fueled core will be 1.15.

Although swelling has been recently observed in material irradiations in BR-5, no allowance is made for the swelling of cladding or structural materials. The designs of BOR-60 and BN-350 are frozen, but there is time during the next 2 years to change the core design in BN-600. The size of the core vessel has, however, been fixed in BN-600.

Very tight cores

The cores are very tight both within and between fuel assemblies (P/D 1.16 for BN-600). There is only 2 mm (79 mils) clearance between the 140-in.-long hexagonal subassemblies in BN-350 with 1.5-mm (59-mil) pads at the corners. No lateral restraint is provided for the core, except that inherent in the radial blanket and shield. The pressure drop across the larger core is 70 to 80 psi with the wire-wrap spacers. The delta temperature across the core is 170°C (306°F) for BN-600. The plenum design provides for radial entry of coolant through side holes in the lower part of the subassembly, providing a hydraulic holddown.

Minimum shielding is used in the cores, leading to high fluences on the core-support plate and the vessel (e.g., 5×10^{22} nvt for BN-350 support plate after 20 years). The BN-350 core-support plate is also not removable. In BN-600, the fluence on the reactor vessel is low; there is, however, a high fluence ($>10^{22}$) on the basket separating the flowing sodium from the tank sodium.

In BOR-60, approximately 50% of the fuel assemblies have nonreplaceable thermocouples, and one subassembly can be provided with in-core instrumentation. There is less instrumentation of individual assemblies in BN-350 and BN-600 than in BOR-60.

In the current design, one-sixth of the BN-350 core will be refueled every 2 months and one-fifth of the BN-600 every 5 months. The tight lattice design combined with a short refueling interval results in a high breeding ratio and

low fissile inventory. However, the 2-month refueling interval for BN-350 (based on earlier considerations) is now recognized to be too short. Design modifications, including increasing the compensator control strength, are under consideration to permit longer refueling intervals for later cores.

The core and blanket design of the BN-350, in which the upper and lower axial blanket pins are separated from the core fuel pins, limits the burnup on their calculations to about 5%, because of the lack of space in the core fuel pin for an adequate fission-gas plenum. For later cores, this limitation will be removed by using fuel pins similar to the BN-600, with core and axial blanket pellets in the same can and a plenum (70-cm length for BN-600) below the lower axial blanket. Details of the fuel, fuel pin, and subassembly parameters are given in Table III.E.1.

The core-design differences of BN-600 (compared with BN-350) are

1. Flatter core (H/D ratio 1/3, as compared with 7/10)
2. Six safety rods, as compared with three
3. Nineteen compensating rods, as compared with six
4. BN-600 fuel pin designed for 10% burnup with large gas plenum and thicker cladding.

The only equipment and techniques developed for location of failed fuel for any of the breeder reactors involve removal and external heating of each fuel assembly. On the basis of BR-5 operating experience with failed fuel, the BOR-60 circuit is designed to operate with 100 failed fuel pins. Testing of vented fuel is one of the main objectives in the early program on BOR-60.

C. NUCLEAR SAFETY

The reactor-safety philosophy being followed is to:

1. Concentrate on prevention of accidents through reliability and operability of components and systems.
2. Use backup safety methods deemed to be sufficiently dependable that they result in a clear-cut safety advantage.
3. Omit intricate design backups for accidents determined to be virtually impossible. In this category are propagation of coolant flow loss beyond a single fuel assembly or progression of small accidents into large ones with bulk fuel meltdown and redistribution and explosive energy release (design basis accident in the U.S. sense).

Based on the general safety philosophy indicated above, Soviet fast reactors use engineered safety features of which the following are examples:

1. The primary system in BN-350 is double-contained up to stop valves in hermetically sealed tanks or pipe jackets in order to catch leaking sodium and thereby prevent loss of core cooling in the event of a vessel or pipe rupture. The BN-600 reactor tank is surrounded by a safety vessel with clearance adjusted so that, in the event of a leak in the reactor tank, the sodium level cannot drop below the outlet nozzle above the core.

2. The BN-350 reactor uses three safety control rods, each having a system of detectors and electronic circuits independent of the other two. This allows for one inoperable control safety system, since two such rods would be sufficient to take the reactor from full to zero power in an emergency and to allow the reactor to be taken slowly to the refueling condition.

3. Interlocks are provided to prevent occurrence of accidents from operator errors—such as turning on a primary pump during refueling (when control rods are in the core and disconnected from their drives) which could lift the control rods from the core.

4. Subcriticality is monitored during refueling using a neutron source and a detector in the pulse mode.

5. The inlet plenum design of the core support plate and fuel assemblies, providing for radial entry of coolant into several small openings, provides hydraulic holddown of the fuel assemblies and minimizes the possibility of a total blockage of coolant flow to the assembly.

6. An independent source of energy, batteries with 10-hr capacity for BOR-60, and diesel generator sets are provided to prevent loss of site power.

The Soviets have evaluated propagation-failure type accidents rather extensively and have taken sufficient measures in design of the inlet plenum to prevent coolant loss to fuel assemblies. The experimental evidence obtained from the BR-5 experience (in which fuel pins failed in 19 assemblies with no more than two failed pins in any one assembly) and the similitude evaluation of steam-generator leak tests, coupled with analysis conducted, indicate that propagation is highly unlikely.

A loss of power to the primary sodium pump is considered the most serious design basis accident, and strong steps are taken, in this event, to ensure scram and to ensure that sodium flow continues. These steps include system design for natural circulation.

The determination that there will be no design basis accident in the U.S. sense, involving possible large explosive energy release or bulk meltdown and redistribution of fuel, is based on the engineered safeguards which prevent small incidents from occurring and, allowing that some such incidents may occur, prevent them from becoming large accidents.

The extent to which the Soviets conduct analyses to determine potential explosive energy releases from unlikely accident events—or use a fault-tree approach in which probabilities are assigned to identified accident sequences—was not determined from the discussions on safety. The Soviets indicated that at least some accident analyses are performed and the results sometimes used for design changes—but such changes are not made for “impossible” accidents.

Meltdown structures below the core are not provided, because a total loss of coolant, which would result in a molten core, is not considered credible. Based on the assumption that propagation will not occur and that there will be no design basis accident in the U.S. sense, the reactors are not designed with plug catchers, blast shields, or containment for large explosive energy release. Fuel assemblies are not highly instrumented. (For example, only half of the BOR-60 assemblies have an outlet thermocouple—one for each such assembly—and the thermocouple is neither replaceable nor in the safety circuit. Large control-rod reactivity worths, in excess of 1% delta k for a single rod, are not considered unsafe, because a rapid ejection of a control rod is considered incredible, except possibly during refueling in the event of a pump being turned on. This is prevented by an interlock on the pump, as discussed above. There is no specific requirement on the magnitude of the Doppler coefficient.

Steel buildings are not used for reactor containment now because the Soviets feel that they do not offer an assured dependability for prevention of radioactivity release to the environment. Use of steel containment buildings for the future will be periodically reviewed.

The Soviets build their reactors at selected sites free from earthquakes. Apparently special precautions are taken for earthquake effects.

Present reactor sites are located within a few kilometers of populated areas as indicated by the following examples.

1. Beloyarsk (BN-600 and integral superheaters): 1½ to 2 miles from 13,000 population.
2. Shevchenko (BN-350): 3 to 4 miles from 60,000 population (slated to grow to 250,000).
3. Melekhess (BOR-60 and VK-50): 3 miles from the large population of Melekhess (>200,000).
4. Novovoronezh (five PWR plants by 1975): 2 miles from Novovoronezh, 25 miles from the 500,000 population of Voronezh.

Investigations to improve physical data on fuel and understanding of heat-transfer processes pertinent to accident analysis are being conducted at Obninsk, but the main purpose of such work is to improve reactor performance under normal operating conditions. Development work is also being carried out on statistical methods—noise analysis—for monitoring reactor disturbances not ordinarily picked up by control instruments, that may provide early warning of impending safety-system failures.*

In summary, Soviet safety practice is focused on reliability and operability of components and systems through adequate design margins and engineered safeguard features which prevent small accidents from occurring and, for cases considered to be reasonable, from developing into serious accidents. Backup design features are provided if considered necessary and dependable, otherwise omitted as introducing unnecessary complexity, which tend to detract from safety.

*A. I. Mogilner, “Statistical Methods of Monitoring the Technological Safety of Power Reactors,” JPRS-48331, *CEMA Symposium on Atomic Power Stations with Fast Reactors* (July 1969). See also *Soviet Atomic Energy*, Vol. 26, No. 6, p. 569 (June 1969).

D. COMPONENTS

1. General

Components were largely similar in design to those in the U.S. and other countries. Where deviations were noted, they were in the direction of fabrication simplification. Materials are similar to those in non-U.S.S.R. plants, with only minor changes in alloy composition. Where possible, design and fabrication paralleled nonnuclear practice as closely as possible. The component design for BOR-60 and BN-600 appeared to follow the same general design philosophy, with allowances for size and the fact that one is a piped and one a pot type system. However, the heat-transfer system components for BN-350 were quite different in design approach and bore little resemblance to those of the other two reactors.

2. Pumps

The sodium pumps for BOR-60 and BN-600 main circulating systems are similar to U.S. (Fermi) designs,

employing sodium-lubricated hydrostatic bearings at the impeller end, and oil-lubricated radial and thrust bearings outside the sodium and sodium vapor space. Rotating gas seals to retain the inert-gas blanket over the sodium pool appear to parallel U.S. mechanical design, with graphite stators running against a nitrided steel shaft journal. The seals are lubricated by hydrocarbon oil with provisions for preventing oil leakage into the sodium. As pump capacity is increased, as in BN-600 [8000 tonnes/hr (32,000 gpm), 7 atm (103 psi) ΔP], double-suction, single-stage configuration is planned. The BN-350 pump design (14,000 gpm) is quite different, having an overhung impeller with very complex shaft bearing and cooling arrangements, and a short shaft compared to U.S. practice. The pump arrangement in this piped system indicates that the primary pump drive will be placed under a biological shield in normal operation, with internal shielding to protect against radiation from residual (or radioactively decayed) sodium after loop shutdown, when contact repair or removal activities would be performed. The internal shield would, of course, reduce the radiation dose to the electrical drive equipment and lubrication systems.

Overhung shaft, NaK freeze seal BN-350 pump

Double-inlet Fermi-type BN-600 pump

3. Stop Valves

Relatively small valves (up to 300 mm) were globe-type, single-bellows sealed. The shaft bellows seal was backed up by graphite stem packing; no bellows leak detectors were noticed or described. Larger valves, as in the 600-mm (24 in.) BN-350 main system, were double-disc wedge gates, with provisions for draining between the discs. These larger valves employed *water-cooled* stem freeze seals, backed up by mechanical packing in both primary and secondary systems. Water cooling was used if there were two independent barriers between water and sodium, implying a cooling-coil type arrangement around the valve stem and support. Stellite valve trim was used, as is U.S. practice. A common feature of all valves observed or described was the bulkiness of the valve body in contact with sodium. All valves up to 600 mm (24 in.) (the largest used) had forged bodies with no apparent effort to reduce the mass of body metal. BOR-60 valve bodies (200 mm, 7.9 in.) were bored-out rectangular forgings; BN-350 valve bodies (600 mm, 24 in.) were bored-out forgings roughly cylindrical, forged down to about 1-in. flats on the outside. Seat leakage criteria were reasonable, up to 200 g/hr (0.44 lb/hr) in BN-350, and up to 15 liters/hr (about 33 lb/hr) in BOR-60. (BN-600 has no primary or secondary stop valves in the current design.)

Check valves, easily removable

4. Check Valves

Check-valve internals were removable in the 200- and 600-mm (8 and 24 in.) sizes, which were installed in the piped systems. These generally followed U.S. practice in hydraulic damping of valve-disc closing, pin and bearing materials, and provision for low-pressure drop during natural convection. Though no check valves were observed, the sketches presented indicated a simple and straightforward approach to their construction.

Only an artist's conception of the BN-600 check valves was available; they are installed inside the pump casing, below the pump impeller in the discharge. Though the detail design was not available, it is considered feasible to perform the design so as to make the check valves removable.

5. Intermediate Heat Exchangers

With the exception of BN-350, intermediate-heat-exchanger design appeared to follow conventional practice with tube bundles removable through the head and sodium introduced into the bottom of the tube bundle through a downcomer. The BN-350 design is unique, in that a large flat-sided shell, reinforced by an external framework, is used, with three tube bundles in series, each removable from the top after disconnecting interconnecting piping. This design has been reported previously* and has been carried through production apparently in parallel with the more conventional BOR-60 installation and planned BN-600 design.

6. Steam Generators

Two different types of steam generators were displayed—the once-through “serpentine” type used in BOR-60 and planned for BN-600, and the bayonet-tube natural-circulation type observed in BN-350. In the once-through steam generators, provision for tube plugging was made, but tube bundles were not removable. Attention had been paid to minimizing thermal shock on tube sheets in contact with sodium in both designs. Tube-joint attachments were of the front-face-welded, trepanned tubesheet type similar to U.S. practice of a number of years ago. Crevice-free joints were not displayed, nor was there discussion of their development. Steam-generator testing was not performed at any of the laboratories visited; such testing, which has been previously reported on 3-MWt

*Leipunsky, A. I., et al., “Sodium Technology and Equipment in the BN-350 Installation,” *Proceedings of the London Conference on Fast Breeder Reactors Organized by the British Nuclear Energy Society, May 17-19, 1966*, Pergamon Press, Inc., New York (1967), p. 797.

BN-350 bayonet steam generator

models, was done at the factory producing the steam generators and responsible for their detailed design.

Sodium-water reaction testing has been performed, indicating that the steam-generator design could withstand the complete rupture of one tube. Further sodium-water reaction work would not include additional "large-leak" testing, but would include small-leak and material-wastage experiments on a wider variety of materials than stainless steel, which had been previously tested. The experimental work in sodium-water reactions performed to date had resulted in a reaction product relief system involving a rupture disc on the sodium expansion tank (not on the steam generator itself), punctured by an electrically operated knife, which received its actuating signal from pressure transducers in the steam generator and superheater.

7. Piping

Sodium piping was 18 wt % chrome, 9 wt % nickel composition. Such piping as was observed was mounted on supporting saddles, rather than being hung from spring hangers. No final conclusion should be drawn from this, however, as piping runs were incomplete in the BOR-60 steam-generator installation and barely started in BN-350.

A major innovation was the use of bellows in the BOR-60 200-mm (8-in.)-diam secondary vertical runs of hot-leg piping. This permitted a compact piping arrangement in the steam-generator rooms, as observed on a model. The bellows were formed "U-span" type, of eight plies of 0.35-mm-thick material, supported by rings in each convolution.

Bellows are not being used for BN-350 or BN-600 piping exposed to sodium.

Thermal insulation for piping was stated to be primarily SiO_2 in fiberglass form. Electric resistance heaters were used for preheating piping, although reactor vessels were heated by hot gas.

Primary piping on BOR-60 and BN-350 was doubly contained out to the stop valves, with bellows accommodating differential expansion of the outer pipe.

E. FUELS AND MATERIALS

1. General

Soviet development of fast-reactor fuels has emphasized fabrication and evaluation of enriched-uranium oxide or carbide. Irradiation of a uranium carbide core continues in BR-5. Uranium oxide will be used for initial loadings of

BOR-60, BN-350, and BN-600. (Pu,U)O₂ assemblies have been tested on a relatively small scale in BR-5 and in loops of thermal test reactors SM-2.*

This approach strongly affects most aspects of U.S.S.R. fuel-development programs and complicates comparison of Soviet progress with that of other countries: (1) fabrication technology is still centered about uranium fuels; (2) fundamental information being developed in many studies is for uranium oxide and will be different for mixed oxide; (3) thermal recycle of plutonium is not planned, and results of larger-scale, fast-flux tests of mixed oxides will not be available for several years; and (4) reprocessing of fuel will involve existing plants for some time to come.

Fuels and materials parameters selected for the fast reactors are compiled in Table III.E.1. Highlights of the BOR-60 and BN-350 fuel systems are indicated in Figs. III.E.1 and III.E.2. Other details of fuel designs are given in Appendix A for the specific reactors.

2. Cladding and Structural Materials

The Soviet approach to cladding development has been summarized in such recent reports as *CEMA Symposium on*

TABLE III.E.1. Parameters for Four U.S.S.R. Fast Reactors

		BR-5 UC	BOR-60 UO ₂	BN-350 UO ₂	BN-600 UO ₂
Fuel-subassembly Design Parameters					
Pellet					
Pu(U + Pu)	%	0	0	0 (1st core)	0 (1st core)
²³⁵ U/U Total	%	90	90	90	-
OD	mm	7.65	5.1 ± 0.1 ^a	4.82 ^b	5.9 (+0.0, -0.2)
	in.	0.30	0.202	0.190	0.23
ID	mm	0	1.7	Annular	1.8
	in.	-	0.068	-	0.071
Length	mm	-	10	-	-
	in.	-	0.394	-	-
Diametral Gap	mm	0.10-0.45	-0.2	0.6 ^b	0.15-0.5
	in.	0.004-0.018	-0.008	0.024	0.004-0.020
Pellet End Shape		-	Flat	-	Flat
Pellet Density	g/cm ³	12	10.2 ± 0.2	10.5	10.4
	% TD	88.0 ^b	93.4 ± 1.8 ^b	95.8 ^b	95.2 ^b
Smear Density	g/cm ³	10.7-11.7 ^b	9.1-9.3 ^b	8	-
	% TD	78.5-85.6 ^b	83.1-84.8 ^b	73.0 ^b	70.8-83.1 ^b
O/M		5.2-5.3% ^c	-	-	-
Goal Burnup	%	2	10	5	10
Cladding					
Material	-	321 and 316	Stainless	Stainless	Stainless
OD	mm	8.55	6.1	6.1	6.9
	in.	0.347	0.240	0.240	0.272
ID	mm	7.75	5.3	5.4	5.9
	in.	0.305	0.209	0.213	0.232
Thickness	mm	0.4	0.4	0.35	0.5
	in.	0.016	0.016	0.014	0.020
Overall Pin Length	mm	500	-	-	-
	in.	19.7	-	-	-
Gas-plenum Length	mm	50	150	-	800
	in.	1.97	5.9	-	31.5
Peak Clad Temp	°C	580	-	-	-
Hot-spot Clad Temp	°C	-	800	-	690
Spacers					
Bottom Support	-	Grid	Grid	-	-
Spiral Spacer	mm	Wire	0.4 x 0.7 Oval Wire	Wire on Alter- nate Pins	Round Wire
Top Support	-	Grid	None	-	-
Triangular Pitch	mm	9.05	6.7	7.0	8.0 ^b
	in.	0.357	0.264	0.276	0.305
Clearance between Pins	mm	0.5	0.6	0.85	1.1 ^b
	in.	0.020	0.024	0.033	0.043
Pitch/Diameter Ratio	-	1.06	1.1	1.15	1.16
Spiral Spacer Material	-	Stainless	-	-	-
Spiral Axial Pitch	mm	-	-	100	-
	in.	-	-	3.94	-

*In about two years, mixed oxide will be included in the next loading of the BR-5, after the reactor is modified for higher power.

FUELS AND MATERIALS

TABLE III.E.1 (Contd.)

		BR-5 UC	BOR-60 UO ₂	BN-350 UO ₂	BN-600 UO ₂
<u>Core Design Parameters</u>					
<u>Subassembly</u>					
OD across Flats	mm	26.1	44	96	96
	in.	1.03	1.73	3.78	3.78
Thickness	mm	0.3	1	2	-
	in.	0.012	0.040	0.079	-
Triangular Pitch	mm	-	45	98	98
	in.	-	1.77	3.86	3.86
Overall Length	mm	833	1495	3500	-
	in.	32.8	59	138	-
No. in Inner Zone	-	} 100	} 80	109	} 397
No. in Outer Zone	-			90	
No. in Radial Blanket	-	-	~172	440	-
No. of Control Subassemblies	-	-	7	12	27 ^c
Hex Can Material	-	-	Stainless	Stainless	Stainless
Pins/Subassembly	-	7	37	169	127
<u>Coolant</u>					
Type	-	Sodium	Sodium	Sodium	Sodium
Maximum Velocity	m/sec	-	~8	~8	-
Flow Direction	-	Up	Up	Up	Up
Inlet Temperature	°C	-	360-480	300	380
Bulk Outlet Temperature	°C	-	600	500	550
<u>Power</u>					
Total Thermal	MWt	5	60	1000	1470
Total Electric	MWe	-	-	150 ^d	600
Thermal in Core	MWt	-	-	890	-
Peak	kW/l	285	1100	840	840
Average	kW/l	210	-	480	550
Peak	W/cm	300-360	550	440	530
	kW/ft	9.1-11	16.7	13.4	16.1
Hot Spot	W/cm	-	-	500	690
	kW/ft	-	-	15.2	21
<u>Core</u>					
Diameter	mm	-	404	1495	2048
	in.	-	15.9	58.9	80.7
Radial Blanket Thickness	mm	-	-	600	-
	in.	-	-	23.6	-
Peak Flux, Total	sec	0.48	3.7	~10	-
10 ¹⁵ n/cm ²					
Active Core Height	mm	380	400	1060	700
	in.	15.0	15.7	41.8	27.6
<u>Radial Blanket Subassembly</u>					
<u>Pellet</u>					
²³⁸ U/U Total	-	-	100	100	100
<u>Cladding</u>					
OD	mm	-	14.5	14.2	14.2
	in.	-	0.571	0.560	0.560
ID	mm	-	-	-	13.4
Thickness	mm	-	-	0.5	0.4
	in.	-	-	0.020	0.015
Blanket Column Length	mm	-	900	-	-
	in.	-	39	-	-
<u>Spacers</u>					
Triangular Pitch	mm	-	15.2	14.8	-
	in.	-	0.600	0.583	-
Clearance between Pins	mm	-	0.7	0.6	-
	in.	-	0.028	0.024	-

^aThis tolerance inconsistent with unground pellets.

^bCalculated value.

^cControl and safety.

^dAn additional amount is used for the desalination plant associated with BN-350.

Fig. III.E.1. BN-350 Fuel System.

Fig. III.E.2. BOR-60 Fuel System.

Atomic Power Stations with Fast Reactors, Vols. I and II (July 1, 1969), JPRS 48330, and *Proceedings of the IAEA Symposium on Irradiation Damage to Reactor Materials*, Vienna, Austria, June 2-6, 1969.

In discussions with this U.S. Delegation it was revealed that U.S.S.R. alloy development has included determination of some performance limits in a fast flux, using a large number of fuel pins. As described elsewhere, failures occurred in many (38) carbide pins clad in Type 321 stainless steel and irradiated in BR-5. Failures did not occur with pins clad in "Russian Type 316" (16 Cr, 15 Ni, 3 Mo, 0.8 Nb). Selection of this alloy composition for breeder reactor cladding was on the basis of 15% greater time to rupture than for AISI Type 316 at high temperatures out-of-reactor. The corrosion behavior and the ductility of the "Russian Type 316" are almost the same as for AISI Type 316. Examination of some "Russian Type 316" cladding after irradiation to 2×10^{22} nvt has shown the onset of swelling (0.5%). Both the Obninsk and Melekh programs are now being reoriented to this swelling problem. The desirability of adding 0.005% boron to the "Russian Type 316" cladding has been investigated, but apparently is still uncertain. Boron may provide further improvement in high-temperature properties of the "Russian Type 316."

3. Fabrication

The PuO_2 fuel for the first BR-5 loading has been the only significant quantity of plutonium fuel produced. Essentially all physics work, such as that in BFS-2, is with enriched uranium. Major questions asked of U.S. Delegates seemed to center about relatively simple problems such as optimum binder or lubricant and whether it is possible to eliminate water in the powder processing.

The next BR-5 core, for operation at 10 MWt, will include 60 kg of $(\text{Pu,U})\text{O}_2$ pellets operating with a power of 900 kW/liter of core and maximum flux of 1.5×10^{15} nv.

The first BN-350 core, containing annular pellets of 90% ^{235}U -enriched UO_2 , is expected from the factories early in 1971. Fuel-design details are provided in tables and figures in Appendix A.3; further fabrication information was not made available.

Trends affecting fuel fabrication include:

- Adoption of annular pellets.
- Use of low-density fuel [as low as 71% smear density (nominal 80%) calculated] for accommodation of fuel swelling.
- Large pellet-cladding gaps, several-fold greater than commonly used in the U.S.
- Nonvented fuel, contrary to some reports.
- No dishing or grinding of pellets.
- Unbonded wire spacers on fuel pins.
- Coprecipitation of mixed oxide will be used, rather than mechanical blending of PuO_2 with UO_2 .

There has been an interesting range of densities considered for oxide fuels, as indicated in Fig. III.E.3.

Details of carbide-fuel fabrication were not revealed, except for that information shown in Table III.E.1. There was no indication of plans for fabrication of future carbide-fuel loadings in spite of the promising BR-5 performance of fuel clad in Russian Type 316 steel.

Fuel-pin closures are by tungsten-inert gas welding. Closure defects were reported to have caused failures in breeder reactor fuel as well as in thermal reactors.

Wire spacers are attached to the side of the end cap by a tungsten-inert gas spot weld.

Many of the fuel-design changes anticipated for BN-600, compared to BN-350 (see Table III.E.1) will affect fuel fabrication. These include:

- Increased pellet diameter (5.9 vs. 4.8 mm).
- Increased cladding-wall thickness (0.5 vs. 0.35 mm).
- Longer gas plenum (700 mm vs. ?).
- Diametral fuel-cladding gaps of 0.15 to 0.5 mm, vs. 0.6 mm, calculated.

It was stated that the possibility of using vented fuel is being given "great attention," but design or fabrication details were not made available.

Coolant channels are now fabricated with corners which

Fig. III.E.3. Trends in Oxide-fuel Density.

are relatively sharp compared to what was originally reported to be acceptable.

4. Irradiation Testing

BR-5 has been the principal U.S.S.R. fast-flux testing facility. It will be shut down in mid-1971 for about twelve months to permit operation at 10MWt. The loops in thermal-reactor facilities SM-2 and MR also are used to support LMFBR programs, but surprisingly little mention of them was made during this visit.

Fuel-coolant reactions, swelling of failed fuel, and release of activity are still of concern, but they are not considered serious problems because of the results of lengthy operation of BR-5 with failed fuel. Some of the fast-flux test information is of limited usefulness to design and operation of other reactors with originally conceived types of fuel, irradiation conditions, and instrumentation, as indicated in Fig. III.E.4.

The "No. 1 Fuel Problem" to several Soviet experts is the interaction of cladding with fission products at high exposures and temperatures. Other experts indicated that swelling of cladding is an equally important problem. Past efforts have concentrated on swelling of the fuel material.

A uranium carbide core has operated in BR-5 since May 1965, to a peak of 5.2% burnup. Comparison of the

Fig. III.E.4. Comparison of Temperature and Fluence in the Most Highly Irradiated Core Components for Swelling Specimens.

performance of the carbide with oxide fuel is not considered significant because the power ratings were so low; the operating conditions are indicated in Table III.E.1 and Fig. III.E.5. Uranium carbide (5.2-5.3% carbon) clad in the Russian Type 316 did not fail. Thirty-eight of 53 sub-assemblies clad in Type 321 steel failed between 3.6 and 5.2% burnup.

Examination of nineteen BR-5 oxide-fuel assemblies for which failures were previously reported has revealed only 1-2 defective pins per subassembly which, as for the carbide failures, is considered important evidence that propagation of fuel failure will not be a serious problem.

Other noteworthy irradiation-testing developments have included:

- Irradiation of an oxide-fuel pin to 18% burnup of the heavy atoms, under conditions not reported in detail. The maximum burnup for plutonium-bearing oxide (PuO_2) in BR-5 continues to be 6-7%.
- Intense effort to complete a thermocoupled sub-assembly for evaluation in BOR-60, possibly in September 1970.
- Evaluation in BOR-60 and other facilities of low-smear density, annular pellets to reduce effects of fuel swelling.
- Use of oval wire spacers in the BOR-60 loadings.

1. Operation began Jan 1959 (Ref. 2, p. 6).
2. Oct 1960, Burnup = 2 a/o, Clad Rupture indicated (Ref. 2, p. 8).
3. Feb. 22, 1961, Burnup = 3.2 a/o (Ref. 2, p. 10).
4. April 29, 1961, Burnup = 3.9 a/o (Ref. 2, p. 11).
5. June 17, 1961, Burnup = 4.55 a/o (Ref. 2, p. 11).
6. Steepest slope on burnup-time curve, assumed to correspond to full power operation of reactor.
7. Sept. 1, 1961, Reactor shutdown, Burnup = 4.85 a/o (Ref. 3, p. 171).
8. Nov 1964, Shutdown to load carbide core, Burnup = 6.5, 1.4, and 0.8 a/o (Ref. 3, p. 171).
9. May 1965, Started up on carbide core (Ref. 1, p. 243).
10. June 25, 1965, Reached full power operation (Ref. 4, p. 185).
11. Burnup = 6.7 a/o (Ref. 1, p. 243).
12. Oct. 1, 1967, Burnup = 3.7 and 2.8 a/o in UC and UO₂ (Ref. 1, p. 243).
13. March 1, 1967, Burnup = 2 a/o (Ref. 4, p. 1270).
14. 81 PuO₂ Fuel Assemblies (Ref. 1, p. 243).
15. Leaks in 18 Subassemblies (Ref. 1, p. 243).
16. Shutdown Sept 1961; restarted March 1962 (Ref. 1, p. 240).
17. Restarted with 63 PuO₂ Fuel Assemblies (Ref. 1, p. 243).
18. Restarted with 50 UO₂ Subassemblies (may be error) (Ref. 5, p. 4).
19. Unloaded 59 Pu and 21 U Subassemblies from core (Ref. 3, p. 175).
20. 17 PuO₂ Subassemblies had leaks (Ref. 1, p. 243).
21. Shutdown Nov 1964, restarted May 1965 (Ref. 3, p. 71-72).
22. Four PuO₂ Subassemblies reloaded in core (Ref. 1, p. 243).
23. Two leaky PuO₂ Subassemblies (Ref. 1, p. 243).
24. Two PuO₂ and two UO₂ Subassemblies reloaded (Ref. 3, p. 182).
25. Coolant outlet from 450°C to 500°C, Dec 1960 (Ref. 4, p. 1272).
26. Temp conditions (Table 3, Ref. 1).
27. Power conditions (Table 2, Ref. 1).
28. Calculated power level = 20% during March 1962-Nov 1964 based upon reported burnup values.
29. BU = 3.6 a/o, March 1968 (Ref. 6).
30. BU = 4.5 a/o max, 1969 (Ref. 7).
31. BU = 5.2 a/o max, Early 1970 (Ref. 7).
32. 22 Subassemblies; 321 Subassemblies had leaks (Ref. 7).
33. 16 Subassemblies; 321 Subassemblies had leaks (Ref. 7).

References:

1. Leipunsky, A. I., *et al.*, "Results of Eight Years Operation of the BR-5," *CEMA Symposium*, Obninsk, Dec 1967, Translated in JPRS-48330, pp. 238-251.
2. Leipunsky, A. I., *et al.*, "Operational Experience with the BR-5 Reactor," *IAEA Conf. on Operational Experience with Atomic Power Reactors*, Vienna, June 1963, Translated in AEC-TR-5890.
3. Leipunsky, A. I., *et al.*, "Experience Gained from the Operation of the BR-5 Reactor 1964-5," *BNES Conf.*, London, May 1966, Paper 2/3.
4. Leipunsky, A. I., *et al.*, *Operating Experience with the BR-5 Reactor over the 1959-1966 Period*, *Atomnaya Energiya* 23(6), 503-511 (1967), Translated in *J. Soviet At. Energy*, 23(6), 1270-1278 (1967).
5. *Nuclear Reactors in the Soviet Union*, Report of Reciprocal Trip to U.S.S.R., 1964, WASH-1060.
6. Lavrencic, D., *Com. Naz. Energ. Nucl. Noticz.* 15(4), 51-65 (1969).
7. U.S. Team visit to U.S.S.R., June 1970.

Fig. III.E.5. The BR-5 Operating Schedule.

- Contrary to previous reports, specimens for metal-swelling studies had not been inserted in BOR-60.

- Identification of failed elements relatively quickly by heating a suspected subassembly to 300°C in a sodium-filled can and measuring the gas activity ($\sim 1000 \mu\text{R/sec}$ versus $1.5 \mu\text{R/sec}$ for a leaktight element).

- For BR-5 oxide fuel, it has been established that (1) the cold gap closes at 0.6% burnup, with a maximum linear pin power rating of 450-700 W/cm; (2) the average restrained fuel-swelling rate was 1.5% per 1% burnup; (3) there was a peak in the swelling versus fuel-center temperature curve; (4) restraint occurs at medium temperatures with various thickness to diameter ratios, no changes being observed in cladding diameter for a thickness-to-diameter ratio of 0.05 and a temperature of 950°F.

- No leaching of oxide or carbide fuel has been observed in failed fuel.

Some of the more interesting testing developments anticipated during the coming year include:

- Exposures of the BR-5 uranium carbide beyond the present 5.2% burnup.

- Testing of both instrumented and vented fuel assemblies in BOR-60.

- BOR-60 irradiation data on low-fuel density, as-sintered (not ground), nondished, annular pellet fuel.

- Evaluation of Russian stainless steel irradiated to a fluence of 7×10^{22} nvt in BR-5.

- Shutdown of BR-5 in preparation for operation to 10 MWt.

- Evaluation of both boron carbide and tantalum as control materials in BOR-60.

BN-350 Pin Dimensions (mm)

It is now planned to start up BN-600 with a UO_2 -fueled core and to convert later to a plutonium-fueled core with mixed oxide. Each fuel subassembly will be contained in a 2-mm-thick duct with 1.5-mm pads for lateral spacing at outside corners of subassembly cans above core centerline.

5. Fuel Recycle

Reprocessing studies in general appear to have a relatively low priority. Pyrochemical and fluoride volatility

processes are being investigated, principally at Kurchatov, but existing aqueous reprocessing plants will be used for the initial cores enriched with either ^{235}U or plutonium. Decisions have not been made whether to locate subsequent reprocessing plants near breeder reactor sites or to establish centralized reprocessing facilities. A special group is working on design of shipping containers and other problems of shipment by railway.

6. Summary

The U.S.S.R. fuels and materials programs involve extrapolation from the results obtained with BR-5, carried out with relatively low heat ratings, fuel burnups, and coolant temperatures. Irradiation of both oxide and carbide fuels beyond failure of many pins has produced valuable data, particularly evidence that failure propagation is not as serious as is sometimes assumed. Very little experience has been obtained in fabricating $(\text{Pu},\text{U})\text{O}_2$ fuel. The importance of voidage, relation of gas plenum to core length, and the effect of stainless steel swelling on fuel-pin and fuel-assembly behavior are only just being realized. Fuel assembly designs for BOR-60 and BN-350 have been frozen; in addition, there cannot be any translation of BOR-60 fuel behavior and performance to the first cores for BN-350.

Existing reprocessing facilities will be used for the foreseeable future.

Major changes (from earlier reports) noted in the U.S.S.R. program are:

- Use of low-smear density fuel, achieved with annular pellets and a large, cold, diametral gap.
- Recognition of the metal-swelling problem, but continued reliance on U.S.S.R. modified stainless steel compositions.
- Increasing concern with $(\text{Pu},\text{U})\text{O}_2$ fabrication problems.
- Even closer packing of core components.
- Emphasis on studies of fuel cladding-fission product compatibility.

F. SODIUM TECHNOLOGY

The U.S.S.R. is developing a good technology base for using sodium as a fast-reactor coolant. Sodium-technology development is carried out mainly at Obninsk. The following indicates the status of technology in various areas.

1. Receiving and Handling of Incoming Sodium

At BOR-60, the charged sodium was heavily contaminated with paraffin, having been shipped under paraffin in 1000-liter drums. After heating to 200°C for 3 hr under vacuum, a "slagging" operation was performed to skim paraffin from the surface of the sodium. Paraffin was also removed by adherence to tank walls, by filtration, and by cold trapping. Since the primary system is not to be operated at a high temperature for some time, continual removal by cold trapping should remove any remaining paraffin hydrocarbons before operation at a high temperature breaks them down.

At BN-350, sodium is received in 1000-liter barrels, which are placed in electric furnaces for melting. The incoming sodium handling and purification system is shown in Part 3 of Appendix A. The melted sodium is transferred to a large storage vessel from which it is circulated through a NaK-cooled cold trap to obtain an oxygen concentration of 50 ppm before filling the reactor sodium systems.

2. Sodium Purification

As in the United States and elsewhere, cold traps are used to maintain oxygen concentrations at satisfactorily low levels. The Soviets have extensively investigated* a variety of cold-trap designs, the major objective being to increase their capacity for sodium oxide. The design of the cold traps used at BN-350 and BN-600 is illustrated schematically in Fig. III.F.1. Such cold traps, which will be cooled by NaK, were being installed at BN-350. In the cold traps used at BN-350 and BN-600, only one metal wall separates the coolant from the sodium—hence, the use of NaK as the heat-transfer medium. At BN-350, four very large NaK-to-air heat exchangers were mounted on the wall of the main reactor room, exhausting through the wall directly to the outside.

A cold trap of different design is used on BOR-60 (see Fig. III.F.2). The unique feature of this cold trap is that it is steam or water cooled. Low-pressure steam is used to keep the sodium molten and to provide fine control on the trapping temperature. A low-melting lead-base alloy is used as a thermal bond between the steam jacket and the cold-trap vessel. Two metal walls, in addition to the thermal bond, separate the steam or water from the sodium.

*V. I. Subbotin, B. L. Kirolov, and F. A. Kozlov, *The Purification of Sodium from Oxygen and Monitoring of the Oxygen Content in Sodium*, U.S.S.R. Report 701 (1965); *Proceedings of the IAEA Symposium on Progress in Sodium-Cooled Fast Reactor Engineering*, Monaco, Spring 1970.

Fig. III.F.1. Schematic Drawing of, and Temperature Distribution in, Cold Trap Cooled by NaK (Basis of Cold-trap Design for BN-350 and BN-600)

3. Sodium Purity Specifications

The sodium purity specification for oxygen charged to a system was given as 50 ppm. (The concentration of oxygen maintained in an operating system was not learned.) No specification for carbon was obtained, but the carbon concentration in BOR-60 was reported to be 40 ppm, that in BR-5, 150 ppm. These specifications and concentrations for oxygen and carbon are much higher than U.S. practice. According to U.S. experience, these levels of impurities would lead to corrosion and mass transfer problems at the high temperatures specified for BOR-60, BN-350, and BN-600.

4. Sampling

On-line distillation samplers located on main circuit bypasses are extensively used in development work at

Obninsk. Several types have been developed. After the sodium has been distilled away, the residue is analyzed for the element of interest. The limit of detection for oxygen by the vacuum-distillation procedure is 1-5 ppm. Such samplers have not been remotized, and, hence, are used only on nonradioactive systems. One distillation unit is to be installed on the secondary system of BN-350.

5. On-line Monitors

Plugging meters are used on reactor sodium systems to indicate the level of impurities in the sodium. Extensive studies by the Russians of plugging-meter operations has been documented in the literature.^{*,**}

Considerable development of in-sodium hydrogen-diffusion meters has been carried out. Hydrogen that is diffused through a nickel membrane, which is maintained at 450°C, is swept by argon into a thermal-conductivity meter, where the hydrogen concentration is measured. The thickness of the membrane is 1 mm (40 mils), which is about four times the thickness of membranes used in the U.S. and elsewhere. Twelve such meters, one on each steam-generator circuit, are to be installed on BN-350.

Fig. III.F.2. Schematic Design of BOR-60 Cold Trap.

^{*}*Ibid.*, see previous page.

^{**}JPRS-48330, *CEMA Symposium on Atomic Power Stations with Fast Reactors*, Vol. 1 (July 1, 1969), Joint Publications Research Service, U.S. Dept. of Commerce; V. I. Subbotin, F. A. Kozlov, E. K. Kuznetsov, and N. N. Ivanovsky, "Monitoring of Oxygen and Hydrogen Contents in Sodium," *Symposium on Alkali Metal Heat Carriers*, Vienna, 1966.

At BN-350, leakage of 400 g of water into 100 tons of sodium (4 ppm of water or 0.44 ppm of hydrogen) could be detected with a cold trap disconnected from the system. With the cold trap connected, a leak rate of 40 g of water per hour (2.5×10^{-5} lb/sec) could be detected in 40 hr, at which time the total in-leakage of water would be 1600 g. The lowest concentration of hydrogen detectable is much higher than that detectable by U.S. meters. Therefore, for any particular water leak rate, the time of detection is correspondingly longer. The Russians stated that there also would be a long delay time before hydrogen could be detected in the cover gas, because of the long time required for hydrogen to equilibrate between the sodium and cover-gas phases.

The Russians indicated that development of other on-line monitors, e.g., the electrochemical oxygen meter and the carbon-diffusion cell, is continuing.

6. Fission-product Behavior

Because of the large number of cladding failures in BR-5, the Russians have more experience on the behavior of fission products in sodium and on the cleanup of sodium systems from fission products than anyone else in the world. As a result of this experience, permissible limits for radioactivity in LMFBR systems have been established, the criterion being the ability to gain access to equipment for repair and maintenance. The permissible limits are the following:

Cover Gas	Limiting Concentration
^{133}Xe	200 mCi/liter
^{135}Xe	150 mCi/liter
Sodium	
^{131}I	10 mCi/liter
^{137}Cs	5 mCi/liter
Pipe Walls	
^{131}I	5 $\mu\text{Ci}/\text{cm}^2$
Zr/Nb (mostly on walls)	3 $\mu\text{Ci}/\text{cm}^2$
^{140}Ba and La	25 $\mu\text{Ci}/\text{cm}^2$

These values correspond to radiation levels of 200 to 300 $\mu\text{R}/\text{sec}$ on the pipes and 500 to 600 $\mu\text{R}/\text{sec}$ on the pumps and heat exchangers. At these radiation levels, maintenance can be carried out with movable shielding.

When these permissible limits are exceeded, it is necessary to shut down the reactor and institute cleanup operations. The experience at BR-5 has been translated into

a number of fuel failures that can be tolerated at other fast reactors before shutdown would be necessary. At BOR-60, 100 fuel elements could fail before the reactor would have to be shut down.

The activity resulting from activation of corrosion-product elements has also been followed in BR-5 for the sodium conditions prevailing, namely, an inlet sodium temperature of 400-430°C and an outlet sodium temperature of 500°C. Of the corrosion-product elements (namely, manganese-54, cobalt-58, and cobalt-60), only manganese could be detected. Its activity level on the pipe wall was about the same as that of Zr/Nb, or 3 $\mu\text{Ci}/\text{cm}^2$.

The BR-5 sodium and cover-gas systems have not been analyzed for tritium, but the Russians are planning to perform such analyses.

7. Sodium-Water Reactions

At Obninsk, single-tube, complete-rupture experiments had been made in static and flowing sodium, simulating both BOR-60 and BN-350 conditions. Temperature, hydrodynamics, pressure, and tube deformation were measured. The maximum rate found was 3 kg/sec water injection.

In static-sodium tests, an adjacent tube failed. In flowing-sodium tests, adjacent tubes did not fail but "bent aside." Maximum temperature recorded was 1100°C; maximum shock pressure varied from 100 to 150 atm, with a duration of 1 msec. A second, longer peak, coming 4-6 sec after initiation of the reaction (see Fig. III.F.3), built up to about 50 atm in a "high-pressure drop situation," but was stated to be lower for the BN-600 situation. The initial shock wave was considered to have so little total energy as to be "not dangerous," but the secondary peak pressures were taken into account in IHX (and presumably steam

Fig. III.F.3. Pressure Pattern Resulting from Sodium-Water Reaction.

generator) design, based on single-tube rupture. No further sodium-water reaction work was being done at Obninsk, and the test equipment had been disassembled. Sodium-water reaction work was reported to have been published in "Thermal Energetics," a publication of the Soviet Academy of Sciences.

Small-leak work was performed at Melekess, in a rig whose thermal capacity was being upgraded from 400 kW to 1-2 MW. Tests were conducted on 19-tube bundles containing one active tube, to obtain erosion data as a function of temperature and tube spacing. For 500°C circulating sodium and 310°C water injected at 100 atm, an empirical relation for minimum damage to stainless steel tubing had been obtained. Where M is the distance from the water-discharge orifice, and d is the diameter of the water-discharge orifice, minimum damage occurred at $M/d = 14$ to 16. In steam and water tests, there was no "local" damage from the water, but rather a generalized attack. It was planned to continue "small leak" tests on the Incolloys and 2¼ chrome steel after upgrading the rig capacity.

Sodium-water reaction protection systems on BOR-60 and BN-350 used rupture diaphragms on the sodium-expansion tanks, cut by electrically operated knives actuated by sodium pressure signals, as described in Parts 2 and 3, respectively, of Appendix A. The BN-600 system was not described, except to note that 250-m³ separators in series were mounted on top of the steam-generator room for each of the three secondary loops (as observed on a model). Previous statements that there were no valves in either primary or secondary systems appear to rule out the "fast-acting" stop valves in the secondary sodium system, such as are provided on BOR-60 and BN-350.

a. Sodium-frost Formation

In BR-5, in which the outlet sodium temperature is 500°C, no significant difficulty has been encountered with sodium-frost formation. Apparently, no great difficulty with frost formation is anticipated. However, in BN-350 and BN-600, for which sodium outlet temperatures are slightly higher (500-550°C, respectively), the fuel-elevator mechanism can be heated to overcome any problems caused by the clogging of mechanisms with frost.

b. Sodium Removal

At BR-5, sodium is removed from cladding first in "alcohol" (the type of alcohol was not determined), then with a mixture of alcohol and water, and finally with alcohol. At BOR-60, sodium is reacted with a mixture of steam and nitrogen and the reaction products are removed

with a water flush—a removal scheme similar to that used at EBR-II. At some time in the future, removal of the sodium by dissolving it in lead will be tried. The exact procedure for cleaning the subassemblies is to be developed over the next year.

c. Sodium Fires

The Russians were very interested in methods of fighting sodium fires. A leaflet describing MET-L-X compounds was given to Timovaeyev at Shevchenko. Also described to them was the use of shallow trays containing a perforated false bottom and the use of inert-gas blanketing of critical items of equipment.

At BOR-60, certain items of equipment, e.g., the intermediate heat exchangers, are blanketed with nitrogen. A large bin of sodium carbonate was also present for fighting sodium fires.

G. PHYSICS

1. Introduction

The Soviet fast-reactor physics program is similar to that of the U.S. in the balance of effort between data acquisition, integral experiments, code development, and support of design. An important difference in the area of critical experiments and their interpretation is the use of fully plutonium-fueled assemblies in the U.S., with effort focused on design support for the FFTF and the demonstration plant, whereas the Soviet critical assemblies will be fueled mainly with ²³⁵U for the next several years. The initial cores of the BN-350 and BN-600 reactors will be ²³⁵U-fueled.

Fast-reactor physics development and design activities are concentrated mainly at the Institute for Physics and Power Engineering in Obninsk. These activities include:

a. Nuclear-data measurements with electrostatic-type accelerators up to 10 MeV and other neutron sources.

b. Compilation and evaluation of nuclear data.

c. Critical experiments in the two BFS facilities for fast-reactor design mockups and for more basic integral measurements to improve fast-reactor design data and computation methods.

d. Shielding experiments in support of fast-reactor design, using the BR-5 reactor as a neutron source.

USA critical system

e. Development of theory and computer codes and their application in the design of fast reactors.

Important contributions to fast-reactor physics development are made also at other laboratories. Nuclear-data measurements important to fast reactors are conducted with the pulsed-reactor facility at Dubna, and physics theory and computational work carried out at the Kurchatov Institute in support of thermal reactors has found important applications for fast-reactor analysis. At the Scientific Institute for Atomic Energy Reactors near Melekhess, about 50 professional people in the Reactor Department are engaged in work that would be identified as reactor physics in the U.S. This activity has been largely thermal-reactor physics focusing on the VK-50 experimental boiling-water reactor; but it is rapidly shifting to an emphasis on fast-reactor physics oriented toward the BOR-60 experimental fast reactor. The latter includes critical experiments in a coupled fast-thermal assembly, BOR-60 physics startup and operations testing, and participation in data evaluation and development of calculation methods.

2. Nuclear-data Measurements

Cross-section measurements of major importance to fast-reactor design are being conducted in the pulsed-reactor facility at Dubna (EBER-I) and the electrostatic accelerator facilities (2.5, 4.5, and 10 MeV) at Obninsk. Improved measurements of ^{239}Pu alpha and of ^{239}Pu fission cross sections are being conducted at both locations; ^{238}U Doppler activation measurements are under way at Dubna, and preparations are being made there for ^{238}U radiative capture measurements; and at Obninsk measurements are under way for $\bar{\nu}$ and σ_f for ^{239}Pu and ^{235}U and $\sigma(n, n')$ for ^{238}U . Some of the earlier Soviet measurements on ^{239}Pu alpha and $\bar{\nu}$ and ^{238}U $\sigma(n, \nu)$ were reported recently at the Second International Conference on Nuclear Data for Reactors, Helsinki, June 15-19, 1970.

The Van de Graaff and other facilities for cross-section measurements at Obninsk are available to serve the needs of the fast-reactor program in a timely manner. Cross-section measurements at Dubna must compete with solid-state physics and other research activities. Although extensive facilities for nuclear-data measurements exist at many laboratories in the U.S.S.R.* in addition to those at Obninsk or the pulse-reactor facility at Dubna, these other facilities are either devoted to low-energy measurements of main interest to thermal reactors or to research in nuclear or solid-state physics.

3. New Facilities for Nuclear-data Measurements

A second larger pulsed reactor (EBER-II) is being built at Dubna for operation around 1975. EBER-II will be capable of about 5 MW average power (roughly 1000 times greater than EBER-I) for pulses of 5- to 100- μsec duration.

A new linear accelerator, FAKEL, is being built at the Kurchatov Institute in Moscow for operation in 1972. It will be a 60-MeV machine capable of 10- μsec pulse duration and will be used for both cross-section measurements and studies of the physics of solids.

The microtron pulsed electronic accelerator at Obninsk, which is used in part to produce a neutron pulse in the BFS-1 critical facility for the time-of-flight measurements (see Part 6 of Appendix A) also provides a radiation source (by impinging the electron beam on an isolated target) for the study of photonuclear reactions of interest to reactor operation.

The BOR-60 reactor at the Scientific Institute for Atomic Energy Reactors near Melekhess has four beamholes which will be used mainly for solid-state experiments. For the future, one of the beams will be used for integral cross-section measurements by placing materials to be measured at various distances from the reactor in a tube

European and Soviet system

*Atomic Energy in the Soviet Union, Trip Report of the U.S. Atomic Energy Delegation (May 1963).

extracting the neutron beam. These measurements will be correlated with neutron spectrum determinations along the tube length made with scintillation detectors and integrally with threshold and resonance detectors.

4. Nuclear-data Evaluation

Most of the work on collection and dissemination of data for U.S.S.R. atomic reactors is performed by the Nuclear Data Center at Obninsk, which plays a role similar to the Brookhaven National Neutron Cross Section Center in the U.S. This parallel applies also in the data-evaluation activities centered at Obninsk but conducted also at many other laboratories. Guidance for the activities of the Nuclear Data Center is provided by a Nuclear Data Committee of distinguished physicists from different institutes throughout the U.S.S.R. Two coordinating groups made up of scientists from different institutes have been formed, one for the coordination of nuclear-data measurements and the problems of compilation and evaluation of experimental data—the other for the systematic collection of nuclear-data requirements and the construction of

multigroup systems of nuclear constants. The major activity of the Nuclear Data Center of interest to fast-reactor design is the preparation now under way of an evaluated nuclear-data file, similar to the U.S. Evaluated Nuclear Data File (ENDF/B), with processing computer codes to utilize the file for generation of multigroup constants.

5. Integral Critical Experiments

Critical experiments in support of the U.S.S.R. fast-breeder-reactor program are conducted mainly in the two critical facilities, BFS-1 and BFS-2 (Big Physics Stands) located at the Institute for Physics and Power Engineering in Obninsk. BFS-2 is the new large facility capable of handling mockups for breeder power reactors in the 1000-MWe range. (Figure III.G.1 is a top view of the BFS-2 facility.) Some critical experiments in support of the operation of the BOR-60 reactor are also done in the coupled fast-thermal critical assembly located at the Scientific Institute for Atomic Energy Reactors in Melekess. These facilities and general plans for their near-term use are described in Part 6 of Appendix A.

Fig. III.G.1. Top View of the BFS-2 Critical Facility at Obninsk.

6. Shielding

Neutron transmission through fast-reactor shielding configurations is measured with the shielding facilities of the BR-5 reactor in Obninsk—the B-2 beamhole and the KTK fission-plate facility. Part 1 of Appendix D discusses the facilities, the experiments conducted, and the shielding-design requirements.

7. Calculation Methods

Fast-reactor physics calculations of neutron fields are done with up to 26 energy groups using one- and two-dimensional diffusion theory, P_1 or P_2 , and one-dimensional S_N with up to 26 groups and eight angles. The 26-group ABBN set of nuclear-data constants, updated to take into account recent cross-section measurements, is the basic data file for reactor calculations. The ABBN type of file will be generated from pointwise data when the

evaluated data file and its processing codes become available.

Monte Carlo calculations with nearly 10^5 case histories have been conducted for analysis of shielding configurations. Fuel-burnup calculations are performed in multi-group for a one-dimensional model, and in one-energy group for a three-dimensional model. (The latter is believed to be the BEPR code developed for burnup studies on water reactors—or a modification thereof.)

The Physics and Power Engineering Institute at Obninsk has an M-220 computer, and the Scientific Institute for Atomic Energy Reactors near Melekhov has a BESM-3 computer. These machines are smaller and slower than the advanced BESM-6 computer, which is available at the Kurchatov Institute in Moscow and is used extensively for calculations pertaining to the design and operation of water-reactor plants. The Institute at Obninsk expects to obtain a BESM-6 computer in the relatively near future.

IV. SOVIET WATER-REACTOR TECHNOLOGY

A. INTRODUCTION

As noted in the report on the 1964 visit, the two thermal-reactor types that have received most attention in the U.S.S.R. are the pressurized-water reactor at the Novovoronezh Station and the nuclear superheat reactor at the Belayarsk Station. The technology on PWR's is derived from that carried out for the Lenin ice-breaker at the Kurchatov Institute, Moscow; the latter Institute is still responsible for the PWR fuel-development work in the MR reactor loops. The graphite-moderated nuclear superheat reactors at Belayarsk are based on the early graphite-water power station at Obninsk. An experimental boiling-water reactor (VK-50) has also been in operation at Melekess.

Overall, the PWR is now adopted as the main thermal nuclear station reactor. In addition to the planned capacity of 2500 MWe at Novovoronezh, two 440-MWe units are being built at Murmansk and two 440-MWe units at Oktomberyan in Armenia. These 440-MWe standard units are also being built in East Germany, Bulgaria, Czechoslovakia, and Finland.

B. PRESSURIZED-WATER REACTORS

Unit No. 1 at Novovoronezh (210 MWe, six loops) has now been in operation for 5 years. After 2 years of low availability, it operated satisfactorily until December 1969 when it was shut down for 9 months for examination of fuel and internals. Unit No. 2 (365 MWe, eight loops) went on line in December 1969, the main changes being dry refueling and an increase in the core capacity. Dry refueling has caused difficulties due to spread of radioactivity in the reactor hall and wet refueling will be used for Units 3, 4, and 5. Units 3 and 4 (440 MWe each, six loops) are being built, and Unit 5 (1000 MWe, four loops) has been designed. The schedule is for Unit 3 on line in 1971, Unit 4 on line in 1973, and Unit 5 on line in 1975. The growth in annual power output and operating capacity is shown in Fig. IV.B.1.

Movable fuel assemblies with boron-steel followers, used for burnup compensation in the Novovoronezh PWR's, give good local power peaking factors (1.13 local peaking when the fuel portion is in the core). The movable fuel assembly is in a hexagonal channel—similar to a normal stationary fuel assembly. The control-rod follower is actually a water hole surrounded by the boron-steel cylinder (1.8% boron) to prevent power peaking.

Soluble poison (boric acid) is now being evaluated and will be used in Novovoronezh PWR Units 3 and 4 to enable

A Novovoronezh PWR Mark II fuel bundle

a reduction in the number of control rods to 37 (from 73 in Unit 2). Fig. IV.B.2 shows the state of construction for Unit 3.

Management of fuel and control rods for an entire operating interval between refuelings is precalculated on a large computer (the BESM-6 at Kurchatov). A three-dimensional, one-energy-group burnup code, BEPR, developed at Kurchatov is used in such fuel-management calculations. The code calculates, at burnup intervals specified by input data, isotopic concentrations, power distribution, and control-rod positions to maintain criticality in accordance with a preselected mode of control-rod management.

Fig. IV.B.1. Growth in Annual Power Output and Operating Capacity of Soviet Pressurized-water Reactors.

Fig. IV.B.2. State of Construction of PWR Unit 3 at Novovoronezh.

The fuel-development work for pressurized-water reactors is carried out at Kurchatov Institute, Moscow. PWR fuel is now standardized with annular UO_2 pellets and zirconium-1% niobium as cladding wire wrapped with zirconium-2½% niobium for the subassembly can. Experimental fuel assemblies have been irradiated satisfactorily to 63,000 MWd/T peak in the Kurchatov in-pile loops. Experience at Novovoronezh Unit 1 has been limited to 13,000 MWd/T average discharge burnup, and approximately 10% of the subassemblies have failed. These failures are attributed to metallurgical defects (faulty welds).

Completion of the five PWR units by 1975 will provide 2,500-MWe capacity at Novovoronezh. No additional units will be constructed, because of cooling limits on the Don River which permit 5°C allowable rise in summer and 3°C rise in winter downstream from the outlet (about 15-20°C rise permitted at the outlet). For Units 1 and 2, the Don River will be used directly for condenser cooling; but seven natural-circulation cooling towers will be built for Units 3 and 4, three for each unit, together with one spare. An artificial lake (1500 acres) will be built for cooling Unit 5. The radioactivity release criterion from the station is 200 Ci/day. Operations to date with Unit 1 have given values of 30-40 Ci/day. The Soviets do not consider plutonium thermal recycle economical until a total of 10,000 MWe PWR capacity has been installed.

C. NUCLEAR SUPERHEAT REACTORS

Two boiling-water nuclear-superheat units are in operation at Beloyarsk. Unit 1 was started in April 1964 at an initial power of 75 MWe, which has since raised to 106 MWe and 510°C steam. Unit No. 2 was started in December 1967, reaching a power of 195 MWe and a thermal efficiency of 37.5%. They are the first reactors using steam cooling as well as boiling in regular operation, with a graphite-moderated, tube-type reactor similar to the APS-1 which operated in 1954. Operation of both units at over 80% load factor has continued since 1967, producing over 5.5 billion kWh.

The most difficult operating problem has been the removal of a damaged or leaking channel. This has been solved by a new channel design which gives very low failure rates, and a special device for rapid extraction of damaged channels. One channel was reported to have failed at noon the day before the Delegation's visit, but was replaced, and the plant was up to 180 MWe power during the plant tour. Leakers are removed into a 35-40-ft dacron "sock," which elastically seals itself to the element and prevents spread of particulate radioactivity.

Discharge of gas up the stack averages 250 Ci/day,

against a permissible "norm" of 1000 Ci/day. The radioactivity is mostly argon (80%).

Fuel burnups have reached 6000 (average) and 8000 (peak) MWd/T, with a design target of 10,000 MWd/T. Higher burnups (to 20,000-30,000 MWd/T) are expected in the future, on the basis of test work being done at Obninsk.

The water-treatment methods used attain exceptionally high standards of water purity to protect the austenitic materials in the reactor (for example, 50 ppb of total solids and less than 15 ppb of chloride). A further feature is the tenfold reduction in the consumption of hydrazine in feedwater (relative to initial operation) by injection into only the next-to-last feedwater heater.

D. NATURAL-CIRCULATION BOILING-WATER REACTOR

There is a relatively small effort at Melekes on the VK-50 natural-circulation boiling-water reactor (Fig. IV.D.1) which is inherently limited to ratings below about 100 MWe.

Improvements in power flattening and in lower hydraulic resistance of the fuel spacers have permitted increased power density. Initially, the reactor operated at 28 kW/liter, but this has been increased to 37 kW/liter. The Soviets expect to reach 50 kW/liter and 270 MWt.

The original plan to install integral superheat channels has been dropped in favor of the objective of a simple and reliable power source. This work appears to be aimed at remote locations requiring simple and reliable small units (up to 100 MWe)* rather than at the economics of large-scale units (which requires forced-circulation designs used in BWR's in the U.S.). Some interesting technical features include the following:

1. Fuel cladding is Zr-1% Nb, which has performed well up to peak burnups of 16,000 MWd/T. Two leaks occurred which did not grow significantly in 150 days of additional full-power operation.
2. A grid spacer with parallel wires is used at two different elevations, said to give a flow resistance less than half that of honeycomb grid-type spacer.
3. Calcium and magnesium deposits on the fuel (from condenser in-leakage) are removed by a boric-citric acid mixture which does not attack either zirconium or stainless steel.
4. In-air refueling is used (rather than underwater as originally designed).

*There is one natural-circulation BWR in the U.S., the 70-MWe Humboldt Bay Reactor.

Fig. IV.D.1. The Boiling-water Reactor, VK-50.

V. ITINERARY AND SOVIET PERSONNEL

A. ITINERARY

Monday, June 15

6:40 p.m. Arrival at Moscow Sheremetyevo Airport; met by Mr. Phillipov (State Committee), Miss Kondratova (guide), Mr. Barnashov (translator) and Mr. Harben (First Secretary, U.S. Embassy)

10:00 p.m. Arrival at hotel

Tuesday, June 16

7:48 a.m. Departure by plane from Vnukhovo Airport, Moscow, to Ulianovsk

10:23 a.m. Arrival at Ulianovsk; departure to Melekes

12:48 p.m. Arrival at Melekes

3:10-6:00 p.m. Visit the Atomic Reactors Research Institute

Evening Visit with Dr. Kazachkovsky and the "English Language Club" at the Culture Center

Nuclear fission symbol—Melekes

Wednesday, June 17

9:00 a.m.-6:40 p.m. Continuation of tours of reactors, sodium facilities, and hot cells, and of discussions

Evening Dinner with Dr. Kazachkovsky and staff, and Dr. D. Sokolov of the State Committee

ITINERARY

Thursday, June 18

10:45 a.m. Departure for Moscow by plane
12:00 Noon Arrival in Moscow
9:40 p.m. Departure to Voronezh by train

Friday, June 19

8:10 a.m. Arrival at Voronezh
11:00 a.m.-Evening Visit the Novovoronezh Power Plant
Evening Dinner with Chief Engineer Samoilov and staff
10:50 p.m. Departure for Moscow

Saturday, June 20

9:08 a.m. Arrival in Moscow
2:50 p.m. Departure from Moscow to Mineralny Vody by plane (Tu 104)
4:00 p.m. Arrival at Mineralny Vody
6:15-8:15 p.m. Tour of Piatigorsk

Sunday, June 21

6:15-8:45 a.m. Flight to Shevchenko
Morning-Afternoon Visit the BN-350 and desalination installations; discussions
8:00 p.m. Dinner with Director Yurchenko, Dep. Dir. Timofeev, Chief Engineer Vasilenko, *et al.*

Monday, June 22

10:55 a.m. Departure from Shevchenko to Mineralny Vody
5:00 p.m. (plane delay) Departure for Moscow
7:03 p.m. Arrival Vnukhuvo Airport, Moscow

Tuesday, June 23

8:41 a.m. Departure by train from Kievsky Station for Obninsk
10:40 a.m. Arrival Obninsk
Morning-Afternoon Visit the Physics and Power Institute (BFS-2, BR-5 tours; discussions)
8:00 p.m. Visits to two homes and one guest house for dinner

Wednesday, June 24

9:00 a.m.-4:30 p.m. Continuation of discussions at Physics and Power Institute
5:00 p.m. Dinner with Dr. Kuznetsov and staff
7:22 p.m. Departure to Moscow
9:42 p.m. Arrival Moscow

Thursday, June 25

8:05 a.m. Departure by plane from Moscow
12:10 p.m. Arrival Sverdlovsk
1:20 p.m. Arrival Beloyarsk
Afternoon Visit the Atomic Power Plant at Beloyarsk
Evening Tour of Beloyarsk
Dinner with Chief Engineers Ivanov and Verettenikov and staff

Friday, June 26

9:00 a.m.-5:30 p.m. Continuation of tour of Beloyarsk plant
6:00 p.m. Visit Dom Kultura "Rovesnik" at Beloyarsk
6:20 p.m. Visit Sports Center and Stadium at Beloyarsk
8:00 p.m. Departure to Moscow
11:30 p.m. Arrival Moscow

V. P. Nevsky (center), Director of the Kurchatov Station and of the town of Beloyarsk, and V. Barnashov, translator (left center), describe the BN-600 reactor to delegates Dickinson (left), Wensch, and Murray (right).

Saturday, June 27

Day City sightseeing (State Committee as host)

Evening Cocktails at apartment of Mr. Harben, U.S. Scientific Attache

Departure of Drs. Zebroski and Evans from Moscow

Sunday, June 28

9:00 a.m.-12 Noon Review of notes at U.S. Embassy

Afternoon Sightseeing

Wednesday, July 1 Departure of remaining members of Delegation

B. U.S.S.R. SCIENTISTS, ENGINEERS, TECHNICIANS, AND ADMINISTRATORS MET BY U.S. DELEGATION

1. Atomic Reactors Research Institute, Melekes; June 16-17

Prof. O. D. Kazachkovsky *Head of the Institute*

N. V. Krasnoyarov *Deputy Director*

V. Afanasiev *Head of Dynamics Laboratory*

Y. Soloviev *Chief Engineer of VK-50*

A. Zabelin *Head of Chemical Laboratory for VK-50*

S. Votinov *Chief of Materials Department*

V. I. Klimenkov *Chief Radiometallurgical Laboratory*

V. Prokhorov *Materials*

A. M. Smirnov *Deputy Chief, Superintendent of BOR-60*

Monday, June 29

10:10 a.m. Visit the Kurchatov Atomic Energy Institute

1:30 p.m. Luncheon with Dep. Dir. Zelenkov and staff

3:00-6:00 p.m. Continuation of discussions at Kurchatov Institute

Evening Moscow Circus (State Committee as host)

Tuesday, June 30

10:00-11:30 a.m. Meeting with State Committee for the Utilization of Atomic Energy in the U.S.S.R.

V. Gryazev	<i>Head of Reactor Department</i>	G. Pomerantsev	<i>Deputy Chief Engineer of the Reactor</i>
G. Gadzhiev	<i>Physicist</i>	I. Shvedenko	<i>Chief of the Instrumentation</i>
Shkurov	<i>Services and Facilities Chief</i>	N. Tverdovsky	<i>Chief of the Dosimetry Service</i>
E. Borisjuk	<i>Chief Engineer of BOR-60</i>	A. Khudiakov	<i>Chief Specialist of Mechanics of the Reactor</i>
I. Lebedev	<i>Chief Materials Laboratory</i>	I. Vakhnia	<i>Chief of the Desalination Plant</i>
Demayanovich	<i>Chief Engineer</i>	N. Skorikov	<i>Chief Engineer of the Dosimetry Service</i>
Gavrilin	<i>Valve Test</i>	V. Kolosov	<i>Engineer of the Service of the Instrumentation Chief</i>
Mazanov	<i>Sodium-Water Test</i>	V. Pavozkov	<i>Chief Engineer of the Technical Information Department</i>
Renata Aleksandrova Barkova	<i>Foreign Delegations Asst. to Kazachkovsky and Guide at Melekess</i>	M. Yepikhin	<i>Chief of the Electrical Service</i>
<i>2. Novovoronezh Atomic Power Station, Novovoronezh; June 19</i>		V. Akelkin	<i>Engineer of the Technical Information Department</i>
S. Samoilov	<i>Chief Engineer</i>	F. Martiushev	<i>Chief Aide</i>
Prozorovsky	<i>Chief of Construction</i>	A. Tiulenev	<i>Chief of the Construction Services</i>
L. Golubev	<i>Deputy Chief Engineer</i>	<i>4. Physics and Power Institute, Obninsk; June 23-24</i>	
Sedov	<i>Deputy Chief Engineer</i>	Prof. V. A. Kuznetsov	<i>Director of the Institute</i>
M. Portny	<i>Deputy Chief Engineer</i>	D. M. Ovechkin	<i>Chief Engineer of the Institute</i>
Malkov	<i>Chief of Operational and Technical Department</i>	A. G. Sokolov	<i>Deputy Director</i>
V. Bagai	<i>Engineer</i>	P. S. Otstavnov	<i>Assistant Director on International Cooperation</i>
Aleshkevich	<i>Engineer</i>	V. V. Orlov	<i>Chief of the Department of Fast and Power Reactors</i>
<i>3. BN-350 Reactor and Desalination Plant, Shevchenko; June 21</i>		I. G. Morozov	<i>Chief of the Scientific Department (Zinoviev reports to him)</i>
D. S. Yurchenko	<i>Director</i>	M. F. Troyanov	<i>Chief of the Nuclear Engineering Laboratory</i>
A. Timofeev	<i>Deputy Director</i>		
K. Vasilenko	<i>Chief Engineer of the Reactor</i>		
G. Semionov	<i>Deputy Chief Engineer of the Plant</i>		
R. P. Baklushin	<i>Deputy Chief Engineer of the Reactor</i>		

V. ITINERARY AND PERSONNEL

A. N. Yegorov	Scientist	M. Ya. Kulakovsky	Heads up shielding (design and BR-5 experiments)
V. B. Litkin	Senior Scientist		
L. P. Abagyan	Physicist (shielding and nuclear data evaluation)	5. U.S.S.R. Visitors at Obninsk, June 23-24	
A. A. Rineysky	Scientist (steam generator designer)	V. V. Stekolnikov	State Committee
V. I. Matveyev	Physicist	I. S. Golovnin	Kurchatov, Fuels
N. N. Aristarkhov	Chief of BR-5 Reactor	E. G. Novinsky	Gorki Institute, Designer
A. V. Karpov	Chief of a Laboratory	N. P. Agapova	Kurchatov, Materials
V. P. Zinoviev	Chief Engineer BFS-1 and -2	Yu. B. Bibilashvili	Kurchatov, Materials
V. N. Bykov	Chief of the Solid State Physics & Radiation Damage Department	V. Schevchenko	Kurchatov, Reprocessing
A. T. Ladigin	Scientific Worker	V. I. Shiryayev	Gorki Institute, Designer
Y. I. Likhachiov	Scientific Worker (physics)	S. G. Gusev	State Committee
A. G. Kormilov	Scientific Worker	V. A. Sidorenko	Kurchatov, PWR Design
P. L. Kirillov	Division Manager in Heat Transfer (steam generator designer)	Titov	Melekess
P. A. Ushakov	Chief of the Heat Transfer Laboratory (works for Kirillov)	V. V. Zotov	Kurchatov, Materials
S. N. Belov	Scientific Worker	6. Dinner at P. L. Kirillov's Home, June 23	
A. I. Mogilner	Chief of a Department (experimental physicist; specialist in noise measurements and analysis)	U.S.	U.S.S.R.
F. A. Kozlov	Chief of the Sodium Technology Laboratory	L. Burris	I. Golovnin
L. A. Kochetkov	Chief of a Department	R. Dickinson	A. Rineysky
Y. I. Bagdasarov	Scientific Worker (reactor safety)	E. Evans	P. Kirillov
V. M. Poplavsky	Sodium-Water Reactions		P. A. Ushakov
			Y. I. Likhachiov
			D. Phillipov
			V. Barnashov
			Mrs. Kirillov
			Mrs. Ushakov

7. Dinner at P. S. Ostavnov's Home

U.S.	U.S.S.R.
P. Greebler	L. P. Abagyan
W. McDonald	F. A. Kozlov
P. Murray	A. Mogilner
	S. Gusev
	P. S. Ostavnov
	M. F. Troyanov

8. Dinner at Guest House

U.S.	U.S.S.R.
G. Wensch	D. Ovechkin
A. Amorosi	A. Sokolov
J. Yevick	V. Orlov
E. Zebroski	L. Kotchetkov
	V. Bykov
	V. Litkin
	I. Morozov

9. Atomic Power Plant, Beloyarsk;
June 25-26

V. P. Nevsky	<i>Director of the plant</i>
B. G. Ivanov	<i>Chief Engineer of Units 1 and 2</i>
G. A. Verettenikov	<i>Chief of Unit 3 (BN-600)</i>
M. P. Alekseev	<i>Deputy Chief Engineer of Maintenance</i>
G. A. Schapov	<i>Chief of Radiochemical Laboratory</i>

10. Kurchatov Atomic Energy Institute, Kurchatov;
June 29

A. G. Zelenkov	<i>Deputy Director of Institute</i>
V. A. Sidorenko	<i>Deputy Chief</i>
Platonov	
S. A. Skvortsov	<i>Deputy Chief, Pressurized Water Reactors</i>
S. P. Ryazantsev	<i>Deputy Chief, Design Test Reactors</i>
S. G. Gusev	<i>State Committee</i>
A. Novikov	<i>Head, Codes and Theory for Water Reactors</i>

11. U.S.S.R. State Committee for Atomic Energy
Moscow; June 30

I. I. Smolin	<i>Deputy Chief, Department of Foreign Relations</i>
D. D. Sokolov	<i>Deputy Chief Engineer, Department of Foreign Relations</i>
J. V. Arkhangelsky	<i>Chief of Section in Foreign Relations</i>
D. P. Phillipov*	<i>Chief of a Section in Foreign Relations</i>
S. G. Gusev	<i>Worker in Foreign Relations</i>
D. F. Khoklova	<i>Worker in Foreign Relations</i>
V. Kondratova*	<i>Aide</i>
V. Barnashov*	<i>Translator</i>

*Traveled with U.S. Delegation throughout the trip.

APPENDIX A

FAST REACTORS AND DESALINATION PLANT

1. THE BR-5 REACTOR AT OBNINSK

a. Operations

The BR-5 reactor is a facility at the reactor development center at Obninsk, whose primary mission is to provide support to the fast breeder program by a wide variety of activities. To a lesser extent, the facility is also engaged in the development of thermal water reactors.

Figure A.1.1 shows the U.S. Delegation with some members of the Obninsk staff.

The BR-5 has operated with an enriched- ^{235}U uranium carbide core (5.2-5.3% carbon) since May 1965. After completion of the present work, modifications will be made to double the power to 10 MWt. These modifications will consist of increasing the heat-removal capability of the secondary system. The coolant will be changed from NaK to sodium, and two additional sodium-to-air heat exchangers will be installed. The mechanical pumps will also be replaced by electromagnetic pumps to improve reliability and increase the plant factor. The modifications will start in mid-1971 and are estimated to take about a year. Operations will then be resumed with a plutonium oxide (60 kg of plutonium) core. The maximum flux at the center of the core will thus be increased from the present 8×10^{14} to 1.5×10^{15} n/cm²/sec. The power per unit volume will be 900 kW/liter. The sodium-outlet temperature will be kept at 500°C, but the inlet temperature will be lowered to 350°C. In Fig. A.1.2, Mr. N. Aristarkhov is shown presenting details of the BR-5 to the U.S. Delegation.

b. Carbide-fuel Behavior

Two types of stainless steel cladding were used in the helium-bonded carbide core—Type 321 and the U.S.S.R. variant on Type 316 (16 wt % Cr, 15 wt % Ni, 3 wt % Mo, 0.8 wt % Nb). No cladding failures occurred in either case, up to 3% burnup. Between 3.6 and 5.2% burnup, however, 38 out of 53 assemblies with Type 321 stainless steel failed. In spite of high carbide swelling, no assemblies containing the U.S.S.R. alloy have failed. This alloy has been selected for the BN-350 uranium oxide fuel.

Figure A.1.3 shows the arrangement of a BR-5 fuel element.

c. Oxide Fuel Examination

Examination of the 19 failed oxide assemblies after 6% burnup had revealed only one or two failed pins per assembly. The highest-burnup oxide subassembly had been examined in detail. This subassembly had been irradiated for 5 years in BR-5 to a burnup of 6.1% and a fluence of 2.4×10^{22} nvt at a maximum cladding temperature of 580°C. Only one pin—the central pin—in the 19-pin assembly had failed. Longitudinal cracks extending the entire length were found on opposite faces of this pin. Failed-fuel propagation had not occurred, and this experience had provided the basis for the assumption that fuel-pin-failure propagation is not a problem with fast-breeder fuel. Extrusion of the fuel into the cracks in the cladding confirmed the substantial plasticity of PuO_2 under irradiation. In spite of the bad failure, there was no fuel-coolant reaction, and the sealing effect of the plastic PuO_2 resulted in no entry of fuel into the coolant stream.

Examination of cladding from the irradiated oxide subassemblies is showing the onset of swelling ($0.5\text{--}1\% \Delta V/V$) after irradiation to a fluence of 2×10^{22} nvt. This first indication of swelling in the U.S.S.R. program has resulted in increased attention being given to this problem at Obninsk. Cladding irradiated to 4×10^{22} nvt will be examined later in 1970, and specimens irradiated to 7×10^{22} nvt by the end of 1971.

d. Failed-fuel Detection

The permissible limits for radioactivity set by the ability to maintain equipment in BR-5 have been determined and are described in the body of this report.

Delayed-neutron detection and cover-gas analysis for xenon have been used at BR-5 and will be used at BN-350 and BN-600 for detection of fuel failures. To locate subassemblies containing failed elements at BR-5, each subassembly is removed, after a cooling time of 10 days, and tested for leak-tightness in an out-of-reactor furnace. Each subassembly is placed in a sodium-filled can, which is placed in a secondary container surrounded by a furnace. The assembly is heated to 300°C, and argon is passed through the cover-gas space to a counting chamber where the gas activity is measured. If no leaking fuel pins are present, the activity level is 1-5 mR/sec. If leaking pins are present, the activity level is greater than 1000 mR/sec.

Fig. A.1.1. Obninsk Staff and U.S. Delegation. Left to right: Front row—Titov, J. Yevick (U.S.), D. Ovechkin, G. Wensch (U.S.), P. Murray (U.S.), L. Burris (U.S.), A. Sokolov, P. Otstavnov, A. Mogilner, V. Shiryayev, V. Litkin; Second row—V. Barnashov, E. Evans (U.S.), W. McDonald (U.S.), V. Bykov, V. Orlov, Yu. Bibilashvili; Third row—E. Novinsky, R. Dickinson (U.S.), A. Amorosi (U.S.), E. Zebroski (U.S.); Fourth row—I. Golovnin, V. Zotov, N. Agapova; Fifth row—D. Phillipov, V. Stekolnikov, P. Greebler (U.S.), V. Kondratova, Unknown.

Fig. A.1.2
Mr. N. Aristarkhov of Obninsk
Staff Briefing the U.S. Delegation
on Details of the BR-5 Reactor.

Fig. A.1.3. The BR-5 Fuel Subassembly. Top—general arrangement of the fuel element. Lower right—cross section of the fuel element. Lower left—fuel pin and its spacing wire.

Subassemblies containing no leaking pins are returned to the reactor. If leaking pins are detected, the can containing the subassembly is sealed and put into storage. There are two such leak-checking devices at BR-5, and two subassemblies can be tested per day in each. The entire core can thus be checked in 15 days. When subassemblies containing failed elements are removed from the core, the delayed-neutron and cover-gas activity levels decrease by factors of 20-30.

e. Behavior of ^{23}Ne

There is a wide difference in the concentration of ^{23}Ne activity in the cover gases of BR-5 and EBR-II. A ^{23}Ne activity of 12 ± 1 mCi/liter in the cover gas of BR-5 had been reported when operating at 100 kW, which contrasts with only 0.03 mCi/liter of ^{23}Ne activity in the argon cover gas of EBR-II operating at 50 MW. The reported concentration of 12 ± 1 mCi/liter of ^{23}Ne in BR-5 cover gas was confirmed. The reason for this high concentration at BR-5 is that the ^{23}Ne is evolved primarily into a small volume of gas (given as 250 liters) above the sodium in the primary pump bowl, from which the gas is sampled continuously. Only grab samples are taken of the cover gas in the reactor. By the time these samples are analyzed, the ^{23}Ne activity, which has a half-life of only 40 sec, has decayed away. The evolution of the ^{23}Ne into a small volume accounts for most of the difference between the ^{23}Ne concentrations at EBR-II and BR-5. Because of good mixing in the pump, there is a much more rapid interchange of neon between sodium and the gas in the pump than there is between the relatively quiescent sodium in the reactor and the reactor cover gas. A comparison of the rate of interchange of gas between reactor cover gas and sodium and pump cover gas and sodium was determined with xenon. The xenon activity in the reactor cover gas was much lower than that in the pump cover gas.

The sodium in BR-5 has also been analyzed for ^{23}Ne activity, and the total amount of ^{23}Ne present in BR-5 (sodium plus cover gas) under steady-state conditions agrees with theoretical predictions. Only by knowing the total amount of ^{23}Ne in EBR-II versus that in BR-5 would it be possible to determine if there are still unexplainable differences.

f. Circuit Behavior

During the 10-year period of operation of BR-5, a leak was found in the 1 wt % Cr-18 wt % Ni-9 wt % Ti (Type 321) stainless steel drain line of the primary sodium system. The location of the leak was at the boundary of a 40- to 20-mm reducing section. A through-crack extended over half the diameter of the connection, and there were many microcracks on the internal surface. The failure was reportedly due to "inadequate freedom of movement of the drainage pipe." However, the reported microcracking on the internal diameter would indicate the possibility of intergranular attack of some form.

Longitudinal cracks were also found in the 1 wt % Cr-18 wt % Ni-9 wt % Ti stainless steel steam tubes of the evaporator section. The cracks were located in the U-shaped steam tubes in the transition region between evaporating and superheating. The tubes were of double-walled construction, with mercury in the annulus. Cracking was revealed to be intergranular, but there were no reports of tests for any contaminant. The Soviets attributed this failure to "intercrystalline corrosion under conditions of thermal cycling and variable stresses in downcomer tubes." Stress-corrosion cracking is also a possible reason for this failure, since cracking occurred in an area where corrosion products could concentrate.

2. THE BOR-60 REACTOR AT MELEKES

a. Introduction

The BOR-60 reactor (see Fig. A.2.1) is an experimental fast reactor at the NIAR site near Melekess, with a maximum power of 60 MWt. There are several objectives for the BOR-60 reactor. The first is to provide a stepping stone in design and operating experience in going from the 5-MWt BR-5 to the Soviet demonstration plants BN-350 (1000 MWt) and BN-600 (600 MWe and 1700 MWh). The second is to use this reactor as a test ground for the development of fuel and components for demonstration reactors. BOR-60 is being operated so as to obtain a maximum of information for the demonstration reactors.

The reactor will be operated at a sufficiently high power level that significant information can be obtained on material and fuel performance, such as stainless steel swelling. Of especial importance will be its use in the development of fuel elements and materials for the cores of high-performance reactors, and of fuel elements which will operate reliably at high thermal and neutron flux densities and at high temperatures.

The design of the BOR-60 was begun at the end of 1963. Construction was started in May 1965 and was completed in September 1968. The reactor went dry critical in December 1968. Filling with sodium started in September 1969, and the reactor was brought to criticality with sodium in December 1969. It has been operating with air-dump heat exchangers at approximately 20 MW since March 1970 and is scheduled to operate at its normal power of 60 MW at the end of 1970. It will be operated with steam generators which are prototypes of the BN-600 units. Prototype fuels of both BN-350 and BN-600, will be inserted prior to operation at 60 MW.

b. Plant Design and Safety

The plant is a loose-coupled pipe system (see Fig. A.2.2), which uses bellows in the secondary system to accommodate differential expansion (see Fig. A.2.12 below). Not only is the plant power output similar to that of EBR-II, but a number of its features resemble those of EBR-II. The core is similar in size, 15.7 in. high and 15.9 in. in diameter compared to the 14 X 20 in. of EBR-II, and design, but has 90% enriched UO_2 fuel. The fuel pin is spiral wire wrapped. The fuel subassemblies have a hydraulic holddown, and both the subassembly and core support are based on the same general principle as in EBR-II. Fuel elements are removed directly from the core without in-reactor storage through double rotating plugs. The fuel is handled and

Fig. A.2.1. BOR-60 Reactor Plant.

cleaned in a somewhat similar manner as in EBR-II. In the primary system two parallel heat-transport loops are used. The intermediate system contains a 30-MWt air dump and two 30-MWt steam generators, all arranged in parallel. The steam generators are similar to those of Fermi, being of the once-through type, single-walled, and with a somewhat similar tube configuration. However, there is a moisture separator between the evaporator and the superheater sections.

The plant was designed for the operating conditions given in Table A.2.1 and for the following transient conditions:

1. 150 cycles of $10^\circ\text{C}/\text{sec}$ ($18^\circ\text{F}/\text{sec}$) (these cycles are from 600 (1112°F) to 480°C (896°F) and are transient scram conditions);
2. 700 cycles for $1^\circ\text{C}/\text{sec}$ ($1.8^\circ\text{F}/\text{sec}$) (these cycles are from 600 (1112°F) to 480°C (896°F) and are power cutback conditions);
3. 1000 cycles of $60^\circ\text{C}/\text{hr}$ ($108^\circ\text{F}/\text{hr}$) (these cycles are from 300 (572°F) to 600°C (1112°F), covering normal load changes as well as normal startup and shutdown conditions).

The Soviets are quite careful to design their reactors to take account of the consequences of likely malfunctions and accidents. They believe that loss of power is something that can happen a number of times over the life of a reactor and that the core should be well protected against this accident. They realize that the safety rods may fail to scram under these conditions, and therefore provide redundancy both in the number of safety rods and circuits. Three independent safety circuits and three safety rods are used even though one, or at best two, are required to control sodium temperatures on a loss of power satisfactorily. BOR-60 is designed with the following features to assure

Fig. A.2.2. Schematic Layout of BOR-60. 1—Reactor, 2—Intermediate heat exchanger, 3—Primary-circuit pump, 4—Steam generator, 5—Overflow tank, 6—Secondary pump, 7—Air-cooled heat exchanger, 8—Overflow tank, 9—Mixer, 10—Turbine, 11—Condenser, 12—Condensing pump, 13—Reheater, 14—Degasser, 15—Feed pump, 16—High-pressure reheater, 17—Cooling tower, 18—Circulation pump.

core cooling under loss of power to the main primary-pump drive motors:

1. Shift to low-power operation on auxiliary power supply;
2. Flywheel incorporated in the pump-motor drives;
3. Backup power for 10 hr of operation, consisting of batteries and diesel generator;
4. Operational procedures to bring in auxiliary power if normal power is not recovered within 10 hr;
5. Natural-circulation capability to handle decay heat within a short time after pump power is lost.

The Soviets believe that pipe rupture can take place, and therefore they provide for secondary containment of the primary system up to stop valves. This is done so that a leak in the primary system will not drop sodium below the outlet pipes.

Half of the subassemblies are monitored by single nonreplaceable thermocouples. These thermocouples can be of use in monitoring the test fuel as well as in averaging sodium outlet temperatures. Since the Soviets do not believe that propagation failure will take place, they are not counting on these thermocouples for prevention of propagation failure. There are instruments for fuel-failure detection on the outlet pipe of the IHX (cold-leg primary circuit), based on detecting delayed neutrons. In addition, fission-gas detectors, located in the cover gas, detect Xe-133 and Kr-85. Shielding has been based upon operation with 100 failed fuel pins that release all of their gaseous fission products.

There is one subassembly which can be completely instrumented. There are vertical beam holes in the reactors.*

*A. I. Leipunsky *et al.*, "The BN-350 and BOR Fast Reactors" *Fast Breeder Reactors*, Proc. London Conf. on Fast Breeder Reactors Organized by the British Nuclear Energy Society, May 17-19, 1966, Pergamon Press, Inc., New York (1967), p.243.

TABLE A.2.1. Design Conditions of BOR-60

Reactor thermal power	~60 MWt
Coolant	Sodium
Fuel	UO ₂
Fuel enrichment	90%
Design exit sodium temperature	600°C (1112°F)
Entering sodium temperature	360-480°C (680-896°F)
Maximum burnup	10%
Maximum power density	1100 kW/liter
Critical mass of ²³⁵ U	176 kg (388 lb)
Maximum sodium velocity	~8 m/s (~26 ft/sec)
Total sodium flow	~1200 m ³ /h (5300 gpm)
Active core height	40 cm (15.75 in.)
Initial core diameter	40.4 cm (15.9 in.)
Number of fuel assemblies	80 ^a
Wt ²³⁵ U per assembly	2.15 kg (4.74 lb)
Maximum neutron flux	3.7 x 10 ¹⁵ ncm ⁻² s ⁻¹
Safety rods	3
Compensating rods	2
Regulating rods	2
Control material	B ₄ C
Enrichment in ¹⁰ B isotope	80%
Maximum heat flux	2.65 x 10 ⁶ kcal/m ² h (9.77 x 10 ⁵ Btu/ft ² -hr)
Maximum can temperature with allowance for possible overheating	800°C (1472°F)
Effectiveness of three safety rods	3.3%—3 rods
Effectiveness of temperature compensator	1.0%—2 rods
Effectiveness of burnup compensator	1.7%—2 rods
Effectiveness of two control rods	0.48%
Temperature coefficient of reactivity	-4 x 10 ⁻⁵ /°C
Power coefficient of reactivity	1.5 x 10 ⁻⁴ /MW

^aThis is the value reported in E. V. Borisouk *et al.*, *Energie Nucléaire* 12(1), 33 (Jan-Feb 1970) for the initial charge. In A. I. Leipunsky and O. D. Kazuchkovsky, *Physics Notes Concerning Reactor Experience with Fast Neutrons in BOR*, SEE-187, Physics and Engineering Inst., Obninsk (1969), the number of fuel assemblies for initial wet critical is given as 68.

c. Core Design

The core of the BOR-60 reactor (see Figs. A.2.3 and A.2.4) consists of 80 hexagonal assemblies with a dimension across the flats of 44 mm (1.73 in.), arranged in a pitch of 45 mm (1.77 in.). Each assembly (see Fig. A.2.5) consists of thirty-seven fuel elements, arranged in a triangular lattice with a pitch of 6.7 mm (0.264 in.), giving a P/D = 1.1. The wall thickness of an assembly is 1 mm (0.0394 in.). The fuel pins are shown in Fig. A.2.6 and their characteristics in Table A.2.2. The fuel elements are spaced by means of oval spiral wrapped wires. Linear power density is 550 W/cm.

Criteria used in the design of the fuel include a limit of 2% creep strain in the cladding and the initial operating temperature of 500°C (932°F). The clad material is U.S.S.R.-modified Type 316 stainless steel: 16 Cr, 15 Ni, 3 Mo, and 0.8 Nb.

The radial blanket is made up of assemblies with an outside configuration similar to the core assemblies (see Fig. A.2.5 and A.2.6). Each contains seven blanket elements, 14.5 mm (0.571 in.) in diameter, arranged with a pitch of 15.2 mm (0.599 in.) in a triangular lattice (P/D = 1.05). The thickness of the blanket-element can is 0.35 mm (0.014 in.). The elements contain pellets of depleted uranium dioxide for a total length of 900 mm

(35.46 in.). The mean blanket thickness is 150 mm (5.91 in.). There are 259 assemblies altogether in the core and blanket.

The top ends of the assemblies are free, which ensures the possibility of free thermal elongation along the axial direction; movement in the radial direction is possible only within the limits of the accumulated engineering tolerances. Just as with EBR-II, the design did not accommodate swelling, and for a research reactor with a low ratio of flux to burnup swelling probably need not be taken into account at this time. Core and material behavior will be carefully monitored. The bottom end of the fuel assemblies is designed as a hydraulic holddown and has side holes for passage of sodium. Throttling of the flow of sodium to each assembly is in accordance with its heat output and is achieved by means of slots in the throttling sleeves of the inlet plenum, covering a given number of side holes in the bottom ends of the assemblies. The flow through the blanket elements is supplied from the low-pressure inlet plenum and enters vertically from the bottom of the subassembly; flow is controlled by an orifice located in the bottom of the subassembly (see Fig. A.2.5). A labyrinth seal is used to minimize leakage along the subassemblies from the high-pressure plenum (see Figs. A.2.5 and A.2.7). This labyrinth seal has the novel feature of using a wire spring in an acme thread. The pressure drop tends to tighten the spring and reduces leakage. It was pointed out that sometimes a subassembly has to be rotated a small amount to uncoil the spring from the wall.

Pertinent information about core design is given in Table A.2.1. The peak flux for BOR-60 is approximately the same as in EBR-II, namely, 3.7×10^{15} nv. A small pitch-to-diameter ratio for fuel spacing has been chosen to make this flux as high as possible and to simulate BN-350 and BN-600 conditions. The radiation times to obtain both fuel and material information will be the same as required in EBR-II. It is expected that several years will be required before a fluence of 10^{23} nvt is achieved. Significant swelling information should be available from BOR-60 by the end of 1972.

Variations of clad materials, of void deployment, and gap will be studied for both UO₂ and PuO₂ fuels. Up to 30 assemblies are to be inserted in BOR-60 prior to the end of 1970. The first group of tests will also include larger elements with lower heat power density to obtain information on blanket-material performance. The first group will contain only uranium oxide. A group to be installed at a later date will contain mixed plutonium and uranium oxides, and some will have ribs instead of wire spacers. Vented elements will be tested, but these have third priority. The Soviets pointed out that the first loading will

SYMBOL	TITLE	NO.
SAFETY ROD	3
REGULATING ROD	2
SHIM ROD	2

Fig. A.2.3. Core Design of BOR-60.

Fig. A.2.4. The BOR-60 Reactor. 1—Fuel exchange cask, 2—Drive for the control rod system, 3—Gate of fuel exchanger cask, 4—Fuel exit port of fuel exchanger cask, 5—Basket support for core, 6—Steel shielding, 7—Core, 8—Reflector, 9—Plates for the gamma heat shield, 10—High pressure plenum, 11—Reactor vessel, 12—Secondary containment, 13—Sodium inlet, 14—High-pressure inlet zone, 15—Low-pressure inlet zone, 16—Sodium outlet duct, 17—Supporting flange, 18—Control rod, 19—Thermocouple, 20—Rotating plugs, 21—Dip seal, 22—Rotating plug drive mechanisms.

Fig. A.2.5. BOR-60 Fuel and Blanket Assemblies.

Fig. A.2.6. BOR-60 Fuel Element.

TABLE A.2.2. Characteristics of Fuel Element and Fuel Pellet for BOR-60

Fuel Element	
Overall Length	1100 mm (43.31 in.)
OD	6.1 mm (0.240 in.)
Active Length	400 mm (13.76 in.)
Length of each Natural Uranium Blanket	100 mm (3.94 in.)
Length of Gas Plenum	
At bottom	150 mm (5.91 in.)
At top	100 mm (3.94 in.)
Clad thickness	0.4 mm (0.016 in.)
Thickness of Annular Gap	~0.1 mm (0.0039 in.)
Fuel Pellets	
Fuel	90% enriched UO_2
Diameter	5.1 ± 0.01 mm (0.201 in.)
Length	10 mm (0.394 in.)
Diameter of Central Hole	1.7 mm (0.072 in.)
Density	10.2 ± 0.2 g/cm ³

Fig. A.2.7. Inlet Plenum in BOR-60 Blanket and Fuel Assemblies.

include material variations for both cladding of the fuel element and for the subassembly materials as well as for reactor-vessel materials. Most of these materials are to be studied from a metallurgical point of view.

d. Reactor Compartment

The reactor vessel is cylindrical. The diameter at core level is only 1100 mm (3 ft 7.31 in.), and the maximum diameter is 1500 mm (4 ft 11 in.). The vessel is 6500 mm (21 ft) high and has a wall thickness of 16-22 mm (0.66 to 0.79 in.). The vessel part is enclosed in a leaktight housing to prevent draining the reactor in the event of a leak. The housing is also used for warming up the reactor with gas.

The vessel has one inlet duct 300 mm (11.82 in.) in diameter, welded into the elliptical bottom head and two outlet ducts, also 300 mm in diameter, above the level of the tops of the assemblies.

The internals of the vessel are made completely removable, as shown in Fig. A.2.4. The thermal sleeve for the

outlet pipes is split to accommodate this. The core support structure (see Fig. A.2.8), inlet plenums, and the vertical shielding are all fastened to the top of the vessel and removable as a unit. If necessary, these can be replaced, but maintenance procedures have not been worked out.

The sodium enters the reactor vessel through the bottom head. The inlet plenum is divided into a high-pressure and a low-pressure zone. Sodium flows from the high-pressure zone to the space between the support plates, where sodium then flows into the core subassemblies and to the low-pressure plenum. The low-pressure zone is below the lower support plate and acts to reduce the upward force on the core subassemblies, as well as the plenum area which feeds the blanket elements, the thermal shielding, and reactor vessel wall.

Fig. A.2.8. Core-support Structure of BOR-60. 1—High-pressure distribution header, 2—High pressure plenum, 3—Subassembly flow regulator, 4—Intake-manifold Structure, 5—Pressure-drop regulator for cooling control rod, 6—Pressure-drop regulator for low-pressure plenum, 7—Gasket, 8—Pressure-flow regulator for vessel-wall coolant, 9—Pressure-flow regulator for shield coolant, 10—Flange for radial restraint, 11—Core clamp (radial restraint).

The radial thermal shield (see Fig. A.2.9) between the core and the reactor vessel consists of concentric steel cylinders in sodium, to a total thickness of 24 cm (9.5 in.), which reduces the total flux at the inner reactor vessel wall to a calculated value of 7×10^{13} nvt. Below the core, shielding consisting of the lower fuel-assembly extensions reduces the calculated integrated flux at the vessel support plate to 1×10^{14} nvt. Measurements of neutron penetration through shielding configurations consisting of alternate layers of steel and graphite were made using the BR-5 beam-hole shielding facility at Obninsk; these measurements served to refine and confirm the computational methods used for the near-core shield design. The following are fluxes and fluences at the support plate and reactor vessel wall:

	Support Plate	Reactor Vessel Wall
Flux (nvt)	10^{14}	7×10^{13}
Fluence (nvt), 20 years, 50% plant factor	3×10^{22}	2×10^{22}

Some benefit can be obtained by thermal annealing of the support plate and the vessel. Swelling and creep at the support plate and the vessel wall for their operating temperature and fluence is not expected to be significant.

Radiation monitoring during early operation of the reactor indicated only a single inadequacy in the biological shield design, namely, excessive neutron streaming in the gap between the outer rotating shield plug and the reactor vessel. The resultant radiation dose rate in the reactor head cavity prevented access to the cavity during power operation. A design objective was to allow essentially unlimited access during operation for maintenance that may be required on the control-assembly drives. Additional boron-steel shielding has now been provided around the gap to permit about 25 min of personnel access into the reactor head cavity during full-power operation; it is planned to add sufficiently more shielding to permit several hours of access. Penetration shielding experiments conducted recently with the BR-5 shielding facility (see Part I of Appendix D) are providing guidance for the required shielding modifications.

e. Fuel Handling

Refueling is somewhat similar to EBR-II except that the fuel is removed directly from the core into a removal

coffin. Double rotating plugs are used. The plugs are made of alternate layers of steel and borated graphite to provide the necessary biological shielding. Gastight sealing of the plugs is achieved by the use of freeze seals fitted with a bismuth-tin alloy. During operation this alloy is solid. All the drives of safety and control elements, and the fuel-transfer-tank connection are on the small rotating plug.

The charging of new assemblies and the discharging of the spent ones are carried out through a special port in the small rotating plug. By rotating the large and small plugs, the port is rotated to the chosen core or blanket position. Then the charge-discharge machine is brought into place (see Fig. A.2.10). This is a coffin with a gripping device, mounted on a gantry crane and able to move along coordinate axes. The charge-discharge machine is connected and sealed to the port in the small plug; by means of the gripper, the assembly to be replaced is raised into the coffin in an inert atmosphere, in which it is transferred for external storage.

Transportation and insertion of new assemblies into the reactor are carried out by the same machine. Assemblies to be loaded are previously heated to about 300°C in an electric furnace.

Fig. A.2.9. Reactor Vessel Shielding in BOR-60.

Fig. A.2.10. Fuel Handling in BOR-60. 1—Container, 2—Argon-air heat exchanger, 3—Air blower, 4—Closure of container, 5—Bridge, 6—Carriage, 7—Compressor.

To take away the decay heat from the fuel element, there is a closed gas circuit in the charge-discharge machine, with the corresponding equipment. The decay heat to be removed is stated to be 5 kW. After the subassembly is lifted above the sodium level in the reactor, forced argon gas is used for cooling. The subassembly heats up at a rate at $200^{\circ}\text{C}/\text{min}$ ($360^{\circ}\text{F}/\text{min}$) in static argon, so there is at least a minute to establish cooling.

The fuel is cleaned in the transfer cask by steam and nitrogen. (At some later time, washing the subassemblies with lead will be tried.) The exact procedure for cleaning the subassemblies is to be developed over the next year. No refueling has taken place; however, four blanket elements have been removed and are in decay storage.

During fuel unloading, the pumps are interlocked to Power Flow operation, since the control rods can be floated out of the reactor if the pumps are at full flow. It was stated that a subassembly can be loaded into the reactor in 70 min and that the time for loading from storage to in-reactor position is 3 hr.

It was pointed out that the subassembly cooling line will be used to help detect failed cladding. The subassembly will be allowed to heat up in the cask and the gas-cooling line will be used as a sniffer line. Neon will be the element detected.

f. Components

In addition to the use of BOR-60 for the development of fuel materials, it is also the source of valuable experience for the heat-transfer components of larger plants.

1) Pumps

BOR-60 pumps are of the single-stage, single-suction Fermi type, capable of producing a flow rate 500 T/hr of sodium (2,200 gpm) with a total maximum pressure rise of 12 atm (176 psi). Continuous speed control from 350 to 1500 rpm is achieved by an ac-dc motor generator system; it was stated that the lower limit of 350 rpm was set to avoid rubbing of the hydrostatic radial bearing in the lower end of the pump. Pony motors for emergency cooling are not used; it was stated that battery power was available for 10 hr of operation subsequent to total power loss, which was considered the most serious credible accident.

The secondary pumps were observed in operation at 1000 rpm, and appeared to be both quiet and smooth in operation. It was stated that maximum amplitude of vibration was 3 to 10μ , although two resonances in the primary pump with a 60μ maximum amplitude had been discovered. These vibration levels were considered satisfactory to the designers and operators.

Shaft seals for the inert gas cover were stated to be graphite running against a nitrided shaft journal, lubricated with "turbine oil," presumably a hydrocarbon. "Fluorolub" was stated not to have been considered or tested.

2) Stop Valves

BOR-60 stop valves in 100-, 200-, and 300-mm (3.94-, 7.87, and 11.8-in.) pipe diameter had been tested in sodium (200 mm is main piping diameter). The valves were globe type, single-bellows-sealed with the bellows backed up by graphite packing. Shafts were backseated; Stellite was used on seats and disks. Valve bodies were rectangular forgings, bored out to accommodate inlet and outlet piping and the valve disk (sketch in Sect. III.D.). No apparent attempt had been made to reduce the mass of metal of the valve bodies, all corners were square. Each type of stop valve had been subjected to a 2000-hr test in sodium at working temperature (600°C or 1112°F) and pressure (12 atm or 176 psi), but no thermal-shock tests had been performed nor had sodium flow tests been made. The valve testing consisted of 1500 open and closing cycles, with a leak rate after test of 15 liters/hr ($\sim 0.5\text{ ft}^3/\text{hr}$) at temperature and pressure considered acceptable. Valve tests were carried out at Melekes in a sodium rig described in Appendix D.

3) Check Valves

The BOR-60 check valves (See sketch in Sect. III.D) were installed in line with the pump discharge but not integral with the pump. Though not observed, they were sketched and appeared to be fabricated again from forgings. The internals were removable through removal of a heavy seal welded plug to which the valve disk and hinge pins were attached. Removal of this plug and its attachments would permit access to the check valve seat. The pin was stated to be XH35BT material operating against a Stellite journal. The valve disks were stated to hang open 17° to permit convective cooling with minimum pressure drop. Disk motion in the check valve was damped by a vane counterbalance attached to the disk, operating in sodium.

4) Intermediate Heat Exchangers

The intermediate heat exchanger is similar to the Fermi type, with a vertical, removable tube bundle. The floating lower head contains the inlet plenum into which the secondary sodium is introduced by a central downcomer tube. Primary sodium is in the shell and is contained around the tube bundle by a shroud. Tube walls are 2 mm (79 mils) thick; it was stated that there were no baffles on the shell side of the heat exchanger, only three tube supports. The central channel was stated to accommodate differential expansion by use of a bellows in the lower end. The exchangers are blanketed with nitrogen.

5) Steam Generators

Both BOR-60 steam generators (see Fig. A.2.11) are of the same, once-through type as described in Ref. 2 (paper 3-1 of the London Conference). This is different from the previously reported plans to install a natural-circulation prototype of the BN-350 steam generator in one of the two loops. One steam-generator assembly was observed in place, with installation proceeding in parallel with reactor operation. It was stated that testing of the BN-350-type steam generator was done "at the factory". Steam-generator tubing was made up of tubing 8 to 12 m (26 to 39 ft) long, butt welded. The steam generators are in two sections, evaporating and superheating, both made of 2-1/4 Chrome-1. Moly unstabilized steel of 0.1 to 0.12% carbon content. Tubing is arranged in "serpentine" fashion. Steam exits from the evaporator section with 5°C (9°F) superheat into a separator which was stated to be capable of handling steam down to 90% quality, and is introduced into the superheating section. A volume of $350\text{ m}^3/\text{hr}$ (1540 gpm) of sodium flows through the shell sides of the steam generators with an average velocity of 1 m/sec (3.3 ft/sec). Feedwater is introduced at 215°C (419°F). A maximum

heat flux of $800,000\text{ kcal/m}^2/\text{hr}$ ($295,000\text{ Btu/ft}^2\text{-hr}$) and an average heat flux of $400,000\text{ kcal/m}^2\text{-hr}$ ($147,000\text{ Btu/ft}^2\text{-hr}$) were the design basis. Tubes were rolled and welded into the tubesheet with a trepanned T X 2T front-face welded joint. Tubes are of 16-mm (0.63-in.) OD with 2-1/2-mm (98-mil) wall thickness. Tubesheets are protected by steel plugs through which the tubes pass, and are fastened by bolts to the sodium side of the tubesheets with spacers, permitting a few mm of stagnant sodium between the plug and the tubesheet. Tubes are on a 25-mm (1 in.) square pitch and are a total length of 18 m (59 ft) in the evaporator section, being somewhat longer in the superheater section. There are 60 coils in the evaporator section and 90 in the superheater section. Tubes are supported by vertical plates with semicircular cutouts supporting individual tubes.

Relief from a sodium-water reaction was provided at the sodium-expansion tank, which was the only free surface in the system (the steam generators do not contain a gas space). A rupture diaphragm, about 10 in. in diameter, with an electrically operated puncturing knife actuated by a pressure transducer at the steam generator, is the means of pressure relief. Thus, pressures generated within the steam generator must travel from the steam-generator shell through about 15 ft of 200-mm (7.9 in.) piping to the expansion tank before venting. Although not then installed, it was stated that a tank for separation of sodium-water reaction products would be connected to the rupture-disk discharge piping.

6) Piping—BOR-60 and BN-350

Piping to carry sodium was stated to be of 18 chrome, 9 nickel composition. Such piping as was observed was mounted on supporting saddles, rather than being hung from spring hangers. No final conclusion should be drawn from this, however, as piping runs were incomplete in the BOR-60 steam-generator installation and barely started in BN-350.

A major innovation was the use of bellows in the BOR-60 200-mm (7.9-in.)-diam secondary system (see Fig. A.2.12). This permitted a compact piping arrangement in the steam-generator rooms, as observed on a model. Bellows were used only in vertical runs of hot-leg piping; they were stated to allow $\pm 150\text{ mm}$ (5.9 in.) of travel axially only and were set to be in the neutral position at operating temperature. The bellows were formed "U-span" type, of 8 plies of 0.35-mm (13.7-mil)-thick material, supported by rings in each convolution. Currently installed bellows, being used in the air-blast heat-exchanger system, are longitudinally welded plies. Those for use in the steam-generator installation will be formed of seamless

Fig. A.2.11. Direct-flow Steam Generator for BOR-60.

Fig. A.2.12. Secondary Piping Expansion Bellows (BOR-60 Representational Only—Not for Scaling).

material. One bellows available for inspection contained six convolutions in each of two sections. The bellows sections were covered externally with a guard ring for mechanical protection; stops were provided to limit axial travel of the bellows.

It was stated that use of bellows was not currently being considered for BN-350 or BN-600.

Thermal insulation for piping was stated to be primarily SiO_2 in fiberglass form. Electric-resistance heaters were used for preheating. In-line filters for preoperational sodium testing were provided (in addition to filters in dummy fuel elements). These in-line filters were stated to be made up of 8 layers of stainless steel mesh with holes 0.15 mm (6 mils) in diameter. Filters were utilized as the systems were stated not to have been "chemically cleaned" after assembly, and sodium was used as the cleaning agent.

Primary piping for BOR-60 and BN-350 was doubly contained out to the stop valves, with bellows accommodating differential expansion.

7) Sodium-purification Systems

The cold trap used for BOR-60 is generally of conventional design, but is steam or water cooled. A schematic drawing is given in Sect. III.F.

g. Operation

As stated in Sect. 2.a of this appendix, it was important to get the BOR-60 started as soon as possible to take advantage of any operating experience that might be applied to the design of BN-350 or BN-600.

The superintendent of BOR-60 has been quite pleased with operation to date. He had been in charge of ORBUS, an organic-cooled reactor, and found little difference between operating procedures of that and BOR-60. He and his operating crew were able to adapt quite readily to the change in reactors and were quite pleased that a sodium-cooled reactor seemed as passive to handle as an organic-cooled reactor.

Some of the experiences are worth relating. Sodium was shipped to the site in 1,000-liter (264 gal) drums; apparently the sodium was prevented from oxidizing by coating the sodium with a layer of paraffin. When the drums were heated, quite a bit of the paraffin was transferred along with the sodium into the plant storage tanks. It took two months to clean up the sodium from the paraffin. The sodium was put into 1,000-liter (264 gal) drums which were

heated to 200°C (392°F) for 3 hr, and a vacuum was pulled on the tank. This provided a "deslagging" process. A great deal of the paraffin adhered to the storage tank wall, and much of the paraffin accumulated on filters that were located at the bottom of the tanks. Sodium was transferred from these tanks into other, small vessels prior to transfer to the main sodium-storage tanks. The sodium was then circulated through the cold trap of the primary system, so it was expected that any residual paraffin was removed. The carbon content at the end of this time was measured at 40 ppm. Since the primary system is not to be operated at high temperature for some period of time, the Soviets expected that the sodium can be fairly well purified of hydrocarbons before operation at a high temperature breaks them down.

Prior to introducing sodium into the primary system, a vacuum was pulled on the system of 0.2 mm Hg. The core was loaded with filter dummy subassemblies similar to those used in Fermi, with a filter size of 0.15 mm (0.006 in.). The system was filtered for three days. The reactor was loaded with fuel in nine days, and after one month of minimum physics experiments the reactor was operated at 20 MW. At the time of our visit, the reactor had been operated for 45 days at 20 MW at a temperature of 350°C (662°F) and to a burnup of 0.6%.

Two problems of a thermodynamics nature were encountered during startup and preoperational analysis. The first has to do with the natural-circulation capability of the system. It was found that the plant is not capable of the natural-circulation heat removal expected; as a result, greater reliance needs to be directed to auxiliary power.

The second problem is that both piping stresses and stresses at the outlet nozzle of the reactor are greater under transient shutdown of the reactor than expected. The plant has no thermal heat capacity in the outlet of the reactor, and therefore reactor scram will cause very large transients to be felt by surfaces exposed to the reactor outlet temperature. The thermal stresses at the reactor outlet nozzle might have been aggravated by the difference in inertia of the vessel wall circuit and the core flow circuit, and by reduction of effectiveness of the nozzle thermal baffle by splitting it to make the basket removable. The primary system is outfitted with strain gauges capable of limited life at 400°C (752°F). Therefore, extensive transient tests were being conducted at 20 MW and a sodium outlet temperature of 350°C (662°F) to establish the actual strains in the system, and from this the mode of operation of the plant. Results at the time of the visit indicated that flow would have to be programmed on reactor scram to minimize thermal transients.

h. Physics Startup Testing

To expedite achievement of 20 MW, low-power physics testing was held to the minimum essential for assurance that the reactor could be taken to power operation safely. Measurements included distributions of spatial fission rate using ^{235}U foils and fuel pins, reaction rates in threshold and resonance activation detectors for neutron-spectrum analysis, and temperature coefficient of reactivity. Over most of the core the measured distribution of the radial fission rate agreed very well with that predicted from calculations, but the measured value was about 10% higher than that calculated for the core side of the core-reflector interface. The neutron spectrum deduced from the measurements (~ 350 keV mean fission energy) agreed overall with the calculated spectrum, although differences of the order of 10% occurred over small energy intervals. The measured temperature coefficient, $-4.1 \times 10^{-5} \Delta k/k/^{\circ}\text{C}$, due mainly to core radial expansion, fuel axial expansion, and sodium density reduction, was about 12% lower than that predicted by calculation. Additional physics testing, including reactivity oscillator measurements of the reactor frequency response, will be conducted when the reactor is started up again and taken through successive steps of increasing power level. Measurements will be made to detect, in particular, any changes in the reactivity coefficient that may occur with operating time and with power cycling, such as those observed in earlier small fast reactors (Rapsodie, Dounreay, and EBR-II).

Prior to startup a mockup critical experiment was run in the BFS-1 facility at Obninsk. The actual and predicted

critical masses of the assembly differed by only 3%; and this close agreement between calculation and experiment gave confidence in the predicted reactor fissile loading.

Some of the major operations to be undertaken are as follows:

(1) Before end of 1970:

- Insertion into the reactor of fuel subassemblies and materials, including an instrumented subassembly, for materials and fuel data on BN-350 and BN-600;

- Establishing safer and simpler reactor operations;

- Conducting steam-generator studies;

- Achieving first 450°C (752°F) outlet at 25-27 MW; and 60-MW operation at 500°C (932°F) at a flux of 3.7×10^{15} nv.

(2) Over the next year:

- Load some mixed-oxide fuels;

- Use of fuel-loading machine which can clean the assemblies with lead.

(3) Eventually:

- Operate with vented fuel.

3. THE BN-350 REACTOR AT SHEVCHENKO

The BN-350 is a 1000-MWt fast breeder reactor, providing the equivalent of 350 MWe in steam, part of which will be used for providing electricity (about 150 MWe) and the remainder in the desalination plant. This reactor plant is located several kilometers south of Shevchenko, on the Mangyshlak Peninsula of the eastern shores of the Caspian Sea (see Fig. A.3.1). This area is a desert with below-sea-level depressions. Manganese, phosphorites, sulphate and chloride salts, brown coal, oil, and gas are located in this region, providing the incentive to industrialize the area. Three oil towns of Yeralievo, Uziem, and Jetibuy will be served by the BN-350 complex. Large quantities of fresh water (not available sufficiently from wells) are required for the industries, population, and local agriculture. The new town of Shevchenko, located south of old Fort Shevchenko, now contains 60,000 people and is expected to grow to 250,000. It is essentially dependent on desalinated water.

Due to a variety of reasons, the level of the Caspian Sea, the largest inland body of water in the world (139,000 sq mi and 4,300 mi of shore line) is gradually falling. Since 1929, the Caspian Sea has decreased in area by about 6,000 sq mi, 80% of the decrease occurring in the northern Caspian. Predictions are that the sea will fall another 7ft by the year 2000. Reasons given were: (1) the main sources of supply, the Volga and the Ural Rivers, have been diverted in many ways to other uses; and (2) evaporation. The sea is as much as 3280 ft deep at the southern end, but only 15-30 ft deep at the northern end. The Kara-Bogoz-Gol section alone evaporates about 2 billion gallons per year.

a. Reactor Site and Buildings

The reactor site is about 3 km (1.9 mi) east of the Caspian Sea. The condensing water for the condensers in the desalinating plant and the condenser for the condensing turbine to be installed will be supplied by seawater. The seawater contains about 14,000 ppm of dissolved solids.

The reactor building, the steam-generator building, the control and service building, the decay pond, the T-G building, and the desalination plant are essentially above ground (see Figs. A.3.2 and A.3.3). For example, the BN-350 lowest level is only 13 ft below grade. This appears to be conventional construction for all thermal power plants in the U.S.S.R.

The reactor plant arrangement is in many respects similar to the U.S.S.R. conventional fossil-fueled power

Fig. A.3.1. BN-350 Site Location.

plants as evidenced by a comparison of Fig. A.5.2 (p. 83) with Figs. A.3.4 and A.3.2.

The framework of the buildings is prefabricated of concrete beams, except for the thick shielded walls, which are poured in place. Fairly ingenious methods are used to tie the columns together by welding studs at column intersections which are not located at column/beam intersections and by tying the beams to the columns by means of bolts.

The reactor building is not gas-tight. It is a mill-type building. Reliance for containment is placed on (1) the fuel element, (2) the reactor vessel and plug, (3) secondary containment around the reactor vessel and piping up to the stop valves, and (4) the biological shielding. Noteworthy are the hermetically sealed IHX units and the primary pumps—six steel-lined inert-gas-filled cells with cooling between the walls.

The large amount of space devoted to the nuclear-steam-supply system and its components is evidenced by the total of 10 million cu ft of space occupied by the reactor building or 10 cu ft/kWt allocated to this system. This compares with 2 cu ft/kWt for Fermi, a 450-MWt design. The large number of components accounts for this large volume of space (see Table A.3.1).

b. Reactor Core and Fuel

See Tables III.A.1 and III.E.1 for core and fuel data.

Annular pellets of UO_2 of 17 and 26% enrichment are loaded in a 6.1-mm (0.24 in.)-OD tube. The fuel pellet and fuel pin are shown in Fig. A.3.5. The fuel pin is welded to a

Fig. A.3.2. Vertical View of BN-350. 1—Reactor, 2—Water pool for fuel decay and storage, 3—Refueling cell, 4—Two transfer rotors, 5—Fuel wash stations.

Fig. A.3.3. Plan View of BN-350, Reactor Building Only. Symmetrical layout—3 loops to the right of the reactor; 3 to the left of the reactor. 1—Reactor; 2—Refueling cell; 3—Wash cell; 4—Three of 6 IHX compartments, 2 shells in each compartment; 5—Three of 6 steam generator compartments; 6—Secondary dump tanks. For B-B see Fig. A.3.17.

TABLE A.3.1. Number of Major Components in BN-350 Reactor Building

	No. of Components
Primary pumps	6
Secondary pumps	6
IHX shells	12
IHX tube bundles	36
Evaporators	12
Superheaters	12
Surge tanks	6
Cold traps	12
NaK-air heat exchangers	4
Large stop valves	24
Large check valves	6
No. of major components in reactor bldg. (excluding reactor plug, fuel handling, and control-rod drives).	136

NOTE: There are six primary sodium loops and six secondary sodium loops.

bottom grid and is free to expand upward. Annular pellets were chosen:

1. To operate fuel below melting.
2. To provide void volume other than in pellet-to-cladding gap or in fuel porosity.

Large tolerances on pellet-to-cladding gaps permit the use of unground pellets. The pitch-to-diameter ratio of about 1.15 is low in comparison to that for fast cores in the U.S. The physical separation of the axial blanket rods from fuel pins limits the gas volume and thus the burnup. In BN-600 the fuel pin incorporates the axial blanket and a longer gas plenum with consequent higher burnup limits, hopefully 10 at. %. The BN-350 core subassembly is shown in Fig. A.3.6.

Figure A.3.7 shows the core cross section. The 1870-liter core, with length-to-diameter ratio of about 0.7, operating at an average of 480 kW/liter, has a low sodium fraction of 31.1% and a high fuel fraction of 44.7%. The core comprises 200 hexagonal subassemblies; each subassembly is 3.5 m (11.5 ft) high and 96 mm (3.78 in.) across the flats. Spacing of 2 mm (0.079 in.) between the cans allows for only 0.5-mm (0.020-in.) free movement since pads at corners are 1.5 mm (0.059 in.) thick (see Fig. A.3.8).

Core restraint is provided by the core support plate, subassembly cans, radial blanket subassemblies, and radial shielding. The Soviets expect that in the next 2 years they will gain sufficient experience from BOR-60, BR-5, and some BN-350 operation to determine future core designs for BN-350.

Fig. A.3.4. Layout of Main Building of a Fossil-fired Steam Power Station with Type PVK-200 Turbines and 640 t/hr Boilers.

Fig. A.3.5

BN-350 Fuel Pins. Spiral spacer wire, 0.85 mm diameter, is welded at each end presumably to end cap. Pitch of spiral ≈ 100 mm. The pitch/diameter ratio = 1.15. The pellet-to-clad gap = 0.1 to 0.5 mm. Initial density = 10.4 g/cm^3 and smeared density = 8 g/cm^3 . The pellets are unground, not dished.

Fig. A.3.6. BN-350 Core Subassembly. 1—Handling head, 2—Axial blanket, 3—Core section, 4—Acme thread seal, 5—Nozzle, 6—Fuel pin.

Fig. A.3.7. Cross Section of BN-350 Core. 1—Safety rods, 2—Burnup rods, 3—Temperature-compensation rod, 4—Regulator rods, 5—Core fuel subassemblies, 6—Radial blanket—Inner zone, 7—Radial blanket—Outer zone, 8—Core fuel subassemblies—Peripheral storage, 9—Steel shield rods.

Fig. A.3.8. Cross Section of BN-350 Subassembly Can.

The subassemblies are held down against upward hydraulic forces by means of a hydraulic holddown incorporated in the support plates and the lower nozzle of each subassembly, as shown in Fig. A.3.9. A further detail is shown in Fig. A.3.10.

c. Reactor Vessel

The reactor vessel (see Figs. A.3.11-A.3.13) was shipped in nine sections by rail and water and assembled on site but not in place. The six sections shown in Fig. A.3.13 can be increased to nine shipped by halving each of the three large sections. The inside walls have been ground to a semi-polished finish. There are three plates on the circumference of the vessel at the sodium outlet. In each plate (about 5×20 ft), two nozzles were forged (pierced).

The vessel support, originally columns, appears now to be a cylinder, or else flexible plates.

d. Rotating Plugs

The two rotating plugs (see Figs. A.3.11 and A.3.12) rotate on ball bearings. A 138°C (280°F) melting-point tin-bismuth eutectic alloy is used in an electrically heated liquid seal during rotation, solid during operation, with a rubber seal as a backup only during normal operation.

There is a sealed machinery dome above the plugs, but only for venting of normal radioactive leakage.

e. Refueling System

The refueling system (see Figs. A.3.11, A.3.12, and A.3.14-A.3.17) consists of:

- (1) A double rotating plug with drives.
- (2) An in-vessel (vertical gripper) fuel-handling machine.

- (3) Two 17° inclined elevators for fuel movement between the reactor and a position below the refueling-cell handling-machine gripper.

- (4) An exit port to shielded refueling cell.

- (5) A refueling-cell fuel-transfer machine capable of pickup and deposit, and translation along the narrow refueling cell.

- (6) A shielded refueling cell.

- (7) Two transfer rotors cooled by NaK for interim transfer storage.

- (8) A shielded wash cell above the wash stations.

- (9) A fuel-handling transfer machine in the wash cell.

- (10) An exit port to the decay-storage water pool.

- (11) A decay-storage water pool.

- (12) Five exit-port plug-handling machines above the refueling and wash cell; some may also be used for transferring new fuel.

- (13) External drive for the new fuel-guide tube-support mechanism.

The in-vessel fuel-handling machine is shown in Figs. A.3.11, A.3.12, and A.3.14. The in-vessel machine can lift and rotate and has a holddown. The elevator, inclined 17° to the vertical, shown in Figs. A.3.11, A.3.12, A.3.15, and A.3.16, consists of a guide, a pot, a carriage, a drive

Fig. A.3.9. Schematic Diagram of the BN-350 Fuel-subassembly Hydraulic Holddown.

Fig. A.3.11. Vertical Cross Section of BN-350 Reactor.

Fig. A.3.10. Means to Control Leakage in BN-350 Core Subassembly. 1—Flow outlet, 2—Subassembly, 3—Seal (acme threads), 4—Upper support plate, 5—Coolant inlet to bushing, 6—Coolant inlet to subassembly, 7—Leak-through seal.

Fig. A.3.12. The BN-350 Reactor. 1—Reactor vessel, 2—Large rotating plug, 3—Small rotating plug, 4—Central shaft for control rods, 5—Refueling-cell fuel-handling machine, 6—Refueling cell, 7—Elevator, 8—Upper fixed shield, 9—In-vessel fuel-handling machine, 10—Core, 11—Reactor support, 12—Radial shield, 13—Gas heating inlet.

Fig. A.3.13. BN-350 Reactor Vessel Showing Sections Shipped.

Fig. A.3.14. BN-350 In-vessel Fuel-handling Machine Showing Sequence to Latch of Control Subassembly and Holddown of Six Adjacent Subassemblies.

Fig. A.3.15. BN-350 Refueling Elevator.

Fig. A.3.16. BN-350 Refueling System. 1—Large rotating plug, 2—Small rotating plug, 3—Reactor exit port, 4—Refueling-machine carriage, 5—Refueling cell, 6—Wash cell, 7—Wash ports, 8—Used assembly transfer rotor, 9—Fresh assembly transfer rotor, 10—In-vessel fuel transfer machine, 11—Elevator, 12—Wash-cell fuel transfer machine, 13—New fuel pot, 14—Refueling-cell fuel transfer machine.

Fig. A.3.17. Section B-B of Fig. A.3.3. Details of BN-350 refueling cell. 1—Plug removal hoist, 2—Wash cell, 3—Electrical supply line to emergency blower on refueling machine, 4—Assembly transfer mechanism, 5—Refueling cell, 6—Assembly transfer mechanism lead screw, 7—Rack drive, 8—Grapple (grab) drive, 9—Square or splined "feed" shafts, 10—Reactor exit ports, 11—Exit ports for transfer rotors, 12—Drive for guide tube support mechanism.

rod, and a chain transmission. The carriage travels 16.5 ft at 10 ft/min and requires a 0.83-kW drive.

Figure A.3.17 shows the refueling and wash-cell mechanisms.

The rack and grapple drives, one set located outside one end of the refueling cell, another outside one end of the wash cell, are connected to an Assembly Transfer Mechanism (ATM) lead screw and to "feed" shafts. The ATM operates the refueling-machine grapple; the "feed" shafts propel the refueling-machine carriage along the cell. The plug-removal hoist, of which there are five, probably also can move new fuel in and out of the "new fuel pot." The drive for the new fuel-guide tube-support mechanism shown in Section A-A of Fig. A.3.17 probably is used in the process. The five plug-removal mechanisms are located one above each of the exit ports, excluding the wash-cell ports. The refueling scheme sequence is shown in Fig. A.3.18. Refueling is accomplished with natural circulation of the sodium systems after 3 hr of shutdown at 500°F. Refueling of one-sixth of core at 2-month intervals (designed to maintain large breeding ratio and to minimize reactivity requirements) is by means of the double rotating plug and a vertical grapple in the eccentric plug which moves fuel between the core, the blanket, surrounding storage positions, and a transfer pot; two inclined (17°) refueling elevators contain a transfer pot at the bottom of the elevators. Each elevator can perform dual functions. The transfer pot, closed at the bottom, is always full of sodium and is always in a vertical position as it travels up to the part where the head of the fuel subassembly is grappled by the refueling-cell fuel-transfer machine grapple above the level of sodium. The windowless refueling cell is a narrow (about 1-meter-wide) shielded cell tangential to the reactor and just over the reactor sufficiently to provide for fuel exit and entrance from and to the reactor. The refueling-cell fuel transfer machine transfers fuel between the elevators and transfer-rotor containers cooled with NaK. Two transfer-rotor containers are available, one for spent fuel elements and one for new fuel elements. Subassemblies in the refueling cell are all handled directly in argon. The core fuel has decayed sufficiently in the reactor (about 2 months) to allow for natural circulation of argon in the refueling cell for fuel cooling. In an emergency, forced argon circulation can be provided in the cell. There are two cells—a refueling cell and a wash cell. The fuel-transfer machine in the wash cell can transfer fuel from the spent fuel-transfer rotor to the wash positions and to the exit port for the port-water decay-storage pool. The refueling mechanisms have been tested thoroughly through thousands of cycles. There are apparently no provisions for sticking of subassemblies or their misalignment.

Blocks to right—location of subassembly
Blocks to left—machines used

Fig. A.3.18. Movement of BN-350 Spent-fuel Subassembly.

Fig. A.3.19. BN-350 Primary and Secondary Sodium Systems. R—Reactor, P—Pump, IHX—Intermediate Heat Exchanger, S—Superheater, E—Evaporator, S.T.—Surge Tank.

f. Sodium Coolant Systems

BN-350 is cooled by six primary coolant loops and six secondary coolant loops. One loop for each circuit typical of all loops is shown in Fig. A.3.19. Two IHX tanks each containing three secondary tube bundles, are connected in parallel on both the primary and secondary sides.

Shutoff valves, sealed with water-cooled freeze seals, are located in each loop. Each primary circuit, the two IHX tanks, stop valves, pumps, check valves, and associated piping are located in a hermetically sealed steel-lined cell. Primary system material is equivalent to Type 304 stainless steel. The cells are cooled by an in-between-wall cooling system. Cell atmosphere is nitrogen.

Secondary containment is placed around the reactor vessel and the primary piping up to the stop valves.

The reactor vessel is heating with gas circulating between the vessel and the secondary containment. Electrical heating is used on the IHX.

g. Intermediate Heat Exchangers

Each Intermediate Heat Exchanger (see Fig. A.3.20), of which there are 12, consists of a flat-sided shell, strong-backed as shown in Fig. A.3.21, with three tube bundles which terminate in three biologically shielded heads providing access to the tube bundles.

One IHX assembly (two IHX's) was observed in place, but not piped up at BN-350. Only the casings were installed. U-tube bundles in which the secondary sodium flows were delivered in sealed containers ready for installation when the plant is substantially complete. These tube bundles are removable. Shielding caps permit disconnecting and removal of a vertical bundle, working directly over the bundles, and avoiding direct radiation from residual pri-

mary sodium. Also noteworthy was the heavy reinforcement by beams of approximately 3-in. web width on the flat vertical side of the IHX shells (see Fig. A.3.21). These beams were welded together in a connected pattern approximately 1 ft apart vertically and horizontally and bolted to the shell at the edges. Thermal insulation and electric resistance heaters were to be attached to the outside of these beams.

h. Primary Sodium Pumps

The primary sodium pumps (shown in Fig. A.3.22), about 330 ft of head, each with a drive weighing about 86 tons and a rated delivery of 14,000 gpm, are driven by a squirrel-cage 1700-kW motor of two speeds (1000 and 250 rpm). Sodium level is maintained at least 12 in. above the impeller; the motor is not started below this level which is maintained by a level regulator located in a separate tank adjacent to the pump. Main bearings are hydrostatic with seven self-adjusting, oil-lubricated segments. Facings are stellite. The bearing design is based on BOR-60 data; the largest bearing tested is 150 mm (5.9 in.) in diameter. Pumps were tested full scale at "the factory." Ten-thousand-hour tests of the oil-lubricated graphite against a nitrided steel gas seal have been successfully performed.

i. BN-350 Stop Valves

The 20- and 24-in.-diam electrically-operated stop valves (manual override) on the BN-350 were not installed, but at least eight had been delivered to the site and were observed on the reactor floor. They are double-disk wedge-type gate valves with a drainable cavity between the disks, and with water-cooled NaK freeze seals around the stem, both in the primary and secondary stop valves. The freeze seals were backed up by packing of an undetermined

Fig. A.3.20. BN-350 Intermediate Heat Exchanger. 1—Biological shield, 2—Tube bundle, 3—Shell.

Fig. A.3.21. BN-350 IHX Strongback.

Fig. A.3.22. BN-350 Primary-system Sodium Pump.

type. Valve bodies appeared to be bored-out forgings of stainless steel. These large forgings had not been machined on the outside, but were forged down to roughly cylindrical shape with approximately 1-in. flats. No information was available from site personnel concerning tests, as all testing was done at "the factory" and valves were delivered ready for installation. The valve inlets and outlets were capped with rather loose-fitting stainless steel dust covers. Paper 5B1 of the 1966 Fast Breeder Reactor Conference in London* stated that BN-350 valve freeze seals would be NaK-cooled; it was emphasized at the site, however, that the valve seals were water-cooled. Leakage of valves on test is less than one-half pint of sodium per hour.

j. BN-350 Check Valves

The BN-350 check valves are similar to BOR-60 design, with counterbalanced disks.

k. Steam Generators

The steam generators (see Fig. A.3.23) consist, for each of six secondary loops, of two evaporators and two superheaters in parallel. One surge tank is connected from the gas space of the two evaporators as a means to vent the reaction products of a sodium-water reaction.

One set of steam generators was observed in BN-350, mounted but not piped.

The general arrangement is as previously published, using bayonet (Field) tube evaporators and counterflow U-tube superheaters. No separator is installed between the evaporator and superheater, steam separation taking place above the upper tubesheet in the evaporator section by double rings of W-type steam separators. The downcomer tube inside the evaporator bayonet tube is attached to a 600-mm-long fitting projecting above the tubesheet by a duct which permits the introduction of feedwater into this central downcomer tube (as shown in a sketch on page 21). Bayonet plugging is accomplished by removing the fitting, which when withdrawn brings out the central tube also. The bayonet is then plugged. Bayonet tubes are rolled and welded into a front-face-weld, trepanned t X 2t joint. There are six steam-generator units; five furnish full power, one serves as a spare. Evaporator and superheater material is low alloy (1 wt % chrome, 2 wt % molybdenum).

Provisions for sodium-water reaction relief are similar to those in BOR-60, with an electrically operated knife puncturing a diaphragm in the sodium expansion tank.

Steam generator design details for BN-350 are as follows:

Evaporator heat-transfer area (per pair)	800 m ² (8611 ft ²)
Superheater heat-transfer area (per pair)	500 m ² (5382 ft ²)
Evaporator thimble length	5 m (16 ft 5 in.)
Thermal capacity (per unit)	200 MWt
Steam production	600 tonnes/hr (1.6 × 10 ⁶ lb/hr)
Steam pressure	50 atm (735 psi)
Steam temperature	435°C (815°F)
Feedwater temperature	158°C (316°F)
Evaporator-thimble diameter (OD)	32 mm (1.26 in.)
Evaporator-thimble thickness	2 mm (0.079 in.)
Evaporator-downcomer-tube diameter	16 mm (0.63 in.)
Evaporator-downcomer-tube thickness	1 mm (0.039 in.)
No. of evaporator tubes per evaporator	816
Evaporator-tube triangular spacing	44 mm (1.73 in.)
Superheater-tube diameter	16 mm (0.63 in.)
Superheater-tube thickness	2 mm (0.079 in.)
No. of superheater tubes per superheater	805
Superheater-tube triangular spacing	23 mm (0.91 in.)
Evaporator recirculation ratio	4:1
Evaporator steam quality	1/4% moisture
Evaporator heat flux (maximum)	900 × 10 ³ kcal/m ² -hr (3.3 × 10 ⁵ Btu/ft ² -hr)
Superheater heat flux	40 × 10 ³ kcal/m ² -hr (1.5 × 10 ⁴ Btu/ft ² -hr)
Shell-side sodium ΔP	0.93 kg/cm ² (13.2 psi)

l. Sodium Service Systems

Sodium is brought to the site in 1-m³ (264-gal) drums, heated in electric furnaces, and recirculated through cold traps to less than 50 ppm of oxygen by weight (see Fig. A.3.24). Two sodium drain tanks per three loops are used in the secondary system. Twelve vertical drain tanks are used to drain the primary system. Service valves are

*Leipunsky, A. I., et al., "Sodium Technology and Equipment in the BN-350 Installation," *Proceedings of the London Conference on Fast Breeder Reactors Organized by the British Nuclear Energy Society, May 17-19, 1966*, Pergamon Press, Inc., New York (1967), p. 797.

Fig. A.3.23. BN-350 Steam Generator. 1—Saturated steam inlets, 2—Superheated steam, 3,7—Sodium inlet and outlet, 4—Feedwater inlet, 5—Water level, 6—Sodium level, 8—Burst diaphragm, 9—Evaporator, 10—Superheater.

bellows sealed. Einstein-Szilard pumps are used in the sodium service system. Plugging indicators are the sole means of checking oxygen on-line.

m. Status of BN-350 as of June 21, 1970

The BN-350 (MWe equivalent) plant is in various stages of construction.

(1) As shown in Fig. A.3.13, the reactor vessel is in place. The portion containing the rotating plug is not in place. The U.S. Delegation was the first foreign delegation in the vessel. There were no reactor-vessel internals.

(2) The refueling cell was on the floor but not in place, and not shielded.

(3) No refueling mechanisms or other reactor internals were in place or visible.

(4) The reactor radial thermal shield is being assembled at the site in six sections, which will be welded in place. One section was being assembled on the floor.

(5) The core support plates are on site.

(6) The 12 IHX shells are in place.

(7) Some of the IHX tube bundles are on the reactor floor in hermetically sealed tanks, under inert gas, ready to be put into the shell and welded. The estimate is that it will take 1-2 weeks to complete this job.

(8) All primary and secondary pump tanks are in place. Pump drives or internals were not visible.

(9) Some of the stop valves (two per circuit) and check valves (one per circuit) are on the floor or in place.

(10) Not all, but a major portion, of the primary piping is in place.

(11) Since the Delegation was shown an uncompleted steam generator assembly, it may be assumed that remaining units are not in place.

(12) The reactor machinery dome was not visible.

(13) Two of the four secondary dump tanks were in place. The status of the other two is unknown.

(14) The 12 vertical primary loop dump tanks were not visible.

(15) Four air dumps for NaK-to-air cooling are in place.

(16) Of the four turbine-generator units, one has been operating, supplied by an oil/gas boiler. The other three units are not in place and were not visible in the plant or in the yard.

(17) Of the three operating rooms—(1) main control room, (2) refueling control room, and (3) auxiliary systems control room—only the third had any equipment; this consisted of partially installed panels.

(18) Though electric furnace heaters for cold traps were in place, there was no evidence that the cold traps were in place.

(19) Only a small portion of electrical-instrumentation wiring appeared to be in place.

(20) The reactor superstructure was essentially completed. The turbine-generator building still requires considerable work. Columns and beams were in place for the three remaining turbine-generator units, but there were no wall panels, exterior or interior.

n. Schedule

Director Yurchenko stated that power operation of BN-350 will start in December 1971.

Further information indicated that construction will be completed at the end of 1970, that criticality could be achieved by mid-1971, and approach to power would take place during the last half of 1971.

Fig. A.3.24. BN-350 Sodium Receiving System. 1—To vacuum, 2—Gas outlet, 3—Gas inlet, 4—To secondary system drain, 5—Secondary system drain tanks, 6—To atmosphere, 7—NaK outlet, 8—NaK inlet, 9—Electromagnetic pump, 10—Oxide plugging indicator, 11—Plugging-indicator heat exchanger, 12—Air blower, 13—Cold trap, 14—Transport container, 15—Steam trap.

Fig. A.4.1. Details of Desalination Plant at Shevchenko.

4. THE DESALINATION PLANT AT SHEVCHENKO

The town of Shevchenko has a population of 60,000 (north of BN-350 site, as shown in Fig. A.3.1) and is dependent on desalinated water. Consumption is about 120 gal/day per person. Desalinated water is also used to supply industry and agriculture in the area, in particular, three oil towns of Yeralievo, Uziem, and Jetibuy. At present there are four desalination banks, two in the older desalination plant north of the BN-350 site and two at the BN-350 site. The BN-350 site has two banks of five evaporators each. The first bank has been operating since December 1968, and the second since June 1969. They are now supplied with steam from a 50-MWe oil/gas-fired fossil unit, operating at 60 MWe.

a. Inlet and Outlet Sea Canals

Caspian Sea water (14,000 ppm total dissolved solids, 13-15°C; 55.4-59.0°F) is received at the site by an open channel having a 3-km-long (1.9 mi) inlet, as shown in Fig. A.4.1. The outlet is a natural open channel discharging into the Caspian 20 km (12.4 mi) from the plant. Ecological disturbances are considered to be minimal.

b. Desalination Bank Schedule

The desalination bank schedule is as follows:

Now installed	m ³ /day	10 ⁶ gpd
Bank No. 1		
Oct 1963		
At old site		
4 evaporators	5,000	1.2
Bank No. 2		
April 1967		
At old site		
5 evaporators	15,000	3.5
Bank No. 3		
Dec 1968		
At BN-350 site		
5 evaporators	15,000	3.5
Bank No. 4		
June 1969		
At BN-350 site		
5 evaporators	15,000	3.5

Now being installed

	m^3/day	10^6 gpd
Bank No. 5		
At BN-350 site		
10 evaporators		
(10 on site;		
2 in place)	17,000	4.5
Bank No. 6		
At BN-350 site		
10 evaporators	17,000	4.5

Now planned at BN-350 site

Bank No. 7		
At BN-350 site		
10 evaporators	17,000	4.5
Bank No. 8		
At BN-350 site		
34-stage evaporating unit	15,000	3.5
Total	116,000	28.7

The final distillate of 3-3.5 mg of salt per liter is mixed with well water to obtain mineralized drinking water, increasing mineral content to 30-50 mg/liter. The degradation for other uses was not given.

The desalination units to date are of the multiple-effect vertical-tube-type evaporation units in which the seawater boils in each of the stages using the regenerative principle in all but the first evaporator unit, where direct heat is applied in the form of steam.*

c. Cost of Distilled Water

The Soviets estimate that desalinated water from the No. 3, 4, and 5 banks (10 evaporator units each) will cost 15 to 18 kopecks per cubic meter. This is 60 to 72 kopecks per 1000 gal or, at \$1.11 per ruble, 67 to 80 cents per 1000 gal. Based on a 2 to 4 factor for wages and other

indexes, this could actually amount to \$1.34 to \$3.20 per 1000 gal, based on a U.S. economy.**

d. Performance

The following performance in terms of heat input per cubic meter of distilled water has been experienced or predicted at the desalination plant on the BN-350 site.

Bank No. 1. 0.12 to 0.13 Gcal/m³—experience†
(~235 Btu/lb)

Bank No. 2. 0.12 to 0.13 Gcal/m³—experience
(~235 Btu/lb)

Bank No. 3. 0.073 Gcal/m³—predicted
(~135 Btu/lb)

Bank No. 4. 0.073 Gcal/m³—predicted
(~135 Btu/lb)

Bank No. 5. 0.073 Gcal/m³—predicted
(~135 Btu/lb)

Bank No. 6. 0.065 Gcal/m³—predicted
(~120 Btu/lb)

e. System Operation

The vertical-tube distillation process for obtaining distilled water operates on the principle shown in Figs. A.4.2 and A.4.3. Steam from the power plant at about 260°F flows on the shell side of the first vertical heat exchanger (evaporator), condensing as it circulates down and around the tubes, transferring heat to the brine flowing upward inside the tubes. The condensate returns eventually to the power plant after being utilized to an extent in regenerative-type heat exchangers. The steam evaporated from the seawater in the first evaporator flows to the shell side of the second evaporator, where its latent heat of condensation is used to evaporate the brine from the first evaporator. In this process, the temperature of the steam produced is reduced in stages:

Evaporator No. 1: 100-105°C (212-221°F)

Evaporator No. 2: 90-92°C (194-197.6°F)

Evaporator No. 3: 79-82°C (174-177.6°F)

*Bank No. 8 is multistage flash. Hot seawater is flashed to vapor successively in chambers at progressively lower pressures with the condensate heating feedwater in countercurrent flow.

**See Plant Design for basis of economics.

†1 Gigacalorie = 10⁹ calories = 1162.8 kWh; 1 m³ = 264.17 U.S. gal.

- ① POWER PLANT TURBINE EXHAUST
- ② EVAPORATOR
- ③ REGENERATIVE HEAT EXCHANGERS
- ④ FLASH POTS
- ⑤ SEED SETTLERS
- ⑥ MAIN CONDENSER
- ⑦ AUXILIARY CONDENSER

- ⑧ DEAERATOR
- ⑨ NONCONDENSIBLE GAS EJECTOR
- ⑩ DISTILLATE COOLERS
- ⑪ PARTICULATE FILTER
- P PUMP
- ⋈ VALVE

- BRINE
- CONDENSATE RETURNING TO POWER PLANT
- STEAM FROM BRINE
- STEAM FROM POWER PLANT
- FRESH WATER
- SLURRY OF CHALK
- NONCONDENSIBLE GASES

Fig. A.4.2

Shevchenko Desalination Plant—Scheme
of Bank of Five Evaporator Units at
BN-350 Site.

Evaporator No. 4: 66-68°C (151-154.4°F)

Evaporator No. 5: 44-45°C (111-113°F)

To achieve such steam temperatures requires approximately the following saturated-steam pressures. (The boiling point of seawater is in the order of a degree above that of fresh water.)

Exit top of No. 1: 15-17 psia

Exit top of No. 2: 10.5 psia

Exit top of No. 3: 7.0 psia

Exit top of No. 4: 4.0 psia

Exit top of No. 5: 1.4 psia

In other words, units 2 through 5 operate at a vacuum in the separator units, down to 2.7 in. of mercury in evaporator No. 5.

The average heat-transfer coefficients achieved in a five-evaporator unit are approximately as follows:

Evaporator No. 1: 630-640 Btu/hr-ft²-°F

Evaporator No. 2: 515-565 Btu/hr-ft²-°F

Evaporator No. 3: 500-530 Btu/hr-ft²-°F

Evaporator No. 4: 390-410 Btu/hr-ft²-°F

Evaporator No. 5: 205-270 Btu/hr-ft²-°F

f. Brine Circulation System

Let us now follow the path of the Caspian seawater. It enters a 3-km (1.9-mi)-long open-channel inlet, is screened, and then enters a distribution pool (as shown in Fig. A.4.1). About 3000 T/hr (12,000 gpm) are pumped through the seawater pump house from this pool to the condenser (see Fig. A.4.2) and to the desalination plant. Of the 3000 T/hr (12,000 gpm), about 700 T/hr (2800 gpm) are bypassed to the desalination plant. The seawater contains 13.5 g of salts per liter of water. It is first deaerated in the deaerator, then filtered for particulate matter, and then passed through a series of reheaters before entering the first evaporator.

g. Prevention of Scale Formation by Seeding with Chalk

To prevent or reduce scale deposits on heat-exchanger surfaces (which deposits mainly consist of calcium sulfate, calcium carbonate, and magnesium hydroxide), a slurry of

Fig. A.4.3. Evaporator for Shevchenko Desalination Plant. Heat exchanger—No. of tubes: 3000; tube OD: 1.5 in.; tube material: copper-nickel alloy, NNI 14-1; shield and downcomer material: 18-8 stainless steel; separator-tank material: upper, above brine, carbon steel—lower, in brine, stainless steel; all numbers are approximate.

chalk is pumped into the brine system just after the brine pumps. This slurry amounts to about 500 to 620 T/hr (2000 to 2480 gpm) or about 20 g of chalk per liter of brine. This chalk is analyzed as:

	Initial Charge, %	After a Period of Operation, %
CaO	54.13	46.1
MgO	0.17	7.5
SiO ₂	1.74	1.8
P ₂ O ₃	1.21	0.9
Roast losses	42.60 (mostly CO ₂)	43.63
IO ₃		3.07

The crystals in the chalk act as seed crystals for CaCO₃ and Mg(OH)₂ salts in the brine. Preferential sedimentation of these compounds on the seed crystals occurs if the boiling temperature of the brine in the first evaporator is kept below 115°C (239°F) and the brine salt content in the last evaporator is kept below 50-60 g of salts per liter of brine. Such preferential settling reduces precipitation of brine salts on evaporator tubes. The conditions stated also prevent the precipitation of CaSO₄ by maintaining solubility of CaSO₄ in the circulating solution.

The use of chalk, in preference to sulfuric acid treatment used in the U.S.A., is a matter of economics. As the brine is recirculated by natural circulation in each of the evapora-

tors, its salt concentration is increased by evaporation. To further prevent scale formation inside the tubes of the evaporator's heat exchanger, the boiling zone is maintained above the upper tube sheet of the heat exchanger by submerging the tube sheet below the brine level. The brine first enters a standpipe section above the upper tube sheet, and the submergence head is sufficient to prevent brine boiling inside the tubes. The Soviets indicated there have been no significant deposits of scale in heat-exchanger tubes since start of operation in 1963. The deposit that does occur is easily cleaned.

A diffuser is placed over the brine standpipe to more uniformly distribute the vapor stream across the separator surface. The separator is a reverse louver type. The first evaporator may contain a 2-ft layer of Raschig rings (about $15 \times 15 \times 2.5$ mm, i.e., $0.59 \times 0.59 \times 0.1$ in.) above the separator to further ensure moisture separation.

The brine from the fifth (last) evaporator goes into two parallel conical-type settlers, where the chalk seed and the salts the seed picked up can settle into a slurry and be repumped back into the system. Excess amounts of settled material can be withdrawn by a blowdown. The cleared brine with a salt content about 50 g/liter flows out through a cylindrical trough on the top periphery of the cones. The carbon steel conical settlers, coated on the inside with epoxy, are about 40 ft in diameter and 28 ft high.

Figure A.4.3 shows outside downcomers in each evaporator. These downcomers are sized to ensure high recirculation ratios of 100-200. High recirculation ratios measurably reduce the amount of seawater required in the total process, making each evaporator unit quite effective. These high recirculation rates also ensure high velocities within the evaporator heat-exchanger tubes, thereby increasing heat-transfer effectiveness. A pump in the last evaporator promotes circulation and increases the heat-transfer effectiveness. The pump is not shown in the schematic diagram.

h. Venting of Noncondensable Gases

To improve heat-transfer coefficients, and to equalize operation of adjacent evaporators, it is necessary to adequately vent noncondensable gases. The vent system for this, including the ejector, is shown in Fig. A.4.2.

i. Bank of 10 Evaporator Units

The first of three banks of 10 evaporator units each is now being installed. The 10-unit bank, as compared to a five-unit bank, will provide for a further decrease in final temperature in the 10th evaporator, about 22°C (72°F). The performance of the plant in terms of gigacalories of heat per ton of distilled water for the 10-unit bank is 0.073, as compared to 0.130 for the five-unit bank. This is accomplished by:

(1) Larger number of evaporator units; lower vacuum with lower last-stage temperature.

(2) Increased heat transfer per unit of heat-transfer surface, using roughened tubes or a similar effect tube structure.

For the forward-feed type of distillation using evaporators as in the first banks at Shevchenko, the performance ratio is proportional to the number of evaporators in series. Therefore, the performance for the 10-unit bank should be improved by a factor of two in comparison with the five-unit bank.

j. 34-stage Flash-distillation Bank

For a forward-feed evaporator-type distillation unit, (1) the performance ratio is not independent of the number of stages (evaporator units); (2) in general, the surface utilization cannot be increased; (3) the auxiliary equipment such as preheaters keeps increasing; and (4) engineering is complex. The evaporating surface can be eliminated by flash distillation. The Soviets are installing a 34-stage flash-distillation unit after the first five banks of multieffect evaporator units are installed. For flash distillation, the performance ratio R is approximately equal to n/j , where n = total number of stages and j = number of heat-rejection stages. R is now dependent on n/j . The optimum value of n will occur when the cost of dividing the plant into n stages overtakes the benefit of reduced heat-transfer surface. The U.S.S.R. will continue to use seeding in the flash-evaporation bank as a means of preventing scaling. The heat utilization in gigacalories per cubic meter of distillate will further decrease. The inlet seawater to the first stage will again be about 105°C (221°F).

Fig. A.5.1. General View of Model of Kurchatov Station at Beloyarsk. Building 1 is the BN-600 unit. Building 2 is superheat units 1 and 2. Building 3 is a plant (electrolytic) for making hydrogen used to cool the generators. Building 4 is for fuel storage. Building 5 is for general stores and instrumentation. The complex at building 6 (existing) handles liquid and dry irradiation waste. Area 7 is a burial ground with capacity for 10 primary cold traps. Building 8 is for feedwater treatment.

5. THE BN-600 REACTOR AT BELOYARSK

A highlight of the visit to Russia was the visit to BN-600 at Beloyarsk and the associated discussions of the design and status of the plant. This reactor represents the most advanced design thinking in Russia on large fast reactors.

a. Status

The plant has been under construction since about the end of 1968. A firm schedule for completion has not been set, apparently because of uncertainties in the delivery dates of some components (supplied by several different ministries). The estimated completion date was stated to be mid-1973 to mid-1975. Construction work was proceeding at a moderate pace with a construction force of about 450 workers. About half of the stated $60,000 \text{ m}^3$ ($78,477 \text{ cu yd}$) of concrete are in place. Construction of the reactor building is up to about 12 m (39 ft) above grade, and slightly above the bottom of the cell containing the main reactor tank. (See Figs. A.5.8-A.5.10 later.) The bottom and side walls of the steel cell liner have been erected. The framework of the steam generator and turbine

buildings are complete, but none of the major components are yet on the site, except for 150-m^3 (39,626 gal) sodium dump tanks.

b. Site

The station is situated immediately alongside the two operating units of the Beloyarsk power station, on a site about 1 by 2 km (0.62 by 1.24 mi) (see Fig. A.5.1) and about 2 km (1.24 miles) from a sizable village. Cooling for all three units is provided by a man-made lake of about 10 km^2 (2500 acres) area, which is also used for recreational purposes.

c. Summary of General Design Features

The reactor, primary pumps, and intermediate heat exchangers are mounted in a primary tank 12.8 m (42 ft) in diameter. The tank is supported by a ring girder, which in turn supports a bridge-truss structure spanning the bottom of the tank. The main components in the tank (except the top deck and the rotating plugs) are supported from the truss structure inside the tank. Differential expansion of the

truss and the top deck are accommodated by bellows up to 2.7 m (9 ft) in diameter, which take up to 1.5 cm (0.6 in.) of lateral displacement. The bellows are situated above the top deck and are conveniently inspectable and replaceable.

The core is surrounded by 0.5 m (1.64 ft) of lateral blanket and steel shielding, which nests into a primary flow shroud. This is surrounded by steel cylinders for additional lateral shielding of the main components and tank.

Refueling is by means of a narrow open cell, which partly overhangs the main tank. Three relatively simple mechanisms are used.

(1) A vertical grapple, mounted on the double rotating plug, which lifts fuel and places it in a storage ring at the edge of the blanket.

(2) An elevator, which raises the fuel and moves it laterally to a position under the refueling cell. (Steps 1 and 2 are done under sodium.)

(3) A vertical grapple, which lifts fuel from the elevator into the argon-filled cell and translates within the cell to storage, cleaning, and shipping-cask stations.

The reactor is situated in a vault with a carbon steel

liner. No "containment" is provided for large accidents, although the reactor building is probably vented through a stack, in accordance with usual practice for reactors in Russia. The pumps and intermediate heat exchangers appear to be of conventional design. The steam generators are scaled-up versions of the BOR-60 units, although one alternative is to use small modules similar in design to those for BOR-60. Evaporating and superheating shells are joined in an unbalanced "H" configuration, and a third shell provides reheating. Provisions for sodium-water reaction are made; the design basis is that only one tube rupture is accommodated by the relief system.

d. Plant Description

The fast reactor is called Unit 3 of the Kurchatov Station at Belayarsk, immediately adjacent to the west of the presently operating superheat Units 1 and 2.

Figure A.5.1 shows a model of the site, with the cooling reservoir off to the right (west). There is considerable sharing of facilities between BN-600 Unit 3 (to the left) and Units 1 and 2 (the two stacks on the right).

Figure A.5.2 shows a model of the BN-600 unit itself. Area 1 is the reactor building, about 47 m (154 ft) high. Area 2 is the steam generator (and auxiliaries) building, and

Fig. A.5.2. View of BN-600 Model from South. Area 1 is the reactor building, about 47 m (155 ft) high. Area 2 is the steam generator (and auxiliaries) building. Area 3 is the turbine building. Area 4 is the switchgear yard (220 kV).

Fig. A.5.3. Section View of BN-600 Plant.

3 is the turbine building. Area 4 is the switchgear yard (220 kV). Figure A.5.3 shows a general section view of the plant, including the reactor building on the right, the turbine hall on the left, and the steam-generator building in between. An enlarged version of the two-shell steam generator used for BOR-60 is visible.

Figure A.5.4 shows the primary tank and nozzles for primary pumps, IHX's, and secondary sodium piping and refueling ports. A bottom ring girder supports the primary tank. The "Top-hat" structure encloses the control and instrumentation mechanisms.

Figure A.5.5 shows a schematic sectional view of the primary tank, reactor core, and IHX's. A ring girder supports the outer main tank (by means of rollers) and in turn supports the inner tank. A nozzle in the outer tank provides for heating (or cooling) the inner tank with gas flow. The inner tank contains a massive truss also supported

by the ring girder. The reactor core, IHX's, radial shielding and pump shells are all supported by the truss structure. Only the double eccentric shield plug is supported by the tank itself.

Figures A.5.6 and A.5.7 show a schematic diagram of the refueling system and a model of the reactor.

Figure A.5.8 shows a general view of the plant construction. The steel liner of the main reactor vault is visible in the center, protruding above the partially completed reactor building. To the left is one of the sodium fill and dump tanks [150 m³ (39,626 gal)]. To the right is the framework of the steam-generator building. Figure A.5.9 shows a closer view of the reactor vault liner. Figure III.A.1 shows the steam-generator building with the turbine building behind it. Figure A.5.10 shows the turbine building, which is structurally almost completed.

Fig. A.5.4. Primary Tank and Reactor Cavity, BN-600.

Fig. A.5.5. Sectional View of Primary Tank, Reactor Core, and IHX's for BN-600.

Fig. A.5.6. Fuel-handling System Section for BN-600.

Fig. A.5.7. Model of BN-600 Reactor, Primary Tank and Refueling System.

Fig. A.5.8. General View of BN-600 Plant Construction, June 1970.

Fig. A.5.9. BN-600 Primary Reactor Cell and Primary Vault Liner under Construction.

Fig. A.5.10. Steam Turbine Building for BN-600.

e. Core and Fuel Cycle

The general characteristics of the reactor are summarized in Table A.5.1 (compiled partly from *The BN-600 Fast Reactor**).

The annular fuel-element diameter and wall thickness have been increased (relative to BN-350 design) to 6.9 mm (0.272 in. versus 0.25 in. for BN-350) with a thicker wall (0.016 in. versus 0.014 in. for BN-350). A large 800-mm (2.62-ft) gas plenum has been added (versus almost no plenum in BN-350). Pitch-to-diameter ratio of pins is 1.16, very tight relative to U.S. designs. Fuel smeared density is 80%, including a center hole. With these changes from BN-350, the nominal target burnup is 10%; and 5% is considered attainable in the first core. The fuel-pin bundle

is contained in a 2-mm (79-mil)-thick can with 1.5-mm (59-mil) pads for lateral spacing. Groups of four, five, or seven bundles are held together by bushings in the exit plenum, carrying a fuel-handling knob for the module. No other lateral restraint is provided for the core. Vented fuel is being given strong consideration. It is planned to start up BN-600 (as well as BN-350) with a wholly UO_2 -fueled core, but to convert later to a plutonium-uranium mixed-oxide core. The core is pancake with height-to-diameter ratio of about 1/3. The breeding ratio for a plutonium-fueled BN-600 has been calculated at 1.37, but the breeding ratio of the initial UO_2 -fueled "converter" core will be ~ 1.15 .

*Leipunsky, A. J., Golovnin, I. S., Kugushev, N. M., et al., *The BN-600 Fast Reactor*, Technical Meeting No. 4/5 of NUCLEX 69, Basel, Switzerland, October 6-11, 1969.

A. FAST REACTORS

TABLE A.5.1. Data for BN-600 Reactor

Core	
Diameter x height, m	2.05 x 0.70 (6.73 x 2.30 ft)
Enrichment zones	
Outer: No. of subassemblies	162
Inner: No. of subassemblies	235
Radial blanket, m	0.4 (1.31 ft)
No. of pins per blanket subassembly	37
Pin diam/wall, mm	14.7/0.4 (0.58 in./15.7 mils)
No. of ribs per pin	3
OD over ribs, mm	15.25 (0.60 in.)
Fissile Loading ^{235}U /or ^{239}Pu , kg	1700 or 1200 (3748 or 2646 lb)
Enrichment ^a	
^{235}U Inner/Outer (Zone), %	~21/~28
^{239}Pu Inner/Outer, %	~14.5/~20
Radial shield materials	Graphite in steel
Fuel Assemblies	
No. of fuel assemblies	397
No. of pins per subassembly	127
Distance across flats, mm	96 (3.78 in.)
Pitch, mm	98 (3.86 in.)
No. of subassemblies per module	4, 5, or 7
Fuel Pin	
Type	Nonvented, plenum
OD/wall, mm	6.9/0.4 (0.27 in./15.7 mils)
Pitch, mm	8.05 (0.317 in.)
Type of spacing	Wire wrap
Material	OX16H15M3Nb ^a
Pellets density, g/cc	10.4 (annular)
OD/ID, mm	5.7 to 5.9/1.8 (0.22 to 0.23 in./0.071 in.)
Gap, mm	0.1 to 0.5 (4 to 20 mils)
Active length, m	0.70 (2.30 ft)
Axial blanket, m (each)	0.4 (1.31 ft)
Plenum, m	0.8 (2.62 ft)
Breeding Ratio	
Pu Fuel or U Fuel	~1.37/1.15
Specific power peak/avg, kW/liter	840/550
Doubling time (Pu fuel only)	10 yr
Control Rods/Absorber/%ΔK	
Automatic No./Material	2/B ₄ C
No. of shims/Material/% Δ K	19/B ₄ C or Ta/6.1%
No. of safety rods/Material/% Δ K	6/B ₄ C/4.0%
Rod strength, % Δ K (UO ₂ core)	0.2
Temperature Coefficient (isothermal)	
%/ $^{\circ}\text{C}$	-1.64 x 10 ⁻⁵
Power Coefficient	
%/MW	-1.0 x 10 ⁻⁵
Power System	
Thermal power, MWt	1470
Electric power (nominal), MWe	600
Thermal efficiency, gross/net %	42/39.2
Core temperature	
Inlet/outlet, $^{\circ}\text{C}$	380/550 (716/1022 $^{\circ}\text{F}$)
Secondary temperature, IHX	
Inlet/outlet, $^{\circ}\text{C}$	320/520 (608/968 $^{\circ}\text{F}$)
Primary flow, tons/hr	24,000 (96,000 gpm)
Secondary flow, tons/hr	21,000 (84,000 gpm)
Feedwater	
Temperature, $^{\circ}\text{C}$	240 (464 $^{\circ}\text{F}$)
Heaters, No. low/high pressure	3/5
Steam Conditions	
Temperature (steam generator outlet), $^{\circ}\text{C}$	505 (941 $^{\circ}\text{F}$)
Pressure, kg/cm ²	140 (1991 psi)
Reheat temp (S.H. outlet), $^{\circ}\text{C}$	505 (941 $^{\circ}\text{F}$)
Reheat pressure, kg/cm ²	24 (341 psi)
Flow, tons/hr	1920 (3.84 x 10 ⁶ lb/hr)
Turbine	
Type	K-200-130
Rating, MWe	200
Generator	
Type	TFB-200

^aEstimated enrichments for alternate U²³⁵ or Pu²³⁹ core loadings.

The ^{235}U start, even for this "second generation" plant, reflects the smaller availability of by-product plutonium from a relatively small thermal-reactor program, in contrast to the surplus of plutonium from power reactors in the West. However, the UO_2 start also simplifies the project in several respects:

- (1) Delays the requirement for large-scale plutonium-fuel fabrication facilities, and the associated fabrication costs.
- (2) Simplifies some issues in reactor safety.
- (3) Simplifies fuel-testing requirements.
- (4) Simplifies equipment maintenance or modification during early operation of the plant.
- (5) Reduces flux/burnup ratio (20 to 30%).

These differences probably compensate for the initial low breeding ratio and poorer initial fuel-cycle economics.

f. Fuel-handling System

The fuel-handling system for removing spent fuel assemblies from the reactor and for loading fresh assemblies consists of two rotating plugs, two handling devices or grapples mounted on the rotating plugs, two elevators, a fuel-handling cell exterior to the reactor, and two fuel-handling drums. These operate as follows:

(1) The rotating plugs are mounted on the tapered part of the reactor vessel. The two grapples can remove any core or blanket assembly to or from either of the two elevator positions. (The discussions indicated that there may be an intermediate position, 120° from the elevator positions and closer to the center of the core. This may be done to overcome interference by the two grapples, but this is an uncertain conclusion.)

(2) The two elevators operate, as shown in Figs. A.5.6 and A.5.7, in two adjacent narrow vertical passages that connect the reactor vessel with a cantilevered fuel-handling cell. The elevators are inclined 15° from the vertical. As the fuel assembly is raised through the passage, it is kept in the vertical position.

(3) At the upper positions of the elevators, the assemblies are transferred to or from a fuel-transfer mechanism. When retracted into the fuel-handling cell,

these can be rotated to place fuel in (or remove fuel from) the fuel-storage drums. Figure A.5.6 shows the rotating frame and tubes in which the grapple moves.

At the point of transfer from the elevator to the cell fuel-handling mechanism, the subassembly protrudes 800 mm above the coolant, so that the transfer mechanism does not come into contact with sodium.

Transferring an assembly from the reactor vessel to a storage drum takes 5-6 min. Transfer through the cell under an inert atmosphere is done without special cooling. If required, however, spent assemblies could be cooled by an argon-gas blower during transfer through the fuel-handling cell.

(4) One fuel-handling drum is used for fresh fuel; the other, for discharged fuel, is filled with sodium. "Lazy-Susan" devices in these drums allow charging or discharging from one position.

Another transfer mechanism (see Fig. A.5.6) is used to remove fuel from the drums and deliver it to other stations, i.e., for fuel cleaning, examination, or extended storage (presumably a water pool).

g. Components

(1) *Pumps.* Detailed design information was apparently not available on these pumps; the design was stated to have been just completed. The general configuration is similar to the Fermi pumps. The pumps were described as double suction, single stage of 8000-T/hr (32,000-gpm) capacity at 7 atm head (approximately 103-psi pressure rise). Pump shaft seals were described as graphite, running against a nitrided shaft, similar to BOR-60 and BN-350. An artist's concept of the pump is shown on page 19.

(2) *Check Valves.* Only an outline-type description of these check valves was available; they are installed integral with the pump casing, below the pump impeller in the pump discharge. These are the only valves required in the BN-600 primary system. Stop valves are not currently being designed into the secondary system.

(3) *Intermediate Heat Exchangers.* Only an artist's concept of the IHX's in BN-600 was available (Fig. A.5.5). This description indicated that the IHX was connected to the primary-coolant pump inlet by a sodium-expansion loop, rather than by using a bellows or siding joint. The tube bundles are removable, but it is necessary to cut the secondary piping in order to replace IHX tubing, typical of an all-welded system. Detailed design drawings were not

available; the statement was made that layout and construction was done with only outline drawings, and that working drawings would be delivered with the equipment.

h. Steam Generators

The final design and arrangements of these units, and especially of the dump and vent system, appear to be under active revision, despite the erection of the building framework for their mounting at Beloyarsk. A reference design was described. Later, alternate design and materials were also described.

The modular steam generators of the BOR-60 design were the basis for BN-600, with these differences:

(1) The evaporators and superheaters would contain a gas space at the top, to act as a secondary-system surge tank.

(2) The superheater shell would also contain a reheat section.

(3) Large shells totaling 480 MWt/loop (versus 30 MWt/loop for BOR-60) comprise the reference design.

The reference steam generator module was described as being made up of two sections (similar to BOR-60), with these characteristics:

Evaporator heat-transfer area	1500 m ² (16,146 ft ²)
Superheater heat-transfer area	1000 m ² (10,764 ft ²)
Reheater heat-transfer area	1500 m ² (16,146 ft ²)
Evaporator tube length	30 meters (98.4 ft)
Superheater tube length	20 meters (65.6 ft)
Pitch	25 mm (1 in.) rectangular
Thermal capacity per set	480 MWt
Steam production per set	640 T/hr (1.3×10^6 lb/hr)
Steam pressure	140 atm (2057 psi)
Reheat steam pressure	25 atm (367 psi)
Reheat steam temperature	505°C (941°F)
Feedwater temperature	240°C (464°F)
Evaporator steam superheat	20°C (36°F)
Evaporator-reheater dimensions	12 × 3.8 m (39.4 × 12.5 ft)
Evaporator reheater weight	150 tons
Superheater dimensions	16 × 3.8 m (52.5 × 12.5 ft)

Superheater weight	200 tons
Evaporator shell and tube material IX2M	2¼ wt % chrome— 1 wt % molybdenum (not stabilized)
Superheater shell and tube material X18H9	18 wt % chrome— 9 wt % nickel (not stabilized)
Evaporator orificing ΔP	9-10 atm (13.2-14.7 psi)
Tubing	Seamless, butt-welded, 8-12 m between butt welds.

The evaporator units are to be shipped separately and assembled on-site. A startup steam separator is included between evaporator and superheater, but this separator would be bypassed during normal power operation.

Niobium stabilized materials were also being considered, and materials with 35-45% nickel (generally similar to Incoloy) also were being evaluated.

A further design alternate includes "modularization," using 30-MWt units exactly like BOR-60, but with an added section paralleling the superheater for reheat. This would have the advantages of:

(1) Taking direct advantage of BOR-60 construction and operating experience.

(2) Reducing damage in event of sodium-water reaction.

(3) Simplifying construction. (Note: Furnace heat treatment of a 150-ton assembly 12 × 3.8 m may be a limiting factor.)

The alternatives were presented as being under consideration, with no indication that the final decisions for fabrication have been made. The steam-generator hall under construction at Beloyarsk appeared large enough to permit considerable freedom of design. It did not appear that any internal structure was in place in the building framing.

i. Turbine

Each of the three secondary loops would supply steam to a hydrogen-cooled 200-MWe turbine-generator set (Type TKB-200, manufactured in Leningrad).

6. THE FAST CRITICAL FACILITIES

Critical experimental facilities BFS-1 and BFS-2 and the coupled fast-thermal facility for BOR-60 provide the integral experimental information required for the Soviet fast-reactor program.

a. BFS-2

BFS-2, the newest and largest of the fast critical facilities, has the capacity to accommodate mockups of large breeder power-reactor cores. It has been in operation at Obninsk since late 1969 and is currently being used for critical experiments on a mockup of the BN-600 reactor. The BFS-2 facility (see Fig. III.G.1 for top view) consists of a tank with cylindrical vertical tubes on a hexagonal grid and with concrete shielding surrounding the tank, similar to the BFS-1* except that the BFS-2 tank (5-m diameter by 3-m height) has approximately seven times the volume of the BFS-1 tank. There are 10,000 vertical tubes in the BFS-2, each of 5-cm diameter with 0.1-cm-thick stainless steel walls. Reactor materials for the mockup configurations are loaded into the tubes from the top in the form of disks—like large poker chips of 4.7-cm OD and 0.5- to 1.0-cm thickness. The vertical tubes are very tightly packed in the hexagonal lattice (1.02 pitch-to-diameter ratio), with 0.8-cm solid steel rods in the space between the tubes to reduce the void volume. A movable graphite thermal column is provided on one of the outer flats of the hexagonal-tube lattice and is used for instrument calibration. The actual radial space readily available for core mockups is approximately 4 m equivalent diameter.

The materials in disk form are used interchangeably in BFS-1 or BFS-2. Both reactors are equipped with loading machines. However, the BFS-2 loading system is more fully automated since the disk assemblies can be completely loaded into the tubes, whereas the BFS-1 requires manual positioning of the disk assemblies into the proper tube after the assemblies have been hoisted by crane from the transfer pits to a position above the top of the core. (See WASH-1060, p. 13, for a description of the BFS-1 mechanical loading arrangement.) The BFS-2 machine for loading the disk materials into the vertical core tubes (see Fig. A.6.1) is also used for positioning movable fission chambers or threshold-activation detectors in any desired location in the assembly and as a driver for pile oscillator measurements.

*WASH-1060, *Nuclear Reactors in the Soviet Union, Report of Reciprocal Exchange Trip to U.S.S.R. by U.S. Atomic Scientists, December 11-22, 1964*; A. I. Leipunsky *et al.*, "Experimental Studies on Fast Reactor Physics Performed in the Testing Unit of the BFS," IPRS 48331, *CEMA Symposium on Atomic Power Stations with Fast Reactors*, July 1969.

Fig. A.6.1. Loading-Unloading Machine for the Fast Critical Assembly, BFS-2.

BFS-2 has about 100 movable assemblies which can be used as control rods. In the present mockup of the BN-600 reactor, these assemblies have the same composition as the rest of the core assemblies to minimize perturbations in configuration. The foreground of Fig. III.G.1 shows the tops of a closely spaced group of six such BFS-2 control assemblies which represent the effects of a larger single control assembly, close to that of the BN-600 design. The actual BN-600 design has 27 B_4C control rods: six withdrawn safety rods, 19 compensator rods for burnup and power ascension, and two automatic controllers. The active core height is 70 cm, the diameter is 205 cm, and the blankets are 40 cm thick. Two core zones of different enrichment are used for power flattening.

Sufficient ^{235}U (90% enriched uranium metal clad in aluminum) in 5-cm-thick disks is now available at Obninsk to run both BFS-1 and BFS-2 with combined fissile inventory in the neighborhood of 3000 kg. A small amount of plutonium (at least 100 kg) as PuO_2 in disks 0.5-cm-thick steel-clad is also available. (The actual amount of

fissile material on hand—either ^{235}U or plutonium—was not explicitly stated.) Disks of sodium clad in stainless steel have now been fabricated in sufficient quantity so that sodium can be used in at least a portion of the mockup assembly instead of being represented wholly by aluminum as in earlier critical configurations. Boron carbide is available in sufficient quantity for safety-rod mockup experiments; UO_2 clad in aluminum is available in sufficient quantity for blanket-region mockup, as well as other materials required for complete compositional mockups of large ^{235}U -fueled cores.

Although the BFS-2 is being used for critical experiments on a mockup of the BN-600 reactor (approximately 1700 kg of ^{235}U), it will be used later for mockup experiments on even larger fast breeder reactors in the 1000-MWe range with fissile masses of about 3000 kg of ^{235}U . Near-term emphasis will be on ^{235}U -fueled assemblies, since it is now planned to start the BN-600 with a UO_2 -fueled core. Only small amounts of plutonium (about 100 kg or whatever larger amount is available) will be used in zoned experiments to obtain integral physics information for design of the later plutonium cores in BN-350 and BN-600. The BFS-2 has provision for forced-air cooling of the assembly tubes, enabling the facility to handle fully plutonium-fueled assemblies.

Studies being conducted on the present BN-600 mockup in BFS-2 include critical mass, spatial fission power distributions, worths of compensator and safety control rods and perturbation effects of control rods on power

distribution, central reactivity worths and statistical weights of reactor materials, neutron lifetime (Rossi alpha), and neutron-spectrum measurements with scintillation detectors. Some bulk sodium voiding reactivity measurements may be conducted; but this is not urgent because the positive sodium void effect is not large in the ^{235}U -fueled core. Doppler measurements have not yet been planned for BFS-2 by either reactivity oscillator or activation methods. (Doppler measurements by the activation method—change in absorption rate of a heated sample—were performed earlier in the BFS-1,* but no oscillator Doppler measurements; and the BFS facilities have not been equipped with a Doppler oscillator mechanism.) Figure A.6.2 shows U.S. delegate Greebler discussing nuclear problems with Soviet scientists at the BFS-2 facility.

b. BFS-1

The BFS-1 facility and early critical experimental program was described in the report on the 1964 trip to the U.S.S.R. (See WASH-1060.) During most of the past 6 years, the BFS-1 has been used for a series of mockup critical experiments in support of the BN-350 and BOR-60 reactor designs and planned operations. The earlier of these assemblies (No. 15, 16, and 17 for the BN-350) and their experimental results were described at the 1966 London Conference* and the 1967 Karlsruhe Symposium** on fast-reactor physics. The more recent critical assemblies have been a mockup of the BOR-60 (No. 21), followed by a more detailed mockup of the BN-350.

The BFS-2 is now handling the engineering mockup experiments in support of the U.S.S.R. design and development programs for large fast breeder reactors. The smaller BFS-1 facility will be used in the near future mainly for more basic integral experiments designed to improve nuclear data and calculation methods used for the design of fast reactors, and for experiments to study heterogeneity effects in support of the BFS-2 mockup experiments. The BFS-1 has been equipped for time-of-flight measurement of the neutron spectrum, using an 800-m flight tube (with intermediate stations at 50 and 230 m from the reactor) and a microtron-pulsed electronic accelerator to produce a source of neutrons from a target located near the center of the core assembly. The resolution of the time-of-flight spectrometer at the 800-m station is 2 to 3 nsec/m,

Fig. A.6.2. Physicist P. Greebler (center), U.S. Delegation; with V. P. Zinoviev (left), Chief Engineer for BFS Facilities; and V. V. Orlov (right), Chief, Department of Fast and Power Reactors, Physics and Power Institute, Obninsk.

*Leipunsky, A. I., *et al.*, "Experimental and Theoretical Investigations of the Physics of Fast Reactors," *Proceedings of the London Conference on Fast Breeder Reactors Organized by the British Nuclear Energy Society, May 17-19, 1966*, Pergamon Press, Inc., New York (1967), p.445.

**Bondarenko, V. V., *et al.*, "A Study of Heat Release Distribution in Fast Reactors with Absorber Rods," *Karlsruhe Symposium on the Physics and Safety Problems of Fast Reactors*, November 1967, Paper SM-101/62.

corresponding to an energy resolution of 0.5% for 1-keV neutrons and 15% for 1-MeV neutrons. The microtron accelerates electrons up to 30 MeV and can generate very short pulses ($\sim 1\text{-}\mu\text{sec}$ width with 5-500 pulses/sec, $\sim 10^{10}$ neutrons from the target per pulse).

Design and analysis of integral experiments to be done in BFS-1 for systematic improvement of nuclear data will be based on a principle similar to the method used in the U.K. and described by Rowlands and MacDougall.* In this approach, measured integral quantities that are relatively easy to interpret by analysis (such as critical mass k_{eff} , reaction rate ratios, and neutron spectra) obtained from many critical assemblies, are used to adjust differential cross sections. This adjustment is done by computing the sensitivity of the measured quantities to the various differential cross sections and then changing the differential data within specified uncertainty limits to give a best fit between the measured and calculated integral quantities.

Integral measurements for evaluating the ^{239}Pu alpha value have been made using the method described by Anderson *et al.*** of measurement of relative activation rates and reactivity worths. Measurements based on the null-reactivity method have not yet been conducted, but it is planned to run such experiments later.

Relative merits of "clean" integral experiments based on simple compositions and geometries versus engineering-

mockup integral experiments are debated within the U.S.S.R., just as in the U.S. The use of BFS-2 mainly for the mockup experiments and BFS-1 for the more basic experiments (plus supplementary measurements in support of the larger mockups) represents a balance between these two approaches which is not far different from the U.S. use of the four major ANL fast critical facilities.

Discrepancies between calculated and measured central reactivity worths for fuel materials and boron are typically 10-20%—but the calculated worths could readily be adjusted to compensate such a difference within the present uncertainty range of the nuclear data and also of the computational methods. Accuracy of critical-mass predictions for recent critical assemblies was stated to be about 1-2% in equivalent k_{eff} —which is about the same as in the U.S.

c. Coupled Fast-thermal Critical Assembly

Critical experiments in support of the BOR-60 reactor have been conducted in a coupled fast-thermal critical assembly located at the Melekess Scientific Institute. The assembly consists of a 12-cm-diam central fast zone simulating the BOR-60 composition, surrounded by a buffer and a larger thermal zone. These experiments included measurements of material central reactivity worths and reaction rates, fission-rate spatial distributions, and neutron spectrum with threshold and resonance activation detectors. However, the actual engineering-mockup critical assembly for BOR-60 was carried out in the BFS-1 critical facility (Assembly 21) at the Physics and Power Engineering Institute in Obninsk. (See Physics Startup Testing in Part 2 of Appendix A.) The Scientific Institute at Melekess may have a full fast critical assembly in support of BOR-60 operation and for other integral experiments by 1972.

*Rowlands, J. L., and MacDougall, J. D., "The Use of Integral Measurements to Adjust Cross Sections and Predict Reactor Properties," *British Nuclear Energy Society Conference on the Physics of Fast Reactor Operations and Design*, June, 1969.

**Anderson, T. L., *et al.*, "Measurement of the Effective Capture-to-Fission Ratio in Two Different Fast Reactor Spectra," *Proceedings of Symposium on Fast Reactor Physics, Karlsruhe, 1967*.

APPENDIX B

SOVIET WATER REACTOR TECHNOLOGY

1. THE PRESSURIZED-WATER REACTORS AT NOVovorONEZH

a. Unit 1

Unit 1 at the Novovoronezh Nuclear Power Station was the first pressurized-water reactor in the U.S.S.R. and has operated since 1964 at 210-MWe capacity, 47-kWt/liter power density, and 100-atm (1470-psi) water pressure; it has six loops and three 70-MWe turbines. The unit operated satisfactorily for 5 years and was shut down in November 1969 for general inspection of the integrity of fuel and reactor internals. Restartup is scheduled for August 1970. From 1964 to 1968, the plant load factor increased from 25 to 83%. During early operation of the plant, fractures in the auxiliary piping occurred due to vibration fatigue. A number of fuel assemblies also failed after reaching burnups of 10,000-15,000 MWd/T (30 assemblies failed out of a total of 350 loaded in this period). These failures are attributed to metallurgical defects. To date the average discharge burnup is 13,000 MWd/T, and the maximum burnup (peak pellet) in one assembly is 30,000 MWd/T.

There are 37 control assemblies, each of the same dimensions as a fuel assembly; six are safety rods, and 31 are shim rods for power regulation and burnup compensation. The control rods are flux traps, a large water hole surrounded by a thin steel cylinder containing 1.8% boron, which prevents power peaking in the adjacent fuel assembly. The compensation rods have fuel-assembly followers, which give a local power-peaking factor of only 1.13 when the fuel follower is in the core. The safety rods have zirconium followers. Each fuel assembly contains 90 pins (10.2-mm OD) clad in a zirconium-1% Nb alloy, wire-wrapped, in a zirconium-2½% Nb alloy can; the fuel enrichment is 2%.

b. Unit 2

This unit is similar to Unit 1 in configuration; i.e., the vessel size, the number of assemblies in the core, and the outside fuel-assembly dimensions are the same, but the number of loops was increased to eight. However, the power output has been increased to 365 MWe with a power density of 73 kW/liter by using a smaller pin diameter and a tighter lattice. (Fig. B.1.1 shows the U.S. Delegation and the Novovoronezh staff in the Unit 2 control room.) The fuel is also changed to annular pellets of 3% average enrichment, and the number of control assemblies is increased to 73. The higher reactivity and control available

allow 28,000-MWd/T average discharge burnup. Two-zone enrichment is used to flatten the radial power distribution. The unit started operation in December 1969 and reached 365 MWe in 3 months. A change to dry refueling on this unit has caused problems with excessive radioactivity. The reactor top deck is shown in Fig. B.1.2. Wet refueling will be used on subsequent units. The turbine-generators for Units 1 and 2 are shown in Fig. B.1.3.

c. Units 3, 4, and 5

Units 3 and 4 are 440-MWe reactors, each having six loops; Unit 3 is scheduled to start up in 1971, and Unit 4 in 1973. The reactor pit and the pressure vessel for Unit 3 are shown in Fig. IV.B.2. Unit 3 will have five 88-MWe turbines, whereas Unit 4 will have two 220-MWe turbines. Both will use burnable poisons of boron steel. To help flatten the power, the burnable-poison elements will be concentrated in the inner zone. The use of soluble poisons in the coolant is necessary in Unit 3 to reduce the number of control assemblies from 73 to 37. Target burnup for Units 3 and 4 is 40,000 MWd/T averaged for discharge subassemblies. (Photographs of some areas of construction are shown in Figs. III.A.2., B.1.4, and B.1.5.)

Unit 5 is a 1000-MWe reactor scheduled for 1975 operation and will have four loops and two turbines.

d. Components

On pumps, there are no significant developments in existing plants; however, Unit 5 is planned for the use of controlled leakage pumps rather than the glandless (canned motor) pumps now being installed. The pumps are tested at the factory under service conditions. (The Unit 5 pumps will have a capacity of 88,000 gpm.) Instead of the currently used piloted hydraulic operators, packed-valve stems are being considered for Unit 5. The difficulties reported in auxiliary piping on Unit 1 were due to vibration and subsequent failure. These pipes were 25 mm (1 in.) in diameter. The problem was solved by increasing the diameter to 30 mm (1.18 in.) and revising the design of the pipe supports. In the steam generators, the steam pressure is steadily increased with succeeding plants, 60 atm (900 psi) being planned for Unit 5. The same design principles are used (horizontal, but increased in size). The steam-plant equipment was conventional; welded condenser tube joints and copper-nickel alloy tubes with carbon-steel shells were used.

Fig. B.1.1. Novovoronezh Unit 2 Control Room with U.S. Delegation and Soviets; left to right: V. Kondratova, L. Golubev, M. Portny, R. W. Dickinson (U.S.), Director S. Samoilov, Unknown, P. Greebler (U.S.), V. Bagai, Unknown, L. Burris (U.S.), P. Murray (U.S.), A. Amorosi (U.S.), Malkov, W. B. McDonald (U.S.), E. L. Zebroski (U.S.), G. W. Wensch (U.S.), two unknown reactor operators, Prozorovsky.

Fig. B.1.2. Control-drive Area of PWR Unit 2 with "Tophat" Removed.

Fig. B.1.3. Novovoronezh Units 1 and 2 Turbine-Generator Hall.

Fig. B.1.4. Novovoronezh Unit 3 Reactor Hall, with Pressurizer Vessel Shown at Left.

e. Station Layout

Completion of the five units by 1975 will provide a 2500-MWe capacity at Novovoronezh. No additional units will be constructed, because of cooling limits on the Don River which permit 5°C (9°F) allowable rise in summer and 3°C (5.4°F) rise in winter downstream from the outlet (about $15\text{--}20^{\circ}\text{C}$ ($27\text{--}36^{\circ}\text{F}$) rise permitted at the outlet). For Units 1 and 2, the Don River will be used directly for condenser cooling; but seven natural-circulation cooling towers will be built for Units 3 and 4—three for each unit, together with one spare. An artificial lake (1500 acres) will be built for cooling Unit 5. The radioactivity-release criterion from the station is 200 Ci/day. Operations with Unit 1 have given values of 30-40 Ci/day. The annual reactivity release is plotted in Fig. B.1.6. The Soviets are not considering plutonium thermal recycle now. When the total PWR capacity reaches 10,000 MWe, plutonium can be economically recycled.

f. PWR Construction Program

The construction program for pressurized-water reactors in the U.S.S.R. is as follows: At Novovoronezh, 2500 MWe total; at Oktomberyan (Armenia), two 440-MWe units; and at Murmansk, two 440-MWe units; giving a total capacity of 4250 MWe in operation by 1975.

2. THE SUPERHEAT REACTORS AT BELOYARSK

a. Status

Two nuclear superheat operating plants were observed at the Kurchatov Power Station. Unit 1 started operating

April 1964 at 70 MWe, with an average steam-outlet temperature of 410°C (770°F), and has reached 106 MWe and 510°C (950°F) steam temperature. Unit 2 started operating December 1967 and reached 170 MWe in July 1967 and 195 MWe in 1969, with outlet steam up to 516°C (961°F). These are graphite-moderated tube-type reactors, and their general description, layouts, and initial operating parameters have been published.* The first unit operated initially at only 75 MWe and at a maximum steam temperature of 410°C (770°F), below the design targets of 100 MWe and 510°C (950°F). In September 1967, with the installation of an improved design of channels using oxide instead of metal fuel in superheat channels, the unit reached 106 MWe with average exit-steam temperature of 510°C (950°F). Peak local steam temperatures reached $535\text{--}545^{\circ}\text{C}$ ($995\text{--}1013^{\circ}\text{F}$). Operation of both units has continued since 1967 with steadily increasing load factors averaging over 80% and reaching 89.8% for Unit 2 in 1969, and over 5.5 billion kWh cumulative output.

b. Plant Design

Plant design has also been described previously.* Unit 1 was an experimental outgrowth of the tube-type, graphite-moderated APS-1 with an indirect evaporating cycle supplying steam to a direct superheating pass which fed the turbine. In Unit 2, both evaporation and superheat passes are direct-cycle. A flow diagram for Unit 2 is shown in Fig. B.2.1. A cross section of the reactor is shown in

Fig. B.1.5. Novovoronezh Turbine-Generator Hall for Units 3 and 4.

*N. A. Dolezhal *et al.*, Geneva Conf. 1958; R. I. Aleshchenkov *et al.*, *Atomnaya Energiya* 16, 489 (1964); N. A. Dolezhal, *et al.*, *Atomnaya Energiya* 27, 379 (1969).

Fig. B.1.6. Annual Concentration of Radioactivity Release from Novovoronezh Station.

Fig. B.2.1. Thermal Circuit of the Second Unit of the Beloyarsk Nuclear Power Station: 1) Reactor; 2) Steam Superheat Channels (SSC); 3) Evaporative Channels (EC); 4) Circulating Pump; 5) Emergency Pump; 6) Steam Separator; 7) Reduction-Humidifier Device; 8) Deaerator; 9) Turbogenerator; 10) Condenser; 11) Condensate Scrubber; 12) Tank Expander; 13) Feed Pump; 14) Feedwater Preheater; 15) Superheat Regulator (PVD-9).

Fig. B.2.2, and the evaporator and superheater elements are shown in Fig. B.2.3.

The layout of the plant equipment is conventional. There is, however, no containment, and building ventilation goes to a stack. The activity release appears nominal (mostly by argon activation) and was stated to be lower than for other water-cooled reactors. The main plant parameters are shown in Table B.2.1.

Originally a rail-mounted refueling machine with operations inside a shielded cab was used for refueling. This worked well but has been replaced by open-cell fuel handling from the overhead crane for Units 1 and 2. Both television and a lead-glass viewport are used to guide refueling operations.

Fig. B.2.4 shows the arrangement on the reactor top deck of Unit 2. Manned entry to the top deck is freely used, even while the reactor is at power. Low-pressure breathing air-supply taps are provided around the cell perimeter for use with air-suits when contamination is present.

BOILING-WATER REACTOR

Fig. B.2.2. Section of Superheat Reactor Core. 1—Layout of graphite; 2—Bottom plate; 3—Top plate; 4—Water protection; 5—Top flooring; 6—Pipelines; 7—Steam separator; 8—Floor mounted machine; 9—Corridor for servicing valves; 10—Room housing drives of control system rods; 11—Superheated steam headers; 12—Main pipelines.

Fig. B.2.3. Optional Superheat Reactor Fuel Elements.

TABLE B.2.1. Main Plant Parameters of Beloyarsk Superheater Reactors

	Unit 1 (AMB-100)	Unit 2 (AMB-200)
Startup date	Apr 1964	Dec 1967
Boiling/superheat cycle	Indirect/direct	Direct/direct
Turbine	100 MW: reheat Type K 100-90	2 x 1200 MW: reheat Type K 100-90
Steam flow, T/hr (lbs./in.)	400 (800,000)	800 (1,600,000)
Pressure, atm (psi)	90 (1323)	90 (1323)
Temp, °C (°F)	500 (932)	500 (932)
Bleed points (nonregulated)	8	8
Manufacturer	Leningrad Metal Works	
Generator, H ₂ -cooled, kV	10.5	10.5
Distribution, kV, service/transmission	0.38: 6.3/220	0.38: 6.3/220
Thermal efficiency, %	33.2	37.5
Peak/avg power (two zones)	1.46	1.28-1.30
Operating factor, %, 1968-1969	69-76	80-89
First reload	-June 1967	-
Max power level reached, MWe	106	195
Initial steam temp	410 °C (770 °F)	501 °C (934 °F)
Pressure at outlet, atm	100	110
Steam temp peak/avg, °C (°F)	519/512 (966/954)	524/516 (975/961)
Short-time local peak, °C (°F)	535 (995)	545 (1013)
No. of channels: evap/SH	732/266	732/266
Cooling tube: diameter/wall, mm (in.)	9.4/0.6 (0.370/0.024)	12.0/0.6 (0.472/0.024)
Channel flow control ^a	Orifice tube	Orifice tube
Channel leak detection	F.P. in graphite gas atmosphere	
Core size, d x h, meters	7.2 x 6.0	7.2 x 6.0
Fuel burnup, peak/avg, MWd/t	8500/6500	13,000/10,000 (design)
Fuel material		
Evaporator	U-9 Mo	U-9 Mo or U-3 Mo
Superheater	70% UO ₂ in Mg	70% UO ₂ in Mg
Cladding	-321 SS	-321 SS
Uranium-235 %/tons U	1.8/67	3.0/50
Refueling interval, months	3-4	3-4
Refueling time for 30-40 channels, days	3	3
Fuel residence time, yr	-4	-2
Control rods/safety rods	78/100	78/100
Design agency	Tyeplo-electro-project Institute	
Technology	Obninsk Physics and Power Institute	
Construction	Ural-Energo-Troy Trust, Ministry of Power	

^aOriginally used rotameter and control valves. Flow measurement now only on Unit 2. Temperature of every channel is monitored by Chromel-Copel thermocouples.

Fig. B.2.4. Beloyarsk Superheat Unit 2 Reactor Top Deck; New Fuel and Instrumentation Thimbles on Right Wall.

c. Refueling

The most difficult operating problem in the first years of operation was the extraction of damaged fuel channels. These apparently crack the graphite, and the removal of such channels was initially responsible for a considerable part of plant outage. The Soviets have developed a special grapple device, operating from the crane hook, which permits pulling the graphite from a leaking channel and replacing it within 24 hr. Damaged channels are replaced by surrounding the channel with a heavy cotton "sock" as it emerges from the reactor core. Elastic bands seal the sock to the channel and at the bottom sufficiently well to prevent the spread of particulate radioactivity.

Fuel burnups to date exceeded 6500 MWd/T channel average and about 8500 MWd/T local peak, and design target values of 20,000 MWd/T average are considered to be reasonably practical on the basis of experimental tests. On the basis of small-scale tests done at Obninsk, higher-burnup targets up to 30,000 MWd/T are expected to be attained in the future. Oxide fuel is now used in the superheat channels, and there have been no failures since 1968. The oxide fuel consists of a 70 vol % dispersion in magnesium metal, and annular fuel elements have circumferential ridges about 1 mm high at about 1-cm intervals along the length of the fuel (see Fig. B.2.3). Fuel channels are 6.0 m long, with two enrichment zones for radial power flattening.

Flat power distributions have been attained by the use of control rods, including weakly absorbing or "gray" rods. The ratio of radial peak to average is 1.46 for Unit 1 and 1.28 to 1.30 for Unit 2. Power distribution is measured by (1) in-core sensors (neutron and gamma), and (2) compari-

sons of relative reactivity with identical sections of control rods.

d. Water Chemistry

Of special interest is the successful use of very small amounts of hydrazine in the feedwater heating train to reduce the oxygen level in the feedwater heater and correspondingly to produce a substantial reduction in iron-bearing corrosion products in the primary coolant system.

The feedwater heater system consists of a string of nine heaters, the first five made of brass (low-temperature heaters) and the last four of carbon steel. The condenser is made of an alloy called "Milchyor," which is a copper-nickel-iron alloy in common use for condensers and low-temperature feedwater heaters. The main problem is the corrosion of the carbon-steel feedwater heaters. The condensate coming to the feedwater heaters (after de-aerating) contains 100-150 μg per kilogram (100-150 ppb) of oxygen when the reactor is operating.

Hydrazine hydrate is injected at the rate of 70-90 g/hr (0.15-0.20 lb/hr) into feedwater heater No. 8. (The temperature at this feedwater heater is near 200°C (392°F).) The corrosion of carbon-steel feedwater heaters is considered very serious if special water treatment is not used. With the injection of hydrazine, the condensate level is reduced to 20 μg of oxygen per kilogram (20 ppb). This addition lowers the content of iron corrosion products from 150 to 20-30 ppb.

With hydrazine treatment, the required flow to the deionizer is only 2-3% of the total flow to the purification system. Under these conditions, the specifications of the total solids content determine the purification flow rate. Chloride is controlled to a limit of less than 10 $\mu\text{g/kg}$ (ppb), although the permissible "norm" is 30 $\mu\text{g/kg}$ (ppb). (See Table B.2.2.)

The salts in the feedwater are due to a leakage rate of about 0.003% from the condenser, where the total flow is 800 tons/hr (1.6×10^6 lb/hr); the actual level of total solids is about 50 $\mu\text{g/kg}$ (ppb). Chloride is routinely maintained at the exceptionally low level of 3-4 ppb. Hydrazine injection at 45 ppm has also been tried experimentally, reducing the oxygen content of the steam to about 30% of normal, or about 50 ppb.

In the closed circuit of Unit 1, radiolysis is suppressed by the use of ammonia which, under the action of irradiation, decomposes to produce hydrogen. An addition of 0.5-1.5 mg/liter of ammonia suppresses radiolysis, with 20 mg/kg of hydrogen being formed in the steam in the

TABLE B.2.2. Plant Materials and Parameters

	Unit 1 (AMB-100)	Unit 2 (AMB-200)
Primary System Materials	Carbon steel Cr-Ni steel Stainless steel Cu-Ni-Fe ^a	Carbon steel Cr-Ni steel Stainless steel Cu-Ni-Fe ^a
Condenser material	Brass ^a	Brass ^a
Feedwater heaters material	Carbon steel	Carbon steel
1 to 5	215 (419)	215 (419)
6 to 9	-	0.003% of 800 T/hr
Feedwater temp, °C (°F)	0.5 (1000)	6.5 (13,000)
Condenser Leakage	(closed loop only)	-
Purification system flow T/hr (lbs/hr)	Closed loop	Open loop
Water purity, actual/spec, ppb	80/100	-50/500
Total solids	<10/15	3-4/15
Chloride	-	-20/80
Iron	<10/50	-15/50
Sodium	80/100	-30/150
Silica	0.5-1.5 ppm NH ₃	70-90 g/hr Hydrazine
Water additives	in closed loop	to heater No. 8
Specific activity; condensate, C/liter after -1-hr decay	10 ⁻⁵ to 10 ⁻⁶	10 ⁻⁵ to 10 ⁻⁶
Activity of Primary Equipment	Mostly ²⁴ Na	-
Pump casing, μR/sec (operating)	-	100
μR/sec (shut down)	-	20
Turbine (operating), μR/sec HP/LP ^b	1.5/2.0	5.0/4.0
Activities deposited	Co, Fe, Cr, V, Mn, Zn, Mg	-
Main activities	⁶⁰ Co (80%); ⁵¹ Cr; ⁵⁴ Mn; ⁵⁸ Co; ⁶⁵ Zn	-
Specific activities d/m-cm ² ⁶⁰ Co	-	-
at core inlet piping after:	-	-
460 days operation	0.9 x 10 ⁶	-
780 days operation ^c	1.6 x 10 ⁶	-
Liquid Waste Disposal	4 tanks, 400 m ³ each (100,000 gal.)	Ferroconcrete; 4 mm, SS liner (0.16 in.)
Pulp disposal	2 tanks, 400 m ³ each (100,000 gal.)	-
Discharge norm (to river), Ci/liter	-	-
Without alpha	10 ⁻⁹	10 ⁻⁹
With alpha or ⁹⁰ Sr	10 ⁻¹³	10 ⁻¹³
Gas discharge: avg/max/norm, μCi/sec x 10 ⁵	-	0.2/0.5/1.06

^aTrade name Milchyor, also Latoon, a silver alloy, in some places.^bHigh-pressure casing/low-pressure casing.^cAt this time, the specific activities of ⁵¹Cr, ⁵⁴Mn, and ⁵⁸Co are 32, 10, and 4% respectively of the ⁶⁰Co value.

superheat pass. Under these conditions, corrosion products in the water in the boiling pass are 50-80 μg/kg, with a chrome-nickel steel system and 0.5-ton/hr cleanup bypass flow. The pH is controlled to 8.5-9.0. The total amount of iron-chrome-nickel in the water is limited, and silicate, chloride, and sodium concentrations are also controlled as shown in Table B.2.2. Consideration is being given to converting Unit 1 to an open cycle for both boiling and superheating, as in the design in Unit 2.

The water-treatment plant was described using the cascade of heaters and evaporators which produce 100 liters/hr (26 gal/hr) of concentrate at 30 g/liter. Cleanup factors of 10⁴ to 10⁵ are regularly obtained. Wastes are stored in a battery of four 400-cubic-meter (100,000-gallon) storage tanks. Discharge to river or ground water is limited to less than 10⁻⁹ Ci/liter without alpha contamination, or 10⁻¹³ Ci/liter if alpha activity or strontium is present.

The mixed active gases go to a gas holder through a condenser and are held in the gas holder until the activity level is acceptable. The nominal stack limit is 1000 Ci/day, but the average discharge is less than 250 Ci/day (of which about 80% is argon).

The capital cost for Unit 1 was given as 704 rubles/kW. (This includes the costs of the site and of the 20-sq km (5000-acre) artificial lake.) For Unit 2, a capital cost of

224 rubles/kW and power costs of 0.85 to 0.9 kopeck/kWh were quoted. Of this, 0.6 kopeck/kWh was stated to be the fuel cost at the present burnup levels. The amortization rate ranged from 6 to 12%, depending on the expected life of the item being capitalized, but the average amortization rate was not given. Using zirconium fuel cladding in the evaporator channels and increasing burnups to levels attained experimentally would cut fuel costs to less than one-half of the above figure. These improvements, together with increase in plant size to 600-800 MWe, would be required for superheat reactors to become competitive with conventional BWR's or PWR's.

3. THE BOILING-WATER REACTOR AT MELEKES

a. Status

The VK-50* natural-circulation BWR at Melekess started up in September 1965 and reached 140 MWt in December 1965. The first refueling was carried out in 1967 after 500 full-power days of operation. In October 1969, improved fuel and control of power distribution permitted an increase of power to 200 MWt. Because of turbine limitations, the electric output is only 30 MWe. Further improvements to 270 MWt are planned, and design studies of 100-MWe units have been carried out.

b. Reactor Description

The reactor design is flexible for an experimental plant which can be operated either on natural circulation or forced circulation, dual cycle, with steam pressures of 450-1500 psi. The pressure vessel (see Fig. B.3.1) is 12.5 ft in diameter and 36 ft high; the fuel is 2% enriched UO₂ in 0.4-in.-OD rods, Zr-1% Nb clad of 24-mil thickness, 1.5-cm (0.6-in.) pitch, 126 rods per bundle. The primary system material is equivalent to Type 321 stainless steel. Control rods use a flux-trap principle (a water hole surrounded by steel-2% boron cylindrical shell) with fuel followers for 31 shim rods and zirconium followers for six safety rods.

The top entry of poison-section control rods initially gave a large power peak in the lower portion of the core. (U.S. designs use bottom entry for the poison section.) Using spatially variable burnable poison elements, with the poison concentrated in the lower half of the core, reduced the axial peaking factor from 2.3 to 1.7. The burnable poison elements are rods of Zircaloy-boron alloy inserted into six positions of each fuel assembly. Use of fuel followers for the shim rods results in a low local peaking factor.

*The designation BK-50 is also used, but should be translated as VK-50. It is a natural-circulation BWR generally similar to the 70-MWe Humboldt Bay Reactor in the U.S. which has operated commercially since 1963.

Fig. B.3.1. VK-50 Natural-circulation Boiling-water Reactor Vessel and Core.

The reduced axial power peaking, combined with a change in spacer design that gives a lower hydraulic resistance, permitted an increase of the core power density from 28 to 37 kW/liter and of the reactor power from 150 to 200 MWt.

c. Burnable-poison Control

Use of burnable poison has also permitted long operating intervals between refuelings. A run design for 250 days of operation actually went 400 days, with the reactivity increasing slightly during the first 100 days. The boron-zirconium alloy burnable poison is in separate rods inserted into six positions of each fuel assembly. To improve the axial power distribution, there is a higher concentration of boron in the lower portion of each rod than in the upper portion.

d. Stability

The reactor shows excellent stability to reactivity changes, recovering from a variety of induced transients by inherent stability. Control-rod movement is required only once every 2 or 3 days (a 20- to 30-mm (0.80- to 1.20-in.) rise in the rods). Power is controlled by turbine inlet valve only (usually without motion of control rods) for up to 40% change in power in 20 sec. Automatic controls are eliminated.

e. Measurement of Core Power Level

Core power level is measured by two methods involving insertion into thimbles:

(1) Activation of a copper wire, which is irradiated for several minutes, removed, allowed to decay for 1 day, and counted.

(2) Small chambers, which provide direct measurement of the current created by beta recoils from neutron captures by rhodium. The betas created by irradiation of rhodium wire (in a vacuum) create sufficient current in the wire for direct measurement. It takes about 10-12 min to traverse the core.

f. Fuel

Fuel burnup is up to 16 MWd/kg (peak) using Zr-1-Nb cladding; two fuel-element failures have been experienced, but did not interfere with continued operation.

The first recharging of fuel was carried out after 500 effective days of operation, at which time 30 assemblies were removed from the central core region. For the last 150 days, failed elements were present, as determined by fission-gas measurements. Fission-gas emission was up to 80-100 Ci/day; the maximum allowable was 3000 Ci/day. Fuel and loading are carried out after 10 days' cooling directly to air. Although the first unloading contained two failed fuel elements, the operation was carried out successfully with no measurable contamination in the reactor building.

APPENDIX C

TEST AND SPECIAL-PURPOSE REACTORS

1. THE SM-2 TEST REACTOR

The SM-2 is a beryllium reflected, light-water-moderated and -cooled test reactor. It is located at the Scientific Research Institute of Atomic Energy Reactors at New Melekess, in the Ulianovsk Region.

The reactor was originally made critical in 1961. The flux levels developed at the original power level of 60 MWt, average power density of 1600 kW/liter, and maximum power density of 4500 kW/liter were 2.5×10^{15} thermal and 1×10^{15} fast.

The reactor is fueled with UO_2 fabricated into nickel-clad plates.

The reactor was described in WASH-1060 (published in 1964). Since 1964 some changes were made in SM-2 to improve its performance. The height of the active core was increased to 350 mm (14 in.) resulting in a power increase to greater than 75 MWt. The control system was improved, and the beryllium oxide reflector was replaced with a reflector fabricated from metallic beryllium.

Figures C.1.1-C.1.4 show the latest available drawings of the SM-2 Test Reactor.

2. THE RPT (MR-2) MATERIALS TEST REACTOR

The RPT (MR-2) Materials Test Reactor, located at the

Kurchatov Atomic Energy Institute in Moscow, was built in 1952. The status of this reactor in late 1964 was published in WASH-1060.

This reactor was remodeled between 1962 and 1964 to operate at 20 MW with multiple experimental loops. The power was doubled in 1967, and at present the total power (40 MW reactor plus loops) is greater than 50 MW. A 4-month shutdown for remodeling is scheduled at the end of 1971.

Any of the reactor's 27 fuel channels can be operated as a driver fuel channel or an experimental loop. Multiple channels can be used for complex experimental loops. Figure C.2.1 shows the top of the reactor with control rods and top loop connections.

The reactor has operated at 80% plant availability since 1967. Eight experimental loops are now installed, each loop containing five to seven experimental devices. The greater part of the experiments have been directed toward confirming design and materials choices for future light-water reactors, although one loop has been devoted to direct conversion, another to the investigation of fuels for high-temperature, gas-cooled reactors in a helium loop, and a third to organic coolants.

Table C.2.1 shows the characteristics of the loops. Figures C.2.2-C.2.6 show important features of the reactor and loops.

TABLE C.2.1. Characteristics of Loops in RPT (MR-2) Material Test Reactor

Loop Designation	Type	Power, kWt	Pressure, kg/cm ²	Circulation Rate, tons/hr	Coolant Temp, °C	Cladding Temp, °C	No. of Channels Used
PVK-200	Steam-H ₂ O	2500	200	300	365	400	5
PVO-200	H ₂ O	1500	200	30	356	365	3
PVU-200	Steam-H ₂ O	1500	200	30	365	400	3
PO	Organic	1000	20	30	400	450	2
PG	He	1000	100	2	800	1500	1
PV(TS)	Steam-H ₂ O	1500	100	30	310	350	7
PV(TS) _m	H ₂ O	200	200	5	352	365	2
PPG	Gas	50	10	-	-	1700	1

Fig. C.1.1. SM-2 Reactor. 1—Control rods, 2—Assembly storage, 3—Overloading mechanism, 4—Actuator of overloading mechanism, 5—Floor of reactor hall, 6—Rotating plug, 7—Vertical experimental channel, 8—Control mechanism, 9—Upper shield plate, 10—Top reactor space, 11—Loop channel, 12—Vessel cover, 13—Reactor vessel, 14—Deflector, 15—Shield screens.

Fig. C.1.2. Longitudinal Section of SM-2 Test Reactor. 1—Horizontal experimental channel, 2—Control-rod systems, 3—Spare cells for assemblies with heat-liberating elements, 4—Mechanism of in-vessel loading, 5—Fuel exit port, 6—Actuator for safety mechanism, 7—Cover, 8—Penetration for control-system mechanism, 9—Active zone with deflector, 10—Shield, 11—Reactor vessel.

Fig. C.1.3. Horizontal Section of SM-2 Test Reactor.

Fig. C.1.4. Fuel Subassembly for SM-2 Reactor. 1—Spacer, 2—Laminar fuel element, 3—Shaft, 4—Flow duct, 5—Head for grapple.

Fig. C.2.1
Top View of the RPT (MR-2) Materials
Test Reactor Showing Control Rods and
Top Loop Connections.

Fig. C.2.2. MR Channel—Movable and Nonmovable. 1—Support tube, 2—Bayonet lock, 3,4—Seals, 5—Water inlet and outlet, 6—Finned tube, 7—Channel, 8—Movable channel, 9—Cadmium screen, 10—Telescoping lift, 11—Extension rod.

Fig. C.2.3. Scheme of Tandem Loop—MR Reactor. 1—Working channel, 2—Tubes, 3—Aluminum-alloy isolation section, 4—Stainless steel sections, 5—Bellows, 6—Thermocouples, 7—Thermocouple tubing and control for seal, 8—Vacuum and helium line, 9—Valve connections, 10—Location of reactor core.

C. TEST REACTORS

Fig. C.2.4
MR—Plan View of Core.

- 1 ○ Working channel with movable experimental section
- 2 ⊗ Working channel—fixed experimental section
- 3 ⊙ U channel
- 4 ⊖ Coolant channel
- 5 ⊕ Materials-irradiation channel
- 6 ⊕ Channel for movable ionization chamber
- 7 ⊕ Channel for nonmovable ionization chamber
- 8 ● Safety rod
- 9 ● Automatic control rod
- 10 ○ Shim rods
- 11 ▨ Beryllium
- 12 ▩ Graphite in aluminum tubes
- 13 ▨ Aluminum

Fig. C.2.5. Plan View of MR Reactor. 1—Headers with valve operators, 2—Support plate, 3—Core barrel, 4—Beryllium, 5—Graphite in aluminum tubes, 6—Protector plate, 7—Support, 8—Working channel with nonmovable experimental section, 9—Working channel with movable experimental section, 10—Cadmium screen in aluminum tube, 11—Central U channel, 12—Loop connection, 13—Materials-irradiation channel, 14—Rod for removing control rods and experiments, 15—Connector, 16—Channel with safety rods, 17—Channel for ionization chamber, 18—Pipe to water pool, 19—Collector for water pool.

Fig. C.2.6. MR Reactor Circuit Diagram. 1,2—Pumps, 3,4—Normal and emergency coolant pumps, 5—Pumps for system-cleaning water of mechanical and ionic impurities and corrosion products, 6—Heat exchanger, 7—Filter, 8—Chemical hermetic seal control, 9—Cloth filter, 10—Cation filter, 11—Cation and anion filter, 12—Volume compensation (pressurizer), 13—Gas-water ejector, 14—Separator, 15—Surge tank, 16—Platinum catalyzer, 17—Induction heater, 18—Pump, 19—Hydrogen detector, 20—Rotameter, 21—Tanks for mixing helium and hydrogen, 22—Helium tank, 23—Vacuum header, 24—Intermediate tank, 25—Vacuum pump, 26—Gas header.

APPENDIX D

PRINCIPAL SPECIAL FACILITIES

In this appendix are collected descriptions of special facilities visited by the Power Reactor Delegation.

1. SHIELDING FACILITIES

Shielding experiments are conducted with the B-2 beamhole and KTK fission-plate facilities of the BR-5 reactor. In the B-2 beamhole facility, neutrons from the outside of the 40.6-cm (16-in.)-thick nickel reflector of the BR-5 enter a 25-cm (9.8-in.)-diam beamhole of about 2.4-m (96.5-in.) length in the concrete shield. Up to 50-ton shields can be accommodated in a tank that can be rolled up to the end of the beamhole. The energy-integrated neutron source (average energy 50 keV) is about 8×10^{13} n/sec, corresponding to a beam flux of 1.7×10^{11} n/cm²-sec, of which 7% is greater than 1.5 MeV. The KTK fission-plate facility provides a lower integrated flux ($\sim 1.5 \times 10^8$ n/cm²-sec) than the B-2 beamhole. Lewin *et al.** give a good summary of U.S.S.R. shielding facilities and work through 1967, and include a cross-section view of the B-2 beamhole facility.

Recent experiments with the BR-5 reactor shielding facility have emphasized mockups of long, narrow shield penetrations, such as the stepped annular gap of the rotating-plug configuration of the BOR-60 and BN-350 reactors. A typical experimental configuration was illustrated as a stepped annular gap of 2-cm (0.8-in.) annular thickness penetrating through a 100-cm (39.37-in.)-thick shield of iron and graphite. Calculations on such shield configurations using the method of first-collision sources agree with the measured neutron transmission to within a factor of 2 or 3. (The method of first-collision sources considers all first collisions as new neutron sources and then uses neutron-age theory for a homogenized shield to calculate the fraction of each source that penetrates to the outer surface of the shield. In the absence of a specific computer code to do the job, such calculations are now performed by hand. Monte Carlo codes are available, but are unsatisfactory for this type of problem.)

Earlier shielding experiments with the BR-5 facility in support of the BN-350 shielding design included measured neutron transmission through shields of iron-graphite mixtures (or alternate layers of these materials), concrete and iron oxide mixtures, and aluminum penetrations (simulating sodium pipes). Results of these earlier experiments

have been published in *Atomnaya Energiya* (see, for example, Veselkin *et al.***).

The uncertainty margin allowed for fast-reactor shielding design was stated to be a factor of 3 for most of the reactor and plant, and a factor of 10 for the shielding above the top of the core where the serious penetration gaps occur. Shielding criteria for the BN-350 and BN-600 allow essentially unlimited access above the top of the reactor vessel for maintenance or repair of control-rod drives during full-power operation. (A similar access criterion was also set for the BOR-60, but an excessive streaming of neutrons between the shield plug and the reactor vessel was found and is now being corrected—see Part 2 of Appendix A.) Core radial shielding in the BN-600 reactor vessel is required to limit activation of the secondary sodium to the extent that there is essentially unlimited access to the secondary system without having to shield the secondary pipes and components.

2. SODIUM HEAT-TRANSFER FACILITY AT OBNINSK

Heat-transfer phenomena in alkali metal systems (sodium, potassium, and cesium) are under study in a well-equipped laboratory at Obninsk, which apparently had only recently been occupied.

The Soviets displayed three small pool-boiling rigs, which used electron-bombardment heaters in cylinders approximately 0.75 m (30 in.) high, using the reflux condenser principle in a closed system. Thermocouples were placed in the pool, but as far as could be determined from descriptions and diagrams shown, there was no evidence of thermocouples on the boiling surface. Boiling sodium, potassium, and cesium had been investigated and experimental data gathered on burnout heat flux, boiling heat-exchange coefficients, temperature fields, and wall-temperature distribution. Work so far had been performed on boiling surfaces composed of 1X18H9T steel with various surfaces; under consideration was additional work with different materials and with other surface finishes. No forced-convection sodium-boiling work had been done, but a test rig was shown that was in the final stages of completion and that is to provide single-channel boiling at 1-2 kg/sec mass flow at several meters per second. This rig has a NaK-cooled condenser. X-ray image intensifiers will be used to examine the phenomena visually.

The boiling-sodium experiments were conducted in concrete cells with viewing windows and controlled from outside the cells; some of the optical and measuring equipment used in these cells was manufactured by

*J. Lewin *et al.*, *A Survey of Recent Soviet Radiation Shielding Work*, ORNL-RSIC-23 (Sept 1968).

**A. P. Veselkin *et al.*, *Energy Distribution of Fast Neutrons Outside of Reactor Shielding*, Soviet Atomic Energy 26 (6) 607 (June 1969).

C. H. F. Mueller in Hamburg, West Germany. Another boiling-potassium rig was a forced-convection rig utilizing "power bursts" rather than steady-state conditions. The objective of the power-burst experiments was to determine burnout heat flux in both horizontal and vertical channels. Wall-temperature changes at burnout were being recorded in an attempt to derive an understanding of thermal stresses under various liquid-metal-boiling conditions, including burnout. Only potassium boiling at atmospheric pressure had been studied, but tests with sodium were planned. In another boiling rig, flow choking by sodium vapor was being measured. Electrically heated, aluminum oxide-insulated molybdenum heaters were used; the heaters were in a 19-pin configuration of which seven pins were heated, the pins were inside a tube and were 1.1 m (43 in.) long. Flow choking was measured by pressure and thermocouple measurements at the inlet and outlet of the channels. The pressure was measured by strain-gauge devices, and flow was measured with an electromagnetic flowmeter. The heat flux intended for the heaters was $600,000 \text{ kcal m}^2/\text{hr}$, but no data had been acquired yet because of persistent failure of the heaters. Samples of failed heaters were displayed. The individual pins appeared to be approximately 5 mm (0.2 in.) in diameter, and the complete 19-pin bundle appeared to be about 40 mm (1.6 in.) in diameter.

Another test rig, designed to measure the temperature field in the triffute between circular pins on a triangular pitch was displayed. This was a rather ingenious rig for measuring the steady-state temperature distribution; no transient conditions were attempted. The three electrically heated cylinders produce 100,000 to 150,000 $\text{kcal/m}^2/\text{hr}$ in flowing NaK at $50 \text{ m}^3/\text{hr}$. An Einstein-Szilard pump was used; the NaK was purified by cold trap. Temperature profiles were measured by a "scissors" thermocouple, which could be pushed up and down the triffute through a rubber seal. The two thermocouples could be spread apart to measure temperatures readily in the channels, and could also be rotated. So far, only NaK at 80°C had been used as difficulties had been experienced with the rubber packing around the movable thermocouple rod above this temperature. The heated channel measured was about $2\frac{1}{2} \text{ m}$ (98 in.) long, which was stated to be greater than 100 hydraulic diameters of the triffute. A somewhat similar rig, stated to have been used for BN-350 fuel-blanket tests, was present, but was now inactive.

There was evidence of a fairly recent NaK fire, which had been cleaned up before the team's arrival. Dry powder had been used to put out this fire.

Another rather large heat-transfer rig was exhibited, in which full 37- and 61-rod subassemblies about 1.5 m (58 in.) long could be tested. All rods were heated, and

three were instrumented. This rig was said to be capable of simulating fins on tubing, wire wrapping, and pins touching the wall of the subassembly can. Axial temperature was read by thermocouples on the instrumented tubes; these tubes could be rotated and temperatures read out at various radial positions. Thermocouples were clad in 0.2-mm (0.08-in.) stainless steel sheathing; the heaters were Nichrome, and the rig operated at below 100°C . The data gathered by this rig are to be published in July 1970.

3. SODIUM COMPONENT TEST FACILITY AT MELEKESS

The group was shown three test rigs, one for valves, one for small-leak sodium-water reactions, and one for water heat-transfer tests. The valve test rig appeared to have at least two stations on which valves of 100-, 200-, and 300-mm size (3.9, 7.9, and 11.8 in., respectively) had been tested in sodium. Tests were essentially in static sodium, with no provision for thermal shock. Valves were tested at $500\text{--}600^\circ\text{C}$ with 12-atm pressure across the seat. A 2000-hr test, consisting of 1500 open-and-close cycles, was performed on each type of valve. The valve-acceptance criterion was stated to be a maximum leakage rate 10-15 liters/hr across the seat after completion of the tests.

The "small-leak" stand was enclosed and appeared to be approximately 15 ft (456.2 cm) square and 10 ft (304.8 cm) high. The rig had a capacity of 400 kW during previous testing and was now shut down for upgrading to a heat capacity of 1-2 MW. An electric furnace produced sodium at 500°C , with water injection at 310°C and 100 atm. Heat-transfer tests were not performed in this rig, although it could accommodate 19-tube bundles with one active tube in its earlier configuration. It was on this rig that the tube spacing-to-hole-diameter relationship had been obtained, as described in Section III. F.7.

A pressurized-water heat-transfer test rig was located adjacent to the two sodium rigs. This apparatus had an electric capacity of 1 MW; heat-transfer elements were heated by direct current to achieve high heat fluxes. This rig had been used to identify "critical thermal loads" for PWR fuel elements (presumably burnout data), and also to check distribution of steam velocity along superheater elements (presumably for the Beloyarsk Nuclear Superheat Reactors). During the PWR tests, the data developed by Tong of Westinghouse had been checked and found accurate.

4. HOT CELLS AT MELEKESS

The hot-cell complex at Melekes, which consists of 36 cells, was toured. (The design of these cells was

D. SPECIAL FACILITIES

described previously in WASH-1060.) Of recent interest is the use of an easily strippable floor covering of 5-mm (0.2-in.)-thick polyethylene, the joints between sections being sealed to form an impenetrable layer that could easily be removed in case of a spill. This easily strippable covering was used extensively at other laboratories also—on stairways and stair railings, as well as on floors. Also note-

worthy at Melekes was the use of sulfur to mount specimens for grinding and polishing. Inert atmospheres were not used in the cell, but for certain special operations, the equipment was operated within a "localized" inert atmosphere. The cells are designed to handle 100,000 Ci of activity, but in some cells the activity limit is only 10,000 Ci.

APPENDIX E

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APPENDIX F

MATERIAL GIVEN TO U.S. DELEGATION

BROCHURES AND PHOTOGRAPHS

Beloyarsk—Book of 29 Photographs; Eight Photographs of Charts in Conference Room

Novovoronezh—Book of Photographs

Brochures—(a) Beloyarsk, (b) Novovoronezh, (c) Lenin Ice Breaker, (d) Melekess

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