ACCIDENTS AND TRANSIENTS IN FAST BREEDER REACTORS

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M.A. Michigan State University, 1987

MASTER OF ARTS

In
Translation and Interpretation

Graduate Division
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Translator’s Introduction

This text was chosen by my original thesis advisor, Mr. Arthur Braunstein, because I believe he felt it would help prepare me to face many of the difficulties that I would encounter when doing translation work in my chosen profession.

The author is I. A. Kuznetsov, who has researched transients in current fast reactors in conjunction with colleagues from the Physics and Energy Institute, using both the BN-350 and BN-600 reactors.

The book is divided into two parts. Part One analyzes transient conditions in fast reactors. Part Two is devoted to fast reactor accidents. All of Part Two, two chapters from Part One, the Introduction and the Conclusion have been translated. These sections were chosen because the remainder of the book is heavily laden with mathematical formulas and very little text. Therefore, they offered very little opportunity for translation experience. The translated portions focus upon reactor design, reactor protection systems, reactor operation, residual heat removal, design basis accidents, the shutdown of various reactor components, and reactor startup and shutdown.

The reactors referred to in this book are the BN-350 and BN-600 fast neutron reactors. Most of the reference sources that were used referred to these two reactors as the BN-350 reactor and the BN-600 reactor. Therefore, this designation has also been used in this translation.
This book is significant in that it is useful for operators, technicians, and those involved in reactor automation and monitoring. Moreover, the book is also useful for the young specialist just beginning his career in the field of nuclear energy.

Because this was a field in which I had no background, I spent many hours researching nuclear energy in general, before I could even begin to understand some of the problems and difficulties discussed in this book. I consulted with both American and Russian specialists in the nuclear industry. I used specialized books, conference papers, dictionaries and encyclopedic articles, all of which complimented each other in helping my understanding and translation of the topic.

There were four main difficulties in doing this translation. The first was deciding upon which terminology to use. Oftentimes there exist two or even more terms for the same concept. For example, the term “reactor time constant” and “reactor period” both refer to a rise or fall in the neutron flux density. When a problem such as this occurred, it was necessary to devise a standard for choosing a term, rather than randomly picking one of the two. Therefore, the decision was made that the term which is the most widely used and most current would be chosen. In the above case “reactor time constant” was chosen over “reactor period” because it has become the preferred term in the
Another example is the term “fuel rod,” which is sometimes called “fuel pin” or "fuel element." In this instance, “fuel rod” provided the most accurate and up-to-date description of a unit of nuclear fuel, and therefore the term “fuel rod” was chosen. And lastly, a third example was the choice between “reactor trip” and “reactor scram.” At first “reactor scram” appeared to be the term most widely in use, but upon closer examination, this was not the case. The term “reactor scram” is a hold-over from the early days of nuclear reactor protection systems, and current sources now say that the term “reactor trip” is preferred to “reactor scram.” In all cases, of course, either alternative would have correctly represented the concept in question. For the sake of consistency, and to be as accurate and precise as possible, the most current term was chosen.

The second main difficulty was dealing with Soviet terms that have no direct equivalent in English. Such a term is the “fast-acting reactor protection system.” Of course, the term “reactor protection system” is well known. The difficulty therefore arose in choosing an adjective to modify the term that will be understandable and precise in a case where one now does not exist. “Fast-acting” was chosen because it seemed to best describe the fact that the system would respond extremely rapidly for reactor safety. Other alternatives were “rapid” or “quick,” but neither seemed to fully explain the meaning as well as “fast-
acting" did. Because the text also mentioned a "slow-acting" reactor protection system, it seemed important to make a clear distinction between these two terms in the translation.

The third main difficulty was dealing with processes and problems that form the main bulk of Soviet attention whereas in the United States they are given less attention. For example, the discussion on design basis accidents focused on certain aspects of the problem which are of greatest concern in the former Soviet Union and are only briefly mentioned by U.S. sources, or not dealt with in any depth. This problem was not insurmountable, of course, but made it difficult to make the correct word choice, or difficult to understand the process or problem under discussion.

The fourth difficulty which was encountered involved the fact that the book which was translated was very highly technical, and not something to be undertaken by an amateur translator. Years of experience in translating, and interpreting work in the field of nuclear reactors have helped to overcome this barrier so that the translation could be completed. Indeed, sometimes nothing can replace the experience which is only acquired over time.

There is only one term that the reader might have a problem understanding. This term is “neutron guide tube.” All sources that I consulted seem to feel that it is a tube which serves as a
medium for the transport of neutrons. However, they could not be more specific.

There is only one word in the Russian text which appears to have been misspelled. In the English text it is the word “filled” and is located in paragraph 6 of Chapter 12. In the Russian text, instead of spelling it “zapolnenie” it was spelled “zapolenie.”

I would like to thank my current thesis advisor, Mr. Michael Gillen, for his extreme patience, and my husband Paul, for all his help.

Andrea E. Bergeron
The following work is a translation of
AVARIYNYE I PEREKHODNYE PROTSESSY V BYSTRYKH REAKTORAKH

by

I. A. Kuznetsov

Moscow
ENERGOATOMIZDAT
1987
Introduction

On June 27, 1954, the world's first 5-MW nuclear power plant went into operation in the city of Obninsk. Thirty years later, 374 nuclear power plants operating in 25 different countries throughout the world were producing approximately 250 GWe. The power output of the more than 40 nuclear power plants operating at that time in the U.S.S.R. exceeded 28 GWe. For the most part, they were either pressurized water reactors (PWR) or graphite-moderated, pressure-tube boiling water thermal reactors (GMBWR). Only two Soviet nuclear power plants are currently operating fast reactors: the BN-350 reactor in the city of Shevchenko and the BN-600 reactor in Unit 3 of the Beloyarsk Nuclear Power Plant. Although fast reactors contribute only a small percentage of total current nuclear power, they have the fastest potential for growth. Not only are they capable of generating heat, which can be converted into electrical energy, but they can also "breed" surplus nuclear fuel, i.e., plutonium. The term "surplus" means that the amount of secondary nuclear fuel produced in a fast reactor exceeds the amount of fuel consumed in one.

The ratio between the number of fuel nuclei produced in a reactor to the number of fuel nuclei consumed in the reactor over the same time period is called the breeding ratio (BR). Nuclear fuel must be bred because 99.3% of
naturally-occurring uranium is uranium-238, which will not fission in a thermal reactor. Only 0.7% of naturally-occurring uranium is uranium-235, which can serve as nuclear fuel in these kinds of reactors.

There are two possible fuel cycles that can breed nuclear fuel. The first fuel cycle involves irradiating uranium-238 with neutrons. Uranium-239 is formed when the uranium-238 nucleus captures a neutron. Resulting beta decays and transmutes the atom first into neptunium-239 and then into plutonium-239. The second fuel cycle involves irradiating thorium-232 with neutrons. When its nucleus captures a neutron, thorium-233 forms, which then transmutes into palladium-233 as a result of beta decay, and finally into uranium-233.

A plutonium fuel cycle is now being developed for fast reactors because insufficient fast neutrons are produced to maintain the thorium cycle.

The core of any nuclear reactor contains fuel, which is composed of fissile and source nuclides, coolant, and the structural materials of the fuel rods and fuel assemblies. A moderator is also inserted into the core of thermal reactors. The interaction of the neutrons with the nuclei of the moderator, a process known as elastic scattering, slows the neutron energy in such a reactor down to 0.005 - 0.2 eV, a level corresponding to thermal equilibrium with
the surrounding nuclei. The level of neutron energy in a fast reactor is close to the value at which neutrons are released when fuel nuclei fission (about 1 MeV).

The processes which occur in a nuclear reactor cause neutron production and losses. Neutrons are produced when fuel nuclei fission and are lost as a result of neutron leakage and capture by materials of the reactor. Reactor power can be controlled by inducing slight changes in the ratio between these processes, for example, by inserting an additional neutron absorber into the core, or by removing the neutron absorber from the core. When the reactor is operated in the steady-state mode, the number of neutrons produced is equal to the number lost. However, the relationship between the individual components which form the neutron balance in a fast reactor is more favorable than in a thermal reactor. This balance will be examined on the basis of one neutron being captured by a fuel nucleus. For this purpose, the components which form the neutron balance are as follows:

- the number of neutrons produced when the fuel atoms fission;
- the number of neutrons which appear when uranium-238 fissions;
- the number of neutrons lost from the reactor as a result of neutron leakage and radiative capture by nuclei of the structural materials and the coolant;
- the number of neutrons lost when they are captured by uranium-238 nuclei, and the nuclei do not fission.

The first two components have a plus sign, while the
latter two have a negative sign. As has already been stated, when the reactor is operating in the steady-state mode, the sum of these components is equal to zero. Therefore, the higher the value of the first two components and the lower the absolute value of the third, the higher the potential value of the fourth component. Since the calculation is carried out for one fuel nucleus lost as a result of neutron capture, this value is in fact equal to the Breeding Ratio (BR) of nuclear fuel in the reactor. Again, if neutrons are captured by uranium-238 nuclei and nuclear fission does not occur, then nuclei of a new fissile isotope are formed, i.e., plutonium-239.

Table 1 contains sample values for the components of the neutron balance listed above for thermal and fast reactors.

### Table 1. The Neutron Balance in Thermal and Fast Reactors

<table>
<thead>
<tr>
<th>Reactor</th>
<th>Fuel</th>
<th>Neutron Balance Components</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>Fuel Fissioning</td>
</tr>
<tr>
<td>Thermal</td>
<td>^²³⁵U</td>
<td>2.06</td>
</tr>
<tr>
<td></td>
<td>^²³⁵Pu</td>
<td>2.04</td>
</tr>
<tr>
<td>Fast</td>
<td>^²³⁵U</td>
<td>2.36</td>
</tr>
<tr>
<td></td>
<td>^²³⁵Pu</td>
<td>2.95</td>
</tr>
</tbody>
</table>
It is apparent that the primary components of the neutron balance in a fast reactor are more favorable than in a thermal reactor. When a fuel nucleus fissions, an average of $\nu$ secondary neutrons appear. It is necessary to keep in mind that not every neutron captured by a nucleus of fuel causes the latter to fission.

The probability that a neutron will be captured by a fuel nucleus without the nucleus fissioning is specified in neutron physics by a value known as the capture cross section, $\sigma_c$, while the probability that a nucleus will split after neutron capture is known as the fission cross section, $\sigma_f$. The ratio between these processes is very important. The aforementioned clearly shows that the first component of Table 1 is the value $\eta = \nu/(1 + \alpha)$, where $\alpha = \sigma_c/\sigma_f$.

Fast neutrons have a much lower $\alpha$ value and a higher $\nu$ value than thermal neutrons do. This is what enables a fast reactor to breed more fuel. In the U.S.S.R. this fact was first noted in 1949 by A. I. Leipunsky.

Thus, the first component of the neutron balance in a fast reactor will grow as $\nu$ increases and $\alpha$ decreases, unlike in a thermal reactor.

The second positive component, fissioning of uranium-238, is insignificant in a thermal reactor. The reason for this is that uranium-238 is only fissionable by high-energy
neutrons (more than 1 MeV), and there is only a small number of such neutrons in a thermal reactor. However, the percentage of feed-material-induced fission (uranium-238) is relatively high in a fast reactor.

The third component is neutron leakage and neutron capture by all reactor materials, with the exception of uranium-238. Of course, the third component cannot be less than one neutron, i.e., the very same neutron captured by the fuel and used as the basis for the calculations. The amount of neutron leakage and neutron capture by structural materials and coolant is minimized as much as possible. In order to reduce unnecessary neutron leakage, the fast reactor core is encircled by a breeding blanket, i.e., a region made of uranium-238. Within this blanket, additional amounts of nuclear fuel are produced and accumulated. In actual practice, the third component of the neutron balance has a value of at least 1.2. In sum, the nuclear fuel BR for a fast reactor may achieve a value of 2, while the nuclear fuel BR for a thermal reactor (PWR) can reach a value of 0.6 - 0.7. The BR for a metal-fueled fast reactor is greater than for an oxide fuel fast reactor. The oxygen in the oxide fuel slows down the neutrons from fast to thermal, a phenomenon that does not occur with metallic fuels.

In a system of \( n \) consecutive reactors that have a
positive BR and use both natural and bred fuel, it is possible to burn $1 + BR + BR' + BR'' + \ldots$ $BR^n$ kg of fuel for each kilogram of uranium-235. If the $BR < 1$, then the total of a sufficiently large number of members of this series tends towards the value $1/(1 - BR)$. When the $BR > 1$, this total can increase without limit as the number of reactors increases. This means that, in principle, fast reactors permit all the uranium which can be recovered from natural uranium sources to be burned. However, thermal reactors can only burn a portion of this uranium, a quantity represented by $\epsilon_{235}/(1 - BR) \approx 2\%$, where $\epsilon_{235} \approx 0.7\%$, i.e., the percentage of uranium-235 in naturally-occurring uranium.

Not only is it important to achieve a high breeding ratio for nuclear fuel in a fast reactor, but also a high rate of nuclear fuel accumulation. The rate at which fuel accumulates is determined by both the BR and the number of fission events in the fuel per unit of time, and consequently, by the core power density or the unit fuel load. A commonly accepted way to specify the nuclear fuel breeding rate in a fast reactor is with a value called doubling time, or $T_d$. This is the amount of time it takes for total power, or the number of nuclear reactors in the system, to double, provided that all excess fissile material accumulated in existing reactors is used to build additional reactors of the same type. A first approximation has shown
that the doubling time is inversely proportional to the product of a unit fuel load produced in reactors, $Q_T$, multiplied by the breeding ratio, i.e., $Q_T (BR - 1)$.

17 Of course, $T_2$ also depends upon other reactor parameters and system parameters as a whole. These include the fuel burnup fraction, fuel reprocessing time at the factory after the fuel has been discharged from the reactor, and relative losses during this reprocessing. The key is that a high rate of nuclear fuel recovery can only be achieved in a reactor in which a high BR is combined with high power density. These conditions lead to contradictory requirements: those measures and design alterations that could serve to raise the BR impede an increase in the core power density, and vice versa. Any attempts at compromise affect the parameters which are important for reactor operation and safety.

18 Every effort is made to maintain the most hardened neutron (energy) spectrum when fast reactor performance characteristics are being selected. It is true that in order to increase reactor safety, plans for artificially softening the neutron spectrum were discussed, but they were recognized as unacceptable.

19 In order to avoid an appreciable slowing down of the neutrons or their parasitic capture, the relative proportions of structural materials and coolant in the core
must be minimized, and only those materials which have appropriately small cross-sectional values should be chosen. On the other hand, the core power density can be raised by increasing the amount of coolant passing through the core, i.e., by increasing the coolant flow area and its unit volume. Possible coolant choices which have been considered are liquid metals and their alloys (sodium, sodium-potassium, or lithium), gases (helium, carbon dioxide, and other dissociating gases) and even steam. At the present time, fast reactor design engineers from countries all over the world tend to prefer sodium. In its capacity as a coolant, sodium has excellent physical and thermal properties. It is relatively ineffective in slowing down and capturing neutrons. In fact, although sodium's heat capacity per unit volume is four times less than that of water, the advantage of sodium is that its thermal conductivity is almost two orders of magnitude greater than water, and therefore its convective heat exchange is much more intensive. One of the most important advantages of sodium is that its boiling point at atmospheric pressure is around 900°C. This makes it possible to use atmospheric or close to atmospheric pressure in the fast reactor (pressure) vessel. This immediately eliminates the most difficult aspect of ensuring nuclear reactor plant safety—a rupture, or breach of the primary loop. Considerable effort and
expenditures are required in order to deal with this problem in thermal reactors, in which the pressure in the primary loop is high (16 MPa). Atmospheric pressure also makes for more favorable radiation conditions in fast reactors than in thermal reactors. As primary loop pressure increases, the probability of ruptures and leaks in the loop also increases, as well as the severity of the consequences. Even small ruptures and leaks are extremely unlikely in fast reactors.

The indicated advantages of sodium as a coolant also make it possible to use the high parameters of the nuclear power plant steam power cycle with fast reactors. The parameters that are chosen are practically identical to the ones used for traditional, non-nuclear power engineering. The sodium temperature at the outlet of a modern fast reactor is 550°C. Nevertheless, the power density in an oxide fuel fast reactor core can reach 900 - 1000 MW/m³ when the unit volume of sodium in the core is around 30%. The optimal value for the core power density is usually lower than the figure given above. For example, the average power density for the BN-600 reactor core is 550 MW/m³. For the sake of comparison, the average power density of a VVER-1000 reactor core is 110 - 115 MW/m³.

If metallic fuel is used in a fast reactor, limits on the temperature in the center of the fuel rod and the point
The contact between the fuel and the fuel cladding will not allow the core to achieve such a high power density when the coolant outlet temperature is significantly high enough. This is not the case for oxide fuel. In the case of oxide fuel, either the outlet temperature must be lowered by 100°C or more, or the power density must be reduced and the amount of coolant per unit volume must be increased in the core.

At the present time, thorough studies are being made of those power plant designs that will allow the advantages of metallic and high-temperature ceramic types of fuel to be combined.

For now, there are still some problems involved in making the transition from oxide to denser ceramic fuel (carbide or nitride).

The unit volume of sodium in the core can be limited by reducing the gap between fuel rods, and by heating the sodium to a higher temperature. This heating in individual channels of the fast reactor core can reach 250°C – 300°C, while in the reactor as a whole—170°C – 200°C. Consequently, even if only small power deviations occur during transients, marked changes in the reactor outlet temperature still occur. These temperature changes can themselves cause high thermal stresses within the structural elements. This thermal stress problem is complicated by the fact that sodium-cooled fast reactors use austenitic stainless steel
as the matrix structural material for the primary loop. This steel has relatively low thermal conductivity and a high thermal expansion factor. Because a high level of heat is transferred to the sodium, this also causes high thermal stresses in the plant elements contacted by the sodium. Because of this, steps must be taken to reduce transient thermal stresses when designing the reactor and the heat exchange equipment, as well as selecting a plant operating mode and an automatic control system.

25 It is important to note that the problem of transient thermal stress is not attributable to fast reactors alone, but occurs in practically any high-temperature energy installation. This problem is alleviated in a PWR by the fact that the water is only slightly heated (30°C) in the reactor, and heat is transferred to water and steam at a slower rate than it is to liquid sodium. However, this problem is exacerbated in a PWR-type reactor because of the greater thickness of the pressure vessel walls.

26 Because the power density in the fast reactor core is so high, a large surface for transferring heat from the fuel to the coolant is needed. Consequently, the diameter of the fuel rods is small: the external diameter is 6 - 7 mm and the can thickness is 0.4 mm. These dimensions are almost 1.5 times smaller than for PWR-type reactor fuel rods (9 - 10 mm), and two times smaller than for GMBWR fuel rods
(13.5 mm). Later in the book, it will be shown that the
time constant used to describe the thermal lethargy of the
core is proportional to the square of the diameter of the
fuel rod. Therefore, it is possible to conclude that the
thermal reactor core is much more lethargic than a fast
reactor core.

The power level of the fuel in a fast reactor is so
high, that even if the diameter of the fuel rods is small,
their unit load is still 500 – 600 W/cm. Consequently, the
oxide fuel in the center of the fuel rods has a temperature
of about 2000°C, and the fuel rod cladding can reach a
temperature of 700°C. The fast reactor fuel rods and fuel
assemblies must be fabricated out of materials that ensure
they can function at the temperatures mentioned above. The
radiative capture cross section of all nuclei is relatively
small in the fast reactor neutron spectrum. Therefore, there
is a much broader choice when selecting core structural
materials for the fast reactor than the thermal reactor.
One option in particular is stainless steel, which is
practically never used in thermal reactor cores. The
structural materials are compatible both with the fuel and
the sodium. In principle, this should preclude the
occurrence of dangerous events, such as a steam-zirconium
reaction during which hydrogen is formed. This reaction can
occur when the temperature rises in a thermal reactor.
It is necessary to keep in mind that the fuel rods operate in a hostile environment during both normal and emergency operation modes. Temperature and stress fluctuations in the fast reactor fuel rods are relatively large during emergency operation because the coolant in the reactor becomes quite hot. Because of this, more rigorous specifications are required for the reactor protection system response time and for the channel settings of this system. Modern hardware can meet these requirements without difficulty.

The uranium-235 and plutonium-239 fission cross sections in the fast reactor neutron spectrum are approximately two orders of magnitude smaller than those in the thermal reactor neutron spectrum. As a result, neutron flux in a fast reactor markedly exceeds neutron flux in a thermal reactor. Furthermore, more than 90% of the high energy neutrons in a fast reactor \((E > 0.1 \text{ MeV})\) are capable of altering the properties of the structural materials. This explains why neutron irradiation has a greater effect on the structural materials in a fast reactor than in a thermal reactor. Neutron irradiation is known to have three effects on the structural materials: (1) irradiation-induced embrittlement; (2) irradiation-induced creep; and (3) irradiation-induced swelling. Without a doubt, all three adversely affect safe reactor operation since all three play
a key role in determining the performance and operability of the fuel rods, the fuel assemblies, the control and safety system components, and the core as a whole during steady-state and transient operation. In particular, fuel rod strength determines the maximum permissible deviations for all reactor inlet parameters. Therefore, strength also affects specifications for quick response time, performance, and other parameters of the reactor protection system. Irradiation-induced swelling of the fuel rod cladding materials was considered to be one of the possible causes of sodium flow-area blockages in the fuel assembly. These blockages cause the sodium temperature to rise and can even cause the sodium to boil. Calculations and measurements of spent fuel rods removed from operating fast reactors have shown that irradiation-induced swelling cannot cause a marked reduction of coolant flow through the fuel assembly. However, there is no question that this swelling plays a negative role, one of the effects of which is that the fuel rod failure threshold may be lowered during an accident.

The fast reactor critical mass appreciably exceeds the critical mass of a thermal reactor of equal volume because both uranium and plutonium have a small cross section for fast neutron fission. However, the thermal reactor core volume must greatly exceed the fast reactor core volume in order to produce the same power level. Moreover, a large
supply of fuel must be kept for burnup because the thermal reactor fuel becomes highly contaminated with fission fragments. Consequently, when these two types of reactors are operating at equal power output, there is very little difference in the amount of fissile material being charged into these reactors.

Despite the aforementioned, some researchers are of the opinion that a fast reactor core should be watched more carefully in order to prevent a secondary critical mass from forming during an emergency fuel disruption or meltdown. This can happen because the fast reactor core is compact, and the fuel in the core is highly enriched. Contemporary thought on this subject is that such accidents should be considered hypothetical, since the possibility that they will occur is completely ruled out by the reactor design. This thought is based on successful fast reactor design developments and is reflected in standards of the U.S.S.R. and other countries.

The prompt neutron lifetime, a parameter whose values vary markedly in fast and thermal reactors, will be discussed next. At first glance, this difference seemed very important in terms of reactor safety. In fact, the high energy level and the speed of the neutrons in a fast reactor mean that the average time from the appearance of neutrons during the fission event until the neutrons are
captured by nuclei of the core materials and the next generation of neutrons is produced is 4-5 times shorter in a fast reactor than in a thermal reactor. This time interval, \(l\) (the average lifetime of prompt neutrons), is \(10^{-3}\) seconds in a thermal reactor and \(10^{-7} - 10^{-8}\) seconds in a fast reactor. However, practice has shown that this difference has not led to any serious problems because the level of reactivity in all possible accident conditions is significantly lower than the actual proportion of delayed neutrons, \(\beta\). Under these conditions, changes in the neutron flux and, consequently, changes in reactor power are mainly determined by delayed neutrons, whose average lifetime in both thermal and fast reactors is appreciably long: about 10 seconds. Therefore, the indicated difference in \(l\) is only significant for hypothetical accidents in which there are large fluctuations in reactivity and there is a runaway of a prompt critical reactor.

Something should also be said about the varied nature of temperature effects of reactivity in both fast and thermal reactors. In a relatively large-volume fast reactor (more than 2,000 liters), it is specifically the resonance neutrons, despite their relatively small number, that make the greatest contribution to the temperature effects of reactivity. The main role is played by capture resonances and the Doppler effect of feed materials. However, it must
not be forgotten that the structural materials exhibit similar resonances, and the fuel exhibits fission resonances. Placed in order of absolute value, the components of the temperature effects of reactivity are affected by a change in coolant density, coolant displacement from the core, and a change in reactor size.

However, in a thermal reactor the main components of the temperature effects of reactivity are affected by deviations from the initial moderator density value, and by a corresponding change in neutron energy. These effects may not be apparent for a long time. In a PWR-type reactor, the magnitude of the temperature coefficient of reactivity is one order of magnitude higher, and in a GMBWR reactor 1.5 - 2 orders of magnitude higher than in a fast reactor. Nevertheless, as will be shown later, a fast reactor operates very well and reliably in the self-regulation mode, and it is not difficult to control. One of the reasons for this is that the mean free path of fast neutrons is appreciably larger than that of thermal neutrons. As a result, parts of a fast reactor core are more closely interrelated in terms of neutron physics, than in a thermal reactor. So far, there has therefore been no problem with spatial neutron flux distribution instability. Even during extreme realignments of the control rods, distortions of the power density field in a fast reactor are not very great.
However, these distortions can be quite large in a thermal reactor, even to the point where a local supercriticality may occur. Attempts to control spatial instability tend to make the automatic control system for these distortions much more complex.

There is one additional factor that can simplify fast reactor control during transient operations. It is the fact that xenon and samarium poisoning do not occur in a fast reactor because fission fragments have a small capture cross section in the fast reactor neutron spectrum.

Substances containing hydrogen and carbon have high moderating capability. When they are inserted into the fast reactor core, they significantly soften the neutron spectrum. Therefore, even if only a small quantity of these substances enters the core, it can cause a significant rise in reactivity. However, if a moderating substance enters the core of a relatively large-volume fast reactor, this can produce a negative effect, i.e., it can reduce reactivity. This can be accounted for by the effect all the plutonium isotopes have, plutonium being the preferred fuel for fast reactors. All fast reactors in operation or on the drawing board are designed with features to prevent moderating substances such as pump lubricating oil from entering the core.

A discussion of thermal and fast reactors would not be
complete without mentioning the problem of after-heat removal. After-power in both fast and thermal reactors results from beta and gamma decay of the fission products, and radiative capture products as well. When the reactor is brought to the required power level, the concentration of these products gradually begins to increase, and eventually reaches a peak steady-state value when the number of decaying nuclei equals the number of newly formed nuclei. Fission fragment decay constants lie within the range of a fraction of seconds to several million years. Therefore, it is virtually impossible to achieve the indicated equilibrium for some of these constants. However, if total power generated as a result of decay is chosen as the criterion, then it is fairly certain that a steady state will be reached within 100 hours of reactor operation. Further, in both thermal and fast reactors, the component of heat caused by fission fragment decay and radiative capture products is 6 - 6.5% of nominal power. There is also very little difference in the rates at which after-power decays after both reactors have been stopped. Because of this, after-power removal from a fast reactor does not pose any additional problems even though the energy level is high in this type of reactor. Moreover, heat removal is a relatively simple process because of sodium's excellent thermal and physical properties as a coolant and the high heat capacity.
of the loops. This problem once again demonstrates the advantage of low pressure in the sodium loops of a fast reactor, which avoids difficulties such as those caused by rapid coolant loss if the loop ruptures in a thermal reactor, which should be designed with measures that would limit and replace these losses in order to remove residual heat after the rupture takes place.

In sum, it can be concluded that fast reactors not only have better prospects in the nuclear power industry because of their ability to breed excess nuclear fuel, but also because their performance characteristics promise to make them one of the safest and most easily controlled types of nuclear reactors. This is very important because expenses for safety will continue to grow as nuclear power develops in the future and corresponding specifications inevitably become stricter. Therefore, safety will become an increasingly important factor during comparative assessments of various nuclear reactors.

It was almost 40 years ago when the idea of the nuclear reactor was born. Since then, time has primarily been spent on researching fast neutron physics, mastering the technology of sodium as a coolant, searching for structural materials, and choosing a fuel. Because of this, fast reactors have not had time to prove their worth. The main problem is that they are still more expensive than thermal
reactors. However, efforts of scientists, engineers and designers in the U.S.S.R. and overseas have reduced the differences in their costs. The circumstances listed above should be favorably impacted by these efforts.

The first nuclear reactors in both the U.S.S.R. (BR-1 and BR-2) and overseas (Clementine in the US, and Zephyr and Zeus in Great Britain) were, in point of fact, assemblies that were small in size and used to study their neutronics. Then the time came to build experimental sodium-cooled fast reactors. This stage lasted for various lengths of time in different countries, and in some countries experimental fast reactors continue to be built to this day. The list of experimental fast reactors includes the following. In the United States they are the 0.2 MWe EBR-I (1951), the 61 MWe Enrico Fermi (1963), the 20 MWe EBR-II (1965), and the 400 MWth FFTF (1979). In the Soviet Union they are the BR-5 and BR-10 (1958) operated at 5-8 MWe, and the 60 MWth BOR-60 (1969). In Great Britain it is the 15 MWe DFR (1962). France had the 40 MWth Rapsodie (1967). For the Federal Republic of Germany there was the 21 MWe KNK-2 (1977). Japan had the 100 MWth JOYO (1977), and India had the 15 MWe FBTP (1986). The 140 MWth Italian PEC is under construction.

Practical experience gained when these reactors were operated allowed the construction of fast power reactors to be undertaken. The first reactor of this type was the
Soviet BN-350 reactor, which was brought into power operation in June 1973 in the city of Shevchenko. A small percentage of its power output was earmarked for generating 150 MWe of electrical energy, and the remaining large percentage to produce 120,000 ton/day of distilled seawater. After the BN-350 reactor, the British PFR reactor and the French Phénix reactor became operational in 1974. The electrical capacity of each one of these reactors was 250 MWe. The BN-600 reactor, which was the next Soviet sodium-cooled fast reactor, came on line at the Beloyarsk Nuclear Power Plant in April 1980. At that time its electrical capacity was already 600 MWe. The Super Phénix reactor, which was started-up in 1986 in France, had an electrical capacity of 1,200 MWe. At the present time, the SNR-300 fast reactor is under construction in West Germany. It will have an electrical capacity of 327 MWe. The MONJU reactor with a capacity of 300 MWe is also under construction in Japan.

When the first fast power reactors were being designed and built, various concepts and options for the design of the reactor, heat exchange equipment, and the plant as a whole were compared and selected. These types of choices are still made today. For example, whether the fast reactor primary system should have an integrated-type of construction (pool-type) or a loop-type design has not been
decided once and for all. There are supporters for both types of configuration in a number of different countries. By researching and developing these configurations further, newer lines of reasoning and arguments in support of each one of them can be presented. Only the BN-350 reactor in the U.S.S.R. has a loop-type design in both the primary and secondary loops. Beginning with the BN-600 reactor, the integrated-type of primary loop has been used. This type of integrated configuration has also been used in the design of next-generation Soviet reactors, i.e., the BN-800 and the BN-1600 reactors. At the present time, an integrated-type construction completely dominates in French and English designs. The loop-type design, however, still predominates in American, West German, and Japanese designs.

Countries such as the U.S.S.R., France and Great Britain, which lead the industry in this technology, have designed fast neutron reactor plants whose unit capacity has reached 1200 - 1600 MWe. There are plans to build even more power stations of this type. Therefore, when fast reactors, or thermal reactors for that matter, are being designed, the quantity and quality of efforts expended on researching and substantiating reactor operational safety increases.

During the initial design phase of the BN-350 reactor, studies of safe reactor operation not only included demonstrating the operability of all the reactor’s elements
during normal plant operation, they also involved an analysis of all possible accident conditions. The list of these accident conditions was compiled by sorting out the main unit failures and plant assembly failures, and overlapping those failures considered most probable by the specialists. However, as the number of units being designed increased, and the spectrum of engineering decisions which had to be made when creating nuclear reactors broadened, it became necessary to systematically organize reactor operational safety studies and regulate this work with the use of specific documentation. Such standardization was designed to eliminate oversights or mistakes in work, and to simplify and improve scrutiny of designs by regulatory agencies. This very same approach was not only used for fast reactor safety studies, but for thermal reactor safety studies as well.

This work could not be systematically organized without preparing standardized documents, which is what was done. The main ones are Pravila yadernoy bezopasnosti (PBYa-74) [nuclear safety regulations], which was published in 1974, and Obshchie polozheniya obespecheniya bezopasnosti atomnykh stantsiy pri proektirovanii, sooruzhenii i ekspluatatsii (OPB-82) [general provisions ensuring nuclear plant safety during design, construction, and operation], which was published in 1982. These documents regulate performance
characteristics of nuclear reactor systems responsible for reactor safety. They also regulate research on safety during the design and start-up phases, and measures to ensure reactor safety during operation. These documents contain organizational and engineering requirements which must be met in order to ensure nuclear power plant safety. All of PBYa-74 and most of OPB-82 contain requirements that apply to all types of reactors. Only special sections of OPB-82 contain additional safety requirements for nuclear power plants with reactors of various types and applications.

A nuclear power plant is considered to be safe if equipment and organizational measures limit both the irradiation of plant personnel and the population to within acceptable limits, and also the amount of radioactive products in the environment during normal operation and all possible accidents. Radiation safety standards (NRB-76) have established maximum permissible doses of radiation to which plant personnel and the population can be exposed, as well as the limit of acceptable radioactive products which can be contained in the environment.

When studying reactor safety, a very important issue that must be taken into account is the list of equipment and plant system failures, including possible combinations or overlaps. The project engineer, the chief designer, and the
plant research supervisor use the single-failure principle when conducting engineering studies of all the plant components in order to compose the list of possible accident initiating events. If necessary, these studies should consider data on the reliability of the relevant components. However, using existing methodologies to determine whether or not a given failure may occur can only give the fixed responses "yes" or "no." Probability studies for analyzing reactor safety have only just begun to be applied for the purpose of designing nuclear power plants.

Therefore, each initiating event is either an isolated failure in the plant's systems, an external event, or human error, which can cause the operational safety limit to be exceeded. An initiator can also include any secondary failures resulting from the primary one. Accident chronology from initiation should be researched in detail by using mathematical modeling and, if necessary, experiments should be conducted in order to establish the ways and means necessary to ensure plant safety. The research findings are described in a document entitled, Tekhnicheskoe obosnovanie bezopasnosti sooruzheniya i ekspluatatsii atomnyy stantsii [A technical study of nuclear power plant structural and operational safety], which is included as part of the plant design, and is examined and approved by federal nuclear reactor safety regulatory agencies.
OPB-82 prescribes the following priorities to possible propagation of failures in the plant's safety systems, i.e., in those systems that alert that an accident is taking place and mitigate its consequences. The nuclear power station design should provide both hardware components and administrative measures to ensure reactor safety if any one of the design-based initiating events should occur, compounded by a failure of any of the following components in the safety systems not dependent on the initiating event: an active or passive component that has mechanical moving parts. Active components in this case are those components whose functioning depends upon the normal operation of another device such as a control device or the power source. Passive components are those components whose functioning does not depend upon the normal operation of other devices. In addition to a failure of one of the components listed above, another type of failure must be considered. This is a previously undetected failure of devices that are not monitored when the nuclear power plant is operating and that affect accident chronology. In individual cases where such devices are shown to have a high level of operational safety, the possibility that they will fail may be overlooked.

In the above instances, very high standards are applied to the safety control systems. These systems should be
multi-channeled, with all channels being independent so that any individual failures in these systems, including common-mode failures, will not disrupt their performance.

Monitoring and control of the reactor and other plant systems during normal operation and during accidents are carried out from the unit control panel. It should be possible to start up the safety systems and receive data on the status of the reactor from a back-up control panel if, for some reason (fire and so on), it is not possible to perform this function from the unit control panel.

Before the core is designed, the acceptable number of damaged fuel rods, the degree of their damage, and associated levels of primary coolant radioactivity must be established. These levels are used to determine what type of radiation shielding is necessary. At the same time, the specifications of the core and the design of the reactor and other primary equipment should preclude the formation of a critical mass during any type of accident, including one which leads to a disruption of the reactor core or a fuel meltdown. If this condition cannot be fulfilled, then it is necessary to show that with the given design, a core disruption or a fuel meltdown leading to criticality will not occur even if additional failures take place in the safety systems or additional demands are made, depending on the type of reactor. At the present time, it has not been
possible to prove that a critical mass will not form in fast reactors when nuclear fuel melts down during an accident. There is a section in OPB-82 containing additional safety requirements for nuclear power plants with reactors of various types and purposes. This section requires a justification that fast reactor core melting and upsets will not lead to the formation of a critical mass even if, in addition to the failures listed above, a failure of an active component or safety system device occurs. Because of this, several systems contain a redundant back-up of the most important components.

The following two accidents must be independently examined as a design basis accident (DBA) for fast reactors. The first accident is a rupture of the primary loop piping along a section that is not double-walled. The second accident is a narrowing or blockage of the coolant flow area in one individual fuel assembly as a result of material swelling, the deposition of impurities from the coolant, or intrusion of foreign objects into the reactor, all of which can lead to reduced coolant flow through the assembly, fuel rod damage or melt down, and propagation of the damage to one row of adjacent fuel assemblies. At the present time, the reactor design and core specifications that are specifically selected are the ones that practically eliminate the causes of the aforementioned fuel assembly
damage. Nevertheless, it is postulated that the DBA we have just discussed, along with a fuel assembly fuel meltdown, can occur in the current fast reactor designs pending the formulation of rigorously strict evidence to the contrary. In this respect, a case in point is the DBA investigated 15 - 20 years ago for foreign designs of fast reactors. It involved a loss of all primary circulating pumps with an attendant total failure of the reactor protection system. Studies have shown that the probability of such an accident occurring is extremely low. Therefore, in a number of countries that are developing fast reactor designs, it has been decided to transfer this type of accident to the hypothetical category, which can be seen in the relevant documents.

The final justification of nuclear power plant safety is completed during the physical and power start-ups of the reactor. Start-up programs are used to conduct tests that help to establish the reliability and accuracy of theoretical and mathematical research, and make determinations about those parameters of the reactor and other plant components that could otherwise only be found by calculations.

The objective of this book is to give the reader an idea about the scope of research and the types of methodologies used to research transient and accident
conditions in fast reactor operation, i.e., methods necessary to substantiate reactor safety. This book is intended for nuclear reactor operating personnel who are becoming more active and useful assistants to the research supervisor, the chief designer, and the project engineer at all stages of this type of research. As the material is presented, the focus will be on the physics of phenomena and on the parameters describing them from the standpoint of safety. Analytical procedures and theoretical models have been simplified and are cited so that they may be used to interpret the findings on the one hand, and on the other to give the reader the basis to make his own, independent judgments.

Transient and accident conditions of fast reactor operations cannot be described without mentioning reactor design, layout, and normal operating modes. So that the reader will not have to refer to other sources, these issues will be briefly touched upon in Section 1.* Further, the book will discuss ways to approximately calculate transient conditions in nuclear power installation components. Section 2 is devoted to an analysis of accident conditions in fast reactors.

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* TN: Only Chapters 1 and 7 of Section 1 have been translated.
1. Fast Reactor Design and the Heat Transport System

1. Depending on the configuration of their primary loop, existing fast reactors can be divided into integral-type (pool-type) and loop-type. A pool-type configuration means that almost all the primary equipment is located in one tank. In plants with a loop-type arrangement, the intermediate heat exchangers (IHXs) and primary pumps are located outside of the reactor tank.

2. Typical examples of a plant with a loop-type and a pool-type configuration are the BN-350 and BN-600 reactors, respectively. Therefore, describing them gives a fairly good idea of what these two types of reactors are like.

3. The BN-350 reactor has a thermal capacity of 1000 MWth. Some of its parameters are given in Table 2.

4. The reactor is placed in a stainless steel tank with a maximum diameter of 6 m, height of 13 m, and wall thickness of 30-40 mm (Figure 1). The lower portion of the tank (the reactor pressure vessel) is a pressure chamber. Sodium is fed to the pressure chamber by the primary pumps at a pressure of 1 MPa. Sodium is brought to the chamber by six 500-mm diameter pipes. There are six, since that is the number of heat removal loops.
Figure 1. The BN-350 reactor: 1 - reactor vessel; 2 - large rotating plug (shield); 3 - small rotating plug (shield); 4 - central column with the control and protection system mechanisms; 5 - assembly transfer mechanism; 6 - transfer chamber; 7 - loading-unloading elevator; 8 - upper stationary shielding; 9 - reloading mechanism; 10 - core; 11 - reactor support; 12 - side shielding; 13 - header chamber.
### Table 2. Main Parameters of the BN-350 and BN-600 Reactors

<table>
<thead>
<tr>
<th>Parameter</th>
<th>BN-350</th>
<th>BN-600</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Capacity</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Thermal power, MW</td>
<td>1,000</td>
<td>1,470</td>
</tr>
<tr>
<td>Electrical rating, MW</td>
<td>150</td>
<td>600</td>
</tr>
<tr>
<td><strong>Primary and secondary sodium temperatures</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Reactor inlet, °C</td>
<td>300</td>
<td>380</td>
</tr>
<tr>
<td>Reactor outlet, °C</td>
<td>500</td>
<td>550</td>
</tr>
<tr>
<td>Steam generator inlet, °C</td>
<td>450</td>
<td>520</td>
</tr>
<tr>
<td>Steam generator outlet, °C</td>
<td>270</td>
<td>320</td>
</tr>
<tr>
<td>Sodium flow through reactor, t/h</td>
<td>14,100</td>
<td>24,000</td>
</tr>
<tr>
<td><strong>Live steam parameters</strong></td>
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<td></td>
</tr>
<tr>
<td>Pressure, MPa</td>
<td>5</td>
<td>14</td>
</tr>
<tr>
<td>Temperature, °C</td>
<td>435</td>
<td>505</td>
</tr>
<tr>
<td><strong>Core dimensions, m</strong></td>
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<td></td>
</tr>
<tr>
<td>Equivalent diameter</td>
<td>1.58</td>
<td>2.05</td>
</tr>
<tr>
<td>Height</td>
<td>1.06</td>
<td>0.75</td>
</tr>
<tr>
<td>Height of axial shielding, m</td>
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<td>0.4</td>
</tr>
<tr>
<td>Number of enrichment zones</td>
<td>2</td>
<td>2</td>
</tr>
<tr>
<td>Number of assemblies in these zones</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Low enrichment zone</td>
<td>109</td>
<td>209</td>
</tr>
<tr>
<td>High enrichment zone</td>
<td>117</td>
<td>162</td>
</tr>
<tr>
<td><strong>Number of assemblies in the side shielding</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>1</td>
<td>412</td>
<td>380</td>
</tr>
<tr>
<td><strong>Average core power, MW/m³</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>440</td>
<td>550</td>
</tr>
<tr>
<td><strong>Maximum linear power of core fuel rods, kW/m</strong></td>
<td>44</td>
<td>53</td>
</tr>
</tbody>
</table>

*In addition, 120,000 tons of fresh water are produced daily.

Above the header chamber is a pressure header, which consists of two horizontal, perforated plates joined together by cylindrical tubes and throttling sleeves. The sodium enters the header through special orifices in the lower plate.

The inner drum in the header divides it to form a high pressure zone and a low pressure zone. The lower ends of the core and blanket fuel assemblies are mounted into the
slots of the upper and lower plates of the pressure header. When this is done, the core fuel assemblies and the two inner rows of the blanket assemblies, all of which release a comparatively high amount of energy, are set into the high pressure zone. The remaining blanket fuel assemblies are mounted into the low pressure zone. The side walls of the fuel assemblies have slots for passage of sodium to the fuel rods. The throttling sleeves in the header close off a given number of these holes. The number of holes open for the sodium to pass through varies for each assembly. This ensures that as the sodium flows between the assemblies, its distribution is correctly patterned after a radial power density distribution in the reactor, meaning that sodium temperature at the outlet therefore becomes equalized in a radial pattern.

During initial charging each core fuel assembly contains 169 fuel rods, each of which is 6.1 mm in diameter. These rods are arranged in a triangular lattice with a pitch of seven millimeters. The stainless steel fuel cladding is 0.4 mm thick. Uranium dioxide serves as the fuel. In order to equalize core power density, a two-region approach to fuel enrichment is used. The upper and lower axial blanket fuel rods abut the ends of the core fuel rods. The former have an outside diameter of 12 mm. The stainless steel fuel cladding is 0.4 mm thick. Thirty-seven lower and 37 upper
axial blanket fuel rods are placed in each fuel assembly.

The core is surrounded by side shielding (breeding region). This shielding is composed of assemblies that have the same outward configuration as the fuel assemblies of the core. The side blanket fuel rods have an outside diameter of 14.2 mm, and the stainless steel fuel cladding tube is 0.5 mm thick. There are 37 fuel rods in each side blanket fuel assembly.

When sodium flows through the side blanket and the core fuel assemblies, it enters the upper portion of the reactor vessel, which has an approximate volume of 100 m$^3$. The inner surface of the reactor vessel is lined with stainless steel plates, which act as a thermal shield to prevent against large thermal stresses in the vessel when the sodium temperature undergoes rapid changes. The thermal shielding plates have a combined thickness of 60 mm. The reactor vessel is cooled by sodium which travels from the pressure vessel through the clearance between the reactor vessel support wall and the thermal shield. The outside of the reactor vessel is surrounded with a 10 mm-thick guard vessel. The clearance between the reactor and guard vessel is selected so that the level of sodium in the vessel will be sufficient to maintain sodium circulation through the fuel assemblies if the clearance space begins filling up with sodium as a result of a vessel leak.
The upper portion of the vessel has six pipes with a diameter of 600 mm each. Sodium is removed along these pipes to the IHXs.

**Figure 2.** BN-350 heat flow diagram: 1 - reactor; 2 - IHX; 3 - primary main coolant pump; 4 - steam superheater; 5 - evaporator; 6 - secondary main coolant pump; 7 - cold trap for oxides.

Figure 2 shows the heat flow diagram of the BN-350 plant. It has six identical heat removal loops, five of which are operational while the sixth acts as a standby. Sodium is transported from the reactor along suction pipes to the IHXs. Each of these is a shell-and-tube heat exchanger with counter flow and transverse flow of coolant (Figure 3). Primary sodium flows through the shell (the space between the tubes of the heat exchanger), and
secondary sodium flows through the tubes. Secondary sodium pressure in the IHX exceeds primary sodium pressure no matter what the operating mode. This ensures that radioactive coolant does not leak into the secondary loop. A main coolant pump (Figure 4) is located downstream of the IHX in each loop. The pump draws sodium in directly from the suction piping. When sodium leaks out of the discharge portion, it passes through the seal and into the pump tank. From there the sodium travels through the leak drain tank which has a liquid level switch and again to the suction intake line.
Each GTsN-1 pump outlet has a check valve. The sodium travels from the pumps along piping which is 500 mm in diameter to the reactor pressure vessel. The piping is 50 meters in length from the reactor to the IHX.
Secondary sodium is delivered from the IHX to the steam generator. The steam generator of each loop consists of two evaporators and two steam superheaters installed in parallel for sodium and water-steam. Secondary sodium is received by the steam superheater from the IHX. Each steam superheater has 805 U-tubes for the steam to flow through. Sodium flows through the shell counter to the flow of steam. Sodium is fed to the evaporator after passing through the steam superheater. In closed-vessel-type evaporators, Field tubes
are used as heat-transfer elements. Of course, these tubes operate on the principle of natural circulation of the steam-water mixture (Figure 5). Sodium enters each evaporator vessel at the bottom and exits through a flange at the top of the evaporator. In the evaporator, there is a gas blanket located above the free surface of sodium. This gas blanket is linked to a special gas reservoir shared by both evaporator vessels. The reservoir is designed to partially collect the sodium discharged from the steam generator, as well as hydrogen, in the event of an accidental steam generator pipe rupture. The secondary pumps are located on the "cold" side of the steam generator, downstream from it, along the coolant flow path. These pumps have the same design as the primary pumps. The primary and secondary pumps are electrically operated. The motors are the squirrel-cage induction-type with dual speed (they have two windings in the stator). The pump has two rotational speeds. One is furnished by the primary winding and is rated at 1,000 rpm. The other rotational speed is geared down to 250 rpm. The latter is used for plant shut-down auxiliary operations (such as transferring and removing after-power from the reactor). Each circuit of the plant's primary and secondary loops is equipped with isolating valves, i.e., gate valves with a remotely controlled electric drive. The primary gate valves are located on the
suction and discharge sides of the loop, i.e., upstream of the IHX and downstream from the pump, but before the check valve. The primary piping along both the suction and discharge sections from the reactor to the isolating valves is double-walled, which prevents a large amount of sodium from being released in the event that a pipe ruptures.

Figure 5. BN-350 reactor steam generator diagram: 1 - evaporator; 2 - superheater; 3 - superheated steam; 4 - separator; 5 - water level; 6 - sodium level; 7 - Field tube; 8 - drain tank; 9 - safety membrane.

The reactor control and protection system contains the following absorber rods: (1) three emergency shutdown rods with a reactivity worth of 1.5 β for each of these rods; (2) two automatic control rods with a reactivity worth of 0.3 β for each rod; and (3) one rod to compensate for temperature and power reactivity effects. The reactivity
worth of the rod is 1.5 $\beta$. In addition, the reactor control and protection system has six fuel rods to compensate for the effect of fuel burnup (shim rod).

15 The reactor protection system is subdivided into the fast-acting reactor protection system and the slow-speed reactor protection system. When the fast-acting reactor protection system is activated, the current to the electromagnets is broken. The emergency shutdown rods, which are suspended from the electromagnets, drop into the core by virtue of their own weight. The drop is accelerated by the use of special accelerating springs.

16 It takes one second to insert the emergency shutdown rods into the core using this method. When the fast-acting reactor protection system signal is actuated, the automatic control rods and the temperature and power reactivity effects rods are simultaneously motor-driven into the core at maximum possible speed. Also, when the fast-acting reactor protection system is tripped, the primary and secondary coolant circulating pumps are geared down from rated to a reduced rotational speed.

17 In the slow-speed reactor protection system mode, the rods are motor-driven into the core at a rate of 1.3 mm/s. One of the control rods also moves downward at a minimum speed of 10 mm/s, which is the speed at which the rods move when they are remotely controlled. When such low speeds are
used to insert the absorber rods into the core, the rate at which reactor power and sodium outlet temperature are reduced will be accordingly slower, than if the fast-acting reactor protection system had been functioning.

In addition to reactor power, other aspects of the plant under automatic control are the following: maintaining water levels above the Field tubes in the evaporators, and maintaining steam pressure in the steam generator. Water levels are sustained by well-known 3-position controllers, which receive signals about changes in water and steam flow rates, and changes in the water level in the evaporator. On the basis of this information, the controllers regulate the positioning of the steam generator feed valves. Small-diameter, bypass feed lines are used when the after-power removal mode kicks in. At this time the amount of power supplied to the steam generator, and therefore the feed water flow rate through it, are markedly reduced. The steam generator pressure is maintained by controllers at a natural, constant level, which affects the valves that change the flow of steam from the steam generator.
Figure 6. The BN-600 reactor: 1 - roller support of the reactor; 2 - safety tank; 3 - reactor vessel; 4 - main coolant pump; 5 - protective cover; 6 - drives for the reactor control and protection system; 7 - rotating plug; 8 - central rotating column; 9 - IHX; 10 - radiation shielding within the tank; 11 - core; 12 - breeding blanket; 13 - header chamber; and 14 - inlet pipe for the gas heating.
System.

Figure 7. Intermediate heat exchanger of the BN-600 reactor: 1 - secondary sodium inlet; 2 - downcomer; 3 biological shielding; 4 - primary sodium inlet; 5 - vessel 6 - primary sodium outlet; 7 - lower manifold; 8 - distribution grate; 9 - heat transfer pipe; 10 - spacing grate; 11 - expansion bend; 12 - inner shell; 13 - upper tube plate; 14 - heat exchanger support; 15 - secondary sodium outlet.
Figure 8. Primary pump of the BN-600 reactor: 1 - check valve; 2 - lower scroll; 3 - impeller; 4 - upper scroll; 5 - hydrostatic bearing; 6 - shaft; 7 - caisson.
In the standard operating mode, feed water is fed to the steam generator by electric feed pumps (PEN), one of which is located on each heat removal loop.

The BN-350 reactor operates in the power-generating range when the flow of coolant in the primary and secondary loops is maintained at a constant level. In addition to rated duty on five heat-removal loops, it is also possible to operate at reduced power on four or three loops.

The BN-350 is among the pilot fast breeder reactors.

The next-generation BN-600 reactor can already be referred to as an industrial reactor in light of its characteristics and designated purpose (see Table 2). With the exception of certain auxiliary systems, all the reactor's primary equipment (the reactor, the IHXs, and the primary reactor coolant pumps) is located in one vessel. This tank is cylindrical in shape, has an elliptically shaped top and bottom (heads) with a truncated cone on each end, is 12.8 m in diameter, and is 12.6 m high. The body wall along the cylindrical portion is 30 mm thick. Rotating plugs are installed on the (top) head of the reactor. The reactor is refueled through these plugs. The reactor pressure vessel is surrounded by a safety tank, which prevents the sodium from falling to a dangerously low level if there is a leak.

Heat is transferred from the reactor to the steam
generator along three parallel loops in a three-coolant-loop system. In the primary system (in the reactor vessel), each loop contains two IHXs and a main coolant pump with a controlled check valve and two parallel sections of pipe which connect it to the reactor pressure header. Figure 6 shows a cross-sectional view of the reactor.

Primary sodium passes through the reactor fuel assemblies and enters the space above the reactor. From there, it flows between the vertical tubes of the radiation shielding which surrounds the core and the side shield. It then proceeds to the inlets of the IHX, which is a vertical shell and tube heat exchanger with a so-called "floating head." The design of the IHX is shown in Figure 7. Primary sodium enters from the top of the IHX and flows downward across the tubes on the outside. Secondary sodium enters the IHX from above. It descends through the center tube to the lower head and then ascends through the heat-transfer pipe. When primary sodium leaves the IHX, it enters three discharge chambers. Each of these chambers merges the draw-off from the two IHXs and is connected to the suction-side of one of the reactor circulating pumps, which are mounted in the reactor pools. The primary loop contains a submersible pump with a sodium-operated lower hydrostatic bearing (Figure 8). The two-way suction rotor impeller is connected to the vertical bracket of the shaft. The length
of the pump shaft is 7.6 m and it has a maximum diameter of 0.68 m. Sodium leaking from the hydrostatic bearing is drained through special openings to the pump suction. The pump has a free surface of sodium in the (reactor) vessel. The difference between the sodium level in the reactor vessel and that in the pump is determined by the hydraulic resistance of the IHX. The pump has an isolating check valve that is 850 mm in diameter, and a hydraulic driver. This valve closes when the pump stops operating, while the two other pumps of the loop maintain operations. It takes three seconds for the valve to close. The pump is driven by an induction motor that has a wound rotor. The speed of the motor is controlled by an asynchronous isolator stage between its rated capacity of 970 rpm down to 250 rpm. Primary sodium is delivered along two pipelines from each pump to the reactor pressure header. The delivery lines are double-walled, which safeguard against the dangerous consequences of a loss of pipe integrity. The sodium is then fed from the pressure header to the fuel assembly modules. Each module contains up to seven fuel assemblies. The dimensions of the feed groove in the module for each fuel assembly determine the flow rate of the sodium through the module. In order to equalize the power density in the core, doubly enriched fuel is used. This is done in the BN-350 reactor as well. The size of a turnkey hexagonal fuel
assembly is 96 mm. The pitch between their emplacement is 98 mm. Each core fuel assembly contains 127 fuel rods with a pitch of 8.05 mm in a triangular lattice. The fuel rods are spaced by using wire spacers wrapped around the cladding. A cylindrical fuel rod has stainless steel cladding with an outside diameter of 6.9 mm and wall thickness of 0.4 mm. The cladding is filled with uranium dioxide inserts. Axial shields made out of depleted uranium dioxide are located above and below the core. The core and axial shield fuel rods are located in individual tubes. The overall length of a fuel rod is 2.445 m. The lower part of the fuel rod contains a gas cavity designed to collect gaseous fission products of the fuel.
Figure 9. BN-600 reactor core fuel assembly.

A side fuel assembly (the side breeding blanket) contains 37 fuel rods which have an outside diameter of 14.1 mm and a cladding thickness of 0.4 mm. The cladding is a steel tube with helical fins and a diameter of 15.25 mm (including the fins). The tops of the core and side blanket fuel assemblies are equipped with special grappler heads for the fuel assembly transfer mechanism. The fuel assemblies have an outlet hole for sodium located in the upper portion on the side surface (Figure 9). Fuel assemblies are reloaded by remote control mechanisms under an inert gas blanket in a closed loop from the reactor to an external storage structure. The transfer mechanism is guided to each fuel
assembly by the aforementioned rotating plugs—one large and one small.

Piping which is 800 mm in diameter delivers secondary sodium from the IHX to the steam generators of the three loops. The length of the piping for the various loops ranges from 40 - 50 m. The BN-600 reactor steam generator is a once-through, sectional-type steam generator in which steam is superheated by sodium (Figure 10). This steam generator contains eight identical sections connected in parallel for sodium and for water-steam. Further, each section contains three modules: an economizer-evaporator, a primary steam superheater and a reheater. All modules of the steam generator are structured like heat exchange devices with a straight-line tube bundle and a lens expansion joint on the housing. The height of the feed water heater-evaporator tube bundle is 15 m, and that of the steam superheater is 12 m. The evaporator tube bundle is fabricated out of 1X2M pearlitic steel and the steam superheater out of 0X18H9 austenitic stainless steel. When austenitic steel is under pressure it is subject to corrosion. Therefore, the plant's heat flow diagram and operating modes are designed to prevent the ingress of moisture into the steam superheater in all possible cases. The evaporator outlet steam temperature should exceed the saturation temperature by at least 20°C. The safety features
of the automatic control system ensure this limitation is met. It is this limitation which greatly influences the start-up and shut-down schedule for the steam generator.

All modules of the steam generator use countercurrent flow of water (or steam) and sodium so that the water or steam flows in the tubes while sodium flows outside the tubes. When secondary sodium enters each section of the steam generator, it is divided into two parts. One part flows to the primary steam superheater and the other to the reheater. The sodium flows upward from the bottom of the superheater, enters the top of the economizer-evaporator and flows downward. Conversely, feed water enters the economizer-evaporator from the bottom, and fresh steam issues from the lower chamber of the superheater. The secondary loop of the BN-600 reactor does not have isolating valves on the main, large-diameter piping. This type of valve is installed on the 300-mm-diameter pipes for supplying sodium to each section of the steam generator and removing it from each section. There are isolating and throttling valves installed on the steam-water pipes, as well as quick-closing valves to dump water and steam from the steam generator and to cut off the steam generator from the tertiary loop in terms of the water and steam supply during an emergency rupture of the heat-transfer pipe and an incursion of water into the sodium.
Figure 10. BN-600 nuclear power plant steam generator section: 1 - evaporator; 2 - steam superheater; 3 - intermediate steam superheater.
Figure 11. Equipment placement in the heat removal loop of the BN-600 reactor (the length of the horizontal piping has been arbitrarily shortened): 1 - IHX; 2 - main coolant pump; 3 - buffer tank; 4 - main and intermediate steam superheaters; 5 - evaporator.

When secondary sodium leaves the steam generator, it travels upwards to a buffer tank. A secondary pump is
located downstream from the buffer tank. The pump is somewhat simpler in design than the primary pump. It has a closed impeller with one-way suction. The impeller is connected to the vertical bracket of the shaft, which rotates on two bearings. The upper is a hydrodynamic oil bearing and the lower is a hydrostatic sodium bearing. An induction motor with a rotor drives the pump in both the secondary and primary loops. Its rotational speed is also maintained within the range of 250-1000 rpm. The secondary loop configuration is complex enough to support natural coolant circulation.

The secondary loop piping has a sharp incline before reaching the buffer tank and pump (Figure 11). The way this particular configuration affects natural sodium circulation will be examined later in the text. Feed water is delivered to each steam generator by three electric feed pumps (PEN), two of which operate while the third is held in standby. If one of the two operating pumps should fail, then the standby pump would switch on.

Live steam from each steam generator is directed to the K-200-130 turbine. After passing through the high-pressure cylinder of the turbine, the steam enters the reheater, and from there returns to the intermediate pressure cylinder. Therefore, each heat removal loop contains a turbine generator unit. Figure 12 shows the plant heat flow
diagram. Because of high steam power cycle parameters, it is possible to use standard engine room equipment at the plant, such as high and low pressure regenerative heaters, condensers, de-aerators, circulating pumps, condensate pumps, feed pumps, and a turbine.

Like the BN-350 reactor control and protection system, the BN-600 reactor control and protection system contains emergency shutdown elements, automatic control elements, temperature and power effect shim elements, and burnup elements.

**Figure 12.** Heat diagram of the third unit of the Beloyarsk Nuclear Power Plant with a BN-600 reactor: 1 - reactor; 2 - primary pump; 3 - IHX; 4 - secondary pump; 5 - steam generator; 6 - buffer tank; 7-9 - high-pressure (HP), intermediate-pressure (IP), and low-pressure (LP) turbine stages, respectively; 10 - condenser; 11 - condensate pumps; 12 - low-pressure regenerative heater; 13 - deaerator; 14 - feed pumps; 15 - high-pressure heaters.
In this case, the number of emergency shutdown rods is equal to five. Their total worth is 4.3 $\beta$. In addition to these rods, an additional emergency shutdown absorber rod has been incorporated. It has a small reactivity worth of 0.4 $\beta$. This rod serves to reduce reactor power when one of the three heat removal loops becomes disconnected.

The two automatic control rods have a total worth of 0.8 $\beta$. With an operating stroke of 800 mm, their maximum speed is 70 mm/s in the control mode. One of the control rods is used to maintain reactor power, and the second is kept as a standby.

In order to compensate for temperature and power effects, as well as fuel burnup, there are 19 regulating and shim rods that have a total reactivity worth of 8.5 $\beta$. The emergency shutdown rods are made out of highly enriched boron carbide. The regulating and shim rods are made of europium oxide. The drive mechanisms and guide tubes of the control and protection system elements are installed on the reactor small rotating plug. The rods of the control and protection elements are housed in the guide tubes. The control and protection system guide tubes are very long. As a result, they are connected by plates at several locations up the tubes, and encased in an overall shell with multibarrier heat containment to avoid vibration and movement, and to provide protection from the flow of hot
sodium. This guide tube assembly is located within a cylindrical shell with an ellipsoidal bottom and is part of the control and protection system. It is called the central rotating column. Twenty thermocouples also pass through this column. They are used to monitor sodium temperature as it exits the core fuel assemblies. In addition to these thermocouples, the reactor is equipped with thermocouples in two groups of four to monitor the median mixed temperature of sodium at the reactor outlet. These are the so-called tank thermocouples. This type of thermocouple is located in the region where sodium from the core overflows into the IHXs. Thermocouples are also mounted at the inlet and outlet of each IHX.

35 The sodium flow rate through the reactor is measured with a flowmeter mounted at the core bypass. This 40-mm pipe joins the reactor pressure header with its outlet mixing chamber. It has been shown that during all operating modes the change in flow rate of sodium through the bypass pipe is virtually proportional to the flow rate of sodium through the reactor. The bypass flow meter signal is used as part of the reactor protection system.

36 The sodium level in the primary loop is measured with electromagnetic level gauges in the reactor vessels and circulating pumps.

37 The BN-600 plant automatically maintains reactor power
(the neutron flow through the ionization chamber), primary and secondary pump rotational speed, sodium temperature at the steam generator outlet, and steam pressure in the steam generator. Reactor power is maintained with a traditional neutron control rod, as is the case in the BN-350 reactor. Sodium temperature at the steam generator outlet is maintained at a prescribed level by the feed water flow rate through the steam generator. In order to ensure high-quality performance when the flow rate of feed water is reduced, the flow is transferred from the main feed pipe by first bypassing it to a pipe of smaller diameter, and then to another pipe of even smaller diameter. Steam pressure in the steam generator is maintained by the turbine control valves. The reason why this occurs is because the plant operates at base load, and the speed at which the turbines rotate is prescribed by the frequency at which the current changes in the line power.
7. Operation of the Monitoring, Control, Emergency Alarm, and Protection Systems During Transients

Because of the fast breeder reactor features described above, an analysis of fast breeder reactor transient operating modes must consider the dynamic quality of the monitoring, control, emergency alarm, and protection systems. This means that it is necessary to account precisely for changes in system parameters over time and for delays in control signal generation, since these values have a marked impact on deviations from the standard values of reactor performance and nuclear power plant performance as a whole.

Figure 23. Change in temperature of an element as the temperature of the coolant washing over it increases in a linear fashion.
The reactor parameters are measured by sensors with varying degrees of lag time. It is usually the temperature sensors, such as the thermocouples and thermometer resistors, which have the greatest lag time. In the liquid metal loops it is an accepted practice to improve reliability by placing thermocouples that already have their own sheath, into an additional process sheath, which is a part of the loop. As a result, an effective thermocouple time constant achieves high values. For example, in the BN-350 reactor vessel the thermocouple time constant is 8 - 10 seconds, and in the primary loop of the BN-600 reactor it is 20 - 25 seconds. At the same time, thermocouples with short lag times are now being installed in reactors. However, there is not enough operating experience with these thermocouples, nor information about their reliability, to use them in the reactor protection system. Most often a thermocouple can be considered to be a single-capacitance link. Therefore, when there is a change in the temperature of the coolant washing over the thermocouple, linear law indicates that a change in the thermocouple output signal time is described by equation \(88\), where \(\tau_0\) in this case is the time constant of the thermocouple. Figure 23 illustrates this equation, which shows that the delay in the thermocouple signal relative to the coolant temperature measured by it achieves \(\tau_0\) in time. As a result, if the
aforementioned thermocouples within the process sheathes are used to generate emergency signals, a high level of performance will not be achieved in the temperature channel of the reactor protection system. In order to improve the speed with which this channel responds, it is necessary to switch to sensors with a shorter time lag, or use electronic devices to correct for the thermocouple readings. The latter method promises to be beneficial no matter which sensor is used. Thermocouple signals must be processed by some electronic device, since it is not an acceptable option for contemporary fast reactors to have an emergency signal issued by the contacts of a secondary device, as was done in the past, due to lack of reliability, accuracy, and speed. At a minimum, one of the jobs of this device should be comparing the thermocouple readings with a given threshold value and generating a signal to stop the reactor when the value is exceeded. By slightly broadening these functions, a correction can be made in the measured temperature for the lag time of the sensor by using an equation resulting from 79:

\[ t_{\text{act}} = t_{\text{meas}} + \tau_c \cdot dt_{\text{meas}}/d\tau, \]  

(228)

where \( t_{\text{meas}} \) and \( t_{\text{act}} \) are the measured and actual values of the temperature, respectively.

*TN: Equation 79 is \( \frac{d\tau}{d\tau} = \frac{\theta - \tau}{\tau_0} + an(\tau) \)
However, most often equation 228 should not be used directly to adjust the thermocouple readings, especially if the adjusted value for the temperature is further used to generate the reactor protection system signal. Signals of thermocouples located behind the heated areas usually fluctuate severely, or "make noise."

These fluctuations are caused by random deviations in the heat and mass transfer parameters in the indicated areas. Figure 45 plots the sodium temperature distribution density in the BN-600 reactor vessel. The temperature value is plotted along the X-axis, and along the Y-axis is the probability that the temperature will fall within a specific temperature interval, referenced to the width of the temperature interval, given there is a sufficiently long enough period of time to record the temperature. The graph shows that the thermocouple signal "makes noise" within the range $\pm 8.5^\circ C$. This noise can increase substantially after the thermocouple readings have been adjusted according to equation 228. Moreover, these adjustments also increase the electrical interference penetrating into the thermocouple signals. Therefore, in equation 228 the correction $\tau_0 \frac{dt_{meas}}{dt}$ is passed through a filter which limits the enumerated random additions:
\[ t_{\text{act}} = t_{\text{meas}} + \frac{\tau_0}{\tau_f} e^{-\tau_f/\tau_f} \int_0^\tau \frac{dt_{\text{meas}}}{d\tau} e^{\tau_f/\tau_f} d\tau \] (229)

Of course, the filter time constant, \( \tau_f \), should be less than the sensor time constant. Consequently, the maximum time delay that can result in the temperature measuring circuit is reduced to \( \tau_f \). The response time of the systems using this temperature is increased accordingly.

Figure 45. Sodium temperature distribution density in the BN-600 reactor vessel.

The coolant flow rate in the liquid metal loops is measured with electromagnetic flowmeters. Even when pipe diameters are between 500 - 800 mm, the lag time of these flowmeters is negligible, and is estimated not to exceed 0.1 - 0.2 seconds. This is significantly less than the
typical amount of time it takes to change the coolant flow rate in the fast reactor loops. Therefore, the actual error in the readings of the flowmeters themselves is insignificant. However, the flowmeter signals can also fluctuate heavily due to disturbances in the coolant velocity field by various obstacles in the flow. The noise band can consist of a few percent to a dozen or more percent of the flow rate median value. Loud signal noises make it difficult to use flowmeters in the reactor automatic control and protection system. Filtration is an effective way to reduce these fluctuations, and it works for the temperature sensors as well. It is important to have enough filtering to stabilize the output signal, but not create an unacceptable delay in the operation of the coolant flow rate protection system, for example. In order to clarify the circumstances in each specific case, special studies are needed. In the majority of cases the electromagnetic flowmeter signal noises are of relatively high frequency. In order to reduce the percentage of these noises in the output signal to the required value, filters with a time constant of 0.5 - 1 second are usually sufficient.

Flowmeters on a small diameter bypass line (40 - 50 meters) connected in parallel to the core are used in the primary loop to measure the flow rate of sodium in reactors with a primary pool-type design. In the broad
picture, the flow rate of sodium through this pipe is almost exactly proportional to the flow rate of sodium through the core, and the reactor as a whole.

\[ \tau_N = \frac{a_i}{\Delta p_{oi} + \Delta p_o} \ll b \]  

It is often asked whether such a measuring system has a large lag time over long distances of bypass piping (15 - 20 meters). In answer to this question, it is possible to make use of the results obtained in Chapter 5.* When the sodium flow rate suddenly drops to zero through the reactor, the drop in the sodium flow rate through the bypass conforms to the shape of a hyperbola. The hyperbola time constant, i.e., the amount of time it takes to reduce the flow through the bypass by a factor of two, is defined by equation 180, and is 0.1 - 0.2 seconds. A momentary decrease in the flow of sodium through the reactor is impossible. However, this example shows how long it takes for the flow through the bypass to respond to a change in the flow through the reactor. As has been shown, this time lag is slight.

Reactor power is measured by using an ionization chamber. The power signal is proportional to neutron flow at the location where the chamber is installed. Therefore,

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*TN: Chapter 5 has not been translated.
it is also proportional to the main component of reactor power, which is determined by the fissioning of fuel nuclei at a specific moment in time. The second power component changes slowly during the process of reactor operation and achieves a range of 6.5 - 7% at full power. This component is linked to the decay of fission products and is not registered by the compensated ionization chamber. This must be taken into account in certain situations. The lag time of the measuring circuit, which includes the ionization chamber, is insignificant. However, intense interference can occur in this circuit, which is designed for small current changes ($10^{-6} - 10^{-4}$ A). This interference is usually of the high frequency variety. Thus, a filter with a time constant of 0.1 - 0.2 seconds is sufficient to radically suppress this interference, in most cases. Therefore, the dynamic error when measuring reactor power is insignificant.

The reactor period is also measured with an ionization chamber. The signal is passed through a logarithm device and is then differentiated. As a result a value is obtained which is inversely proportional to the period:

$$\frac{d\ln (n)}{d\tau} = \frac{1}{n} \frac{dn}{d\tau} = \frac{1}{\tau_R}$$

However, in this form the computational results cannot be used in the reactor protection system. This type of
algorithm may cause significantly inaccurate deviations of the reactor period as a result of interferences in the measuring circuit. At low power levels, the errors result from statistical fluctuations of the neutron flow, and consequently, the ionization chamber current. Filtration eliminates this hazard. For the reactor period meter circuit, usually dual filtration is used, i.e., a circuit consisting of two sequential single capacitance filters. The reactor period protection system channel is mainly designed to be used at low power levels, when fluctuations of the neutron flux are large as a result of the statistical nature of fission. Fairly large filter time constants are selected, i.e., about 10 seconds. Therefore, during a reactor runaway the signal at the period meter output reflects the actual period with a significant time lag.

\[
\frac{n}{n_0} = \frac{\beta}{\beta - \rho} e^{\frac{\frac{\rho_{\lambda}}{\bar{\beta}} - \rho}{1 - \rho}} = e^{\frac{n_{\lambda}}{\bar{\beta}} \cdot \frac{\rho_{\lambda}}{\bar{\beta}} - \rho}
\]

Let us examine the change in the output signal of the aforementioned period meter after various jumps in reactivity. For simplicity's sake when describing reactor power versus time after reactivity jumps, equation 5 will be used. After taking the logarithm and calculating the
differential of this function, the result is passed twice through a single capacitance filter with a time constant, $\tau_r$. As a result, a figure inverse to the measured period is obtained:

$$\frac{1}{\tau_r} = e^{-\frac{\tau}{\tau_r}} \frac{\tau}{\tau_r} \ln \frac{\beta}{\beta - \rho} + \frac{1}{\tau_R} \left[ 1 - e^{-\frac{\tau}{\tau_R}} \left( 1 - \frac{\tau}{\tau_R} \right) \right]$$  \hspace{1cm} (231)

In the first few moments of time, the period markedly differs from the actual reactor period, $\tau_R = (\beta - \rho)/\rho \lambda$. As an illustration, the calculation will be performed with $\tau_r = 10$ seconds. Suppose that the reactor protection signal is issued when the measured period is reduced to 20 seconds. As a result of the calculation, the time lag from the moment reactivity is inserted until the emergency signal occurs was determined. This time lag is represented in Figure 46. If $\rho = 0.28 \beta$, which in the approximate calculation corresponds to an established reactor period of 20 seconds, the time lag tends toward an infinitely large value. It must be said that when short lag time filters ($\tau_r < 7.5$ seconds) are used in the period meter circuit, a prompt reactivity insertion causes the emergency signal for the reactor period to be issued in the situation under study, even with established periods of more than 20 seconds, as a
result of the initial power jump up to $\beta(\beta - \rho)$ due to prompt neutrons. This jump, as can be seen from equation 231, also causes the measured period to be shortened. When the reactivity is $0.52\beta$ (reactor period of 5 seconds), the time lag is 6.2 seconds.

Note the reactor power level value when the reactor period emergency signal is generated. The relationship between this power and its initial value is also shown in Figure 46 when the fast-acting reactor protection system signal is issued. This relationship has a minimum value equal to 3.16 when $\rho \approx 0.4\beta$ (the reactor period is 10 seconds). The reactor power protection system is designed to respond when reactor power exceeds 12% of normal power. Based upon the values given, it can be concluded that under these conditions any reactivity jumps at a power level of 4% above rated will cause the reactor period protection system to respond after the reactor power protection system. However, this conclusion is only valid when the filter time constants used in the calculations are equal to ten seconds in the reactor period measuring circuit. When reactor power levels are high, there is no need to have such sluggish filters, and it is possible to switch to faster computational circuits. At high power levels the neutron flux statistical fluctuations are insignificant. As was already stated, the interference which can occur in the
ionization chamber current measuring circuit does not require low level frequency filters to suppress it. In addition, random period fluctuations associated with reactor input parameter noise during steady state operations are far from dangerous values. Figure 47 shows the time change in the value inverse to the period for the BN-350 reactor in the steady state mode of operation. The minimum reactor period in this case is 400 seconds.

In order to generate the protection system signals, process parameters are also used, such as the coolant level in the reactor, speed of the primary circulating pumps, and pump motor electrical supply voltage. The dynamic characteristics of the level meters and the tachometers (for measuring pump level and speed) are more than sufficient to ensure that the reactor protection system responds quickly enough. In order to measure the electrical voltage of the circulating pump motors during a power supply failure, it is necessary to account for the voltage created by the motors of the coasting pumps and other equipment connected to the very same power sources. An emergency voltage signal should be issued after a time delay in case power is restored after a short-term power interruption. The voltage created by the motors during coastdown can interfere with correctly determining the power interruption.
Figure 46. Fast-acting reactor protection system period signal delay (1) versus reactor power (2) when this signal is issued from inserted reactivity.

Figure 47. Change in value inverse to the period, over time, in the reactor steady state operating mode.
After the alarm signal parameter actually deviates to a set value, not only does sensor sluggishness contribute to a time lag in the fast-acting protection system, but also the ultimate amount of time it takes for the logic units to respond and the delay in releasing the emergency trip rod electromagnets. It usually takes no more than 0.1 - 0.15 seconds for the logic units to function. The delay for the emergency trip absorber rod electromagnets to be released may be from 0.2 - 0.3 seconds because of transient electrical processes in the electromagnetic supply circuit and the residual magnetization. All these additional components in the time lag of the fast-acting protection system must be considered when analyzing transients. These components have a marked effect on temperature deviations in the reactor core in the most serious emergency situations.

When the protection system rod electromagnets are finally released, the rods begin to be inserted into the core. The trip rods are inserted into the core by gravitational pull and the accelerating and absorbing springs. In all, it takes about 1 second for the rods to be inserted into the core as they move through the sodium. Figure 48 shows movement of the trip rods as a function of time when they drop into the core from various heights.
Figure 48. Change in protection system rod positioning when the rod is dropped into the core from various heights.

If the reactor parameter deviations are small and do not cause the protection system to function, the prescribed operating mode is sustained by the automatic control system. At a nuclear power plant with fast reactors, the automatic control rods usually control the following: reactor power level, coolant temperature at the reactor outlet, sodium and steam temperature at the single-pass steam generator outlet, the water level in the steam generator with natural circulation, steam pressure before the turbine or in the steam generator, feed water pressure, and speed of the turbine and primary and secondary circulating pumps.
Section 2

SYSTEM AND EQUIPMENT MALFUNCTIONS DURING ACCIDENTS
IN FAST REACTOR NUCLEAR POWER PLANTS

8. Accident Conditions Considered in the Design; Goals and Capabilities of Reactor Emergency Protection Systems

A list of possible reactor system failures and equipment malfunctions that should be identified in the design are compiled using the criteria for nuclear reactor safety described in the Introduction. The list is formulated by doing engineering studies of the heat-removal system, power supply system, control and safety system, and refueling system. These studies will include reports on the failure rates of each one of the individual system components. Generally, the anticipated accident conditions include the following: leaks in the reactor vessel; ruptures of the piping and loop equipment; a rupture of the neutron guide tubes; a shutdown of the heat removal loop (primary and secondary pumps); an IHX leak; failures and mispositioning of the control rods and shutdown rods; reactor fuel assembly meltdown causing melting and damage to an adjacent row of fuel assemblies; the penetration of moderator into the core; passage of gas bubbles through the core; steam generator coolant pipe rupture accompanied by water surging out into the sodium; fuel assembly drop into a "critical" reactor during refueling; feed pump shut off and other disruptions of feedwater delivery to the steam
generator; turbine shut off; turbogenerator cut off from the power system; a massive power failure, etc.

The most hazardous failures of the safety system and system components involved in accident clean-up are used when these situations are analyzed in the design. The design frequently examines accident conditions using stricter assumptions relative to safety system failures, than regulatory documents dictate. Accident conditions cannot be analyzed without considering transient conditions of the plant. If necessary some of the estimates can be tested by experimental models and bench tests. The most common transient conditions will be studied in this chapter in order to illustrate research findings.

One of the most important reasons for studying accident conditions and transient conditions in reactors is to determine which parameters of the reactor protection system are essential for safety control. All channels of this system are studied separately and as a total unit.

Regulatory documentation indicates that a safety system is a system designed to prevent or limit damage to the fuel, the fuel cladding tubes, and the primary components, and to avert nuclear accidents (i.e., accidents which cause damage to the fuel rods or expose operating personnel to potentially harmful doses of radiation).

When the reactor is being designed, in a practical
sense even stricter limitations are imposed on reactor protection system performance. In all possible accident conditions this system should provide a warning when the temperature of the fuel, the fuel cladding tubes, and coolant in the reactor core exceeds safety limits. Natural limiters are used for fuel and coolant temperatures. For example, the fuel should not become molten in any of the fuel rods, and the coolant should not begin to boil in any of the reactor channels. As the temperature rises towards 1000°C, sodium may begin to boil as it leaves the fast reactor core. Such a temperature increase would cause a massive fuel rod rupture. In addition, when sodium boils in the core this is more often than not destabilizing and is accompanied by large pressure fluctuations that can accelerate damage to the fuel rods. It is less apparent that fuel melting should be prevented at all costs, even if only limited quantities of fuel in the central portion of the fuel rod cores of some of the most heavily stressed fuel rods melt. Some researchers feel that such a localized fuel meltdown would not be dangerous. However, during the design phase, the indicated fuel temperature limitation is always used.

The fuel rod design engineer establishes temperature limitations to be used during an accident for the fuel cladding tubes. It was determined that for fast reactors
the fuel cladding tube temperature could rise to 800°C for up to 10 seconds. This temperature limitation is roughly the same as the temperature load limit of the fuel rod, although the figures are not completely identical. Recently, studies of accident conditions have focused on the tensile strength of fuel rods. Transient stress and bowing of the cladding tubes caused by changing pressure of the gaseous fission products and mechanical interaction of the fuel rod core with the cladding tube were computed as the tubes expand with the changing temperatures. The values of these parameters will be used to conclude how well the accident safety channel operates with respect to the accident condition under study. When cladding deformation results from the transient loads mentioned above, limiting deformation to the elastic field is now advocated, as well as preventing an instantaneous plastic deformation in the cladding. However, there is some doubt that even this somewhat flexible limitation is necessary. The reason deviations in the fuel rod parameters during an accident are allowed is because the fuel rods have high tolerance levels to withstand damage.

Before analyzing specific accident conditions, the reactor protection system channel limits are usually established during disturbances of all the reactor input parameters. The reactor protection system channels
monitoring reactor power and the reactor time constant are activated during reactivity disturbances. The various types of possible disturbances due to reactivity effects are conventionally divided into two types: momentary and slow. A disturbance is considered momentary if the fuel rods undergo only a minor temperature change caused by a corresponding power deviation during the disturbance. All other disturbances are referred to as slow. To all intents and purposes, this means that the time it takes for reactivity to be inserted during a momentary disturbance is much less than the fuel rod time constant. An example of such disturbances would be a situation in which the coolant transports a small amount of a moderator substance or gas bubbles through the core.

An example of a slow disturbance is the movement of the reactor control devices due to a false alarm. While the main indicator of a momentary disturbance is its magnitude, for a slow disturbance such a fundamental indicator is the reactivity insertion speed. The magnitude of a slow disturbance usually does not affect the maximum deviation in reactor parameters during a transient when the reactor protection system is activated. Of course, the strength of the disturbance coupled with the power effect of reactivity should be less than the worth of the reactor protection system elements. The speeds at which reactor power and
temperature change in the core during large reactivity disturbances are high—tens of degrees per second. Therefore disturbance limits essentially depend on the delay in the fast-acting reactor protection system response.

Acceptable disturbances for the power safety channel increase as the reactor power level at which they occur decreases, until 10% is reached, and then they begin to drop. Disturbances that are dissipated by the time constant safety channel exceed acceptable disturbance limits for the power safety channel at low power levels. Transients are computed to determine the magnitude of the difference. Figure 49 depicts a sample of the computations for BN-350 reactor transients. Combining all similar computations makes it possible to determine the relationship between reactivity disturbance limits and various parameters such as reactor power, operating delay of the fast-acting reactor protection system, and the sluggishness of the time constant measuring channel. Figure 50 depicts the permissible levels of momentary disturbances and the rate of slow reactivity disturbances for the BN-350 reactor versus the total response delay of the fast-acting reactor power protection system. The temperature of the fuel cladding tubes is taken to formulate the criteria for disturbance limits. The reactor power output is at rated capacity. The graph shows that during the actual delays, an analysis of which was
completed in the preceding paragraph, the momentary disturbance limit is $0.2 \times 10^{-2} \text{s}^{-1}$, and the slow disturbance limit is $0.4 \times 10^{-2} \text{s}^{-1}$. Other fast reactors also have similar reactivity disturbance limits.

Figure 49. A transient in the BN-350 reactor when reactivity is inserted at a rate of $0.5 \times 10^{-2} \text{s}^{-1}$, and the fast-acting reactor protection system provides a delayed response at the power increase signal: $a$ - temperature in the center of the fuel rods; $\delta$ - temperature of the fuel cladding tubes; $\sigma$ - stress in the fuel cladding tubes relative to initial level; $\cdots$ - 0.5 second delay; $\cdots\cdots\cdots$ - 0.8 second delay; $\cdots\cdots\cdots\cdots$ - 1 second delay.
Figure 50. Acceptable momentary disturbance values (1) and reactivity insertion rate (2) versus the delay in the fast-acting reactor protection system response at the power increase signal for the BN-350 reactor.

The calculations described above are verified by experiments during reactor start-up. These experiments measure the actual fast-acting reactor protection system delay for various channels and therefore suggest feasible reactor protection systems. Figure 51 depicts changes in the parameters of the BN-600 reactor during testing of the reactor time constant protection channel. The experiment was conducted with the minimum controllable power level. A control rod was used to cause reactor runaway. The time
constant of the period meter was somewhat less than expected.

Figure 51. A change in the parameters of the BN-600 reactor during testing of the reactor time constant protection system: 1 - reactor power related to initial level; 2 - time interval for reactor power to double; 3 - calculated time for power to double at the measuring device outlet.

In addition to the operating modes which have already been described, operating modes in which the reactor safety system does not function will also be studied. It is possible to determine the maximum rate at which reactivity can be inserted so that a runaway of a prompt critical reactor does not result, even if the fast-acting reactor protection system does not function. This means that even at such a rate the negative reactivity feedback will still be able to counteract a sufficient portion of the
disturbances that have already begun. The maximum rate at which reactivity can be inserted is an important parameter of the reactor, because it allows basic estimates of safety limits to be made in the case of hypothetical accidents. For example, even with a rated output of 1%, the maximum rate at which reactivity can be inserted is $0.75 \times 10^{-2} \text{ s}^{-1}$ for the BN-350 reactor.

12 Even in the event of an accident, the maximum rate at which reactivity can be inserted is far below safety limitations. For instance, if a control rod is mispositioned in the BN-350 and BN-600 reactors, the reactivity insertion rate is $(0.2 - 0.3) \times 10^{-3} \text{ s}^{-1}$. This rate is more than one order of magnitude less than safety limits.

13 The channel of the reactor protection system that monitors coolant flow is the most important channel. In the core the rates at which the temperature rises during the most dangerous failures of sodium circulation through the reactor have reached $100^\circ \text{C/s}$. Moreover, operating experience indicates that accidents involving various coolant circulation failures in the loops are the most common type of accidents. A drop in the rate of coolant flow through the reactor is the most dangerous when all the primary pumps shut down at the same time. A shutdown such as this may be due to a power supply failure from the pump motors. It is for just such a contingency that the protection system
channels, which function during a loss of coolant flow through the reactor, have been designed. In such situations the alarms of the fast-acting reactor protection system in existing nuclear plants are activated by:

- A loss in the total coolant flow-rate in the primary pumps;
- An increase in reactor power versus coolant flow through the reactor;
- An increase in coolant flow in normally operating primary pumps;
- A reduction in the speed of the primary pumps;
- A drop in electrical voltage of the primary pump motors; and
- A shutdown of a prescribed number of primary pumps.

The reactor process flow diagram and control system are taken into account when a specific list of alarm signals is being selected. This ensures that the reactor protection system operates reliably and reduces the possibility of false activation.

The maximum-allowable delay for protection to be activated is determined by calculating the transient during a simultaneous shutdown of all the primary pumps while the fast-acting reactor protection system is triggered. This calculation has been done for the BN-600 reactor, and the findings are presented in Figure 52. These findings indicate that the maximum-allowable time interval between pump shutdown and initial movement of the safety absorber rods is 1.6 – 1.7 seconds. If the time for the sodium flow rate to drop to the reactor protection system threshold level is
subtracted from this time interval, then 0.5 – 0.6 seconds is an acceptable delay for the logic units and electromagnets of the accident shutdown rods to emit an alarm signal. It may take from 0.2 – 0.3 seconds for the electromagnets to disengage. Therefore it can be concluded that all other components in the alarm circuit should operate within 0.2 – 0.4 seconds. Standard equipment is used to guarantee such a quick response.
Figure 52. Shut down of three main coolant pumps of the BN-600 reactor primary loop, followed by the activation of the fast-acting reactor protection system, which is actuated by the signal that monitors an increase in power by 20% versus sodium flow rate: 1 - reactor power; 2 - sodium flow-rate through the reactor; 3 - power versus flow-rate; 4 - maximum temperature limit of the fuel cladding tubes; 5 - maximum temperature limit of the fuel; 6 - sodium temperature at the fuel assembly outlet; 7 - reactor vessel sodium temperature; - - - - one second response delay of the fast-acting reactor protection system; - - - - 1.5 second response delay; - - - - - - 1.8 second response delay.

In a nuclear power plant operational safety feasibility study, it is not enough to anticipate only the most dangerous power supply failures to the recirculation pumps, but any sufficiently likely failures. By focusing on the whole spectrum of accident situations and their expected frequencies, it is possible to determine the probable load on the fuel rods and mechanical equipment of the plant during an accident.
In order to avoid unnecessary reactor shutdowns, the alarm signal monitoring voltage dips in those sections of the switchgear from which the pump motors are fed is equipped with a time delay. This delay prevents the signal of the fast-acting reactor protection system from being tripped during brief voltage dips when the supply is switched over to the reserve sections. On the one hand, the indicated delay should be long enough to ensure that the power switch-over has enough time to take place during the most likely interruptions of the power supply. On the other hand, the indicated delay should be within safety limits for temperature deviations in the core. An examination of the transient makes it clear when it is necessary to shut down the reactor in each specific case. Figure 53 plots transient curves in the primary loop of the BN-350 reactor when a short circuit occurs in sections of the switching gear and the electric differential protection of the busbars is activated. In this case the sodium flow rate drops only slightly through the reactor, which makes it possible to avoid a reactor shutdown.
Figure 53. Calculated change in the primary parameters of the BN-350 reactor when a short circuit occurs in sections of the switching gear and the electric differential protection of the busbars is activated: $a$ - sodium flow through the reactor; $b$ - sodium flow rates in individual loops; $---$ - flowmeter reading in one of the loops; $c$ - pump rotational speed.

As has already been stated, the channel of the reactor protection system that monitors the sodium temperature at reactor outlet is often very sluggish. This sluggishness is
caused by the lag of the temperature sensors and by the tank volumes in which these sensors are placed, and sometimes this sluggishness is caused by the transport coolant. The reactor protection system channel that monitors the temperature must make adjustments in the relevant components based on the transients, otherwise this channel will not be very effective. Figure 54 illustrates the permissible rate of power increases for a reactor of the BN-600 type versus the time constant of the temperature sensor placed in the reactor vessel. A permissible rate is considered to be one which does not lead to a temperature increase above 800°C in the core fuel cladding tubes when the fast-acting reactor protection system is activated by the temperature alarm signal. If the sensor time constant is 20 seconds, the rate at which the power can increase safely is 0.3%/s. If a correction is made for the sluggishness of the sensor within the reactor vessel, the rate at which the power can increase safely is 1.5 - 2%/s.

There are some accidents whose dangers can only be avoided by ensuring that the fast-acting reactor protection system is activated by the outlet temperature. These types of accidents include, for example, one in which the coolant temperature rises on entering the reactor because feedwater delivery to the steam generator has been stopped. This also includes accidents involving distorted readings of the
ionization chamber monitors. The temperature channel will respond much more quickly if the relevant adjustments are made, especially during such disturbances.

Figure 54. Permissible rate of reactor power increases versus the time constant of the temperature sensor in the reactor protection channel, which is shutting down the reactor.

Of course, when the fast-acting reactor protection system is activated, steps are taken to decrease the thermal stresses in the components of the reactor structure. This is done by reducing the rate at which the coolant temperature is lowered by converting the primary pumps to a reduced speed at the signal from the fast-acting reactor protection system. The curves in Figure 55 show thermal stress changes in the IHX casing and the outlet piping of the BN-350 reactor in two situations: after the fast-acting
reactor protection system is triggered and the primary pumps are operating at normal speed, and when the pumps have been switched to one-quarter speed. When the pumps are switched to a reduced speed, the stress is reduced by about a factor of two.

Figure 55. A change in the BN-350 reactor parameters with the fast-acting reactor protection system in the activated mode and without converting the main coolant pumps to a reduced speed (Curve 1; with the main coolant pumps shut
down in order to reduce their speeds within two seconds after the fast-acting reactor protection system signal is given (Curve 2); and at the exact same time as the signal is given (Curve 3): $a$ - reactor power and sodium flow rate through the reactor; $\dot{\theta}$ - sodium temperature at the reactor vessel outlet ($\dot{\theta}$) and average sodium temperature at the fuel assembly outlet (TBC); $\theta$ - thermal stress in the IHX and the outlet piping.
9. Reactor Operations in the Self-Regulation Mode

Permissible changes in reactor parameters in the self-regulation mode. Typically, a fast power reactor is stable during operation and is easy to control in the self-regulation mode. This feature is based upon the fact that the power density field in the core is highly stable, and the power and temperature effects of reactivity in this type of reactor have no positive components. It is only possible to have positive components in small-volume experimental reactors. Another factor which plays an important role is the fact that the response times are very quick for all feedback components of reactivity. The amount of time it takes for them to respond to changes in reactor power, coolant flow rate or input temperature is measured in seconds, and only some relatively small components are measured in tens of seconds. Under these conditions, the reactor can sustain large disturbances for all the input parameters, including reactivity, coolant flow rate, and coolant input temperature, without dangerous temperature deviations in the core. Based on the equations cited in Chapter 4,* it is possible to identify which disturbances will cause the sodium to boil, the fuel in the core to melt down, or the fuel cladding temperature to rise to maximum

*TN: Chapter 4 has not been translated.
permissible limits. These disturbances will be evaluated and compared with possible values for design basis accidents.

2 When disturbances in reactivity occur, deviations in reactor power and core temperatures are proportional to the disturbance-to-power-effect ratio. For the BN-350 and BN-600 reactors, a disturbance which would lead to a fuel meltdown in the most stressed fuel rods is equal to $(0.19 - 0.3) \times 10^{-2}$. One which would cause the peak temperature of the fuel rod cladding to rise to $800^\circ C$ is equal to $(0.15 - 0.18) \times 10^{-2}$, and the sodium in the central channels of the core to start boiling — $(0.6 - 0.8) \times 10^{-2}$. It is apparent that these disturbances are large. For the sake of comparison, it is possible to say that when a centrally-located control rod is withdrawn from the core, this leads to a disturbance of $(0.1 - 0.14) \times 10^{-2}$. The most unfavorable shift of molten fuel within one fuel assembly causes a disturbance of $0.1 \times 10^{-2}$.

3 When the sodium flow rate through the reactor drops, reactor power is reduced as a result of core heating when the reactor is in the self-regulation mode. A power reduction significantly slows down temperature increases in the core. Figure 56 plots sodium heating established in the self-regulation mode in a BN-600 reactor that was initially operated at nominal power and coolant flow rate, versus
final coolant flow rate. For comparison's sake, heating versus flow rate has been plotted at constant power on the same graph. The calculations show that in the self-regulation mode, sodium will boil in the reactor core when the coolant flow rate is reduced to 20%, and when the power level is constant—to 44%. It must again be emphasized that these results pertain to the final steady state. During a transient, when the flow rate is sharply reduced, temperatures will exceed the established values.

Figure 56. BN-600 reactor power (curve a) and sodium heating in the reactor (curve b) versus sodium flow rate through the reactor when power and flow rate are reduced in the self-regulation mode: 1 - when the hydrodynamic effects of reactivity are absent; 2 - when the value of these effects equals $5 \times 10^{-4}$; 3 - when constant power is automatically maintained.
When the coolant temperature changes at the reactor inlet, temperature deviations in the core will also greatly depend upon the rate of the disturbance while the reactor is in the self-regulation mode. It was already stated in Chapter 4 that a slow increase in coolant temperature at the input to the BN-350 and BN-600 reactors means that the peak temperature values for fuel, coolant, and fuel cladding in the cores are reduced or remain almost unchanged. However, if the input temperature increases quickly enough, core temperatures may initially rise during a transient. Therefore, if the inlet temperature rises by 200°C and the time constant is 20 seconds, exponential law says there will be a dynamic coolant temperature rise at the BN-350 reactor outlet by 70°C over the rated value. The temperature will rise by 50°C with a time constant of 50 seconds. In existing fast reactors, the most effective IHX time constants are 20 - 30 seconds. These time constants determine the maximum rate at which the temperature rises, if, for example, there is a loss of sodium circulation in the secondary loop.

The primary main coolant pump shuts down, but the reactor protection system does not function. Until recently, a primary main coolant pump disconnect in conjunction with a protection system failure in several foreign reactor designs was looked upon as a design basis.
accident. Calculations showed that such an accident rapidly caused the coolant to boil and be ejected from the core, which then lead to a runaway reactor as a result of the positive sodium void effect of reactivity. In the majority of countries today, this type of accident is categorized as a hypothetical accident, since it will only occur if an improbably large amount of equipment and the reactor protection systems fail. In addition, this raises the question whether it is possible to totally eliminate coolant boiling in this operating mode by choosing appropriate reactor physical and dynamic properties. In order to do this, it is necessary to reduce reactor power at a fast enough rate after the pumps have failed by taking advantage of negative temperature reactivity effects.

Suppose that when the main coolant pumps shut down and then coast down, natural coolant circulation begins in the primary loop. At the same time, the coolant flow rate through the reactor is reduced almost by a factor of two. Equation 166 obtained in Chapter 4 shows that under these conditions in the steady state, the reactor coolant will heat up to the degree to which the power reactivity effect exceeds the flow effect, independent of the coolant flow rate during natural circulation. Reactivity effect data, which is also given in Chapter 4, shows that after the sodium flow rate falls, sodium will heat up in the BN-350
reactor by a factor of 3.8, and the temperature at the core outlet will rise to 920°C. These calculations were made for an initial reactor power of 650 MW. The sodium boiling temperature is 960°C at the pressure which exists at core outlet.

\[ \Delta \theta / \Delta \theta_0 \approx \frac{K_{M0}}{K_{G0}} \]  (166)

For the situation under consideration, sodium in the core will be heated up by a factor of 2.7 hotter in the BN-600 reactor. If the given inlet temperature does not change, the core sodium outlet temperature should increase to 970°C. This means sodium leaving the core will start boiling. Calculations made with equation 166 show that in order to limit sodium heating in the reactor during a shift from forced to natural coolant circulation in the self-regulation mode, it is necessary to try to increase the absolute value of those reactivity effects components which are associated with coolant and steel temperatures and reduce those which are associated with fuel temperature. Therefore, it is a misconception to think that increasing the negative Doppler effect always has a favorable outcome. Note that all other conditions being equal, this effect increases temperature deviations in situations associated with a reduction in the flow rate. Because of this, temperature deviations in the situation under consideration are less in a reactor with
metallic fuel, where the Doppler effect is relatively weak, than in a reactor with oxide fuel (if \( K_M < 0; K_G < 0 \)).

First of all, the quantitative evaluations made above refer to a quasi-stationary process. Second, they refer to the simplest of cases in which flow rate and power are reduced, but the reactor input coolant temperature does not change and no outside reactivity disturbances occur. A combined change in the given indicators will be examined in order to evaluate their effect on temperature deviations in the core.

As is shown in Chapter 4, the reactivity balance equation for the final steady state can be written as follows:

\[
K_M(n - 1) + K_{G0} \delta G + k_i (T - T_o) + \rho_B = 0. \tag{232}
\]

In the process of writing this equation, it was assumed that when reactor power and flow rate change, the coolant input temperature changes from the value \( T_o \) to \( T \), and outside reactivity disturbances equal to \( \rho_B \) are introduced into the reactor. Given the above, the following is obtained:

\[
\theta = \theta_o + \Delta \theta_o \delta G \left( \frac{K_{M0} - K_{G0}}{K_{M0} + K_{G0} \delta G} \right) + \delta T \left[ 1 - \frac{k_i \Delta \theta_o (1 + \delta G)}{K_{M0} + K_{G0} \delta G} \right] - \frac{\rho_B \Delta \theta_o (1 + \delta G)}{K_{M0} + K_{G0} \delta G}
\]

When the reactor changes from standard to natural coolant circulation, and \( \delta G \approx 1 \), then
\[ \theta = \theta_0 + \Delta \theta_0 \frac{K_{M0} - K_{G0}}{K_{G0}} + \delta T \left( 1 - \frac{k_i \Delta \theta_0}{K_{G0}} \right) - \frac{\rho_W \Delta \theta_0}{K_{G0}}. \] (233)

It is apparent that in the majority of cases involving no marked reactivity effect resulting from fuel assembly bowing in a non-uniform temperature field, \( k_i \Delta \theta_0 / K_{G0} > 1 \). This means that a reduction in reactor inlet coolant temperature will lead to an increase in core outlet temperature in the situation being studied.

During an emergency, a change in the reactor inlet temperature largely depends on the way reactor power is removed. Heat can be removed by using a forced-circulation straight-tube steam generator, or a steam generator with natural water circulation and a steam/water mixture. In addition, special emergency cooling air heat exchangers connected to the reactor primary or secondary loop can be used. Using this design, the reactor inlet temperature may drop all the way down to the temperature in the deaerator if the coolant temperature is not well-maintained at the steam generator outlet. In this respect, a steam generator with natural circulation works better. In this case, steam generator outlet coolant temperature drops no lower than the water saturation temperature, which corresponds to the pressure in the steam generator. Of course, the pressure must be maintained automatically. However, during emergency operations, this problem can be solved in a much simpler
way. Instead of maintaining the temperature of sodium at a constant level at the steam generator outlet, the feed water flow rate can be changed by small amounts.

11 With the air heat exchanger design, this problem is solved by choosing the appropriate thermohydraulic performance characteristics of the air path. Therefore, using this design means that reactor inlet coolant temperature is solely determined by reactor power level. In the self-regulation mode, a reduction in the inlet temperature will delay a power drop. Therefore, an uncontrolled reduction in temperature in this case is only possible if mistakes were made about the path's performance characteristics.

12 The results of the assessments made on steady-state show that even if inlet temperature remains unchanged in the BN-350 and BN-600 reactors, a power reduction in the situation being studied is not sufficient to avoid a dangerous temperature increase in the core. It must be considered that temperature deviations during transients should actually be even greater. Also, it is important to account for the energy release resulting from fission fragment decay. The ratios cited are valid until the next one, reactor power, exceeds after-power. When natural circulation is established, it is possible that the neutron component of power will drop to zero, and further
temperature changes will be determined by a drop in after-power. This is all taken into account when calculating transients. Figure 57 shows the results of such calculations for the BN-600 reactor. They show that if the reactor protection system suffers a total failure, sodium will start boiling within 26 seconds at the core outlet. If the pumps fail, there is not enough negative temperature feedback for self-quenching. However, equation 233 shows that the insertion of even a small amount of negative reactivity can significantly improve the situation. When specific values are substituted for the constants in equation 233, the following equations are obtained for a maximum coolant temperature deviation from the initial figure at the core outlet of the BN-350 reactor:

\[ \delta \theta = \theta - \theta_0 = 454 - 28T + 855 \cdot 10^2 \rho_B \]

and the BN-600 reactor:

\[ \delta \theta = 370 - 2.098T + 860 \cdot 10^2 \rho_B \]

These equations show that for each degree the inlet temperature is reduced, the outlet temperature rises by two degrees. From this it is apparent that in this accident case study, if only two absorber control rods with a worth of \(- (0.2 - 0.26) \times 10^{-2} \) are inserted into the core, each one markedly reduces temperature deviations. The transient will
be calculated when an operating control rod is at first located halfway into the core, and a standby control rod is at its upper edge. Therefore, inserting these rods into the core causes a negative reactivity of 0.4 x 10^{-2}. These results are also contained in Figure 57. They show that a maximum sodium temperature in this process is 850°C, i.e., this temperature is less than the boiling point.

13 A simultaneous failure of all reactor protection system rods when the pumps also fail can only be caused by massive core damage and large displacements of the guide tubes and sleeves of the reactor control and protection system elements as a result of seismic effects. In these circumstances, it is less likely there will be a failure of the control rods which are inserted into the core by the electric motors (as opposed to gravity pull) when the fast-acting reactor protection signal is given.

14 The maximum temperature during a transient falls as the flow reduction rate also decreases i.e., as the coastdown constant of the pumps increases. However, if this constant increases, it causes the temperature to drop at a faster rate after the fast-acting reactor protection system functions, which thereby causes an increase in thermal stress in the structural elements.
Figure 57. Change in BN-600 reactor parameters in the self-regulation mode when the primary pumps are disconnected:  

- reactor power (dotted lines) and sodium flow through the reactor (solid lines), relative to nominal values; 
- the temperature effects of reactivity (solid lines), and reactivity inserted by the control rods (dotted lines); 
- sodium temperature at the core outlet; 

1, 2 - complete pump disconnect; 3 - reduction in pump rpms; 1, 3 - no absorber rods are inserted into the core; 2 - control rods inserted into the core.
Because of the way energy is supplied to the pumps in the BN-350 and BN-600 reactors, the more likely situation in which the pumps begin operating at $\frac{1}{4}$ of rated speed will be examined, instead of examining the situation in which the pumps stop after being shut down. The transfer to a reduced speed occurs when the fast-acting reactor protection system signal is given. If the pumps begin operating at a reduced speed and none of the absorber rods are inserted into the core, reductions in reactor power in the self-regulation mode are enough to prevent sodium from boiling. In the self-regulation mode, the maximum temperature reaches 920°C in the BN-600 reactor (Figure 57). Inserting one control rod reduces the maximum temperature to 820°C. Consequently, even in the most unlikely hypothetical situations, an accident will proceed without catastrophic results.

Experiments have been conducted on transients in fast reactors when the pumps are cut off and reactor power changes in the self-regulation mode. Such experiments for the EBR-II and Rapsodie reactors are described in the literature. The primary pumps were shut down in the Rapsodie when the power level was 22 MW. During the transient, the reactor outlet sodium temperature rose from 400°C to 550°C and then began to drop. In the central channel of the core, the temperature rose to 705°C.

A similar experiment was conducted in the BN-350
reactor with a very low initial power level (400 kW). At first, the primary pumps operated at a reduced speed, and then they were disconnected. In order to increase power deviations and thereby enhance measurement accuracy, a positive reactivity disturbance ($\rho = 5 \times 10^{-3}$) was inserted all at once by moving a control rod. In the primary loop, natural circulation began to be established, and reactor power changed in the self-regulation mode. The sodium temperature at the reactor inlet remained constant for a dozen minutes or so. The reactivity balance equation for this operating mode looks as follows:

$$\rho = \rho_n + K_{\text{rr}} \delta G$$  \hspace{1cm} (234)

The initial leg of the process was used for evaluation. During this period the natural circulation flow rate was determined only by the temperature in the reactor, as well as hydraulic and inertial characteristics of the loop. These performance characteristics are well-known, because the results of independent experiments are available. Therefore, only reactor power had to be measured to determine the natural circulation flow rate through the reactor. Next, equation 234 gave the reactivity flow rate coefficient, which was the main goal of the experiment since this coefficient determines acceptable temperature deviations in the operating mode under study. The results of the experiment are shown in Figure 58. In the initial seconds,
deviations in the flow rate coefficient from the established value can be explained by measurement errors.

Figure 58. Change in BN-350 reactor parameters in the self-regulation mode with natural sodium circulation: 1 - reactor power relative to nominal level; 2 - sodium flow rate relative to nominal rate; 3 - sodium heating in the core; 4 - reactivity; 5 - $K_t = 2 \times 10^4 K_\infty / \Delta \theta_c$, where $K_\infty$ is nominal flow effect of reactivity and $\Delta \theta_c$ is the nominal heating of sodium in the core.

A fuel assembly drops into the reactor during reloading. This situation was often examined during fast
reactor safety engineering studies. It was assumed that the fuel assembly would drop into a critical reactor. The subcriticality level of the reactor is closely controlled during reloading. The reactor can only be launched into a critical state accidentally if all the control channels fail. A fuel assembly can only drop into the core, consequently, if there is a malfunction of the refueling equipment. Thus, the situation under study might be the result of many failures overlapping. Therefore, this situation is numbered among the hypothetical ones. The importance of this situation is that emergency shutdown elements are lowered into the core during reloading. After an assembly drops, a power increase is only limited by negative temperature effects of reactivity. Based upon the reactivity balance equation, it is possible to determine the ratio between coolant heating in the reactor, which occurs as a result of the transient after the fuel assembly drops, and its rated value:

\[
\frac{\Delta \theta}{\Delta \theta_0} = \frac{n}{G} = - \frac{B_{\text{ASSEMBLY}}}{K_{G0} + (K_{M0} - K_{G0})G}
\]

(235)

For the majority of reactors, fuel assembly performance is

\[
B_{\text{ASSEMBLY}} \leq |K_{G0}|, \text{ moreover, } |K_{M0}| > |K_{G0}|.
\]

Therefore, the heating limit at any coolant flow rate does not exceed a nominal value. This shows that refueling can be carried out at very
low coolant flow rates sufficient to remove after-power.
10. One Heat Removal Loop of the Unit Disconnects

Heat is removed from a fast reactor along several parallel loops. It is important to have enough loops to ensure that the shutdown of one loop does not force a reactor shutdown. The most likely cause of a loop shutdown is a malfunction of the circulation pumps due to mechanical damage, failures in the pump auxiliary systems and their electric motors, and power outages. Therefore, the systems under discussion are usually arranged so that a failure in any one of them will only take one pump off line.

Figure 59. Single primary pump shutdown mode in the BN-350 reactor: 1 - reactor power; 2 - set power; 3 - sodium flow rate through the reactor; 4 - fuel temperature; 5 - fuel cladding tube temperature; 6 - reactivity; 7 - sodium temperature in the vessel; 8 - sensor temperature in the reactor vessel.
If one primary pump shuts down, the flow of sodium through the reactor will only be slightly reduced. In the BN-350 reactor, the flow rate will be reduced by 11%, and in the BN-600 reactor—by 15%. At rated power, sodium in the core of these reactors is heated up to 200 - 250°C. Therefore, even if the sodium temperature rises together with the temperature of the fuel cladding tubes located at the core outlets, neither should exceed 30 - 40°C, even if the power level remains constant.

Such a temperature increase is not very dangerous. Nonetheless, it was considered best to avoid even these temperature deviations. When the loop shut-off signal is given, the control system transfers the reactor to a reduced power level. In the BN-350 reactor, an automatic controller performs this transfer by changing the set power level (Figure 59). In the BN-600 reactor study, a special absorber rod is initially dropped into the core in order to accelerate the power level reduction, and then the power level is brought to a level which is two-thirds of nominal power by an automatic controller. The rotational speed of the pumps in the two loops that remain operational is reduced in order to reestablish the nominal coolant flow rate in these loops. In the BN-350 reactor, a check valve in the disconnected loop is closed by the coolant flow, and in the BN-600 reactor by the drive servomotor. If the check valve fails in the
disconnected loop, a backwash of coolant will occur. As a result, the flow rate of sodium through the reactor will decrease even more. Also, "cold" sodium from the pressure header will encounter the hot IHX and the reactor vessel, causing thermal stress in their elements. Therefore, if the check valve of the loop fails in conjunction with a loop shutdown, the fast-acting reactor protection system signal will sound, and the reactor will shut down. The reactor circulation pumps will switch to a reduced speed. This will lessen the reverse flow of coolant in the disconnected loop.

Figure 60. Sodium temperature at the IHX outlet and thermal stresses in the heat exchanger elements when one secondary main coolant pump disconnects (solid lines), and when primary and secondary main coolant pumps simultaneously disconnect (dotted lines): 1, 2 - primary and secondary sodium temperatures, respectively, at the IHX outlet; 3 - thermal stresses in secondary outlet piping; 4 - thermal stresses in the IHX housing.
The chief reason why a heat removal loop shuts down can be attributed to a primary or secondary pump failure. The danger associated with major thermal stresses in the IHX and other equipment elements causes the introduction of an interlock which ensures that the primary and secondary pumps in each loop will only shut down in pairs. Otherwise, the rate at which primary and secondary coolant temperatures change at the IHX outlet is too fast. The graph in Figure 60 plots sodium temperature and thermal stress changes in the BN-350 reactor IHX housing when the secondary pump is lost, and when both the primary and secondary pumps are lost at the same time. In the latter case, the rate at which the primary sodium temperature rises at the IHX outlet, and consequently thermal stress in its housing, is on an order of several times lower than in the former case. This illustrates how important the interlocks mentioned above are.

When one heat removal loop is lost, the temperature may be distorted in the reactor vessel. This creates interference in the temperature channel of the controller, if it is turned on. Figure 61 shows sodium temperature changes both in the BN-600 reactor vessel, and at the IHX inlet when loop number five is disconnected. A colder stream of sodium coming from the side shield and entering the IHX of a disconnected loop reduces the temperature of the sodium at the IHX inlet. Therefore, in this instance the temperature begins to fall
immediately after the loop is disconnected. At first, when the sodium enters the IHXs of loops four and six, its temperature begins to rise because the core outlet temperature rises.

![Graph showing sodium temperature changes](image)

**Figure 61.** Sodium temperature changes at the IHX inlet and in the BN-600 reactor vessel when the fifth loop main coolant pump disengages: 1, 2 and 3 - IHX inlet for loops 4, 5 and 6, respectively; 4 - temperature in the vessel.

After one heat removal loop is disconnected, the reactor continues to operate at a reduced power level. If one more loop is lost, the fast-acting reactor protection system will cause the reactor to shut down. When the secondary loop is disconnected in the BN-600 reactor, the check valve does not close. Because of this, it has been asked whether the level of sodium will change in the primary pump that remains operational when one of two, or two of three pumps are lost. The level of sodium drops in the pumps that remain in
operation. This is explained by the fact that the inflow of sodium into the pump tank initially does not change. It is determined by the coolant level differential in the reactor and in the pump. However, the amount of sodium discharged from the tank (pumping capability) increases. A special experiment in which both pumps were disconnected showed that the level to which the sodium dropped in the operational pump was acceptable (Figure 62).

Figure 62. Experimental (solid lines) and calculated (dotted lines) changes in BN-600 reactor primary parameters when the main coolant pumps in loops four and five are lost, and the check valve closes at the outlet of pump four (flow and speeds are referenced to rated values: $a$ - sodium level deviations in the pumps; $\delta$ - pump speeds, bypass flowmeter signal, and sodium flow rate in loop number six.)
11. After-Power Removal from the Reactor during a Power Supply Failure

1. When there is a complete power failure in a nuclear power plant, the main difficulty is removing after-power from the reactor. In such a situation the fast-acting reactor protection system will be tripped and the fission chain reaction in the core will immediately be terminated. As has already been stated, after-power in the reactor is the result of beta and gamma decay of fission products and radiative capture products. The ratio between the "neutron" portion of reactor power and the portion resulting from fragment decay is shown in Figure 21. Only in the first minutes of the process does after-power decrease rapidly. For example, within 1 hour after the fast-acting reactor protection system is tripped, after-power in the BN-600 reactor is approximately 20 MW, and within 10 hours it is 12 MW. This shows that in such situations it is essential for reactor cooling to be reliable and sustainable for a long period of time. After-heat removal from the reactor during a nuclear power plant power failure is called shutdown cooling.

2. When there is a total loss of power, the power supply is interrupted to the motors of the circulating and feedwater pumps. It is for this reason that special equipment and power supplies are provided to activate
coolant circulation through the loops of the plant when a power loss occurs. Furthermore, full use is made of favorable factors such as circulation pump and turbogenerator coastdown, and natural coolant circulation. Pump and turbogenerator coastdown lasts the first 1.5 - 2 minutes of the transient. However, it is precisely at the beginning part of the transient that reactor power is still high, which means that the rate of coolant flow in the loops must be increased. Natural circulation of coolant has a very low head, but nonetheless it is enough for an extended heat removal operation.

Figure 21. The change in after-power in the reactor after it has been shut down by the fast-acting reactor protection system (the numbers by the after-power decay curves indicate the reactor operating time at nominal power before shutdown): —— calculations using the Way-Wigner formula; - - - - updated calculation for the BN-600 reactor;
Because of the isolated layout of the BN-350 reactor, its power supply is sub-divided into 3 categories: a system power supply, an independent power supply, and an emergency power supply. This does not include a reliable power supply from batteries for the most critical needs. The system power supply also includes the turbogenerators of the nuclear and thermal power plants themselves. The independent power supply consists of two 6-MW turbogenerators that are not connected to the main system. Emergency power is supplied by diesel generators, which start up when the plant loses total power. The power from the diesel generators is supplied to units of the independent power supply. The following pumps are used to supply feedwater to the BN-350 steam generators in various operating modes: (1) main feedwater pump (PEN); (2) auxiliary feedwater pumps (APEN); and (3) coolant (emergency feedwater) pumps (NAR). The PEN motors are connected to units of the system power supply, and the APEN and NAR motors are connected to units of the independent power supply. The APENs supply water to the steam generator from the feedwater suction header along piping that is independent of the primary water supply piping. The NARs can supply water to the pressure header from both the suction header and the water supply tanks, or along independent discharge lines to the steam generator. This type of power and water supply set-up allows for safe
Transients occurring when system power is lost have been studied at various power levels. When the power supply is lost, the fast-acting reactor protection system is tripped, and the primary and secondary pumps are converted from 1000 rpm to 250 rpm. A quick drop in the amount of sodium flowing through the steam generator causes a deterioration in the steam quality in the evaporative channels, a reduction in steam generation, and therefore a brief drop in the water level both in the evaporators and in the saturated steam pressures. This occurs because the control system cannot immediately restore the index value. It takes 30 - 40 minutes after the fast-acting reactor protection system is actuated for the temperature to actually level off to a useful degree.

The switchover from forced to natural circulation in the primary and secondary loops has been researched when a total loss of all power occurs, i.e., an interruption of the system and independent power supplies. Figure 63 shows the results of one of the experiments that was conducted on one loop. At the outset, the loop pumps operated at a speed of 250 rpm. Reactor power was lowered to the minimum controllable level while the pumps were shut down. Then a gradual power increase was initiated. This led to a temperature increase and the buildup of natural sodium
circulation in the loops. When the power level reached seven megawatts, the sodium flow rate rose to 3% of nominal in the primary loop and 2% of nominal in the secondary loop. It was calculated that approximately 25 MW is the maximum power that can be removed from the reactor using natural circulation in the five loops without exceeding the rated temperature at core outlet. Experiments were also conducted on the BN-350 reactor. They involved cooling secondary sodium by using natural air convection through the intermodular space of the "Nadezhnost" steam generator without supplying water to it. The experiments confirmed the calculations, which indicate that air cooling can remove up to 3.5 MW of heat from each steam generator.

Figure 63. Results of measuring BN-350 reactor parameters when power increases during natural sodium circulation in one heat removal loop: 1 - sodium temperature at core outlet; 2 - sodium temperature at reactor outlet; 3 - sodium temperature at IHX inlet; 4 - sodium temperature at IHX outlet; 5 - primary sodium flow rate; 6 - reactor power; 7 - secondary sodium flow rate.
In the BN-350 plant, the level of steam pressure in the steam generator must be accurately maintained when forced circulation is replaced with natural circulation in the secondary loop. If control is poorly maintained, a sharp drop in the sodium flow rate through the steam generator will cause a sharp drop in the generator's steaming capacity, and consequently in steam pressure. This in turn causes a reduction in the water saturation temperature in the steam generator with natural circulation of the steam-water mixture in the Field tubes, and consequently, a reduction in the sodium temperature in the evaporator. If the sodium is circulating normally, then it is flowing upwards in the evaporator. Therefore, a sodium temperature drop in the evaporator causes a reduction in the natural circulation delivery head and also in the sodium flow rate. A drop in the flow rate of sodium will again lead to an additional reduction of the steaming capacity, steam pressure, water saturation temperature, sodium temperature, and so forth. This could lead to a dangerous reversal of natural sodium circulation in the secondary loop. This has been shown in stability studies using traditional methods, and by direct calculations of transients in the secondary loop. The change in the natural circulation sodium flow rate was calculated in the secondary loop of the BN-350 reactor plant during steam pressure disturbances in the
steam generator. The results are shown in Figure 64. If the pressure control system maintains pressure with a high degree of accuracy (curve I), then the sodium flow rate is stable. However, if the flow control valve does not move because there is a malfunction, then natural circulation will be reversed (curve II). Natural circulation is not used when the reactor is operating because of the reasons listed above, as well as the relative complexity of the secondary loop configuration. In some situations this can prohibit natural coolant circulation from developing. Natural circulation of primary coolant in the BN-350 reactor is very reliable and stable.

**Figure 64.** A change in the natural circulation flow rate of secondary sodium and the change in steam pressure in the steam generator when the task of the pressure regulator changes: 1 and 2 - flow rate and pressure, respectively, during normal control system operations; 3 and 4 - the same parameters during failure and shutdown of the control valve, respectively.
The main loop equipment is also used for emergency shutdown cooling of the BN-600 reactor. The "total power loss" signal is sent along two independent time-delay circuits when the voltage dips on two out of three 6-kW auxiliary units, or when two out of three 220-kV air circuit breakers are switched off, and the stop valves of the appropriate turbines close. This signal triggers the fast-acting reactor protection system and the diesel-generators. The main coolant pump gears down to 25% of normal speed. All equipment that is not used in emergency shutdown cooling is disconnected from the 6-kV auxiliary power units. Special APENs are used to deliver feedwater to the steam generator during emergency shutdown cooling. In order to reduce the power level of these pumps, the pressure in the steam generator is automatically reduced to 5.0 MPa by releasing steam to the atmosphere. The pressure dump lasts for 20 seconds. Power is supplied to the APEN motors from the 500-kW diesel generators. The 1800-kW diesel generators serve as a power supply for the main coolant pump and the service water pumps. It takes 18 - 20 seconds to gear down the pumps. It takes 18 - 20 seconds to start up and maintain the working load for the 500-kW diesel generators, and 40 - 45 seconds for the 1800-kW diesel generators. Before the diesel generators start up, electrical power is supplied to all consumers by turbogenerator coastdown. After the
turbine stop valves are closed, mechanical coastdown is used. Tests have shown that at a load of 6.5 MW, a turbogenerator in coastdown can supply power to the equipment attached to it for 45 seconds. During this same time span, the rotational speed of the turbogenerator shaft, and consequently, the frequency of the current, decrease to 7% of nominal values. The electrical system of the turbogenerators has a special device for superexcitation in order to maintain the required voltage in the auxiliary units when the turbogenerators gear down. During each scheduled preventive maintenance cycle, the transmission of all the electrical signals required for emergency shutdown cooling is checked. According to the previously described algorithm, sodium temperature changes in the loops during emergency shutdown cooling are practically the same changes that occur when the fast-acting reactor protection system is tripped without total loss of all power to the plant.

The design of the plant does not provide for natural sodium circulation in the reactor loops of the BN-600 reactor during emergency shutdown cooling. Natural circulation has been experimentally tested in order to determine its stability, quantitative characteristics, and potential usage for raising the emergency shutdown cooling reliability level during equipment failures.
The first experiment was conducted at a low power level while sodium was heated in the loops to 16°C. The experiment was unsuccessful because stable natural circulation was not established in the secondary loop. This was due to the fact that changing the flow rate of the feedwater was not enough to maintain the temperature of sodium at the steam generator outlet. During the experiment, the sodium outlet temperature dropped. Since the sodium passes along a large ascending leg of piping behind the steam generator, the natural circulation delivery head began to drop in the secondary loop after the cooled sodium entered this leg of pipe. The loss of delivery head caused an additional reduction in the sodium flow rate and temperature at the steam generator outlet. As a result, coolant circulation in the secondary loop could not be maintained. However, such an experiment is not representative because it does not duplicate actual conditions. If the experiment took place under actual conditions, sodium would initially be heated up to quite high temperatures in the loops, and a reduction in heating would only occur following a significant drop in power density after natural circulation has fully developed. It should be pointed out here that the same experiment conducted in the reactor primary loop produced stable natural circulation of the sodium despite the low initial
heating. The natural circulation flow rate was 2 - 2.5% of nominal flow at a 1% power level.

The next experiment was conducted on one heat removal loop while secondary sodium was initially heated to 170°C. In this case the natural circulation flow rate of sodium in the loop was stable and reached 15% (Figure 65). The next experiment involved three loops with initial sodium heating in the loops to 120°C. Stable natural circulation was established in the primary loop at a flow rate of 3%, and in the secondary loop at 6%.

**Figure 65.** Change in temperature and sodium flow rate in the shut down secondary loop of the BN-600 reactor as natural coolant circulation develops: 1 - sodium temperature at pump inlet; 2 - sodium temperature at steam generator inlet; 3 - sodium flow rate referenced to the initial value before the pump was shut down.

In order to recreate what actually happens during emergency shutdown cooling, an experiment was conducted in which the reactor was shut down when the power was initially at 50% of normal power. The experiment was conducted in
three loops, and the initial heating of the sodium in the reactor was 140°C. Both in the primary and secondary loops, stable natural circulation of sodium was established. Some of the findings of this experiment are depicted in Figure 66.

Figure 66. Temperature change in the BN-600 reactor as natural sodium circulation develops (initial reactor power was 50% before the shut down): 1 and 2 - sodium temperature at the primary sodium inlet and outlet from the primary IHX; 3-5 - natural circulation flow rates in the secondary loops; 6 and 7 - primary sodium temperature at the steam generator inlet and outlet.

Thus, studies of natural coolant circulation in the loops of the BN-600 reactor have produced sufficiently good results and indicate that it is possible to use natural circulation during emergency cooling. The design of future
fast reactors calls for natural coolant circulation in the primary loop of the reactor when certain equipment failures occur. Although there have been hopeful computational and experimental results, at this time reactors are not being designed for natural circulation in the secondary loop because natural circulation is sensitive to the quality of steam generator parameter control in accident conditions, and because of the comparatively complex configuration of the secondary loop.

An important feature of next-generation fast reactor emergency cooling systems is that they will be designed with seismic design criteria contained in the new regulatory documents on nuclear power plant safety. Since it is not economically feasible to construct the third loop with a seismic design, emergency cooling will be carried out by special seismically-resistant air heat exchangers connected to the secondary circuit of each loop. Electromagnetic pumps will circulate the sodium through the heat exchangers, and air will be recycled through natural circulation. Since failures in the emergency cooling system are possible, General Safety Rule 82 requires redundancy of the electromagnetic pumps, diesel-generators, and some of the valves.
12. **Design Basis Accidents (DBA)**

In the introduction it said that two types of accidents are separately regarded as design basis accidents (DBA) in a fast reactor. The first accident involves a blockage of the sodium flow area in one of the fuel assemblies with subsequent coolant boiling, fuel meltdown in the fuel assembly, and damage propagation to one row of surrounding assemblies; or it involves a rupture of the primary piping that is not double-walled. The second type of accident (radioactive coolant leakage) actually requires no analysis of any of the transients in the reactor or the loops. In this instance studies involve determining the amount of sodium that leaks from the time the leak occurs until it is discovered and the isolation valves close. All primary piping is fitted with these isolation valves at the outlet of the double-walled pipe sections. Further, the processes of sodium burning in the containment rooms and the spread of airborne contaminants are also calculated, along with the amount of radioactive products lost to the various rooms of the primary loop and to the atmosphere. However, these processes will not be examined here, and it will only be mentioned that in all possible situations airborne contaminant leaks into operational areas and into the atmosphere will be limited to amounts significantly less than the minimum safe level.
In accordance with General Safety Rule 82, a DBA study should be used to provide for safety measures in the reactor design in such a situation. This means that it is necessary to develop a way to detect damage and prevent damage from spreading beyond the row of fuel assemblies immediately surrounding the damaged assembly. This problem can only be resolved by studying the physical properties of sodium boiling, fuel meltdown, and possible damage to the hexagonal can of the fuel assembly, and by determining the minimum amount of time it takes for the accident to spread from assembly to assembly. This minimum time interval is the main reference point that is used when designing a system for monitoring fuel assembly status, detecting localized accidents in the reactor core, and quickly shutting down the reactor.

Although General Safety Rule 82 postulates a DBA resulting from a fast reactor fuel assembly meltdown, it is important to note that this rule does not associate such a meltdown with any specific causes. The reasons proposed for a blockage of the fuel assembly flow area are swelling of the steel, precipitation of impurities from the sodium, or the intrusion of extraneous items into the coolant. Studies show that it is extremely unlikely that there will be a dangerous failure of fuel assembly cooling for the reasons given above. Coolant is delivered to and removed from the
fuel assemblies through many openings in the side surfaces of the shank. It is virtually impossible for all of these openings to simultaneously become blocked by some type of object. Furthermore, it would not be dangerous if even 50% of the flow area outside of the fuel portion of the fuel assembly became blocked. Such a blockage will only cause a 10% drop of the coolant flow rate from the initial level. There will also be no severe consequences if individual cells between the fuel assemblies become clogged.

In the stagnant zone behind the blockage, the coolant will start recirculating, and the temperature will increase. Localized boiling may occur if there are extensive blockages. It was calculated whether sodium would boil if 50% of the sodium flow area in the fuel portion of a fuel assembly was blocked after an initial superheating to 85°C (experimental data show this to be feasible for sodium). The calculations showed that steam bubbles forming behind the blockage and condensing as they are removed to a colder part of the fuel assembly have a lifetime of 0.1 second. In this instance the fuel rods will not dry out completely because a thin film of coolant will remain on the surface. The fuel rods can last a very long time under such conditions. For example, in experiments conducted on the DFR reactor (Great Britain), the fuel rods remained in operation for several hours after the coolant boiled. Of course, if the fuel rods
are superheated for an extended period of time, they can rupture and gas fission products can erupt into the coolant. Since the volume of these products is large at the end of a campaign, it was assumed that if these products enter the space between the fuel rods, a loss of cooling and renewed superheating of the fuel cells can result. Because of this, the consequences of various types of fuel cladding ruptures have been studied. The most detrimental ruptures are those which involve medium-sized ruptures of the fuel cladding tubes accompanied by the release of a relatively long-lasting outflow of gas. However, even under these conditions the superheating of the tubes does not exceed a couple dozen degrees and does not initiate a chain or "avalanche" of fuel rod ruptures. According to the findings, a cladding breach in the fuel rod starts very slowly under such conditions and can be detected when radioactivity increases in the cover gas of the reactor. At a later stage a breach can be detected by the presence of delayed neutrons in the coolant.

Sodium will only boil along the whole cross section of a fuel assembly if the flow rate of sodium through the assembly is reduced by at least a factor of two due to a major blockage of the flow area. Whether the accident progresses after boiling has begun is primarily contingent on whether the boiling is stable. Of course, it is possible to determine if the boiling is stable by the location of the
operating point on a curve that plots assembly pressure loss versus coolant flow rate (Figure 67). If the operating point is located on the ascending branch of the curve, then boiling is stable; however, if pressure in the vicinity of the operating point increases with a decrease in the flow rate, then boiling is not stable. In this case, boiling is accompanied by fluctuations in the pressure and flow rate, as well as a periodic loss of sodium from the fuel assemblies. The approximate criteria for stable coolant boiling in the fuel assemblies can be expressed as follows:

\[
\Delta P_0 > \frac{w_{Na}^2 \rho_{Na}}{2} \frac{(\theta_H - T)c_{Na}}{r_{Na}} \gamma, \quad \text{where}
\]

- \(w_{Na}\) is core inlet sodium flow rate at the start of boiling;
- \(\theta_{so}\) is the boiling point; \(T\) is the temperature of sodium at the reactor inlet; \(c_{Na}\) is the specific heat of sodium; \(r_{Na}\) is the specific heat of steam generation; \(\gamma\) is the ratio of liquid sodium density to the steam density on a saturation scale; and \(\Delta P_0\) is core pressure loss. Computational analyses for existing reactors indicate that sodium boiling in the fuel assemblies is frequently unstable. Boiling can only be stable (see equation 236) when the temperature of sodium increases significantly at reactor inlet. If there is a localized accident in a fuel assembly involving almost no change in the reactor parameters, the pressure differential in the fuel assembly is not adequate to ensure stable
boiling. To a certain extent, this is explained by the low density of sodium steam. When the sodium pressure is close to atmospheric pressure, then $\gamma \approx 1500 - 2000$, whereas for water under the same conditions, $\gamma = 1200$. Therefore, sodium begins to boil and bursts out from the fuel assembly when extensive blockages of the flow area occur. Fuel rod coolant failure leads to a temperature increase in the fuel rod and a fuel meltdown. Molten fuel enters the coolant, which can periodically fill up the assembly during unstable boiling.

![Figure 67. Hydraulic nature of a fuel assembly when sodium is boiling in it: 1 - section of stable boiling; 2 - section of unstable boiling.](image)

Thermal interaction between the molten fuel and the sodium must be considered in future analyses. This interaction is accompanied by pressure pulses which can lead to additional damage to an already damaged fuel assembly and
its neighbor. When molten fuel enters the sodium, it is broken down into tiny particles, which are then dispersed. The molten fuel is broken down by surface tension and inertial loads, which are created during high pressure gradients in the molten fuel/coolant interaction zone (MFCI zone). The more the fuel disperses, i.e., the smaller the pieces become during this process, the more quickly heat is given off to the coolant, and the greater the pressure pulses. The heat transferred to the coolant can be converted to mechanical energy (work). The heat transfer coefficient depends on the rate of heat transfer. If heat is transferred very quickly, the mass of coolant in the MFCI zone will be constrained, and in the initial first few seconds of the process the MFCI zone remains unchanged. During this process, the pressure increase in the MFCI zone is determined by the thermal pressure coefficient. If only sodium and fuel are located in the MFCI zone, then the indicated thermal pressure coefficient should be mainly attributed to the sodium. Studies indicate that a change in the volume of fuel has little effect on pressure. The thermal pressure coefficient of liquid sodium at an approximate temperature of 1000°C is 0.7 - 0.9 MPa/°C. Therefore, significant pressure surges can be expected to occur before the sodium boils a second time due to molten fuel-coolant interaction. During this process the
temperature of sodium can increase by a couple of hundred degrees, and pressure may increase by several dozen megapascals. However, this is the upper limit. It is unlikely that the pressure will rise so high for the following two reasons. First, heat is not transferred from fuel particles to the coolant in an instantaneous manner. Experiments have shown that fuel dispersion leads to the formation of particles that are approximately 0.3 - 0.5 mm in size. Even given ideal heat transfer to the sodium, such particles still have time constants of 0.01 - 0.07 seconds. The formation of a steam blanket on the surface of the particles increases the amount of time it takes to cool them. Even the amount of time indicated above is enough for deformations and shifts in sodium surrounding the MFCI zone. Therefore, this process markedly reduces the pressure peaks in the MFCI zone. Second, the presence of fission gas and sodium vapor is practically unavoidable in the MFCI zone. Fission gas is released from the damaged fuel rods along with molten fuel, and sodium vapor remains when the fuel assemblies are only partially filled with the returning sodium. Vapor also arises during a secondary boiling of sodium in the MFCI zone. Gas and vapor markedly lower the thermal pressure coefficient for this zone.

If sodium, molten fuel, and gas are located in the MFCI zone, and the temperature of the gas and the volume of the
fuel are constant, then the value derived from the pressure in the MFCI zone due to sodium temperature, i.e., the above-mentioned thermal pressure coefficient for the MFCI zone, is:

\[
\gamma \rho = - \frac{\alpha_{Na} (1 - \epsilon_g)}{\beta_{Na} (1 - \epsilon_g) + \beta_g \epsilon_g}
\]

\(\alpha_{Na}\) is the sodium heat expansion coefficient; \(\beta_{Na}\) is the isothermal compressibility of sodium; \(\beta_g\) is the isothermal compressibility of gas; and \(\epsilon_g\) is the volume fraction of gas in the MFCI zone. The isothermal compressibility coefficient for liquid sodium is \(\beta_{Na} = -(3.5 - 4.5) \times 10^{-4}\) MPa\(^{-1}\), and for gas it is \(\beta_g = -1/P_o = 1(1 - 10)\) MPa\(^{-1}\). It is clear from this equation that if the volume fraction of gas in the MFCI zone is only several percent, then the thermal pressure factor decreases exponentially. In other words, an expansion of liquid sodium as its temperature increases is compensated by a compression of gas without a large pressure increase. It goes without saying that a pressure increase in the MFCI zone is also dependent upon the amount of molten fuel and upon the ratio of fuel and coolant masses exchanging heat.
Figure 68. A 1-centimeter fuel rod meltdown accompanied by molten fuel entering the sodium: $a$ and $b$ – pressure in the MFCI zone; $\delta$ and $\varepsilon$ – sagging of the fuel assembly can; 1 and 2 – momentary dispersion of the fuel into the sodium; 3 – time interval of fuel interaction with liquid sodium $= 0.005$ seconds; 1 and 3 – fuel entering the liquid sodium; 2 – fuel
In order to illustrate how the enumerated factors affect the process characteristics, several findings will be cited on how this process was calculated. Figure 68 shows the changes in the MFCI zone parameters during a one-centimeter meltdown of one fuel rod. Given the lack of gas and vapor in this zone and a momentary interaction of fuel with the liquid sodium, the pressure initially peaks in the zone at a maximum of 7.5 MPa; the duration of the peak is $0.5 \times 10^{-4}$ seconds. The walls of the hexagonal cans of the fuel assemblies will not collapse in this case. If the fuel from the meltdown of the fuel assembly falls into the boiling sodium, then the maximum pressure in the pulse is much less than in the previous case, given the reasons mentioned above. The pressure only reaches 0.7 MPa. The extent of the pressure pulses during MFCI is particularly affected by the duration of the interaction and by the dispersion of fuel. Since the pressure pulses are not long in duration, even a small (within hundredths of a second) shift in various sections of the fuel rods and fuel assemblies can significantly lower the magnitude of these pulses. A study including all the enumerated factors indicates that the fuel assembly can cannot be damaged as a result of pressure pulses, even if the fuel in half the assembly melts down.

A simultaneous fuel meltdown over the whole length of
the fuel assembly will cause the hexagonal can to collapse. However, such a meltdown can only occur in the unlikely event of a sodium flow blockage through the entire assembly. This accident is examined in order to determine the upper limits of the process. These limits are vital for developing a detection system for localized accidents in the reactor core and the appropriate devices for a protection system. The sodium boiling process in the fuel assemblies was analyzed when there was a sudden interruption in the flow rate of sodium through the assemblies, and the results are presented in Figure 69. Sodium begins to boil 0.55 seconds after leaving the core, and the fuel assembly will be completely ruined 0.35 seconds later. Heat removal from the surface of the fuel rods comes to a halt after the thin layer of coolant evaporates off their surfaces. In the hot spots of the core this will occur within 0.3 - 0.4 seconds after boiling begins. When a significant portion of the thin sodium layer dries on the fuel rods, the amount of vapor and heat that forms the steam blanket is reduced, and the sodium begins to return to the fuel assembly.

At this stage it is possible that the assembly is not completely filled with sodium. Apparently, a dry section will remain in its central part, and the temperature will continue to grow, resulting in fuel melting. A total meltdown in the most dangerous section will occur within
5 - 6 seconds. This meltdown may further disturb fuel rod cooling, after which melting propagates throughout the fuel rod within one second. Fuel melting can include a meltdown or rupture of the fuel assembly can. It was calculated that in a worst case scenario, a complete penetration of the fuel assembly can occur within 8 seconds. After this, the molten fuel can come into contact with the fuel assembly can abutting the damaged one. In total, the duration of the accident is 15 seconds and includes all the stages listed above from the time the assembly clogs up until the first possible time the molten fuel leaves the damaged fuel assembly. It is this time interval which is used as the reference point for determining the appropriate response time of the control system and reactor protection channels that function during localized accidents in the reactor core. Localized accidents can be detected by traditional means, i.e., flow meters and temperature sensors. However, in order to simplify the design, domestic fast reactors are not currently equipped with this equipment for controlling all fuel assemblies, as was already stated. A rupture and failure of the rods can be detected by the cladding seal monitoring system when the level of radioactivity in the reactor cover gas increases, or delayed neutron sources, i.e., fission products, are present in the coolant. The exceedingly unlikely event of a sudden blockage of the fuel
assembly was used to determine the accident propagation time given above. However, from a practical standpoint, a slowly developing failure may be used.

![Figure 69. Boiling of sodium in a fuel assembly: 1 - limits of the bubble; 2 - temperature of sodium in the bubble.](image)

Heavy reliance is placed on the fuel assembly localized accident detection system, which is based on studies of neutron and acoustical noises in the reactor and an analysis of the reactor reactivity balance. When sodium begins to boil in the reactor core, changes take place in the high frequency range (10 - 200 kilohertz) of the acoustical noise spectrum and in the low frequency range of neutron noise.

The reactivity balance is monitored in the core by using reactivity components associated with core temperature changes due to deviations from initial values of reactor power, coolant flow rate and coolant input temperature. The hydrodynamic effects of reactivity, effects due to movements of the control mechanisms, and fuel depletion are also used
when monitoring reactivity balance. The sum of all the components calculated by the control device is constantly compared to the actual reactivity. Disagreements between these figures indicate that anomalous components have arisen. These components may result from the boiling and release of sodium from the fuel assembly, or from the melting and shifting of fuel in the assembly. In such cases warning and emergency signals are given. Experiments indicate that anomalies in reactor reactivity may be detected with a sensitivity of up to $2 \times 10^{-5}$. In order to assess the suitability of this sensitivity, it can be said that a reactivity change in the BN-600 reactor averages $0.7 \times 10^{-4}$ when sodium is released from one of the fuel assemblies. When the fuel melts in the fuel assembly and runs off to the lower axial screen, reactivity deviations reach $5.4 \times 10^{-4}$. When fuel accumulates in the central section of the fuel assembly, reactivity deviations equal $9 \times 10^{-4}$. Such situations will be reliably detected by the monitoring device.
13. An Interruption in the Supply of Feed Water to the Steam Generators, and Turbine and Turbogenerator Failure

1 The flow of feed water to the steam generators may be stopped as a result of the following: ruptures in the tertiary loop piping or equipment elements, such as the deaerator; an accidental closure of the shut-off valves; a malfunction in the feed controller; or a failure of the feed pumps.

2 In all these cases, the sodium temperature begins to rise at the steam generator outlet. Then it begins to rise at the IHX inlet of the secondary loop, and finally, at the reactor inlet. During normal operations an automatic controller regulates the sodium temperature at the steam generator outlet. Therefore, the primary objective here is to examine upsets in the delivery of feedwater for which the controller cannot compensate. If water stops flowing in one of the heat removal loops, a signal indicating a sodium temperature increase at the steam generator outlet causes the appropriate secondary main cool pump to shut down, followed by the primary one. This will slow down the rate at which the sodium temperature increases and as a result, will prevent large thermal stresses in the housing and outlet tube plates of the steam generator. This also eliminates the possibility that the sodium temperature will rise at the reactor inlet via the shut down loop. Reactor
power will decrease, as it would during any heat removal loop shutdown.

When one of the feed pumps disconnects, followed by a failure of the standby pump, which should have been triggered when the feed pump disconnected, there will not be enough feed water flowing. This calls for a reduction in loop power, if feedwater is independently delivered along each one of the various loops. However, if feed water is supplied to the steam generator from only one header, then the power level of all the loops must be reduced. The use of a common header is now being considered to supply feedwater to the steam generator for fast reactors that are currently being designed. These reactors will have one K-800-130 turbogenerator. Moreover, the plan is to also use turbo feed pumps instead of the electric ones that operate in the BN-600 reactor layout. If one turbo feed pump shuts down, reactor power is lowered together with the flow rate of sodium in the primary and secondary loops, while nominal steam parameters at the turbine inlet are maintained. This emergency power reduction helps avoid dangerous temperature deviations at the steam generator outlet and in the reactor core. A larger emergency power reduction will also be used in other emergency situations that will be discussed later on in the text. It must be noted that a nuclear power plant that has one turbogenerator and uses turbo feed pumps may
totally lose the water supply to the steam generator. In such a situation, additional measures have been provided for emergency shutdown cooling of the reactor. Air heat exchangers attached to the reactor's secondary loop have been chosen to provide cooling. As has already been stated, these heat exchangers should have a seismic-safe design and provide for the emergency removal of heat in case of an earthquake.

**Figure 70.** Reactivity versus time when the BN-600 reactor's power is reduced from 100% to 25%: 1 and 2 - reactivity versus time when power is reduced for 50 and 100 seconds, respectively; 3 and 4 - reactivity which must be inserted by the absorber rods, given the temperature effects of reactivity, in the same situation.

There is one additional situation that calls for an immediate reduction of reactor power. It is when one or more turbogenerators in the nuclear power plant becomes disconnected from the power supply system. This situation is being carefully studied for fast reactors under design.
If high-voltage air circuit breakers shut down as a result of a failure of the power supply system, the nuclear power plant turbogenerator reduces power to an auxiliary power level. At this stage the turbine control system quickly reduces the flow rate of live steam, thereby averting an excursion. The system has been designed to prevent the safety circuit breaker from responding during the sudden load dip from rated to auxiliary load. The surplus steam produced by the steam generator—but not admitted after the turbine control valve closes—is dumped to the condenser and partially exhausted to the atmosphere through safety valves. In order to limit feedwater losses at this time, it is necessary to reduce reactor power as quickly as possible and bring steam generator output in line with turbine steam consumption. While the power is being reduced, the live steam parameters are maintained at a constant level. In order to do this, the coolant flow rate in the primary and secondary loops must be reduced simultaneously with, and in an amount proportional to the power reduction. In fast reactors, the flow rates are kept within the range of 100% to 25%. Consequently, it is possible to maintain the rated parameters of live steam within the very same range. Somewhere between 100 - 150 seconds is selected to reduce power from 100% to 25% of normal power. The linear response
function introduced in Chapter 2* can be used to explain the change in reactivity as power is being reduced within a specified range according to linear law, and then maintained at a constant level:

\[
\bar{\rho} = (1 - 1/0.25) U_{\Delta \tau} (\tau) = -3 U_{\Delta \tau} (\tau), \text{where}
\]

\(\Delta \tau\) is the amount of time it takes to reduce power. This reactivity change is depicted in Figure 70. In addition, the temperature effects of reactivity must be compensated for during a transient. Therefore, reactivity which should be inserted into the reactor to support the given power reduction law \(\bar{\rho}_{PC}(\tau)\) differs from the derived value \(\bar{\rho}(\tau)\) by the value of these effects, which themselves are a function of time (Figure 70). An analysis shows that the automatic control rods and shim rods can provide the necessary reactivity insertion. As has already been pointed out, one of the major advantages of fast reactors is that their power density field is highly stable. Therefore, even substantial repositionings of individual control elements do not lead to major warping and distortions of the relative power density distribution in the core during this type of transient. This is illustrated in Figure 71, which contains the results of measuring deviations from the initial power density value

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* TN: Chapter 2 has not been translated.
in individual fuel assemblies of the BN-600 reactor when one of the shim rods is raised as high as it can go and another is lowered as far as it will go. Even in this situation the power density will not deviate by more than 11% of the initial level. Therefore, an emergency power reduction will not cause dangerous temperature changes in any of the fuel assemblies if constant, even heating of the coolant is maintained in the reactor. Figure 72 graphs the results of calculating a transient in a BN-600 reactor during an emergency power reduction to 25% of normal power. Despite the fact that both power and coolant flow rate were reduced very rapidly, sodium temperature was maintained with a fair degree of accuracy as it left the reactor. The automatic (control) system reduces reactor power down to 25% of normal power. The operator may reduce the power level even further, down to auxiliary level. It is possible that the emergency power reduction mode will not only be used when the turbogenerator becomes disconnected from the power supply system, but also when a turbine fails for a short period of time.
Figure 71. Deviation in power density from the initial values when the KC-11 shim rod is raised as high as it will go and the KC-17 rod is simultaneously lowered as far as it will go (the length of the arrows is proportional to the deviation in power density from the initial value before the rods are repositioned; the right-hand side shows the size of the arrows which correspond to a deviation in power density by 5% from the initial value): \( \sigma \) - growth in power density; \( \pi \) - reduction in power density.
Figure 72. Extreme reduction in reactor power from 100% to 25% when the turbogenerator is disconnected from the power supply: 1 and 2 - reactor power and coolant flow rate through the reactor, respectively, relative to the nominal values; 3 and 4 - reactivity, and the reactivity inserted by the absorber rods, respectively, 5 - sodium temperature at the reactor outlet.
14. Reactor Startup and Shutdown, Planned Changes in Reactor Power, and Stable Reactor Operations in the Steady-State Mode

Limits are placed on the rates at which reactor power and coolant temperature change in the loops when the reactor is started up and shut down. When reactor power is increased from the minimally controllable level to 1% of normal power, the reactor period is usually held steady at 60 - 100 seconds. On the one hand, a period of this duration ensures safety. On the other hand, it shortens the time of reactor startup. Power is further increased at a steady and acceptable rate of change for the reactor outlet coolant temperature. Originally, the limits used for the rate of coolant temperature changes in the loops were borrowed from the field of conventional (non-nuclear) energy industries. In their case, the acceptable rate at which the temperature can increase during the startup of many plants was set at 60 - 70°C/hr. This rate was established through experience. It allows stresses to be limited in the power plant elements to a fairly safe level. These stresses are caused by thermal expansion and displacement of piping and heat exchanger devices, as well as uneven heating of individual components or sections of these elements. A more thorough analysis showed that this rate of temperature change is not always acceptable for a nuclear power plant. When calculations and
measurements were performed on a transient temperature field in the reactor pressure vessel, the IHXs, and the steam generator, they showed that the given rate must be reduced to 10 - 20°C/hr for certain operating modes and in certain ranges of temperature change.

Stresses in the fuel cladding tubes are very often a limiting parameter in the fuel rods. These stresses result from mechanical interaction with the fuel cores. The fuel expansion temperature coefficient is 1.5 times lower than the temperature coefficient of steel expansion. However, the average fuel temperature increase is one order greater than the increase in fuel cladding temperature when reactor power increases, for example, in the median core cross-section. Therefore, the annular space between the fuel and the cladding in the fuel rods shrinks during reactor startup. The most unfavorable circumstance would be an increase in reactor power after the reactor had been operating for a fairly long period of time (at least several days) at reduced power. When this occurs, the mass transport of fuel in the fuel rods leads to a sharp reduction in the annular space between the core and the fuel cladding, until the space completely disappears. After this, an increase in power, as has already been stated, leads to faster thermal expansion of the fuel rod cores compared to the cladding, and to more elongated stresses in the latter.
Under these conditions, it is necessary to limit the rate at which power increases, and to stop at interim levels in order to relax stresses caused by core and cladding creep.

The most experience has been acquired researching these conditions with oxide fuel rods in a thermal neutron reactor. Linear loads of the water-moderated, water-cooled power reactor (VVER) fuel rods are reasonably high. In the VVER-1000 reactor, for example, these loads reach 415 W/cm, which is quite close to the linear loads of fast reactor fuel rods. Therefore, it is known that experience with thermal reactors can be used to optimize operating modes of fast reactor fuel rods. Permissible rates of power level changes in VVER-type reactors are 3 - 4% per minute when these rates are used to regulate daily and weekly energy system load schedules.

At some foreign nuclear power plants with thermal reactors, limits of up to 3% per hour have been set for the rate of a power increase after a long period of operation at reduced power levels. It is apparent that there can be a large difference in possible limits on power change rates. This can be explained by the fact that the condition of the fuel rods is not identical before the power is increased, nor are their parameters. This experience is being used to develop guidelines for fast reactor startup. The selected
limits for the rate at which power can be increased are checked during reactor operation.

During a reactor shutdown, the fuel rods do not require power reduction rate restrictions, since the fuel cores will shrink in size faster than the fuel cladding in these modes.

Initially, reactor power is increased at a constant, but reduced sodium flow rate relative to the rated sodium flow rates in the primary and secondary loops of the plant. After nominal sodium heating has been achieved in the reactor, reactor power continues to rise while the flow rate increases at the same time. During a planned reduction in reactor power, limits are placed on the rate at which coolant temperature changes and power increases occur in the loops. As has already been said, the reasons why there are different limits for a power increase and for a reactor shutdown are the dissimilar conditions under which the fuel rods operate, as well as differences in the thermomechanical processes in the plant components.

All thermal power plants place restrictions on reactor startup rates and transition rates from one power level to another. This also applies to nuclear power plant turbines. For a turbine, the following types of startup exist: (1) cold start, (2) semi-cooled start, and (3) hot start. It usually takes about 100 hours to totally cool down the steam turbines of a nuclear power plant. If the plant is shut
down for less than five to ten hours, a subsequent startup is conducted from a hot state. After a shutdown of 20 - 40 hours, the turbine is in an intermediate, semi-cooled state.

Currently, fast reactors are designed to operate in a base-line mode. Therefore, their turbines are most often started up after a power peak from a cold state. There are exceptions for the BN-600 reactor, where the turbines are started up in disconnected heat removal loops after a short down time.

Reactor operational stability in the steady state will now be discussed. Previously, it was stated that a fast reactor is easy and convenient to operate, both when its parameters are automatically controlled, and in the self-regulation and remote-controlled modes. To a great degree, this is due to the fact that the reactivity-power channel has a large reserve of stability. Instability means that there is a spontaneous increase in power, or power fluctuations with ever increasing amplitudes after an initial accidental reactivity deviation. In such a situation, an uncontrolled power increase without power fluctuations can occur if the feedback reactivity (i.e., the reactivity temperature effects resulting from a power jump) has a reasonably large positive component that is not offset by negative components. In this instance, feedback reactivity is totaled from the initial disturbance that
causes a rapid increase in reactor power. Previously, it was stated that temperature and power reactivity effects of a large fast reactor do not have positive components that could lead to this type of instability.

During negative temperature and power reactivity effects, there is only one type of possible instability, i.e., spontaneous power fluctuations of ever increasing amplitude. Spontaneous fluctuations such as these are called autofluctuations. They occur when feedback reactivity is in counter phase, i.e., there is a shift in phase by $\pi$ relative to the initial reactivity fluctuation. In addition, the amplitude of the feedback reactivity is equal to or greater than the initial fluctuation. Autofluctuations can be dangerous when their frequency and amplitude are large. It must be said that feedback reactivity has components responding to a power change with various time and phase shifts. The main components are affected by quick temperature changes in the core, immediately following a change in reactor power density. When this occurs in a reactor employing oxide fuel, the most significant component affected by the Doppler effect cannot have a phase lag of more than $\pi/2$ relative to power changes. No matter what the frequency is, power will also not lag in phase by more than $\pi/2$ relative to reactivity. Within the range of essentially dangerous fluctuation frequencies, power almost completely
corresponds in phase to the reactivity. Consequently, feedback reactivity in such reactors cannot have a phase lag of \( \pi \) relative to the initial disturbance. Therefore, power autofluctuations are impossible in this reactor.

As a result of reactivity feedback components that pass through the external heat removal system and are due to fluctuations in the coolant temperature at the reactor inlet, low frequency power fluctuations can occur in some reactors in the self-regulation mode. These fluctuations have a period of three to four minutes. The reason why these fluctuations arise is clarified in Figure 73. Suppose a reactivity of \( \rho_b \) is inserted into the reactor. As a result, reactor power will deviate by \( \rho_b/K_n \) after a relatively short-term transient. Coolant temperature at the reactor outlet will deviate by \( \Delta\theta_0 \rho_b/K_n \). After the temperature wave passes through the heat removal system and enters the reactor, a disturbance of \( A_0 \Delta\theta_0 \rho_b/K_n \) will occur. Due to temperature effects, the disturbance induces a reactivity deviation of \( k_A A_0 \Delta\theta_0 \rho_b/K_n \). The fluctuations will damp out if the secondary changes in reactivity are less in amplitude than the initial reactivity changes, i.e., if

\[
\left| k_A A_0 \Delta\theta \right| < |K_n|
\]

In the BN-350 and BN-600 reactors, this condition is known to exist. In the above equation, \( K_n = \) power effect of
reactivity; \( k_t \) = temperature reactivity coefficient; \( \Delta \theta_0 \) = coolant heating in the reactor; and \( A_{0_T} \) = ratio of the deviation in reactor coolant inlet temperature to the outlet temperature deviation. This ratio is dependent upon the heat removal system performance characteristics. Usually it does not exceed 0.3 - 0.4 at the very lowest frequencies.

12 The value of \( A_{0_T} \) drops very quickly as the frequency increases due to the large heat lag time in the heat removal system. It is for this very reason that if such fluctuations are indeed possible, they occur at very low frequencies of about 0.004 - 0.005 Hz. Of course, these fluctuations are not dangerous since the control system can easily suppress them.

13 It must be said that the control system can successfully correct an instability even during positive power and temperature coefficients of reactivity. As was already stated, such reactivity coefficients are not possible in a large, fast power reactor. Therefore, this statement only holds true with a small-volume research reactor. During a positive power effect, the automatic proportional neutron power control rod provides stability if the rate at which it operates satisfies the condition:

\[
v_{pc} \geq \frac{H_{pc}}{B_{pc}} \Delta n, \quad \frac{\lambda K_n}{1 + \lambda \tau_0}, \quad \text{where}
\]
\( H_{pc} \) = working (power) stroke of the control rod; \( B_{pc} \) = worth of the control rod; \( \Delta n_{c} \) = control rod proportionality zone; 
\( \lambda \) = the average decay constant of the delayed neutron precursor-nuclei; and \( \tau_{0} \) = the time constant typical for the power effect.

Figure 73. Change in successive deviations in reactivity and power when the disturbance passes through the heat removal system: 

- \( a \) - damping out the fluctuations when \( |K_{M}| > |k_{e}A_{oT}| \); 
- \( b \) - increase in fluctuation amplitude when \( |K_{M}| < |k_{e}A_{oT}| \).
Analyzing this equation shows that even at a moderate rate of 50 - 100 mm/s, the automatic control rod ensures reactor stability during a positive reactivity effect commensurate with $\beta$. 
Conclusion

1 Even a short description of accidents and transients allows those factors which ensure safe fast reactor operation to be identified.

2 The low pressure levels in the liquid metal loops and the high boiling point of the sodium used to cool the fast reactor virtually preclude accidents involving a massive release of radioactive products from the primary loop. It is these circumstances that simplify the problem of heat removal from the reactor. A nuclear power plant with fast reactors produces less radioactive effluent during operation than other plants with other types of reactors.

3 The core construction materials make it possible to use high parameters for the nuclear power plant steam power cycle. These materials are very compatible with the fuel and coolant, and preclude dangerous phenomena, such as a steam-zirconium reaction at the very highest temperatures that occur during accidents.

4 Current regulatory documentation postulates a design basis accident (a rupture or melting of one core fuel assembly with the damage propagating to one row of surrounding assemblies) that does not entail severe consequences. Moreover, design and experimental research, as well as operating experience, show that such an accident is extremely unlikely.
The fast reactor protection system prevents any damage to the core when major deviations in any of the reactor inlet parameters occur, i.e., when there is virtually a prompt reactivity jump of up to $0.3 - 0.4\beta$; when there are even larger reactivity changes at a rate of up to $0.6 - 0.7\beta/s$; when all the circulation pumps simultaneously shut down; and when the flow of feedwater to the steam generator stops. The loss of one heat removal loop does not call for a reactor shutdown. It is possible to sharply reduce reactor power down to house load within 2 - 2.5 minutes when the nuclear power plant turbogenerator is separated from the power system until a set minimum level is achieved while the turbine is shut down for short periods of time. This is possible because fast reactors have excellent power cycling maneuverability and a highly stable power density field. The problem of non-steady state thermal stresses in the fast reactor structural elements and heat removal equipment has been solved.

Operating experience has shown that fast reactors are simple to control when the automatically-operated controllers are used, as well as in the remote control or self-regulation modes. When a reactor is in the self-regulation mode, it compensates for significant reactivity disturbances, coolant mass flow or coolant inlet temperature disturbances without dangerous temperature deviations in the core. If all the circulation pumps are lost, and a hypothetical reactor
protection system failure occurs at the same time, the reactor provides effective self-quenching of power density in the core. If all the pumps shut down, sodium boiling can be prevented in the reactor by simply inserting an absorber rod with a worth of 0.5 - 0.6β into the core. If, after shutdown, the pumps convert to a minimum rotational speed and are powered by reliable sources, as provided for by the control circuit, then sodium will not boil, even if no absorber rods are inserted. In this instance, a failure of the reactor protection system is practically impossible since this system is initiated by 6 - 12 independent channels.

7 Sodium has good thermal and physical properties which, when taken together with the circumstances given above, guarantee reliable emergency shutdown cooling of a fast reactor. Natural coolant circulation through the reactor has been successfully used in reactor emergency shutdown cooling systems, along with coastdown of the circulation pumps and turbogenerators, and the storage of heat in the loops.

8 Everything that has been said shows that fast reactors have the potential to become one of the safest types of reactors in the nuclear power industry. The goal of operation is to take advantage of the performance characteristics, design, and layout of fast reactors in order to optimize their operating modes and draw on experience to obtain knowledge useful when designing future plants.
Personnel who work at nuclear power stations will find themselves in unusual situations in which they will be forced to make decisions, or determine what the consequences will be if a modification or change is made to one of the operating modes. To help with this work, some engineering methods have been suggested to calculate reactor kinetics as they relate to reactivity and power, non-steady state modes of heat exchange in the reactor, heat exchanger devices, piping, mixing chambers, transient hydraulics, and transient thermal stresses in the structural elements in the elastic region and the elastic-plastic region. Some of the methods are based on approximated semianalytical solutions. They permit the calculation of transients with large, five-to-ten second intervals, or even longer, as the nature of the process calls for. A sophisticated computer is not needed in order to carry out these calculations. These methods are illustrated by sample calculations, and in the appendix there are several simple programs based on these examples for microcomputers like DVK, which are most often used in training centers. The programs are interactive and can be used as the simplest form of training when studying nuclear reactor kinetics both with and without feedback reactivity.

The results from such approximate calculations and operational observations may beneficially complement each other.
Translator's Bibliography


